

UNITED STATES NUCLEAR REGULATORY COMMISSION

REGION I

Inspection Report Nos. 50-387/95-12; 50-388/95-12

License Nos. NPF-14; NPF-22

Licensee: Pennsylvania Power and Light Company
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Allentown, Pennsylvania 18101

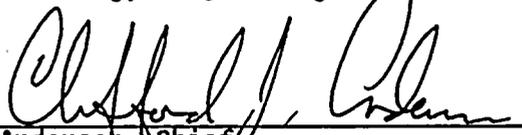
Facility Name: Susquehanna Steam Electric Station

Inspection At: Salem Township, Pennsylvania

Inspection Conducted: May 16, 1995 - July 4, 1995
July 10, 1995 - July 13, 1995

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8/2/95
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EXECUTIVE SUMMARY

Susquehanna Inspection Reports
50-387/95-12; 50-388/95-12
May 16, 1995 - July 4, 1995

Operations

The licensee completed power uprate testing for Unit 1 in a controlled and safe manner. Test results showed lower containment vibration levels than during the Unit 2 power uprate testing. Anomalies encountered during the test program were properly dealt with by operators and with engineering support. (Section 2.1)

Operators appropriately responded to an electrical transient caused by the failure of the Unit 2 non-vital instrument Uninterruptible Power Supply (UPS) 2D240 on June 6th. The instruments were supplied from the maintenance bypass until a thorough engineering evaluation of the UPS was completed and corrective actions taken. (Section 2.1)

A walkdown of the Standby Gas Treatment System found the system in good physical condition, and the inspector noted that the licensee is working toward resolution of a damper actuator reliability problem. (Section 2.2)

A review of open bypasses and jumpers (temporary modifications) found that the licensee's program for controlling open bypasses was functioning satisfactorily. (Section 2.3)

Maintenance/Surveillance

The High Pressure Core Injection (HPCI) system on-line maintenance safety assessment adequately addressed the risk considerations and proposed conservative mitigative measures. In the past few months, the licensee had successfully completed various on-line maintenance activities on both units. However, events during the HPCI on-line maintenance indicated that improvements were needed in the licensee's process for planning, coordination and control of sequential tasks, system configuration and equipment status. The root causes and corrective actions are being investigated by PP&L. (Section 3.1.1)

Engineering/Technical Support

The sizing of thermal overload protection devices for HPCI system motor-operated valves was reviewed because the inspector noted that the devices, which are normally bypassed, are relied upon for the operability of the HPCI system valves during the quarterly Reactor Core Isolation Cooling (RCIC) flow surveillance. The devices were found to be adequately sized based on the expected current requirements of the motor-operators under design basis conditions. (Section 4.1)

(EXECUTIVE SUMMARY CONTINUED)

Plant Support

Acceptable performance in the Health Physics and Security areas was observed during this period.

A review of twelve fire protection related Licensee Event Reports (LERs) found that PP&L is taking effective corrective actions for previously identified weaknesses in the fire protection program. (Section 6.1.1)

Safety Assessment/Quality Verification

After reviewing three licensee condition reports and two associated NRC notifications related to instrument failures that effected core thermal power calculations, the inspector noted that the immediate actions taken by the operating shift did not identify two additional instrument errors effecting the calculation. These instrument errors were subsequently discovered during implementation of corrective actions a month and a half later. The Corrective Action Team (CAT), consisting of the Functional Unit Managers and the VP Nuclear Operations, routinely provides very good front end involvement and direction for resolution of condition reports. However, the inspector observed that the CAT directed actions for these events did not address the broader short term implication of the deficiency or interim actions that could be taken pending the formal root cause and subsequent corrective actions. (Section 6.3)

The inspectors identified that eight supplemental LERs from PP&L were overdue. Report information concerning the cause of the events and corrective actions are sometimes deferred in initial LERs due to expanded licensee reviews, such as Event Review Teams, continuing beyond the 30 day report period. The late supplemental LERs are overdue by at least six months with some more than one year. Also the status of the corrective actions for each LER was not readily available. This issue is unresolved pending information from PP&L regarding their process for tracking supplemental LERs and implementation of corrective actions. (Section 6.1)

A total of 20 LERs and five unresolved items were reviewed and closed based on the inspectors assessment of the licensee's corrective actions.

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DETAILS

1. SUMMARY OF FACILITY ACTIVITIES

Susquehanna Unit 1 Summary

This report period began with Unit 1 in a power ascension following its 8th refueling outage and implementation of a 50 MWe power uprate. Repair work on the 'C' reactor feed pump (RFP) seals had delayed the power ascension and limited reactor power to 80%. Power was further reduced to 40% for leak repairs on the 'B' RFP discharge check valve flange. By May 29th, both RFPs were returned to service and the Power Urate Program (PUP) testing had resumed. During PUP tests, two anomalies were encountered: A core flow calibration problem which led to the incorrect adjustment of the recirc MG set scoop tube stops, and entry into Technical Specification (TS) 3.0.3, that required actions to be initiated within one hour to place the unit in at least hot shutdown within the following twelve hours and cold shutdown in subsequent 24 hours. The TS limiting condition for operation was exited after the scoop tube stops were reset. A down power to 60% power was necessary to fix a loose control circuit cable connection on turbine control valve #3. On June 10, operators increased reactor power to the uprated 100% core thermal power of 3441 MWt. Operation at or near the new 100% power level continued throughout the remainder of the period. Routine power reductions were made to accommodate control rod sequence exchanges and turbine control valve testing.

The licensee completed their review and approval of the PUP test results on June 30, 1995, and all critical test criteria were satisfied. At greater than 103 Mlbm/hr core flow, less vibration and containment noise were observed in Unit 1 than in Unit 2 during its power uprate test program in June 1994. Significant noise and vibrations were noted in Unit 2 in June 1994 when core flow was increased from 100 Mlbm/hr to 108 Mlbm/hr (see combined NRC Report 50-387/94-11; 50-388/94-12).

Susquehanna Unit 2 Summary

Unit 2 began and ended this inspection period at 100% power. In the interim, power reductions were made in response to several minimum power generation requests by the load dispatcher and routine operations that included turbine valve testing and control rod sequence exchanges. On June 7th power was lost to the non-vital instrument loads supplied from UPS 2D240 when its output breaker opened unexpectedly. Two ENS notifications were made in June regarding instrumentation problems that resulted in minor core thermal power calculation errors.

2. PLANT OPERATIONS (71707, 92901, 93702, 40500)¹

2.1 Plant Operations Review

The inspectors routinely observed the conduct of plant operations to independently verify that the licensee operated the plant safely and according to station procedures and regulatory requirements.

Control room indications and plant systems were independently observed by NRC inspectors to verify that plant conditions were in compliance with station operating procedures and Technical Specifications (TS). Control room alarms and bypass indication system (BIS) warnings were routinely reviewed and discussed with operators; Operators were cognizant of control board indications and plant conditions. Control room and shift manning were in accordance with TS requirements.

The inspectors observed that plant management emphasized the need to execute the power uprate testing in a safe and controlled manner. Portions of the testing and status meetings were observed. Discussions with the managers of Operations and Nuclear Systems Engineering revealed that they were well informed of ongoing issues and had taken appropriate actions to resolve them.

Operators properly responded to a loss of non-vital instrument distribution panels 2Y218 and 2Y219 when the output breaker for Uninterruptible Power Supply (UPS) 2D240 tripped open unexpectedly and the UPS failed to transfer to its alternate supply. After verifying that the plant was stable and that no fault existed on the instrument bus, power was returned to the distribution panels via the maintenance bypass switch, and all systems were restored. As a result of this failure and those previously encountered (reference NRC Report 50-387/95-08), the licensee is performing a more detailed review of the non-vital instrument UPS. The referenced NRC Report cited a recent post-scrum recovery event during which the UPS did not transfer to its alternate supply, when required, due to an inoperable battery and not being aligned to the alternate supply as described in the Final Safety Analysis Report (FSAR). The licensee's corrective actions will be reviewed in a future inspection along with the response to the Notice of Deviation from an FSAR commitment issued in report 95-08.

The inspectors conducted regular tours of the various plant areas and periodically reviewed logs and records to ensure compliance with station procedures, to determine if entries were correctly made, and to verify correct communication of equipment status. These records included various operating logs, turnover sheets, blocking permits, and bypass logs. The inspector observed plant housekeeping controls including control and storage of flammable material and other potential safety hazards. Posting and control of radiation, high radiation, and contamination areas were appropriate.

¹ The inspection procedure from NRC Manual Chapter 2515 that the inspectors used as guidance is parenthetically listed for each report section.



The inspectors performed 50.5 hours of backshift and deep backshift inspections during the period. The deep backshift inspections covered licensee activities between 10:00 p.m. and 6:00 a.m. on weekdays, weekends, and holidays.

2.2 Standby Gas Treatment System ESF Walkdown

The Standby Gas Treatment System (SGTS) is designed to treat gas being released from the primary and secondary containment before being released to the environment, and to maintain post accident offsite radiation dose within the 10 CFR Part 100 limits. During this inspection period, the inspector completed a walkdown of the accessible portions of this Engineered Safety Feature (ESF) system to verify its operability. The system procedures were reviewed and equipment was inspected to identify conditions that might degrade performance of the system. The inspector also observed portions of preventive maintenance on the 'A' SGTS train and the monthly SGTS operability surveillance (SO-070-001).

During the system walkdown the inspector noted that the hydraulic damper actuators HDM-07552A(B) and HDM-07553A(B) were running continuously and that their actuator housings were very hot. The inspector questioned the effects of this high temperature on the hydraulic actuator oil and internal components. Discussions with the system engineer revealed that these actuators are performing as designed, but the vendor did not consider the consequent temperature increase. As a result, the actuators have a history of failures greater than the industry average. The licensee considers these failures to be primarily a reliability problem since the actuators fail to their safe position and do not impact the operability of the SGTS. Maintenance personnel believe the actuator's seals are degrading at an accelerated rate due to the high temperature caused by the excessive heat generated by the actuator. The system engineer has been tracking the failures since 1990 and the 1994 semiannual system status reports indicate that a modification (DCP #94-9040) has been approved to install a different model actuator that will not run continuously and eliminate these recurring failures.

No additional deficiencies were identified during this ESF walkdown and system review. The inspector concluded that the system is in good physical condition, and that the licensee is working toward resolution of the damper actuator problem. The inspector had no further questions.

2.3 Jumper/Bypass Log Review

On June 19, 1995, there were 26 bypasses installed in Unit 1, including those on systems common to both Units. At Susquehanna, bypasses are controlled via procedure NDAP-QA-484, Nuclear Department Bypass Program. The inspector noted that out of the 26 existing bypasses, 15 were initiated in 1995, 9 in 1994 and 2 in 1993. The NDAP provides criteria for safety evaluation and requires a semi-annual review of open bypasses that includes justification for keeping bypasses open for more than six months and a plan for closure.

From a sample review of the open bypasses and walk down of the control room panels, the inspector concluded that the licensee addressed 10 CFR 50.59 considerations, evaluated impact on technical specification equipment operability, and that the control room panels contained clear indication of equipment status made inoperable by a bypass. The inspector reviewed the control room drawings to ensure that appropriate drawing change notices (DCN) were posted. Except for one DCN all others were posted. The inspector checked several other control room drawings for posted DCNs, and did not find any other discrepancy. Following notification, the licensee immediately posted the DCN. Based on this finding, the inspector noted the above discrepancy as an isolated event.

Regarding the semi-annual review for open bypasses, the inspector reviewed a licensee's report for a period of July 1 through December 31, 1994. As required by the NDAP, this report addressed justification for keeping certain bypasses open for more than six months and a plan for closure. The report was reviewed by the Plant Operations Review Committee (PORC) (meeting number 95-007) and found acceptable. The inspector concluded the licensee's program for controlling open bypasses was satisfactory.

3. MAINTENANCE AND SURVEILLANCE (62703, 61726, 92902)

3.1 Maintenance Observations

The inspector observed and/or reviewed selected maintenance activities to evaluate whether the work was conducted in accordance with approved procedures, regulatory guides, Technical Specifications, and industry codes or standards. The following items were considered, as applicable, during this review: Limiting Conditions for Operation were met while components or systems were removed from service; required administrative approvals were obtained prior to initiating the work; activities were accomplished using approved procedures and quality control hold points were established where required; functional testing was performed prior to declaring the involved component(s) operable; activities were accomplished by qualified personnel; radiological controls were implemented; fire protection controls were implemented; and the equipment was verified to be properly returned to service.

Maintenance observations and/or reviews included:

- WA P50972, Standby Gas Treatment System (SGTS) Fan Lube And Inspection, June 21, 1995
- WA P42867, SGTS Supply Plenum Damper Solenoid Replacement, June 21, 1995
- WA P46127, Residual Heat Removal System Motor Operated Valve F004B MOV Actuator Overhaul, June 19, 1995
- WA S53776, Mechanically Clean Laundry Drain Sample Tank Effluent Line, June 18, 1995.

Based on the sampled portions of the above maintenance, the inspector concluded that the work was conducted and completed appropriately, with due concern for plant safety and procedures.

3.1.1 HPCI On-line Maintenance

On June 26, 1995, the licensee commenced a four day on-line maintenance work window for the Unit 2 High Pressure Core Injection (HPCI) system. The work scope of the outage included breaker maintenance, a check valve inspection, local leak rate testing of containment isolation valves, turbine overspeed trip mechanism inspection and testing, calibration of various instruments, and a snubber inspection. SSES procedure NDAP-QA-0900, Assessment of On-Line Work Windows, provides guidance for evaluation and documentation of the technical basis for performing on-line work. The work plan was reviewed by Nuclear Technology in Nuclear Engineering and a safety assessment was prepared following the guidelines of the NDAP.

The safety assessment analyzed the risk significance of the HPCI system being unavailable for seven days in terms of changes to the plant core damage frequency for ATWS and design basis transients. Plant risk was evaluated in terms of the probabilities of an initiating event and resulting core damage. For the risk significant events, core damage frequency alone was judged to be about 100 times more likely than core damage with containment failure. Hence, containment failure was not considered. Other preplanned work activities and emergent work, including maintenance and surveillance tests, were reviewed for their impact on the probability of initiating an event and the plants ability to mitigate an event. The licensee concluded that with HPCI out of service for seven days, a 40% increase in risk of core damage will result. Compensatory actions were recommended to increase the reliability of systems to reduce the risk. With the compensatory measures in place, the overall risk of core damage increased by only 10%.

The inspector reviewed the plant control room operator and Unit Supervisor turnover sheets and licensee's surveillance test database to determine if recommended mitigative actions for the HPCI work window were taken. The safety assessment recommended removing the monthly functional test of the Automatic Depressurization System (ADS) accumulator backup gas low pressure alarm channels (SI-225-201) from the work window. However, the inspector found that this surveillance was performed on June 28, 1995 within the HPCI work window. The licensee indicated, upon a further review of the work scope, that the maintenance planners and I&C had determined that the surveillance could be performed within the work window. However, this deviation from the safety assessment recommendation was not communicated to engineering. The licensee initiated a condition report and an Event Review Team (ERT) was commissioned to investigate root causes and the required corrective actions.

The inspector also noted that three events occurred during the work window. The first one involved a water hammer in the HPCI discharge line while restoring from a local leak rate test (LLRT) alignment. It appears that a void existed in the pipe between the HPCI pump discharge check valve and the discharge valve, such that when the discharge valve was opened a water hammer

resulted. After a walkdown and review of the system configuration, the licensee concluded no damage resulted from the event and the system was operable.

The second event involved loss of status control of four containment isolation valves in the HPCI system. The breakers for these valves were inadvertently closed following breaker maintenance. Since the valves were still not declared operable, Technical Specification 3.6.3 requires the valves to be closed and deactivated with their power source removed. The valves remained closed during the event, and containment isolation was maintained.

A third event involved hangers for drained portions of system piping that were not pinned as required by plant procedure. The licensee had performed a safety evaluation that indicated such pinning was not needed. However, at the time of the event, the plant procedure requiring pinning was not changed.

The licensee initiated condition reports for all these events. At the end of the inspection period, the licensee was continuing with their review for root causes and needed corrective actions.

The inspector concluded that the risk considerations of on-line HPCI maintenance were conservatively addressed in the licensee's safety assessment. Although the impact of the three events to actual plant safety was minimal, the incidents raised questions regarding the adequacy of the licensee's process for controlling the planned evolutions, system configuration and status control. This item will remain unresolved pending completion of licensee's review for root causes and needed corrective actions. (URI 50-388/95-12-01)

3.2 Surveillance Observations

The inspector observed and/or reviewed the following surveillance tests to determine whether the following criteria, if applicable to the specific test, were met: the test conformed to Technical Specification requirements; administrative approvals and tagouts were obtained before initiating the surveillance; testing was accomplished by qualified personnel in accordance with an approved procedure; test instrumentation was calibrated; limiting conditions for operations were met; test data were accurate and complete; removal and restoration of the affected components were properly accomplished; test results met Technical Specification and procedural requirements; deficiencies noted were reviewed and appropriately resolved; and the surveillance was completed at the required frequency.

Surveillance observations and/or reviews included:

- S0-250-002, Reactor Core Isolation Cooling Pump Quarterly Flow Verification, June 2, 1995
- S0-070-001A, Monthly Standby Gas Treatment Surveillance, June 21, 1995

Based on observation of selected portions of the above surveillances, the inspector concluded that they were completed with appropriate consideration for safe plant operation and administrative control.

4. ENGINEERING (71707, 37551, 92903)

4.1 HPCI MOV Thermal Overloads

The Reactor Core Isolation Cooling (RCIC) pump quarterly flow surveillance (SO-250-002) directs operators to place the HPCI Division II motor operated valve (MOV) thermal overload (TOL) bypass switch to the test position. This places TOL protection in service for almost all of the HPCI valves in order to protect a common HPCI/RCIC valve F011, during alignment of the RCIC test flow path to the condensate storage tank. The inspector observed that during the RCIC surveillance, the capability of the HPCI MOV TOLs is relied upon for operability of the valves and that this electrical alignment is not typical when HPCI is operable. Technical Specification (TS) 3.8.4.2.1 requires that the TOLs for these valves be continuously bypassed and has an allowed outage time of 8 hours before the valves must be declared inoperable. Entry to this allowed outage time is controlled by a procedure step requiring operator's sign off. The inspector questioned whether the current MOV capability calculations for the HPCI valves had considered that the TOLs would be in the circuit under design basis conditions.

PP&L engineers in the Allentown office provided documentation regarding the sizing of the TOLs and stated that the resistance of the TOL heaters had been accounted for in the MOV voltage drop calculations. The information provided shows that the current expected during valve seating and pullout can be withstood by the TOL for a duration sufficient to accomplish the safety function and still provide protection against thermal damage of the motor. The inspector noted that PP&L's MOVs are set up such that the thrust at torque switch trip is greater than the calculated value required under design basis conditions. The fact the MOVs are periodically stroked with the TOLs in service for IST testing provides reasonable assurance that the TOL circuitry is functional. Based on the information provided by PP&L, the inspector concluded that the operability of HPCI is not compromised during the RCIC surveillance and the licensee had performed an adequate technical evaluation of the condition.

5. PLANT SUPPORT (71750, 71707, 92904)

5.1 Radiological and Chemistry Controls

During routine tours of both units, the inspectors observed the implementation of selected portions of PP&L's radiological controls program to ensure: the utilization and compliance with radiological work permits (RWPs); detailed descriptions of radiological conditions; and personnel adherence to RWP requirements. The inspectors observed adequate controls of access to various radiologically controlled areas and use of personnel monitors and frisking methods upon exit from these areas. Posting and control of radiation areas, contaminated areas and hot spots, and labelling and control of containers holding radioactive materials were verified to be in accordance with PP&L



procedures. Health Physics technician control and monitoring of these activities was satisfactory. Overall, the inspector observed an acceptable level of performance and implementation of the radiological controls program.

5.2 Security

PP&L's implementation of the physical security program was verified on a periodic basis, including the adequacy of staffing, entry control, alarm stations, and physical boundaries. These inspection activities were conducted in accordance with NRC inspection procedure 71707. The inspector reviewed access and egress controls throughout the period. No deficiencies were identified.

6. SAFETY ASSESSMENT/QUALITY VERIFICATION (40500, 90700, 90712, 92700)

6.1 Licensee Event Report Review

The inspector reviewed LERs submitted to the NRC office to verify that details of the event were clearly reported, including the accuracy of the description of the cause and the adequacy of corrective action. The inspector considered whether further information was required from the licensee, whether generic implications were involved, and whether the event warranted onsite follow up.

Overview

The inspector found that eight LERs which require supplemental information have not been updated in a timely manner. The examples below are beyond the expected submittal dates by at least six months, and in most cases greater than a year.

Unit 1 and Common LERs:

91-015-00	High Pressure Coolant Injection system inoperable due to broken steam control valve pilot
93-007-00	Potential plugging of ECCS suction strainers by containment debris
93-008-00	Reactor scram following turbine trip on high vibration
94-012-00	Loss of fire detection / suppression
94-008-00	Class 1E 125 VDC, 250 VDC and 480 VDC load centers outside dynamic design basis
94-015-00	Postulated failures of SGTS components are outside of design basis

Unit 2 LERs:

93-008-00	Auxiliary load shed scheme for ESS busses surveillance missed
94-005-00	Instrument line excess flow check valve failure - TS required shutdown

The inspector acknowledged that complex problems can take longer to evaluate and correct than the 30 days allowed for reporting under 10 CFR 50.73, however, the number of long overdue supplemental LERs raises questions regarding the licensee's tracking of the issues.

The inspector questioned whether the late supplemental reports were the result of an administrative tracking problem or delays in the implementation of the corrective actions. The licensee was able to provide the current status of corrective actions for some of the issues, however, the status of other issues was not readily available. In response to the inspector's observation the licensee agreed to review the status of corrective actions related to the outstanding supplemental reports. Issues reported to the NRC under 10 CFR 50.73 typically involve a significant condition adverse to quality and for this reason the inspector was concerned about the timely implementation of corrective actions. Pending assessment of corrective actions for the conditions reported in the LERs with overdue supplemental reports and pending receipt of information from the licensee regarding their process for tracking supplemental LERs and implementation of corrective actions, this issue is unresolved. (URI 50-387;388/95-12-02)

The following LERs were reviewed, in accordance with the guidance provided in NRC Inspection Procedure 92700, and the licensee's corrective actions were considered adequate:

Unit 1

94-009-00 Isolation System Response Time Testing Not Completed

On June 7, 1994 the licensee discovered that the 18 month TS 4.3.2.3 requirement for response time testing of several functions (TS Table 3.3.2-1, functions 1.a.1, 7.a, 1.b, and 7.e) was not met due to the misapplication of a TS 4.3.2.3 allowance for testing redundant circuits during alternate outages. The trip functions in question do not have redundant channels. Once identified, the licensee completed the test within 24 hours allowed by TS 4.0.3. The licensee determined that the error occurred during a procedure format revision and has subsequently reviewed all Unit 1 and Unit 2 Time Response Tests that use the redundant channel allowance.

The safety significance of the missed time response testing of the non-redundant channels is low, as discussed in NUREG 1366, Improvements to Technical Specification Surveillance Requirements. The licensee has confirmed that all other surveillances that credit the redundant channel testing are, in fact, for redundant channels. The inspector agreed with the licensee's reportability analysis and considered the corrective actions adequate.

95-004-00 Personnel Access Hatch Surveillance Not Completed

During an initial containment entry while Unit 1 was in hot shutdown, the interior containment door interlock was not tested prior to containment entry. An oversight by operations personnel who had tested the outer door interlock, resulted in a missed surveillance of the inner door interlock. Procedure NDAP-QA-0309, Primary Containment Access Control, has been revised to include

two testing verifications, one for the exterior door interlock and the other for the interior door interlock. In addition, the procedure was also revised to ensure that an operator is part of the containment entry team.

Unit 2

94-003-01 'S' MSRV Acoustic Monitor Inoperable

The acoustic monitor for the "S" Main Steam Relief Valve (MSRV) was declared inoperable and could not be repaired without the Unit shutdown. Because of a power emergency, discretionary enforcement from the applicable Technical Specification was granted by the NRC. In addition, there were many other means available in the Control Room to determine if the MSRV had lifted. The acoustic monitor was repaired during the 6th refueling outage.

94-004-00 HPCI Inoperable Due To Broken CST Level Switch

During testing of a Condensate Storage Tank level switch a states link screw broke. This caused the level switch to be inoperable and consequently the High Pressure Coolant Injection (HPCI) System to be inoperable although the system could function manually if needed. The repair was made within the time frame of the limiting condition for operation and HPCI was declared operable.

94-008-00 Emergency Switchgear Room Cooling Inoperable

On May 25, 1994, both Unit 2 emergency switchgear room cooling subsystems were declared inoperable due to an intermittent control logic problem discovered during an 18 month loss of offsite power (LOOP) surveillance test. A "relay race" in the cooling system control logic was found to randomly result in a false motor high temperature signal. This false signal could have prevented the automatic restart of the system following a LOOP, however, manual initiation was still possible. The "relay race" occurred when power was restored following the simulated LOOP and was caused by millisecond differences in relay pick-up and drop-out times.

The licensee modified the control circuit to prevent a false signal from occurring upon re-energization and the system was successfully tested. The inspector agreed with the licensee's reportability determination and considered their corrective actions appropriate for the deficiency identified.

95-003-00 Shift Average Maximum Power Level Exceeded

On February 10, 1995, the licensee identified that a calibration error had been introduced during the units sixth refueling outage which included implementation of a power uprate. The error was the result of a calculation based on feed water flow instrument line locations that were depicted incorrectly in original plant drawings. The error resulted in exceeding the rated power over an eight hour shift by 0.024%, less than the instrument tolerance of 0.25%.

The inspector concluded that the LER contained the required information and that the licensee's corrective actions were appropriate for this event. The safety consequences of this error were minor and no previous occurrences were identified.

95-005-00 Reactor Scram Following Turbine Trip On Load Reject

NRC review of this event is documented in Inspection Report 50-387/95-08 and the inspector found the LER description of the event and corrective actions adequate.

95-008-00 Shift Average Licensed Core Thermal Power Exceeded

On June 9-10, 1995, instrumentation problems were discovered that effected the unit's heat balance calculation in a non-conservative direction. The combined impact of these instrument failures caused a 0.12% error in the indicated core thermal power.

The inspector concluded that the licensee's reporting this event was consistent with the NRC's August 22, 1980 guidance entitled "Discussion of Licensed Power Level." The LER description of the event and corrective actions were considered appropriate. The inspector found the licensee's root cause investigation to be thorough and the corrective actions comprehensive. No previous events were identified concerning instrument failures at SSES that effected core thermal power calculations.

6.1.1 Fire Protection LER Review

A separate review of fire protection related LERs was conducted to assess the adequacy of the corrective actions taken and to determine if any commonality exists among the issues.

In all of the LERs listed below the licensee has shown a good questioning attitude, and effected a thorough root cause analysis and corrective actions. The issues in the LERs result from weaknesses in licensee's past fire protection program and were discovered as a result of upgrades implemented by the licensee. The licensee's corrective actions were verified by the inspector on a sampling basis.

Unit 1

92-015-03 Fire Barriers Not Inspected and Not Installed

NRC Inspection 92-23 identified that Kaowool fire rated barriers were not being inspected as part of the licensee's surveillance program. This was due to an oversight during procedure development and drawing inaccuracies. Because of generic industry concerns regarding Thermo-Lag and Kaowool, the licensee declared these fire barriers inoperable in 1992 and has posted appropriate fire watches. The licensee stated that they will continue with fire watches until the generic issues with these fire barriers are resolved

between the NRC and the industry in general. Fire wrap issues identified by Inspection 92-23 which have not been closed are included in the NRC's inspection followup system.

94-003-00 Fire Barrier Not Sealed; Fire Watch Not Performed

An electrical conduit which penetrates from the lower cable spreading room to the north cable chase was discovered to be not sealed as required. Upon discovery of this condition an appropriate fire watch was established. The apparent cause was that the construction drawing erroneously identified the penetration as embedded rather than through wall. The penetration has since been sealed and the drawing corrected. In addition the licensee's Appendix R Project has performed a 100% inspection of all Technical Specification fire rated penetration seals which has established a base line for the penetration seal inspection program.

94-004-00 Fire Protection Damper Not Installed in a Fire Rated Barrier

During a surveillance of fire dampers, a fire damper which should have been installed in the Control Building Computer Room duct work was found to be not installed. This was apparently an original construction deficiency as the drawings show the existence of a damper. This was not discovered in previous inspections as another damper located nearby in the same duct work was mistaken for the damper in question. The licensee has established a roving fire watch for the area until a damper can be installed. In addition, all fire dampers have been labeled to assure proper identification for surveillance inspections. The licensee plans to install a damper as per the existing drawing by September 1995.

94-006-00 Halon Fire Protection for Panel Inoperable With the Panel Door Left Open

A panel door to an instrument panel located in the Control Building was left open after installation of test equipment which rendered the Halon protection for that panel inoperable. After the condition was discovered, the test equipment was removed and the door closed. As permanent corrective action, the procedure NDAP-QA-0441, Fire Protection System Status Control, which governs fire protection status control, was revised to include guidance as to requirements for enclosures to Halon protected equipment. Doors for the Halon protected cabinets have been marked to clearly identify the Halon protection and that special permission is required to leave the doors open. In a separate corrective action, all fire doors in the plant have had large signs painted on them clearly identifying them as fire doors and not to be blocked open.

94-007-00 and 01 Inoperable Fire Barrier

In the course of investigating a penetration problem, an opening was noticed by an engineer in a penetration between the drywall and Q-Decking in the Lower cable spreading Room. Although the penetration was sealed, air flow was detected from the penetration, indicating an incomplete seal. The condition had existed since original construction and was not previously detected

because the area is almost inaccessible for close up physical inspection. The gap was sealed and similar ceiling joint penetrations were also inspected to verify similar conditions did not exist. The fire barrier inspection procedure, SE-013-007, 18 Month Inspection of Unit Common Fire Barriers, was revised and significantly upgraded to include more specific inspection criteria and providing means to inspect penetrations which are difficult to access.

94-016-00 Fire Watches Not established Within One Hour After the Loss of Fire Detection and Suppression

One channel of the Simplex Fire Protection System failed when a transponder card in the system failed due a lightning strike power surge. The licensee was not able to establish the appropriate number of fire watches within the one hour time frame. As a result of this event, procedure OI-AD-013, Simplex Problem/Failure Response, has been significantly upgraded to ensure that fire watches can be established in a timely manner when automatic fire detection and suppression is lost. Also, many persons are given fire watch training so that sufficient personnel will be available on each shift to man emergency fire watches until regular fire watch personnel can be called in. Backshift drills have been successfully conducted on March 25, 1995 and July 12, 1995, to test the procedure. Other drills are scheduled until all backshifts are covered. The licensee is also taking actions to upgrade their Simplex Fire protection system to make it less vulnerable to power surges.

95-001-00 Fire Barrier Not Sealed and Fire Watch Not Established

A penetration was installed in the computer room in 1991. The penetration was not recognized as a fire barrier and was left unsealed and no fire watch was established. In July 1992, independent of the above issues, fire watches had been established in the above area due to Thermo-Lag fire barriers being declared inoperable at that time. These fire watches are still in effect. The discovery of the unsealed fire barrier was due to improved fire barrier surveillance procedures. In 1992, as a result of a reorganization, the Conduit, Raceway, and Tubing (CRT) group, was incorporated into the site design group. This allowed for a more stringent review of plant penetration work. Procedure MFI-5210, "Electrical Cable Routing and Raceway Design and Installation Instructions", has been established, and in part, requires an Appendix R evaluation as part of any new wall penetration. Also, Procedure MFP-QA-2308, Design Inputs and Considerations, contains checklists which include Appendix R and fire protection considerations. The licensee plans to seal the penetration at a future date.

95-002-00 Fire Watch Not Completed Within One Hour Time Frame

A firewatch inspection was performed 15 minutes in excess of the required time frame. The individual performing the fire watch became stuck in an elevator. Although this individual was in contact with the Control Room, the communication did not make it clear that a surveillance time constraint would be missed. There was also incomplete communication between the fire watch supervision and operations. Personnel were reinstructed on proper

communications. Also, extra fire watch logs have been placed in the Control Room and the fire watch trailer should a fire watch become delayed in the future.

Unit 2

94-006-00 Fire Rated Elevator Doors Not Inspected

As result of an engineering review for the licensee's fire protection drawing upgrade project, it was determined that certain elevator doors in the Control Building and the Reactor Building were fire rated but had not been included in the surveillance program. The doors were inspected and found to be acceptable. The fire door inspection procedure, SE-013-008, Six Month Inspection of Unit Common Fire Doors, has now been revised to include applicable elevator doors.

95-004-00 Two Fire Zones Not Inspected During 18 Month Surveillance Inspection

Two fire zones located in high radiation areas were not inspected while performing the required 18 month surveillance inspection. Although this problem had been recognized by procedure writers, it was omitted from procedures due to an incomplete turnover of procedure responsibility between two individuals. The inspections have been performed. The licensee has revised the surveillance procedure SE-213-007, 18 Month Inspection of Unit 2 Fire Barriers, to ensure that inaccessible fire zones are inspected when they become accessible.

Units 1 and 2

94-010-00 Access Hatches Between Fire Zones Not Fire Rated

During an inspection for the damaged fire penetration seals, access hatches in both the Unit 1 and Unit 2 Upper Relay Rooms were not fire rated. These hatches, because of their location, are difficult to access. This condition has existed since original construction. Roving fire watches have been established for these rooms. The licensee has not yet made a decision on the repair of the hatches. They were not designed to be fire barriers and may not be able to be modified to be fire rated. In addition, installation of new hatches may be difficult. The licensee is evaluating permanent corrective action. However, the fire barrier 18 month surveillance procedure (SE-13-007) has been significantly upgraded to inspect these types of locations.

6.2 Open Item Followup

(Closed) URI 50-387/91-18-02 Service Water, Relief Valve Failure - Common

On October 14, 1991, a service water safety relief valve (PSV 21022A) on the reactor building close cooling water (RBCCW) heat exchanger failed its setpoint and seat leakage tests. The licensee found that the valve was

clogged with rust and a plastic dust cap, which had apparently been left inside the valve since initial installation during plant construction. The issue was left unresolved pending licensee's evaluation.

The subject valve is one of seventy two heat exchanger thermal relief valves at SSES. The valves are designed to prevent internal damage of heat exchangers if the cooling medium is isolated without a corresponding isolation of the hot side process fluid. In response to the failure of PSV 21022A, the licensee sampled eight additional valves to determine if dust caps had been left inside any other valves during installation (prior to initial startup). No caps were found in the eight valves, however, rust accumulation was found which did impact their actuation pressure. Based on this finding the licensee began replacing all raw water system thermal relief valves as part of the preventive maintenance (PM) plan for the heat exchangers. The replacement valves have soft seats and stainless steel bodies that are expected to improve reseal leakage problems and reduce internal rust accumulation.

Eight of the thermal relief valves identified by the licensee are safety related, providing protection for the RHR service water heat exchangers on both units. The licensee's preventive maintenance activities for these valves implement the inservice testing requirements of ASME OM-10. The inspector concluded that the OM-10 testing provides assurance that these valves are not being degraded by rust (or other muck from the spray pond) and adequately addresses this issue. The licensee's development of a PM and replacement of the thermal relief valves for balance of plant and non-safety systems should improve the reliability of thermal reliefs in those systems. Based on the licensee's actions to evaluate the scope of the problem, the safety impact of the problem, and the long term issue, the inspector considered this item closed.

(Closed) URI 50-387/91-18-03 Secondary Containment Damper Failure

This unresolved item concerns a secondary containment damper that failed to stroke during a quarterly surveillance test. The failure was attributed to the malfunction of the solenoid valve that vents air from the damper's actuator. The solenoid valve was replaced as corrective action for the failure. Also, at the time of the failure an outstanding work authorization existed for an air leak on the solenoid valve. The inspector was concerned that the licensee had not performed a thorough root cause for this solenoid valve failure, and had not documented an operability evaluation for the continued use of a degraded safety-related component prior to its failure.

In the fall of 1991, a number of solenoid valve failures due to coil problems were under investigation (S00Rs 1-87-221, 1-90-099, 1-91-022, and 1-91-024). However, the failure of solenoid valve SV-17524B was different than the previously identified problems in that the coil had not failed. After replacing the valve, the mechanics found that the solenoid operated freely during bench testing. The licensee concluded that the failure was most likely attributable to dirt inhibiting movement of the popet and that the dirt was dislodged during removal of the valve. The licensee had recently addressed the potential for dirt intrusion hampering components supplied from the instrument air system by installing high efficiency particulate filters.

The inspector reviewed deficiency reports (S00Rs and NCRs) for solenoid valve failures issued since 1991 and did not identify any subsequent failures of safety-related solenoid valves due to debris intrusion. This information supported the licensee's position that it was a unique occurrence and that in-line filters would prevent future problems.

Documentation of operability evaluations is required for plant non-conformances identified under the Condition Report (CR) process. NDAP-QA-703, Operability Determinations, issued in March 1995 provides the guidelines for the licensee's assessments and requires documentation of them as part of the permanent CR record. NRC Inspection Report 50-387/95-80 discusses the CR process in greater detail.

The inspector considered the licensee's corrective action acceptable for the circumstances. The actions to prevent recurrence (in-line filters) and absence of similar solenoid valve failures since that time support the licensee's assessment of the SV-17524B failure. The inspector agreed that the failure of SV-17524B was different than the coil related failures and considered the current process for reporting deficiencies and documenting operability evaluations an improved process. Based on this review and assessment, the unresolved item is closed.

(Closed) URI 50-387/92-02-01 Industry Event Review (IER) Program Weaknesses - Common

This item was left unresolved following a hydrogen ignition and contamination injury on January 18, 1992. The inspector was concerned that the licensee's industry experience review program may not have been adequate as it did not address certain generic correspondence related to the issue of flammable gas explosion.

An update and partial resolution of the unresolved issue was discussed in Inspection Report 50-387/95-08 which addressed how the licensee evaluated the effectiveness of their program. The inspector left this item open pending licensee's clarification on disposition of Generic Letter (GL) 91-03, Reporting of Safeguards Events which lists listed examples of safeguards events that do not need to be reported promptly to the NRC Operations Center. The licensee identified that although an internal correspondence indicated that the GL information was to be incorporated into the next revision of the procedure governing NRC notification (AD-QA-424), this never occurred. The licensee's failure to incorporate this information resulted in a prompt notification that was made to the NRC on May 8, 1995 regarding a safeguards event. The notification was not required. The licensee has subsequently incorporated the GL into the reporting procedure NDAP-QA-0720.

The inspector noted that an NRC review of the licensee's IER program was recently documented in Inspection Report 50-387/95-80. The inspection report concluded that, in general, PP&L seemed to successfully deal with industry experience information and that NRC information documents are promptly dealt with. Based on this review and the review discussed in Inspection Report 50-387/95-08, the inspector concluded that the licensee's IER program was acceptable. This item is closed.

(Closed) URI 50-387/90-03-01, One Hour Fire Watches Not Established

This item is being administratively closed based on the review of the Unit 1 LER 94-016-00 and because the issue will also be addressed during NRC review of a related violation (VIO 50-387/94-16-01).

(Closed) URI 50-388/92-29-03, Inoperable RWCU High Flow Isolation Channel

This unresolved item identified a number of weaknesses regarding the licensee's process for requesting enforcement discretion from the NRC. Specifically, the licensee's request for enforcement discretion to allow continued operation of the reactor water cleanup (RWCU) system with the 'B' RWCU high flow channel inoperable did not identify that certain portion of the RWCU piping run was not adequately monitored for line break. Hence, the compensatory measures proposed by the licensee were not adequate. The licensee established an Event Review Team (ERT) to fully evaluate their performance, the this item was left unresolved pending completion of ERT review.

The inspector discussed the results of the ERT review with the licensee. Following the results of the ERT review a document titled Nuclear Regulatory Affairs Guidelines Requests for Enforcement Discretion was prepared. However, the ERT was disbanded before issuance of a report, hence, an ERT report was not available.

The inspector reviewed the guidelines document and noted that this document was not required to be used or referenced in a related nuclear department procedure. The inspector pointed out that without the guideline document being referenced in a nuclear department procedure, the use of the document can not be guaranteed. The licensee committed to incorporate the guidance in a nuclear department procedure, and expected to complete it by the third quarter of 1995.

The inspector noted that recently the licensee submitted a request for enforcement discretion to the NRC for Unit 2 B-excore neutron monitoring system and was considering one for the Unit 2 D-RHR pump. In both of these cases, the licensee's process was timely and the safety assessment (which for the second case writeup was still in draft form) addressed appropriate safety and risk considerations and were of good quality. Based on this review, the unresolved item is closed.

6.3 Corrective Action Process Observation

The inspectors periodically monitor the implementation of the licensee's corrective action process by reviewing condition reports (CRs) and actions taken in response to them. The inspector observed that the two instrument failures effecting core thermal power calculations reported in the Unit 2 LER 95-008-00, were preceded by a failure which had been captured by the condition report process. The LER described failures were identified on June 9-10, 1995, during the implementation of corrective actions for the first event that had been documented in a CR on May 2, 1995. The inspector observed that neither the immediate corrective actions taken by Operations nor the CAT

directed actions specifically addressed timely verification of the other instrument providing input to heat balance calculation. A comparison of the heat balance calculation computer inputs for the two units would have highlighted the two additional instrument errors much sooner. Although the CR process and Corrective Action Team represent a significant improvement, the process is still evolving and plant management is continuing efforts to further improve and refine the program.

7. MANAGEMENT AND EXIT MEETINGS (71707)

7.1 Resident Exit and Periodic Meetings

The inspector discussed the findings of this inspection with PP&L station management throughout the inspection period to ensure timely communication of emerging concerns. At the conclusion of the reporting period, the resident inspection staff conducted an exit meeting summarizing the preliminary findings of this inspection. Based on NRC Region I review of this report and discussions held with licensee representatives, it was determined that this report does not contain information subject to 10 CFR 2.790 restrictions.

7.2 Other NRC Activities

A meeting was held on May 16, 1995, in King of Prussia, Pennsylvania between NRC Region I inspectors and licensee representatives to discuss 1) the upcoming full participation emergency preparedness exercise, 2) corrective actions for past siren failures, and 3) the licensee's action plan to address Emergency Plan awareness issues discussed in a recent inspection report (50-387/95-08).

The following region based NRC inspection activities took place during this period:

<u>Dates</u>	<u>Report No.</u>	<u>Inspection Procedure</u>	<u>Lead Inspector</u>
June 5 - 9	95-14	64704, Fire Protection	Harrison
May 29 - June 9	95-13	37550, Engineering	Dempsey

