April 26, 2001

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

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION - NRC INSPECTION REPORT 05000387/2001-003, 05000388/2001-003

Dear Mr. Byram:

On March 31, 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 April 10, 2001, with Mr. B. Shriver and other members of your staff.

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

No findings of significance were identified.

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 610-337-5185.

Sincerely,

/RA/

Donald Florek, Acting Chief
Projects Branch 4
Division of Reactor Projects

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

Enclosure: Inspection Report 05000387/2001-003, 05000388/2001-003

Attachments: (1) Supplemental Information

cc w/encl:
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G. T. Jones, Vice President - Nuclear Engineering and Support
R. Anderson, General Manager - SSES Operations
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REGION I

Docket Nos.:	05000387, 05000388
License Nos.:	NPF-14, NPF-22
Report No.:	05000387/2001-003 05000388/2001-003
Licensee:	PPL Susquehanna, LLC
Facility:	Susquehanna Steam Electric Station
Location:	Post Office Box 35 Berwick, PA 18603
Dates:	February 11, 2001 to March 31, 2001
Inspectors:	S. Hansell, Senior Resident Inspector J. Richmond, Resident Inspector A. Blamey, Resident Inspector J. Noggle, Senior Health Physicist
Approved by:	Donald Florek, Acting Chief Projects Branch 4 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000387/2001-003, 5000388/2001-003, on 02/11-03/31/2001; PPL Susquehanna, LLC; Susquehanna Steam Electric Station; Units 1&2. Resident inspector and radiation specialist report.

The inspection was conducted by resident inspectors and a regional senior health physicist. No findings of significance were identified. The significance of most findings is 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 website at http://www.nrc.gov/NRR/OVERSIGHT/index.html.

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Report Details

Summary of Plant Status

Susquehanna Steam Electric Station (SSES) Unit 1 operated at or near 100% power through the inspection period.

SSES Unit 2 operated at or near 100% power through the inspection period until March 10. On March 10, Unit 2 was shut down to begin a refueling and maintenance outage.

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

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

The inspectors reviewed PPL's preparations for the March 5th and 6th snow storm. The inspectors performed a plant walkdown of the emergency core cooling systems and on-site electrical distribution systems. The inspectors reviewed and evaluated plant conditions using NDAP-00-0024, revision 2, "Winter Operation Preparations and Severe Weather Operation."

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

- Unit 1 Reactor Coolant Isolation Cooling System
- Unit 1 Residual Heat Removal (RHR) system fuel pool cooling assist mode while the Unit 2 RHR pumps out of service during the Unit 2 refuel outage

b. Findings

No findings of significance were identified.

.2 Complete System Walkdown

a. Inspection Scope

The inspectors performed a complete system walkdown on the Unit 1 standby liquid control (SLC) system to verify equipment alignment. In addition, the inspectors reviewed the Final Safety Analysis Report (FSAR), SLC system design drawings, and issues tracked by the system health report (condition reports, work orders, and maintenance rule issues). These reviews were conducted to identify discrepancies that would impact system operability. The following documents were included in the review:

- CL-153-0011, Unit 1 SLC Electrical Check-off List
- CL-153-0012, Unit 1 SLC Mechanical Check-off List
- CL-153-0013, Unit 1 SLC Containment Check-off List
- ES-150-002, Boron Injection with Reactor Core Isolation Cooling
- M-148, Standby Liquid Control System P&ID
- DBD-042, Design Basis Document for SLC System
- FSAR section 9.3.5, SLC System
- FSAR section 15.8, Anticipated Transients Without a Scram
- Technical Specification and Basis sections 3.1.7, SLC System

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 then performed walkdowns of 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 Reactor Core Isolation Cooling system
- Unit 2 High Pressure Coolant Injection pump and turbine room
- Refuel floor during Unit 2 refuel and maintenance outage
- Unit 2 main condenser and main turbine control/stop valve areas

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

The inspectors reviewed the Unit 1 emergency core cooling system pump room flood protection equipment and pump room level alarm circuits. The following procedures were included in the review:

- NDAP-QA-0302, Section 6.14, “Internal Flooding and Floor Drain Covering”
- ON-169-002, “Flooding in the Reactor Building”
- High Pressure Coolant Injection, Residual Heat Removal, and Core Spray alarm response procedures for “Pump Room Flooded”
- Emergency Operating Procedure EO-100-104, Secondary Containment Control”
- FSAR Section 3.4, ”Water Level Flood Design”

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12)

a. Inspection Scope

The inspectors reviewed the follow-up actions for two system, structure, or component (SSC) issues and the performance of these SSCs, to assess the effectiveness of PPL's maintenance activities. The inspectors verified that problem identification and resolution of these issues had been appropriately monitored, evaluated, and dispositioned 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 corrective actions to verify that the actions were reasonable and appropriate. The following issues and documents were reviewed:

Equipment Issues
- Unit 1 Residual Heat Removal suppression pool cooling motor operated valve, HV-151-F024B
- Standby liquid control system pump relief valve repetitive test and storage tank level indicator failures

Procedures and Documents
- 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 in the Maintenance Rule"
- EC-053-1001, "Determination of Design Basis for SLC Accumulators"
- Condition Reports 98-0611, 98-1488, 98-2663, 245526, 250859, 251120, and 317544
- MT-GM-005, "Safety/Relief Valve Setting"

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 against the requirements of 10 CFR 50.65(a)(4) and the recommendations of NUMARC 93-01 Section 11, "Assessment of Risk Resulting from Performance of Maintenance Activities." The inspectors verified that risk assessments were performed when required and appropriate risk management actions were identified.

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

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

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

- Unit 2 “C” Residual Heat Removal pump discharge check valve, repeat work
- Unit 2 Reactor Protection System motor-generator set, generator replacement
- Unit 2 loss of alternate decay heat removal, due to a blown fuse for the No. 2 Supplemental Decay Heat Removal pump

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

On March 14, 2001, the Unit 2 "A" Reactor Protection System Motor Generator set circuit breaker tripped which caused the Residual Heat Removal system shutdown cooling suction valve, HV-251-F009, to close. This resulted in loss of shutdown cooling. The operators restored shutdown cooling in approximately 37 minutes. Reactor coolant temperature increased by less than 2 degrees during this period.

The inspectors evaluated PPL's response to the unanticipated loss of shutdown cooling by reviewing plant computer and recorder data, operator logs, approved procedures and training. The following procedures were included in this review:

- ON-249-001, "Loss of Shutdown Cooling"
- TP-235-011, "Refuel Outage Decay Heat Removal and Tie-in of SDHR Temporary Cooling Equipment"

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. The inspectors verified that the operability determinations were performed in accordance with NDAP-QA-0703, “Operability Assessments.” The inspectors used the Technical Specifications, Technical Requirements Manual, Final Safety Analysis Report, and associated Design Basis Documents as references during these reviews. The issues reviewed included:

CR 303627	Unit 2 "C" RHR discharge check valve leakage
CR 317544	Unit 2 "B" RHR loop keep-fill pressure dropped below 50 psig during system fill and vent
CR 316281	Unit 2 "A" RHRSW radiation monitor induced voltage from the starter motor
CR 312664	Unit 2 reactor recirculation system sample valve, HV-243-F020 failure to close
CR 320966	Unit 1 Suppression Chamber oxygen concentration increase

b. Findings

No findings of significance were identified.

1R16 Operator Work-Arounds (71111.16)

a. Inspection Scope

The inspectors reviewed the significant control room deficiencies and all items on the operator work-around list to determine if the functional capability of a system, or a human reliability response during an event, would be affected. Operations procedure OP-AD-002, Attachment I, "Operations Standards for Error and Event Prevention," was also reviewed. This review focused on the operators' ability to implement abnormal and emergency operating procedures during postulated plant transients with the existing equipment deficiencies. The review included an evaluation of the cumulative effects of the identified operator work-arounds. The most significant operator work-arounds included:

- RHR service water radiation monitor local operation
- Main steam safety relief valves that leak and result in frequent operation of the RHR suppression pool cooling system and elevated suppression pool water temps
- Modification to the condensate transfer water supply to the emergency core cooling system to prevent system water hammer

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17)

a. Inspection Scope

The inspectors reviewed selected portions of modification DCP/ECO 96-9109, revision 1 and 2, "4KV ESS Switchgear - Unit 1 Seismic Qualification Upgrades." This modification installed seismic supports to maintain the operability of the ESS buses when 4 KV breakers were placed in the test (disconnected) position. The safety evaluation, EC-SQRT-1070, "Anchorage Qualification for 4KV Switchgear Hold Down Bracket," and WCAP-14867, "Westinghouse Equipment Qualification Report" were included in the review. Work authorization C80429, for installation of the bracket in the 1A20102 Bus cubical, was also reviewed to verify that the installation and post-modification testing activities were properly performed. The inspectors observed in-progress modification work on the Unit 2 "A" RHR system.

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 the acceptance criteria to demonstrate that the tested components satisfied the design and licensing bases and Technical Specification requirements. The specific issues reviewed included:

WO 238529	“B” Emergency Diesel Generator load reject test (TP-024-146)
CR 313346	Unit 2 RHR shutdown cooling low pressure permissive switch replacement
SO-024-001	“B” Emergency Diesel Generator run after planned maintenance

b. Findings

No findings of significance were identified.

1R20 Unit 2 Refueling and Maintenance Outage Activities (71111.20)

.1 Refuel Outage Plan Review

a. Inspection Scope

The inspectors reviewed the risk assessment for the scheduled outage plan to verify that PPL had appropriately considered overall plant risk, industry experience, and previous SSES outage problems. The review included PPL's computerized ORAM-Sentinel risk assessment program, NDAP-QA-0612, "Outage Implementation and Assessment," and the SSES Team Manual.

b. Findings

No findings of significance were identified.

.2 Reactor Plant Shutdown Activities

a. Inspection Scope

The inspectors observed selected portions of operator activities during the plant shutdown to reactor hot shutdown, reactor plant cooldown and RHR system transfer and operation in the shutdown cooling mode. The inspectors verified that activities were performed in accordance with approved procedures and training. The inspectors reviewed computer data and operator logs to spot check that the cooldown rate did not exceed the Technical Specification limit of 100 degrees per hour. The following documents were included in the review:

- GO-200-004, "Plant Shutdown to Minimum Power"
- OP-293-002 section 3.10, "Pre/Post Outage Main Turbine Overspeed Test"

b. Findings

No findings of significance were identified.

.3 Control of Outage Activities

a. Inspection Scope

Configuration Management & Risk Management: The inspectors observed portions of equipment and system maintenance and reviewed equipment test procedures. The inspectors verified that system configuration, work control, and maintenance tests were performed in accordance with NRC requirements and PPL procedures. The inspectors also reviewed emergent work and unexpected plant conditions to evaluate outage risk control. The reviewed activities included:

- Restoration of RHR shutdown cooling, following unexpected loss of RPS bus
- PPL response to unexpected reactor scram signal, while shutdown
- PPL response to unexpected partial loss of decay heat removal
- Recirculation loop chemical decontamination
- Control rod drive mechanism change out
- Main steam isolation valve local leak rate testing

Supplemental Decay Heat Removal System Operation: While the service water system was removed from service, a temporary supplemental (alternate) decay heat removal (SDHR) system provided river water cooling directly to the Unit 2 fuel pool cooling heat exchangers. This temporary SDHR system, in conjunction with the Unit 1 fuel pool cooling system, was an acceptable alternate decay heat removal system when both Unit 2 divisions of RHR were removed from service. The Unit 1 RHR system was also used in the fuel pool cooling assist mode. The inspectors performed a walkdown of the SDHR system and those portions of Unit 1 RHR system that would be operated in the fuel pool cooling assist mode. The inspectors observed SDHR system operation and reviewed operating logs, operating procedures, and off-normal procedures to verify that activities were performed in accordance with PPL procedures and appropriate design basis documents. The following documents were included in the review:

- TP-235-011, "Refuel Outage Decay Heat Removal and Tie-in of SDHR Temporary Cooling Equipment"
- OP-011-001, "SDHR System"
- ON-249-001, "Loss of RHR Shutdown Cooling Mode"
- NL-95-001, "Safety Evaluation for Tie-in and Operations of SDHR"
- OP-149-003, "Unit 1 RHR in Fuel Pool Cooling Assist Mode"

b. Findings

No findings of significance were identified.

.4 Plant Operations with the Potential to Drain the Reactor Vessel

a. Inspection Scope

The inspectors reviewed portions of plant operations with the potential for draining the reactor vessel or reactor cavity to verify that the necessary administrative or engineering controls were in-place to prevent an inadvertent loss of reactor coolant.

b. Findings

No findings of significance were identified.

.5 Refueling Activities

a. Inspection Scope

The inspectors observed portions of fuel handing operations, and other related activities to verify that the activities were performed in accordance with the Technical Specification requirements and PPL approved procedures. The inspectors spot checked fuel assembly movement from the refuel platform, to verify that the locations of fuel assemblies were tracked, from core off-load through core reload. The following activities and documents were observed or reviewed:

- Control rod blade replacements
- Fuel handling between spent fuel pool and reactor core
- New fuel receipt inspection and channeling
- Foreign material exclusion control around fuel pools and reactor cavity
- Refueling interlock surveillance checks on refuel platform
- Refuel floor secondary containment integrity during fuel handling operations
- OP-0RF-005, "Refueling Operations"
- OP-181-001, "Refueling Platform Operation"
- ON-081-001, "Fuel Handling Accident"
- ON-081-002, "Refueling Platform Operation Anomaly"

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 that the systems and components were capable of performing their design basis functions. The observed or reviewed surveillance tests included:

SC-176-102	Unit 1 Reactor Coolant System Dose Equivalent Iodine-131 Sample and Analysis
SO-030-001	“A” Control Room Emergency Outside Air Supply System Monthly Operability Test
SO-253-004	Unit 2 SLC System Quarterly Flow Verification Test
SO-100-006	Unit 1 Drywell Floor Drain Leakage Calculation

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

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 FSAR and Technical Specifications, and assessed the adequacy of the 10 CFR 50.59 safety evaluations. The inspectors also assessed configuration control of the temporary changes by reviewing selected drawings and procedures to verify that appropriate updates had been made. The inspectors compared the actual installations against the temporary modification documents to verify that the implemented changes were consistent with the approved documents. The inspectors reviewed selected post-installation test results to confirm that the actual impact of the temporary changes had been adequately verified by test. The following temporary modifications and documents were included in the review:

<u>Temporary Modifications</u>

TMOD 284420	Unit 2 Temporary Drywell HEPA Units for Zone II Supply Duct
TMOD 295983	Unit 2 Disable Door 102 Alarm and Interlock
TMOD 295435	Unit 2, Install I-Beam in the Truck Bay Door to Maintain Secondary Containment
TMOD 295268	Removal of the Refuel Water Transfer Pump Suction Pressure Trip for the Reactor Flood-up
IC-280-005	Unit 2 Temporary Reactor Vessel Level Indication

<u>Procedures and Documents</u>

- NDAP-QA-1218, "Temporary Modifications"
- ICC-LT-B21-2N027-2, "Transmitter Calibration with Reactor Vessel Head Removed"
- 50.59 Screening Determination for IC-280-005, dated 01-22-99
- NL-97-005, "Bypass to Run Electrical Cables Under the Unit 2 Truck Bay Door"
- TS 3.6.1.1, "Primary Containment"
- TS 3.6.1.3, "Primary Containment Isolation Valves"
- TS 3.6.4.1, "Secondary Containment"
- RACT 271265, “Removal of the Refuel Water Transfer Pump Suction Pressure Trip”
- NDAP-QA-0409, "Door, Floor Plug, and Hatch Control"
- FSAR Section 6.2.3.2, Secondary Containment Design
- PCWO 226957, "Install I-Beam Under the Unit 2 Truck Bay Door"
- PCWO 295734, "Seal Penetration on the Unit 2 Truck Bay Door I-Beam"

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY
Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

a. Inspection Scope

The inspectors reviewed the access control program (required by Technical Specifications and 10 CFR 20.1601) by examining the controls established for exposure in significant areas, including postings, barricades and locking controls for access to radiologically significant areas. In-plant areas and activities reviewed included control rod drive (CRD) replacement, drywell access to locked high radiation areas, transient dose rate controls for chemical decontamination processing equipment, and refueling floor hot particle control zones.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope

The inspectors reviewed As-Low-As-Reasonably-Achievable (ALARA) performance in accordance with 10 CFR 20.1101(b). Areas reviewed included an evaluation of ALARA planning for 4 of the 5 highest exposure outage tasks: drywell scaffolding, drywell health physics, drywell insulation, and CRD replacement. Interviews were also conducted with the applicable scaffolding and insulation foreman, drywell lead health physics technician, and CRD replacement maintenance and health physics staff. In addition, mechanical maintenance’s methodology for performing CRD scram volume discharge header hydrolyzing and plans and initial activities associated with the chemical decontamination of the recirculation piping system were reviewed.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

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

- Reactor Coolant System dose equivalent Iodine-131 specific activity
- Reactor Coolant System leak rate measured by the drywell leakage calculation
- Residual Heat Removal System Unavailability
- Safety System Functional Failures

b. Findings

Residual Heat Removal System Unavailability

The inspector could not determined the accuracy of this PI at this time because a frequently asked question (Draft FAQ 19.3) submitted by PPL to NRC, in accordance with the reactor oversight process, regarding counting unavailability of the residual heat removal system (RHR) while in suppression pool cooling has not been answered. One position is that RHR should be considered unavailable while in shutdown cooling because, if a simultaneous loss of cooling accident and loss of offsite power occurred, the RHR system response would cause portions of the drywell spray header, the Injection header, and reactor vessel head spray header to be emptied and the resulting water hammer upon automatic restart of the RHR pumps would be of sufficient magnitude to render the subsequent function of the associated RHR loop indeterminate. A second position is that since that PPL's probabilistic risk analysis determined that the simultaneous occurrence of a loss of coolant accident and loss of offsite power during suppression pool cooling operation resulted in a core damage frequency less than 10 E-6, it was beyond the plant design basis (because the risk was low) and did not need to be counted. This issue will remain open until the FAQ is resolved. This item will be tracked as a unresolved item. **(URI 05000387, 388/2001003-01)**

Safety System Functional Failures

The inspector could not determine the accuracy of this PI at this time due to apparent conflict in guidance in determining whether a failure of an instrumentation system should be counted as a safety system functional failure (SSFF). The NEI 99-02 guidance defined a SSFF as any event or condition that could have prevented fulfillment of a safety function needed to shutdown the reactor, remove residual heat, control the release of radioactive material, or to mitigate the consequences of an accident. NUREG 1022, revision 2, section 3.2.7 [50.73(a)(2)(v)], stated that a failure of a system used only to warn the operator, where no credit is taken for it in any safety analysis, and it does not directly control any safety functions is not reportable as a SSFF. This conflict resulted in uncertainty as to whether two LERs should be considered a SSFF.

The inspectors identified that LER 05000387,388/2000-001-00, "Inadequate Hydrogen-Oxygen Analyzer System Design" described a failure of post accident monitoring instrument that potentially could have prevented a safety function needed to mitigate the consequences of an accident and that PPL did not count in the NRC PI data as an SSFF event. The inspectors determined that the hydrogen-oxygen analyzer system is identified in the emergency procedures and the information from this system is directly used by the operators to start the manually initiated hydrogen recombiner system, a system necessary to mitigate the consequences of a design basis accident. In LER 05000388/2000-002-00, "Inadvertent Containment Radiation Monitor Isolation Valve Closure During Maintenance Activities", failure of the containment radiation monitor could have prevented timely initiation of emergency plan activities.

PPL Licensing reviewed this issue and concluded that since these systems provided no active function (e.g., indication only), then the systems were not required to mitigate the consequences of an accident and were not within the scope of the NEI 99-02 guidance for reporting as an SSFF event. The inspectors concluded that information from NRC Headquarters is required to resolve the conflicting guidance to determine whether the failure of these post accident monitor systems should be counted as an SSFF. The resolution of this item is pending a response from NRC Headquarters and will be tracked as a unresolved item. **(URI 05000387,388/2001003-02)**

4OA5 Other

.1 Independent Spent Fuel Storage Installation Safety Evaluations (60857)

a. Inspection Scope

The inspectors reviewed a 10 CFR 72.48 screening determination for a procedure change which allowed the use of heated compressed air to blowdown a dry shielded canister during the water draining process. The inspectors verified that PPL's conclusions were consistent with NRC requirements and PPL approved procedures and that a 72.48 safety evaluation was not required. The documents reviewed included:

- 50.59 and 72.48 Screening Determination for ME-0RF-144 PCAF 2000-4570, rev-3
- Transnuclear West Safety Review Screening Form SRS 72-1387, dated Dec 27, 1999

- Condition Report 313749, "Utilizing Hot Air during Blowdown of Dry Fuel Storage Canisters"
- ME-0RF-144, "Dry Fuel Storage - Dry Shielded Canister Draining, Vacuum Drying, and Helium Fill"

b. Findings

No findings of significance were identified.

4OA6 Meetings

.1 Exit Meeting Summary

On April 10, 2001, the resident inspectors presented the inspection results to Mr. B. Shriver and other members of your staff who acknowledged the findings.

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

.2 Regulatory Conference for the Health Physics White Finding

On March 1, 2001, the NRC and PPL met in the Region I Headquarters in King of Prussia, PA, to discuss the root cause and risk significance of potential personnel radiation exposures related to the Susquehanna fuel pool cleanup project. The meeting was open to the public.

Attachment 1 **SUPPLEMENTAL INFORMATION**

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000387,388/2001003-01	URI	Residual Heat Removal System Unavailability PI Verification (section 4OA1.2)
05000387,388/2001003-02	URI	Safety System Functional Failure PI Verification (section 4OA1.3)

Opened and Closed

NONE

Closed

NONE

LIST OF ACRONYMS USED

ALARA	As-Low-As-Reasonably-Achievable
CFR	Code of Federal Regulations
CR	Condition Report
CRD	Control Rod Drive
DBD	Design Basis Document
FAQ	Frequently Asked Question
FSAR	[SSES] Final Safety Analysis Report
HPCI	High Pressure Coolant Injection
KV	Kilovolts (1000 volts)
LER	Licensee Event Report
LOOP	Loss of Off-site Power
NRC	Nuclear Regulatory Commission
PI	[NRC] Performance Indicator
PPL	PPL Susquehanna, LLC
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
RPS	Reactor Protection System
SDHR	Supplemental Decay Heat Removal
SDP	Significance Determination Process
SLC	Standby Liquid Control
SSC	Structure, System, or Component
SSES	Susquehanna Steam Electric Station
SSFF	Safety System Functional Failure
TS	Technical Specification
URI	[NRC] Unresolved Item