November 26, 2001

Mr. Robert G. Byram
Senior Vice President and
Chief Nuclear Officer
PPL Susquehanna, LLC
Susquehanna Steam Electric Station
2 North Ninth Street
Allentown, Pennsylvania 18101

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION - NRC INSPECTION REPORT 50-387/01-10, 50-388/01-10

Dear Mr. Byram:

On November 17, 2001, the NRC completed an inspection at your Susquehanna Steam Electric Station Units 1 and 2. The enclosed report documents the inspection findings which were discussed on November 20, 2001, with Mr. R. Anderson, General Manager, and other members of your staff.

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

No findings of significance were identified.

Since September 11, 2001, the Susquehanna Steam Electric Station has assumed a heightened level of security based on a series of threat advisories issued by the NRC. Although the NRC is not aware of any specific threat against nuclear facilities, the heightened level of security was recommended for all nuclear power plants and is being maintained due to the uncertainty about the possibility of additional terrorist attacks. The steps recommended by the NRC include increased patrols, augmented security forces and capabilities, additional security posts, heightened coordination with local law enforcement and military authorities, and limited access of personnel and vehicles to the site.

The NRC continues to interact with the Intelligence Community and to communicate information to PPL Susquehanna, LLC. In addition, the NRC has monitored maintenance and other activities which could relate to the site's security posture.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publically Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm.html (The Public Electronic Reading Room).

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

Sincerely,

/RA/

Mohamed Shanbaky, Chief
Projects Branch 4
Division of Reactor Projects

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

Enclosure: Inspection Report 50-387/01-10, 50-388/01-10

Attachment 1 - Supplemental Information

cc w/encl:
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R. Anderson, General Manager - SSES Operations
R. L. Ceravolo, General Manager - Plant Support
G. A. Williams, General Manager - Nuclear Assurance
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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket Nos.:	05000387, 05000388
License Nos.:	NPF-14, NPF-22
Report No.:	50-387/01-10, 50-388/01-10
Licensee:	PPL Susquehanna, LLC
Facility:	Susquehanna Steam Electric Station
Location:	Post Office Box 35 Berwick, PA 18603
Dates:	October 1, 2001 to November 17, 2001
Inspectors:	S. Hansell, Senior Resident Inspector J. Richmond, Resident Inspector
Approved by:	Mohamed M. Shanbaky, Chief Projects Branch 4 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000387-01-10, 05000388-01-10; on 10/01-11/17/2001; PPL Susquehanna, LLC; Susquehanna Steam Electric Station; Units 1&2. Resident Inspector Report.

The report covered a 7 week period of resident inspection. No findings of significance were identified.

A. <u>Inspector Identified Findings</u>

No findings of significance were identified.

B. <u>Licensee Identified Violations</u>

Violations of very low safety significance which were identified by PPL have been reviewed by the inspectors. Corrective actions taken or planned by PPL appear reasonable. These violations are listed in section 4OA7 of this report.

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Report Details

Summary of Plant Status

Susquehanna Steam Electric Station (SSES) Unit 1 began the period at full power. On October 25, 2001, reactor power was reduced to approximately 74% when normal cooling to the primary containment was unexpectedly lost. The Unit was returned to full power on October 26, and operated at or near full power for the remainder of the report period; with exceptions for control rod pattern adjustments, control rod drive maintenance, and main turbine control valve testing.

Unit 2 was operated at or near full power for the report period; with exceptions for control rod pattern adjustments, main steam isolation valve testing, and main turbine control valve testing.

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

1R04 Equipment Alignments (71111.04)

.1 Partial System Walkdowns

a. Inspection Scope

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

- Unit 1 and Unit 2 High Pressure Coolant Injection (HPCI) systems, following identification of a steam leak on the Unit 2 HPCI steam admission valve HV-255-F001 (CR 361132)
- Plant protected area boundary and installed security barrier changes, in response to the heightened security condition

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05Q)

a. Inspection Scope

The inspectors reviewed the Fire Protection Review Report to determine the required fire protection design features, fire area boundaries, and combustible loading requirements for the areas examined during this inspection. The inspectors walked down these areas to assess PPL's control of transient combustible material and ignition sources, fire detection and suppression capabilities, fire barriers, and any related compensatory measures. These areas included:

- Unit 2 High Pressure Coolant Injection system, turbine room warmer than normal due to a steam leak on the steam admission valve (CR 361132)
- Unit 1 and Unit 2 125/250 Volt DC battery and battery charger rooms

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Re-qualification Training (71111.11Q)

a. Inspection Scope

On November 6, 2001, the inspectors observed licensed operator performance in the simulator during Training Scenario 01-05-02, revision 0, "Security Event, Station Blackout with Unit 1 in Mode 3, then subsequent Station Power Restoration from T-10," to assess licensed operator performance. The inspectors assessed the operators' adherence to Technical Specifications, emergency plan implementation, and the use of off-normal and emergency operating procedures. The inspectors compared their observations to ON-000-010, revision 2, "External Security Threat," and ON-104-001, revision 13, "Unit 1 Response to Loss of All Off-Site Power." The inspectors observed the PPL training critique of the operators' simulator performance to identify discrepancies and deficiencies in training. In addition, the inspectors reviewed the ability of the simulator to model the actual plant performance.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12Q)

a. Inspection Scope

The inspectors evaluated the follow-up actions for selected system, structure, or component (SSC) issues and reviewed the performance of these SSCs, to assess the effectiveness of PPL's maintenance activities. 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(a)(1) and (a)(2), "Requirements for Monitoring the Effectiveness of Maintenance." In addition, the inspectors reviewed selected SSC classification, performance criteria and goals, and PPL's corrective actions that were taken or planned, to verify that the actions were reasonable and appropriate. The following issues and documents were reviewed:

Equipment Issues

- Unit 1 "A" Turbine Building Closed Cooling Water (TBCCW) pump inadequate discharge pressure (CR 360250)
- Unit 1 Reactor Building Chilled Water (RBCW) system failure resulted in a loss of normal drywell cooling, followed by an apparent failure of the standby pump start feature for Reactor Building Closed Cooling Water system (RBCCW), during automatic re-alignment of RBCCW to the drywell coolers (CRs 363442, 363443, and 363410)

Procedures and Documents

- Maintenance Rule Basis Documents for RBCW, RBCCW, and TBCCW
- System Health Reports for RBCW, RBCCW, and TBCCW
- NDAP-QA-0413, "SSES Maintenance Rule Program"
- EC-RISK-0528, "Risk Significant SSCs for the Maintenance Rule"
- EC-RISK-1054, "SSC Availability Performance Criteria for the Maintenance Rule"
- EC-RISK-1060, "Acceptable Number of Failures for Risk Significant SSCs"

b. Findings

No significant observations or findings were identified.

1R13 Maintenance Risk Assessment and Emergent Work (71111.13)

a. Inspection Scope

The inspectors reviewed the assessment and management of selected maintenance activities to assess 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 verify 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. The inspectors assessed those activities to evaluate whether appropriate implementation of risk management actions were performed in accordance with the following PPL procedures:

- NDAP-QA-1902, "Maintenance Rule Risk Assessment and Management Program"
- NDAP-QA-0340, "Protected Equipment Program"
- PSP-22, "Susquehanna Sentinel Program"
- SSES Team Manual

In addition, the inspectors compared the assessed risk configuration to the actual plant conditions and any in-progress evolutions or external events to evaluate whether the assessment was accurate, complete, and appropriate for the issue. The inspectors performed control room and field walkdowns to verify whether the compensatory measures identified by the risk assessments were appropriately performed. The selected maintenance activities included:

- Unit 1 "A" TBCCW pump inadequate discharge pressure (CRs 360250, 361210, 361164, 361828, 363125, and 364048, and WO 360261)
- Unit 1 emergent degraded conditions on TBCCW, RBCW, and RBCCW, on October 26, 2001 (CRs 363442 and 363443)

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-routine Plant Evolutions & Events (71111.14)

.1 Unit 1 Loss of Drywell Cooling

a. Inspection Scope

On October 25, 2001, the Unit 1 "A" reactor building chiller tripped, followed by a failure of the "B" chiller to automatically start and load. Normal primary containment (drywell) cooling was temporarily lost and the air temperature inside the drywell increased to approximately 140 °F, which exceeded the Technical Specification (TS) limit of 135 °F. Drywell air temperature exceeded the TS limit for about 5 hours. Reactor building closed cooling water system automatically realigned to supply the drywell coolers, and operators manually reduced reactor power to 74%. The operators manually started and loaded the "B" chiller; the drywell temperature returned to below the TS limit.

The inspectors reviewed operating logs, plant computer data, and interviewed plant operators for this unplanned event to independently determine what occurred and evaluate the initiating cause. The inspectors assessed personnel performance during this event to evaluate whether the operator response was appropriate and in accordance with procedures and training. In specific, the inspectors reviewed ON-134-001, "Loss of Drywell Chillers," and condition reports 363410, 363442, and 363443.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

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

- Unit 2 reactor feedwater leading edge flow meter inoperable (CR 360111)
- Unit 2 HV-214-F022B, "B" Inboard Main Steam Isolation Valve, fast stroke time troubleshooting plan and re-test (CR 351409)
- Unit 2 reactor vessel Low Pressure Permissive signal relay with high contact resistance, for "B" Residual Heat Removal (RHR) system Low Pressure Coolant Injection (LPCI) mode (CR 363575)
- Unit 2 High Pressure Coolant Injection system, turbine room warmer than normal due to steam leak on the steam admission valve (CR 361132) and the turbine exhaust rupture diaphragm rupture disc improperly insulated (CR 361369)

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors observed portions of post-maintenance testing activities in the field, and reviewed the test data at the job site. The inspectors observed whether the tests were performed in accordance with the approved procedures, and assessed the adequacy of the test methodology, based on the scope of maintenance work performed. In addition, the inspectors assessed 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 evaluate whether the acceptance criteria were satisfied. The maintenance activities reviewed included:

- Unit 1 "A" Instrument Air Compressor re-test following DCP 312524 (WO 326662)
- Unit 2 "B" RHR LPCI mode Low Pressure Permissive functional test, following relay rework, using partial surveillance test SI-280-301, "Reactor Vessel Pressure Channel Calibration Check for PIS-B21-2N021A-D and PS-B21-2N021E&G, Core Spray and LPCI Permissive" (WO 363674)
- Unit Common "B" Control Structure Chiller and Chill Water System, after heat exchanger tube cleaning (OP-030-001)

- Unit Common Emergency Service Water flow balance, after the "A" Emergency Diesel Generator flow control valve replacement (TP-054-076)
- Unit Common "E" Emergency Diesel Generator 13.8 KV Electrical Breaker, 0A550, "E-EDG Test Feeder Breaker to Bus 10," after inspection and maintenance (WO 359610)

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

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

- Unit 2 "A" Core Spray Loop quarterly flow verification, following maintenance (SO-251-A02 on Oct. 11, 2001)
- Unit 2 Main Steam Isolation Valve quarterly exercising (SO-284-003 on Oct. 12, 2001)
- Unit Common "E" Emergency Diesel Generator 2 year battery surveillance test and battery charger test (SM-102-E03 on Oct. 16-17, 2001)
- Unit 2 "A" Residual Heat Removal Service Water quarterly flow verification, (SO-216-003 on Nov 8, 2001)

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Performance Indicator Verification

a. Inspection Scope

The inspectors reviewed PPL's performance indicator (PI) data to verify the PI accuracy and completeness. The inspection examined PI data, PI data summary reports, and plant records from the third quarter of 2000 to the third quarter of 2001. The PI data was evaluated with the guidance contained in Nuclear Energy Institute (NEI) document 99-02, revision 1, "Regulatory Assessment Performance Indicator Guideline." The following indicators and PPL documents were included in this review:

NRC Performance Indicators
- Emergency Response Organization Drill/Exercise Performance (DEP)
- Alert and Notification System Reliability (ANS)
- Emergency Response Organization (ERO) Readiness

PPL Documents
- Siren Testing 2001 Trending Report
- Bloomsburg, Hazleton, and Wilkes-Barre Siren Test Data for July/August 2001
- EP-AD-022, "Nuclear Emergency Planning Performance Indicators"
- Emergency Director ERO Participation for the 2000/2001 time frame
- PPL Emergency Drill and Critique Observations during the 2000/2001 time frame

b. Findings

No findings of significance were identified.

.2 (Closed) URI 50-387,388/01-003-02 Safety System Functional Failures

a. Inspection Scope

The inspectors reviewed the NRR response to Unresolved Item (URI) 50-387,388/01-003-02, "Safety System Functional Failure PI Verification." The URI described an apparent conflict in guidance in determining whether a failure of an instrumentation system should be counted as a safety system functional failure (SSFF).

b. Findings

As described in NEI 99-02, revision 1, the SSFF PI comprises events or conditions reported under 10 CFR 50.73(a)(2)(v). NUREG-1022, revision 2, "Event Reporting Guidelines," contains an example to illustrate that the failure of a system used only to warn operators, where no credit is taken for the system in any safety analysis and it does not directly control any safety function, is not reportable per 50.73(a)(2)(v). The NRR staff determined that reports of such failures have been included in the Safety System Failure PI in the past. However, such instrument system failures (as described in the URI) are not required to be reported, and are likewise, not required to be included in the SSFF PI. URI 50-387,388/01-003-02 is closed.

4OA3 Event Follow-up (71153)

.1 (Closed) LER 50-388/01-005-00 Multiple Test Failures of Main Steam Safety Relief Valves

On April 12, PPL determined that the as-found lift setpoint for six main steam safety relief valves (SRVs) on Unit 2 failed to open within the required Technical Specification (TS) actuation pressure setpoint tolerance. TS 3.4.3, "Safety Relief Valves," provided an allowable pressure band of +/- 1% for an individual SRV. Four of the SRVs opened below the required band (actual range was -1.25 to -2.18%) and two opened above the required band (actual was +1.16 and +1.53%). PPL determined that the cause of the actuation failures was due, in part, to exposure to in-service operation and seat leakage during operation. All eight SRVs were replaced with valves that were tested and verified to open within the allowable TS band.

This finding effected the Mitigation Systems Cornerstone and was considered to have very low safety significance (Green) using the Significance Determination Process (SDP), because the SRVs would have functioned to prevent reactor vessel over pressurization during the most limiting postulated reactor pressure events. This finding was more than minor because, if left uncorrected, the SRV lift setpoints would further degrade. This PPL identified violation is discussed in Section 4AO7. This issue was documented in PPL's corrective action program as condition report 327430. This LER is closed.

.2 (Closed) LER 50-388/01-003-00 Secondary Containment Bypass Leakage Exceeded

On March 19, 2001, PPL determined that the Unit 2 as-found Secondary Containment Bypass Leakage (SCBL) was 11.13 standard cubic feet per hour (scfh), which exceeded the Technical Specification (TS) leakage limit of 9 scfh. PPL determined the cause of the excessive leakage was due to abnormally high valve stem friction on the "B" Drywell Spray Valve (HV-251-F016B), which reduced the valve's seating force. The valve was re-worked, and satisfactorily re-tested.

This finding affected the Barrier Cornerstone and was considered to have very low safety significance (Green) using the Significance Determination Process (SDP), because the finding did not represent an actual open pathway in the physical integrity of primary containment, and did not affect the control room barrier function. This finding was more than minor because, if left uncorrected, the bypass leakage could increase and potentially result in increased off-site radiation dose consequences. This PPL identified violation is discussed in Section 4AO7. This issue was documented in PPL's corrective action program as condition report 320683. This LER is closed.

4OA6 Meetings

.1 Exit Meeting Summary

On November 20, 2001, the resident inspectors presented the inspection results to Mr. R. Anderson, General Manager, and other members of your staff, who acknowledged the findings.

The inspectors asked PPL whether any items discussed during the exit meeting should be considered proprietary. No proprietary information was identified.

4OA7 Licensee Identified Violations

The following findings of very low significance were identified by PPL and are violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600 for being dispositioned as Non-Cited Violations (NCVs).

NCV Tracking Number	Requirement Licensee Failed to Meet
50-388/01-010-01	Technical Specification (TS) 3.4.3, "Safety Relief Valves (SRVs)," required an allowable pressure band of +/- 1% for opening individual SRVs. Six of eight SRVs tested for Unit 2 on April 12, 2001, opened outside of the TS band, as described in condition report 327430.
50-388/01-010-02	Technical Specification 3.6.1.3, "Primary Containment Isolation Valves," specified that the combined leakage rate for all Secondary Containment Bypass Leakage (SCBL) paths shall not exceed 9 scfh. On March 19, 2001, the as-found SCBL for Unit 2 was 11.13 scfh.

If you deny these non-cited violations, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, Region I, the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001, and the NRC Resident Inspector at the Susquehanna Steam Electric Station.

ATTACHMENT 1

a. List of Items Opened, Closed and Discussed

Opened

None

Opened and Closed

50-388/01-010-01	NCV	Six of eight safety relief valves failed to satisfy the Technical Specification requirements for as-found opening setpoint (section 4OA7)
50-388/01-010-02	NCV	Secondary Containment Bypass Leakage exceeded the Technical Specification limit (section 4OA7)

Closed

50-387,388/01-003-02	URI	Safety System Functional Failure PI Verification (section 4OA1.2)
50-388/01-005-00	LER	Multiple Test Failures of Main Steam Safety Relief Valves (section 4OA3.1)
50-388/01-003-00	LER	Secondary Containment Bypass Leakage Exceeded (section 4OA3.2)

Discussed

None

b. List of Documents Reviewed

None

c. List of Acronyms

ANS	Alert and Notification System
CFR	Code of Federal Regulations
CR	Condition Report
DEP	[Emergency Response Organization] Drill/Exercise Performance
ERO	Emergency Response Organization
F	Fahrenheit
FSAR	[SSES] Final Safety Analysis Report
HPCI	High Pressure Coolant Injection
LPCI	Low Pressure Coolant Injection
NCV	Non-cited Violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PI	Performance Indicator
PPL	PPL Susquehanna, LLC
QA	Quality Assurance
RBCCW	Reactor Building Closed Cooling Water
RBCW	Reactor Building Chilled Water
RHR	Residual Heat Removal
SCBL	Secondary Containment Bypass Leakage
scfh	standard cubic feet per hour
SDP	Significance Determination Process
SRV	Safety Relief Valve
SSC	Structure, System, or Component
SSES	Susquehanna Steam Electric Station
SSFF	Safety System Functional Failure
TBCCW	Turbine Building Closed Cooling Water
TS	Technical Specification
URI	[NRC] Unresolved Item
WO	Work Order