August 17, 2001

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

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION - NRC INSPECTION REPORT 50-387/01-08, 50-388/01-08

Dear Mr. Byram:

On August 11, 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 August 16, 2001, with Mr. R. Ceravolo, General Manager SSES - Maintenance, 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 NRC identified one violation that was evaluated under the significance determination process, and was determined to be of very low safety significance (Green). However, because of the very low safety significance and because this issue was entered into your corrective action program, the NRC is treating this issue as a Non-cited Violation, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny this non-cited violation, you should provide a response within 30 days of the date of this letter, with the basis for your denial, 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.

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

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

Sincerely,

/RA/

Mohamed Shanbaky, Chief
Projects Branch 4
Division of Reactor Projects

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

Enclosure: Inspection Report 50-387/01-08, 50-388/01-08

Attachment 1 - Supplemental Information

cc w/encl: B. L. Shriver, Vice President - Nuclear Site Operations
G. T. Jones, Vice President - Nuclear Engineering and Support
R. Anderson, General Manager - SSES Operations
R. L. Ceravolo, General Manager - SSES Maintenance
G. A. Williams, General Manager - Nuclear Assurance
G. D. Miller, Manager - Nuclear Plant Services
R. R. Sgarro, Supervisor, Nuclear Licensing - SSES
M. M. Golden, Manager - Nuclear Security
P. Nederostek, Nuclear Services Manager, General Electric
A. M. Male, Manager, Quality Assurance
H. D. Woodeshick, Special Assistant to the President
G. DallaPalu, PP&L Nuclear Records
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Commonwealth of Pennsylvania
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REGION I

Docket Nos.:	05000387, 05000388
License Nos.:	NPF-14, NPF-22
Report No.:	50-387/01-08, 50-388/01-08
Licensee:	PPL Susquehanna, LLC
Facility:	Susquehanna Steam Electric Station
Location:	Post Office Box 35 Berwick, PA 18603
Dates:	July 1, 2001 to August 11, 2001
Inspectors:	S. Hansell, Senior Resident Inspector J. Richmond, Resident Inspector J. Noggle, Senior Health Physicist
Approved by:	M. Shanbaky, Chief , Projects Branch 4 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000387/2001-008, 05000388/2001-008, on 07/01-08/11/2001; PPL Susquehanna, LLC; Susquehanna Steam Electric Station; Units 1&2. Emergency Preparedness Drill Evaluation.

The inspection was conducted by resident inspectors and a regional senior health physics inspector. The inspection identified one Green finding that was considered a non-cited violation. 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 are indicated by “No Color” or by the severity level of the applicable violation. The NRC’s program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process web site at http://www.nrc.gov/NRR/OVERSIGHT/index.html.

A. Inspector Identified Findings

Cornerstone: Emergency Preparedness

- **Green**. The inspectors identified a non-cited violation for failure to adequately describe, in approved procedures, the communication steps needed to alert or activate emergency personnel under each class of emergency. (10 CFR 50.47(b)(5), “Emergency Plan Procedures,” and 10 CFR 50 Appendix E, Section IV C., "Activation of Emergency Organization")

  This violation was of very low safety significance because some inadequacies contained in the control room communicator emergency procedure may have contributed to a delayed notification of emergency response personnel. The delayed notification contributed to emergency response facility activation times in excess of the emergency plan requirements. (Section 1EP6)

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Report Details

Summary of Plant Status

Susquehanna Steam Electric Station (SSES) Unit 1was operated at or near full power for the report period, with exceptions for control rod pattern adjustments and main turbine control valve testing.

Unit 2 began the period at full power. On July 12, 2001, reactor power was reduced to approximately 89% to perform a power uprate test program after the installation of a new reactor feedwater flow element. The unit was returned to full power on July 13, and operated at or near full power for the remainder of the report period, with exceptions for control rod pattern adjustments and main turbine control valve testing.

On August 7, 2001, PPL engineering determined that the reactor heat balance input for the reactor vessel exit steam calculations, contained a non-conservative value. The value for the process computer calculation resulted in a core thermal power output that was approximately six megawatts (approximately 0.2 % of reactor thermal power limit) greater than the maximum license limit for Susquehanna Units 1 and 2. With the use of the appropriate steam moisture fraction factor in the calculations, both Units decreased their reactor thermal power by approximately 7 megawatts each in order to operate within the license thermal power limit.

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

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

During the period of August 8 to 11, 2001, the inspectors reviewed operator actions for high ambient temperatures. The inspectors walked down numerous plant areas to assess potential hot weather vulnerabilities. The inspectors utilized PPL procedure ON-000-005, "Hot Weather," to evaluate operator actions and plant conditions.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignments (71111.04)

.1 Partial System Walkdowns

a. Inspection Scope

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

• Unit Common diesel and motor driven fire pumps, water supply, and discharge to fire main header.
• Unit 1 “A” and “B” control rod drive (CRD) systems and high pressure coolant injection (HPCI) system during the standby liquid control (SLC) testing. At the time Unit 1 was in the third highest of 4 risk levels due to the SLC testing. The CRD and HPCI systems were posted with adequate barriers in place to prevent inadvertent work on the systems.
• “E” emergency diesel generator (EDG) with the “C” EDG out of service for a six week maintenance outage. The “E” EDG was substituted for the “C” EDG and performed the function of one of four Technical Specification required EDGs.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

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

- Unit 1 & 2 upper and lower relay rooms
- Unit Common refuel floor, during inspection of irradiated fuel
- Unit 1 & 2 13Kv switchgear rooms (plant startup buses)
- Unit 1 & 2 125VDC & 250VDC battery rooms
- Unit 1 & 2 condensate pump rooms

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12)

a. Inspection Scope

The inspectors evaluated the follow-up actions for selected system, structure, or component (SSC) issues and reviewed the performance of these SSCs, to assess the effectiveness of PPL's maintenance activities. The inspectors reviewed PPL's problem identification and resolution actions for these issues to evaluate whether PPL had appropriately monitored, evaluated, and dispositioned the issues in accordance with PPL procedures and the requirements of 10 CFR 50.65(a)(1) and (a)(2), "Requirements for Monitoring the Effectiveness of Maintenance." In addition, the inspectors reviewed selected SSC classification, performance criteria and goals, and PPL's corrective actions that were taken or planned, to verify that the actions were reasonable and appropriate. The following issues and documents were reviewed:

Equipment Issues

- “C” emergency diesel generator (EDG) lube oil cooler found with debris partially blocking 35 tubes on the emergency service water (ESW) cooling side of the heat exchanger. Reviewed the heat exchanger functional failure requirements related to the partially blocked tubes (CR 345243)
- Unit 2 bypass indication system (BIS) for HPCI (CR 346987)
- Unit 2 emergency service water (ESW) supply removed from the reactor building closed cooling water (RBCCW) and turbine building closed cooling water (TBCCW) heat exchangers, to support refuel outage maintenance, and not restored for 3 months following the outage (CR 342049)

Procedures and Documents

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

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessment and Emergent Work (71111.13)

a. Inspection Scope

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

The inspectors reviewed scheduled and emergent work activities with licensed operators and work coordination personnel to verify whether risk management action threshold levels were correctly identified. The inspectors assessed those activities to evaluate whether appropriate implementation of risk management actions were performed in accordance with the following PPL procedures:

- NDAP-QA-1902, "Maintenance Rule Risk Assessment and Management Program"

- NDAP-QA-0340, "Protected Equipment Program"
- PSP-22, "Susquehanna Sentinel Program"
- SSES Team Manual
- OP-245-001, "Reactor Feed Pump and RFP Lube Oil System"

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

- The Unit 2 reactor feedwater level control electronic circuit card replacement with the three feedwater pump controls in manual operation (WO 347913)
- Unit 1 1D131 maintenance bypass switch operation, during 1D130 uninterruptible power supply maintenance (WO 318952)
- Unit 1 “B” control rod drive (CRD) pump seal line leak repair. The plant risk was the second highest of four risk levels during the time the “B” CRD pump was out of service for the leak repair (WO 347434)

b. Findings

No findings of significance were identified.

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

.1 Inspection of Irradiated Fuel Assembly

a. Inspection Scope

During the period of July 9 to 20, 2001, PPL utilized contractor personnel to perform detailed inspections of two irradiated fuel assemblies. Pre-selected individual fuel rods were removed from fuel assemblies, inspected, then re-installed in the fuel assemblies.

The inspectors assessed personnel performance during this non-routine planned evolution to evaluate whether the plant personnel performance was appropriate and in accordance with procedures and training. The inspectors reviewed OP-ORF-007, "Underwater Fuel Inspection and Repair," safety evaluation NL-01-014, "Irradiated Fuel Inspection with Framatome Equipment," work order 211347, and condition report CR 344077, "Fuel rod could not be removed from bundle." The inspectors observed portions of the inspection activities, including disassembly and reassembly of a fuel assembly, and interviewed personnel performing the inspection activities.

b. Findings

No findings of significance were identified.

.2 Unit 1 Unexpected Single Control Rod Scram during Surveillance Testing

a. Inspection Scope

On July 15, 2001, Unit 1 control rod 22-51 scrammed from position 12 to full-in, during performance of SO-158-001, "Weekly Manual Scram control switch Functional Check." Reactor power decreased from 100% to 98.4% as a result of the individual rod scram. PPL entered this issue into their corrective action program as CR 344576.

The inspectors reviewed operating logs and interviewed plant operators for this non-routine unplanned event to independently determine what occurred and evaluate the initiating cause. The inspectors assessed personnel performance to determine whether the operator response was appropriate and in accordance with procedures and training.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

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

- "C" emergency diesel generator lube oil cooler found with debris partially blocking 35 tubes on the emergency service water cooling side of the heat exchanger (CR 345243)
- Unit 2 HPCI system steam supply line vibration (CRs 343168 and 343791)

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17)

.1 Unit 2 Power Up-rate

a. Inspection Scope

The inspectors reviewed the Unit 2 Power Up-rate modification. The modification installed a Caldon Leading Edge Flow Meter (LEFM) which allowed a more accurate measurement of reactor feedwater flow by using an ultrasonic flow measurement system. With the LEFM system in service, the maximum operating reactor power level was increased from 3441 Megawatts-thermal to 3489 Megawatts-thermal.

The inspectors reviewed the post-modification (power ascension) test procedures and test acceptance criteria to assess whether the testing would verify that system performance characteristics were within the predicted design assumptions, and demonstrate that operation of the plant at the higher power level was acceptable. The inspectors observed the shift test briefing and reviewed PPL's risk management for the power ascension and testing activities to verify whether appropriate risk management actions had been identified and adequately performed. In addition, the inspectors observed portions of the power ascension and testing activities to verify whether the activities were properly performed, in accordance with approved procedures. The inspectors reviewed the test data to evaluate whether the test acceptance criteria were satisfied, and whether any unintended system interactions had been identified. The following documents were included in the reviewed:

Procedures and Documents
- NE-2000-001, revision 1, "PPL Licensing Topical Report"
- PPL Susquehanna Unit 2 License Amendment No. 169
- FSAR Section 14.3, "Power Uprate Test Program"
- NRC Safety Evaluation Report for License Amendment 169
- PPL Letter to NRC, PLA-5300 Attachment 2, "Power Ascension Testing"
- TP-299-008, "LEFM Power Uprate Master Test Procedure"
- TP-200-015, "LEFM Steady State Data Collection"
- TP-293-025, "LEFM - Turbine Pressure Regulator Testing"
- Engineering Specification M-1515, revision 2, "Power Uprate Startup Testing"
- General Electric Startup Test Specification 22A6950, revision 0
- OP-AD-001, revision 21, section 6.3, "Compliance with 100% Thermal Power License Condition"
- ON-245-005, revision 1, "LEFM Failure"
- ON-200-004, revision 7, "Reactor Power Greater than 100%"
- Condition reports 347774, 348606, and 348811

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 test data. The inspectors assessed the adequacy of the test methodology, based on the scope of maintenance work performed, and evaluated whether the acceptance criteria demonstrated that the tested components satisfied the design and licensing bases requirements. The specific issues reviewed included:

- The Unit 2 reactor feedwater level control electronic circuit card replacement with the three feedwater pump controls in manual operation. The post maintenance test verification included the feedwater pump controllers being switched between manual and automatic operation.
- “D” emergency diesel generator following a cylinder head replacement to stop a jacket water leak, OP-024-001 Section 3.23.
- Unit 1 "B" control rod drive pump seal line leak repair (WO 347343)

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors reviewed selected surveillance tests, test data results, and the applicable Technical Specification requirements. In addition, the inspectors observed the performance of portions of surveillance tests to verify whether the systems and components were capable of performing their design basis functions. The observed or reviewed surveillance tests included:

- Unit 1 functional test of average power range monitor (APRM) channel “E”, SI-178-209E
- Unit 1 calibration of local power range monitors (LPRMs), RE-1TP-012
- Unit 1 "A" core spray loop quarterly flow verification (SO-151-A02)
- Diesel and motor driven fire pump monthly run (SO-013-001)
- Unit 1 & Common filter exhaust train deluge system functional test (SI-113-261)

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

On July 31, 2001, the inspectors observed the operation support center (OSC) and technical support center (TSC) performance during an emergency preparedness exercise. The simulated emergency plan (E-Plan) event included the activation of the emergency operations facility (EOF) and the technical support center (TSC) when an Alert classification was declared by the control room emergency director. The control room simulator was used for the exercise with the exception of the emergency notification telephone pushbutton located in the main control room.

The inspectors assessed the operators' adherence to emergency plan implementation procedures, the response to simulated degraded plant conditions, and site accountability. In addition, the inspectors observed PPL's critique of the exercise on August 6, 2001.

Subsequently, on August 7, 2001, the inspectors reviewed the nuclear emergency response organization (NERO) communication systems that are used to activate the emergency responder pagers. The inspectors reviewed the following documents and procedures:

- EP-PS-100, "Emergency Director Control Room;"
- EP-PS-126, "Control Room Communicator;"
- Security Emergency Drill Log;
- Security Telephone/TNS Check Off List;
- Operations Instruction "Hot Box" 00-147 and 01-104;
- Condition Report Nos. 349077, 349081, 349068, and 349073;
- Operation Support Center Drill Log.

b. Findings

The inspectors identified a Green non-cited violation because PPL did not have procedures in place to adequately describe the communication steps needed to alert or activate emergency personnel under each emergency class.

Emergency procedure EP-PS-126, "Control Room Communicator Flow Chart," is used to alert station personnel for simulated and actual events. The inspectors determined that the procedure used to notify PPL's Nuclear Emergency Response Organization (NERO) of a radiological event was inadequate which contributed to the delay of the activation of emergency facilities during the emergency drill conducted on July 31, 2001. The delayed notification contributed to the untimely activation of the emergency response facilities as delineated in the E-Plan. The EOF was staffed in 105 minutes versus the E-Plan requirement of 90 minutes, and the TSC was staffed in 64 minutes versus the E-Plan required 60 minutes. PPL activates their NERO by using an emergency notification telephone pushbutton located in the main control room and a telephone notification system (TNS) operated by security personnel. The NERO push button was installed in the control room as part of corrective actions associated with activating emergency facilities in a timely manner (NRC Inspection Report Nos. 05000387/2000-010 and 05000388/2000-010).

Subsequent to the drill, the inspectors reviewed the control room communicator procedure guidance contained in EP-PS-126. The procedure contained the following direction for the communicator: "Activate NERO pagers using Telephone HOT BUTTON." The control room phone has 20 pushbuttons, a green "NERO" pushbutton is dedicated for the pager call out. The procedure did not describe the sequence of pushing the NERO hot button or the length of time the phone should remain off of the receiver to ensure the pager activation was completed successfully. When the inspectors discussed the time the phone needed to remain off the receiver to ensure the NERO pagers were activated, the plant control operators (PCOs) stated times of approximately 10 seconds, 30 seconds, and not sure of the required time.

An operation department information notice, Hot Box No. 00-147, was issued when the control room hotbutton was installed in 2000. The Hot Box stated that the PCO's should remove the telephone handset, push the GREEN activation button, pager activation takes approximately 10 seconds, reset the green button, and return the telephone handset to the cradle. On August 10, 2001, PPL performed a test of the control room phone to determine the actual pager activation time. The time recorded from pressing the NERO push button until pager activation was approximately 32 seconds. PPL performed a temporary procedure change to EP-PS-126 to provide written direction for the control room phone activation of the NERO pagers. The information contained Hot Box No. 00-147 was also corrected.

In addition to the control room phone, the Telephone Notification System (TNS) and the Alpha Mate system are also available to activate the NERO pagers. The TNS is normally activated during an emergency when requested by the control room communicator. The TNS pager activation and NERO response time is approximately 15 to 20 minutes longer than the control room NERO pushbutton activation time. The Alpha mate system is a backup to the TNS system. It's pager activation and NERO response time is approximately 30 minutes longer than the control room NERO pushbutton activation. Both systems are operated and tested by the security department using a security check off list. The systems are not operated and tested with controlled procedures as required by emergency response program requirements.

In accordance with 10 CFR 50.47(b)(5) PPL shall have procedures for notification of emergency personnel by all organizations and 10 CFR Part 50, Appendix E.IV.C. requires, in part, that the communication steps needed to alert or activate emergency personnel under each class of emergency shall be described. Following the guidance of Manual Chapter 0610*, Appendix B, this NRC identified issue was determined to be more than minor because if left uncorrected, there was a potential impact on public safety in that the inadequate procedure could have delayed the notification of emergency response personnel and mitigation of potential plant events. NRC Inspection Manual Chapter 0609, Appendix B, was used to assess the risk significance of this finding. Although 10 CFR 50.47(b)(5) is considered a risk significant planning standard, the inspector determined this issue was of very low safety significance (Green) because PPL would have been able to activate the NERO pagers using the

Telenotification System even though the activation times could be approximately 15 minutes slower than the control room NERO push button. While this finding was a failure to "meet" a regulatory requirement (Sheet 1, Appendix B, MC 0609) it was not considered a failure to "meet" a planning standard. This violation 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) and was documented in PPL's corrective action program as condition reports 349077 and 349081. **(NCV 05000387,388/2001-008-01)**

2. RADIATION SAFETY
Cornerstone: Occupational Radiation Safety

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope

The Susquehanna Unit 2 Spring 2001 refueling outage exposure performance was evaluated through a review of a draft of the Unit 2 10th Refueling and Inspection Outage ALARA Report. The five highest exposure jobs were reviewed with respect to estimates. These included: reactor pressure vessel nozzle in-service inspection (ISI), insulation work in the drywell, scaffolding work in the drywell, temporary shielding, and main steam isolation valve work in the drywell. Additional outage post-job reviews that were evaluated included: drywell snubber work, vessel weld ISI, drywell residual heat removal valve work, and drywell main steam relief valve work.

Other high exposure jobs were reviewed that resulted in greater than 5 person-rem. In particular, radiation work permit (RWP) 2001-2317, "Recirculation one-inch piping modifications," was reviewed in detail based on the 84% dose overrun and associated work proficiency issues. This review included interviews with involved project engineers, ISI, and maintenance staff.

Revision to outage job exposure estimates were reviewed by reference to initial outage RWP exposure estimates, applicable ALARA in-progress reviews, and final outage RWP results. This was performed to verify the exposure estimates, that they were not inappropriately adjusted, and that the job exposure estimates were a valid standard to measure outage job exposure performance.

ALARA program assessments (by both internal and external licensee organizations) were reviewed for 1999 and 2000. These included corporate exposure performance analyses and annual Nuclear Assurance Services ALARA performance assessment reports. This review was with respect to the annual radiation protection program review requirement as specified in 10 CFR 20.1101(c).

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA3 Event Follow-up (71153)

.1 (Closed) LER 05000388/2001-004-00: Control Rod Drive (CRD) System Seismic Island Check Valves Did Not Meet Local Leak Rate Test Acceptance Criteria. A previous failure of the CRD seismic island check valves was discussed in NRC Inspection Report 50-387,388/2000-004. No new issues were identified during this review; no violations of NRC requirements were identified. This LER is closed.

4OA6 Meetings

.1 Exit Meeting Summary

On August 16, 2001, the resident inspectors presented the inspection results to Mr. R. Ceravolo, General Manager SSES - Maintenance , and other members of your staff who acknowledged the findings.

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

ATTACHMENT 1

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

NONE

Opened and Closed

05000387, 388/2001008-01	NCV	Inadequate procedures in place to describe the communication steps needed to alert or activate emergency personnel. (Section 1EP6)

Closed

05000388/2001-004-00	LER	Control Rod Drive (CRD) System Seismic Island Check Valves Did Not Meet Local Leak Rate Test Acceptance Criteria

LIST OF ACRONYMS USED

ALARA	As Low As is Reasonably Achievable
CFR	Code of Federal Regulations
CR	Condition Report
EDG	Emergency Diesel Generator
ESW	Emergency Service Water
FSAR	[SSES] Final Safety Analysis Report
HPCI	High Pressure Coolant Injection
ISI	In-Service Inspection
Kv	Killivolts
LOOP	Loss of Off-site Power
NCV	Non-cited Violation
NERO	Nuclear Emergency Response Organization
NRC	Nuclear Regulatory Commission
PPL	PPL Susquehanna, LLC
RBCCW	Reactor Building Closed Cooling Water
RHR	Residual Heat Removal
SDP	Significance Determination Process
SLC	Standby Liquid Control
SSC	Structure, System, or Component
SSES	Susquehanna Steam Electric Station
TBCCW	Turbine Building Closed Cooling Water
TSN	Telephone Notification System
TS	Technical Specification