March 6, 2001

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

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION - NRC INSPECTION REPORT 05000387/2001-002, 05000388/2001-002

Dear Mr. Byram:

On February 10, 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 February 23, 2001, with Mr. B. Shriver 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.

Based on the results of this inspection, the inspectors identified one issue of very low safety significance (Green). The issue was determined to involve a violation of NRC requirements. However, because of its very low safety significance and because it has been entered into your corrective action program, the NRC is treating this issue as a Non-cited violation, in accordance with Section VI.A of the NRC's Enforcement Policy. If you deny this non-cited violation, 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.

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If you have any questions please contact me at 610-337-5233.

Sincerely,

/RA/

Curtis J. Cowgill, Chief
Projects Branch 4
Division of Reactor Projects

Docket Nos. 05000387, 05000388
License Nos. NPF-14, NPF-22

Enclosure: Inspection Report 05000387/2001-002, 05000388/2001-002

Attachments: (1) Supplemental Information
(2) NRC's Revised Reactor Oversight Process

cc w/encl:
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REGION I

Docket Nos.:	05000387, 05000388
License Nos.:	NPF-14, NPF-22
Report No.:	2001-002
Licensee:	PPL Susquehanna, LLC
Facility:	Susquehanna Steam Electric Station
Location:	Post Office Box 35 Berwick, PA 18603
Dates:	January 1, 2001 to February 10, 2001
Inspectors:	S. Hansell, Senior Resident Inspector J. Richmond, Resident Inspector A. Blamey, Resident Inspector
Approved by:	Curtis Cowgill, Chief Projects Branch 4 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000387/2001-002, 5000388/2001-002, on 01/01-02/10/2001; PPL Susquehanna, LLC; Susquehanna Steam Electric Station; Units 1&2, Personnel Performance During Non-routine Plant Evolutions and Events

This inspection was conducted by resident inspectors. The inspection identified one green issue of very low safety significance and was classified as a non-cited violation. The significance of findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, “Significance Determination Process” (SDP) (see Attachment 2). Findings for which the SDP does not apply are indicated by “no color” or the severity level of the applicable violation.

A. Inspector Findings

Cornerstone: Barrier Integrity

- Green. On February 1, 2001, the NRC identified that for an 8 hour period PPL staff had sufficient information to declare the Unit-2 reactor core isolation cooling inboard steam supply primary containment isolation valve inoperable but did not declare the valve inoperable. During that time period PPL did not complete required actions in Technical Specification 3.6.1.3 “Primary Containment Isolation Valves,” to close and de-activate another automatic valve in the flow path within four hours. A non-cited violation was identified because the 8 hour time period between the occurrence of the inoperable valve and PPL’s declaration (discovery) that the valve was inoperable exceeded the four hour time period allowed in technical specification to complete required actions.

  This violation was considered to have very low safety significance using the Significance Determination Process because, the barrier function of the control room was not effected and the finding did not represent an actual open pathway in the primary containment since the redundant isolation valve remained operable or closed during this event.

Report Details

Summary of Plant Status

Susquehanna Steam Electric Station (SSES) Unit 1 began the period at full power and operated at full power for the entire report period except for a planned power reduction to 75% power on February 4, 2001, for control rod pattern adjustment. Unit 1 returned to 100% power on February 5, 2001. SSES Unit 2 operated at or near full power for the entire period with exceptions for control rod pattern adjustments, control rod drive maintenance and testing, and to repair a loose ground wire connection on a 500 KV transmission line.

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

1R04 Equipment Alignments (71111.04)

a. Inspection Scope

The inspectors performed partial system walkdowns to verify system and component alignment and note any discrepancies that would impact system operability. The inspectors verified selected portions of redundant or backup systems/trains while a system was 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 2 high pressure coolant injection system during planned maintenance on the Unit 2 reactor core isolation cooling system.
- “A,” “C,” & “D” emergency diesel generators (EDGs) during maintenance on the “E” EDG jacket water low temperature condition (CR 310533).

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

The inspectors reviewed PPL’s Fire Protection Review Report, revision 9, dated August 8, 1997, to determine the required fire protection design features, fire area boundaries, and combustible loading requirements for the areas examined during this inspection. The inspectors then performed walkdowns of these plant areas to assess PPL’s control of transient combustible material and ignition sources, fire detection and suppression capabilities, fire barriers, and any related compensatory measures. The areas included:

Fire Zone 2-1B	Unit 2 Core Spray Pump Room
Fire Zone 2-2A	Unit 2 Remote Shutdown Panel Room
Fire Zone 2-1G	Unit 2 Reactor Building Sump Room

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11)

a. Inspection Scope

On January 17, 24, and 31, 2001, the inspectors reviewed licensed operator performance during annual simulator examinations and PPL's critique of the operators' performance. The inspectors focused on the operating crews' satisfactory completion of crew critical tasks. Critical tasks are limits placed on key reactor plant parameters that will ensure safety margins are maintained during the simulated malfunctions. Also, the evaluation included the operators' adherence to Technical Specifications, emergency plan implementation, and the use of emergency operating procedures. In addition, the ability of the simulator to model the actual plant performance was reviewed.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12)

a. Inspection Scope

The inspectors reviewed PPL's follow-up actions for selected structure, system, or component (SSC) issues, to assess the effectiveness of PPL's maintenance activities. The inspectors reviewed the performance of selected SSCs to verify that problem identification and resolution of Maintenance Rule related issues had been appropriately monitored, evaluated, and dispositioned in accordance with the requirements of 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance," and PPL procedure NDAP-QA-0413, "SSES Maintenance Rule Program." In addition, the inspectors reviewed selected SSC classification, performance criteria and goals as listed in PPL analysis EC-RISK-0528, "Risk Significant Systems, Structures, and Components for the Maintenance Rule and Generic Letter 89-10 Components," EC-RISK-1054, "SSC Availability Performance Criteria for the Maintenance Rule," and EC-RISK-1060, "Acceptable Number of Failures for Risk Significant SSCs in the Maintenance Rule." The specific issues included:

CR 311835 Repeat Maintenance on the residual heat removal system motor operated valve HV151F028A.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessment and Emergent Work (71111.13)

a. Inspection Scope

The inspectors reviewed PPL's assessment and management of selected maintenance activities to assess the effectiveness of PPL's risk management for planned and emergent work. The inspectors compared PPL's risk assessments and risk management actions against 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," dated February 2000, to verify that 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 that risk management action threshold levels were correctly identified, and that appropriate implementation of risk management actions were performed, in accordance with PPL procedures NDAP-QA-1902, "Maintenance Rule Risk Assessment and Management Program," NDAP-QA-0340, "Protected Equipment Program," PSP-22, "Susquehanna Sentinel Program," and the SSES Team Manual. The inspectors reviewed the assessed risk configuration against the actual plant conditions and any in-progress evolutions or external events to verify that the assessment was accurate, complete, and appropriate for the issue. In addition, the inspectors performed control room and field walkdowns to verify compensatory measures, identified by the risk assessments, were appropriately performed. The specific plant configurations included:

January 5, 2001	"E" emergency diesel generator (EDG) substitution for "B" EDG
January 16, 2001	Scheduled unavailability period for "E" EDG fuel oil transfer to the "D" EDG, while the "B" EDG fuel oil storage tank was unavailable
January 17, 2001	“E” EDG inoperable for unplanned starting air filter repairs
January 26, 2001	Unit 1 Startup Bus 10 under-voltage protective relay circuit work (WO 255210)
January 31, 2001	“E” EDG potential inoperability due to low jacket water temperature while the Unit 1 RCIC and Unit 2 Division I LPCI swing bus was inoperable

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

On February 1, 2001, the inspectors reviewed PPL's actions during removal of the Unit 2 reactor core isolation cooling (RCIC) system from service to support planned maintenance work. Specifically, the inspectors reviewed PPL's response to the failure of the RCIC inboard steam supply primary containment isolation valve (PCIV) to fully close. The review included the application of the applicable Technical Specifications and event report requirements.

b. Findings

The inspectors identified that PPL staff had sufficient information to determine that the Unit-2 RCIC inboard steam supply PCIV was inoperable and they did not make the decision regarding the operability until the resident inspector asked about the valve. At 11:30 p.m. on January 31, 2001, operators closed the RCIC inboard steam supply PCIV to depressurize the RCIC steam line. Operators expected the RCIC steam line to depressurize through a steam trap, however the RCIC steam line remained pressurized. At 1:45 a.m. on February 1, operators opened a one inch steam line drain valve. Although the RCIC inboard steam supply PCIV indicated closed and the steam line drain valve was open, the RCIC steam line still remained pressurized. At 2:35 a.m. on February 1, operators closed the RCIC outboard steam supply PCIV and the steam line depressurized as expected. The operations staff identified that the RCIC inboard steam supply PCIV had significant leakage and requested an engineering evaluation to determine whether the magnitude of the leakage exceeded allowable containment penetration leakage limits. The operations staff waited for the engineering evaluation to determine whether the RCIC inboard steam supply PCIV was inoperable. After primary containment integrity was questioned by the resident inspectors, PPL declared the RCIC inboard steam supply PCIV inoperable at 7:30 a.m. on February 1. PPL entered Technical Specification (TS) 3.6.1.3 "Primary Containment Isolation Valves " and completed the TS required actions by 9:43 a.m. The engineering staff subsequently completed their evaluation and confirmed that the valve leakage was excessive which rendered the valve inoperable.

The inspectors concluded that, as a result of the length of time between the occurrence and declaration of the inoperable RCIC inboard steam supply PCIV, PPL did not complete TS required actions within the TS 3.6.1.3 Condition A completion time. Specifically, TS 3.6.1.3 Condition A, states, in part, that if a PCIV is inoperable, isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve within 4 hours. Since the RCIC steam supply inboard PCIV was inoperable as of 11:30 p.m. on January 31, the RCIC outboard steam supply PCIV should have been closed and de-activated prior to 3:30 a.m. on February 1. On February 1, PPL closed the RCIC outboard steam supply PCIV at 2:35 a.m. and electrically de-activated the PCIV at 9:43 a.m. TS 3.6.1.3 Condition F states, in part, if the required action and associated completion time of Condition A is not met, be in MODE 3 within 12 hours. Since PPL did not complete the Condition A actions within 4 hours, a Unit-2 reactor shutdown would have been required to be completed by 3:30 p.m. the same day. Although PPL did not complete the TS required actions within the completion time for Condition A, they did complete the TS required actions for Condition A prior to having to complete the required actions for Condition F.

Although PPL completed the TS required actions for Condition A prior to having to complete the required actions for Condition F, PPL violated TS 3.6.1.3 for reasons stated in Section 8.1.1 of the NRC Enforcement Manual. A violation occurred because the 8 hour time period between the occurrence of the inoperable valve and PPL's declaration (discovery) that the valve was inoperable exceeded the four hour time period allowed in TS to complete the required actions for Condition A. This violation is more than minor because if PPL had not de-activated the redundant PCIV under the same conditions, it could have become a more significant safety concern because TS would have required the plant to be shutdown. This violation affects the Barrier Integrity Cornerstone and was considered to have very low safety significance (green) using the Significance Determination Process because, the barrier function of the control room was not effected and the finding did not represent an actual open pathway in the primary containment since the redundant isolation valve remained operable or closed during this event. This violation of TS 3.6.1.3 is being treated as a Non-Cited Violation (NCV), consistent with Section VI.A of the NRC Enforcement Policy, issued on May 1, 2000 (65FR25368). This issue is documented in PPL's corrective action program as condition report 310810. **(NCV 05000388/2001002-01)**

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. The inspectors verified that the operability determinations were performed as required by procedure NDAP-QA-0703, “Operability Assessments.” The inspectors used the Technical Specifications, Technical Requirements Manual, Final Safety Analysis Report, and associated Design Basis Documents as references. The specific issues included:

CR 306015	Unit 2 "A" core spray pump motor meggar unexpected results
CR 309098	"D" EDG, "E" EDG cylinder head test valve problems
CR 310533	“E” EDG potential inoperability due to jacket water low temperature
CR 310810	Unit 2 RCIC steam line pressure did not decrease

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 and reviewed selected PPL test data. The inspectors assessed the adequacy of the test methodology for the scope of the maintenance work which had been performed. The inspectors assessed the adequacy of the test acceptance criteria to demonstrate that the tested components satisfied the design, licensing bases, and Technical Specification requirements. The specific issues reviewed included:

CR 304332 Secondary containment zone 2 damper failure (HD27576A), SO-234-001, " 92 day Secondary Containment Isolation Damper Time Test"

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors reviewed the applicable Technical Specifications and observed the performance of selected portions of surveillance tests. The inspectors observed portions of the tests in the plant and reviewed selected test results to verify that the systems and components were capable of performing their safety functions. The observed or reviewed surveillance tests included:

SO-250-002 Quarterly Reactor Core Isolation Cooling (RCIC) Flow Verification
SO-034-001 Quarterly Zone III Isolation Damper Timing

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

.1 Annual Performance Indicator Verification (71151)

a. Inspection Scope

The inspectors reviewed PPL records to assess the accuracy and completeness of selected NRC performance indicator (PI) data. The records reviewed included selected Technical Specification limiting condition for operation logs, system surveillance tests, and condition reports for the previous 36 months. The specific indicators included:

- High Pressure Coolant Injection System Unavailability
- Reactor Core Isolation Cooling System Unavailability

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up (71153)

.1 (Closed) LER 05000388/00-005-00 Inadvertent Engineered Safety Feature Actuation Due to Reactor Protection System (RPS) Electrical Protection Assembly (EPA) Breaker Trip

On December 5, 2000, the Unit 2 “B” reactor protection system (RPS) power was lost due to an EPA breaker trip. The failure resulted in a RPS “B” half scram and corresponding containment isolations. PPL replaced the EPA logic card and monitored EPA performance for several days. This is the first inservice EPA breaker logic card failure since 1991. No new issues were identified in this review; no violations of NRC requirements were identified. This issue was documented in condition report CR 299462. This LER is closed.

4OA6 Meetings

.1 Exit Meeting Summary

On February 23, 2001, the resident inspectors presented the inspection results to Mr. B. Shriver and other members of your staff who acknowledged the findings.

The inspectors asked PPL whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

Attachment 1

SUPPLEMENTAL INFORMATION

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None.

Opened and Closed

05000388/2001002-01	NCV	Unit 2 Reactor Core Isolation Cooling System Inboard Steam Supply Primary Containment Isolation Valve Failed to Close (1R14)

Closed

05000388/00-005-00	LER	Inadvertent Engineered Safety Feature Actuation Due to Reactor Protection System (RPS) Electrical Protection Assembly (EPA) Breaker Trip (4OA3)

LIST OF ACRONYMS USED

CFR	Code of Federal Regulations
CR	Condition Report
EDG	Emergency Diesel Generator
EPA	Electrical Protection Assembly
FR	Federal Register
FSAR	[SSES] Final Safety Analysis Report
HPCI	High Pressure Coolant Injection
KV	Kilovolts (1000 volts)
LER	Licensee Event Report
LPCI	Low Pressure Coolant Injection
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
PCIV	Primary Containment Isolation Valve
PI	Performance Indicator
PPL	PPL Susquehanna, LLC
RCIC	Reactor Core Isolation Cooling
RPS	Reactor Protection System
SSC	Structure, System, or Component
SSES	Susquehanna Steam Electric Station
TS	Technical Specification

Attachment 2

NRC's REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety	**Radiation Safety**	**Safeguards**
• Initiating Events • Mitigating Systems • Barrier Integrity • Emergency Preparedness	• Occupational • Public	• Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.