UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
2100 RENAISSANCE BOULEVARD, SUITE 100
KING OF PRUSSIA, PENNSYLVANIA 19406-2713

February 25, 2013

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 EVALUATION OF CHANGES, TESTS, AND EXPERIMENTS AND PERMANENT MODIFICATIONS TEAM INSPECTION REPORT 05000387/2012007 AND 05000388/2012007

Dear Mr. Rausch:

On December 14, 2012, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at the Susquehanna Steam Electric Station, Units 1 and 2. The enclosed inspection report documents the inspection results, which were preliminarily discussed on December 14, 2012, with Mr. J. Helsel, Plant General Manager, and other members of your staff. The final inspection results were discussed during a teleconference exit meeting with Mr. R. Franssen, Nuclear General Manager – Engineering, and other members of your staff on January 25, 2013.

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. In conducting the inspection, the team reviewed selected procedures, calculations and records, observed activities, and interviewed station personnel.

Based on the results of this inspection, no findings were identified.

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, Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Paul G. Krohn, Chief
Engineering Branch 2
Division of Reactor Safety

February 25, 2013

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 EVALUATION OF CHANGES, TESTS, AND EXPERIMENTS AND PERMANENT MODIFICATIONS TEAM INSPECTION REPORT 05000387/2012007 AND 05000388/2012007

Dear Mr. Rausch:

On December 14, 2012, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at the Susquehanna Steam Electric Station, Units 1 and 2. The enclosed inspection report documents the inspection results, which were preliminarily discussed on December 14, 2012, with Mr. J. Helsel, Plant General Manager, and other members of your staff. The final inspection results were discussed during a teleconference exit meeting with Mr. R. Franssen, Nuclear General Manager – Engineering, and other members of your staff on January 25, 2013.

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. In conducting the inspection, the team reviewed selected procedures, calculations and records, observed activities, and interviewed station personnel.

Based on the results of this inspection, no findings were identified.

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, Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,
/RA/
Paul G. Krohn, Chief
Engineering Branch 2
Division of Reactor Safety

DOCUMENT NAME: \\nrc.gov\nrc\R1\Office\DRS\Engineering Branch 2\Pindale\SusquehannaModsReport2012007.docx
ADAMS ACCESSION NUMBER: **ML13056A477**

☑ SUNSI Review		☑ Non-Sensitive ☐ Sensitive		☑ Publicly Available ☐ Non-Publicly Available	
OFFICE	RI/DRS	RI/DRP	RI/DRS		
NAME	SPindale	MGray	PKrohn		
DATE	02/06/13	02/08/13	02/25/13		

OFFICIAL RECORD COPY

Docket No. 50-387, 50-388
License No. NPF-14, NPF-22

Enclosure:
Inspection Report 05000387/2012007 and 05000388/2012007
w/Attachment: Supplemental Information

cc w/encl: Distribution via ListServ

Distribution w/encl: (via e-mail)
W. Dean, RA
D. Lew, DRA
D. Roberts, DRP
P. Wilson, DRP
C. Miller, DRS
J. Clifford, DRS
M. Gray, DRP
A. Rosebrook, DRP
J. Ayala, DRP
P. Finney, DRP, SRI
J. Greives, DRP, RI
S. Farrell, DRP, AA
J. Cassidy, RI OEDO
RidsNrrPMSusquehanna Resource
RidsNrrDorlLpl1-2 Resource
ROPreports Resource

U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No:	50-387; 50-388
License No:	NPF-14; NPF-22
Report No:	05000387/2012007 and 05000388/2012007
Licensee:	PPL Susquehanna, LLC (PPL)
Facility:	Susquehanna Steam Electric Station, Units 1 and 2
Location:	Berwick, Pennsylvania
Inspection Period:	November 26 through December 14, 2012
Inspectors:	S. Pindale, Senior Reactor Inspector, Division of Reactor Safety (DRS), Team Leader K. Mangan, Senior Reactor Inspector, DRS J. Richmond, Senior Reactor Inspector, DRS
Approved By:	Paul G. Krohn, Chief Engineering Branch 2 Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000387/2012007 and 05000388/2012007; 11/26/12 - 12/14/12; Susquehanna Steam Electric Station, Units 1 and 2; Engineering Specialist Plant Modifications Inspection.

This report covers a two week inspection of the evaluations of changes, tests, or experiments and permanent plant modifications. The inspection was conducted by three region based engineering inspectors. 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.

No findings were identified.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R17 Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications (IP 71111.17)

.1 Evaluations of Changes, Tests, or Experiments (26 samples)

a. Inspection Scope

The team reviewed three safety evaluations to determine whether the changes to the facility or procedures, as described in the Updated Final Safety Analysis Report (UFSAR), had been reviewed and documented in accordance with Title 10 of the *Code of Federal Regulations* (CFR) Part 50.59 requirements. In addition, the team evaluated whether PPL Susquehanna, LLC (PPL) had been required to obtain U.S. Nuclear Regulatory Commission (NRC) approval prior to implementing the changes. The team interviewed plant staff and reviewed supporting information including calculations, analyses, design change documentation, procedures, the UFSAR, the Technical Specifications (TS), and plant drawings to assess the adequacy of the safety evaluations. The team compared the safety evaluations and supporting documents to the guidance and methods provided in Nuclear Energy Institute (NEI) 96-07, "Guidelines for 10 CFR 50.59 Evaluations," as endorsed by NRC Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments," to determine the adequacy of the safety evaluations.

The team also reviewed a sample of twenty-three 10 CFR 50.59 screenings for which PPL had concluded that a safety evaluation was not required to be performed. These reviews were performed to assess whether PPL's threshold for performing safety evaluations was consistent with 10 CFR 50.59. The sample included design changes, calculations, and procedure changes.

The team reviewed the safety evaluations that PPL had performed and approved during the time period covered by this inspection (i.e., since the last plant modifications inspection) not previously reviewed by NRC inspectors. The screenings and applicability determinations were selected based on the safety significance, risk significance, and complexity of the change to the facility.

In addition, the team compared PPL's administrative procedures used to control the screening, preparation, review, and approval of safety evaluations to the guidance in NEI 96-07 to determine whether those procedures adequately implemented the requirements of 10 CFR 50.59. The reviewed safety evaluations and screenings are listed in the Attachment.

b. Findings

1. Technical Specification Surveillance Testing of Secondary Containment and Standby Gas Treatment System

Introduction: An unresolved item (URI) was identified because additional NRC review and evaluation is needed to determine whether the technical specifications (TS) and associated bases documents were adequate to provide reasonable assurance of operability of the secondary containment boundary and the standby gas treatment system (SGTS).

During a review of a modification to the secondary containment boundary, the team identified a potential deficiency related to the adequacy of testing performed to comply with TS surveillance requirements (SR) 3.6.4.1.4 and 3.6.4.1.5, which were associated with verifying secondary containment and SGTS operability. The team questioned whether the allowed configuration for secondary containment during testing could mask potential leakage between the secondary containment zones (through a common boundary) and could potentially make secondary containment inoperable.

Description: At Susquehanna, the secondary containments are divided into three zones. Zone 1 is associated with Unit 1, Zone 2 is associated with Unit 2, and Zone 3 encompasses the spent fuel area. These zones are normally maintained at $\geq$ 0.25" water vacuum via the normal non-safety related ventilation lineup, and the ventilation flow is discharged to the atmosphere via an unfiltered but monitored release path. During a postulated design basis accident, the secondary containment ventilation system is designed to isolate the affected unit's zone (either Zone 1 or Zone 2) and the spent fuel pool zone (Zone 3). These two zones are automatically placed on recirculation and are discharged through the SGTS via common ventilation ductwork. The unaffected unit's zone does not isolate and remains on the normal ventilation lineup. During a postulated design basis accident combined with a loss-of-offsite power, all three zones would automatically isolate and be placed on recirculation with discharge via the SGTS.

The team reviewed the TS SRs used to verify the operability of secondary containment boundary and the SGTS, as well as the surveillance procedure used to meet TS SRs 3.6.4.1.4 and 3.6.4.1.5. The team found that any of three ventilation line-ups were allowed in the procedure. The acceptable configurations were;

1. All three zones on recirculation connected to the SGTS.
2. Zone 1 (2) and Zone 3 on the recirculation system connected to the SGTS with Zone 2 (1) operable (> 0.25" water column vacuum) and on normal ventilation.
3. Zone 1 (2) and Zone 3 on the recirculation system connected to the SGTS with Zone 2 (1) inoperable (atmospheric pressure).

The TS SR acceptance criteria specified that if the SGTS can maintain the tested zone at a vacuum $\geq$ 0.25" of water in the required time (3.6.4.1.4) and the flow rate through the SGTS is less than the established limit (3.6.4.1.5), then the SRs are considered met. The team found that, as a matter of routine, PPL tested the SGTS as per Items 1 and 2 above to provide reasonable assurance of operability of secondary containment and the SGTS system, and that the test results met the TS surveillance acceptance criteria.

The team determined that when the all three zones configuration is used (Item 1 above), the boundaries between zones are not subject to any testing (i.e., the SGTS maintains all three zones at about the same pressure). This configuration does, however, test the secondary containment exterior boundaries and the SGTS for a design basis event with a loss-of-offsite power.

Similar to the Item 1 configuration, for either of the two zone configurations (i.e., with the unaffected unit's non-safety related system in operation, Item 2 above), the common boundary between the zones is not tested. Specifically, the team's review of the test results for this two zone configuration identified that the unaffected unit's vacuum was better than the unit's vacuum that was being established by the SGTS. As a result, if there was an opening between the zones (at the common boundary), the normal ventilation lineup would be assisting the SGTS, thus invalidating the test.

The team determined that the configuration with one zone inoperable (Item 3 above) represented a valid surveillance test in that it would ensure that potential leakage between the zones would be small enough such that the SGTS could maintain sufficient secondary containment vacuum. However, because this is not the normal configuration, the potential for leakage at the common boundary between the affected and non-affected zones could exist since normal ventilation creates a better vacuum on the non-affected unit. This potential leakage would be discharged without filtration to the environment via the normal ventilation system, and manual operator action would be necessary to stop this discharge.

In response to the team's concerns, PPL conducted field walkdowns to visually confirm the absence of openings at the internal boundaries. In addition, PPL reviewed the results of recently completed surveillance test data, which indicated the normal ventilation system that was in-service on the zone that was not being tested, was not significantly more negative than the zone tested by the SGTS. In addition, PPL confirmed that operators would receive a control room alarm in the event that the normal ventilation system experiences a high radiation condition in its effluent discharge path, and that the operators would respond and manually place the SGTS in service on that unit. The team determined that the results of PPL's review demonstrated reasonable assurance that the SGTS was capable of performing its intended function.

The team will coordinate with the NRC's Office of Nuclear Reactor Regulation to review the adequacy of PPL's SGTS testing methodology to ensure secondary containment boundary integrity. Pending resolution of this issue and determination of any potential enforcement actions, this item is an Unresolved Item. **(URI 05000387/2012007-01, 05000388/2012007-01, Adequacy of Secondary Containment and Standby Gas Treatment System Testing)**

.2 Permanent Plant Modifications (9 samples)

.2.1 Disable Steam Leak Detection Differential Temperature Isolation

a. Inspection Scope

The team reviewed modification EC-1338391 that disabled the steam leak detection differential temperature isolation signal. The differential temperature signal was originally designed to detect steam leaks in secondary containment and isolate the

associated safety systems in order to stop the leak. PPL determined that the differential temperature signals for all the safety systems could inadvertently actuate due to a rapid change in outside air temperature or a single failure of one temperature probe. PPL removed the isolation signal but retained the differential temperature alarm so that operators would be alerted to a steam leak in the room. PPL evaluated the change against the licensing bases of the plant and concluded that the isolation function for a steam leak for each safety system could be provided by the high temperature probes located in each room, and that the requirement to identify steam leaks was unaffected by the modification.

The team reviewed the removal of the differential temperature isolation signal to determine that the design bases, and licensing bases, and performance capability of the isolation function had not been degraded. The team interviewed design engineers and reviewed calculations, and evaluations to determine if the capability of the high temperature isolation function and the differential temperature alarm function met design and licensing requirements. Additionally, the team reviewed post-modification testing results and associated maintenance work orders to confirm that the modification was appropriately implemented. Finally, the team reviewed the results of the modification to determine if the previously identified single failure vulnerability had been corrected. The 10 CFR 50.59 screening determination associated with this modification was also reviewed as described in Section 1R17.1 of this report. Documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2.2 Install Two Secondary Containment Bypass Leakage Boundary Valves

a. Inspection Scope

The team reviewed modification EC-1386053 that installed two new secondary containment bypass leakage (SCBL) boundary valves in order to change the location of the existing SCBL test boundary for a residual heat removal pipe penetration. The modification was performed to address excessive leak rates of the credited SCBL boundary valves for one primary penetration. PPL had identified that the leakage from this penetration path challenged the TS limits of 9 standard cubic feet per hour for SCBL boundary leakage. PPL installed the new boundary valves to address limitations in the isolation capabilities of the existing large diameter isolation valves located at the primary containment penetration. In addition to the two new 3-inch valves, the modification added two existing 4-inch valves to the SCBL boundary to provide a complete boundary for the penetration. The valves were installed to provide a better isolation boundary and thus reduce overall SCBL leakage. Additionally, the modification reduced the testing requirements for the SCBL boundary and reduced dose rates for maintenance personnel during testing.

The team reviewed the modification to confirm that the design and licensing bases and performance capability of the SCBL boundary had not been degraded by either the installation of the new valves or the additional design requirements assigned to the previously installed valves. The team interviewed design engineers, reviewed the modification package, and reviewed vendor data to determine if the capabilities of the

newly credited valves met the design and licensing bases. The team also reviewed the changes to the newly credited SCBL boundary to ensure that all potential primary and secondary containment bypath paths were evaluated and all potential leakage was evaluated against TS limits. Additionally, the team reviewed post-modification testing results and associated maintenance work orders for the new valve installation to verify the valves were installed correctly. Finally, the team walked down the valves with the system engineer to verify the modification was performed as described in the modification package. The 10 CFR 50.59 screening determination associated with this modification was also reviewed as described in Section 1R17.1 of this report. Documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2.3 Install New Time Delay Relays in Emergency Switchgear Fan Control Circuits

a. Inspection Scope

The team reviewed modification EC-643549 that installed time delay relays in the control circuit for the emergency switchgear ventilation fans. The ventilation fans supply cooling to the emergency switchgear during design basis events. There were two fans in each ventilation train with one fan in operation and the other in standby. If the operating fan were to fail, the standby fan would automatically start in response to a low flow or high temperature signal. PPL implemented the control circuit modification to ensure that a sufficient time delay would exist between a failure of the operating fan and the start of the standby fan, so that the accompanying low flow alarm signal in the control room would activate and alert operators of the failure of the primary fan. PPL had identified that without a sufficient time delay, the second fan could start before the low flow alarm activated.

The team reviewed the modification to confirm that the design and licensing bases and performance capability of the ventilation system had not been degraded by installation of the relay or the delay in starting of the standby fan. The team interviewed design engineers and reviewed vendor data, control circuit drawings, and evaluations to determine if the modified control circuit met the design and licensing requirements. Additionally, the team reviewed post-modification testing results, and associated maintenance work orders to verify that the relay modification was appropriately implemented. The 10 CFR 50.59 screening determination associated with this modification was also reviewed as described in section 1R17.1 of this report. Documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2.4 Remove ‘E’ Emergency Diesel Generator Turbocharger Overspeed Trip Function

a. Inspection Scope

The team reviewed modification EC-1193270, which changed the ‘E’ emergency diesel generator (EDG) control logic by permanently removing the turbocharger overspeed trip and alarm function. The team noted that the other (four) EDGs did not have a similar trip function. This trip function was automatically bypassed when the EDG was operated in the emergency mode, and only provided a protective trip in the test mode of operation. All other protective trips were unaffected by this design change.

The team reviewed the modification to determine whether the design and licensing bases and performance capability of the EDG had been degraded by the modification. The team assessed PPL's technical evaluations and design details, including installation specifications, to determine whether the EDG would function in accordance with the modification's assumptions, and with design and licensing requirements. Drawings and procedures were reviewed to verify whether they were properly updated. The team also reviewed the completed work order to assess whether installation activities were performed as specified by the modification's design. Post-modification test results were reviewed to verify that the acceptance criteria had been met. A review of condition reports was performed to determine whether there were any reliability or performance issues associated with the post-modification configuration. Additionally, the 10 CFR 50.59 screening determination associated with this modification was reviewed as described in Section 1R17.1 of this report. Documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2.5 Emergency Service Water Pump Auto-Start Circuit Modification for Fire Protection

a. Inspection Scope

The team reviewed EC-1305801 that modified the automatic start circuits of all four emergency service water (ESW) pumps to prevent a loss of control power to the pump's auto-start circuit, due to a postulated main control room fire. PPL determined this modification would mitigate a multiple spurious operation (MSO) vulnerability, which had been identified during PPL's recent plant specific post-fire safe shutdown re-analysis. PPL determined that multiple fire-induced shorts to ground on control cables routed within the main control room could have resulted in a loss of control power to ESW pump circuits and a subsequent loss of the ESW pump auto-start function. Specifically, this modification split the automatic and manual start circuits and placed the auto-start circuit on separate control power fuses, isolated from other portions of the circuit located within the control room.

The team reviewed the modification to determine whether the design and licensing bases and performance capability of the ESW system had been degraded by the modification. The team assessed PPL's technical evaluations and design details, including installation specifications, and interviewed engineering personnel to determine whether the ESW system would function in accordance with the modification's assumptions and the design and licensing bases. Drawings and procedures were reviewed to verify that they were properly updated to reflect the post-modification design and operation. The team also reviewed completed work orders to assess whether installation activities were performed as specified by the modification's design. The post-modification test results were reviewed to verify that the acceptance criteria had been met. A review of condition reports was performed to determine whether there were any reliability or performance issues associated with the post-modification configuration. Additionally, the 10 CFR 50.59 screening determination associated with this modification was reviewed as described in Section 1R17.1 of this report. Documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2.6 Main Steam Isolation Valve Key Lock Switch Modification for Fire Protection

a. Inspection Scope

The team reviewed EC-1455549 that modified the logic control circuits of the main steam isolation valves (MSIVs) to prevent both MSIVs in the same steam line from failing to close or re-open due to a postulated main control room fire. PPL determined this modification would mitigate a MSO vulnerability, which had been identified during PPL's recent plant specific post-fire safe shutdown re-analysis. PPL determined that multiple fire-induced hot shorts in control cables routed within the main control room could have resulted in multiple MSIVs failing to close or spuriously re-opening by maintaining the opening logic circuits energized or re-energizing the circuits. Specifically, this modification added a key lock switch, located outside the control room, in the control power circuit for each MSIV division (i.e., two switches; one for inboard MSIVs, and one for outboard MSIVs). Operation of either switch would de-energize the control logic for a single division, and therefore prevent both MSIVs in the same steam line from re-opening.

The team reviewed the modification to determine whether the design and licensing bases and performance capability of the main steam system had been degraded by the modification. The team assessed PPL's technical evaluations and design details, including installation specifications, and interviewed licensed operators and engineering personnel to determine whether the MSIVs would function in accordance with the modification's assumptions, and with design and licensing requirements. Drawings and procedures were reviewed to verify whether they were properly updated to reflect the post-modification design and operation. The team also reviewed completed work orders to assess whether installation activities were performed as specified by the modification's design. The post-modification results were reviewed to verify that the acceptance criteria had been met. In addition, the team walked down the key lock switches to independently evaluate material conditions and configuration control with the approved design, including an independent assessment of emergency lighting for

operator access and egress and switch operations. A review of condition reports was performed to determine whether there were any reliability or performance issues associated with the post-modification configuration. Additionally, the 10 CFR 50.59 screening determination associated with this modification was reviewed as described in Section 1R17.1 of this report. Documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2.7 Unit 1 Reactor Feed Pump Turbine Upgrade

a. Inspection Scope

The team reviewed modification EC-770912 that evaluated and replaced the reactor feed pump (RFP) turbine to support extended power uprate (EPU) activities. New RFP turbines capable of operating at higher speeds were required to permit the RFP to achieve the increased feedwater system flows required at EPU conditions.

The team reviewed the modification to verify that the design and licensing bases of systems had not been degraded by the RFP turbine replacement. The team confirmed that the components met the appropriate quality standards. The team also reviewed post-modification testing of the equipment to verify proper operation and interaction with the existing system controls. The team also conducted a walkdown of the affected components. The 10 CFR 50.59 screening determination associated with this modification was reviewed as described in Section 1R17.1 of this report. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2.8 Extend the Disc Stop on Emergency Service Water System Check Valves

a. Inspection Scope

The team reviewed modification EC-1297470 that extended the disc stop on the 'E' diesel generator emergency service water (ESW) system check valves (011513 and 011514). This modification was implemented to reflect severe service conditions and to eliminate adverse consequences of small buildup of microbiologically induced corrosion on the internal surface of the installed valves. Specifically, the modification reduced the valve travel by a small amount to eliminate the potential for contact between the disc and valve body while maintaining sufficient ESW cooling water flow for the diesel generator.

The team reviewed the modification to verify that the design and licensing bases of plant systems had not been degraded by extending the check valve stops by about ¾ inch. The team reviewed the associated documents and associated evaluation to verify that adequate ESW cooling water flow could be maintained. The team also reviewed the results of the post installation inspection and testing to determine the adequacy of the

modification. The team also conducted a walkdown of the 'E' diesel generator to verify the overall material condition of the valves. Finally, the 10 CFR 50.59 applicability determination associated with this modification was reviewed as described in Section 1R17.1 of this report. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2.9 Residual Heat Removal Service Water Isolation Valve Replacement

a. Inspection Scope

The team reviewed modification EC-363142 that replaced a residual heat removal (RHR) service water isolation valve. The replacement butterfly valve was similar to the existing butterfly valve, which became obsolete. The replacement valve design was bolted in place differently than the original valve and it utilized a different seal material.

The team reviewed the modification to verify that the design and licensing bases and performance capability of the RHR service water system had not been degraded by the modification. The team interviewed engineering staff and reviewed technical evaluations associated with the modification to determine whether the valve would perform consistent with the valve that was replaced. The team reviewed drawings and the post-modification testing to ensure that the valve was properly installed and tested. The team also performed a walkdown of the installed valves to verify the installation was completed as designed and to assess the material condition of the equipment. Additionally, the 10 CFR 50.59 screening determination associated with this modification was reviewed as described in Section 1R17.1 of this report. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems (IP 71152)

a. Inspection Scope

The team reviewed a sample of condition reports (CR) associated with 10 CFR 50.59 and plant modification issues to determine whether PPL was appropriately identifying, characterizing, and correcting problems associated with these areas, and whether the planned or completed corrective actions were appropriate. In addition, the team reviewed CRs written on issues identified during the inspection to verify adequate problem identification and incorporation of the issues into the corrective action system. The CRs reviewed are listed in the Attachment.

b. Findings

No findings were identified.

4OA6 Meetings, including Exit

The team presented the preliminary inspection results to Mr. J. Helsel, Plant General Manager, and other members of PPL staff at a meeting on December 14, 2012. Following additional in-office review, which included support from the Office of Nuclear Reactor Regulation staff, the team conducted a final exit meeting on January 25, 2013, via teleconference with Mr. R. Franssen, Nuclear General Manager – Engineering, and other members of PPL staff to discuss the final results of the inspection. The team returned proprietary information reviewed during the inspection and verified that this report does not contain proprietary information.

ATTACHMENT

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

T. Case, Licensing Engineer
M. Cummings, System Engineer
S. Geiger, System Engineer
T. Gorman, Senior Design Engineer
R. Pistolas, Design Engineer
W. Morris, Licensed Senior Reactor Operator
T. Wales, Design Engineer
R. Weitzel, Design Engineer

NRC Personnel

P. Finney, Senior Resident Inspector - Susquehanna
J. Grieves, Resident Inspector - Susquehanna

ITEMS OPENED, CLOSED AND DISCUSSED

Opened and Closed

05000387,388/2012007-001	URI	Adequacy of Secondary Containment and Standby Gas Treatment System Testing (Section 1R17.1.1)

LIST OF DOCUMENTS REVIEWED

10 CFR 50.59 Evaluations

50.59-SE-00013, ECs 940986, 940983, 864462, and 910695, Integrated Control System and Reactor Feed Pump Turbine Speed Control System, Revision 0
50.59-SE-00016, LDCN 4906 – Unit 1 Cycle 17 COLR Revision 1 – Initial Core Loading, Revision 0
50.59-SE-00017, HUXY Code Change for Unit 1 Cycle 18 Licensing Analysis, Revision 0

10 CFR 50.59 Screened-out Evaluations

AD 00511, RCIC Turbine Overspeed Trip Testing with Auxiliary Steam (TP-150-004, TP-250-004), 9/30/09
AD 00820, Loss of Service Water Procedure Revision, 4/17/12
SD 00750, Temporary Sealing of Doors A24 or A25 Using Nashua Type 398N Nuclear Duct Tape, Revision 1
SD 00854, EC 181241, ‘D’ ESW Pump Motor Replacement, Revision 1
SD 00895, SE-024-D02, ‘D’ EDG Overspeed Trip Test, Revision 1

SD 00918, EC 1185628, Change 1D135 Battery type from Valve Regulated Lead Acid to Vented Lead Acid, Revision 0
SD 00981, EO-100-030, Station Blackout Procedure Revision, Revision 0
SD 00991, SO-193-001, Quarterly Turbine Valve Cycling, Revision 1
SD 01000, Deletion of Steam Leak Detection Differential Temperature Isolation, Revision 0
SD 01006, Remove Triaxial Peak Accelerographs from FSAR, Revision 0
SD 01050, EC 1305834, Appendix R Modification MSO #N7, 4kV Breaker Trip Function Control Power, Revision 4
SD 01081, Fuel Rod Corrosion Thickness Measurement System Dual Frequency and Profilometry, Revision 0
SD 01138, NDAP-QA-1130, Issuance of New Cable Aging Management Program, Revision 0
SD 01154, EC 1330732, Appendix R Modification MSO #2G, Sump Room / Core Spray Room Watertight Door Addition, Revision 1
SD 01157, EC 1464882, Cyber Security Infrastructure Upgrades, Revision 2
SD 01159, Opening Valve HV1(2)51F027A/B During Wetwell Spray Operation, [OP-1(2)49-004 and OP-1(2)16-001], Revision 0
SD 01200, Unit 1 Cycle 18 COLR Revision 0 - Initial Core Loading, Revision 0
SD 01206, CO 07-001-1505315-0, Removal of Appendix R Light 2ELU2052, Revision 0
SD 01222, HPCI System 1(2)C601 [AR-1(2)14-001, HPCI System 1(2)C601], Revision 0
SD 01225, RHR Shutdown Cooling [OP-149-002], Revision 2
SD 01247, Use of Head Spray Line as Alternate RPV Injection Path, Revision0
SD 01275, Engineered Safeguard Service Water Pumphouse HVAC, Revision 0
SD 01281, Revise Diesel Fuel Operating Procedure, Revision 0

<u>Modification Packages</u>
EC-1193270, Permanently Remove the ‘E’ EDG Turbo Overspeed Trip, Revision 0
EC-1297470, Extend the Disc Stop on 011513 and 011514, Revision 0
EC-1305801, Appendix R Modification MSO #5i, ESW Pump Auto Start Function, Revision 1
EC-1338391, Disable Steam Leak Detection Setpoint to Remove Single Point Vulnerability, 1/7/11
EC-1386053, Install Two New SCBL Test boundaries on Unit 1 RHR System, 12/13/11
EC-1455549, Appendix R Modification MSO #2b, MSIV Key Lock Switches, Revision 0
EC-363142, Butterfly Valve, Size 20” Complete with Operator for Replacement Valve, RHR Heat Exchanger to RHR Service Water Isolation Valve, Revision 2
EC-643549, Correct Coordination Deficiencies Between the Emergency Switchgear Room Low Flow Switches and Associated Time Delay Relays, 3/19/09
EC-770912, Unit 1 Reactor Feed Pump Turbine Upgrade, Revision 0

<u>Calculations, Analysis, and Evaluations</u>
EC-023-1012, Evaluate Impact on Use of Ultra Low Sulfur Diesel Fuel on the Diesel Generator Fuel Oil Storage and Transfer System, Revision 1
EC-048-1022, Performance Test Results for the Unit 1 Dresser-Rand RFP Turbines, Revision 0
EC-049-0001, Pressure Drops in the RHR System for Various Modes of Operation, Revision 8
EC-RADN-1141, Radiological Consequences of an Instrument Line Break, Revision 2
EC-TRAN-0513, Transient Safety Analysis Design Report GEZ-7127, Revision 0
FPSD 00081, EC 1365628, Appendix R MSO 8-Hour Emergency Lights - Unit 1, Revision 1
FPSD 00085, EC 1455549, Appendix R MSO MSIV Key Lock Switches, Revision 0
FPSD 00087, EC 1365628, Appendix R MSO 8-Hour Emergency Lights - Unit 2, Revision 1
ICS-044, Integrated Control System Failure Modes and Effects Analysis, Revision 0
M-1560, Specification - Refurbishment and Replacement of the Reactor Feed Pump, Revision 3

Condition Reports

0315490	1388942	1647042*	1650607*
1114053	1408577	1647130*	1650657*
1151749	1429288	1647175*	1650998*
1189638	1480485	1647181*	1651030*
1237762	1510713	1647414*	1651408*
1249250	1510856	1649337*	1651707*
1266341	1521397	1649978*	1652040*
1285359	1631367	1650026*	1652983*
1294964	1646699*	1650026*	1652994*
1296646	1646795*	1650033*	1653272*
1382300	1646962*	1650591*	

(* denotes NRC identified during this inspection)

Drawings

B-106516, Sht. 2 and 12, Logic Diagram Reactor Building Zone 1 HVAC Switchgear and LC Room Cooling Fan IV-222B, Revision 11 and 1
E-106228, Sht. 12, Process Sampling, Revision 16
E-106256, Sht. 1 and 2, Residual Heat Removal, Revision 64 and 53
E-106258, Sht. 1, Fuel Pool Cooling and Cleanup, Revision 41
E-106280, Sht. 1, Reactor Building Air Flow Diagram Zone III, Revision 33
E-106450, Sht. 2, Heating and Ventilating Reactor Building Unit-1, Area 25, Revision 21
E-146, Sht. 2, 'A' ESW Pump Schematic, IDCN 35
E-146, Sht. 4, 'B' ESW Pump Schematic, IDCN 29
E-146, Sht. 8, 'B' ESW Pump Schematic, IDCN 31
E-146, Sht. 6, 'C' ESW Pump Schematic, IDCN 37
E-16, Sht. 5, Circuit Breaker DC Control Power Diagram, IDCN 18 & 19
E-16, Sht. 9, Circuit Breaker DC Control Power Diagram, IDCN 19 & 20
E-205954, Sht. 2, Unit 2 Fire Doors and Fire Dampers, Revision 5
E-205956, Sht. 2, Unit 1 Fire Doors and Fire Dampers, Revision 5
FF61604, Sht. 32, 'E' EDG Control Schematic, IDCN 1
FF61604, Sht. 40, 'E' EDG Control Schematic, IDCN 2
FF61604, Sht. 43, 'E' EDG Control Schematic, IDCN 7
FF65136, Sht. 11, ANSI Class 300 Check Valve, 10 Inch 380-15-WE(40)-X, Revision 2
M-111, Sht. 4, Emergency Service Water System, Revision 4
M-112, Sht. 1, RHR Service Water, Revision 50
M-124, Sht. 7, 'E' EDG Auxiliaries, IDCN 8
M-127, Sht. 2, Unit 1 Feed Pump Turbine Steam, Revision 10
M-127, Sht. 3, Unit 1 Feed Pump Turbine Steam, Revision 7
M-151, Sht. 1, Unit 1 Residual Heat Removal, Revision 68
M-151, Sht. 2, Unit 1 Residual Heat Removal, Revision 53
M1-B21-131, Sht. 1, MSIV Elementary Diagram, IDCN 8
M1-B21-131, Sht. 2, MSIV Elementary Diagram, IDCN 7
M1-B21-131, Sht. 7, MSIV Elementary Diagram, IDCN 9
M1-B21-131, Sht. 8, MSIV Elementary Diagram, IDCN 7
M1-B21-131, Sht. 12, MSIV Elementary Diagram, Revision 15
M1-B21-131, Sht. 13, MSIV Elementary Diagram, Revision 13
W0125791, Sht. 2, 20 Inch, 150 Stainless Steel Lugged Butterfly Valve, Revision 0

Licensing Documents
UFSAR 5.4.1, Reactor Recirculation Pumps, Revision 65
UFSAR 7.7.1.3, Recirculation Flow Control System, Revision 65
UFSAR 15.4.5, Recirculation Flow Control Failure with increasing Flow, Revision 65
TRM Basis 3.3.10, Reactor Recirculation Pump MG Set Stops, Revision 3
NRC Safety Evaluation Report, Safe Shutdown Analysis and Associated Circuits (accession No. 9711040196), dated 10/21/97

Procedures
MFP-QA-1220, Engineering Change Process, Revision 14
NDAP-QA-0412, Leakage Rate Test Program, Revision 15
NDAP-QA-0423, Station Pump and Valve Testing Program, Revision 24
NDAP-QA-0726, 10 CFR 50.59 and 10 CFR 72.48 Implementation, Revision 13
NDAP-QA-1218, Temporary Changes, Revision 11
NDAP-QA-1220, Engineering Change Process, Revision 8
ON-100-009, Control Room Evacuation, Revision 24, 25, 26, and 27
OP-023-001, Diesel Fuel Oil System, Revision 33
OP-116-001, RHR Service Water, Revision 32
OP-149-002, RHR Shutdown Cooling, Revision 48
OP-149-004, RHR Containment Cooling, Revision 23
OP-216-001, RHR Service Water, Revision 27
OP-228-001, Engineered Safeguard Service Water (ESSW) Pumphouse HVAC, Revision 9
OP-249-004, RHR Containment Cooling, Revision 24
OP-ORF-007, Underwater Fuel Inspection and Repair, Revision 10
SC-023-002, New Diesel Fuel Oil Receipt Analysis, Revision 11
SE-070-011, 24 Month Secondary Containment Drawdown and Inleakage Surveillance Test Zone I, II, and III, Revision 12
SI-079-326, 24 Month Calibration of Refuel Floor Wall Exhaust Duct High Radiation Monitor Channel R-D12-2K609A, Revision 16
SI-079-330, 24 Month Calibration of Refuel Floor Wall Exhaust Duct High Radiation Monitor Channel R-D12-2K609B, Revision 14
SO-000-010, Monthly Zone III Integrity Verification, Revision 32
SO-100-010, Monthly Zone I Integrity Verification, Revision 31
SO-200-010, Monthly Zone II Integrity Verification, Revision 30
SUS-ISTPLN-100.0, SSES Unit 1 In-service Testing Program Plan, Revision 5

Work Orders

0466070	1296329	1505048
0737870	1338541	1505824
0737925	1466113	1631489
1240366	1466121	
1260816	1483138	

Miscellaneous
ANSI N510, Testing of Nuclear Air-Cleaning Systems, 1975
ANSI/ASME N509, Nuclear Power Plant Air Cleaning Units and Components, 1976
EC 1320608, Remove Triaxial Peak Accelerographs, dated 2/14/11
EC-059-1024, Design Requirements for and Evaluation of Potential Secondary Containment Bypass Leakage Pathways, Revision 9
EMF-2361(P)(A), EXEM BWR-200 ECCS Evaluation Model, Revision 0

FS-184, Fuel Rod Corrosion Thickness Measurement System Dual Frequency and Profilometry, Revision 5
J1116, Procurement of a Seismic Monitoring System Replacement, dated 12/3/08
LDCN 4892, Take Exception for Triaxial Peak Accelerographs in Revision 1 of Regulatory Guide 1.12, dated 10/22/10
LDCR 4906, Unit 1 Cycle 17 COLR Revision 1 - Initial Core Loading, Revision 1
Letter, USNRC to PPL, Amendment No. 21 to Facility Operating License NPF-14, SSES, Unit 1, dated 3/23/84
NIMS Component Records 1A20403FU23, 1A20403FU24, 1C201BFU217, and 1C201BFU218, dated 12/11/12
PLA-4488, Proposed Amendment No. 203 to License NPF-14 and No. 161 to License NPF-22, Conversion of the SSES Technical Specifications to the ISTS NUREG 1433, dated 8/1/96
RIR 129476, Receipt Inspection Report for P/N 060922, dated 12/19/06
Safety Evaluation by NRR Related to Amendment No. 178 to Facility Operating License NPF-14 and Amendment No. 151 to Facility Operating License NPF-22
System Health Report, Diesel Generators, 2nd Quarter 2012
System Health Report, RHR Service Water, Unit 1, 2nd Quarter 2012
System Health Report, RHR Service Water, Unit 2, 2nd Quarter 2012
US4616, GeoSIG Seismic Monitoring System for NPP Susquehanna, Revision 5
XN-CC-33, Report Regarding the Exelon Nuclear Company ECCS Non-Jet-Pump-BWR Fuel Heatup Model by the Office of Nuclear Reactor Regulation, dated 3/6/75

Surveillance and Modification Acceptance Tests
SE-159-200, MSIV Logic System Functional Test, performed on 5/22/12
SE-170-11, Secondary Containment Drawdown and In-leakage Surveillance Test, Zones I and III, performed on 8/19/11
SE-270-011, 24-Month Secondary Containment Drawdown and Inleakage Surveillance Test Zones II and III, performed on 1/26/12
SO-024-014, Monthly Diesel Generator ‘E’ Operability Test, performed on 9/3/10
TP-054-107, ‘C’ ESW Pump Test/EC 1305801, 4kV Breaker Modification, performed on 8/28/12
TP-134-053, Unit 1 Post Maintenance Test for ECO 643549, performed on 5/14/10

LIST OF ACRONYMS

ADAMS	Agencywide Documents Access and Management System
CFR	Code of Federal Regulations
COLR	Core Operating Limits Report
CR	condition report
DRS	Division of Reactor Safety
EDG	emergency diesel generator
EPU	extended power uprate
ESW	emergency service water
IMC	Inspection Manual Chapter
MSIV	main steam isolation valve
MSO	multiple spurious operation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
PARS	Publicly Available Records
PPL	PPL Susquehanna, LLC
RFP	reactor feed pump
RHR	residual heat removal
SCBL	secondary containment bypass leakage
SGTS	standby gas treatment system
SR	surveillance requirement
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report