January 30, 2004

Mr. Bryce L. Shriver
Senior Vice President and Chief Nuclear Officer
Susquehanna Steam Electric Station
PPL Susquehanna, LLC
769 Salem Blvd., NUCSB3
Berwick, PA 18603-0467

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION - NRC INTEGRATED INSPECTION REPORT 05000387/2003005 AND 05000388/2003005

Dear Mr. Shriver:

On December 31, 2003, the US Nuclear Regulatory Commission (NRC) completed an inspection at your Susquehanna Steam Electric Station Units 1 and 2. The enclosed integrated inspection report presents the results of that inspection, which was discussed with R. Anderson, Vice President - Nuclear Operations, and other members of your staff on January 9, 2004.

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.

As a result of a recent equipment problem associated with bolting of the "A" emergency diesel generator (EDG) governor, the previously identified open unresolved item associated with an improperly tightened bolt on the "D" EDG governor will remain unresolved. Additional inspection is needed to further understand the performance deficiency, extent of condition, any potential common mode failures, and PPL actions. The safety significance of the performance deficiency will be determined following the completion of the inspection.

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

Since the terrorist attacks on September 11, 2001, the NRC has issued five Orders and several threat advisories to licensees of commercial power reactors to strengthen licensee capabilities, improve security force readiness, and enhance controls over access authorization. In addition to applicable baseline inspections, the NRC issued Temporary Instruction 2515/148, “Inspection of Nuclear Reactor Safeguards Interim Compensatory Measures,” and its subsequent revision, to audit and inspect licensee implementation of the interim compensatory measures required by the Order. Phase 1 of TI 2515/148 was completed at all commercial nuclear power plants during calendar year (CY) ‘02, and the remaining inspection activities for Susquehanna were completed in April 2003. The NRC will continue to monitor overall safeguards and security controls at Susquehanna.

In accordance with 10CFR2.790 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 the NRC’s document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

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

Enclosure: Inspection Report 05000387/2003005, 05000388/2003005
w/Attachment: Supplemental Information

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REGION I

Docket Nos.:	50-387, 50-388
License Nos.:	NPF-14, NPF-22
Report No.:	05000387/2003005, 05000388/2003005
Licensee:	PPL Susquehanna, LLC
Facility:	Susquehanna Steam Electric Station
Location:	769 Salem Boulevard Berwick, PA 18603
Dates:	September 28, 2003 to December 31, 2003
Inspectors:	S. Hansell, Senior Resident Inspector J. Richmond, Resident Inspector F. Jaxheimer, Resident Inspector J. Caruso, Sr. Operations Engineer J. Noggle, Senior Health Physicist N. McNamara, EP Inspector D. Florek, Sr. Project Engineer S. Chaudhary, Reactor Engineer G. Johnson, Reactor Engineer P. Presby, Reactor Engineer W. Schmidt, Senior Reactor Analyst
Approved by:	Mohamed M. Shanbaky, Chief Projects Branch 4 Division of Reactor Projects

CONTENTS

SUMMARY OF FINDINGS

IR 05000387/2003005, 05000388/2003005; 09/28/2003 - 12/31/2003; Susquehanna Steam Electric Station, Units 1 and 2. Temporary Modifications and Identification and Resolution of Problems.

The report covered a 3 month period of inspection by resident inspectors and announced inspections by a regional senior health physicist, senior operations engineer, senior reactor analyst, reactor engineers, senior project engineer, and an emergency preparedness specialist. Four Green non-cited violations (NCVs) were identified. The significance of most findings are indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609 "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. NRC Identified Findings

Cornerstone: Barrier Integrity

- **Green** The inspectors identified a non-cited violation of very low safety significance of Technical Specification 5.4.1, because PPL did not adequately implement alarm response procedure written instructions to evaluate and correct indicated low differential pressure (D/P) for the refuel floor secondary containment.

  This finding affects the Barrier Integrity cornerstone and is more than minor because it is associated with the human performance attribute and adversely affects the objective of the Barrier Integrity cornerstone to provide reasonable assurance that physical design barriers provide protection against a radiological release. This finding is of very low safety significance because the finding only represented a potential degradation of the radiological barrier function provided for the spent fuel pool.

  This finding was related to the Human Performance cross-cutting area because operators did not adequately implement alarm response procedures to evaluate and correct indicated low D/P for the refuel floor secondary containment. (Section 1R23)

- **Green** The inspectors identified a non-cited violation of very low safety significance of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Actions," because PPL did not promptly identify a condition adverse to quality. From July to December 2003, multiple evaluations by PPL did not identify that an American Society of Mechanical Engineers (ASME) fail-safe closure test was required to be performed on main steam isolation valves. The required test had not been performed since 1994.

  This finding affects the Barrier Integrity cornerstone and is more than minor because, similar to example 1.c in the NRC Inspection Manual 0612, Appendix E, "Example of Minor Issues," a required surveillance test was not performed. This finding is of very low safety significance because the finding did not represent an actual open pathway in the physical integrity of the reactor containment.

A contributing cause of this finding is related to the Problem Identification and Resolution cross-cutting area because, although PPL had multiple opportunities, PPL did not promptly identify a condition adverse to quality regarding ASME testing for the main steam isolation valves. (Section 4OA2.2)

- **Green** The inspectors identified a non-cited violation of very low safety significance of Technical Specification 5.5.6, "Inservice Testing Program." Since initial plant startup, PPL did not perform valve seat leakage testing on the SDV vent and drain valves, and did not have an adequate justification that any leakage through these valves would be inconsequential.

  This finding affects the Barrier Integrity cornerstone and is more than minor because, similar to example 1.c in the NRC Inspection Manual Chapter 0612, Appendix E, "Example of Minor Issues," a required surveillance test was not performed. This finding is of very low safety significance because the finding did not represent an actual open pathway in the physical integrity of the reactor containment.

  A contributing cause of this finding is related to the Problem Identification and Resolution cross-cutting area because PPL's corrective actions for a similar finding were narrowly focused and limited in scope. (Section 4OA2.3)

- **Green** The inspectors identified a non-cited violation of very low safety significance of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Actions." PPL did not promptly identify a condition adverse to quality and did not enter it into its corrective action program as a condition report. Specifically, following changes made to the Probabilistic Risk Analysis (PRA), PPL did not identify the need to perform an evaluation utilizing the current PRA to verify that a 1998 change to Technical Specification 3.1.8 action statements was still valid.

  This finding is more than minor because it is associated with the configuration control attribute and affects the objective of the Barrier Integrity cornerstone to provide reasonable assurance that physical design barriers provide protection against a radiological release. This finding was determined to be of very low safety significance because it did not result in an actual open pathway in the physical integrity of a fission product barrier.

  A contributing cause of this finding is related to the Problem Identification and Resolution cross-cutting area in that PPL had prior opportunities to identify and correct this issue. (Section 4OA2.4)

B. Licensee Identified Violation

A violation of very low safety significance, which was identified by PPL, has been reviewed by the inspectors. Corrective actions taken or planned by PPL have been entered into PPL's corrective action program. This violation and corrective actions are listed in Section 4OA7 of this report.

Report Details

Summary of Plant Status

Susquehanna Steam Electric Station (SSES) Unit 1 began the inspection period shutdown in a maintenance outage. The unit was restarted on September 28, and achieved full power on October 2. The reactor operated at or near full power for the remainder of the inspection period, except for control rod pattern adjustments.

Unit 2 operated at or near full power during the inspection period, except for control rod pattern adjustments, reactor feed pump maintenance, and reactor recirculation motor generator maintenance.

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

1R04 Equipment Alignments

1. Partial System Walkdowns (71111.04Q - 3 samples)

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. This inspection activity represented 3 samples. The walkdowns included the following systems:

- Units 1 and 2 all EDGs and startup transformer T-20 prior to removing T-10 transformer for replacement
- Units 1 and 2 T-10 startup transformer during replacement, and one time Technical Specification change to allow one offsite power source out for 10 days
- Units 1 and 2 reactor building secondary containment Zones 1, 2, and 3 during intermittent low building differential pressure

b. Findings

No findings of significance were identified.

2. Complete System Walkdowns (71111.04S - 1 sample)

a. Inspection Scope

The inspectors conducted a detailed review of the alignment and condition of the Unit Common emergency service water (ESW) system, during a related review of a "D" emergency diesel generator (EDG) availability issue. The inspectors used PPL procedures and other documents listed in Attachment 1 to verify proper system alignment. This inspection activity represented 1 sample.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

1. Routine Plant Area Observations (71111.05Q - 6 Sample)

a. Inspection Scope

The inspectors reviewed PPL's fire protection program to determine the required fire protection design features, fire area boundaries, and combustible loading requirements for selected areas. The inspectors walked down those areas to assess PPL's control of transient combustible material and ignition sources, fire detection and suppression capabilities, fire barriers, and any related compensatory measures to assess PPL's fire protection program in those areas. The inspectors reviewed the respective pre-fire action plan procedures for the inspected areas. This inspection activity represented 6 samples. The inspected areas included:

- Unit 1 Reactor Building High Radiation area for the high pressure coolant injection (HCPI) system Steam Supply and RHR F015 valve
- Unit 1 & 2 startup and auxiliary 13kV bus rooms
- Unit 2 HPCI and reactor core isolation cooling (RCIC) system rooms
- Unit 1 & 2 4kV bus rooms
- Unit 1 Fire Zone 1-5A-S, Standby Control Area & Nuclear Boiler Instrumentation Racks
- Unit 2 Fire Zone 2-5A-N, Standby Control Area & Nuclear Boiler Instrumentation Racks

b. Findings

No findings of significance were identified.

2. Station Fire Brigade Performance (71111.05A - 1 Sample)

a. Inspection Scope

On December 9, the inspectors observed an announced fire brigade drill in the radiological control area. The fire was a simulated oil fire at the Unit 2 hydrogen seal oil skid. The inspectors assessed PPL’s strategies to fight a fire on-site and to evaluate the readiness of PPL to prevent and fight fires.

The inspectors observed the fire brigade members don protective clothing and turnout gear. In addition, the inspectors observed the fire fighting equipment brought to the fire area scene to evaluate whether sufficient equipment was available for the simulated fire. The inspectors observed fire fighting directions and radio communications between the brigade leader, brigade members, and the control room. The inspectors reviewed the post drill critique to evaluate if the drill objectives' acceptance criteria were satisfied. This inspection activity represented 1 sample.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed PPL's external flood analysis, flood mitigation procedures, and design features to verify whether they were consistent with the PPL design requirements and industry standards. The inspectors walked down selected risk significant plant areas, including the moats and surrounding areas for large on-site tanks. The inspectors evaluated the condition and adequacy of flood detectors, sump pumps, sump level alarm circuits, and other flood protection design features to assess whether the flood protection design features were adequate and operable. During the walk downs, the inspectors also evaluated whether there were any unidentified or unanalyzed sources of flooding, including holes and unsealed penetrations in floors and walls. This inspection activity represented 1 sample. The specific areas included:

- "A" through "D" Emergency Diesel Generator Rooms
- Emergency Service Water (ESW) Building
- ESW / RHRSW Valve Vault (external only)
- Unit 1 and 2 Condensate Storage and Refueling Water Tanks

The inspectors reviewed PPL's flood mitigation procedures, flood alarm response procedures, and selected preventative maintenance tasks for flood detectors and flood barriers to evaluate whether component functionality was routinely verified. In addition, the inspectors reviewed PPL's corrective action program, including system health reports, and interviewed selected maintenance personnel to verify whether previous flood related issues had been appropriately identified, evaluated, and resolved. The following procedures were included in the review:

- NE-94-001, Section 5.2, "Susquehanna Individual Plant Evaluation for External Events - Floods"
- FSAR Section 2.4.2, "Hydrologic Engineering - Floods"
- FSAR Section 3.4, "Water Level (Flood) Design"
- EC-RISK-1024, "External Flood Effects"
- EC-PIPE-1032, Section 4.2, 4.3, and 4.5, "Moderate Energy Pipe Crack Evaluation - ESW Pumphouse, ESW Valve Vault, and Diesel Generator Bldg."
- RTPMs 426227 and 516607, "RHR / ESW Manhole Monthly Inspection"

b. Findings

No significant findings were identified.

1R11 Licensed Operator Requalification

1. Routine Licensed Operator Requalification (71111.11Q - 1 Sample)

a. Inspection Scope

On November 18, 2003, the inspectors observed licensed operator performance in the simulator during the operator re-qualification training. The inspectors compared their observations to Technical Specifications, emergency plan implementation, and the use of emergency operating procedures. The inspectors' evaluation focused on the operating crew's satisfactory completion of crew critical tasks, and satisfactory implementation of the emergency plan and emergency action level (EAL) classifications for the simulated plant conditions. Critical tasks are operational limits placed on key reactor plant and containment parameters that will ensure safety margins are maintained during the simulated malfunctions. The review included a comparison of the simulator's ability to model the actual plant performance. The inspectors also evaluated PPL's critique of the operators' performance to identify discrepancies and deficiencies in operator training. This inspection activity represented 1 sample. The observed training scenario included:

- Operating Crew performance during emergency plan response training

b. Findings

No findings of significance were identified.

2. Biennial Licensed Operator Requalification Program (71111.11B - 1 Sample)

a. Inspection Scope

The following inspection activities were performed using NUREG-1021, revision 8, "Operator Licensing Examination Standards for Power Reactors," Inspection Procedure 71111.11, "Licensed Operator Requalification Program," and NRC Manual Chapter 0609, Appendix I, "Operator Requalification Human Performance Significance Determination Process," as acceptance criteria, 10 CFR 55.46 Simulator Rule (sampling basis). These inspection activities were performed for both units.

The inspectors reviewed documentation of operating history since the last requalification program inspection. The inspectors also discussed facility operating events with the NRC resident staff. Documents reviewed included NRC inspection reports and 11 Condition Reports (450281, 452936, 453216, 477808, 464851, 479495, 482335, 506241, 508508, 510950, 512239) that involved human performance and Technical Specification compliance issues.

The inspectors also verified that documented Requalification Training Schedule changes were made, specifically to address events. A sample of five training records were reviewed to verify completion of this training.

The Inspectors reviewed three RO and three SRO comprehensive biennial written exams administered in 2002 (i.e., administered exam weeks 1, 3, and 6). In addition, the inspectors reviewed three sets of Scenarios and JPMs administered during this current exam cycle (i.e., weeks 1, 3, and 6) to ensure the quality of these exams met or exceeded the criteria established in the Examination Standards and 10 CFR 55.59.

The inspectors observed the administration of operating examinations to two crews (i.e., Bravo Shift). The operating examination consisted of two simulator scenarios for each crew and one set of five job performance measures (JPMs) administered to each individual.

For the site specific simulator, the inspectors observed simulator performance during the conduct of the examinations, reviewed simulator performance tests (e.g., steady state performance tests, selected transient tests, and licensed operator re-qualification program scenario-based tests), and discrepancy reports to verify compliance with the requirements of 10CFR55.46. Specific documents, tests, and data reviewed are listed in Attachment 1 to this report.

Conformance with operator license conditions was verified by reviewing the following:

- Attendance records for the most recent year training cycle
- 10 medical records (5 SRO; 5 RO) and confirmed all records were complete, that restrictions noted by the doctor were reflected on the individual's license and that the exams were given within 24 months.

- Proficiency watch-standing and reactivation records. A sample of six licensed operator watch-standing documentation was reviewed for the current and prior quarter to verify currency and conformance with the requirements of 10CFR55.

Remediation training records for the prior two years were reviewed by assessing five instances of evaluation failures, which included one operating exam crew failure.

PPL's feedback system The inspectors interviewed Instructors, training/operations management personnel, and licensed operators (i.e., 2 training supervisors, 2 instructors, 2 evaluators, and licensed operators 4 ROs and 4 SROs) for feedback regarding the implementation of the licensed operator requalification program to ensure the requalification program was meeting their needs and responsive to their noted deficiencies/recommended changes. In addition, 16 plant and industry events/changes were reviewed to ensure that these issues were adequately addressed in the Requalification Training Program.

On October 31, 2003, the inspectors conducted an in-office review of PPL's requalification exam results. These results included the annual operating test only (i.e., the comprehensive written exam was administered last year). The inspection assessed whether pass rates were consistent with the guidance of NRC Manual Chapter 0609, Appendix I, “Operator Requalification Human Performance Significance Determination Process.” The inspectors verified that:

- Crew failure rate on dynamic simulator was less than 20%. (Failure rate was 0%)
- Individual failure rate on the dynamic simulator test was less than or equal to 20% (Failure rate was 0%)
- Individual failure rate on the walk-through test (JPMs) was less than or equal to 20% (Failure rate was 1.7%)
- More than 75% of the individuals passed all portions of the exam (98.3% of the individuals passes all portions of the exam)

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

1. Routine Maintenance Effectiveness Observations (71111.12Q - 2 Samples)

a. Inspection Scope

The inspectors evaluated PPL’s work practices and follow-up corrective actions for selected system, structure, or component (SSC) issues to assess the effectiveness of PPL's maintenance activities. The inspectors reviewed the performance history of those SSCs and assessed PPL’s extent of condition determinations for these issues with potential common cause or generic implications to evaluate the adequacy of PPL’s corrective actions. The inspectors reviewed PPL's problem identification and resolution actions for these issues to evaluate whether PPL had appropriately monitored, evaluated, and dispositioned the issues in accordance with PPL procedures and the requirements of 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance." In addition, the inspectors reviewed selected SSC classification, performance criteria and goals, and PPL's corrective actions that were taken or planned, to verify whether the actions were reasonable and appropriate. This inspection activity represented 2 samples. The following issues were reviewed:

Equipment Issues

- Units 1 and 2 reactor building secondary containment intermittent low building differential pressure issues, CR 527005
- Units 1 and 2 Common Harmony 68AW lubricating oil low viscosity issue (Multiple components)

Procedures and Documents

- Interim Oil Control Process, revision 0, issued 12/4/2003
- Harmony 68 and Gulfcrest 32 oil chemical sample and viscosity results
- CR 529334, Unexpected viscosity values, Operability Assessment
- CR 529658, RHRSW Pump Low viscosity - Upper Motor Bearing Sample
- CR 530069, Programmatic Weaknesses in the SSES Oil Control Program

b. Findings

No significant observations or findings were identified.

2. Biennial Maintenance Effectiveness Periodic Evaluation (71111.12B - 1 Sample)

a. Inspection Scope

The inspector reviewed PPL's periodic evaluations required by 10 CFR 50.65 (a)(3) for Units 1 & 2, to verify that structures, systems and components (SSCs) within the scope of the maintenance rule were included in the evaluations and also to ensure that balancing of reliability and unavailability was given adequate consideration. The inspector reviewed PPL's most recent periodic maintenance rule evaluation report which covered the period from December 31, 2000, through December 31, 2002.

The inspector selected safety significant systems that were in an (a)(1) status to verify that (1) goals and performance criteria were appropriate, (2) industry operating experience was considered, (3) corrective action plans were effective, and (4) performance was being effectively monitored. As of December 15, 2003, there were four SSCs in an (a)(1) status, out of which two were in the process of monitoring, and two in the status of development of corrective action. The inspector also reviewed PPL's assessment of the balance between reliability and availability for these systems:

- 480V Motor Control Centers, System 206, Unit 2
- 120V Lighting and Misc. Distribution, Systems 107/207, Unit 1 & 2
- Containment and Suppression, Systems 159/259, Units 1 & 2
- Process and Area Radiation Monitoring, System 079, Common Units 1 & 2

The inspector reviewed the following (a)(2) high risk significant systems to verify that performance was acceptable:

- Control Structure HVAC, System 030, Common Units 1 & 2
- Zone I HVAC Supply, System 134A, Unit 1
- Zone IIA HVAC Supply, System 234A, Unit 2
- Reactor Non-nuclear Instrumentation, System 280, Unit 2

This inspection activity represented 1 sample.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments & Emergent Work Evaluation (71111.13 - 4 Samples)

a. Inspection Scope

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

The inspectors reviewed scheduled and emergent work activities with licensed operators and work-coordination personnel to verify whether risk management action threshold levels were correctly identified. In addition, the inspectors compared the assessed risk configuration to the actual plant conditions and any in-progress evolutions or external events to evaluate whether the assessment was accurate, complete, and appropriate for the issue. The inspectors performed control room and field walkdowns to verify whether the compensatory measures identified by the risk assessments were appropriately performed. The inspectors also utilized General Electric Service Information Letter, SIL 600, as a technical reference for recirculation pump vibration at increased speed. This inspection activity represented 4 samples. The selected maintenance activities included:

- Unit 1 "A" EHC Pump filter cleaning during main turbine startup and generator sync to grid, CR 514318
- Units 1 and 2 T-10 outage, TP-003-012
- Unit 2 Recirculation flow and vibration testing, CR 510260, and AR 517380
- Unit Common fire protection system work during period of heightened security awareness

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15 - 3 Samples)

a. Inspection Scope

The inspectors reviewed operability determinations that were selected based on risk insights, to assess the adequacy of the evaluations, the use and control of compensatory measures, and compliance with the Technical Specifications. In addition, the inspectors reviewed the selected operability determinations to verify whether the determinations were performed in accordance with NDAP-QA-0703, "Operability Assessments." The inspectors used the Technical Specifications, Technical Requirements Manual, Final Safety Analysis Report (FSAR), and associated Design Basis Documents as references during these reviews. This inspection activity represented 3 samples. The issues reviewed included:

- Unit 1 inadvertent control rod drive (CRD) uncoupling and control rod blade excessive friction, CR 513102, 10/7
- Unit 1 higher than normal ECCS keepfill pressure, CR 514658
- Unit 2 “B” Recirculation Pump High Vibration during High Speed Stop testing, CR 523932

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19 - 5 Samples)

a. Inspection Scope

The inspectors observed portions of post maintenance testing activities in the field to determine whether the tests were performed in accordance with the approved procedures. The inspectors assessed the test's adequacy by comparing the test methodology to the scope of maintenance work performed. In addition, the inspectors evaluated the test acceptance criteria to verify whether the test demonstrated that the tested components satisfied the applicable design and licensing bases and the Technical Specification requirements. The inspectors reviewed the recorded test data to determine whether the acceptance criteria were satisfied. This inspection activity represented 5 samples. The post maintenance testing activities reviewed included:

- Unit 1 Main Turbine Oil Pressure Trip Test after replacement of the Control Room Test Switch, WO 512669, on 9/29
- Units 1 and 2 off-site power transformer T-20 tap changer inspection, ERPM 451631, WO 491747
- Unit 1 "B" CRD pump replacement, TP-155-014 "pump performance curve"
- Units 1 and 2 off-site power transformer T-10 operability testing following transformer replacement, SO-013-016, SI-013-250, and TP-003-010
- "E" EDG Post Maintenance Testing following Governor linkage bolt replacement

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22 - 3 Samples)

a. Inspection Scope

The inspectors observed portions of selected surveillance test activities in the control room and in the field and reviewed the test data results. The inspectors compared the test result to the established acceptance criteria and the applicable Technical Specification or Technical Requirements Manual operability and surveillance requirements to evaluate whether the systems were capable of performing their intended safety functions. This inspection activity represented 3 samples. The observed or reviewed surveillance tests included:

- Unit 2 Standby Liquid Control Quarterly Surveillance after "B" SLC pump maintenance, 11/19/03
- Unit 2 RCIC Flow Surveillance, SO-150-002, 10/31/03
- Unit 1 RHRSW division 1 DC control power automatic transfer logic test, SE-116-312, 12/22/03

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modification (71111.23 - 2 Samples)

a. Inspection Scope

The inspectors reviewed temporary plant modifications to determine whether the temporary changes adversely affected system or support system availability, or adversely affected a function important to plant safety. The inspectors reviewed the associated system design bases, including the Final Safety Analysis Report (FSAR), Technical Specifications, and assessed the adequacy of the safety determination screenings and evaluations. The inspectors also assessed configuration control of the temporary changes by reviewing selected drawings and procedures to verify whether appropriate updates had been made. The inspectors compared the actual installations to the temporary modification documents to determine whether the implemented changes were consistent with the approved documents. The inspectors reviewed selected post installation test results to verify whether the actual impact of the temporary changes had been adequately demonstrated by the test. This inspection activity represented 2 samples. The following temporary modifications and documents were included in the review:

- Fire Protection System, with 18 isolated OS&Y Valves, for Charcoal Bed fire stations - change of deluge stations from automatic to manual response
- Unit 2 Reactor Bldg. ventilation air supply blocked with plastic sheeting to maintain Zone 2 negative pressure

During November 2003, the inspectors reviewed PPL's practice of using plastic sheeting to cover the reactor building (RB) HVAC intake filters, to control building differential pressure (D/P) within design specifications. Inspectors walked down the RB HVAC local control panel indications to evaluate the temporary modification impact on interfacing system performance. During the walkdown, the indicated D/P for the refuel floor secondary containment momentarily degraded to less than the minimum allowed Technical Specification value and caused a local low D/P alarm. The inspectors reviewed PPL's response to observed secondary containment low D/P alarms. In addition, the inspectors reviewed PPL's evaluation and associated corrective actions for previously identified HVAC problems. The specific procedures and documents reviewed are listed in Attachment 1 to this report.

b. Findings

Introduction

The inspectors identified a non-cited violation of very low safety significance of Technical Specification (TS) 5.4.1, because PPL's did not adequately implement alarm response procedure written instruction to evaluate and correct indicated low differential pressure for the refuel floor secondary containment.

Description

On November 19, the inspectors observed that the Unit 2 RB and refuel floor secondary containment ventilation were degraded because:

- Secondary containment D/P gauges in the RB were fluctuating below the TS minimum allowed value of 0.25 inches water gauge (wg)
- Low D/P alarms were sounding (setpoint was 0.15 inches wg) at the local control panels 2C275 and 2C276
- Secondary containment D/P gauges in the main control room were fluctuating below the TS minimum allowed value of 0.25 inches wg (safety related standby gas treatment system gauges)
- Main control room alarms were sounding for Unit 2 RB and refuel floor "Fan Room Trouble"

The licensed plant control operators (PCOs) responded to the control room fan trouble alarms and observed that the control room indicators for Unit 2 RB and refuel floor secondary containment D/P were fluctuating to less than the local panel alarm setpoint of 0.15 inches wg. The PCOs determined that the building ventilation supply and exhaust fans were still running and that there were windy conditions outside. Although the PCOs could not determine the cause for the secondary containment low D/P condition from the control room, they assumed that the changes in D/P were caused by windy weather conditions. The non-licensed nuclear plant operator (NPO) did not respond to the 2C275 and 2C276 local panel alarms in the RB until approximately one and one half hours after the control room alarm had been received.

The inspectors determined that an NPO was required to respond to the local panel alarms in the RB to acknowledge, determine, and correct the cause of a secondary containment low D/P condition. Alarm response procedure LA-2276-001 required specific actions to be performed in the RB, which could not be performed in the main control room, to evaluate secondary containment integrity and correct the low D/P condition, including:

- Observe position of the exhaust fan damper position
- Notify PCO that alarm condition could constitute TS LCO under SR 3.6.4.1.1
- Upon direction of the PCO, verify refuel floor integrity by performing SO-000-010
- Ensure instrument air supplied to the refuel floor exhaust fan discharge damper and control panel 2C276

- If low D/P exists with outside air temperature low and dropping, then adjust outside air supply louvers in the closed direction to control D/P
- Block individual air supply filters as necessary (plastic bags are used to perform this step)
- If low D/P cannot be corrected, enter ON-234-002, "Low Reactor Building Differential Pressure"

When the NPO arrived at the local panel, the wind conditions had stabilized and the low D/P condition cleared. The inspectors determined that operators did not adequately implement written alarm response procedures because for approximately 90 minutes operators did not take required actions to maintain secondary containment differential pressure at greater than or equal to 0.25 inches of vacuum.

Analysis

This finding is a performance deficiency because operator response to a local panel alarm instructions is required by Technical Specification 5.4.1. This finding is more than minor because it is associated with the human performance attribute and adversely affects the objective of the Barrier Integrity cornerstone to provide reasonable assurance that physical design barriers provide protection against a radiological release. Specifically, PPL did not adequately evaluate and correct an off-normal condition when the indicated refuel floor secondary containment D/P was degraded. This finding affects the Barrier Integrity cornerstone because it is associated with the integrity of the secondary containment.

This finding is of very low safety significance (Green) using the NRC IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." A Phase-1 significance determination evaluation screened this finding as Green because the finding only represented a potential degradation of the radiological barrier function provided for the spent fuel pool area.

This finding was related to the Human Performance cross-cutting area because operators did not adequately implement alarm response procedures to evaluate and correct indicated low D/P for the refuel floor secondary containment.

Enforcement

Technical Specification 5.4.1, "Administrative Controls - Procedures," requires, in part, that written procedures shall be established and implemented as recommended in NRC Regulatory Guide (RG) 1.33 Appendix A. RG 1.33 Appendix A, section 4.j(2) required procedures for Containment Ventilation System, and Appendix A section 5 required procedures for abnormal and off-normal conditions. PPL alarm response procedures AR-206-001.C16 and AL-2276-001.C08 provided instructions to evaluate and correct off-normal conditions for refuel floor secondary containment.

Contrary to the above, on November 19, 2003, refuel floor secondary containment experienced periods of indicated low differential pressure and plant operators did not adequately implement alarm response procedure written instructions to evaluate and correct indicated low differential pressure for the refuel floor secondary containment.

Because this violation is of very low safety significance and PPL entered this finding into their corrective action program (CR 528151), this violation is being treated as a non-cited violation (NCV), consistent with Section VI.A of the NRC Enforcement Policy. **(NCV 05000387,388/2003005-01)**

1EP4 Emergency Action Level & Emergency Plan Changes (711114.04 - 2 Samples)

a. Inspection Scope

A regional in-office review was conducted of PPL submitted revisions to the emergency plan, implementing procedures and Emergency Action Level (EAL) changes which were received by the NRC during the period of January through December 2003. A thorough review was conducted of aspects of the plan related to the risk significant planning standards (RSPS), such as classifications, notifications and protective action recommendations. A limited review was conducted for non-RSPS portions. These changes were reviewed against 10 CFR 50.54(q) to ensure that the changes did not decrease the effectiveness of the plan, and that the changes as made continue to meet the standards of 10 CFR 50.47(b) and the requirements of Appendix E. These changes are subject to future inspections to ensure that the impact of the changes continues to meet NRC regulations. The inspection was conducted in accordance with NRC Inspection Procedure 71114, Attachment 04, and the applicable requirements in 10 CFR 50.54(q) were used as reference criteria. This inspection activity represented 2 samples.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY
Cornerstones: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

a. Inspection Scope

Walkdowns of the Unit 1 and Unit 2 spent fuel pools were performed to verify that there were adequate physical controls for highly activated reactor hardware in storage.

Walkdowns of the Unit 1 and Unit 2 reactor and turbine buildings were conducted during September 29 - 30, 2003, to verify the radiological postings using a radiation survey instrument and to verify the adequacy of locked high radiation areas by challenging the locked doors and gates. On September 30, 2003, an inventory of high and very high radiation area keys was conducted at the Unit 2 radiological controlled area access point with respect to procedure HP-TP-311, revision 23, "Locking, Barricading and Key Control," and Technical Specification 5.7 requirements.

The inspector reviewed one NRC Performance Indicator issue that was documented in Section 4OA2. There were no internal exposures >50 mrem CEDE during 2003.

This inspection activity, when considered with previous inspections in this area, completes all available samples in this inspection area.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151 - 16 Samples)

a. Inspection Scope

Reactor Safety Indicators (14 samples)

The inspectors reviewed PPL's performance indicator (PI) data, for the period of September 2002 through September 2003, to verify whether the PI data was accurate and complete. No problems were noted with the PI accuracy and completeness. The inspectors examined selected samples of PI data, PI data summary reports, and plant records. The inspectors compared the PI data against the guidance contained in Nuclear Energy Institute (NEI) 99-02, revision 2, "Regulatory Assessment Performance Indicator Guideline." This inspection activity represented 14 samples. The following indicators and PPL documents were included in this review:

Mitigating Systems Cornerstone Performance Indicators

- Units 1 & 2 Emergency AC Power System Unavailability
- Units 1 & 2 Residual Heat Removal System Unavailability

NRC Initiating Events Performance Indicators

- Units 1 & 2 Unplanned Scrams per 7000 Critical Hours
- Units 1 & 2 Scrams With Loss of Normal Heat Removal
- Units 1 & 2 Unplanned Power Changes per 7000 Critical Hours

NRC Barrier Integrity Performance Indicators

- Units 1 & 2 Reactor Coolant System (RCS) dose equivalent iodine specific activity
- Units 1 & 2 RCS Identified leak rate measured by the drywell leakage calculation

PPL Documents

- Units 1 & 2 Control Room Logs
- NDAP-QA-0737, "Regulatory Performance Assessment"
- Technical Specification 3.4.4, "RCS Operational Leakage"
- SO-100/200-006, "Shiftly Surveillance Operating Log"
- SC-176/276-102, "Reactor Coolant Dose Equivalent Iodine-131"
- Units 1 & 2 Licensee Event Reports

Occupational Radiation Safety Performance Indicator (1 Sample)

The inspector sampled PPL submittals for the Occupational Exposure Control Effectiveness PI, for the period of January through September 2003. The PPL records reviewed included those used by PPL to identify occurrences of locked high radiation areas, very high radiation areas, and unplanned personnel exposures. Additional records reviewed included electronic dosimeter alarm computer logs, exposure deviation reports, and radiation protection department condition report records addressing individual exposures. This inspection activity represented 1 sample. The inspector also interviewed PPL personnel that were accountable for collecting and evaluating the PI data.

Public Radiation Safety Performance Indicator (1 Sample)

The inspector reviewed a listing of relevant effluent release reports for the past four (4) calendar quarters, for issues related to the public radiation safety performance indicator, which measures radiological effluent release occurrences per site that exceed 1.5 mrem/quarter whole body or 5.0 mrem/quarter organ dose for liquid effluents; 5mrads/qtr gamma air dose, 10 mrad/quarter beta air dose, and 7.5 mrads/quarter for organ dose for gaseous effluents. This inspection activity represented 1 sample.

The inspector reviewed the following documents to ensure that PPL met all requirements of the performance indicator from the fourth quarter 2002 through the third quarter 2003:

- monthly projected dose assessment results due to radioactive liquid and gaseous effluent releases
- quarterly projected dose assessment results due to radioactive liquid and gaseous effluent releases
- SC-099-001, revision 0, "Dose Assessment"

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

1. Routine PI&R Review

a. Inspection Scope

During this inspection period, the inspectors performed a screening review of each item that PPL entered into its corrective action program, to assess whether there were any unidentified repetitive equipment failures or human performance issues that might warrant additional follow-up.

The inspectors reviewed 3 Condition Reports (489772, 506164, and 508790) that were initiated from August through September 2003 and were associated with the occupational radiation safety cornerstone. The condition reports were evaluated against the criteria contained in Technical Specification 5.7 and 10CFR20. The purpose of the review was to evaluate PPL's effectiveness at properly identifying, characterizing, investigating and resolving problems in implementing PPL's radiation protection program. Condition report 506164 identified a Performance Indicator occurrence associated with a failed high radiation area door locking mechanism. The root cause analysis identified less than adequate corrective actions were taken following a similar occurrence in 1996. Corrective actions to replace high radiation area door locks with "hotel" style locks were not completed and most likely would have prevented this occurrence.

The inspector reviewed PPL's corrective action program through the review of condition reports associated with the maintenance activities (listed in attachment 1). A detailed review of problems in Doble testing of the engineered safeguards auxiliary transformer was performed to assess the effectiveness of problem resolution activities. The inspector verified that problems and concerns in the maintenance area were identified, documented, evaluated, and entered into the corrective action system.

b. Findings

No findings of significance were identified

2. Annual Sample Review - Failure to Perform Required ASME Testing on MSIVs (71152 - 1 sample)

a. Inspection Scope

During the April 2003 Unit 2 refueling outage, the "D" inboard main steam isolation valve (MSIV) failed post modification testing, due to a measured/calculated packing friction of 5200 pounds-force which exceeded the design limit of 5000 pounds-force. The inspectors reviewed PPL's evaluation and associated corrective actions for the MSIV excessive packing friction. The inspectors also reviewed the design basis for the MSIV friction limit. The documents reviewed are listed in Attachment 1of the report.

b. Findings

Introduction

The inspectors identified a non-cited violation of very low safety significance (Green) of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Actions," because PPL did not promptly identify a condition adverse to quality. From July to December 2003, multiple evaluations by PPL did not identify that an American Society of Mechanical Engineers (ASME) fail-safe closure test was required to be performed on main steam isolation valves. The required test had not been performed since 1994.

Description

The inspectors determined that both the inboard and outboard MSIVs were valves with fail-safe actuators and had a safety function to fail closed on loss of power (pneumatic or electric) based on the MSIV's safety design bases and system description. FSAR section 5.4.5.1, "Safety Design Bases - Main Steam Line Isolation System," stated that the MSIVs used "local stored energy (compressed air and springs)" to close. FSAR section 6.2.4.2, "Containment Isolation System Design," stated that the MSIVs were spring-loaded, pneumatic piston operated valves "designed to fail closed on loss of pneumatic pressure or loss of power."

The inspectors identified that, since 1994, PPL had not performed fail-safe closure testing of the MSIVs. Technical Specification (TS) 5.5.6, "Inservice Testing Program," required the implementation of an inservice test program for ASME Code components. The ASME Code required that valves with fail-safe actuators be tested by observing the operation of the valve actuators upon loss of power. At Susquehanna, this applied to both the inboard and outboard MSIVs and, at a minimum, required testing of the inboard MSIVs in a "springs only" configuration.

Instead of performing a fail-safe closure test, PPL performed a packing friction test after MSIV maintenance, during each refueling outage. The packing friction test consisted of a partial valve stroke and determination of the valve's friction during the partial stroke. PPL had incorrectly concluded that this post maintenance friction test, in conjunction with a PPL engineering analysis, satisfied the intent of a "springs only" closure test.

PPL's combination of analysis and packing friction testing did not satisfy the requirements of the ASME code, in part, because the actual test did not demonstrate the valve's ability to fully close upon loss of valve actuating power. PPL entered the missed inservice surveillance test for the MSIVs into their corrective action program as condition report 515179.

Between July and December 2003, PPL performed multiple evaluations, but incorrectly concluded that the ASME fail-safe closure test requirement did not apply to the MSIVs. The inspectors concluded that PPL's initial assessments were narrowly focused and limited in scope.

- PPL's initial assessment concluded that the MSIVs were not valves with fail-safe actuators, and were not subject to ASME fail-safe testing, July 2003
- AR-WGA 488945 approved PPL Site Engineering position paper "MSIV Testing," which concluded OM-10 fail-safe testing did not apply to the MSIVs, because the FSAR stated that "MSIV closure is springs and air," not springs and/or air, dated October 10, 2003
- In response to NRC observations, Engineering Work Request (EWR) 511687 approved PPL Design Engineering position paper "50 second MSIV 'Springs-Only' Jet Impingement Closure Time," which stated that inboard MSIV closure by springs-only was required by design analysis, dated September 26, 2003
- After the NRC discussed EWR 511687 with PPL Site Engineering on October 1, AR-WGA 488945 was revised, and concluded that ASME fail-safe testing was required for inboard MSIVs, but not for outboard MSIVs, dated November 3
- CR 515742 performed a comprehensive Level-2 Apparent Cause Evaluation for "MSIV Design and Testing Issues," including ASME fail-safe testing of the MSIVs, and reiterated the previous response AR-WGA 488945, that ASME fail-safe testing was not required for outboard MSIVs, dated November 11, 2003
- CR 515179 performed a Level-2 Apparent Cause Evaluation for "ASME Code MSIV Fail Safe Testing," which did not come to a different conclusion than AR-WGA 488945, dated November 21, 2003
- PPL senior management reviewed design and site engineering position papers and previous assessments, and determined that the ASME OM-10 fail-safe testing was required for both the inboard and outboard MSIVs, CR 536480 dated December 30, 2003

Analysis

This finding is a performance deficiency because PPL's evaluations did not promptly identify a condition adverse to quality, associated with performance of an ASME fail-safe closure surveillance test of the MSIVs, as required by 10 CFR50 Appendix B Criterion XVI. This finding affects the barrier integrity cornerstone because it is associated with the integrity of the reactor containment. This finding is more than minor because, similar to example 1.c in the NRC Inspection Manual Chapter 0612, Appendix E, "Example of Minor Issues," a required surveillance test was not performed.

This finding is of very low safety significance (Green) using the NRC IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." A Phase-1 significance determination evaluation screened this finding as Green because the finding did not represent an actual open pathway in the physical integrity of the reactor containment.

A contributing cause of this finding is related to the Problem Identification and Resolution (PI&R) cross-cutting area because PPL had multiple opportunities but did not promptly identify a condition adverse to quality regarding TS required ASME testing for the MSIVs.

Enforcement

10 CFR 50, Appendix B, Criterion XVI, "Corrective Actions," required, in part, that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. Contrary to the above, from July to December 2003, multiple evaluations by PPL (AR-WGA 488945, EWR 511687, CR 515742, CR 515179, and CR 536480) did not identify a condition adverse to quality, in that TS required ASME fail-safe closure testing on the MSIVs had not been performed since 1994.

Because this violation is of very low safety significance and PPL entered this finding into their corrective action program (CRs 515179, 515742, and 536480), this violation is being treated as a non-cited violation (NCV), consistent with Section VI.A of the NRC Enforcement Policy. **(NCV 05000387,388/2003005-02)**

Observations

Inadequate MSIV Operational Readiness Testing

PPL had previously concluded that a post maintenance friction test, in conjunction with a PPL engineering analysis, satisfied the intent of a springs only. CR 515179 Level-2 Apparent Cause Evaluation for "ASME Code MSIV Fail Safe Testing," stated, in part, that the inboard MSIVs were qualified to close without pneumatic assist by calculation, which was confirmed by test MT-083-009, "MSIV Packing Friction Determination," each refuel outage. The packing friction test consisted of a partial valve stroke, and did not measure the valve's friction during the last 25% of the valve closure stroke.

The inspectors determined that industry operating experience identified the last 10% of valve closure as a critical characteristic to monitor because the valve's main poppet seated at approximately 10% from fully closed (weight of main poppet assists actuator in closing). Therefore, the inspectors concluded that PPL's friction test did not provide reasonable assurance of MSIV operational readiness, because the test was performed after valve maintenance, and did not demonstrate that an MSIV could fully close on loss of power.

3. Annual Sample Review - SDV Vent & Drain Valve Leakage Testing Deficiency (71152 1 sample)

a. Scope

Inspectors reviewed PPL's evaluations and actions for an apparent deficiency associated with Technical Specification (TS) 3.1.8, "Scram Discharge Volume Vent and Drain Valves." The inspectors also reviewed the material history, and maintenance and surveillance activities associated with the Scram Discharge Volume (SDV) pressure boundary, as well as the basis for any PPL established administrative limits for those activities.

b. Findings

Introduction

The inspectors identified a non-cited violation of very low safety significance (Green) of Technical Specification 5.5.6, "Inservice Testing Program." Since initial plant startup, PPL did not perform valve seat leakage testing on the SDV vent and drain valves, and did not have an adequate justification that any leakage through these valves would be inconsequential.

Description

The inspectors identified that PPL did not perform any type of check for potential valve seat leakage through the SDV vent or drain valves. Seat leakage from the SDV valves was a pathway for reactor coolant to outside of the secondary containment. The valves are hard piped into the 2700 gallon reactor building (RB) sump, and the sump is automatically pumped to the rad waste building. The inspectors determined that valve seat leakage might not be detected until the leak rate became more than minor, because potential leakage would go into the large RB sump and be automatically pumped to radwaste outside of secondary containment. Since the SDV could be pressurized to reactor pressure for an extended period of time during potential transients and accidents, the inspectors concluded that leakage through these valves could result in an unmonitored pathway to outside of the secondary containment. PPL did not include any leakage through this pathway in its off-site radiological dose analysis calculation

Valve seat leakage testing is required for primary containment isolation valves (PCIVs), by 10 CFR 50 Appendix-J, and for ASME Category A valves, by TS 5.5.6 for ASME Inservice Test (IST) program. ASME Code defined Category A as valves where seat leakage was limited to a specific maximum amount, and Category B as valves for which seat leakage was inconsequential. PPL had previously determined that since there were no valve specific regulatory leakage requirements (e.g., the valves were not PCIV per TS 3.6.1.3, or reactor coolant pressure boundary per TS 3.4.5), then the valves must be Category B, by default. PPL did not have any test data, evaluations, or radiological consequence analysis to justify that the SDV valve leakage is inconsequential. Therefore, the inspectors concluded that PPL did not have adequate justification to classify the valves as Category B valves.

The inspectors identified that TS 5.5.2, "Primary Coolant Sources Outside Containment," requires program controls to limit leakage from those portions of systems outside containment that could contain highly radioactive fluids during an accident. The systems included the Scram Discharge System. The program requires preventative maintenance, periodic inspection, and integrated leak testing for each system. PPL had previously assumed zero seat leakage for the SDV valves, and had concluded that the ASME IST inspection of the SDV piping satisfied the TS 5.5.2 program requirements. The inspectors concluded that reactor coolant leakage through the SDV vent or drain valves, into the reactor building sump, and outside the secondary containment, was required to be considered for use in PPL's off-site radiological analysis.

Analysis

This finding is a performance deficiency because PPL did not perform an ASME valve seat leakage test of the SDV vent and drain valves, as required by Technical Specifications. This finding affects the barrier integrity cornerstone because it is associated with the integrity of the secondary containment. This finding is more than minor because it is similar to example 1.c in the NRC Inspection Manual Chapter (IMC) 0612, Appendix E, "Example of Minor Issues," in that a required surveillance test was not performed.

This finding is of very low safety significance (Green) using the NRC IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." A Phase-1 significance determination evaluation screened this finding as Green because the finding did not represent an actual open pathway in the physical integrity of the reactor containment.

A contributing cause of this finding was related to the Problem Identification and Resolution (PI&R) cross-cutting area because PPL's corrective actions for a similar finding were narrowly focused and limited in scope. In 2001, the NRC (NRC IR 50-387,388/2001-05) identified that PPL had classified emergency service water valves that interfaced with the non-safety related service water system as ASME Category B valves, did not perform any seat leakage checks, assumed that any valve leakage was inconsequential, but did not have any basis for their assumptions.

Enforcement

Technical Specification 5.5.6, "Inservice Testing Program," required the implementation of an inservice test program for ASME Code Class 1, 2, and 3 components. 10 CFR Part 50.55a(f), "inservice testing requirements," required, in part, that licensees implement the requirements of ASME Operations and Maintenance (OM) Standards Part 10, "Inservice Testing of Valves in Light-Water Reactor Power Plants." The ASME OMa-1988 Addenda to the OM-1987 Edition of OM, section 1.4 required that valves be categorized and tested according to each valve's distinguishing characteristics. Category A valves were designated as valves with seat leakage limited to a specific maximum amount, and Category B as valves for which seat leakage was inconsequential. OM 10, section 4.2.2.3 stated, in part, that Category A valves, which perform a function other than containment isolation, shall be seat leakage tested at least every 2 years.

Contrary to the above, since initial plant startup, PPL had not performed seat leakage testing on the SDV vent and drain valves, and did not have any test data, evaluations, or radiological consequence analysis to justify that the potential leakage through the valves would be inconsequential during an accident. Because this violation is of very low safety significance and PPL entered this finding into their corrective action program (CRs 526987 and 530955), this violation is being treated as a non-cited violation (NCV), consistent with Section VI.A of the NRC Enforcement Policy. **(NCV 05000387,388/2003005-03)**

4. Annual Sample Review - Scram Discharge Volume Valve Technical Specification Actions (this finding was part of the same Annual Sample Review as 4OA2.3 above)

a. Scope

The inspectors reviewed PPL's actions to evaluate the validity of a significant revision to an engineering design analysis that was used in support of an amendment to Technical Specification (TS) 3.1.8 Scram Discharge Volume (SDV) vent and drain valves. The TS amendment was approved by the NRC in 1998 to establish new SDV vent and drain valve requirements. The inspector reviewed (AR 435428 dated November 2002) which directed the examination of TS 3.1.8 requirements in light of the new Probabilistic risk analysis assumptions and results. The inspector reviewed the risk evaluations performed as part of AR 435428, to understand how risk was determined and utilized as a factor in the prioritization of corrective actions. The inspector assessed whether interim corrective actions, such as compensatory measures, were considered or taken to mitigate the effects of potential TS deficiency.

Inspectors utilized industry standards and PPL procedures to evaluate the significance of the issue and to independently evaluate the need and effectiveness of interim or compensatory actions. These standards included NRC Administrative Letter 98-10, "Dis-positioning of Technical Specifications that are Insufficient to Assure Plant Safety," and Generic Letter 91-18, revision 1, which describe the NRC position and expectations for the discovery of an improper or inadequate TS Action. Inspectors also reviewed

NDAP-QA-0702, "Action Request and Condition Report Process," to evaluate the appropriateness of actions taken to identify, prioritize and resolve the issue.

b. Findings

Introduction

The inspectors identified a non-cited violation of very low safety significance (Green) of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Actions,." PPL did not promptly identify a condition adverse to quality and did not enter it into its corrective action program as a condition report. Specifically, following changes made to the PRA, PPL did not identify the need to perform an evaluation utilizing the current PRA to verify that a 1998 TS 3.1.8 change is still valid.

Description

In 1998, NRC approved PPL's request to change TS 3.1.8 which added a required action to open inoperable SDV vent or drain valves. This TS revision was justified by a PPL PRA analysis which, at the time, concluded that an Anticipated Transient Without Scram (ATWS) event contributed to more than 75% of the total core damage frequency (CDF). Subsequently, PPL revised the PRA analysis several times and the risk contribution associated with the ATWS event was re-evaluated to be to less than 4% of total CDF. In November 2002, inspectors questioned whether the TS 3.1.8 required actions continued to be supported by current PRA analysis. As of November 2003, PPL did not identify the issue to perform an evaluation utilizing the current PRA to verify that the TS change made based on the 1998 TS amendment request is still valid and did not enter the issue into its corrective action program as a condition report.

Analysis

This finding is a performance deficiency because PPL did not promptly identify a condition adverse to quality and enter it into its corrective action program as a condition report, as required by NDAP-QA-0702. This finding affects the barrier integrity cornerstone because it is associated with the integrity of the reactor containment. This finding is more than minor because it negatively impacts the configuration control and barrier performance attributes of the Barrier Integrity cornerstone to provide reasonable assurance that physical design barriers provide protection against a radiological release.

This finding is of very low safety significance (Green) using the NRC IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." A Phase-1 significance determination evaluation screened this finding as Green because the finding did not represent an actual open pathway in the physical integrity of the reactor containment.

A contributing cause of this finding was related to the Problem Identification and Resolution (PI&R) cross-cutting area, in that PPL had prior opportunities to identify and correct this issue.

Enforcement

10 CFR 50, Appendix B, Criterion XVI, "Corrective Actions," required, in part, that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. NDAP-QA-0702 requires that conditions adverse to quality be entered into PPL's corrective action program as a condition report. Contrary to the above, PPL did not promptly identify a condition adverse to quality and did not enter it into its corrective action program as a condition report. Specifically, following changes made to the PRA, PPL did not identify the need to perform an evaluation utilizing the current PRA to verify that a 1998 change to TS 3.1.8 actions was still valid.

Because this violation is of very low safety significance and PPL entered this finding into their corrective action program (CRs 526980, 528114, and 539183), it is being treated as a non-cited violation (NCV) consistent with section VI.A of the NRC Enforcement Policy. **(NCV 05000387,388/2003005-04)**

Observations

No Procedure to Perform Off-normal Operation

The inspectors identified that TS 3.1.8 Bases described the specific ability to manually close the SDV vent and drain valves as a capability available during a plant event or transient. However, PPL does not have a procedure for manual closure of the Unit 1 SDV vent and drain valves. Inspectors observed that Unit 1 did not have local hand wheels on the SDV vent and drain valves to allow local operation during an event or transient. Inspectors determined that a procedure was necessary because the action was complex (i.e., might require lifting wires, disconnecting air lines and a coordinated response by more than one individual), and because the action to close the valves needed to be taken as a response to a plant event or transient.

As a result of this performance deficiency, operators cannot provide a reliable manual backup for the automatic closure function to the SDV vent and drain valves, as specified in the TS Bases. PPL had described one potential success path for closing the valves which involved a set of coordinated actions to hold test switches in the main control room while concurrently performing local actions to isolate and vent the valve's air supply. Inspectors determined that the required actions for manual valve closure was not contained in any procedure.

This observation is related to the Problem Identification and Resolution (PI&R) cross-cutting area because there were previous opportunities for this issue to be identified, including a prior NRC finding that involved the Unit 2 SDV vent and drain valves and recent PPL risk evaluations for TS 3.1.8 Actions. In each case, corrective actions for the related issues were narrowly focused, limited in scope, and did not involve a detailed look at the reliability to perform risk significant functions.

TS 5.4.1, "Administrative Controls - Procedures," required, in part, procedures be established and implemented as required by Reg Guide (RG) 1.33, revision 2. RG 1.33

Appendix A (4.g) required procedures for Control Rod Drive and (5) required procedures for abnormal and off normal conditions. TS 3.1.8 Bases described an operator action during an off normal condition. PPL did not have a procedure to perform the described action. This violation did produce a small increase in risk since not providing the capability of manual closure of the SDV vent and drain valves during an event or transient did reduce the reliability of the barrier integrity function. However, this issue was determined to be a minor violation because the procedural deficiency has had no actual impact on plant safety. PPL entered this issue into their corrective action program as condition report 528458.

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

1. (Closed) LER 05000387/2003004-00 Loss of Control Structure Chiller Safety Function due to Spurious Chiller Trip

On June 11, 2003, at 12:17 p.m., the “A” control structure chiller tripped while the “B” control structure chiller was inoperable for planned maintenance. With both the “A” and “B” control structure chillers inoperable, the “A” and “B” trains of the control room emergency outside air supply system and the “A” and “B” trains of the control room floor cooling system were inoperable. PPL commenced a controlled shutdown of Unit 1 and Unit 2 as required by Technical Specification 3.0,3. At 1:20 p.m., PPL restored the “B” train of the control room emergency outside air supply system and “B” train of the control room floor cooling system to an operable condition and suspended the controlled shutdown. PPL determined that the most likely cause of the “A” control structure chiller trip was from a loose connection found within a circuit for high bearing or high motor temperature. The inspectors performed an in-office review of this LER and identified no findings of significance. PPL documented the event in CR 4791566. This LER is closed.

4OA4 Cross Cutting Aspects of Findings

Cross-References to Human Performance Findings Documented Elsewhere

Section 1R23 describes a finding where plant operators did not adequately implement alarm response procedures to evaluate and correct indicated low differential pressure for the refuel floor secondary containment.

4OA5 Other

1. (Update) URI 05000387,388/2003004-03 Maintenance Instructions Not Implemented to Torque a "D" EDG Governor Bolt

NRC Inspection Report 50-387,388/2003-004, Unresolved Item (URI) 50-387,388/2003-004-03, identified that PPL had not implemented written work instructions to tighten a linkage bolt on the "D" emergency diesel generator (EDG). The finding was unresolved pending completion of a risk significance determination. Subsequent to this inspection period, on January 25, 2004, the "A" EDG governor mounting bolts were found loose during a surveillance test. The "A" EDG governor was installed in August 2003, using a similar maintenance activity to the one that installed the "D" EDG governor in July 2000. Governor bolts that were required to be tightened to a torque value during both installation activities subsequently were found loose. The performance deficiency for the "D" EDG (maintenance instructions not implemented to torque EDG governor bolts) may be broader in scope that originally identified. As a result, the URI will remain open and additional inspection will be performed to further understand the performance deficiency, extent of condition, any potential common mode failures, and PPL actions. The safety significance of the performance deficiency will be determined following the completion of the inspection.

4OA6 Meetings, Including Exit

On January 9, 2004, the resident inspectors presented the inspection results to R. Anderson, Vice President - Nuclear Operations, and other members of your staff, who acknowledged the findings.

On January 22, 2004, the resident inspectors presented the safety significance determination results for "D" EDG Finding (URI 50-387,388/2003-04-03) to R. Anderson, Vice President - Nuclear Operations, and other members of your staff, who acknowledged the findings.

The inspectors confirmed that proprietary information was not provided or examined during the inspection.

4OA7 Licensee-identified Violations

The following violation of very low safety significance (Green) was identified by PPL and is a violation of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as a Non-Cited Violation.

- Technical Specification 5.7 required high radiation areas greater than1 Rem/hour to be locked. Contrary to this, on September 3, 2003, the Unit 1 door-504 on 749 foot-elevation, with access to high radiation areas greater than Rem/hour, was found unlocked. PPL entered this violation into their corrective action program as condition report 506164. This violation is of very low safety significance because it did not constitute an ALARA finding, did not result in a personnel over-exposure, did not create a substantial potential for an over-exposure, and did not compromise PPL's ability to assess dose to workers.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINT OF CONTACT

1R11 "Biennial Licensed Operator Requalification Program"

B. Boesch, Supervisor Nuclear Training
M. Jacopetti, Simulator Instructor
C. Dodge, Simulator Supervisor
B. Stitt, Unit Supervisor Requalification
F. Tarselli, Simulator Instructor
C. Hess, Simulator Instructor

2OS1 Access Control to Radiologically Significant Areas

R. Smith, Radiation Protection Manager
R. Kessler, Senior Health Physicist

4OA1 Public Radiation Safety Performance Indicator

F. Hickey, Plant Chemist

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None

Opened and Closed

05000387,388/2003005-01	NCV	PPL did not adequately implement alarm response procedures for a refuel floor secondary containment low differential pressure condition. (section 1R23)
05000387,388/2003005-02	NCV	PPL had multiple opportunities, but did not identify a condition adverse to quality associated with ASME fail-safe closure testing of main steam isolation valves. (section 4OA2.2)
05000387,388/2003005-03	NCV	PPL did not perform leakage testing on the scram discharge volume vent and drain valves, and did not have any test data, evaluations, or radiological consequences analysis to justify their assumption that the potential leakage through the valves would be inconsequential during an accident. (section 4OA2.3)

05000387,388/2003005-04	NCV	PPL did not identify the need to perform an evaluation using the current probabilistic risk analysis and did not enter the issue into the corrective action program. (section 4OA2.4)

Closed

05000387/2003004-00	LER	Loss of Control Structure Chiller Safety Function due to Spurious Chiller Trip. (section 4OA3.1)

Discussed

05000387,388/2003004-03	URI	Maintenance Instructions Not Implemented to Torque a "D" EDG Governor Bolt (section 4OA5.1)

LIST OF DOCUMENTS REVIEWED

(Not Referenced in the Report)

1R04 Equipment Alignments

- EC-054-058, revision 0, "ESW Pump Maximum Flow Rate for Continuous and Short Term Operation"
- EC-054-0511, revision 0 and 1, "Determine if One ESW pump is Adequate for D/G cooling"
- E-011, Sheets 1 thru 4, ESW PI&D
- TP-054-076, "ESW Loop A and B Flow Balance Procedure"
- ON-104-001, revision 13, "Unit 1 Response to Loss of All Offsite Power"
- ON-111-001, revision 12, "Loss of Service Water"
- FSAR Section 9.2.5, and Table 9.2-3, "Definition of ESW Flows"
- ESW System Health Report and Pump Performance Trending Data
- Condition reports 98-2716, AR-524407

1R11 Biennial Licensed Operator Requalification Program

Documents

- NTP-QA-31.2, revision 8, “Licensed Operator Requalification Program Implementation"
- STCP-QA-328, revision 4, ”Operator Requalification Written Exams"
- STCP-QA-323, ”Simulator Performance Evaluation”
- NTP-QA-14.5, “Student Remediation”
- NTP-QA-71.1, revision 2, "Simulator Certification"
- NSEI-AD-506, revision 8, "Computer System Problem Reporting"

Simulator deficiency report (CSPR) priority schemes

- All currently open CSPRs (modifications = 28; priority "2" urgent = 2; priority 3 moderate = 3; priority 4 minor = 46; priority 5 enhancements; priority 6 others = 90)
- All CSPRs closed in last two years (modifications = 84, priority 1critical = 33, priority "2" urgent = 84, priority 3 moderate = 86, priority 4 minor = 112, priority 5 enhancements and priority 6 others = 25)

Normal operations tests

- 5101, "Plant Startup - Cold to Hot Standby (including operations at hot standby)
- 5102, "Nuclear Startup From Hot Standby to Rated Power (including turbine startup and generator synchronization)
- 5106, "Plant Shutdown From Rated Power to Hot Standby (including load changes) Followed by a Plant Cooldown To Cold Shutdown Conditions"

Transient tests

- 5303, "Simultaneous Closure of All Main Steam Isolation Valves"
- 5305, "Single Recirc Pump Trip"
- 5306, "Main Turbine Trip (maximum power level which does not result in an immediate reactor scram)"
- 5308, "Maximum Size Reactor Coolant System Rupture Combined With a Loss of Offsite Power"

Steady state performance tests

- 5501, "Steady State Performance - 36% Power"
- 5502, "Steady State Performance - 80% Power"
- 5503, "Steady State Performance - 100% Power and One Hour Stability"

Real time tests

- 5601, "Simulator Computer Real Time Test"

Malfunction tests

- 5223, "Loss of Service Water"
- 5240, "Generator Trip (Negative Phase Sequence Trip - Generator Load Reject)"

Surveillance tests performed on the simulator

- SO-100-011, "Reactor Vessel Temperature and Pressure Recording"
- SO-131-001, "RWM Op Demo Startup Following System Failure"
- SO-100-007 Attachment A, "Daily Surveillance Log"
- SO-156-005, "RSIS Rod Withdrawal Block After Initial Rod Withdrawal"
- SO-156-007, "Control Rod Coupling / Full In Indication Checks"
- SO-150-005, "24 Month HPCI Flow Verification"
- SO-150-002, "Quarterly RCIC Flow Verification"
- SO-152-002, "Quarterly HPCI Flow Verification"

1R12 Biennial Maintenance Effectiveness Periodic Evaluation

- PLI-92351, "Maintenance Rule (a)(3) Assessment 4th Quarter 2000 through 4th Quarter 2002"
- Condition Reports 1998-1538, 192005, 92721, 264998, CRA 272381, CRA 208583, 397287, 388074, 75367, 56816, 43541, 347800
- System Health Reports for System 206, "480 Motor Control Centers," Systems 107and 207, "120V Lighting and Misc. Distribution," Systems 159 and 259, "Containment and Suppression," Systems 079, 170, and 279, "Process and Area Monitoring," System 030, "Control Structure HVAC," and System 280, "Reactor Non-Nuclear Instrumentation"
- NDAP-QA-0413, revision 6, "Maintenance Rule Program,"
- Miscellaneous Documents and Computer Database for PM and corrective maintenance plans and frequencies

1R23 Temporary Plant Modification

- FSAR 9.4.2.1.1, "Design Basis - Reactor Building HVAC"
- Technical Specification and Bases 3.6.4.1, "Secondary Containment"
- NDAP-QA-0312, "Control of LCOs, TROs, and Safety Function Determination Program"
- OP-AD-001, section 6.2, "Technical Specifications and Technical Requirements"
- ON-234-002, "Low Reactor Building Differential Pressure"
- Units 1 and 2 Control Room Logs for November 19, 2003
- LA-2276-001 C08, 2C276 Local Panel Alarm Response -Zone-3 Hi-Lo Diff Press
- AR-206-001 C16, Control Room Alarm Response - 2C276 HVAC Trouble Alarm
- PPL Memo from A. Piemontese, System Engineering, to T. Markowski, Operations Assistant Manager, "Guidance on RB HVAC Zone 3," dated December 14, 1989
- Engineering Work Request 365717
- Condition Reports 357291, 446191, 527005, 528151, 528759, and 528762

4OA2 Identification and Resolution of Problems

4OA2.2 <u>Failure to Perform Required ASME Testing on MSIVs</u>

- FSAR Sections 3.6.1, 5.4.5, 6.2.4, 9.3.1.5, and 15.6.5
- Technical Specifications 5.5.6
- NDAP-QA-0423, "Station Pump and Valve Testing Program"
- EC-083-0516, "MSIV Jet Impingement Closure Analysis"
- EC-VALV-1128, "MSIV AOV Program Design Bases"
- NL 91-030, Safety Evaluation for an FSAR revision to the description of the MSIV closure capability
- NUREG 1482, "Guidelines for Inservice Testing," section 4.2.4, "MSIVs"
- General Electric Service Information Letters (GE SIL):
  - 477, "MSIV Closure," December 13, 1988
  - 482, "MSIV Closure Testing Requirements," February 22, 1989
  - SC02-07, "Inboard MSIV Closure for Large Break LOCAs," June 6, 2002
- NRC Information Notices:

  - 85-84, "Inadequate Inservice Testing of MSIVs"
  - 88-51, "Failures of MSIVs"
  - 94-08, "Potential for Surveillance Testing to Fail to Detect Inoperable an MSIV"
- Work Order 371368 and MT-083-009, "MSIV Packing Friction Determination"
- Condition Reports 367013, 450386, 464052, 488848, 488945, 515179, 515742, and 536480

4OA5 Other

4OA5.1 URI 50-387,388/2003-04-03

- Technical Specification 3.8.1, "AC Sources-Operating"
- Maintenance Rule Bases Document for emergency diesel generators
- MT-GM-015, "Torquing Guidelines"
- Condition Reports 460227, 498084, 498436, and 504149
- Work orders 265805, 460312, 460811, and 460834

LIST OF ACRONYMS

ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
CDF	Core Change Frequency
CFR	Code of Federal Regulations
CR	Condition Report
CRD	Control Rod Drive
D/P	Differential Pressure
EAL	Emergency Action Level
EDG	Emergency Diesel Generator
EP	Emergency Preparedness
ESW	Emergency Service Water
EWR	Engineering Work Request
FSAR	[SSES] Final Safety Analysis Report
HPCI	High Pressure Coolant Injection
HVAC	Heating, Ventilation and Air-Conditioning
ICM	Interim Compensatory Measures
IMC	[NRC] Inspection Manual Chapter
JPM	Job Performance Measure
LER	Licensee Event Report
MSIV	Main Steam Isolation Valve
NCV	Non-cited Violation
NPO	Nuclear Plant Operator
NRC	Nuclear Regulatory Commission
PCIV	Primary Containment Isolation Valves
PCO	Plant Control Operator
PI	[NRC] Performance Indicator
PI&R	Problem Identification and Resolution
PPL	PPL Susquehanna, LLC
QA	Quality Assurance
RB	Reactor Building
RCIC	Reactor Core Isolation Cooling
RG	[NRC] Regulatory Guide
RHR	Residual Heat Removal
RO	Reactor Operator
RSPS	Risk Significant Planning Standards
SDP	Significant Determination Process
SDV	Scram Discharge Volume
SRO	Senior Reactor Operator
SSC	Structure, System, or Component
SSES	Susquehanna Steam Electric Station
TS	Technical Specification
WO	Work Order