U. S. NUCLEAR REGULATORY COMMISSION

REGION I

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Facility:	Susquehanna Steam Electric Station (SSES)
Location:	P.O. Box 35 Berwick, PA 18603-0035
Dates:	April 8, 1997 through May 19, 1997
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EXECUTIVE SUMMARY

Susquehanna Steam Electric Station, Units 1 & 2
NRC Inspection Report 50-387/97-03, 50-388/97-03

This integrated inspection included aspects of licensee operations, engineering, maintenance, and plant support activities. The report covers a 6-week period of resident inspection.

Operations

- Unit 2 refueling and inspection outage activities were inspected to determine proper movement and placement of fuel assemblies. Two minor events occurred during the fuel movement activities. The two events were: 1) an impact between a single blade guide and a reactor vessel flange cover and 2) the failure of a safety chain clip resulting in a portion of the clip being dropped into the fuel pool. Each of the events was responded to aggressively by the licensee. The licensee also identified and resolved generic issues with human performance related to refueling activities. Despite the two events and the refueling related issues, refueling activities were well supervised and were conducted in a safe and conservative manner.

- On May 2, 1997, with Unit 2 in cold shutdown, a SPING alarm was received. The operators responded appropriately, the Shift Supervisor made conservative decisions which ensured the safety of the unit and the public. The inspector determined that one of the plant alarm response procedures was inadequate in that it did not contain reasonable validation criteria, and it did not agree with and delayed entry into the Emergency Plan.

- Approximately seven lighted control room annunciators correspond to chronic conditions in SSES ventilation systems. The licensee currently has a corrective action plan in place to correct these conditions. Previous action plans were determined to be ineffective in that there was very little progress made from approximately 1988 until the present to correct these specific alarmed annunciators. There is no present impact on Technical Specification operability requirements, compliance with NRC requirements or compliance with selected SSES design standards.

- The licensee has directed considerable efforts toward the resolution of plant conditions that are documented in condition reports (CRs). Upper levels of PP&L management, including the Vice President - Nuclear Operations and the General Manager - Nuclear Engineering, were observed routinely addressing the resolution of individual CRs. The licensee demonstrated during this inspection report period that they were capable of addressing and resolving CRs adequately in the short term.

Maintenance

- In general, maintenance activities observed during this report period were adequately controlled and performed in accordance with station procedures. In the case of the spent fuel pool temperature monitor maintenance activities, the work was well performed and controlled.

- A Unit 2 'C' reactor feed pump turbine bearing was replaced for corrective maintenance. The replacement bearing did not meet the vendor specified clearances and the System Engineer unilaterally revised the procedural clearance limits in the field. The System Engineer did not effectively communicate to the Unit Supervisor (US) that he had not met the bearing clearance limits when he was requesting and performing a post maintenance acceptance test. The test was terminated by the US due to high temperature. The bearing was brought into tolerance and a successful post maintenance acceptance test was performed. The reactor feed pump is not safety related and its failure would result in bounded plant transients. There was no impact on the safe operation of the unit. Therefore, no violations of NRC requirements occurred.

- The monthly surveillance test for the 'C' EDG was generally performed according to approved surveillance test procedures. However, the inspector observed a failure to comply with alarm response procedures for a known equipment condition associated with an oscillating jacket water standpipe level indication. This failure to follow procedures is being treated as a non-cited violation.

- The performance of the quarterly surveillance test for the Division I core spray system was generally well-controlled. However, the inspector concluded that the practice of venting the core spray pumps immediately prior to starting them for their quarterly surveillance test was a preconditioning action. The failure to perform the core spray surveillance test under suitably controlled conditions is considered a violation of NRC requirements. In addition, the inspector observed that the methodology for performing independent verifications within the test procedure was weak and did not meet licensee expectations.

- Two maintenance activities associated with restoration from the Unit 2 refueling outage had the potential for personnel injury. Both issues were adequately resolved by the licensee. No personnel injury occurred, there was no impact on safety related equipment, and no violations of NRC requirements occurred.

- During fuel movement activities, a fuel assembly was suspended (less than one foot) above the reactor vessel fuel support piece without the ability to raise or lower it through normal means. Maintenance activities were initiated on the Unit 2 rod control system to resolve this condition. The inspector observed that the maintenance was performed using an "information only" SSES Training Department drawing that was not authorized for use. This failure to use controlled drawings was characterized as a non-cited violation.

- As a result of weaknesses identified in the under vessel maintenance activities during the Unit 1 refueling outage, and condition reports written by the licensee, the inspector observed/reviewed under vessel activities during the Unit 2 refueling outage. The licensee issued and resolved a number of condition reports and took adequate corrective actions. No violations of NRC requirements were identified.

- Maintenance on safety-related instruments performed by the Emergent Work Action Crew (EWAC) was observed to be commensurate with the scope and complexity proscribed by the administrative procedure for minor maintenance. Appropriate documentation and communication practices were noted.

- PP&L's efforts to remove foreign materials from the Unit 2 containment following the refueling outage were very good. The final containment walkdown and inspection by the Operations and Maintenance department managers was viewed as strength. Based on the areas reviewed during the inspector's containment walkdown, PP&L was effective in restoring equipment (hatches, insulation, hangars, etc.) to the condition required for plant operation.

- PP&L's efforts to reduce foreign debris in the Unit 2 suppression pool during the Spring 1997 refueling outage were through. The compensatory actions requested by the NRC in conjunction with deferral of the final resolution of Bulletin 96-03 were implemented by PP&L. The Unit 2 suppression pool cleanout results were consistent with the assumptions contained in PP&L's existing operability evaluation that addressed suction strainer clogging.

- Reactor building ventilation system (RBVS) back draft isolation dampers are safety related components within the non-safety related system. Although the RBVS is addressed by the maintenance rule program at SSES, the function of the BDIDs was not included in the licensee's scoping document. A determination of the significance of not including the BDIDs in the RBVS maintenance rule scope will be tracked as an unresolved item.

- The Nuclear Assessment Services (NAS) audit of the Test Control Program provided an adequate review of post maintenance testing as required by the Operational Quality Assurance Manual. However, the inspector considered the audit sample size (15 packages) small relative to the number of safety related work authorizations processed in a two year period (22,000 packages). The lack of an NAS audit in the minor maintenance area was considered a weakness in testing program oversight required by the Quality Assurance Manual.

Engineering

- A Unit 2 cycle 9 core reload hydraulic performance evaluation (separate from thermohydraulic performance) for the Atrium 10 fuel was reviewed and determined to be adequately bounded by analysis.

- After a safety related 4160 volt breaker failed to operate when required, the licensee identified a possible new failure mode involving a personnel protection

device referred to as a tripper lever. Although problems with tripper levers were previously identified in NRC Information Notice (IN) 96-50, this failure was different since it occurred after the breaker had been racked in and cycled. The licensee's response to the IN was very conservative and aggressive. The involvement of first and second line engineering management in this issue was laudable. The generic aspects of the issue have been forwarded to NRR for review.

- The licensee maintained the capability to utilize emergency alternate water sources identified in the SSES Emergency Plan and discussed in the FSAR. No violations of NRC requirements were identified.

- In four instances, PP&L failed to perform safety evaluations prior to making changes to the facility as required by 10 CFR 50.59. The following examples were identified: 1) blocking open doors for rooms with high energy line break protective features, 2) increasing the normal 250 Vdc system float voltage, 3) installing temporary test equipment on the emergency diesel generators, and 4) cross connecting the normal and backup fire protection systems. This was identified as a violation of NRC requirements.

Plant Support

- The licensee met the requirements of its security plan with respect to vital area door access. The licensee's surveillance activities were carefully and well performed. Some aspects of general employee training could be improved to make the operation of the door alarm system clearer to plant employees. No violations of NRC requirements were identified.

- On May 5, 1997, the licensee cross connected the normal and backup fire protection systems and the systems remained in the cross connected condition at the end of this report period. This alignment constitutes a change to the normal fire protection system that is described in the FSAR and TS 3/4 7.6. This change was not preceded by an evaluation to determine if an unreviewed safety question would result from the cross tie of the two fire protection systems and is example number (4) of the 10 CFR 50.59 violation.

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Report Details

Summary of Plant Status

Unit 1 operated throughout this inspection period at 100 percent power. The Unit 2 eighth refueling and inspection outage (8RIO) lasted 56 days and ended on May 5, 1997. On May 10, 1997, Unit 2 closed its generator output breakers ending the refueling outage and reached 100% power on May 16, 1997. Unit 2 operated at 100% power throughout the remainder of the inspection period.

I. Operations

O1 Conduct of Operations[1]

O1.1 Unit 2 Refueling Activities

a. Inspection Scope (71707)

Unit 2 8RIO refueling activities were inspected to determine proper movement and placement of fuel assemblies. In addition, several refueling related issues, and two events were reviewed for adequate licensee corrective action. The two events involved a single blade guide making contact with the reactor vessel flange cover, and retaining a clip on the refueling bridge which failed and dropped into the fuel pool near new and used fuel.

b. Observations and Findings

The inspector reviewed a number of procedures, drawings, reports and corrective action documents including:

GE Drawing A-17599-D, Refueling Platform

QS Surveillances 97-035, 97-036

OP-ORF-005, Refueling Operations

Significant Operating Occurrence Report (SOOR) 93-347

Condition Reports (CRs) 97-0809, 1182, 1293, 1222, 1271 and 1275 related to human performance

CRs 97-1175, and 1182 related to refueling bridge condition

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[1]Topical headings such as O1, M8, etc., are used in accordance with the NRC standardized reactor inspection report outline. Individual reports are not expected to address all outline topics.

CRs 97-1222 and 97-1256 related to impacting the reactor pressure vessel flange protector with a single blade guide

Single Blade Guide Movement

SSES Significant Operating Occurrence Report (SOOR) 93-347 addressed a double blade guide handle that was bent during a 1993 transfer activity. The double blade guide was bent during movement above the reactor vessel flange. During the 1993 movement of the double blade guide, an operator on the bridge noticed that there was not enough clearance to move over the reactor vessel flange. The operator attempted to stop the movement of the bridge but was unable to prevent impact between the double blade guide and the reactor vessel. The root cause of the event was determined to be that the governing procedure, RE-081-032, Refueling Operations, did not require the mast to be in the full up position prior to entering the transfer canal. The governing procedure was modified to include the requirement to have the mast in the full up position prior to movement in certain areas of the reactor vessel and/or spent fuel pool. Similar events occurred on four occasions prior to the 1993 SOOR.

Following replacement of the aforementioned bridge, the operating procedures for the bridge were modified. The procedures were modified to account for the flexibility that the new bridge offered during fuel movement (movement in more than one direction at a time).

On April 11, 1997, the licensee was conducting a transfer of a single blade guide to reactor core location 49-34, in the automatic mode of the refueling bridge. The single blade guide impacted the reactor pressure vessel flange protector during the move. The refueling bridge platform had started to slow down per design near its target position. Concurrently, the grapple started to lower the blade guide into the core. Because of the combination of these motions, the location in the core that the single blade guide was to be placed, and the light mass of the single blade guide, which allowed it to glide while being moved, its placement was not vertical during the lowering process. The operator recognized the situation (in a manner similar to the 1993 event) and stopped lowering the mast but was unable to prevent the impact. The operators entered offnormal procedure, ON-081-002, Refueling Platform Operation Anomaly, and performed inspections of the grapple and flange protector.

The inspector determined that OP-ORF-005, revision December 9, 1996, Refueling Operations, contained precautions that disagreed with the body of the procedure. The precautions stated that the operators were not to use semiautomatic or automatic mode when moving double blade guides, while the body of the procedure stated that the operators are not to use these modes when moving double blade guides or single blade guides. This inconsistency in the procedure resulted in a misunderstanding by refueling operators who thought that it was acceptable to operate the refueling bridge in automatic and/or semiautomatic while moving a single blade guide.

Technical Specification (TS) 6.8.1 requires that written procedures be established and implemented for applicable procedures recommended in Appendix 'A' of Regulatory Guide 1.33 Revision 2, February 1978. Regulatory Guide 1.33 Appendix 'A' item 1.l requires procedures for refueling operations. Contrary to the above, OP-ORF-005, Refueling Operations, was inadequate in that it did not clearly control the movement of single blade guides and as a result, a single blade guide impacted the reactor pressure vessel flange cover. This is example (a) of **VIO 387,388/97-03-01.**

Human Performance During Refueling Activities

During the Unit 2 8RIO activities, a number of events were identified by the licensee in condition reports. Among these condition reports, the inspector identified a trend related to human performance. The condition reports that relate to human performance are listed in paragraph 1.1.6 of this report. The inspector identified this issue in parallel with the SSES second line refueling supervisor. The SSES supervisor documented the human performance issue in CR 97-1293 and took very aggressive corrective actions.

Refueling Bridge Cab Entry Chain Clip

On April 9, 1997, while moving spent fuel from the rector vessel to a storage location in the spent fuel pool, an operator leaned on the safety chain for the refueling bridge cab entry. The clip broke and a portion of the clip landed in the fuel pool near a new fuel assembly (Atrium 10). The licensee stopped fuel movement activities and developed a plan to locate the missing portion of the clip. The licensee subsequently located the missing portion of the clip and determined that there was no impact on the new fuel near where it fell. The inspector determined that the actions of the licensee were responsive and very conservative. The SSES second line refueling supervisor took charge of the issue and implemented a number of very conservative actions that resulted in the retrieval of the item from the pool and overall improvement of the condition of the refueling bridge.

The inspector identified that the applicable GE Drawing A-17599-D, Refueling Platform, indicated a specific part number for the correct clip. It was determined that the clip that broke and possibly the chain on which it was connected did not correspond to the part number identified in the GE drawing. Because of the responsive corrective action by the licensee, the conservative action of the second line refueling manager, and the determination that there was no impact on the new fuel, no violation of NRC requirements were determined to occur.

c. Conclusions

Unit 2 8RIO refueling activities were inspected to determine proper movement and placement of fuel assemblies. The fuel movement activities involved several refueling related issues that were documented by the licensee with condition reports, and two events. The two events were: 1) an impact between a single blade guide and a reactor vessel flange cover and 2) the failure of a safety chain clip resulting in a portion of the clip being dropped into the fuel pool. Each event was

responded to aggressively by the licensee. The licensee also identified and resolved generic issues with human performance related to refueling activities. Despite the two events and the refueling related issues, refueling activities were well supervised and were conducted in a safe and conservative manner. Second line management supervision of refueling activities and refueling bridge physical condition were considered a strength.

O1.2 Unit 2 Turbine Building System Particulate Iodine Noble Gas (SPING) Refueling Activities

a. Inspection Scope (71707)

On May 2, 1997, with Unit 2 in cold shutdown, a Stack Monitoring System Alarm was received. The inspector reviewed/inspected this event to determine if the operators responded appropriately, plant procedures and practices were adequate, and if operator response was similar to previously inspected plant events.

b. Observations and Findings

In order to evaluate the event that occurred on May 2, 1997, it was necessary to examine three aspects of the event.

- The adequacy of the Alarm Response Procedure (AR) in addressing the condition
- The adequacy of the Shift Supervisor's response in resolving divergent procedural guidance
- The determination of whether the event represented a real release with potential impact on the public.

The Adequacy of the Alarm Response Procedure

On May 2, 1997, with Unit 2 in cold shutdown, a Stack Monitoring System Alarm was received. A Plant Control Operator (PCO) responded to the alarm, using alarm response (AR) AR-015-D4, Stack Monitoring System Alarm (OC630), Hi Hi Radiation and initially determined that the cause of the alarm was the Unit 2 turbine building Iodine above 3.59 E-8 micro Ci/cc. AR-015-D4 step 2.2 requires the operator to perform a number of actions including substep 2.2.1b - Notify chemistry to confirm alarm validity and to take appropriate actions. The PCO implemented the proper procedural steps including substep 2.2.1b. Substep 2.2.1c states that for valid alarms, evaluate data for entry condition into EO-100-105, Reactivity Release Control.

Section 5.0 of the SSES Emergency Plan states that an Unusual Event should be declared as soon as it has been indicated and verified. However, it sets the parameters of the time expected to verify the need for an Unusual Event by stating that all reasonable efforts are implemented to make this verification within fifteen

minutes of the initial indication of the event. Because the AR procedure limits the operator to a validation process which could take up to two hours before directing him to the Emergency Plan, the AR does not adequately support the implementation of the Emergency Plan.

Technical Specification (TS) 6.8.1 requires that written procedures be established and implemented for applicable procedures recommended in Appendix 'A' of Regulatory Guide 1.33 Revision 2, February 1978. Regulatory Guide 1.33 Appendix 'A' item 5 requires procedures for emergencies and item 6 requires procedures for abnormal, offnormal or alarm condition. Item 6 further states that the procedures for abnormal conditions should include immediate operator action.

Contrary to the requirement of TS 6.8.1, AR-015-D4, Stack Monitoring System Hi Hi Radiation, was inadequate in that substep 2.2.1b requires the operator to notify chemistry to confirm the validity of a SPING alarm, effectively eliminating a procedural route to the Emergency Plan, prior to the completion of the validation by chemistry. Because the validation process implemented by chemistry can take up to two hours, compliance with the AR inhibits compliance with SSES Emergency Plan which would have all reasonable efforts implemented to complete the verification within fifteen minutes of the initial indication of the event. This is example (b) of VIO 387,388/97-03-01.

This issue is similar to a violation identified in NRC inspection report 387,388/97-01, which also involved the adequacy of operator response to an AR. The events share a common thread in that there appears to be an SSES tendency to delay operator action in certain instances until a validation of a control room alarm is performed. Although this is not an incorrect approach overall, in the two specific cases cited, this tendency was a contributor to the noted weaknesses.

Following a discussion between the SSES Supervisor of Operations, the inspector determined that the licensee had implemented a number of initial corrective actions including a procedural change to the AR (PCAF 1-97-0348) which describes a different method of determining the alarm validity. The initial corrective actions were determined to be very good. The long term corrective actions are as yet incomplete and will be addressed by the licensee's response to the violation indicated in the previous paragraph of this report.

<u>The Adequacy of the Shift Supervisor Response</u>

The Shift Supervisor (SS) determined that it would take up to two hours for chemistry to validate the alarm and that the alarm data, as read, indicated that the limit defining an Unusual Event in EP-PS-100-6, EAL 15.1 release rate at 1.41 E 5 micro Ci/min was being exceeded (note: this is a converted quantity that agrees in scale with the AR scale). He further realized that section 5.0 of the SSES Emergency Plan required that all reasonable efforts be implemented to make a verification of an Unusual Event within fifteen minutes of the initial indication of the event.

The SS chose not to enter the emergency plan and did not declare an Unusual Event. Based on the inspector's discussion with the SS and his supervisor, the SS chose to declare the SPING alarm an invalid alarm based on data other than that called for in the AR. His determination was based on:

> The immediate previous operating history of the unit (greater than 30 days shutdown and completing a refueling period) did not support such a high SPING indication.
>
> Area monitors near the SPING monitors were not alarming and were in fact reading normally.
>
> The mechanical vacuum pump had been removed from service and isolated four hours prior to the alarm and, therefore, a release path was isolated.
>
> Work activities in progress did not include activities that would produce the SPING monitor alarm.

His choice was determined by the inspector to be a conservative one which was later verified to be technically correct. However, the fact that the SS was forced into such a position indicated a procedural weakness that is being addressed as a violation.

Impact of the Event on the Public

The inspector determined that there was no actual release, that, by not declaring an Unusual Event, the Shift Supervisor made a conservative decision, and that there was no impact on the public.

c. Conclusions

The operators responded appropriately, the SS made conservative decisions which ensured the safety of the unit and the public. The inspector determined that one of the plant alarm response procedures was inadequate.

O2 Operational Status of Facilities and Equipment

O2.1 Control Room Annunciators Operability

a. Inspection Scope (71707)

The inspector selected a number of normally alarmed annunciators in the control room and reviewed the impact that they had on the control room operators, the licensee response to the lighted conditions, the impact on Technical Specification

operability requirements, compliance with NRC requirements and compliance with selected SSES design standards.

b. Observations and Findings

The control room annunciators selected for this review each affect SSES ventilation systems. There are a total of seven annunciators in the lighted condition. They include:
AR-106-C16 "1C276 Delta Press Swings
AR-106-D16 "Delta Press Swings"
AR-106-E16 "RW Building HVAC Panel 0C377 System Trouble"
AR-106-F16 "TB Supply Filter Hi DP"
AR-206-C16 "Circ Space Hi/Lo DP"
AR-206-D16 "Area DP Swing"
AR-206-F16 "HVAC Turbine Building Panel 2C175 Trouble"

The inspector determined that the licensee currently has an engineering project scheduled to eliminate these lighted annunciators. The project includes initial completion dates between June and August 1997. The licensee has had similar projects that date back to approximately 1988, which had similar goals and objectives.

The inspector determined by interviews that the lighted annunciators did not distract the operators. Occasionally the corresponding alarms intermittently annunciated. In those situations, the operators were more directly affected. The inspector also determined that the licensee's current response to the lighted conditions was adequate and that it had a corrective action plan in place. Previous action plans were determined to be ineffective in that there was very little progress made to the present to correct these specific alarmed annunciators. The previous plans were eventually abandoned when milestones were routinely missed. No present impact on Technical Specification operability requirements, compliance with NRC requirements or compliance with selected SSES design standards was identified.

c. Conclusions

Approximately seven lighted control room annunciators correspond to chronic conditions in SSES ventilation systems. The licensee currently has a corrective action plan in place to correct these conditions. Relative to these specific alarms, previous action plans were determined to be ineffective over a period of years. There is no present impact on Technical Specification operability requirements, compliance with NRC requirements or compliance with selected SSES design standards.

O2.2 Review of Licensee Condition Reports

a. Inspection Scope (71707)

The inspector reviewed approximately 800 CRs written during this report period and/or associated with the Unit 2 8RIO. The CRs were reviewed for initial licensee response, impact on Technical Specification operability requirements, compliance with NRC requirements and compliance with selected SSES design standards.

b. Observations and Findings

A cursory review of approximately 800 CRs was performed by the inspector during this inspection period. Of these, approximately 100 involved level 2 or 1 conditions. About half of the CRs address equipment failures (400) and half of those (200) were resolved prior to the end of this inspection period. The CR generation rate has dropped since the last inspection period but remains high based on historical data.

The licensee has directed considerable efforts toward the resolution of CRs and the inspector observed PP&L management levels up to and including the Vice President Nuclear Operations and the General Manager Nuclear Engineering routinely addressing the resolution of individual CRs. The licensee demonstrated during this inspection report period that they were capable of addressing and resolving CRs adequately in the short term.

c. Conclusions

The licensee demonstrated during this inspection period that they were capable of addressing and resolving CRs adequately in the short term.

O8 Miscellaneous Operations Issues (92700)

O8.1 Review of Licensee Event Reports

a. Inspection Scope (90712)

The inspector reviewed Licensee Event Reports (LERs) submitted to the NRC to verify that the details of the event were clearly reported, including the accuracy of the event description, cause and corrective action. The inspector evaluated whether further information was required from the licensee, whether generic implications were involved, and whether the event warranted onsite followup.

b. Observations and Findings

The following LERs were reviewed and closed during this inspection period:

(Closed) LER 50-387/97-008-00: Instrument Response Time Testing

On March 26, 1997, with Unit 1 at 100% power and Unit 2 in refueling, PP&L determined that the requirements of TS surveillances 4.3.1, 4.3.2, and 4.3.3 for Response Time Testing were not fulfilled. The failure to satisfy TS surveillance requirements for response time testing resulted in numerous instruments and systems being declared inoperable and required operators to enter TS 3.0.3 at 8:25 p.m. Enforcement discretion was granted by the NRC allowing Unit 1 to exit TS 3.0.3 at 9:00 p.m. This event is discussed in NRC Inspection Report 50-387/97-02.

The licensee's failure to adequately perform the TS surveillances is a violation. This licensee-identified and corrected violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1 of the NRC Enforcement Policy.

(Closed) LER 50-387/97-009-00: Fire Watch Rounds Not Completed On Time

On April 2, 1997, PP&L line management identified that on two separate occasions, roving fire watch personnel did not survey areas as required by TS Action 3.7.7.a. PP&L determined that these events were caused by ineffective on-the-job fire watch training and qualification, and that there was no provision for timely feedback of problems during fire watch rounds.

Short term corrective actions implemented by PP&L included surveys of the missed areas, initiation of refresher training for fire watch personnel, and increased supervisory oversight. Long term actions described in the LER were completion of the refresher training, implementation of a formal training and qualification process, training for fire watch supervisors, and evaluation of feedback methods for fire watch patrols.

The inspector found that a number of long term corrective actions described in the LER have expected completion dates in September 1997. The inspector discussed the completed short term actions with the responsible PP&L manager and concluded that the short term actions for this LER are acceptable. However, to provide assurance that future fire watch rounds will not be missed in other areas of the plant, more comprehensive action will be needed.

This licensee-identified and corrected violation is being treated as a Non-cited Violation, consistent with Section VII.B.1 of the NRC Enforcement Policy.

c. Conclusions

The events reported by PP&L in the LERs reviewed during this period were appropriately reported, and provided an accurate description of the causes and corrective actions. The inspector determined that for the LERs discussed in brief, the corrective actions were reasonable, and that these events require no additional onsite followup.

II. Maintenance

M1 Conduct of Maintenance

M1.1 Planned Maintenance Activity Review

a. Inspection Scope (62707)

A variety of maintenance activities were reviewed on the basis of their complexity, safety (or risk) significance, or other considerations. A sample of work permits, equipment tagouts, procedures, drawings, and vendor technical manuals associated with these maintenance activities were reviewed as part of the inspection. Through observation of the maintenance activities, review of appropriate documentation and interviews with maintenance personnel, the inspector sought to verify that the activities were performed in accordance with procedures and regulatory requirements, that personnel were appropriately trained and qualified, and that appropriate radiological controls were followed.

b. Observations and Findings

The following maintenance activities were reviewed through direct observation and/or review of the completed work packages:

WA S71662	Spent Fuel Pool Temperature Instrument
WA V71171	Spent Fuel Pool Temperature Instrument
MT-GE-005	Circuit Breaker Inspection and Maintenance of 5 and 15 Kv Breakers
WA V70005	LPRM Maintenance

c. Conclusions

With respect to the selection of maintenance activities indicated in this section, the work activities were adequately controlled and observed portions were performed in accordance with station procedures. In the case of the spent fuel pool temperature monitor maintenance activity the work was well performed and controlled.

M1.2 Reactor Feed Pump Repair Activity Review

a. Inspection Scope (62707)

Maintenance activities associated with the repair of the Unit 2 'C' Reactor Feed Pump (RFP) were reviewed on the basis of their potential for catastrophic failure of the pump bearing during inprocess testing and/or normal plant operation.

b. Observations and Findings

The following data were reviewed/inspected during this inspection activity:

WA P61666	Reactor Feedwater Pump (RFP) Disassemble/Inspect
MFP-QA-2309	Design Change Package/Engineering Change Order Preparation
NDPA-QA-206	Replacement Item Evaluations (RIE)
NDAP-QA-502	Work Authorization (WA)
MT-048-001	RFP Disassembly Inspection and Reassembly
TP-245-004	Overspeed Trip Test of RFP

Two aspects of this maintenance activity indicated weaknesses in the licensee's control of balance of plant related work. These aspects were; 1) the acceptance of an out of tolerance bearing and 2) the performance of a test without clear communication to the Unit Supervisor of the out of tolerance condition and/or acceptance criteria of the performance test.

Acceptance of an Out of Clearance Bearing

WA P61666 was written to disassemble and inspect the Unit 2 'C' RFP. During the performance of the WA, the low pressure (LP) bearing was replaced. The as-found clearance of the old bearing met the MT-048-001 calculated clearance limits of 12 to 20 mils. The new replacement bearing did not meet the clearance limits when installed. The clearance of the new bearing was 6 mils. The new bearing was accepted by the System Engineer as out-of-tolerance and the procedural clearance limits were relieved unilaterally by the System Engineer.

The inspector reviewed several plant processes to determine the level of review and approval, design control, and test control that are normally provided at SSES in cases such as the one encountered by the RFP System Engineer. These processes are discussed in the paragraphs below.

Section 6.4.8 of the SSES MFP-QA-2309 allows a process referred to as a minimod for "mundane and simplistic changes" to plant equipment. Minimods permit implementation of a minor change as a maintenance activity using the normal work authorization process and post modification configuration control. Minimods are required to meet NDAP-QA-1202, section 6.2.5, and must be authorized by a modification group lead and the supervisor site modification design. The indicated review and approval assures that the change fully satisfies the design basis of the equipment.

Section 6.6.6 of NDAP-QA-0502 describes the process of WA field changes used by the licensee. The procedure states that field changes cannot result in a change in plant design or configuration.

Section 1.0 of NDAP-QA-0206 describes the process for evaluating non-identical replacement items for use at SSES, including the identification of installation requirements, and the maintenance of configuration control. Section 3.5 describes the exemptions allowed for use of a non-identical item outside of the approved process.

The inspector determined that the use of the out-of-tolerance bearing was a defacto design change because it resulted in a change to the bearing design specifications as established by the SSES procedure. The acceptance of the out-of-tolerance bearing by the System Engineer did not meet the intent of any of the SSES procedures discussed above. Following the failure of the post maintenance test, the RFP was brought into compliance with the acceptance criteria and a subsequent RFP acceptance test was performed. The subsequent acceptable performance of a post-maintenance acceptance test indicated that there was little potential for impact on the safe operation of the unit.

Performance of a Post-Maintenance Acceptance Test

Following the acceptance of the out-of-tolerance bearing, the System Engineer requested and performed (with the assistance of Operations) post-maintenance test TP-245-004. The inspector reviewed the interaction/communication between the Unit Supervisor and the System Engineer prior to the performance of this test. The inspector understands that the System Engineer did not make it clear to the Unit Supervisor that the bearings did not meet the required design tolerances and that the test was intended to run the bearings in. Both the Unit Supervisor (US) and the Plant Control Operator stated that no temperature criteria had been established by the System Engineer and that the US and PCO set an interim temperature limit based on their expectations of the post-maintenance test. The inspector concluded that the US was not provided adequate information by the System Engineer in order to make knowledgeable decisions concerning the testing of plant equipment. This is considered a weakness.

Potential Regulatory and/or Safety Impact of the Maintenance Activity

Failure of the reactor feed pump would result in bounded plant transients and the reactor feed pump is not safety related. The acceptable performance of a post maintenance acceptance test conducted after the initial test failure ensured that there was little potential for impact on the safe operation of the unit from this problem. Therefore, no violations of NRC requirements occurred. However, the inspector was concerned because safety related activities are subject to the same maintenance, engineering and communication restraints as were described in this case.

c. Conclusions

During the performance of maintenance on the Unit 2 'C' reactor feed pump, the LP bearing was replaced. The new replacement bearing did not meet the clearance limits when installed. The new bearing was accepted by the System Engineer who unilaterally revised the procedural clearance limits. The System Engineer did not effectively communicate to the Unit Supervisor that he had not met the bearing clearance limits when he requested and operators performing a post-maintenance acceptance test. The test was terminated by the US on high bearing temperature. The bearing was brought into tolerance and a successful post maintenance acceptance test was performed. The reactor feed pump is not safety related and its failure would result in bounded plant transients. Although these maintenance and communication activities were weak, there was no impact on the safe operation of the unit. Therefore, no violations of NRC requirements occurred.

M1.3 Surveillance Test Activity Sample Reviews

a. Inspection Scope (61726)

The inspectors observed portions of selected surveillance tests involving different technical disciplines for safety-significant systems.

b. Observations and Findings

Through observation and review of records, the inspectors verified that the test activities were properly released for performance, that the test instrumentation was within its current calibration cycle, and that it was being performed by qualified personnel in accordance with approved test procedures. The inspectors also verified that the tests conform to TS requirements and that applicable limiting condition for operations (LCOs) were taken. The following activities were reviewed during this period:

TP-149-060	Unit 1 Shutdown Cooling Suction Flush
SR-200-008	Shutdown Margin Demonstration
OP-261-002	Precoating Reactor Water Cleanup Filter
OP-249-002	Shutdown Cooling Flush
SO-024-001	E Emergency Diesel Generator
SO-200-006	Shift Surveillance Log
SE-170-011	18 Month Secondary Containment Inleakage Test

c. Conclusions

The routine surveillance activities observed during this inspection period were adequately performed.

M1.4 Emergency Diesel Generator Monthly Surveillance Test

a. Inspection Scope (61726)

The inspectors reviewed the conduct of the monthly surveillance test for the 'C' emergency diesel generator (EDG).

b. Observations and Findings

On April 22, 1997, the inspector observed selected portions of the monthly surveillance test for the 'C' EDG. The inspector found that an operator and technicians were actively monitoring the EDG, and the test was performed using approved procedures.

The inspector noted that a "standpipe level high" alarm was lit, which indicated that a high level existed in the jacket water cooling system standpipe. The inspector discussed the alarm with the operator, who pointed out that the standpipe level indication oscillated significantly, causing repeated alarms. The operator had verified that the average level was not increasing; an increasing level would indicate a possible jacket water heat exchanger tube leak. The operator understood the cause of the alarm, and was aware of the alarm response procedure. He also recognized the oscillating standpipe level as a known, expected minor deficiency. Yet, the inspector did not observe any deficiency tags or other documentation indicating that this was a known condition.

The inspector noted that step 2.3 of the local alarm response procedure, LA-0521-003, specified that the operator notify chemistry following EDG shutdown to sample the jacket water. However, the operator stated that he would not notify chemistry because the average standpipe level did not increase.

The inspector observed that the minor equipment condition associated with the oscillating standpipe level was an accepted condition that led the operator to believe that compliance with the alarm response procedure was not necessary. Although a minor issue, this represented an example of failure to follow alarm response procedures. This failure constitutes a violation of minor significance and is being treated as a Non-Cited Violation, consistent with Section IV of the NRC Enforcement Policy.

c. Conclusions

The monthly surveillance test for the 'C' emergency diesel generator (EDG) was generally performed according to approved surveillance test procedures. However, the inspector observed a minor failure to comply with alarm response procedures for a known equipment condition associated with an oscillating jacket water standpipe level indication. This is being treated as a Non-Cited Violation consistent with Section IV of the NRC Enforcement Policy.

M1.5 Unit 1 Core Spray System Quarterly Surveillance Test

a. Inspection Scope (61726)

The inspector observed portions of the Unit 1 core spray system quarterly surveillance test, SO-151-A02, conducted on April 24, 1997.

b. Observations and Findings

The inspector observed an operator performing the preparations, valve positioning, and other actions specified by the test procedure. The test was generally well-controlled.

During the observations, the inspector identified a weakness in the methodology for performing independent verifications of valve positions as part of the test. Specifically, the inspector found that isolation valves for the core spray pump suction pressure gages were operated multiple times during the test without independent verification of each manipulation, as required by the procedure. Although the independent verifications were done by a second operator at the conclusion of the test, this did assure that the interim valve operations were performed as specified. The operator performing the test had questioned the unit supervisor prior to the test on how the independent verifications were to be performed, but due to communications weaknesses or incomplete review of the procedure, the decision was made to perform the verifications only at the conclusion of the test. The inspector brought this issue to the attention of plant management. Operations staff determined that the independent verification steps were not performed according to expectations. Operations also determined that the interim independent verification steps were not necessary and initiated a procedure change to remove them.

Following the start of the 'C' core spray pump, the inspector observed that the discharge check valve indicator did not indicate fully open, as expected. The inspector brought this to the attention of the operator, and condition report 97-1475 was initiated. The licensee's operability determination concluded that this was an indication discrepancy only and, because system flow rates were within specification, there were no operability concerns.

During the test, the inspector observed an action that resulted in the pumps being tested in a condition that was altered from their as-found condition. The inspector noted that the surveillance procedure, SO-151-A02, required the operator to vent the core spray pumps prior to starting them. The purpose of the venting steps apparently was to remove any trapped air that could lead to air binding of the pumps.

The inspector discussed this observation with the operator, who stated that he knew of no instances in which air was actually vented from the pumps during surveillance testing. He stated that the steps provided additional assurance that no air was in the pumps. The inspector also brought this issue to the attention of plant

management, who stated that venting was considered by operations management to be a good operational practice, but was not required. Based on this information, the inspector considered that there were no pump operability issues. NRC Information Notice (IN) 97-16, issued April 4, 1997, discussed several examples of unacceptable preconditioning actions that licensees have performed before Technical Specification surveillance testing. One of the examples cited was a practice of venting residual heat removal pumps immediately prior to conducting surveillance testing. IN 97-16 further notes that equipment should be tested in the as-found condition, and any disturbance or alteration to equipment would be expected to be limited to the minimum necessary to perform the test and prevent damage to the equipment.

The inspector concluded that the practice of venting the core spray pumps immediately prior to starting them for their quarterly surveillance test was an unnecessary preconditioning action. This poor surveillance practice resulted in the pumps being tested in a condition that was different from the as-found condition and thus made questionable the validity of the surveillance test results. The failure to perform the core spray surveillance test under suitably controlled conditions is considered a violation of 10 CFR 50 Appendix B, Criterion XI, Test Control. **(VIO 50-387/97-03-02)**

c. Conclusions

The performance of the quarterly surveillance test for the Division I core spray system was generally well-controlled. The inspector observed that the methodology for performing independent verifications within the test procedure was weak and did not meet licensee expectations. The inspector concluded that the practice of venting the core spray pumps immediately prior to starting them was an unnecessary preconditioning action. The failure to perform the core spray surveillance test under suitably controlled conditions is considered a violation.

M1.6 Maintenance Activities that Resulted in Potential Industrial Safety Situations

a. Inspection Scope (62707)

Two maintenance activities associated with the restoration of Unit 2 8RIO conditions were inspected.

b. Observations and Findings

During a Unit 2 reactor building tour, the inspector identified personnel working below and to the side of a suspended drywell hatch. The hatch, which weighed in excess of 1000 pounds, was supported by a lifting rig. When the inspector questioned the attending first line supervisor, he agreed that the position of the hatch was not safe and stated that it would be repositioned. Upon returning to the work area, the inspector found that the hatch had been moved, but was still in a position to affect workers, if the rig from which it was suspended failed. The

inspector notified the SSES safety organization and the SSES safety organization worked with maintenance line management to resolve the issue adequately.

On April 24, 1997, a nylon sling, which was being used to support a tool box, separated. The sling was suspended from the Unit 1 Reactor Building Crane auxiliary hoist. One end of the box dropped approximately eight feet striking the edge of a stored Unit 2 reactor cavity shield plug. The tool box weighed approximately 4000 pounds. The inspector determined, in parallel with the licensee, that one of the root causes of this event involved weaknesses in the routine testing of nylon slings. SSES safety and maintenance departments adequately responded to the event.

Because no personnel injuries occurred and there was no impact on safety related equipment, no violations of NRC requirements occurred. However, these events constitute a weakness in the industrial safety practices at the site.

c. Conclusions

Two maintenance activities associated with the restoration of Unit 2 8RIO conditions were inspected. Each of the activities had the potential for personnel injury, although no injury occurred. Both issues were adequately resolved by the licensee. No personnel injury occurred, there was no impact on safety related equipment, and no violations of NRC requirements occurred.

M1.7 Maintenance Activities in Support of Refueling Activities

a. Inspection Scope (62707)

During fuel movement activities, a fuel assembly was suspended (less than one foot) above its lower fuel support piece without the ability to raise or lower it through normal means. The inspector observed portions of the refueling activities and the ensuing maintenance support activities.

b. Observations and Findings

On April 15, 1997, a fuel assembly was suspended (less than one foot) above the lower reactor vessel fuel support piece without the ability to raise or lower it through normal means. The inspector observed portions of the refueling activities and the ensuing maintenance support activities. It was determined from the control room that the inability to move the fuel assembly was the result of a rod movement interlock from a lost position on rod 22-47.

The Unit Supervisor initiated a maintenance activity (Work Authorization V71079) to resolve the interlock. The inspector observed portions of the WA and reviewed the following documents:

WA V71079, 22-47 Rod Out Interlock
MT-AD-509, Control of Minor Maintenance
CR 97-1303, Emergency Work Authorization to Clear Refueling Platform Interlocks
NDAP-QA-0500, Conduct of Maintenance
NDAP-QA-0502, Work Authorization System
MI-PS-001, Work Package Standard

Section 6.6.4 of NDAP-QA-502 addresses emergency work authorizations. It states that the Shift Supervisor assumes responsibility during offnormal hours for all groups represented in the Work Authorization procedure. No other allowance is given by NDAP-QA-502 concerning the conduct of work, and there is no relief given for the processes that actually control work in the field under NDAP-QA-0502.

During the performance of work under WA V71079, the inspector noted that the technician was using an "information only" SSES Training Department drawing to guide his activities. In addition, the technician was not documenting his activities on a Status Control sheet, NDAP-QA-502-5, nor was he documenting his activities on an Actions Taken form NDAP-QA-502-3.

The training drawing used by the technician to support WA 71079 was not prescribed by the licensee for work at SSES. Because the licensee was able to determine that none of the activities performed impacted on the operability of the reactor protection system, this issue being treated as a Non-Cited Violation consistent with Section IV of the NRC Enforcement Policy.

c. Conclusions

During fuel movement activities a fuel assembly was suspended (less than one foot) above its reactor vessel fuel support piece without the ability to raise or lower it through normal means. Maintenance activities were initiated on the Unit 2 reactor protection system to resolve this condition. Maintenance activities performed under Unit 2 work authorization WA 71079 on the reactor protection system were performed using an "information only" SSES Training Department drawing which was not authorized for use. This issue being treated as a non-cited violation.

M1.8 Maintenance Activities Resulting in a Plant Transient

a. Inspection Scope (62707)

On February 25, 1997 maintenance activities resulted in a loss of Unit 1 condenser vacuum. The inspector reviewed an SSES Event Review Team (ERT) Report and additional information supplied by the licensee to evaluate the event.

b. Observations and Findings

NRC Inspection Report 387,388/97-02 section M21.b.1 discussed the event. In that section it states that the concrete boring work was directly above the Unit 1

hydrogen analyzer cabinet. Following a review of the ERT and WA C63269, the inspector determined that a more accurate description of the location of the work with respect to the hydrogen analyzer would be approximately 15 feet above and offset approximately 8 feet to the north west. The previous report incorrectly discusses the event. The report should have stated that the 1997 event was similar to a 1996 transient which occurred in anticipation of a loss of condenser vacuum, and that the event was the result of weak circulating water pump maintenance. The weak maintenance included: the failure to remove a buildup of corona discharge material inside a connector box; the use of sealants on the motor connection box; and the failure of preventive maintenance activities to identify the connector cable damage, the corona discharge material or a contaminated standoff insulator prior to failure.

c. Conclusions

The conclusions of NRC Inspection Report 387,388/97-02 section M21.b.1 are unchanged.

M1.9 Maintenance Activities Under the Unit 2 Reactor Vessel

a. Inspection Scope (62707)

The inspector reviewed and observed (through a video link) maintenance activities conducted under the Unit 2 reactor vessel.

b. Observations and Findings

As a result of weaknesses identified in the under-vessel maintenance activities during the Unit 1 refueling outage, and CRs written by the licensee, the inspector observed and reviewed under-vessel activities during the Unit 2 refueling outage. The licensee issued CRs 97-0941, 97-0947, and 97-0936 which addressed split cables identified on Unit 2. Unit 1 had experienced split cables and the licensee's corrective actions included the placement of cable protectors to prevent maintenance related damage. The inspector determined that the licensee's corrective actions were adequate and that the inherent tight quarters under the vessel made maintenance activities very difficult.

c. Conclusions

As a result of weaknesses identified with under-vessel maintenance activities during the 1996 Unit 1 refueling outage, and condition reports written by the licensee, the inspector observed/reviewed under vessel activities during the Unit 2 refueling outage. The licensee issued and resolved a number of condition reports and took adequate corrective actions. No violations of NRC requirements were identified.

M1.10 Emergent Work - Minor Maintenance

Previous NRC inspection observations and non-cited violations identified that the documentation of actions taken and the "as left" condition following minor maintenance on safety related equipment was weak. On May 6, 1997, the inspector observed minor corrective maintenance on a suppression pool temperature indicator, TI-15751, located on the remote shutdown panel in Unit 1. The work was performed by the emergent work action crew (EWAC) in accordance with maintenance procedure MT-AD-509, Minor Maintenance, under work authorization S71572.

The inspector concluded that the scope and complexity of the work observed was similar to the examples provided in the Minor Maintenance procedure. The inspector observed appropriate documentation of the actions taken, appropriate post maintenance testing, and good communication of the as left configuration. During the activity, good supervisory interaction and communication between the work crew and the control room were noted.

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 Material Condition of Plant Equipment and Systems

a. Inspection Scope (62707)

During routine observations of plant operations, the general condition of equipment was examined to determine the effectiveness of licensee controls for identification and resolution of maintenance related problems.

b. Observations and Findings

The general condition of the facilities was discussed routinely with SSES operators and system engineers and was inspected in the field. Five issues were identified that required varying degrees of inspector review. These five issues were:

- Seven annunciated plant conditions associated with Unit 1 and Unit 2 ventilation

  This issue is discussed in section 2.1 of this report

- Unit 2 'D' residual heat removal (RHR) pump oil leak and Division 2 RHR Swing Bus Motor Generator Set slinging oil onto the wall while being internally contaminated

  Each of these issues had been previously identified by the licensee and had corrective action plans (CR 97-1568, 97-1535) in place and analyses to indicate that there was no impact on the safe operation of the involved unit.

- 'E' emergency diesel fuel line abutting a support during operation.

  This issue was identified during a routine resident inspector tour of the diesel building. A CR and a WA were written to resolve this issue. No immediate impact on diesel operability was identified by the inspector or the licensee.

- Bonnet leak on HV10606B

  On a plant tour, the inspector identified that the leakage from the Unit 1 Reactor Feed Pump discharge check was impacting plant cabling. The licensee determined that the affected conduit tray was approximately 170°F, there was no immediate impact, and is completing a long-term evaluation.

c. Conclusions

With the exception of the ventilation annunciator issue, the licensee has initiated adequate corrective actions.

M2.2 Unit 2 Containment Closeout Inspection

a. Inspection Scope (71707)

Prior to final closure of the Unit 2 containment, the inspector performed a walkdown of the drywell to assess PP&L's effectiveness in removal of foreign material, restoration of pipe insulation, cable raceway covers, electrical connections, flanges, piping, and supports.

b. Observations and Findings

In general, the cleanliness of the containment was very good. The inspector performed a containment walkdown in parallel with PP&L management's final tour. A small number of items were identified and removed during this tour. These items included plastic tie wraps, small pieces of wire, and duct tape.

The inspector identified three items that did not appear consistent with the expected equipment configuration during the containment tour:

- Screens protecting the air inlet for containment coolers 2V421A and 2V415A were missing.

- A number of hold-down clips were missing from a section of floor grating on elevation 738.

- A bundle of small gauge wire was installed with tie wraps on existing conduit and structures.

PP&L initiated a CR for each of these items and documented an operability determination. In all three cases, PP&L determined that the items were acceptable for operation and would not impact operability.

c. Conclusions

PP&L's efforts to remove foreign materials from the Unit 2 containment following the refueling outage were very good. The final containment walkdown and inspection by the Operations and Maintenance department managers were viewed as strength. Based on the areas reviewed during the inspector's containment walkdown, PP&L was effective in restoring equipment (hatches, insulation, hangars, etc.) to the condition required for plant operation.

M2.3 Unit 2 Suppression Pool Cleaning

a. Inspection Scope (62707)

The inspector reviewed PP&L's suppression pool cleaning activities during the Unit 2 refueling outage and confirmed implementation of the compensatory actions requested by the NRC in a letter dated February 19, 1997.

b. Observations and Findings

The Unit 2 suppression pool cleaning was performed by divers and included filtering the water and vacuuming the floor, structures, and reactor pedestal openings.

The pool water was filtered/vacuumed using 1.0 micron filters and resulted in the collection of approximately 1100 pounds of wet sludge and 71 pounds of rust particles. Debris collected during this evolution included rust particles, small pieces of tape, tie-wraps, small pieces of paper/plastic, a 25 foot hose, tags, strips of metal, a glove, a boot, small pieces of wire, small pieces of hose and rope, a hard hat, a soda can, a piece of weld guard (approximate 1 square foot), a 4" wire brush, nuts and washers.

PP&L's final inspection of the pool floor found no signs of debris and no signs of silt accumulation. The cleaning resulted in improved visibility (to approximately 11 feet below the water surface) with a fine particulate still suspended in the water. Visibility at the bottom of the pool was reported to be 2 to 4 feet.

All 87 downcomers were inspected and 5 were found to contain floating debris. The debris consisted of a piece of rope, 2 rubber boots, a paper tag, and small pieces of paper/plastic.

PP&L documented the items discovered during the inspection pool cleaning in the CR process. In all cases, PP&L determined that the items would not have prevented the emergency core cooling systems taking suction on the pool from performing their intended function.

The inspector observed PP&L's final inspection in the drywell to verify the proper configuration and condition of insulation. This activity also included a walkdown to verify that all foreign material had been removed from the drywell. These activities were noted to be through and are discussed in more detail in Section M2.2 of this report.

The inspector discussed the Unit 2 suppression pool cleanout and inspection results, and the implications for Unit 1, with the cognizant system engineer. PP&L considers the sludge removed from Unit 2 to be bounded by the operability evaluation (dated November 15, 1995) performed in response to NRC Bulletin 95-02. Based on review of this evaluation, the inspector determined that the Unit 2 suppression pool cleanout results did not invalidate PP&Ls assumptions or conclusions.

c. Conclusions

PP&L's efforts to reduce foreign debris in the Unit 2 containment and suppression pool during the Spring 1997 refueling outage were through. Management involvement in the final inspection of containment was viewed as a strength. The compensatory actions requested by the NRC in conjunction with deferral of the final resolution of Bulletin 96-03 were implemented by PP&L. The Unit 2 suppression pool cleanout results were consistent with the assumptions contained in PP&L's existing operability evaluation for the suction strainer clogging issue communicated in NRC Bulletin 95-02. No information was identified that would invalidate PP&L's conclusion regarding operability of either SSES Units' suppression pool strainers.

M3 Maintenance Procedures and Documentation

M3.1 Maintenance Rule Implementation - Back Draft Isolation Dampers

a. Inspection Scope (62707)

The inspector reviewed PP&L's Design Guide for System Scoping for Maintenance Rule Applicability, GDS-18, to determine if the program for implementation of the Maintenance Rule (10 CFR 50.65) identified the safety function of back draft isolation dampers (BDIDs) in the reactor building ventilation system.

b. Observations and Findings

Piping systems whose failure might generate hazardous environmental conditions are located in rooms which are capable of being isolated from required safety systems. Isolation of these rooms is provided, in part, by automatic BDIDs that actuate on differential pressure between the room and the general reactor building. The inspector was concerned that although pressure switch testing is periodically conducted, there was no evidence that BDIDs were exercised to demonstrate functionality.

10 CFR 50.65(b) states that the scope of the monitoring program required by the rule is to include safety related structures, systems, or components that are relied upon to remain functional during and following design basis events. FSAR Section 3.6.1.1 describes the Susquehanna design basis for a postulated pipe break outside the containment. Piping systems whose failure might generate hazardous environmental conditions are located in compartments which are capable of being isolated from required safety systems. The isolation of those compartments is, in part, accomplished by BDIDs in the reactor building ventilation system.

The inspector's review of GDS-18, Revision 3, System Scoping for Maintenance Rule Applicability, dated July 15, 1996, found that the BDIDs were not identified by PP&L as a maintenance rule function of the reactor building ventilation system. The reactor building ventilation system is identified as being within the scope of maintenance rule and fourteen separate maintenance rule functions of the system are identified. The inspector discussed this finding with the cognizant nuclear system engineering (NSE) supervisor, and the supervisor acknowledged the need for these dampers to be covered by PP&L's program. In response to this issue, CR 97-1648 was initiated by PP&L to address the omission of the BDID function from the maintenance rule program scope and the fact that no testing has been performed that confirms the dampers are capable of closing. As part of the corrective actions for CR 97-1648 the licensee prepared an interim operability determination that concluded that the BDIDs were operable because failure of the equipment was not expected based on the testing of the solenoids and the pressure switches. The inspector determined that the operability determination was weak in that it did not support why the BDIDs were expected to function mechanically when called upon.

At the end of the inspection, the following information was needed to determine the operability of the BDIDs and their status within the Maintenance Rule program.

Industry data regarding the functionality of unexercised dampers is needed.

A determination by the licensee of whether or not the BDIDs have ever stroked on demand.

A determination whether future testing of the dampers is needed.

A determination by the licensee whether the BDIDs should be included in the scope of the reactor Building Ventilation system under GDS-18 criteria, and whether a BDID failure is risk significant.

This issue will remain unresolved pending completion of the CR 97-1648 corrective actions **(URI 50-387, 388/97-03-04).**

c. <u>Conclusions</u>

The back draft isolation dampers are safety related components within the non-safety related reactor building ventilation system. Although the reactor building ventilation system is addressed by the maintenance rule program at SSES, the

function of the BDIDs was not included in the licensee's evaluation of the reactor building ventilation system. There was no performance history to indicate that the BDIDs would function on demand. A determination of the operability and maintenance rule status of the BDIDs in the reactor building ventilation system will be tracked as an unresolved item.

M7 Quality Assurance in Maintenance Activities

M7.1 Review of Post-Maintenance Testing

a. Inspection Scope (62707)

The inspector reviewed the most recent Nuclear Assessment Services (NAS) audit of the test control program at SSES, dated September 11, 1995. Specifically, PP&L's review of post-maintenance testing (PMT) was evaluated against the SSES Operational Quality Assurance Manual policy for Control of Inspection and Testing (OPS-14).

b. Observations and Findings

NAS performs an audit of the test control program every two years as discussed in FSAR Chapter 13.4. The 1995 audit reviewed a sample of fifteen completed work authorizations (WAs), out of approximately 22,000 activities in 1994/1995, to determine the adequacy of post-maintenance test activities. The inspector found that the sample included only WAs that had gone through the work planning process and did not appear to have sampled WAs for minor corrective maintenance performed under the licensee's Maintenance Investigation Instruction (superseded by the Minor Maintenance process).

The 1995 audit found that the post-maintenance functional test requirements, required by NDAP-QA-0482 were defined by the work group and that all WAs reviewed contained test requirements. In addition, the PP&L sample found that test results were properly documented, analyzed, and evaluated against test acceptance criteria to verify completeness and achievement of test objectives.

The inspector concluded that the 1995 NAS audit sampled too few safety related maintenance activities (15 out of 22,000) to provide a representative sample. In addition, the omission of unplanned maintenance (ie., minor maintenance) from the sample was considered a program weakness.

c. Conclusions

The Nuclear Assessment Services audit of the Test Control Program (Audit No. 95-059) provided an adequate review of post-maintenance testing as required by the Operational Quality Assurance Manual. However, the inspector considered the audit sample size to be small relative to the number of safety related work authorizations processed in a two year period and is considered a potential weakness. The lack of an NAS audit in the minor maintenance area was also

considered a weakness in testing program oversight. Based on these weaknesses, the inspector could not determine whether the 1995 NAS audit assessment provided PP&L management a representative assessment of all types of post-maintenance testing.

III. Engineering

E2 Engineering Support of Facilities and Equipment

E2.1 Hydraulic Compatibility of Atrium 10 Fuel in the Current Unit 2 Core Configuration

a. Inspection Scope (37551)

The inspector reviewed the cycle 9 Unit 2 core configuration for hydraulic compatibility and stability. This review was performed separate from the core thermal analysis that was conducted for the most recent fuel-related TS amendment.

b. Observations and Findings

The following proprietary documentation was made available by the licensee for NRC review:

- Thermal Hydraulic Characteristics of the Atrium 10 Fuel Design for Susquehanna
- Susquehanna SES Unit 2 Cycle 9 Hydraulic Compatibility Evaluation
- Susquehanna 2 SQB-8 Design Report Mechanical and Thermal Hydraulic Design for SPC Atrium 10 Fuel Assemblies

The review was conducted to determine if there were conditions that resulted in flow instabilities, reversals or other anomalies (separate from thermohydraulic conditions reviewed under a recent TS change). No such conditions were identified. The inspector did not retain any of the proprietary documentation.

c. Conclusions

The hydraulic performance (separate from thermohydraulic performance) of the Atrium 10 fuel was reviewed and determined to be adequately bounded by analysis.

E2.2 Engineering Support of 4160 Volt Circuit Breaker Operability

a. Inspection Scope (62707)

On April 18, 1997, a PCO attempted to start the Unit 2 'A' RHR pump and received no response. The inspectors reviewed the root cause of this event (CR 97-1363),

the impact of the failure on the operability of other safety related equipment, the licensee's corrective actions, and the generic implications of the failure.

b. Observations and Findings

The subject breaker contains a personnel protection device designed to not allow the breaker to close if it is fully or partially racked out. This protection device is referred to as a tripper lever and was the subject of a previous NRC Information Notice, (IN) 96-50, September 4, 1996. Upon investigation of the failure, the licensee determined that this tripper lever was not in the fully down and level position, causing the breaker not to close when called on to perform by the PCO.

The licensee responded very conservatively to the previous issue discussed in the IN. However, the previous failure mode had been isolated to cases where the breakers had been racked-in just prior to a test. In the SSES RHR case, the breaker had been cycled successfully prior to its failure and had not been racked-out/in just prior to the test. The licensee took immediate actions to ensure the operability of the other safety related breakers on the operating unit and took generic action on all other safety related breakers. Included in the corrective actions was the performance of a number of dimensional measurements of the breakers. Several minor indications were identified through visual inspections and were corrected by the licensee (examples CR 97-1547, and 97-6125).

The inspector reviewed the availability of the Unit 2 'A' RHR pump and determined that no TS violations occurred.

The appropriate technical information was transferred to the SSES NRR Licensing Project Manager for generic review and resolution.

c. Conclusions

Following the failure of a safety related 4160 Vac breaker to perform when requested, the licensee identified a possible new failure mode involving a personnel protection device referred to as a tripper lever. Although the failure mode was different, this same device that was the subject of IN 96-50. The licensee's response to the IN was very conservative and aggressive. The involvement of first and second line engineering management in this issue was laudable. The generic aspects of the issue have been forwarded to NRR for review.

E8 Miscellaneous Engineering Issues (92902)

E8.1 Review of FSAR Commitments

A recent discovery of a licensee operating its facility in a manner contrary to the Updated Final Safety Analysis Report (UFSAR) description highlighted the need for a special focused review that compares plant practices, procedures and/or parameters to the UFSAR description. The inspectors reviewed the applicable portions of the UFSAR and SSES emergency operating procedures that relate to post-accident

alternative water supplies to the core in order to determine if the licensee maintained the capability of accessing these alternative water supplies.

b. Observations and Findings

The inspectors reviewed the following SSES operating and emergency operating procedures:

ES-150/250-002, Boron Injection via Reactor Core Isolation Cooling
ES-013-001, Fire Protection System Cross Tie to RHR Service Water
SO-013-001, Monthly Hose House Inspection
SO-253-004, Quarterly SBLC Flow Verification

The licensee maintained and documented adequate access to the emergency alternate water sources identified in the SSES FSAR.

c. Conclusions

The licensee maintained the capability to utilize emergency alternate water sources identified in the SSES Emergency Plan and discussed in the FSAR. No violations of NRC requirements were identified.

E8.2 (Closed) URI 50-387, 388/96-08-03: Open HELB Room Doors

a. Inspection Scope (37551)

On three occasions, doors for personnel access to rooms equipped with high energy line break (HELB) protective features were blocked open during on-line maintenance activities. The inspector opened the unresolved item pending additional information from PP&L that was necessary to determine if this condition was adequately bounded by existing analysis.

b. Observations and Findings

The plant design includes the ability to sustain a high energy pipe break accident coincident with a single active failure and retain the capability for safe cold shutdown (reference NUREG 0776). The plant was designed in accordance with Branch Technical Position (BTP) ASB 3-1 "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment" and PP&L used separation as the primary means of protection.

At the time the open doors were identified, PP&L was using a TS interpretation (No. 1-92-006) in place to provide operational restrictions for opening doors, hatches, and plugs. This interpretation permitted the opening or removal of HELB room boundaries in support of work activities but prohibited their being left open indefinitely. The inspector's review of PP&L's basis for the interpretation (EWR M10103) found that it did not address generic assumptions regarding the open room boundary dimensions, or the potential impact on environmental qualification

and divisional separation. PP&L's immediate actions (after the third door was identified) included removal of the subject TS interpretation pending a review of its basis.

In response to this issue, PP&L performed operability determinations for the three subject plant areas where doors were blocked open in support of maintenance (reference CRs 96-2191 and 96-0748). The evaluations addressed seventeen aspects of the SSES design basis, including those originally questioned by the inspector. In each evaluation, PP&L concluded that there was no impact on the operability of the subject design features. The inspector's review of these operability determinations did not identify any problems with PP&L's assessment.

Due to the complexities of these assessments, PP&L has determined that a generic assessment to cover all plant areas and combinations of open doors, floor plugs, hatches, etc., is not practical. PP&L management stated that, in the future, a safety evaluation would be performed to support specific work activities that would block open a room boundary that could affect the HELB protective features. The inspector noted that this type of evaluation is now required by NDAP-QA-0409, which was approved on February 27, 1997.

Although PP&L's analysis after the events determined that the plant had been operated within its design basis, the failure to perform safety evaluations before blocking open the HELB room doors is a violation of 10 CFR 50.59. This violation is example (a) of **VIO 97-03-04.**

c. Conclusions

PP&L's failure to perform safety evaluations prior to blocking open doors for rooms with HELB protective features is a violation of 10 CFR 50.59. Subsequent PP&L evaluations determined that blocking open the HELB room doors created no adverse impact on operability or conditions outside the plant's design basis.

E8.3 (Closed) IFI 50-387, 388/96-04-01: Battery Charger Setpoints

a. Inspection Scope (62703)

This inspection followup item was opened pending NRC review of PP&L's formal evaluation of increased 250 Vdc system float voltages.

b. Observations and Findings

In 1994, the 60 month performance discharge test for 125 Vdc battery 2D620, found the battery had 83.5% capacity. Although this met the TS required minimum capacity of 80%, this result was unexpected for a 5 year old battery. PP&L determined that the cause of the capacity degradation was sulfation resulting from low float voltage. As corrective action for this problem, PP&L increased the float voltage for both 125 and 250 Vdc batteries.

NRC Inspection Report (IR) 96-04 reviewed the change in float voltages after discrepancies were identified in plant log acceptance criteria. Section E2.1 of IR 96-04 stated that PP&L calculation EC-088-0530, Revision 1, did not address the acceptability of the 125 Vdc float voltage. However, this was not correct; Calculation EC-088-0530, Revision 1, did not address the acceptability of the 250 Vdc float voltage.

PP&L's review of this issue (CR 96-0475) found that NSE had authorized the increase in float voltage for the 250 Vdc battery (under WAs S51447 and V50816) as a long-term corrective action for the 1994 failure. However, a request for Systems Analysis to evaluate the effects of the increased float voltage on equipment connected to the 250 Vdc battery was not processed.

Revision 2, to EC-088-0530, Attachment 6, evaluated the effects of raising the Class 1E 250 Vdc battery charger float voltage from 265 Vdc to 268 Vdc (each with a band of ±3 volts) to assure that there would be no adverse effects on the safety function of connected components. PP&L concluded that all safety related 250 Vdc equipment can withstand continuous operation at the new float voltage without loss of life or adverse impact to nuclear safety. The inspector reviewed this revision of the calculation, discussed several questions with the cognizant PP&L engineer, and concluded that a technical basis exists for PP&L's conclusions.

PP&L's failure to perform a safety evaluation for the increase in 250 Vdc float voltage is a violation of 10 CFR 50.59. This violation is example b of **Vio 97-03-04.**

c. Conclusions

Revision 2 of PP&L's calculation EC-088-0530, documented the evaluation of 250 Vdc battery float voltage and appropriately considered the potential for degradation of connected safety related equipment. No degradation of the equipments ability to perform its intended safety function was identified.

E8.4 (Closed) URI 50-387,388/95-24-01: Temporary Monitoring Equipment

a. Inspection Scope (37551)

This unresolved item was opened pending PP&L's documentation of a safety evaluation for temporary test equipment (visicorder) used on the emergency diesel generators. The inspector found that the Bypass Program, which controlled temporary monitoring equipment, did not require a formal evaluation for use on safety related equipment, when the monitoring equipment was in place for less than seven days.

b. Observations and Findings

In response to the inspector's findings documented in IR 50-387/95-24 and issues raised by NRC Information Notice 95-13, PP&L took the following actions:

- Administrative procedures governing the Bypass Program (NDAP-QA-0484) and the Work Authorization System (NDAP-QA-0502) were revised to remove the exception that allowed temporary monitoring equipment to be installed for seven days without a safety evaluation.

- A review was conducted to ensure that the Bypass Program required evaluations for installation of temporary monitoring equipment would provide sufficient barriers for concerns raised in IR 95-25 and IN 95-13 (and IN 95-13, Supplement 1).

- Training was conducted for station engineering personnel regarding the changes to the Bypass Program (deletion of the seven day allowance), industry events, IN 95-13, and the 10 CFR 50.59 evaluation requirements for temporary changes.

- Training was conducted for maintenance production supervisors, similar to the training for engineers, to emphasis the procedural changes that require all temporary monitoring instrumentation to be processed as a bypass.

The inspector reviewed the actions taken in response to these issues, the safety evaluation for the diesel generator temporary monitoring equipment and discussed the Bypass Program review process with a cognizant engineer and NSE supervisor. The inspector concluded that the reviews required by the Bypass Program provides controls to ensure that the in-field configuration matches the approved configuration, the reviews addressed relevant design considerations, and that there is no impact on operation of equipment due to the installed monitoring instrumentation.

PP&L failed to perform a safety evaluation to support the installation of temporary test equipment on the emergency diesel generators. This failure constitutes a violation of 10 CFR 50.59, "Changes, Tests and Experiments." This violation is example (c) of VIO 97-03-04.

c. Conclusions

PP&L's failure to perform the safety evaluation required by 10 CFR 50.59 prior to installing temporary test equipment on the emergency diesel generators is a violation. A subsequent evaluation determined that there was no impact on operability. As part of the corrective action for this violation, changes to the Bypass Program and Work Authorization System removed the inappropriate exemption from performing safety evaluations that allowed the violation to occur.

IV. Plant Support

S1 Conduct of Security and Safeguards Activities

S1.1 Access Practices on Vital Doors

a. Scope

The inspector reviewed security surveillance activities intended to determine the proper operation of site vital area access doors.

b. Findings

Surveillance NS-SSP-004, Test Check and Inspection of Security Data System, and NS-SO-004-1, Alarm Log, were reviewed in regards to a recent performance test. PP&L decided to perform the test after questions were raised by the inspector regarding the proper operation of doors and red indicating lights prior to personnel access into vital areas.

The door alarm system was evaluated by the inspector to perform as described in the SSES security plan. The inspector determined that in some instances, personnel may be given the impression that they were inappropriately given access to vital areas, based on the illumination of red indicating door lights. General employee training requires a practice of calling security after receiving two illuminated red indicating lights following security door access attempts. The practice of calling security after receiving two red indicating door lights on sequential door key access attempts may not be uniformly followed in the field. The inspector identified this weakness in the application of general employee training in the field. This weakness may have led to employees misunderstanding the meaning of illuminated red door indicating lights. However, this misunderstanding does not bear on the adequacy of the security system design nor the licensee's implementation of the security plan, both of which were determined to be adequate.

c. Conclusion

The licensee met the requirements of its security plan with respect to vital area door access. The licensee's surveillance activities were carefully and well performed. Some aspects of general employee training could be improved to make the operation of the door alarm system clearer to plant employees. No violations of NRC requirements were identified by the inspector.

F8 Miscellaneous Fire Protection Issues

F8.1 Review of UFSAR Commitments

A recent discovery of a licensee operating its facility in a manner contrary to the UFSAR description highlighted the need for a special focused review that compares plant practices, procedures and/or parameters to the UFSAR description. The

inspectors reviewed the applicable portions of the UFSAR and the SSES Fire Protection Review Report (FPRR) that relate to the back up fire protection system.

b. Observations and Findings

The SSES FPRR, section 4.1, states that, in addition to the normal fire protection system, the SSES site has a backup fire protection system which consists of a 2500 gpm diesel driven fire pump, a jockey pump and a dedicated water supply. The 2500 gpm pump is not part of TS requirements and is isolated from the main yard loop. The backup fire protection system and the normal fire protection system can be cross tied.

The normal and backup fire protection systems are isolated during routine standby alignment. On May 5, 1997, the licensee cross connected the systems and the systems have remained in the cross connected condition to the end of this inspection report period (May 19, 1997). The systems were cross connected to prevent leakage through back flow preventer valve O22-528. The valve had two associated deficiency tags attached to it when the inspectors traced parts of the system in the field. The deficiencies (21690, 21045) were written on April 21, 1997 and May 4, 1997, respectively.

The licensee used operating procedure OP-013-003, step 3.4, to cross connect the backup fire protection system with the normal fire protection system plant yard loop. The inspector walked down portions of the system and reviewed the applicable procedures to determine if the backup system was operated and maintained in the same general condition as the normal fire protection system. The following procedures were reviewed:

TP-013-026, 18 Month Function Test of Common Out Building Sprinkler Systems
CL-013-031, Backup Fire Protection System Electrical
CL-013-032, Backup fire Protection System Mechanical

The inspector determined that the physical condition and procedural requirements of the backup fire protection system were similar to those of the normal fire protection system.

SSES calculation EC-013-0996 evaluated the basis and values for the backup fire protection system jockey pump and diesel pump setpoints. This calculation states that the backup fire protection system shall operate as an integral part of the normal fire protection system by acting as a supplemental source of pressurized water. The normal fire protection system provides sufficient water to satisfy design basis requirements. However, in an extreme case, the demand for pressurized water may exceed design basis expectations. Setpoints should be selected so that when the demand exceeds design basis expectations, the cross-ties may be opened and the backup fire protection system brought on line as a supplemental source of pressurized water. This calculation does not address the cross connection of the two systems for other than extreme conditions and does not address the routine

supply of keep-fill pressure to the backup fire protection system piping from the normal fire protection system.

Based on the licensee description in the SSES FPRR and the three conditions used to calculate system setpoints in EC-013-0996, the inspector determined that the configuration, as described in the FPRR and understood by the NRC, is that the two fire protection systems should normally be operated isolated from each other. Further, according to the PP&L setpoint calculation, the cross-tied condition would be reserved for extreme demand conditions - beyond design basis expectations.

The inspector determined that the normal and backup fire protection systems have remained in the cross connected condition without an adequate safety evaluation. This constitutes a change to the normal fire protection system which is described in the FSAR and TS 3/4 7.6. The change was not preceded by an evaluation to determine if an unreviewed safety question would result from the cross tie of the two fire protection systems and is a violation. This violation is example (d) of **VIO 387/388/97-03-04.**

c. Conclusions

On May 5, 1997 the licensee cross connected the normal and backup fire protection systems and the systems have remained in the cross connected condition at the end of this report period. This alignment constitutes a change to the normal fire protection system which is described in the FSAR and TS 3/4 7.6. The change was not preceded by an evaluation to determine if an unreviewed safety question would result from the cross-tie of the two fire protection systems.

V. Management Meetings

X1. Exit Meeting Summery

The inspectors presented the inspection findings for this report period to members of PP&L management at the conclusion of the inspection on May 20, 1997. the licensee acknowledged the findings presented, with no exceptions taken. No proprietary information is included in this report.

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

387,388/97-03-01	VIO	Two Examples of Inadequate Procedures (Refueling Operations, SPING Alarm Response Procedure)
387,388/97-03-02	VIO	Failure to Perform Core Spray Surveillance Test Under Controlled Conditions
387,388/97-03-03	URI	Omission of the Back Draft Isolation Dampers Function in the SSES Maintenance Rule Program
387,388/97-03-04	VIO	Four Examples of a Failure to Perform a Safety Evaluation Prior to a Design Change

Closed

50-387/97-008-00	LER	Instrument Response Time Testing
50-387/97-009-00	LER	Roving Fire Watch Rounds Not Completed On Time
50-387,388/96-08-03	URI	HELB Room Doors
50-387,388/96-04-01	IFI	Battery Charger Setpoints
50-387,388/95-24-01	URI	Temporary Monitoring Equipment

LIST OF ACRONYMS USED

AR	Alarm Response
BDID	Back Draft Isolation Dampers
BTP	Branch Technical Position
CFR	Code of Federal Regulations
CR	Condition Report
EAL	Emergency Action Level
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EP	Emergency Preparedness
ERT	Event Review Team
EWAC	Emergent Work Action Crew
FPRR	Fire Protection Review Report
FSAR	Final Safety Analysis Report
HELB	High Energy Line Break
IN	Information Notice
LCO	Limiting Conditions for Operation
LER	Licensee Event Report
LP	Low Pressure
NAS	Nuclear Assessment Services
NCV	Non-Cited Violation
NOV	Notice of Violation
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NSE	Nuclear System Engineer
PCO	Plant Control Operator
PMT	Post Maintenance Testing
QA	Quality Assurance
RFP	Reactor Feed Pump
RHR	Residual Heat Removal
RIE	Replacement Item Evaluations
SALP	Systematic Assessment of Licensee Performance
SOOR	Significant Operations Occurrence Report
SPING	System Particulate Iodine Noble Gas
SS	Shift Supervisor
SSES	Susquehanna Steam Electric Station
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
US	Unit Supervisor
WA	Work Authorization