UNITED STATES NUCLEAR REGULATORY COMMISSION

REGION I

Inspection Report Nos.	50-387/91-13; 50-388/91-13
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:	August 13, 1991 - September 23, 1991
Inspectors:	G. S. Barber, Senior Resident Inspector, SSES J. R. Stair, Resident Inspector, SSES

Approved By: _____ 10/9/91

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

Inspection Summary:

Areas Inspected: Routine inspections were conducted in the following areas: operations, radiological controls, maintenance/surveillance testing, emergency preparedness, security, engineering/technical support, safety assessment/quality verification, and Licensee Event Reports, Significant Operating Occurrence Reports, and Open Item Followup.

Results: During this inspection period, the inspectors found that the licensee's activities were directed toward nuclear and radiation safety. No violations or deviations were identified. An Executive Summary is included and provides an overview of specific inspection findings.

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EXECUTIVE SUMMARY

Susquehanna Inspection Reports

50-387/91-13; 50-388/91-13

August 13, 1991 - September 23, 1991

Operations (30702, 71707, 71710)

Operators effectively controlled plant evolutions and identified plant problems.

An Engineered Safety Feature Walkdown of the Unit 2 High Pressure Coolant Injection System (HPCI) was conducted during the period. The inspector determined that the HPCI system was properly aligned and capable of performing its safety function. Several minor discrepancies were noted which were addressed by the licensee. Overall equipment condition indicated good maintenance and housekeeping practices.

Radiological Controls (71707)

Area contamination occurred when air, generated while backwashing the fuel pool demineralizers, was directed into the floor drain system. Consequently, the air blew contamination from the drain system onto a large floor area of 749' elevation of the reactor building. The cause is still being reviewed by the licensee. An unresolved item was generated.

Maintenance/Surveillance (61726, 62703)

The licensee exercised good control of maintenance and surveillance activities. No scrams or ESF actuations were attributable to maintenance or surveillance activities.

Emergency Preparedness (82301)

No emergency preparedness issues emerged during the period.

Security (71707)

Routine observation of protected area access and egress control indicated good control by the licensee.

Engineering/Technical Support (71707, 92720, 93702)

The Unit 1 "A" loop of RHR began pressurizing during the period due to backleakage through the injection line. Data from a previous similar situation involving Unit 2 was compared to the Unit 1

case, and indicated that leakage was a fraction of the TS limit. Procedural and administrative controls have been established to monitor and control the leakage. Further leakage testing and valve repair is planned for the next refueling outage.

A 10 CFR Part 21 Report was issued by Cooper-Bessemer (C-B) concerning Cracked Fuel Injector Nozzles for C-B emergency diesel generators. The faulty injectors were confined to specific lots which P.P.& L. Company determined were not received at the Susquehanna Steam Electric Station. Followup on this deficiency by the licensee was adequate.

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

Licensee Event Reports (LERs) were reviewed during the period. These LERs were clearly detailed and accurate, and appropriate corrective action was either completed or planned within a reasonable period of time. All the LERs were submitted within the required time frame per 10 CFR 50.73.

A total of 67 Significant Operating Occurrence Reports were reviewed during the period.

Details

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, measurement of radiation levels, independent calculation, and selective review of applicable documents. Abbreviations are used throughout the text. Attachment 1 provides a listing of these abbreviations.

1.2 Susquehanna Unit 1 Summary

Unit 1 operated at or near full power throughout the inspection period. Scheduled power reductions were conducted during the period for control rod pattern adjustments, surveillance testing, and maintenance. No ESF Actuations occurred in Unit 1 during the inspection period.

1.3 Susquehanna Unit 2 Summary

Unit 2 entered the inspection period with the unit at 100 percent full power. The unit operated at full power until August 21, when a shutdown was performed in order to replace the "C" phase main transformer. The shutdown was initiated after routine sampling of the transformer oil showed an increased level of combustible gas which is indicative of a potential internal fault. A second sample was taken which confirmed the increase in gas concentration. The transformer was replaced on August 27 and the unit placed back on line on August 28. Full power was reached on August 30 and was maintained throughout the remainder of the inspection period. The removed transformer is scheduled to be shipped back to the vendor for refurbishment. No ESF actuations occurred in Unit 2 during the inspection period.

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 26 hours of deep backshift inspections on August 23 from 2:00 a.m. to 6:00 a.m.; September 16 from 12:00 a.m. to 6:00 a.m.; September 21 from 7:30 a.m. to 5:15 p.m.; and, September 22 from 7:30 a.m. to 3:15 p.m..

2.2 Inspection Findings and Review of Events

2.2.1 Engineered Safety Feature Walkdown - Unit 2 - High Pressure Coolant Injection System

During the period, the inspector independently verified the status of the Unit 2 High Pressure Coolant Injection (HPCI) System. The engineered safety feature (ESF) system status verification included the following:

-- Confirmation that the licensee's system check lists and operating procedure are consistent with the plant as-built drawings, configuration, and labeling.

-- Identification of equipment conditions and items that might degrade performance.

-- Verification of proper breaker positions at local electrical panels and correct indications on control panels.

-- Verification of the flow path and valve line ups necessary for the system to perform its safety function.

-- Verification of good housekeeping in the area of the system equipment.

No major weaknesses were identified during the inspection. However, the inspector noted several minor discrepancies during the inspection and identified them to the licensee for review.

-- Several motor operated valves on the HPCI system were noted as having indicators on their motor operators which did not appear to correlate to actual valve position.

The inspector discussed the "Stem Movement In Inches" indicator with an engineer from the licensee's valve group. The engineer presented vendor documentation which showed the indicator measures compression of springs in the motor operator. These springs cushion overtravel of the motor operator when used for high speed applications. Consequently, the indicators are not intended to depict actual valve position.

-- Two area radiation (rad) monitors, located on the south wall of the HPCI room, had damaged electrical connectors.

The inspector informed the licensee's quality control (QC) department of this hardware deficiency. The licensee stated that the rad monitors were non-safety related and issued work authorizations (WAs) to have the electrical connections repaired.

-- A spring can support for the lube oil cooler relief line was installed at an angle which could effect its performance.

The licensee's QC department investigated the hanger and determined that the angle and the cold setting were acceptable.

-- Two HPCI valves did not have labels, 256F025 "HPCI COND PP DSCH OB DRN ISO" and 256F061A "HPCI BOOSTER PP VENT."

The inspector identified the missing labels to the licensee. The licensee subsequently replaced them.

The inspector performed a followup inspection of the HPCI system to verify that the licensee had initiated corrective actions. The inspector noted that the missing valve labels were replaced and equipment deficiency tags indicated that the area rad monitors had been entered into the WA system. The inspector also determined that the HPCI system was properly aligned in accordance with the operating procedure. The minor discrepancies noted would not have prevented the HPCI system from fulfilling its intended safety function. No additional inadequacies were noted.

3. RADIOLOGICAL CONTROLS

3.1 Inspection Activities

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.

3.2 Inspection Findings

Observations of radiological controls during maintenance activities and plant tours indicated that workers generally obeyed postings and Radiation Work Permit requirements. No major inadequacies were noted.

3.2.1 Fuel Pool Demineralizer Backwashing Causes Large Area Contamination

The licensee discovered contamination on the 749' elevation of the Unit 1 Reactor Building (RB) when a contractor alarmed a Personnel Contamination Monitor (PCM) at approximately 2:00 p.m., August 12. When questioned by licensee health physics personnel, the contaminated individual stated that he was working in 749' elevation of the RB. The area was surveyed and contamination was found, with the heaviest contamination near the floor drains. The licensee reviewed recent evolutions and noted the fuel pool (FP) filter demineralizers (FDs) had been backwashed at 11:30 a.m. that day. Further review noted a problem with program timers used to backwash spent resin from the filter elements into the Backwash Receiving Tank (BRT). The licensee believed that an excessive volume of air passed through the BRT, to its overflow, and into the floor drain system. Residual contamination inside the piping was then blown back up through other floor drains on the

same elevation. Drain traps prevented contaminating other elevations. Air samples taken approximately 3 to 5 hours after the initial timer problem indicated that airborne contamination was less than the concentration values specified in 10 CFR 20. The licensee reviewed the conditions that existed and determined the event was not reportable. Upon review, the inspector concurred with that determination.

Upon discovery of the contamination, the licensee took action to control the affected area. The area was posted and decontamination (decon) was begun. Surveys taken prior to decon showed general contamination levels ranging from 2000 to 80,000 dpm/100cm^2, with the highest levels at the floor drains near the Backwash Receiving Tank. The licensee's Area Contamination Report No. 91-061 indicated localiized radiation levels as 10 mR/hr, gamma; and 360 millirad/hr, beta. Subsequent contamination surveys indicated low levels of contamination in overhead areas, demonstrating the possibility that low level airborne activity might have occurred. However, followup air samples did not indicate any unusual airborne activity. The 749' elevation was subsequently decontaminated and the area was released for normal access.

The licensee formed an Event Review Team (ERT) to identify causal factors and propose corrective actions. Their effort was not complete as of the end of this report period. Therefore, the adequacy of the licensee's root cause identification and corrective action will remain unresolved pending completion and review by the NRC. (UNR 50-387/91-13-01 (Common)

4. MAINTENANCE/SURVEILLANCE

4.1 Maintenance and Surveillance Inspection Activity

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

4.2 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:

-- Eighteen Month Inspection and Maintenance of Emergency Diesel Generator OG501C per Surveillance Authorization (SA) A11451 on August 28.

-- Maintenance and Electrical checks of RPS Transformer 1X201B per WA 511087 on September 13.

-- Replacement of RCIC Room Unit Cooler IV-208B Coils per WA C13382.

-- Replacement of "A" Reactor Building Closed Cooling Water Heat Exchanger 1E201A Tubes per Plant Modification 88-3041A.

4.3 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:

-- SO-054-003, "A" Loop Emergency Service Water (ESW) Quarterly Flow Surveillance, dated 8/15/91.

-- SR-200-001, "Determination of Core Thermal Limits," performed on September 16.

-- SO-070-001, "Monthly Operational Check of Standby Gas Treatment System," performed on September 18.

4.4 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 surveillance and 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. No unacceptable conditions were identified.

5. EMERGENCY PREPAREDNESS

5.1 Inspection Activity

The inspector reviewed licensee event notifications and reporting requirements for events that could have required entry into the emergency plan.

5.2 Inspection Findings

No events were identified that required emergency plan entry. No inadequacies were identified.

6. SECURITY

6.1 Inspection Activity

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.

6.2 Inspection Findings

The inspector reviewed access and egress controls throughout the period. No unacceptable conditions were noted.

7. ENGINEERING/TECHNICAL SUPPORT

7.1 Inspection Activity

The inspector periodically reviewed engineering and technical support activities during this inspection period. The on-site Technical (Tech) section, along with Nuclear Plant Engineering (NPE) in Allentown, provided engineering resolution for problems during the inspection period. The Tech section generally addressed the short term resolution of problems; and NPE scheduled modifications and design changes, as appropriate, to provide long term problem correction. The inspector verified that problem resolutions were thorough and directed to preventing recurrences. In addition, the inspector reviewed short term actions to ensure that the licensee's corrective measures provided reasonable assurance that safe operation could be maintained.

7.2 Inspection Findings

7.2.1 Residual Heat Removal System Valve Leakage - Unit 1

During a routine tour on August 15, the inspector noted that the Unit 1 "A" Residual Heat Removal (RHR) loop was pressurized above the normally maintained pressure, i.e. 150 psig. The observed pressure was approximately 280 psig. From interviews with the system engineer and operations personnel, the inspector learned that the licensee had been aware of the pressurization for about the last ten days; and had analyzed the cause, quantified the leakage, and implemented the appropriate procedures. However, the inspector noted that the licensee had not documented this discrepant operating condition. As a result of the inquiry, the licensee generated SOOR 1-91-206 on August 15 to track resolution of the concern.

The licensee identified the source of the pressurization as back leakage through two series valves that connect the low pressure RHR system to the reactor pressure vessel (RPV). The motor operated injection valve (F015) and the testable check valve (F050) provide the intersystem boundary. The leakage rates through these valves is normally determined by testing during refueling outages.

The inspector questioned the licensee to ensure that TS leakage limit was being met and that the existing leakage was being properly evaluated and quantified. The licensee's evaluation was based on a comparison to the known leakage from a previous similar problem on Unit 2. Previously, the Unit 2 FO15 valve developed a leak that caused system pressurization. As a result, the leakage was monitored and administrative controls were developed to control and limit pressurization. From February 1990 to November 1990, the Unit 2 "B" loop of RHR pressurized between zero and ten times per day. When pressure reached 300 psig, the system was depressurized through the heat exchanger (HX) vents. Leakage rate was determined by leak rate testing during refueling outages. The following pertains:

Outage No.	Date	2F015B (gpm)	2F050B/2F122B (gpm)
2	04/07/88	0.01322	0.009
3	09/26/89	0.37	0.052
-	05/30/90	0.247	---
4	04/91	0.35	0.21

Currently, the frequency of pressurization for the Unit 1 "A" loop of RHR is less than previously observed in the Unit 2 "B" loop of RHR. The maximum leakage determined for the Unit 2 condition was 0.37 gpm. Based on this evaluation, the licensee concluded that the leakage observed in Unit 1 is no more that previously observed in Unit 2. Consequently, the licensee determined that the leakage is within the one gpm leakage specification of TS 3.4.3.2.

In addition to this bounding analysis, the licensee implemented additional administrative controls to

monitor and limit the effects of back leakage through the FO15 isolation valve. The RHR system operating procedure (OP-149-001, Section 3.10) currently requires that the RHR system be vented through heat exchanger vent valves when RHR pressure reaches 300 psig. Each venting evolution is logged by the Shift Technical Advisor (STA) and the Plant Control Operator (PCO). The PCO records depressurization in the unit log; and the STA monitors and performs trending analysis of each depressurization event. Any significant change (increase) in the expected frequency of depressurization requires evaluation by licensee management and operations personnel to determine resolution and corrective measures. The inspector determined the licensee's analysis of the leakage and implementation of a formal administrative system to monitor and control system pressurization appeared acceptable for the present.

The licensee is planning on performing a 1000 psig leakage test for the Unit "A" RHR F015 valve during the next refueling outage (March 1992). The results of the test will be used to determine if disc-seat lapping or other repair is necessary.

The inspector questioned the licensee as to why a SOOR had not been generated when this matter was first observed by the licensee. The licensee indicated that while the staff was aware of the leakage, had analyzed it as less than one gpm, and initiated the proper administrative controls to monitor and limit system pressurization and leakage, a SOOR was appropriate for the circumstance and should have been generated. The inspector noted that the lack of a SOOR did not appear indicative of a programatic deficiency, particularly since the licensee had initiated action.

7.2.2 10 CRF 21 Report - Diesel Generator Cracked Fuel Injectors

An Event Notification was issued by the NRC on April 8, 1991 based on the discovery of cracked fuel injector nozzles on Cooper-Bessemer emergency diesel generators at the South Texas Project (STP) Nuclear Station. The cracked fuel oil injectors allowed fuel oil to drain into the engine lube oil sump and may have adversely affected mechanical functioning of the engines and the capability of the diesels to respond to lengthy emergency duty. STP determined that essentially all of the failed injector nozzles were of the same lot number (150006) and replaced twenty-two of those nozzles in both units as a precautionary measure. Two additional nozzles were found cracked (one each from batch numbers 150009 and 150010) but were from unrelated causes. STP removed all nozzles of these lots from service. The primary root cause of the failure of lot number 150006 was determined to be excessive undersizing of the inner ligament of the nozzles, while the failures from lots 150009 and 150010 were attributed to excessive nitriding, and severe clogging of the spray holes and/or planar inclusions in the microstructure, respectively.

Cooper-Bessemer issued a 10 CFR Part 21 Report to the NRC, dated April 5, 1991 on the cracked injector nozzles. However, this Report did not address the failure mechanism of lot number 150009.

The licensee reviewed the Part 21 Report and STP's Licensee Event Report connected to this defect and determined that they have not received any injector nozzles for use at Susquehanna from either lot numbers 150006 or 150010. The inspector questioned the licensee on their review of the Part 21

Report and whether or not they had reviewed the LER associated with it. Specifically, the inspector questioned whether the licensee had checked to determine if they had any injectors from lot 150009 in conjunction with their review. The licensee verified that they had no injectors from lot 150009. Since the licensee has no injectors from the affected lots and have taken adequate action to ensure that they will not receive any in the future, the inspector considers this issue closed.

8. SAFETY ASSESSMENT/QUALITY VERIFICATION

8.1 Licensee Event Reports (LER), Significant Operating Occurrence Report (SOORs), and Open Item (OI) Followup

8.1.1 Licensee Event Reports

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 determined whether further information was required from the licensee, whether generic implications were involved, and whether the event warranted onsite followup. The following LERs were reviewed:

Unit 1

91-006-00 ESF Actuations Due to RPS Breakers Spurious Trip. This LER documented primary containment isolations and initiations of the "A" Standby Gas Treatment System and the "A" Control Room Emergency Outside Air Supply System. These actuations were due to loss of the primary power supply to the "A" Reactor Protection System when its associated Electrical Protection Assembly breakers spuriously tripped. Although no root cause was positively identified, a probable cause was a faulty power supply. This event was reviewed in NRC Inspection Report 91-07.

91-007-00 "C" Emergency Diesel Start Time Exceeded Technical Specification Requirements. This LER documented a slow start of the "C" Emergency Diesel Generator on June 19. The slow start resulted from voiding of the fuel supply header due to a modification of the header performed in 1990. This event was reviewed in NRC Inspection Report No. 91-07.

91-008-00 Loss of Offsite Circuit Caused Unit Scram and MSIV Isolation. This LER documented a reactor scram and MSIV isolation on July 31. The event occurred due to a loss of power to the unit's startup transformer (T-10) which caused a half-scram signal in Division 1, along with a pre-existing half-scram signal in Division 2. Stabilization of the unit and achievement of cold shutdown conditions were complicated due to the MSIV closure and other equipment problems. This event was reviewed in NRC Inspection Report No. 91-10.

91-009-00 Entry Into Technical Specification 3.0.3 during Radiation Monitor Retest. This LER documented the need to reset a channel trip system in order to test the "B" Main Steam Line Radiation Monitor following its repair. By removing the channel from the tripped condition, entry into T.S. 3.0.3 was required since the action statement of T.S. 3.3.2 was no longer being met. The licensee is pursuing a T.S. change which would allow resetting trip signals in order to perform required surveillance testing following maintenance.

91-010-00 Inadvertent Placement of the Reactor Mode Switch into the Run Position. This LER documented an MSIV isolation signal and RPS actuation due to the accidental placement of the reactor mode switch into the run position when attempting to place it in Startup. Since the MSIVs were in a closed position and all control rods were inserted at the time, no valve or rod movement occurred. This event was attributed to personnel error and a bent reactor mode switch key. The key was subsequently straightened and the event reviewed with all licensed personnel.

Unit 2

91-007-01 ESF Actuations Due to an RPS EPA Breaker Spurious Trip. This revision to LER 91-007 incorporates the results from subsequent investigations which determined that the cause of the power supply loss to the "A" RPS was a faulty capacitor in the EPA logic card.

91-008-00 ESF Actuations Due to RPS EPA Breakers Spurious Trip. This LER documented primary containment isolations and initiations of the "A" Standby Gas Treatment System and the "A" Control Room Emergency Outside Air Supply System. These actuations were due to loss of the primary power supply to the "A" Reactor Protection System when its associated Electrical Protection Assembly breakers spuriously tripped. Although no root cause was positively identified, the most probable cause was a faulty oscillator on the EPA logic card. Due to industry experience with EPA problems, a study was prepared by the vendor (GE) for the BWR Owners Group. This study is undergoing review by the licensee to assess operational or design changes which have a potential to enhance reliability of the RPS. This event was reviewed in NRC Inspection Report No. 91-07.

91-009-00 Reactor Water Cleanup (RWCU) System Isolation While Performing a Routine Functional Test. This LER documented the inadvertent actuation of the RWCU System steam leak detection system during monthly functional testing on the Main Steam Line Tunnel delta-temperature instrumentation. This actuation resulted in closure of the RWCU inboard primary containment isolation valve HV-244-F001. Instrumentation and Control Technicians connected test equipment to the RWCU steam leak detection instrumentation in error due to poor verbal communications. This event was reviewed in NRC Inspection Report No. 91-07.

91-010-00 Reactor Water Cleanup System Isolated Twice Due to Actuation of Steam Leak Detection Instrumentation. This LER documented two RWCU system isolations on June 28. The first occurred when the Riley temperature module meter switch for RWCU penetration room temperature was positioned to read room temperature. The second occurred when a Division II RWCU Penetration room high ambient temperature signal was received. Both occurrences were due to elevated ambient room temperatures as a result of unusually hot weather conditions. The licensee requested and received a Waiver of Compliance on July 12, which allowed raising the RWCU high ambient room temperature trip setpoints during hot weather to prevent future spurious isolations. The waiver will remain in effect until October 15. A Technical Specification change request for a permanent change was previously submitted. These events were reviewed in NRC Inspection Report No. 91-10.

91-011-00 High Radiation Area Door Not Properly Secured. This LER documented discovery on June 26 by a Health Physics technician, that a high radiation area door (DR #142) although closed, was not locking properly because the locking mechanism was not properly seated in the striker plate. The licensee attributed the event to both personnel error and poor physical condition of the door. This door was not included on the station key usage log when used the preceeding day which prevented an independent verification of door closure and proper locking. An investigation performed by the licensee determined that no unauthorized entries into the related high radiation area occurred during the period the door was not properly secured. This event was reported per 10 CFR 50.73 (a)(2)(i)(c) in that it was a condition prohibited by T.S.6.12.2. The report was submitted on August 12 since the event was not determined reportable until July 19 due to a communication error between the Health Physics and Licensing Compliance staffs.

91-012-00 Automatic Unit Scram From Full Power. This LER documented the simultaneous scram and dual reactor recirculation pump trip signals which resulted in an automatic unit shutdown on August 6. The event was caused by water intrusion into an electrical junction box on the No. 2 main turbine stop valve which led to a short-circuit in the stop valve control logic and the partial closure of 3 of the stop valves. The plant responded as designed with no major complications. Appropriate repairs were made and the plant returned to power on August 8. This event ws reviewed in NRC Inspection Report No. 91-10.

No unacceptable conditions were identified.

8.1.2 Significant Operating Occurrence Reports

SOORs are provided for problem identification and tracking, short and long term corrective actions, and reportability evaluations. The licensee uses SOORs to document and bring to closure problems identified that may not warrant an LER.

The inspectors reviewed the following SOORs during the period to ascertain whether: additional followup inspection effort or other NRC response was warranted; corrective action discussed in the licensee's report appears appropriate; generic issues are assessed; and, prompt notification was made, if required:

Unit 1

45 SOORs, inclusive of 1-91-193 through 1-91-240.

Unit 2

22 SOORs, inclusive of 2-91-191 through 2-91-214.

No unacceptable conditions were identified.

9. MANAGEMENT AND EXIT MEETINGS

9.1 Routine 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.

9.2 Inspections Conducted By Region Based Inspectors

Dates	Subject	Inspection Report No.	Reporting Inspector
8/25-8/30	SAQV	91-11	J. Trapp
8/21-8/23	RADCON	91-14	N. Noggle
8/24-8/31	RADCON	91-16	J. Kottan
8/24-8/31	SAQV	91-17	R. Mathew

9.3 Management Meeting - Organizational Effectiveness Review and Other Major Issues

A management meeting was held between PP&L and NRC on August 23 to discuss PP&L's Organizational Effectiveness Review (OER) and other major issues. This list of attendees is provided in Attachment 2 and their presentation. Also discussed, was the licensee's recent enhancements to the Nuclear Department Management System and Power Uprate activities.

Emergency Planning issues and strike preparations were also discussed at the August 23 meeting. The Emergency Plan Full Field Exercise (FFE-3) is scheduled for February 1993 with significant involvement required by the NRC and a number of additional Federal, State and Local agencies. PP&L is well into their planning efforts. The licensee's strike preparations were also reviewed. The licensee indicated that a full complement of recently trained staff could assume bargaining unit positions, if the need arose.

ATTACHMENT 1

Abbreviation List

AD - Administrative Procedure
ADS - Automatic Depressurization System
ANSI - American Nuclear Standards Institute
CAC - Containment Atmosphere Control
CFR - Code of Federal Regulations
CIG - Containment Instrument Gas
CRDM - Control Rod Drive Mechanism
CREOASS- Control Room Emergency Outside Air Supply System
DG - Diesel Generator
DX - Direct Expansion
ECCS - Emergency Core Cooling System
EDR - Engineering Discrepancy Report
EP - Emergency Preparedness
EPA - Electrical Protection Assembly
ERT - Event Review Team
ESF - Engineered Safety Features
ESW - Engineering Service Water
EWR - Engineering Work Request
FO - Fuel Oil
FSAR - Final Safety Analysis Report
HVAC - Heating, Ventilation, and Air Conditioning
ILRT - Integrated Leak Rate Test
I&C - Instrumentation and Control
JIO - Justifications for Interim Operation
LCO - Limiting Condition for Operation
LER - Licensee Event Report
LLRT - Local Leak Rate Test
LOCA - Loss of Coolant Accident
LOOP - Loss of Offsite Power
MSIV - Main Steam Isolation Valve
NCR - Non Conformance Report
NDI - Nuclear Department Instruction
NPE - Nuclear Plant Engineering
NPO - Nuclear Plant Operator
NQA - Nuclear Quality Assurance
NRC - Nuclear Regulatory Commission
OI - Open Item
PC - Protective Clothing
PCIS - Primary Containment Isolation System
PMR - Plant Modification Request
PORC - Plant Operations Review Committee
QA - Quality Assurance

RB - Reactor Building
RCIC - Reactor Core Isolation Cooling
RG - Regulatory Guide
RHR - Residual Heat Removal
RHRSW - Residual Heat Removal Service Water
RPS - Reactor Protection System
RWCU - Reactor Water Cleanup
SGTS - Standby Gas Treatment System
SI - Surveillance Procedure, Instrumentation and Control
SO - Surveillance Procedure, Operations
SOOR - Significant Operating Occurrence Report
SPING - Sample Particulate, Iodine, and Noble Gas
TS - Technical Specifications
TSC - Technical Support Center
WA - Work Authorization

ATTACHMENT 2

PP&L/NRC Management Meeting Attendees

August 23, 1991

Pennsylvania Power & Light Company (PP&L)

R. Byram, Vice President, Nuclear Operations

U.S. Nuclear Regulatory Commission (NRC)

C. W. Hehl, Director, Division of Reactor Projects (DRP)
M.W. Hodges, Director, Division of Reactor Safety (DRS)
J.R. White, Chief, RPS2A, PB2, DRP
G.S. Barber, Senior Resident Inspector, Susquehanna, DRP
P.D. Kaufman, Project Engineer, DRP