UNITED STATES NUCLEAR REGULATORY COMMISSION

REGION I

Inspection Report Nos.	50-387/94-11; 50-388/94-12
License Nos.	NPF-14; NPF-22
Licensee:	Pennsylvania Power and Light Company 2 North Ninth Street Allentown, Pennsylvania 18101
Facility Name:	Susquehanna Steam Electric Station
Inspection At:	Salem Township, Pennsylvania
Inspection Conducted:	May 24, 1994 - July 18, 1994
Inspectors:	G. S. Barber, Senior Resident Inspector, SSES D. J. Mannai, Resident Inspector, SSES F. L. Bower, III, Reactor Engineer

Approved By: ______________________ 8/1/94

J. White, Chief Date
Reactor Projects Section No. 2A,

Inspection Summary: This inspection report documents routine and reactive inspections (during day and backshift hours) of station activities, including: plant operations; radiation protection; surveillance and maintenance; engineering and technical support; and safety assessment/quality verification. Findings and conclusions are summarized in the Executive Summary. Details are provided in the full inspection report.

EXECUTIVE SUMMARY

Susquehanna Inspection Reports

50-387/94-11; 50-388/94-12

May 24, 1994 - July 18, 1994

Operations (71707, 92901, 93702)

During the period, the inspector reviewed the response of the High Pressure Coolant Injection (HPCI) and the Reactor Core Isolation Cooling (RCIC) system response to a postulated High Energy Line Break. Specifically, the expected response of suppression pool suction valves in response to the HELB. Both systems were found to respond in an acceptable manner. Section 2.2.1 pertains.

During the period, an automatic reactor water cleanup (RWCU) isolation on high differential flow occurred. The licensee determined the event constituted an engineered safety feature (ESF) actuation and was reportable. Operators responded to the event promptly and determined the cause. Investigation revealed a valve used for maintenance blocking unseated when system pressure overcame valve spring pressure and allowed leakage flow through open drain valves. Section 2.2.2 pertains.

Engineering/Technical Support (37551, 92903)

During the period, operators discovered a modification design error when the RHR pump discharge check valves failed to close, as required, on two separate occasions. Following the first self-disclosing event on May 29, an Event Review Team was formed. The team identified that a vendor design error was made and was not detected during the modification process relative to a modification to safety-related check valves in the RHR system. The issue remains unresolved pending final licensee corrective actions. Section 4.2.1 pertains.

On June 22, at 9:30 a.m., a significant vibration of the Unit 2 reactor building occurred after recirculation flow was increased from 100 Mlbm/hr to 108 Mlbm/hr, as allowed by the approved power uprate for the unit. The licensee also identified excessive vibration affecting containment instrument gas (CIG) piping and a suppression pool hatch cover. When reactor recirculation (RR) pump speed was lowered to 1515 rpm, (about 100 Mlbm/hr) the vibrations subsided. The licensee took the following prompt actions to address the event: RR pump speed was limited to 1515 rpm, a hold was placed on further power ascension testing, and a task team was formed to investigate. The licensee believes that the vibration was caused by a vane passing phenomena involving hydraulic pulses generated as each pump vane expels a volume of water. The licensee completed their preliminary assessment on June 25 and concluded that there were no operability issues. The NRC is continuing to evaluate the licensee's actions. A September 1994 management meeting is planned. Section 4.2.2 pertains.

Safety Assessment/Assurance of Quality (40500, 90712, 92700, 92701)

Several open items were reviewed regarding the Fire Protection Program performance. Two previous violations and one unresolved item were closed. Two previous violations were updated. Section 6.1 pertains.

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1. SUMMARY OF OPERATIONS

1.1 Inspection Activities

The purpose of this inspection was to assess licensee activities at Susquehanna Steam Electric Station (SSES) as they related to reactor safety and worker radiation protection. Within each inspection area, the inspectors documented the specific purpose of the area under review, the scope of inspection activities and findings, along with appropriate conclusions. This assessment is based on actual observation of licensee activities, interviews with licensee personnel, independent calculation, and selective review of applicable documents.

1.2 Susquehanna Unit 1 Summary

At the start of the report period, Unit 1 was operating at approximately 100% of rated power. On June 10, all 112 emergency offsite notification sirens were inoperable from 9:30 a.m. to 12:00 noon. The system was lost when the local telephone company performed work on telephone lines. A one hour 50.72 ENS phone call was made to the NRC. Emergency Planning personnel also notified state and local emergency management agencies. The licensee documented the problem on Significant Operating Occurrence Report (SOOR) 94-377 and began an investigation. The system was restored at the time of NRC notification. The inspector will evaluate the licensee's corrective actions as a part of a future inspection.

On July 7, the Reactor Water Cleanup (RWCU) system automatically isolated due to high differential flow. The event constituted an Engineered Safety Feature (ESF) actuation. Section 2.2.2 pertains. On July 8, power was reduced to 60% of rated power for a planned control rod sequence exchange and condenser water box cleaning. Power was subsequently returned to 100% on July 9. The reactor remained at 100% power through the end of the report period with the exception of several small power reductions to maintain condenser back pressure within acceptable limits.

1.3 Susquehanna Unit 2 Summary

At the start of the report period, Unit 2 was in Condition 5 in a refueling outage. On May 25, during Diesel Generator LOCA/LOOP testing, the A & B Emergency Switchgear Room Cooling Subsystems were declared inoperable due to the discovery of an improperly wired relay associated with the direct expansion (DX) unit. The relay problem affected the automatic start function of the system intermittently. The relay was replaced. The licensee determined the condition reportable. The inspector will perform final NRC assessment during the LER review process.

Vessel reassembly began on May 24, and Condition 4 was entered on May 28 when the reactor pressure vessel head was fully tensioned. On May 29, after operation of 'B' loop of RHR for testing purposes, keepfill pressure could not be established. Investigation revealed both discharge check valves stuck full open causing the RHR loop to drain to the suppression pool. On May 31, after securing 'A' loop of RHR from shutdown cooling, the 'A' RHR pump discharge check valve failed to close. All three valves required modification to be operable. Section 4.2.1 pertains.

Condition 2 was entered and reactor startup commenced at 11:51 p.m. on June 7. Condition 1 was entered on June 9 at 2:27 p.m. On June 13, during scram time testing control rod 46-19 drifted from position 32 to full out (position 48). Operators reset rod drive control system (RDCS) and exercised the control rod in single notch and continuous withdrawal modes. The inspector will evaluate the licensee's corrective actions after the SOOR resolution is complete.

On June 21, core flow was increased from 100 to 108 million lbs/hr to support power uprate testing. On June 22, with core flow at 108 Mlb/hr, vibration was identified in the reactor building. Following an investigation, the vibration was arrested when reactor recirculation pump speed and core flow were reduced to 1515 rpm and 100Mlb/hr, respectively. Section 4.2.2 pertains.

Power was increased to 100% (core flow at 100 Mlb/hr) on June 26 at 1:26 p.m. to the new licensed rated thermal power limit of 3441 MWT for power uprate. The reactor remained at 100% power through the end of the inspection period except for minor power reductions to maintain condenser back pressure.

2. OPERATIONS

2.1 Inspection Activities

The inspectors verified that the facility was operated safely and in conformance with regulatory requirements. Pennsylvania Power and Light (PP&L) Company management control was evaluated by direct observation of activities, tours of the facility, interviews and discussions with personnel, independent verification of safety system status and Limiting Conditions for Operation, and review of facility records. These inspection activities were conducted in accordance with NRC inspection procedure 71707.

The inspectors performed 9.5 hours of deep backshift inspections during the period. These deep backshift inspections covered licensee activities during between 10:00 p.m. and 6:00 a.m. on weekdays, and weekends and holidays.

2.2 Inspection Findings and Review of Events

2.2.1 HPCI and RCIC Suppression Pool Suction Valve Response to a Postulated High Energy Line Break

The inspector reviewed the response of the High Pressure Coolant Injection (HPCI) and the Reactor Core Isolation Cooling (RCIC) Suction Valves to a High Energy Line Break (HELB). This review was performed to ensure that appropriate containment isolation function was provided during the suction swap over from the condensate storage tank (CST) to the suppression pool. For HPCI, the suppression pool suction valves (HV-E41-1F042 and HV-E41-2F042) for both units respond similarly. If a HPCI start signal is received, the HPCI turbine would accelerate up to speed and begin injecting water into the reactor core. The initial HPCI suction is from the condensate storage tank. When the CST reaches low level, an automatic suction swap over is initiated from the CST to the suppression pool. If a HELB occurred in the HPCI room during this time period, then a HPCI automatic isolation signal would be initiated. This would cause the appropriate suppression pool suction valve to cease opening, while simultaneously inserting a close signal within the valve logic. Thus, a containment isolation signal from a HELB would override the automatic suction swap over.

The RCIC suppression pool suction valves for both units (HV-E41-1F031 and HV-E41-2F031) have no automatic containment isolation function. They do have remote isolation capability and, per Final Safety Analysis Report (FSAR) Table 6.2-12, meet General Design Criteria 56, Primary Containment Isolation. There is no specified closure time nor is automatic isolation of this penetration required. These valves are designed to operate in a harsh environment resulting from a HELB. A manual closure signal would override the automatic suction swap over feature on low CST level.

The inspector reviewed prints, the FSAR, and design basis documents to verify the automatic or manual closure capability of the HPCI/RCIC suppression pool suction valves. The current design assures adequate containment isolation under these conditions. The inspector had no further questions.

2.2.2 Automatic Reactor Water Cleanup Isolation

On July 7, 1994, at 9:47 a.m., the Unit 1 Reactor Water Cleanup (RWCU) system automatically isolated on high differential flow. The licensee determined the automatic isolation constituted an Engineered Safety Feature (ESF) actuation. The operators confirmed the system functioned as required. The proper 10 CFR 50.72 NRC notification was made. SOOR 94-417 documented the event.

The licensee promptly determined the cause of the isolation. During the time of the event, a blocking permit had been hung to support RWCU system maintenance. The licensee's investigation determined that prior to the authorization for maintenance, an air operated valve (AOV) used for isolation leaked when full system pressure overcame spring pressure. This unseated the valve seat which allowed flow into the isolated portion of the system and, consequently, out through open drain valves. The flow was sufficient to cause an automatic isolation of the RWCU system due to high differential flow. Nuclear Department Administrative Procedure, NDAP-QA-323, Standard Blocking Practices, was subsequently changed to specifically prohibit the use of this valve for blocking. The Maintenance Department Valve Team began a review to identify other plant locations where this type valve is utilized. Following completion of activities, the NDAP will be generally revised to preclude the consideration of valves of that type for establishing blocking boundaries.

As a result of the leakage, a portion of the reactor water cleanup holding pump room (about 150 $ft.^2$) was contaminated in the proximity of the liquid radwaste floor drain. One operator became slightly contaminated when manually closing the draining valves in response to the event. Health Physics generated an Area Contamination Report (ACR) and a Personal Contamination Report (PCR). Operators returned the system to service following decontamination efforts of the RWCU holding pump room.

The inspector concluded the operators properly responded to the RWCU isolation. The licensee promptly determined the the cause to be insufficient planning in support of maintenance activities affecting the RWCU system. The inspector found immediate licensee corrective actions were adequate. From discussion with maintenance personnel, this was the first time the valve was used for blocking to isolate full system pressure from a depressurized and vented maintenance boundary. Safety consequence was minimal since the system isolated per design and no personnel were injured or significantly contaminated. The inspector had no further questions.

3. MAINTENANCE/SURVEILLANCE

3.1 Maintenance Inspection Activity

On a sampling basis, the inspector observed and reviewed selected maintenance activities to ensure that specific programmatic elements described below were being met. Details of this review are documented in the following sections.

3.2 Maintenance Observations

The inspector observed and/or reviewed selected maintenance activities to determine that 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.

These observations and/or reviews included:

-- WA H30006, Remove Cattle Chute to Support Reactor Reassembly, dated May 25.

-- WA H30022, Install Steam Separator to Support Reactor Vessel Reassembly, dated May 25.

-- WA V23119, Disassemble/Inspect/Reassemble RHRSW Pump '2A' for 5 Year Inspection, dated June 29.

3.3 Surveillance Observations

The inspector observed and/or reviewed the following surveillance tests to determine that 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 was accurate and complete; removal and restoration of the affected components was 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.

These observations and/or reviews included:

-- SE-224-A02, 18 Month Diesel Generator 'A' Autostart and ESS Bus '2A' Energization on Loss of Offsite Power - Plant Shutdown, dated May 24.

-- SO-156-001, Control Rod Weekly Exercising for Operability, dated May 27.

-- SO-251-002, Quarterly Core Spray Flow Verification, dated June 27.

3.4 Inspection Findings

The inspector reviewed the listed maintenance and surveillance activities. The review noted that work was properly released before its commencement; that systems and components were properly tested before being returned to service and that maintenance activities were conducted properly by qualified personnel. Where questionable issues arose, the inspector verified that the licensee took the appropriate action before system/component operability was declared.

4. ENGINEERING/TECHNICAL SUPPORT

4.1 Inspection Activity

The inspector periodically reviewed engineering and technical support activities during this inspection period. The on-site Nuclear Systems Engineering (NSE) organization, along with Nuclear Technology in Allentown, provided engineering resolution for problems during the inspection period. NSE generally addressed the short term resolution of engineering problems; and interfaced with the Nuclear Modifications organization to schedule modifications and design changes, as appropriate, to provide long term corrective action. The inspector verified that problem resolutions were thorough and directed at preventing recurrence. In addition, the inspector reviewed short term actions to ensure that they provided reasonable assurance that safe operation could be maintained.

4.2 Inspection Findings

4.2.1 Residual Heat Removal (RHR) Pump Discharge Check Valve Modification Problem

On May 29, during restoration from diesel generator testing (SE-224-207), while securing the Unit 2 'B' loop of RHR, discharge check valves for the 'B' and 'D' RHR pumps stuck full open resulting in the draining of the 'B' loop of RHR. These swing check valves are expected to close whenever their respective RHR pump is secured. The unit was in Condition 4, cold shutdown, at the time of the event. This was discovered when operations, while checking the system filled and vented, detected only air at the high point vent. Operations declared the 'B' loop of RHR inoperable and entered Limiting Condition for Operation (LCO) Action Statement for Technical Specification 3.4.9.2 which requires two loops of RHR be available for shutdown cooling. The licensee documented this self-disclosing event on SOOR 94-356 and formed an Event Review Team (ERT).

Two days later the 'A' RHR pump was secured from shutdown cooling. Discharge pressure decreased below keepfill pressure. The 'A' discharge check valve was discovered full open. The inspector noted that all of the Unit 2 RHR pump discharge check valves had been modified during the current Unit 2 regueling outage, and that this condition did not exist previously.

The ERT performed a safety assessment for the RHR pump discharge check valves failing to close when the pumps were shutdown. The safety assessment concluded the condition would not have prevented RHR injection. However, if the RHR pumps were started with the system partially drained, a water hammer could result and cause subsequent damage to the system. Since the drain down occurs only when securing, the system's existing procedures are expected to prevent damage to the RHR system in the event the system needs to be manually restarted. The ERT also considered a scenario where the pump would be shutdown and then automatically restart. This possibility would exist if the system had been in service for testing of suppression pool cooling and a simultaneous LOCA-LOOP occurs. This scenario was previously analyzed during licensee review of potential effects of a water hammer event. The licensee concluded the probability of the event was low. In response to the previous analysis, existing operating procedures limit operation to one loop of RHR in suppression pool cooling or full flow test modes. Additionally, the licensee administratively limits the number of hours the RHR system is operated in this mode.

The ERT identified that the valves failed to close due to binding between the valve disc and valve body. The binding was due to the bonnet stops being too short which allowed excessive disc travel. The ERT identified the root causes to be vendor design error and also noted the modification review process was not sufficient to detect the vendor design error.

The vendor sized the disc stops incorrectly. The stops were short by two inches. PP&L engineering guidance regarding modifications did not specify performing installation checks on vendor supplied subcomponents for critical clearances. The licensee added disc stop extensions and performed clearance checks to ensure no binding would occur at full open. Testing showed the modified check valves passed design flow with normal pressure drop, and stroked and closed properly. The ERT findings were presented to Plant Operations Review Committee (PORC) on June 3. The immediate corrective actions were found acceptable for purposes of Plant Startup. However, PORC directed the ERT to revise and strengthen the actions to prevent recurrence.

The inspector independently reviewed the licensee's safety assessment and concluded the actual safety impact was minimal, but the deficiency in the licensee's modification process to allow such a design error in a safety-related component to go undetected had potential safety significance. The immediate actions to correct the hardware problem were considered adequate. Actions to prevent recurrence were still being evaluated by the licensee at the conclusion of the report period. The issue of proper safety-related modification implementation remains unresolved pending inspector review of final licensee evaluation and corrective actions. (URI 50-387/94-11-01)

4.2.2 Excessive Vibration due to Elevated Reactor Recirculation Pump Speeds

On June 21, at 1:30 p.m., operators raised Unit 2 reactor recirculation (RR) flow to 108 Mlbs/hr for power uprate testing. On June 22, at approximately 8:30 a.m., a planned feedwater system tuneup began. At approximately 9:30 a.m., with RR pump speed at 95% (1570 - 1580 rpm), operators observed vibration in the containment instrument gas (CIG) piping outside containment. They also noted a rhythmic hum from inside the Unit 2 reactor building. Additionally, vibration probe (3Y) mounted on the pump shaft of the RR pumps on the same axis as the pump discharge showed an unexplained increase. Subsequent engineering walkdowns noted the vibration appeared to be coming from the primary containment structure. The suppression pool (SP) hatch covers were also vibrating. At that time, the licensee attributed the excessive containment, reactor building and piping vibration to the elevated RR pump speed. Subsequently, the licensee developed a plan to lower RR pump speed while monitoring for any additional vibrations. At 2:30 a.m., June 23, RR pump speeds were lowered. When both pump's speeds were at 1515 rpm, vibration and noise levels returned to normal.

As a result of the event, the licensee took prompt action to assess the effects of the vibration and to reduce the likelihood of future vibration events. First, RR pump speed was administratively restricted to 1515 rpm (equivalent to about 100 Mlb/hr, core flow). This was the point at which the vibrations subsided to normal levels. Second, a formal hold was placed on additional power ascension testing until management evaluation of the effects of the vibration event could be completed. Third, a task team was formed to determine the cause of the event and to recommend short and long-term actions.

Licensee review of previous industry experience noted a similar problem at Hope Creek (HC) on September 18, 1987 (OE 2255). In this case, excessive vibration caused two leaks to develop on RR flow instruments that were subsequently repaired. In addition, two vent line failures were attributed to this excessive vibration.

The licensee attributed the excessive vibration event at Susquehanna to vane passing phenomena that resulted from increasing RR pump speed to accommodate the core flow required for power uprate. The RR pumps are five vane Byron Jackson centrifugal pumps. As each vane passes the opening at the pump discharge it expels a volume equivalent to the height multiplied by area between the vanes. This results in a series of transmitted hydraulic pulses equivalent to RR pump speed. If the transmitted frequency of these hydraulic pulses is the same as the natural frequency of some attached or internal piping or component, a natural resonance is matched. The displacement amplitude and acceleration of this resonance vibration will be dependent on the energy transmitted through the system. In general, lower frequencies (<60Hz), transmit more energy than higher frequencies (>100Hz). Vibrations at a frequency of 131Hz would be considered high frequency and would have a lower energy content. Thus, transmitted stresses and strains will be low and the resultant vibration becomes more of concern from a high cycle fatigue standpoint.

In this case, a vibration was detected that corresponded to a unique harmonic of RR pump speed between 1570 and 1580 rpm or a vane passing frequency of approximately 131 Hz. Measurements at the SP hatch covers on the evening of June 22 confirmed vibration frequency of 131 Hz. The northwest SP hatch cover was vibrating at that frequency with a peak acceleration of 0.6 g's (acceleration of gravity). After RR speed was reduced below 1515 rpm, the amplitude decreased to 0.02 g's. The measured frequency of vibration observed at the SP hatch covers corresponded to the vane passing frequency of the RR pumps at the time of measurement. It also correlated well with the calculated natural frequency of the hatch covers verified during previous testing.

The licensee's task team performed an immediate assessment of the excessive vibration on plant operation. Specifically, the licensee attempted to determine the effects of operating for 18 hours in this condition. In their June 25 engineering assessment report, the licensee concluded that the vibration event did not result in any damage to any equipment, systems, structures or components; that operation at the current levels poses no threat to nuclear safety; and that continued power operation up to the levels specified in the assessment report was acceptable from a nuclear safety perspective. The bases for the licensee's conclusions are summarized below:

- The event experienced at Unit 2 is very similar to events experienced on numerous occasions at Hope Creek.

- Except as previously noted, physical inspections performed by plant personnel at Hope Creek during outages subsequent to such events have not uncovered any damage as a result of these events to equipment, systems, structures or components.

- The licensee Nuclear Technology group's assessment of the cause of the event at SSES indicates that the physical characteristics of the Reactor Recirculation System causes an amplification of the RR pump vane passing frequency at pump speeds on the order of 1570-1580 RPM. This amplification causes a pressure wave with a frequency equal to the vane passing frequency to be dispersed throughout the Reactor Vessel and, ultimately, into other attached structures.

- GE has indicated that they are aware of this condition relative to the vessel and its internals. GE's assessment is that it poses no threat of damage to the vessel or its internals.

- The Nuclear Technology group's assessment and engineering judgement of the effect of these high frequency loads on the equipment, systems, structures and components (SSCs) outside of the vessel is that only SSCs items with frequencies close to the vane passing frequency have any susceptibility to additional amplification and potential damage. Since the vast majority, if not all, of the equipment, systems, structures and components at SSES do not have fundamental frequencies even close to the vane passing frequency, little, if any, additional amplification will result and the potential for damage is remote.

The inspector reviewed the licensee task team's conclusion, and interviewed personnel to ascertain the adequacy of the licensee's evaluation of potential safety concerns. In considering the potential effects of excessive vibration, the inspector noted that in the HC case, small bore piping and/or instrument tubing was directly affected. The prolonged existence of excessive vibration apparently caused minor cracking at key welds on instrument lines. Thus, even at low stress levels, this vibration contained sufficient energy to fail the instrument lines by high cycle fatigue. The inspector noted that the task team's review did not fully consider the effects of these failures for the Unit 2 event. Their report did not evaluate the instrument line weld failures seen in the 1987 HC event. No attempt was made to calculate or quantify a lifetime fatigue failure penalty to be assessed for this 18 hour event.

This concern and the need for a detailed technical evaluation of this event was discussed with the NRC technical experts. On July 7, a detailed NRC review was begun to evaluate this event. The licensee assessment was reviewed and questioned in a July 14 conference call between NRC and PP&L. Because of the highly technical nature of this issue, NRC has requested that a management meeting be held in September 1994 in Rockville, MD to resolve open issues. The licensee has agreed to this time frame and has also agreed to revise, if necessary, their power uprate submittal for Unit 1 based on the outcome of this meeting. The NRC will continue to evaluate the generic implications of this issue.

5. PLANT SUPPORT

5.1 Radiological Controls

PP&L's compliance with the radiological protection program was verified on a periodic basis. These inspection activities were conducted in accordance with NRC inspection procedure 71707. Observations of radiological controls during maintenance activities and plant tours indicated that workers generally obeyed postings and Radiation Work Permit requirements. No significant observations were made.

5.2 Emergency Preparedness

The inspector reviewed licensee event notifications and reporting requirements for events that could have required entry into the emergency plan. No events were identified that required emergency plan entry.

5.3 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 significant observations were made.

6. SAFETY ASSESSMENT/QUALITY VERIFICATION

6.1 Open Item (OI) Followup

(Closed) VIO 387/92-23-04, Kaowool Installed in a Fire Zone without Automatic Suppression

NRC Inspection 50-387 & 388/92-23 identified that Kaowool was installed in fire zone 0-28H without an automatic fire-extinguishing system. This was a violation of License Condition 2.C.(6) and Deviation Request 17, in that the Kaowool was installed in an area without installed automatic fire suppression and detection. This condition was also a violation of License Condition 2.C.(7), in that the Kaowool was installed in the unsprinkled fire zone, 0-28H, without this Kaowool fire barrier wrap material having been subjected to an ASTM E-119 fire test at an approved testing laboratory.

In the response to the Notice of Violation, PP&L concluded that the reason for the violation was that the drawings for this area were incorrect, and PP&L did not recognize that Kaowool was in this area. PP&L's corrective action was to replace these Kaowool barriers with preformed Thermo-lag fire barrier material. The planned corrective actions to avoid further violations included not using Kaowool for new installations and maintaining existing Kaowool fire-wrap barriers.

The inspector reviewed the completed work package (WA-B30102) that replaced the Kaowool in fire zone 0-28H with preformed Thermo-lag. The new Thermo-lag in fire zone 0-28H was walked down and accessible surfaces were visually inspected. The new Thermo-Lag appeared to be installed in a neat and quality manner.

Discussions with licensee personnel revealed that walkdowns of all the fire barrier wraps in both units had not identified any additional instances where Kaowool was installed in an area without automatic fire suppression and detection. Based on a walkdown of a selected sample of areas in the Unit 1 reactor building and the control structure, the inspector did not identify any additional areas where Kaowool was installed in an area without sprinklers.

Based on the above reviews, this item is considered closed. However, an inspector follow item is being opened pending resolution of NRC Bulletin 92-01 and Generic Letter 92-08 issues concerning the indeterminate nature of the operability of the Thermo-lag used in fire zone 0-28H and the remainder of Units 1 and 2. Additionally, IFI 387 & 388/92-23-03 will continue to track the Region I staff request for NRR to review the "grandfathering" of Kaowool and whether valid and reasonable justification exists to support the continued use of Kaowool material at Susquehanna and other facilities where the product may be applied.

(Closed) URI 50-387 & 388/92-23-05, Gypsum Board Enclosures Fire Resistance Rating

During NRC Inspection 50-387 & 388/92-23, the inspector noted that Gypsum board enclosures were used in lieu of the Thermo-lag fire barrier system to protect four dc circuit breaker boxes in fire zones 0-28B-II and 0-28A. The inspector found that the enclosures were not installed in accordance with the design drawings. Licensee personnel were unable to locate data that would support a one-hour, fire resistance rating for the installed enclosures; and, therefore, the issue remained unresolved pending PP&L providing documentation to support a one-hour fire resistance rating of the installed Gypsum board enclosures. At the conclusion of the inspection, PP&L considered these enclosures operable.

Engineering Evaluation, SEA-ME-409, "Evaluation of One Hour Gypsum Board Fire Barrier," dated January 14, 1993, concluded that the fire resistance rating of this enclosure could be as low as 55 minutes. Nonconformance Report 92-241 documented this issue and determined that the enclosures were inoperable on January 6, 1993. NCR 92-241 also documented that the issue was reportable under 10 CFR 50.73 and that it would be reported in LER 92-015-02. The inspector reviewed this LER and noted that the issue and the corrective actions planned at that time were not clearly documented. After the inspector discussed this issue with licensee representatives, who indicated LER 92-015-02 would be revised to more clearly document the issue and the completed corrective actions. This commitment is being tracked internally by nuclear department Action-Required Item Number 92-23-05 for completion by June 20, 1994.

PP&L took corrective action to replace the Gypsum board enclosures with new fire-rated enclosures in accordance with design change packages DCP 93-9004 and 93-9005. Additional stated purposes of these DCPs were to ensure that the enclosures' fire doors and penetration seals met one-hour, fire-rated designs. PP&L's response to VIO 50-387/92-23-06 stated that the redesigned enclosures would be installed by April 30, 1993.

DCP No. 93-9004, Fire Barrier Enclosure Upgrade, dated February 11, 1993, and the associated safety evaluation for the Unit 1 Gypsum board enclosures were reviewed. Work Authorization (WA) C33120 for the installation of the redesigned gypsum board enclosure around Panel 1D624 was also reviewed. The new Gypsum board enclosures specified in DCP No. 93-9004 were verified to be consistent with Underwriters Laboratory Design Number U505 that has an assembly, fire resistance rating of two (2) hours. DCP No. 93-9004, the associated safety evaluation, and WA-C33120 were found appropriate in scope and detail, and no technical concerns were identified. A visual inspection of the accessible surfaces of the gypsum board enclosures for panels 1D624, 1D644, 2D624 and 2D644 was performed. The installation appeared to be installed in a neat and quality manner. A review of the work authorizations for installing these enclosures indicated that all were installed by the committed date. Based on the above reviews, this item is closed.

(Closed) VIO 50-387 & 388/92-23-07, Fire Barrier Surveillance Inspections

NRC Inspection 50-387 & 388/92-23 identified a violation in that between December 1989 and September 4, 1992, fire-rated assemblies, constructed of Kaowool and Gypsum board, had not been inspected and verified operable as required by Technical Specification 4.7.7.1. The inspector observed that the level of detail contained in the fire barrier surveillance procedures did not sufficiently demonstrate that all acceptance criteria were met. PP&L's planned corrective steps to avoid further violations included: (1) revising the surveillance procedures to address the types of fire barrier material to be inspected and to provide appropriate acceptance criteria for each type of fire barrier by April 30, 1993; and (2) reperforming the surveillance inspection with the revised surveillance procedures by the end of May 1994. Additionally, PP&L planned to provide additional engineering oversight of the inspections and to provide the surveillance inspection personnel with training on the revised procedures prior to their performing the surveillance inspections.

The revised Engineering Surveillance Procedures SE-013-007, "18 Month Inspection of Unit Common Fire Barriers," Revision 1, dated March 7, 1994; SE-113-007, "18 Month Inspection of Unit 1 Fire Barriers," Revision 1, dated November 29, 1993; and SE-213-007, "18 Month Inspection of Unit 2 Fire Barriers," Revision 1, dated December 22, 1993, were reviewed for technical adequacy. For a selected sample of areas in the control structure and the Unit 1 reactor building, the revised engineering surveillance procedures were walked down and compared to the fire protection features drawings.

The inspector reviewed the procedures and noted that the procedures require inspections of structural fire barriers and structural steel fireproofing, as well as wrapped raceway fire barriers. Fire doors, dampers and penetration seals are surveilled separately. The inspector noted that the procedure revisions provided significant improvements in the areas of: (1) describing the types of materials to be inspected; (2) the level of inspection required; (3) locations of materials to be inspected; (4) the inspection acceptance criteria; and (5) the level of detail required to document the inspection activities performed. The last completed test record for each of these surveillances was reviewed. The test records indicate that work requests were generated to correct the identified material deficiencies and procedure change forms were generated for to correct identified procedural errors. The tests were completed prior to the planned completion date of May 1994. Based on the significant improvement in the surveillance procedures and the quality of the current test records, this item is closed.

(Updated) VIO 387/92-23-06, Inadequate Quality Identified in Kaowool and Gypsum Board Fire Barriers

Plant walkdowns during inspection 50-387 & 388/92-23 identified examples where Gypsum board fire enclosures and Kaowool, installed as a cable raceway fire barrier wrap, were not installed in accordance with design specifications. This was identified as a violation of the fire protection program commitment for field personnel to verify that fire protection installations were installed in conformance with design specifications.

PP&L's response to the Notice of Violation concluded that these noncompliances resulted from inadequate surveillance procedure acceptance criteria and discrepancies with initial construction design documents. The corrective actions that PP&L committed to take included performing comprehensive walkdowns of fire-rated raceway wrap and taking action to correct the identified deficiencies by April 30, 1993. The identified discrepancies were documented in Nonconformance Reports 92-200, 92-201 and 93-005. Additional corrective steps identified to avoid further violations included enhancing the surveillance procedures and replacing improperly-installed gypsum board enclosures. A discussion of the inspector's review of PP&L's corrective actions to enhance the surveillance procedures was included in Report Section 3.0 concerning VIO 50-387 & 388/92-23-07. A discussion of the inspector's review of the redesigned gypsum board enclosure installations was included in Report Section 2.0 concerning URI 50-387 & 388/92-23-05.

The inspector reviewed Nonconformance Report (NCR) 92-201 and noted that, in January 1993, while removing the existing Kaowool fire-rated raceway wrap in fire zone 0-28H, PP&L identified a potential generic concern that the removed Kaowool had not been installed in accordance with the installation specification. The three deficiencies identified included: 1) incorrect overlapping and offsetting of joints in the Kaowool, 2) incorrect banding, and 3) failure to seal the raceway wrap barriers to penetrations that it met. The inspector noted that the detailed inspection results, sketches, and photographs were being maintained by the work group engineer and were not included in the record copy of the NCR. The NCR documents PP&L's operability evaluation of this condition and noted that the barriers had been previously declared inoperable due to generic concerns related to the qualification and fire resistance rating of Thermo-Lag. The NCR also documented that the concern would be reported in LER 92-15.

To address the potential generic installation inadequacy of all Kaowool installations, a sample of conduit was selected for disassembly and inspection in accordance with work authorization (WA) B30119. Certain sampled conduits were wrapped, but PP&L had determined that their safe shutdown methodology did not require wrapping these conduits. These select conduits were subsequently inspected by the licensee to determine if any of the three previously mentioned deficiencies existed. The completion of the activity was documented in a WA. The inspector reviewed the completed WA record and found that it was completed on April 8, 1993. The inspector noted that the detailed inspection results required by the WA were not included in the record copy. At the time of this inspection, NCR 92-201 had not been updated to document the inspection results, evaluate the operability and reportability considerations, and identify any additional corrective actions. The inspector noted that the detailed inspection results, sketches, and photographs were being maintained in the work group engineer's personal files. Licensee representatives stated that NCR 92-201 would be updated to reflect this information. The deficiencies identified were similar to those found in fire zone 0-28H and included incorrect overlapping and offsetting of joints in the Kaowool, and incorrect banding.

From discussions with licensee representatives, the inspector noted that PP&L's continuing corrective actions include the planned performance of Design Change Package (DCP) 93-9041 to inspect an additional sample of wrapped raceways. Although originally planned for performance in the third quarter of 1993, DCP 93-9041 was completed in June 1994. Similar deficiencies were identified and are being corrected by the licensee. Discussions with the responsible work group engineer, indicated that at least one of the wrapped cable trays in this sample has been reworked during past modification work; and, therefore, it is expected to have been reworked to the correct design installation specifications. The inspector was concerned with the licensee's failure to include the timely detailed inspection results in NCR 92-20, as well as their lack of additional timely inspections to address these three potentially generic Kaowool installation deficiencies.

During a walkdown of wrapped raceways on the 719' elevation of the reactor building, the inspector noted that there was duct tape applied on the Zetex covering on Raceway F1KY01. Polyken tape is required to be used to seal seams or rips on Kaowool fire barrier wrap. PP&L initiated work Authorization S43649 to remove the duct tape and inspect the fire barrier wrap for damage and to contact fire protection system engineering to evaluate the inspection results. The inspector considered this to be an appropriate corrective action.

Based on the above review, this item remains open pending further review of the original issues and a review of the corrective actions taken to disposition the potentially-generic installation inadequacies identified with the Kaowool fire barrier wrap materials.

(Updated) VIO 387/92-23-08, Drawing Quality

During NRC Inspection 50-387 & 388/92-23, the inspector reviewed a sample of the facilities Fire Protection Features Drawings and conducted a walkdown of the plant. The inspector also compared the Fire Protection Features Drawings to another one of the licensee's drawings (E-294/295) that documented the plant's wrapped raceways. The inspector noted conflicts between pairs of drawings and between drawings and as-installed plant configurations. The inspector found that adequate quality assurance had not been applied to these fire protection design-basis documents.

PP&L's response to the Notice of Violation stated that the errors in the E-294/295 and C-1700 series drawings, used for fire protection activities, resulted from inadequate drawing verification during their original development. PP&L's corrective actions included revising and identifying these drawings as "Quality-F" class drawings. Comprehensive walkdowns of fire-rated raceway wrap were to be conducted to identify and correct the drawing errors. The NRC's reply to PP&L's response to the Notice of Violation questioned whether other fire protection features and their drawings have similar quality deficiencies. To address this concern, PP&L committed to correct, the as-built and control drawings for fire- wrapping material, penetration seals, suppression equipment, detection equipment, and Appendix R-required lighting and barriers (including walls, doors, and dampers). This effort was completed May 20, 1994.

From a review of nuclear department action-required (NDAR) items, the inspector noted that the walkdown of wrapped raceways and updating of the related E-294/295 and C-1700 series drawings was completed October 1993. Discussions with licensee representatives indicated that the remaining fire protection features drawings, for penetration seals, barriers, Appendix R lighting, etc., were completed in PP&L's corporate offices by May 19, 1994. During this inspection period, these revised drawings were not available at the site for detailed inspection. This item will remain open pending detailed review of the revised drawings and verification that PP&L has met their corrective action commitments for this violation.

7. MANAGEMENT AND EXIT MEETINGS

7.1 Resident Exit and Periodic Meetings

The inspector discussed the findings of this inspection with station management throughout and at the conclusion of the inspection period. 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 Inspections Conducted By Region Based Inspectors

Date	Subject	Inspection Report No.	Reporting Inspector
06/27/94 - 07/01/94	Confirmatory Measurements	94-13; 94-14	J. Kottan