May 5, 2006

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

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION - NRC INTEGRATED INSPECTION REPORT 05000387/2006002 AND 05000388/2006002

Dear Mr. McKinney:

On March 31, 2006, the US Nuclear Regulatory Commission (NRC) completed an inspection at your Susquehanna Steam Electric Station Units 1 and 2. The enclosed integrated inspection report documents the inspection results, which were discussed on April 7, 2006, with Mr. R. Saccone, Vice President - Nuclear Operations and other members of your staff.

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

This report documents one NRC-identified finding and one self-revealing finding both of very low safety significance (Green). One finding was determined to involve a violation of NRC requirements. However, because of the very low safety significance and because the issue was entered into your corrective action program, the NRC is treating this finding as a non-cited violation (NCV), consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest the NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington, D.C. 20555-0001; with copies to the Regional Administrator Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the Susquehanna Steam Electric Station.

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

Sincerely,

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

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

Enclosure: Inspection Report 05000387/2006002 and 05000388/2006002
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket Nos.:	50-387, 50-388
License Nos.:	NPF-14, NPF-22
Report No.:	05000387/2006002 and 05000388/2006002
Licensee:	PPL Susquehanna, LLC
Facility:	Susquehanna Steam Electric Station, Units 1 and 2
Location:	Berwick, Pennsylvania
Dates:	January 1, 2006 through March 31, 2006
Inspectors:	A. Blamey, Senior Resident Inspector F. Jaxheimer, Resident Inspector E. Gray, Senior Reactor Inspector J. Furia, Senior Health Physicist P. Finney, Reactor Inspector
Approved by:	James M. Trapp, Chief Reactor Projects Branch 4 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000387/2006-002, 05000388/2006-002; 01/01/2006 - 03/31/2006; Susquehanna Steam Electric Station, Units 1 and 2; Flood Protection Measures, Maintenance Effectiveness.

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

A. NRC Identified Findings and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. A self-revealing non-cited violation was identified for not complying with 10 CFR 50, Appendix B, Criterion III, "Design Control." PPL did not assure that the emergency core cooling system (ECCS) compartments were water tight as described in the Final Safety Analysis Report (FSAR). This resulted in water intrusion into two ECCS compartments simultaneously during an unexpected overflow of the reactor water cleanup backwash receiving tank on August 18, 2004. PPL entered this issue into the corrective action program, re-performed numerous internal flood analysis and concluded that the hatch plugs do not have to be leak tight. In addition, PPL sealed the gaps around the equipment hatch plugs.

  This finding is greater than minor because it is associated with the Mitigating Systems Cornerstone of design control and affects the cornerstone's objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e. core damage). The water intrusion reduced the capability of the Division II core spray system because the system auto start feature was manually disabled for approximately two hours. The inspectors performed a Phase 1 screening using IMC 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations." The finding was determined to be of very low safety significance (Green) because this design deficiency did not result in a loss of function in accordance with Generic Letter 91-18. (Section 1R06)

C Green. The inspectors identified a finding for not implementing Corrective Action procedure, NDAP-QA-702, which requires all actions to correct and prevent recurrence be completed before the closure of a condition report. Following electrolytic capacitor failures at Susquehanna, corrective actions were not completed, which directly contributed to loss of control rod drive hydraulic flow on

February 22, 2006. PPL has entered this issue into their corrective action program.

This finding is greater than minor because it is associated with the equipment performance attribute of the Mitigating Systems Cornerstone. The finding negatively affected the cornerstone objective to ensure the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences because the failure resulted in increased scram times on 20 control rods. The finding was determined to be of very low significance (Green) since the finding does not represent a loss of safety function and is not potentially risk significant due to external events. The cause of the problem is related to the Problem Identification and Resolution cross-cutting area. (Section 1R12)

B. Licensee-Identified Violations.

None.

REPORT DETAILS

Summary of Plant Status

Susquehanna Steam Electric Station (SSES) Unit 1 began the inspection period at or near full reactor thermal power (RTP). Unit 1 remained at or near full RTP except for control rod pattern adjustments until the start of the end-of-cycle power coastdown on February 12th. On February 18th, power was reduced from 97 percent power to just below 70 percent power for control rod friction testing, and a rod sequence exchange. On March 3rd, the unit was shutdown to begin a refueling and maintenance outage. Unit 1 remained in the refueling and maintenance outage for the remainder of the inspection period.

Unit 2 remained at or near full RTP during the entire inspection period with the exception of a power reduction to 70 percent on February 11th for a control rod sequence exchange, and a power reduction to 75 percent on March 10th to repair the 3C feedwater heater normal level control and to perform control rod scram time testing.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01- 1 Sample)

Adverse Weather - Site Readiness

a. Inspection Scope

The Susquehanna site experienced high winds on February 13, 2006, the inspectors reviewed site conditions and PPL's preparations for high winds. This included a tour of the outside areas including the areas with materials and equipment staged for the upcoming Unit 1 refueling outage. Inspectors also reviewed ON-000-002, "Natural Phenomena" and evaluated risk management actions taken in preparation for high wind. The documents reviewed for inspection activities are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04Q - 4 Samples; 71111.04S - 1 Sample)

1. Partial Walkdown

a. Inspection Scope

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

- Unit 1 “A” train of standby gas treatment (SBGT) in operation following inadvertent Div. I containment isolation
- Unit 2 Div. I 250 volt DC and Unit 1 Div. I 250 volt DC batteries/charges and buses
- Unit 2 emergency service (ESW) water, Reactor Building and emergency diesel generator (EDG) cooling
- Unit 2 control rod hydraulic system, Turbine Building and Reactor Building components

2. Complete Walkdown

a. Inspection Scope

The inspectors conducted a detailed review of the alignment and condition of the supplemental decay heat removal system (SDHR). The inspectors reviewed operator rounds, checkoff lists, system operating procedures and system piping and instrumentation diagrams. The inspectors evaluated ongoing maintenance and outstanding condition reports associated with the SDHR system to determine the effect on system health and reliability. The test results obtained before placing the system in service for the Unit 1 refueling outage were reviewed against acceptance criteria and system design documents. The inspectors verified proper system alignment and looked at system operating parameters when there was a change to the available decay heat removal systems for Unit 1.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05Q - 9 Samples)

1. Fire Protection - Tours

a. Inspection Scope

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

- Units 1 and 2 security control center, fire zone 0-83, FP-013-360
- Unit 1 main condenser gallery, fire zone 1-32D FP-113-291
- Unit 1 feedwater heater cells A, B, C fire zones 1-33 (E-G)

- C Unit 1 main steam isolation valve area and main steam pipeway, FP-113-118
- C Unit 1 749' Div. I 4 KV switch gear rooms fire area 1-5F, 1-5G
- C Unit 2 reactor core isolation cooling (RCIC) lube oil deluge and RCIC room
- C Unit 2 residual heat removal (RHR) rooms, FP-213-240
- C Unit 2 Reactor Building standby liquid control area fire zones 2-5A-N and 2-5A-S
- C Unit 2 upper switchgear room, fire zone 2-34A, elevation 714'

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06 - 1 Sample - Internal)

1. Internal Flooding

a. Inspection Scope

The inspectors reviewed selected risk-important plant design features and PPL procedures intended to protect the plant and its safety-related equipment from internal flooding events. The inspectors reviewed flood analysis and design documents, including the final safety analysis report (FSAR), engineering calculations, administrative door and floor plug controls, abnormal operating procedures and plant drawings. The inspectors performed walkdowns in the Unit 1 and Unit 2 Division II core spray and high pressure coolant injection (HPCI) compartments and also on the Reactor Building 683 foot elevations. The inspectors verified that adequate procedures were in place to identify and respond to flooding. In addition, the inspectors completed a review of unresolved item (URI) 05000387/2004004-01, “Equipment Hatch Floor Plugs are not Watertight as Indicated in the FSAR.”

b. Findings

URI 05000387/2004004-01, “Equipment Hatch Plugs are not Watertight as Indicated in the FSAR.”

Introduction: A Green self-revealing non-cited violation was identified for not complying with 10 CFR 50, Appendix B, Criterion III, “Design Control.” PPL did not assure that the emergency core cooling system (ECCS) compartments were water tight as described in the FSAR. This resulted in water intrusion into two ECCS compartments simultaneously during an unexpected overflow of the reactor water cleanup backwash receiving tank on August 18, 2004.

Description: On August 18, 2004, a self-revealing finding was identified because the equipment hatch plugs located in the ceiling of the Unit 1 HPCI system compartment and division II core spray system compartment did not meet the FSAR, Revision 51, Section 3.4 design description. The FSAR Section 3.4, stated, in part, that “Redundant engineered safety features, pumps and drives, heat exchangers and associated pipes, valves and instrumentation in the Reactor Building subject to potential flooding are housed in separate watertight rooms." During this event, water entered the ECCS compartments because the equipment hatch plugs in the ceiling of these compartments were not water tight. Two inches of water accumulated on the floor of the Division II core spray and HPCI compartments because the floor drains in these compartments were isolated, as required by plant design. PPL performed walkdowns of the effected equipment areas and determined that none of the equipment in these two compartments were rendered inoperable as a result of the water impacting the equipment. However, operators disabled the Division II core spray system auto start feature for approximately two hours until the equipment walkdowns were completed.

Large equipment hatches are installed in the ceilings of all the ECCS rooms to allow for removal of equipment during maintenance activities. The equipment hatches are normally required to be closed by the use of large equipment hatch plugs which are placed in these openings. PPL reviewed the design of the equipment hatch plugs after the August 18, 2004 event and determined that the plugs would not provide a water tight seal between the plug and the surrounding floor, even though the FSAR, Section 3.4, describes the compartments as watertight. In addition, the plant specific analysis did not consider any leakage through the equipment hatch plugs. The analysis accounted for leakage around doors and through the floor drain system but did not account for leakage through the equipment hatch plugs or sealed penetrations. Therefore, the analysis assumed the equipment hatch plugs were water tight.

PPL has performed plant configuration verification walkdowns and re-performed numerous flooding analysis and concluded that not having watertight equipment hatch plugs, would not have resulted in placing the plant outside of the design basis. However, PPL did performed a plant "Minimod," ECO 601198, "Use of Sealant Around Hatches & Floor Plugs in Reactor Building," to ensure the equipment floor plugs above these compartments, would not leak.

<u>Analysis</u>: This finding is greater than minor because it is associated with the Mitigating Systems Cornerstone of design control and affects the cornerstone's objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e. core damage). Specifically, the equipment hatch plugs were not consistent with the FSAR, Revision 51, Section 3.4, "Water Level Flood Design," to be water tight. This allowed water to enter the Unit 1, Division II core spray and HPCI compartments and reduced the capability of the Division II core spray system because the system auto start feature was manually disabled for approximately two hours. The inspectors performed a Phase 1 screening using IMC 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations." The finding was determined to be of very low safety significance (Green) because this design deficiency did not result in a loss of operability in accordance with Part 9900, Technical Guidance, "Operability Determination Process for Operability and Functional Assessment."

<u>Enforcement</u>: 10 CFR 50, Appendix B, Criterion III, "Design Control," requires in part that, "Measures shall be established to assure that applicable regulatory requirements and the design basis, as defined in 10 CFR 50.2 and as specified in the license

application, for those structures, systems, and components... are correctly translated into specifications, drawings, procedures, and instructions." Contrary to the above, the design basis information in the license application was not correctly translated into the procedures and instructions in that PPL's FSAR, Section 3.4, Revision 51, stated, in part, that "Redundant engineered safety features, pumps and drives, heat exchangers and associated pipes, valves and instrumentation in the Reactor Building subject to potential flooding are housed in separate watertight rooms." The equipment hatch plugs in the ceiling of the ECCS compartments were not water tight and on August 18, 2004, water entered two ECCS compartments simultaneously. Because this failure to comply with 10 CFR 50, Appendix B, Criterion III, is a very low safety significance and has been entered into PPL's corrective action program, (CR 768233), this violation is being treated as a non-cited violation (NCV), consistent with Section I.A.1 of the NRC Enforcement Policy: **NCV 05000387/20060002-01 Equipment Hatch Plugs are not Watertight as Indicated in the FSAR**

1R07 Heat Sink Performance (71111.07 - 1 Sample)

a. Inspection Scope

The inspectors reviewed PPL's inspection, cleaning, and maintenance activities, and reviewed PPL's evaluation of the as-found conditions for the common emergency service water control structure chiller heat exchanger (0S117). The inspectors verified whether PPL properly evaluated the results to identify adverse trends and ensure adequate heat transfer capabilities. The inspectors compared their observations against PPL's procedures and specifications to assess whether the heat exchanger was capable of performing its safety function under design basis accident conditions.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08 - 4 Samples)

a. Inspection Scope

The inspector observed selected samples of in-process nondestructive examination (NDE) activities. Also, the inspector reviewed documentation of additional samples of NDE and component repair activities which involved welding processes. The sample selection was based on the inspection procedure objectives and risk priority of those components and systems where degradation would result in a significant increase in risk of core damage. The observations and documentation review were performed to verify activities were performed in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requirements.

The inspector observed automatic and manual ultrasonic (UT), and visual (VT) testing activities to verify effectiveness of the examiner, test equipment and process in identifying degradation of risk significant systems, structures and components and to evaluate those activities for compliance with the requirements of ASME Section XI of the Boiler and Pressure Vessel Code.

The inspector noted that nonconforming conditions were repaired and not dispositioned “accept as is” for continued service. For example, two relevant indications in the Unit 1 steam dryer identified as cracks were scheduled for grinding to remove the cracks and welding as the repair method. The preparations for welding these areas under water and the associated weld procedure and planning were reviewed by the inspector. The inspector also reviewed the joint process control instructions, welding instructions and welding procedure specification for the steam dryer repairs by welding.

The inspector observed the calibration for and manual ultrasonic examination (UT) test of the reactor recirculation (RR) pump chemical decontamination 4" diameter line connection field welds VRRB311-14-G and F for the 1A recirculation pump. One of these welds is from a weld-o-let to 4" diameter pipe and the other is between the 4" pipe and a blind flange. These welds were scheduled for inspection as followup to an operating experience finding of a fatigue crack at this location at another plant.

The inspector reviewed the automatic UT testing and analysis of reactor pressure vessel (RPV) recirculation system nozzle N1A to reactor vessel shell weld to confirm that the ASME Code requirements were being met. The examination confirmed that a small acceptable weld inclusion identified in 1997 had not changed in dimension. In addition, the inspector observed the process of scanning a vessel weld, the documentation of the transducers used and the coordination of scan position to weld location.

At the time of the inspection, no radiography of welds had been performed during the outage, therefore no radiographs were reviewed.

The inspector reviewed a sample of video recordings of the remote in-vessel visual inspection (IVVI) of the steam dryer base metal, structural welds, and tie bar welds. This review was conducted to confirm the examiner skill, test equipment capabilities, examination technique, and environment (water clarity) enabled the performance of the visual examination of the selected vessel internals. The inspector concluded that this remote visual examination met the requirements of ASME Section XI.

The PPL visual inspection, of component supports and snubbers was sampled. This included review of the DBB1201-H18 sway strut and the adjacent DBB1201-H11 strut where identified unacceptable conditions resulted in an expansion of the scope of visual examinations in accordance with the ISI program. Both of the strut conditions were corrected and evaluations verified that operability of the related components had not been compromised by the as-found conditions.

b. <u>Findings</u>

No findings of significance were identified.

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

Resident Inspector Quarterly Review

a. Inspection Scope

On February 15, 2006, the inspectors observed licensed operator performance in the simulator during operator requalification training. The inspectors compared their observations to Technical Specifications, emergency plan implementation, and the use of emergency operating procedures. The inspectors also evaluated PPL's critique of the operators' performance to identify discrepancies and deficiencies in operator training. The following training scenario was observed:

- Licensed Operator Requalification Scenerio OP 002-06-03-05, "Automatic Depressurization System Rapid Depressurization and Spray of Containment"

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12Q - 3 Samples)

a. Inspection Scope

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

Equipment Issues

- Units 1 and 2 rod drive control systems 1(2)56;
- Unit 1 refuel platform and system (A1 system) due to functional failures; and
- Unit 2 multiple functional failures for control rod drive (CRD) hydraulic system due to CRD flow controller power supply failure on February 22, 2006.

b. Findings

Introduction: The inspectors identified a green finding for a failure to implement Corrective Action procedure, NDAP-QA-702, which requires that actions to correct and prevent recurrence be completed before the closure of a condition report. Following electrolytic capacitor failures at Susquehanna corrective actions were not completed which directly contributed to loss of control rod drive hydraulic flow on February 22, 2006.

Description: On February 22, 2006, an electrolytic capacitor failure in the Unit 2 control rod drive flow control power supply caused the closure of the CRD flow control valve and the loss of all valve position and flow indication. The power supply failure caused control rod drive cooling flow to be reduced from 60 to 20 gpm which resulted in high temperature alarms on 20 control rod drives. PPL declared the control rod drives with high temperatures “slow” and entered Technical Specification limiting condition for operation (LCO) 3.1.4. which required a unit shutdown within 12 hours. Declaring the 20 control rods “slow” was consistent with vendor guidance which describes the potential for two phase flow (steam flashing) where the CRD hydraulic control unit over piston pipe enters the scram discharge volume. Steam flashing is expected to increase the scram time for control rods with temperatures above 350 degrees Fahrenheit.

PPL identified corrective actions to address the aging of electrolytic capacitors in 2002 following a previous component failure in the reactor core isolation cooling system (RCIC) and a review of published industry operating experience. PPL developed a list of components containing aluminum electrolytic capacitors; however, that list did not appropriately identify the CRD flow control power supply as an electrolytic capacitor component. In response to the Level 1 root cause evaluation (RCE), PPL identified uncertainty in the preventive maintenance adequacy of numerous components containing electrolytic capacitors and developed a plan to address the aging and reliability of these components. The inspectors identified that the corrective action plan to verify appropriate preventive maintenance or perform “run to failure” evaluations were not performed. PPL missed another opportunity to address this issue when additional capacitors failed in 2004 which resulted in a Condition Report (CR) that described an increase in component failure rate due to capacitor degradation. The inspectors found that this CR was closed in 2005 after PPL incorrectly concluded that the corrective actions from the previous plan (2002) were complete and would address this issue

Analysis: This finding is greater than minor because it is associated with the equipment performance attribute of the Mitigating Systems Cornerstone and the finding negatively affected the cornerstone objective to ensure the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. The finding was determined to be of very low significance (Green) since the finding does not represent a loss of safety function and is not potentially risk significant due to external events.

This finding is related to the problem identification and resolution cross-cutting area because PPL did not implement corrective actions in accordance with the corrective action program procedure.

Enforcement: Since the control rod drive flow control valve is not safety-related, there were no violations of NRC requirements. PPL's failure to follow the corrective action procedure directly contributed to the electrolytic capacitor power supply failure which reduced the CRD flow, increasing the control rod scram times on 20 control rods, which negatively impacted the mitigating systems cornerstone on February 22, 2006. PPL entered this issue in the corrective action program (CR# 758337): **FIN 05000388/2006002-02, Incomplete Corrective Actions contribute to CRD Flow control failure**

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13 - 7 Samples)

a. Inspection Scope

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

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

- Units 1 and 2 "C" EDG failure on low engine lube oil and system outage to repair jacket water leak;
- Units 1 and 2 "C" EDG output breaker replacement and modification, 1A20304;
- Units 1 and 2 condensate transfer / ECCS Keep Fill freeze seal PCWO 665043;
- Units 1 "C" 4 KV ES bus undervoltage relay testing, ODM # 750645;
- Unit 1 "A" RHR pump auto start time delay relay PCWO 482599;
- Unit 1 HV151 F004 D, RHR suppression pool suction valve did not open/close during functional logic test; and
- Unit 1 primary containment integrated leak rate test (ILRT), orange risk for decay heat removal (DHR) and yellow risk for vessel inventory make up.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

For the non-routine events described below, the inspectors reviewed operator logs and plant data to determine the sequence of events, how the operators responded, and to determine if the response was in accordance with plant procedures.

Unit 2 Sequence Exchange

On February 12, 2006, at 4:10 am, a control rod sequence exchange was being performed on Unit 2, when a licensed reactor operator noted a discrepancy between the existing control rod pattern and the sequence exchange paperwork. The inspectors reviewed PPL's response to the identified discrepancy including core thermal limits, fuel preconditioning and core stability to verify fuel integrity and Technical Specification limits were met. The inspectors also observed PPL's recovery actions and directly observed completion of the Unit 2 sequence exchange. PPL determined that the final sequence exchange plans were not completely incorporated into the sequence exchange package that was utilized during the sequence exchange and the method used to review the package was not thorough. PPL completed the sequence exchange on February 12, 2006 at 12:25 pm. The safety significance of this event was very low because the core remained with the required core operating limits during the sequence exchange.

ALERT Declaration Due to a Fire Suppression System (Halon) Discharge

On March 1, 2006, at 9:27 pm, PPL declared an ALERT as the result of a fire protection system actuation which automatically discharged the fire suppressant Halon into an occupied building forcing an evacuation of the work area. PPL activated its emergency response centers as required. The Senior Resident Inspector responded to the site to monitor PPL's response and confirm the proper event classification and notifications. The ALERT was exited at 4:33 am. on March 2, 2006, after PPL cleared the area of the Halon gas, deactivated the Halon system, and reset the fire panel and dampers. Section 4OA3, “Event Follow-up,” provides additional information regarding this event.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15 - 6 Samples)

a. Inspection Scope

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

- CR 740488, “B” standby gas treatment system differential pressure (DP) transmitter out of calibration;
- CR 745751, “A” emergency diesel generator (EDG) digital reference unit (DRU) Part 21 Report #2006-01-01;
- CR 745569 and CR 747105, “A” division core resistant mechanism (CRM) return line isolation valve (PCIV);
- CR 751588, Isolation of Battery Room Ventilation;
- CR 763707, 763397, and 763398, Refuel Bridge Hoist Tube Hang Ups; and
- CR 764737, reactor protection system (RPS) and alarm HFA relay termination incorrect.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17 - 2 Samples)

a. Inspection Scope

The inspectors reviewed the system design package and the associated design and licensing documents for (1) ECO 618882, Unit 1 Power Range Neutron Monitoring System and (2) ECO 696686, Unit 1 RHR and Recirculation Instrument Tube Relocation Modification. The Average Power Range Monitor portion of the Power Range Neutron Monitoring System was reviewed. Field implementation activities were observed and compared to the design requirements and installation standards. The inspectors reviewed the results of post modification testing and verified that the affected procedures and design basis documents were appropriately updated.

b. Findings

No findings of significance were identified.

1R19 <u>Post Maintenance Testing</u> (71111.19 - 6 Samples)

a. <u>Inspection Scope</u>

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

- Units 1 and 2 replacement of "A" emergency diesel generator (EDG), governor digital reference unit (DRU) in response to 10 CFR Part 21 #2006-01-01, PCWO 746574;
- Unit 1 testing of core spray system following F001B, F015B and F031B valve maintenance. PCWO 735842, RTPM 638393 and 467119;
- Unit 1 Bus "D" auxiliary undervoltage relay replacement PCWO 760163;
- Unit 1 reactor core isolation cooling (RCIC) overspeed trip mechanism inspection/replacement PMT (TP-150-004);
- Unit 1 "B" main steam isolation valve 22B actuator replacement, CRA 763207; and
- Unit 2 containment radiation monitor valve SV-257102A position and operator repair including local leak-rate test (LLRT), PCWO 742085.

b. <u>Findings</u>

No findings of significance were identified.

1R20 <u>Refueling and Other Outage Activities</u> (71111.20 - 1 Sample)

<u>Refueling Outage</u>

a. <u>Inspection Scope</u>

The inspectors reviewed the outage risk management plan for the Unit 1 refueling outage, conducted March 4, to April 12, 2006, to confirm that PPL had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense-in-depth. During the refueling outage, the inspector observed and / or reviewed the outage activities listed below.

- Plant shutdown and cool down activities.
- Outage configuration controls including:
  1) availability and accuracy of reactor coolant system instrumentation;

2) electrical power alignments;
3) decay heat removal system operation, including fuel pool cooling system and supplemental decay heat removal system;
4) availability of reactor inventory makeup water systems; and
5) secondary containment controls and integrity.

- Drywell, suppression chamber, and refuel floor walkdowns after shutdown and prior to final closeout.
- 4 KV emergency buses and supplemental decay heat removal equipment clearance.
- Fuel handling operations including fuel movement, control of reactivity, fuel assembly tracking, and core verification activities.
- Reactor startup, including plant restart reviews, system restoration and testing, preparation for reactor mode changes, control rod withdrawal, reactor criticality, low power physics tests, reactor coolant system heat up, and reactor power increases.

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

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22 - 6 Samples)

a. Inspection Scope

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

- Units 1 and 2 “B” EDG emergency core cooling system (ECCS) start and load reject testing, SE-024-B01;
- Unit 1 main steam isolation valve local leak-rate test (LLRT) (in progress);
- Unit 1 source range monitor signal to noise and functional test, SI-178-231 and SI-178-215 B and D;
- Unit 1 integrated leak rate test of primary containment, SE-100-003;
- Unit 2, Loop B core spray flow verification, SO-251-B02 inservice testing (IST);and
- Unit 2 reactor core isolation cooling (RCIC) pump and flow verification, SO-250-002.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23 - 1 Sample)

a. Inspection Scope

The inspectors reviewed temporary plant modifications to determine whether the temporary changes adversely affected system or support system availability, or adversely affected a function important to plant safety. The inspectors reviewed the associated system design bases, including the FSAR, Technical Specifications, and assessed the adequacy of the safety determination screenings and evaluations. The inspectors also assessed configuration control of the temporary changes by reviewing selected drawings and procedures to verify whether appropriate updates had been made. The inspectors compared the actual installations to the temporary modification documents to determine whether the implemented changes were consistent with the approved documents. The following temporary modification and documents were included in the review:

- PCWO 665043 installation of two freeze seals on condensate transfer.
- Drawing M 108, Sheet 1, Condensate and Refueling Water System.
- OP-037-001, Demineralized and Condensate Transfer Water System.
- ON-037-001, Loss of Condensate Transfer System.
- RLWO 746060, Replace 008099-Refueling Water Pump Suction Condensate Transfer Supply Isolation Valve.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation (71114.06 - 1 Sample)

a. Inspection Scope

On January 24, 2006, the inspectors observed a full emergency plan drill. The inspectors assessed licenced operator adherence to emergency plan implementing procedures, and their response as well as the Technical Support Center and Emergency Operations Facility staff response to simulated degraded plant conditions to identify weaknesses and deficiencies in classification and notification. The inspectors observed PPL's critique of the Technical Support Center and reviewed the final drill critique presentation to senior management to evaluate PPL's identification of weaknesses and deficiencies. Inspectors reviewed participant rosters to compare against the participant and drill controllers for this same drill scenerio which was previously performed in

August of 2004. The inspectors compared PPL's identified findings against the inspectors' observations to determine whether PPL adequately identified problems.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01 - 7 Samples)

a. Inspection Scope

Based on PPL's schedule of work activities during the Unit 1, 14^{th} refueling and inspection outage (Unit 1-14th RIO), the inspectors selected three jobs being performed in radiation areas, airborne radioactivity areas, or high radiation areas (<1 R/hr) for observation; reviewed radiological job requirements (radiation work permit [RWP] requirements and work procedure requirements); observed job performance with respect to these requirements; and, determined that radiological conditions in the work area were adequately communicated to workers through briefings and postings. The jobs reviewed were: scaffold work in the drywell; nozzle and vessel inservice inspection (ISI); and, ISI piping/hangers/erosion-corrosion outside of bioshield doors.

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

During job performance observations, the inspectors observed radiation worker performance with respect to stated radiation protection work requirements and determined that they were aware of the significant radiological conditions in their workplace, the RWP controls/limits in place, and that their performance took into consideration the level of radiological hazards present.

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

The inspectors identified exposure significant work areas within radiation areas, high radiation areas (<1 R/hr), or airborne radioactivity areas in the plant and reviewed associated controls and surveys of these areas to determine if controls (e.g. surveys, postings, barricades) were acceptable.

The inspectors walked down these areas or their perimeters to determine: whether prescribed RWP, procedure, and engineering controls were in place; whether PPL's surveys and postings were complete and accurate; and, whether air samplers were properly located.

The inspectors reviewed RWPs used to access these and other high radiation areas and identified what work control instructions or control barriers had been specified. The inspectors reviewed electronic personal dosimeter alarm set points (both integrated dose and dose rate) for conformity with survey indications and plant policy.

The inspectors attended planning meetings, reviewed the radiological controls (including the RWP and controlling procedures) implemented, attended the pre-job briefing and made direct observations of the diving operations undertaken in support of steam dryer repairs. The documents reviewed are provided in the Attachment.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors obtained a list of work activities ranked by actual/estimated exposure that were in progress during the current outage and selected the three (3) work activities of highest exposure significance (listed in paragraph 2OS1 above).

The inspectors reviewed the as low as is reasonably achievable (ALARA) work activity evaluations, exposure estimates, and exposure mitigation requirements and determined that PPL had established procedures, engineering and work controls, based on sound radiation protection principles, to achieve occupational exposures that are ALARA.

The inspectors compared the results achieved (dose rate reductions, person-rem used) with the intended dose established in PPL's ALARA planning for these work activities. The documents reviewed are provided in the Attachment.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors verified the calibration expiration and source response check currency on radiation detection instruments staged for use. The documents reviewed are provided in the Attachment.

b. <u>Findings</u>

No findings of significance were identified.

Cornerstone: Public Radiation Safety

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

a. <u>Inspection Scope</u>

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

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

The inspector observed the routine processing (including sample collection and analysis) and release of radioactive liquid waste to verify that appropriate treatment equipment is used and that radioactive liquid waste is processed and released in accordance with procedure requirements; observed the sampling and compositing of liquid effluent samples. In lieu of direct observation, review several radioactive liquid waste release permits, including the projected doses to members of the public. The inspector also observed the routine processing (including sample collection and analysis) and release of radioactive gaseous effluent to verify that appropriate treatment equipment is used and that the radioactive gaseous effluent is processed and released in accordance with RETS/ODCM requirements.

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

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

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

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

The inspector reviewed records of instrument calibrations performed since the last inspection for each point of discharge effluent radiation monitor and flow measurement device and reviewed any completed system modifications and the current effluent radiation monitor alarm setpoint value for agreement with RETS/ODCM requirements. The inspector also reviewed calibration records of radiation measurement (i.e.,counting room) instrumentation associated with effluent monitoring and release activities and reviewed quality control records for the radiation measurement instruments.

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

The inspector reviewed PPL's Licensee Event Reports, Special Reports, audits, and self assessments related to the RETS/ODCM program performed since the last inspection. The inspector determined that identified problems were entered into the corrective action program for resolution. The inspector also reviewed corrective action reports affecting environmental sampling, sample analysis, or meteorological monitoring instrumentation.

b.

c. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems (71152)

1. Review of Items Entered into the Corrective Action Program

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

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

Section 1R12 describes a finding where PPL did not complete corrective actions for the aging of electrolytic capacitors. These corrective actions were developed in 2002 for numerous plant components to address industry operating experience as well as plant specific experience. Not completing corrective actions resulted in another failure which degraded system mitigation functions. Since the cause of this finding was not completing corrective actions, this finding is related to the Problem Identification and Resolution cross-cutting area.

3. Inservice Inspection Activities (71111.08)

a. Inspection Scope

The inspector reviewed the PPL Quality Assurance Audit No. 584591 performed in late 2005 of ISI activities and selected three (3) of the related ARs (717167, 738741 and 738793) for followup to confirm that identified problems were adequately documented with corrective actions initiated. Also, the inspector evaluated effectiveness in the resolution of problems identified during ISI activities.

The inspector reviewed a sample of corrective action reports shown in the attachment, which identified nonconforming conditions discovered during this outage. The inspector verified that flaws and other nonconforming conditions identified were reported, characterized, evaluated and appropriately dispositioned and entered into the corrective action program.

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up (71153 - 1 Sample)

1. Section 1R14 discusses the March 1, 2006, ALERT as the result of a fire protection system actuation which automatically discharged the fire suppressant Halon into an occupied building. In addition to the initial response, the inspectors also followed PPL's troubleshooting and equipment restoration activities. Inspectors monitored command and control as well as the coordination of activities during the transition from the PPL emergency organization back to the normal line management and non-emergency processes. Inspectors collected and communicated event details to Region I management to determine the appropriate agency follow up response in accordance with the risk based and deterministic criteria in Management Directive 8.3.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On April 7, 2006, the resident inspectors presented the inspection results to Mr. R. Saccone, Vice President - Nuclear Operations, and other members of your staff, who acknowledged the findings. The inspectors asked PPL whether any of the material examined during the inspection should be considered proprietary. No information was identified.

4OA7 Licensee-identified Violations

None.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

F. Hickey, Nuclear Chemistry Health Physicist
B. Rhoads, Manager - Plant Chemistry
L. Vnuk, Senior Chemist
R. Kessler, Health Physicist - ALARA
V. Schuman, Radiological Protection Manager
J. Henlsel, Manager - Nuclear Operations
R. Sgarro, Manager - Regulatory Affairs
D. D'Angelo, Manager - Station Engineer
G. Ruppert, Manager - Maintenance
V. Schuman, Manager - Radiation Program
W. Morrissey, Supervising Engineer, Regulatory Affairs
D. Brophy, Senior Engineer, Regulatory Affairs
A. Fitch, Assistant Operations Manager
J. Hirt, Supervisor, Reactor Engineering
R. Bosche, Supervisor - Operations Training

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None.

Opened and Closed

05000387/2006002-001	NCV	Equipment Hatch Plugs are not Watertight as Indicated in the FSAR
05000388/2006002-002	FIN	Incomplete Corrective Actions Contribute to CRD Flow Control Failure

Closed

05000387/2004-004-001	URI	Equipment Hatch Plugs are not Watertight as Indicated in the FSAR

LIST OF DOCUMENTS REVIEWED
(Not Referenced in the Report)

Section 1R01: Adverse Weather Protection

PCAF 2006 - 1086
OI-AD-029, Emergency Load Control
EP-TP-001, Susquehanna EAL bases document
Operations turnover sheets
AR 758005, AR 754193, AR 754085 and AR 754087

Section 1R04: Equipment Alignment

OP-255-001, Control Rod Drive Hydraulic System
TSTF-404A, Rev 0, BWROG83
Maintenance rule basis document - system 55
PI&D drawings, M 2146 and M 2147, control rod drive
AR/CR 694131, crack in bottom half case of SDHR pump #1
AR/CR 757711, SDHR RAD monitor is out of limits specified in SO-100-007

Section 1R06: Flood Protection Measures

ON-235-001, Rev. 25, Loss of Fuel Pool Inventory
ON-269-002, Rev. 2, Flooding in Reactor Building
LA-2206-001, Rev. 6, Fuel Pool Cooling Control Panel 2C206
AR-214-001, Rev. 22, High Pressure Coolant Injection (HPCI) System 2C601

ECO 601198, Use of Sealant Around Hatches & Floor Plugs in Reactor Building
FSAR Section 3.6, Rev. 61, Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping

AR 760869, Discrepancies in the Flooding Analysis

EC-FLOD-0500, Rev. 2, Re-Evaluate Maximum Flood Depth in Reactor Building Pipe/Penetration Room on Elevation 683
EC-FLOD-0001, Rev. 0, Moderate Energy Pipe Breaks - Floods
EC-012-6047, Rev. 0, Removal of Floor Plugs in Conditions 4 or 5

Section 1R07: Heat Sink Performance

M 1453, Heat Exchanger Tube Plugging
H-1001, Heat Exchanger/Condenser Tube Cleaning

Section 1R08: Inservice Inspection Activities

Condition Reports/Action Reports
AR 760476 Support
CRA 760489 Sample Expansion
CR 554430 DBB1201-H18
CR 738741 Records retention process
CRA 757003 Interim ISI records control plan
AR 760475 UT couplant
AR 759273 Pipe socket welds rejected during ISI UT inspection
AR 759221 Pipe socket welds rejected during ISI UT inspection
AR 758819 Steam Dryer cracked condition at the weld VS-B-2 weld
AR 758849 Steam Dryer cracked condition by the 220 deg lifting lug

Procedures and Plans

PPL ISI Program Plan for the Third Ten-Year Inspection Interval, revision 1.
NEPM-QA-1154, Rev 4, page 7. Control of Contractor's NDE qualification and certifications.
P50YP244, R10. Process Specification for Structural Arc Welds in All-Water Environment

Radiograph Review

N/A

NDT Examination Reports

VT-06-570	VT-3 OF DBB1201-H18
VT-06-572	VT-3 OF DBB1201-H11
UT-06-079	UT of VRRB311-14-F, weld-o-let to 4" pipe, 70 deg
UT-06-080	UT of VRRB311-14-F, weld-o-let to 4" pipe, 45 deg
CNF IVVI-06-05	VT identified tack weld crack at 270 deg. Per Exam VT-06-009
CNF IVVI-06-41	VT identified weld crack at 220 deg. Per Exam VT-06-123
CNF IVVI-06-51	VT identified crack near weld VS-A-1, per Exam VT-06-116
CNF IVVI-06-43	VT identified crack near weld VS-B-2, per Exam VT-06-119

NDT Examination Procedures

NDE-UT-020 R2	Automated Ultrasonic Examination of reactor pressure nozzle inner radius and nozzle to vessel welds with the GERIS 2000OD in accordance with Appendix VIII, (GE ID # GE-UT-705 Ver. 5)
NDE-UT-025 R2	Examination of reactor pressure vessel welds with the GERIS 2000OD in accordance with Appendix VIII, (GE ID # GE-UT-704 Version 8)
NDE-UT-001 R6	Manual ultrasonic examination of austenitic pipe welds for IGSCC
NDE-UT-005 R1	Manual ultrasonic examination of vent, drain line & other socket welds
NDE-VT-003 R5	Visual Examination VT-3
NDE-VT-005 R4	Underwater Visual Examination of RPV Internals including the Steam Dryer

Repair-Replacement or NDE related Work Orders

PCWO 667693 Replacement of a socket weld with a welded plug to prevent leakage.
PCWO 676588 ISI preparation and restoration work for VRRB311-14-F & G and 312-3-F & G.
WO 758846 U1 Steam Dryer Repair by Wet Underwater Welding

Welding Procedure Specification

WPS N-516G-3, R7 Welding Procedure Specification for Steam Dryer repair welding.

Section 1R11: Surveillance Testing

ANSI/ANS-56.8-1994, American National Standard for Containment System Leakage Testing Requirements
NRC Regulatory Guide 1.163, Performance-Based Containment Leak-Test Program
TP-159-017, Integrated Leak Rate Test Valve Lineup

IR12: Maintenance Effectiveness

ON-200-006, Loss of reactor heat balance calculation
ON-255-007, Loss of CRO system flow
ESC122K600, CRD instrument power supply design data - NIMS
CR 754044, 753990 and 758337
PCWO 753992, loss of CRD flow indication in control room and Reactor Building
GE SIL No. 173 Supplement 1, September 20, 1999
EWR 363472, end of life failures of aluminum electrolytic capacitors

1R14: Operator Performance During Non-Routine Evolutions and Events

Unit 2 Sequence Exchange
ON-255-001, Control Rod Problems
RE-0TP-103, Attachment A, Pre-Sequence Exchange/Power Maneuver Checklist
AR 751433, Initial rod pattern used to develop sequence exchange sheets difference from actual rod pattern.
Operations Directive 06-02
Operator Logs

ALERT Declaration due to a Fire Suppression System (Halon) Discharge
Security Incident Report 06-03-03
ARs 755777, 756076
EP-TP-001, EAL Classification Levels
EP-PS-101, TSC Emergency Director
Operator Logs

Section 1R15: Operability Evaluations

OFR 746091
MT-158-B01, RPS A2 Channel HFA relay contact inspection
MT-158-B01, RPS B1 channel HFA relay contact inspection
ON-159-002, Containment Isolation

Section 1R17: Permanent Plant Modifications

TS Amendment No. 230, Power Range Neutron Monitoring System Digital Upgrade.
TS 3.3.1.1, Reactor Protection System (RPS) Instrumentation
TS 3.3.1.3, Oscillation Power Range Monitor Instrumentation
TS 3.3.2.1, Control Rod Block Instrumentation
TS 3.4.1, Recirculation Loops Operating

Calculations:
EC-078-1016, Setting and Setpoints for GE NUMAC OPRM Setup and Tuning
EC-078-0511, APRM Flow Biased Scram / Rod Block Setpoint Calculation (APRM 1-APRM 4)
EC-FUEL-1024, SSES Power / Flow Maps with Power Uprate Conditions

Operator Lesson Plans
Replacement GE NUMAC Power Range Neutron Monitoring (PRNM) System
Replacement GE NUMAC Power Range Neutron Monitoring (PRNM) System - Modification Update and Technical Specifications

Installation Documents:
JIP-2801, I&C Installation Procedure for DCP 618882, Install NUMAC Power Range Neutron Monitoring System Unit 1
Site Installation Procedure for Modification Package ECO 696686
PCWO 705772, 700626, “Instrument Tube Routing”

Testing Documents:
SE-149-400, RHR System Leak Quantification Test
TP-178-040, Post Modification Test for PRNMS Operability (DCP-618882)

Section 1R19: Post Maintenance Testing

SE-024-A01, Diesel Generator A Integrated Surveillance Test
PCWO 74674, replace the DRU installed in the “A” DG
SO-251-B02, Core Spray Flow Verification Loop B
SO-151-B02, Core Spray Flow Verification B Loop
SO-151-B04, Quarterly Core Spray Valve Exercising Division II
SO-151-014, Core Spray System Cold Shutdown Valve Exercising
SO-151-015, 24 Month Core Spray System Remote Position Indicator Checks

Section 1R20:Refueling and Other Outage Activities

GO-100-002, Plant Startup, Heatup and Power Operation
GO-100-004, Plant Shutdown to Minimum Power
GO-100-005, Plant Shutdown to Hot/Cold Shutdown
GO-100-006, Cold Shutdown, Defueled and Refueling
Unit 1 Cycle 14A Shutdown Control Rod Sequence
OP-149-002, RHR Shutdown Cooling
SE-104-202, 24 Month 4.16 KV Class 1E Bus 1D (1A204) Offsite Supply Transfer Check
SE-149-002, 24 Month RHR Logic System Functional Test (Div. 2) -Outage (Partial)
TP-105-009, Load Center 1B240 Outage Coordination Procedure
SE-124-207, Unit 1 Division 2 Diesel Generator LOCA LOOP Test
B & D ESW timer relay failures during SE-124-207 (EWR 767582)
OT-155-001, Control Rod Drive Exercising in mode 3, 4 or 5
OT-149-005, Miscellaneous Flushes of RHR Piping Outside Containment
SE-100-002, ASME Class I Boundary System Leakage Test
SE-100-002, Attachment Y and Attachment W Data Sheets
CR 766285
AR 764456, 763877 and 764314
Event Notifications: 42407, 42385 and 42423
MT-181-003, Refueling Platform Mechanical Inspection and Testing Preoutage
MT-299-001, Reactor Building Crane Operating Procedure
ME-ORF-013, Service Platform Installation and Removal
ORAM Risk Color Output Basis, M. Adelizzi, 2/20/06
NDAP-QA-507, Conduct of Refuel Floor
NDAP-QA-503, General Housekeeping and Foreign Material Control
NDAP-QA-0412, Leakage Rate Test Program
NDAP-QA-0302, System Status and Equipment Control
NUREG 0612, Control of Heavy Loads
Core verification, video and final loading pattern.
SR-100-008, In-Sequence Critical and Shutdown Margin Demonstration
SR-155-004, Scram Time Measurement of Control Rods
SR-178-012, LPRM Calibration and Validation
AR 750606, Risk Management Action Plan for Unit 1 Maintenance Activities on Bus 1C
Startup Plant Operating Review Committee meeting (Startup PORC) 3/31/06

Section 1R22: Surveillance Testing

SE-100-003, Primary Containment Integrated Leakage Rate Test
SI-164-307, Semi-Annual Calibration of Recirculation Flow Unit B
SI-164-307, Semi-Annual Calibration - Reactor Recirculation Flow Unit 1 B
SO-150-002, RCIC Flow Verification
SI-164,307, EC-INST-1284, I&C Maintenance Calculation for Flow Unit 1B, FYB311K607B, ICC Flow Unit 1B-1
SI-164-307, EC-INST-1286, I&C Maintenance Calculation for Flow Unit 1B-2

Section 2OS: Occupational Radiation Safety

RWPs & ALARA Reviews:
2006-1320; 2006-1360; 2006-1370; 2006-1372; 2006-1383; 2006-1006

Action Reports (AR):
733098; 732539; 750309; 755414; 755362; 756718; 756780; 756786; 756788; 756791; 757080; 757729; 757709; 757605; 758330; 758999; 759216

Section 2PS1: Radioactive Gaseous and Liquid Effluent Treatment and Monitoring Systems

Susquehanna Steam Electric Station 2004 Annual Radioactive Effluent Release Report
Susquehanna Steam Electric Station Offsite Dose Calculation Manual (2005 revision)
Susquehanna Steam Electric Station Technical Requirements Manual (2005 revision)
Calibration/testing records for the following systems/components:

Effluent Monitors
liquid radwaste discharge radiation monitor
Units 1 and 2 service water radiation monitors
Units 1 and 2 RHR service water radiation monitors
Reactor Building ventilation radiation monitors (low range noble gas & accident channels)
Turbine Building ventilation noble gas monitor (low range noble gas & accident channels)
standby gas treatment system radiation monitors (low range noble gas & accident channels)

Flow Measurement Device
liquid radwaste effluent flow monitor
cooling tower discharge flow monitor
Reactor Building ventilation purge noble gas flow monitor
Turbine Building ventilation purge noble gas flow monitor
standby gas stack flow and sampler flow rate monitors
Quality Assurance Internal Audit Report No. 527452
Self-Assessment Reports Nos. CHM-03-02, CHM-03-05, CHM-05-07
Interlaboratory Cross-Check Results (Analytics) 1st-4th Quarter 2005
Laboratory Germanium Counting System Control Charts (January 22-February 20, 2006)
Monthly Radiological Effluent Surveillance (January 2006)
2005 Liquid Effluent Dose Report
2005 Gaseous Effluent Dose Report
Liquid Effluent Discharge Permits 06-004 & 06-005
Weekly Vent Sample Surveillance:
- Unit 1 Reactor Vent (1/10/06)
- Unit 1 Turbine Building Vent (1/10/06)
- Unit 2 Reactor Vent (1/10/06)
- Unit 2 Turbine Building Vent (1/10/06)
- Standby Gas Treatment Vent (1/10/06)

Tritium and Noble Gas Effluent Samples:
Unit 1 Reactor Building (1/3/06: 1/5/06)
Unit 2 Reactor Building (1/3/06: 1/5/06)
Standby Gas Treatment (1/4/06)

LIST OF ACRONYMS

ALARA	As Low As Is Reasonably Achievable
ASME	American Society of Mechanical Engineers
CFR	Code of Federal Regulations
CR	Condition Report
CRD	Control Rod Drive
CRM	Core Resistant Mechanism
DHR	Decay Heat Removal
DP	Differential Pressure
DRU	Digital Reference Unit
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EDO	Executive Director for Operations (NRC)
ESW	Emergency Service Water
FSAR	[SSES] Final Safety Analysis Report
HPCI	High Pressure Coolant Injection
ILRT	Integrated Leak Rate Test
IMC	Inspection Manual Chapter
ISI	Inservice Inspection
IST	Inservice Testing
IVVI	In-Vessel Visual Inspection
KV	Kilovolts
LCO	Limiting Condition for Operation
LLRT	Local Leak-Rate Test
NCV	Non-cited Violation
NDAP	Nuclear Department Administrative Procedure
NDE	Nondestructive Examination
NRC	Nuclear Regulatory Commission
ODCM	Offsite Dose Calculation Manual
PCWO	Plant Component Work Order
PI	[NRC] Performance Indicator
PI&R	Problem Identification and Resolution
PMT	Postmaintenance Testing
PPL	PPL Susquehanna, LLC
PCIV	Primary Containment Isolation Valve
QA	Quality Assurance
RCE	Root Cause Evaluation
RCIC	Reactor Core Isolation Cooling
RETS	Radiological Effluent Technical Specification
RHR	Residual Heat Removal

RPV	Reactor Pressure Vessel
RPS	Reactor Protection System
RR	Reactor Recirculation
RTP	Reactor Thermal Power
RWP	Radiation Work Permit
SBGT	Standby Gas Treatment
SDP	Significant Determination Process
SDHR	Decay Heat Removal System
SSC	Structures, Systems, and Components
SSES	Susquehanna Steam Electric Station
TS	Technical Specification
URI	Unresolved Item
UT	Ultrasonic Testing
VT	Visual Testing
WO	Work Order