U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Report Nos. 50-387/85-31; 50-388/85-26

Docket Nos. 50-387 (CAT C); 50-388 (CAT C)

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: September 30, 1985 - November 10, 1985

Inspectors: for Jack Strosnider 11/26/85
R. H. Jacobs, Senior Resident Inspector — date

for Jack Strosnider 11/26/85
L. R. Plisco, Resident Inspector — date

Approved By: Jack Strosnider 11/26/85
J. Strosnider, Chief Reactor Projects Section 1B, DRP — date

Inspection Summary:

Areas Inspected: Routine resident inspection (U1 - 201 hours; U2 - 50 hours) of plant operations, licensee events, open items, surveillance, maintenance and ESF walkdown.

Results: ESF walkdown of Unit 1 Core Spray System identified no substantive discrepancies (Detail 2.3); review of 18 month diesel surveillance requirement identified several discrepancies (Detail 5.3); licensee actions following RWCU spill on October 11 were thorough and conservative (Detail 6.0). No violations were identified.

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DETAILS

1. Follow-up on Previous Inspection Items

1.1 Closed) Inspector Follow-up Item (387/83-12-04): Control of Control Structure Boundary Doors

In May 1983, the licensee discovered that a control structure boundary door had been blocked open by cables to facilitate Unit 1 battery testing. This violated the control structure boundary which would inhibit the proper performance of the CREOASS system in a post-accident condition. In Inspection Report 50-387/83-12, the inspector discussed concerns about the controls over the control structure boundary and training of station personnel.

As a result of the incident, the licensee marked each affected door as a control structure boundary door and added a daily check of the doors on an operator rounds sheet. On December 17, 1984, the licensee revised LER 83-076 to require that only non-security control structure boundary doors be checked on operator rounds. Security doors are monitored continuously. The inspector reviewed SO-100-007, Rev. 5, "Daily Surveillance Operating Log" to verify that the doors have been included in the operator's rounds sheets.

1.2 (Closed) Inspector Follow-up Item (387/83-31-03; 388/83-31-03): No Reference to Emergency Procedures in Alarm Response Procedures

In December 1983, the inspector noted that the alarm response procedure (ARP) for primary containment high pressure trip (AR-204-001) had no reference to the appropriate Emergency Operating (EO) procedure to be used during the alarm condition.

The licensee reviewed all ARPs and included, where appropriate, a reference to the associated Emergency Operating or Off-Normal procedure. The inspector reviewed several ARPs to verify that this action was completed.

1.3 (Closed) Inspector Follow-up Item (387/84-38-01; 388/84-47-01): Actions to Ensure Cleanliness of CRD Air System

In December 1984, the scram discharge volume (SDV) vent and drain pilot valve malfunctioned preventing the SDV vent and drain valves from closing. Investigation revealed a small piece of pipe dope was trapped between the valve inlet seat and disc of the pilot valve. Because of the concern of loose particulates in the CRD air system, the licensee performed a freon flush of the CRD air system and during the first refueling outage, cleaned system red brass piping where pipe dope was used. In addition, the licensee implemented PMR 85-1004 on Unit 1 which disassembled and replaced instrument air tubing from the CRD air header to each of the scram pilot solenoid valves (SPSV). Pipe dope had been used in the original tubing to the SPSVs. The flush results on Unit 1 did not identify any significant contamination of the air system. Hence, the licensee does not intend to perform a flush of the Unit 2 air system. Both units now have redundant SDV vent and drain valves and pilot valves, so the likelihood of both SDV pilot valves failing is reduced. PMR 85-1005 will replace instrument tubing at the SPSV on Unit 2 during the first refueling outage. The inspector had no further concerns.

1.4 (Closed) Inspector Follow-up Item (387/83-11-05): Missing Surveillance Requirements in Procedures

In April 1983, the licensee identified that six instruments had not been included in Operations surveillance procedures. These instruments were associated with containment isolation and ATWS recirculation pump trip. In LER 83-45 the licensee committed to review the operator's 12 hour, 24 hour and weekly surveillance procedures to ensure all channel check requirements were included.

The licensee performed the above review and added the missing instruments to the shiftly surveillance procedure. The inspector verified that the instruments were added. Following this occurrence, the licensee committed to perform a comprehensive review of the surveillance program, which included a verification that all surveillance requirements are included in appropriate procedures. This task has been completed and will be reviewed with Violation 387/83-20-01. The inspector also reviewed AD-QA-422, Surveillance Testing Program, to verify that the licensee ensures that surveillance procedures are revised as appropriate when the Technical Specifications are modified.

1.5 (Open) Inspector Follow-up Item (387/84-26-01): Diesel Generator Turbocharger Failure

On September 10, 1984, during monthly surveillance testing, the 'D' emergency diesel generator (EDG) tripped on high vibration. During three subsequent troubleshooting starts, maintenance personnel noted a whining noise from the turbocharger area and the EDG tripped on high vibration several minutes after starting. Partial disassembly of the Cooper-Bessemer ET high pressure turbocharger revealed that the blower end journal bearing and the adjacent thrust bearing were severely damaged. The turbocharger was replaced with a spare and the EDG was declared operable on September 12, 1984. A previous turbocharger thrust bearing failure occurred on the 'C' EDG during surveillance testing in February 1984. (See Inspection Report 50-387/84-07).

The licensee shipped the damaged turbocharger from the September failure to the vendor (Cooper-Bessemer) for evaluation. The vendor concluded that the lack of prelubrication to the thrust bearing was a major contributor to the accelerated bearing wear. The thrust bearing wear progressed with each "dry start" until the bearing geometry was altered to the point that the bearing was no longer capable of sustaining normal load even under normal lubrication conditions. The journal bearing failure was determined by the vendor to be consequential. The vendor recommended that the licensee pursue the proposal of adding a turbocharger prelubrication sequence to the test startups to eliminate the problem. The vendor stated the turbocharger may have been within minutes of impeller-casing contact and subsequent gross failure.

The turbocharger bearings are not prelubricated in the standby mode as are the remainder of the engine bearings. Due to the design of the labrynth bearing seal, lube oil cannot be continuously circulated through the turbocharger when the unit is shutdown because oil would flow past the seals and into the air inlet system. The bearings may not reach rated lube oil pressure until 5-10 seconds after starting.

The licensee has now incorporated a turbocharger bearing prelubrication sequence prior to starting the diesel during scheduled surveillance testing. Operating Procedure OP-024-001, Diesel Generators, was revised in February 1985 to allow manual initiation of turbocharger lubrication prior to manual starts. The surveillance test procedures have also been revised to incorporate the prelubrication sequence. The licensee has additionally issued a Request for Modification (RFM) to change the non-emergency mode starting logic to automatically initiate turbocharger lubrication prior to a diesel start. The licensee presently intends to implement this modification in 1987.

The 'A' and 'B' EDG turbochargers have not been inspected since the two bearing failures. The inspector reviewed the periodic maintenance requirements for the diesel generator (See Detail 5.3) and noted that the turbocharger is only inspected every six years (every fourth 18 month inspection). During the 18 month inspections, a vibration survey is performed, the turbocharger coast down time is checked and the turbocharger lube oil filters are inspected for metallic particles. These checks may not detect a degradation of the turbocharger thrust bearing. The last 18 month inspection was performed in January 1985, and the next inspection due date is September 1986. This inspection will be the fourth inspection, so in accordance with the surveillance procedure, SM-024-002, and the vendor recommendations, the turbochargers will be removed and inspected in 1986.

This item will remain open pending NRC review of the results of the 'A' and 'B' EDG turbocharger inspection and final corrective actions.

1.6 <u>(Closed) Inspector Follow-up Item (388/85-09-01): RCIC System Walkdown Discrepancies</u>

In March 1985, the inspector identified several minor discrepancies with the Unit 2 RCIC system during a system walkdown. These discrepancies were that 1) several RCIC valves were locked closed but the P&ID did not indicate that they were required to be locked closed, 2) two valves were indicated locked open on the P&ID but they are normally open, motor operated valves, and 3) some buckets of oil were left unattended in the RCIC room. The licensee issued Drawing Change Notice (DCN) 85-3262 to correct the above drawing discrepancies and the buckets of oil were removed from the room. The inspector verified the above actions.

1.7 <u>(Closed) Inspector Follow-up Item (387/84-35-03): Scram Discharge Volume (SDV) Vent and Drain Valve Missed Surveillance</u>

In October 1984, the licensee identified that surveillance procedure SO-155-003, SDV Vent and Drain Valve 18 Month Operability, was overdue by approximately 15 months. The cause of the overdue surveillance was an administrative error. The completion date of the previous surveillance was listed as June 2, 1983. The surveillance was actually last performed during preop testing in January 1982. June 2, 1983 was the date that a new surveillance authorization cover sheet was completed since the original surveillance document was misplaced.

The licensee performed a 100 percent verification of the completion dates for all surveillances whose frequency was quarterly or longer. No additional late surveillances were identified. An individual was assigned to conduct an on-going independent review of the surveillance program implementation.

1.8 <u>(Open) Unresolved Item (387/82-36-02): Preservice Inspection of Recirculation System Pipe Welds</u>

Because of ALARA considerations the licensee has proposed to perform volumetric examination of recirculation system sweepolet to riser pipe welds using an ultrasonic examination technique in lieu of the currently mandated radiographic technique using MINAC which was used during the last outage.

Previous attempts to ultrasonically examine the welds proved unsuccessful because of numerous indications which were detected. Those indications were attributed to the geometric and metallurgical condition of the materials comprising the welds, base material and cladding through which the ultrasonic beam must pass. The great number of indications precluded a meaningful interpretation of the data.

The technique currently being proposed by the licensee incorporates the use of a dual element, 45 degrees refracted longitudinal beam focused transducer, and is similar to the technique which was recently developed for the examination of piping which contains corrosion resistant cladding. The transducer active elements were designed to

match, as closely as possible, the acoustic impedance of austenitic stainless steel.

The previously attempted ultrasonic technique was demonstrated on a weld mock-up containing ID and OD notches. The mock-up was recently modified by the addition of two artificially induced cracks in the weld heat affected zone.

A demonstration of the new technique was performed by a Southwest Research Institute Level II individual, and was witnessed by the inspector. The purpose of the demonstration was to show that the two cracks can be detected, and that the crack indications are discernible above the noise level.

Equipment calibration was performed using a calibration block containing notches 10% of material thickness in depth, and system sensitivity was then increased by 6 dB. The resulting noise level in the mock-up maximized at approximately 25% of full screen height at the inside surface of the material. The examination makes use of the half arc technique, so that reflections are disregarded which appear beyond the I.D. location on the cathode ray tube.

The licensee successfully demonstrated that the two cracks can be detected ultrasonically, and that the crack signal to noise ratio was great enough to clearly display the crack signals.

The licensee plans to send the mock-up to the EPRI NDE Center at Charlotte, North Carolina, where it will be radiographed with the MINAC system. The results will be used to assess the relative merits of radiography and ultrasonics regarding detection of the cracks in the mock-up. Additionally, the licensee wants to use the ultrasonic technique on the field welds in the plant during the next outage.

The inspector stated that he agreed with the licensee's plan. He further stated that the preliminary results were promising, but that final judgement must be reserved pending examination of the field welds.

2.0 Review of Plant Operations

2.1 Operational Safety Verification

The inspector toured the control room daily to verify proper manning, access control, adherence to approved procedures, and compliance with LCOs. Instrumentation and recorder traces were observed and the status of control room annunciators were reviewed. Nuclear Instrument panels and other reactor protection systems were examined. Effluent monitors were reviewed for indications of releases. Panel indications for onsite/offsite emergency power sources were examined for automatic operability. During entry to and egress from the protected area, the inspector observed access control, security boundary integrity, search activities, escorting and badging, and availability of radiation monitoring equipment.

The inspector reviewed shift supervisor, plant control operator, and nuclear plant operator logs covering the entire inspection period. Sampling reviews were made of tagging requests, night orders, the bypass log, Significant Operating Occurrence Reports (SOORs), and QA nonconformance reports. The inspector observed several shift turnovers during the period.

2.2 Station Tours

The inspector toured accessible areas of the plant including the control room, relay rooms, switchgear rooms, cable spreading rooms, penetration areas, reactor and turbine buildings, security control center, diesel generator building, ESSW pumphouse, and the plant perimeter. During these tours, observations were made relative to equipment condition, fire hazards, fire protection, adherence to procedures, radiological controls and conditions, housekeeping, security, tagging of equipment, ongoing maintenance and surveillance and availability of redundant equipment.

2.3 ESF Walkdown - Unit 1 Core Spray System

On October 9 - 10, 1985, the inspector independently verified the operability of Unit 1 Core Spray System, Division II. This verification included a complete walkdown of all accessible system mechanical, electrical and instrumentation equipment to assess:

-- conformance of the "as-built" configuration to the system drawings and FSAR;

-- adequacy of the system mechanical and electrical equipment check-off lists;

-- proper position, and remote and local position indication, of valves and breakers, including valve locking devices and end caps;

-- operability of system instrumentation; and

-- conditions which might degrade system operation or performance.

In addition to the above, the inspector reviewed selected sections of Core Spray Logic Functional Test SE-151-001. The review was conducted to ensure that the test, as written, meets the intent of the technical specification and surveillance requirements. Test sections were compared to Bechtel Electrical Schematics and General Electric Elementary Diagrams to determine their accuracy.

The inspector determined that system alignment was consistent with the system check-off lists and the current mode of operation. System equipment was reasonably well maintained and identified. Portions of the surveillance reviewed were technically adequate.

A few minor discrepancies were noted and discussed with the licensee. The licensee initiated action to correct the discrepancies.

3. Summary of Operating Events

3.1 Unit 1

At 3:33 p.m. on October 28, the Unit 1 reactor scrammed from 100 percent power due to low reactor water level. While a Division II half-scram was inserted during routine surveillance testing of reactor vessel level instrumentation, a fuse blew to one of the four groups of Division I scram pilot solenoid valves, causing the rod group to insert. The resulting level transient caused a full reactor scram on reactor vessel low level. After investigation of the blown fuse, the Unit was restarted and criticality was reached at 5:10 p.m. on October 29. The scram ended a site record 138 day continuous run. (See Detail 3.4)

At 7:36 p.m. on October 30, the Unit 1 reactor scrammed from about 64 percent power on control valve fast closure due to a turbine trip. The cause of the turbine trip was high level in the 'B' moisture separator. After investigation and instrumenting the moisture separator, the unit was restarted, reaching criticality at 6:02 p.m. on November 5. After power escalation, the unit operated at full power for the remainder of the period. (See Detail 3.5)

3.2 Unit 2

The Unit 2 reactor scrammed at 8:37 a.m. on October 5 due to a ground fault in the 500 KV system grid caused by a lightning strike. During the switchyard transient one of the Unit 2 synchronizing breakers (2T) was slow to clear the fault, initiating a breaker failure relay scheme. This initiated a generator lockout and subsequent turbine trip. The reactor scrammed due to turbine control valve fast closure. After investigation of the switchyard transient, the startup was initiated at 12:20 a.m. October 6 and criticality was reached at 10:18 a.m. the same day. The Unit operated at or near full power for the remainder of the inspection period.

3.3 Turbine Building Spill

At 10:15 a.m. on October 26, 1985, the on-shift STA identified water flowing out from under the door of the 'B' Condensate Demineralizer room on the 676 foot elevation of the Unit 1 Turbine Building. The STA notified the control room, and operators were dispatched to the scene to isolate the leak. On initial entry into the room by an HP technician, it was noted that water and resin were blowing out of a vessel vent line flange, three of the four flange bolts were missing, and the flange gasket was blown out. Attempts to isolate the demineralizer vessel at the local control panel were unsuccessful and the manual block valves had to be closed to slow the leakage. The leak was not completely stopped until approximately 12:30 p.m. when a new gasket was installed. The spill volume was estimated to be approximately 12,000 gallons. The water and resin covered the 'B' demineralizer vessel room and the adjacent passageway in the Turbine Building, an area of about 3,000 square feet. A small amount of the water also drained down an equipment access hatchway to elevation 656. The water and resin were collected by the Turbine Building drains and processed by the Liquid Radwaste System. General area radiation levels in the demineralizer room were 15-50 MR/HR immediately following the spill. The area was promptly decontaminated and unrestricted access to the area was restored.

The cause of the leak has been attributed to the failure of a 1-1/4 inch vent line flange gasket on the 'B' demineralizer vessel. Maintenance had been performed on the demineralizer the week before, and the work included disassembly of the flanged connection. Reactor power had been reduced to 52 percent from 100 percent at 6:50 a.m. on October 26 in order to perform a rod sequence exchange. The power decrease caused an increase of approximately 150 psig in condensate pump discharge pressure, and may have initiated the gasket failure.

On October 18, 1985 Work Authorization (WA) S55179 was performed to open, clean and inspect the 'B' condensate demineralizer. The vessel was reassembled at approximately 11:00 p.m. October 18, 1985 after the inspection was completed. The work plan specified cutting and installing new manhole gaskets with "GARLOCK 3200 SBR, 3/32 inch gasket material or equal". The work plan was not specific concerning the material to be used for the flange gasket. The piping isometric drawing (SP-GBD-107-7) specifies a 1/16 inch compressed asbestos gasket for the flange. The licensee investigation following the spill determined that a red rubber gasket was actually installed during the flange reassembly. The construction mechanics installed a red rubber gasket based on previous practice with the other demineralizers. Two operational leak tests (approximately 470 psig) were performed on the system after reassembly and no leaks were found. At 4:47 p.m. October 24, the demineralizer was placed in service, and remained in service for approximately two days before the leak occurred. Licensee evaluation following the event determined that the red rubber was an acceptable material for this application.

A similar spill occurred on July 10, 1984 when the 'A' condensate demineralizer blew a gasket and 3,500 gallons of water and resin spilled on the floor and adjacent hallway (SOOR 1-84-278). The root cause was determined to be poor alignment of the vent lines, which required multiple (stacked) gaskets to obtain an adequate seal. Based on this occurrence and previous leakage problems, the vent lines were reworked to provide a proper flange alignment in March 1985. According to the associated WAs, compressed asbestos gaskets were installed following completion of this work. The Unit 2 demineralizers have the same alignment problem and the WAs have been written, but the work is currently scheduled for the Unit 2 First Refueling Outage (August 1986).

The inspectors toured several demineralizer rooms on both Unit 1 and Unit 2 and found that various materials are being used as gaskets for the flanges and manhole covers on the demineralizer vessels.

The results of the inspection were discussed with the Maintenance Supervisor. It appears that although this piping in non-Q, stronger controls need to be in place to ensure the correct gaskets are used for each application. Currently the only station guideline used is to "replace in kind". Additionally, the inspectors stated that review of the schedule for correction of the Unit 2 flanges may be warranted based on this event. The Maintenance Supervisor stated the Unit 2 work schedules would be reviewed for possible upgrade in priority, and an evaluation will be conducted concerning the need for a gasket control program.

3.4 Unit 1 Reactor Scram Due to Blown Fuse

At 3:33 p.m. on October 28, the Unit 1 reactor scrammed from 100 percent power due to low reactor water level. While a Division II half-scram was inserted during routine surveillance testing of the 'D' reactor vessel level instrumentation, a fuse blew to one of the four groups of Division I scram pilot solenoid valves, causing the rod group to insert. The resulting level transient caused a full reactor scram on vessel low level. No ECCS actuations occurred during the scram and no safety relief valves lifted. Just prior to the scram, I&C was performing quarterly surveillance test SI-180-305 on the 'D' reactor vessel level instrument (LIS-B21-N024D) which caused an expected half-scram on Division II. While the half-scram was inserted, fuse C72A-18A to the Group 1 scram pilot solenoid valves blew causing 1/4 of the control rods to scram. The partial scram caused a power decrease and rapid level decrease to the scram setpoint of 13 inches. The operators performed the immediate actions of emergency procedure EO-100-101 and stabilized the plant.

The inspectors observed the operator response and recovery from the control room and attended the post-scram meeting. All systems responded as designed during the scram except for the 'D' IRM which would not drive in. It was later found to have a blown fuse and was returned to service. Review of the sequence of events printout and GETAR's traces indicated that all systems responded as designed.

A similar scram occurred on Unit 1 on July 11, 1983. During I&C surveillance testing of the primary containment high pressure channel 'A', a fuse blew on the Group 4, System B control rod scram pilot solenoid valves. The blown fuse deenergized the Group 4, System B scram pilot solenoid valves and the I&C surveillance deenergized the System A scram pilot solenoid valves, resulting in the insertion of only Group 4 control rods. This 1/4 scram caused reactor power to decrease rapidly from 66 percent to 10 percent, which resulted in a rapid decrease in reactor vessel water level causing a full scram on low reactor vessel water level.

In response to the July 1983 scram an operations instruction was issued which requires the Plant Control Operator to station observers in the upper and lower relay rooms during RPS surveillance tests to verify RPS status light indications during the reset of half scrams. The cause of the blown fuse was found to be that an incorrect fuse type had been installed. All of the other fuses were inspected and found to be correct. At least three occurrences of blown fuses to the scram pilot solenoid valves have been documented. The two occurrences in Unit 1 noted above resulted in reactor scrams. One blown fuse was identified in Unit 2 prior to the start of a surveillance test in July 1985, and a scram was averted due to a precautionary check by the PCO.

The licensee is evaluating the possible causes for the fuse failures and is also investigating several spurious half-scrams and chattering relays that have occurred recently on Unit 2. The F18 fuses are slow blow type FNM15 (15 amp), and electrical maintenance measurements have determined that inrush current of 15 amps will occur occasionally when a half-scram is reset. Licensee review of the FNM time current curve found it suitable for the required slow blow service.

The results of the licensee evaluation will be reviewed in a subsequent inspection. (387/85-31-01)

3.5 Reactor Scram Due to 'B' Moisture Separator High Level

At 7:46 p.m., October 30, 1985, Unit 1 scrammed from 64 percent power due to turbine control valve fast closure on a turbine trip. The turbine tripped on high level in the 'B' moisture separator (MS). Conditions during the scram were normal. No safety relief valves lifted. The recirculation pumps tripped on the end of cycle recirculation pump trip (EOC-RPT) as designed. Lowest vessel level reached was 8 inches.

The inspector reviewed post-trip data, and monitored licensee's corrective actions. Plant computer data indicated that the 'B' MS drain tank level rose very rapidly from the normal level to the trip setpoint of 68 inches and remained above that level for greater than 10 seconds which causes a turbine trip. This corresponds to several hundred gallons of water. Other computer data (i.e. feedwater heater alarms, operation of MS drain valves, etc.) supports the licensee's conclusion that the level rise was due to actual water and not flashing in the drain tank level control system. I&C personnel performed calibration of the MS drain tank level control system which identified that one MS tank valve to the 4B feedwater heater was inoperable. However, this discrepancy does not account for the noted level rise.

During the investigation, it was determined that there was excessive spring can loading on the piping hangers for one of the 42 inch cross-around pipes between the HP turbine and the 'A' MS, indicating that there was water in the line. Manways for all six cross-around pipes were opened and the lines inspected. Some water was found in two of the cross-around pipes and the pipe which had its spring cans displaced was drained prior to opening it. The spring cans for this line returned to normal after draining. The drain valves on these six lines were verified to not be clogged.

The licensee was unable to positively identify the cause of the high MS drain tank level. It is surmised that the source of the water was one or more of the cross-around pipes, but the licensee was unable to identify a forcing function which would have caused the water to enter the MS at such a rapid rate. The plant was at steady state power at the time of the event and no anomalies were noted in plant parameters just prior to the event. The accumulation of water in the cross-around pipes was probably due to condensation which was not fully drained during the unit startup which began on October 29. It could not be determined, based on operator interviews whether the cross-around drain valves were opened and remained open when the turbine was shutdown following the October 28 scram.

The licensee instrumented the MS drain tank level control system to allow monitoring by the G.E. Transient Analysis Recording (GETARS) System. The general operating procedure, GO-100-003, was revised to specify opening the cross-around piping drains for 15 minutes at about 30 percent power, to ensure the pipes are drained. Thermocouples were attached to the drain lines to enable verifying that the drain valves open. Boosters were also installed on the MS emergency dump valves (LV-10232A and B) to enable quicker opening to help control level transients. This modification had been previously performed on Unit 2 following moisture separator level problems. (Inspection Report 50-387/84-38; 50-388/84-47).

The licensee returned the unit to operation on November 6 and escalated to full power. No further anomalies were noted on moisture separator drain tank levels.

4.0 Licensee Reports

4.1 In-Office Review of Licensee Event Reports

The inspector reviewed LERs submitted to the NRC:RI office to verify that details of the event were clearly reported, including the accuracy of description of the cause and 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

85-029-00, Inservice Inspection Surveillance Requirements Not Met

Unit 2

*85-025-00, Reactor Scram Caused by Lightning Strike on 500KV Transmission Line.

*Discussed in Detail 3.2.

4.2 Review of Periodic and Special Reports

Upon receipt, periodic and special reports submitted by the licensee were reviewed by the inspector. The reports were reviewed to determine that the report included the required information; that test results and/or supporting information were consistent with design predictions and performance specifications; that planned corrective action was adequate for resolution of identified problems, and whether any information in the report should be classified as an abnormal occurrence.

The following periodic and special reports were reviewed:

-- RCIC Injections (Unit 1) dated September 27, 1985

-- Monthly Operating Report - September 1985, dated October 14, 1985

4.3 10 CFR Part 21 Report - Yarway Valves

On September 26, 1985, Yarway Corporation provided a notification of a defect to Region I in accordance with 10 CFR Part 21. The notification stated that a cracked stem assembly was detected in a 3/4 inch Yarway Welbond valve at a non-nuclear facility. Five other valves exhibited leakage and Yarway concluded that the leakage was caused by a void in the bar stock used to manufacture the stems. The bar stock used to manufacture the stems is 5/8 inch round bar, AL Tech stainless steel type 416, ASTM A-562-75 condition T, heat number 93876. The report indicated that valves with stems from this bar stock were sent to PP&L. The inspector provided the Part 21 report to PP&L. On October 8, 1985, Yarway Corporation notified PP&L of the defective valve stems. Yarway indicated that 21 1/2-inch and 3 3/4-inch valves with defective stems were supplied to PP&L. The licensee reviewed the information and determined the location of the 24 valves. Of the 24 valves, eleven are in the warehouse, three are installed in the

Unit 2 RHR system, four are installed in Unit 2 RHR pump room cooling piping, four are installed in the Unit 2 RWCU system, one is installed in the Unit 1 offgas system, and one is unaccounted for. Non-Conformance Report (NCR) 85-0462 was prepared to document the condition of these valves. Yarway indicated that new valve stem/disc assemblies will be provided. PP&L intends to replace the installed valve stem/disc assemblies at the next available outage. The remaining valves in the warehouse were tagged with NCR tags and will be returned to Yarway. This issue will be reviewed following disposition of the valves. (388/85-26-01)

5.0 Monthly Surveillance and Maintenance Observation

5.1 Maintenance Activities

The inspector observed portions of 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 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 QC hold points were established where required; functional testing was performed prior to declaring the particular component 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.

Activities observed included:

-- 'D' ESW pump discharge check valve repairs (WA-S55279) performed on October 10, 1985

-- 'C' Diesel generator maintenance performed on October 16 - 18, 1985

-- Replacement of the Unit 2 'C' Reactor Steam Dome high pressure switch (B21-N023C) performed on October 30, 1985. The work was performed under WA-U57406.

5.2 Surveillance Activities

The inspector observed the performance of surveillance tests to determine that: the surveillance test procedure conformed to technical specification requirements; administrative approvals and tagouts were obtained before initiating the test; testing was accomplished by qualified personnel in accordance with an approved surveillance 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 included:

-- SO-100-006, Shiftly Surveillance Operating Log, performed on October 4, 1985

No unacceptable items were identified.

5.3 Emergency Diesel Generator Surveillances

The inspector conducted a review of selected emergency diesel generator (EDG) surveillance procedures to ascertain whether the surveillance of safety-related systems and components is being conducted in accordance with approved procedures as required by Technical Specifications.

5.3.1 18 Month Lockout Feature Surveillance

Unit 1 and Unit 2 Technical Specification Surveillance Requirement 4.8.1.1.2.d.13 states that at least once per 18 months, each of the diesel generators shall be demonstrated operable by verifying that the following diesel generator lockout features prevent diesel generator starting and/or operation only when required:

a) Engine overspeed

b) Generator differential

c) Engine low lube oil pressure

The Unit 1 and Unit 2 Technical Specification/Surveillance Procedure Cross Reference Matrix prepared by the licensee states that procedures SI-024-301 through 304 and SM-024-A02 through D02 implement the requirements of Technical Specification 4.8.1.1.2.d.13. The inspector reviewed the designated surveillance procedures to determine whether the Technical Specification surveillance requirement was adequately implemented.

The following items were identified:

-- Surveillance procedure SI-024-301, Revision 1, "18 Month Calibration of Diesel Generator 'A' Lube Oil Low Pressure PSL-03468A1, A2, A3, A4", performs testing on the low lube oil pressure switch for the diesel generator. The automatic circuitry is not functionally checked for proper operation (i.e. repeater relays and associated contacts).

-- Surveillance procedure SM-024-A02, Revision 0, "18 Month 4KV Diesel Generator 'A' Differential Relay Calibration", performs testing of the 4KV Diesel Generator differential lockout relay (87AG) by removing the relay and reinstalling it after a bench calibration. The automatic lockout feature circuitry is not tested during the surveillance.

-- The inspector could not identify any surveillance testing performed where the engine overspeed lockout feature has been functionally tested, or verified to be operational. The Technical Specification cross-reference also did not note any procedure to meet this requirement.

During discussions concerning the surveillance requirements with the licensee, the inspector learned that an NQA audit finding (Finding No. 0-84-33-01) was issued April 16, 1985 which apparently addressed the same issue. The finding stated that current plant surveillance procedures do not functionally test the three main trip relays for each start circuit as required by the Technical Specifications. At the close of this inspection period, the audit finding was still open, although some preliminary responses have been submitted by the responsible departments. The plant staff believes that the current calibration procedures are sufficient for the low lube oil pressure switches and differential relays. Additionally, the staff believes that verification of completion of the diesel generator load reject test (SE-024-A05) without an overspeed trip is sufficient to show that the overspeed trip will not occur when it is not required.

Since this apparent violation was previously identified by the licensee, a Notice of Violation will not be issued. The item will remain unresolved pending closure of the audit finding and completion of corrective action. (387/85-31-02)

5.4 Surveillance Procedure Review - SO-100-005

The inspector reviewed SO-100-005, Weekly Electrical Power Systems Distribution Operability Verification, to determine if all required AC and DC power distribution system divisions are verified to be energized as required by Technical Specification (TS) 4.8.3.1.1 and 4.8.3.2.1. Amendment 48 to the Unit 1 Technical Specifications and Amendment 14 to the Unit 2 Technical Specifications, which were issued in July 1985, added the 480 volt AC swing bus motor generator

(MG) set, transfer switch and preferred and alternate power sources to the Technical Specifications required for power distribution. When operations reviewed the amendments on July 12, 1985, they indicated that no surveillance procedure changes were required. Inspector review of the applicable surveillance, i.e. SO-100-005, indicated that the preferred power source, the MG set and the transfer switch were not specified in the procedure. The licensee was only verifying that the swing bus was energized and that its alternate source was available. In response to the inspector's concern, the licensee revised SO-100-005 and SO-200-005 to specifically include the MG set, transfer switch and the alternate power sources.

6.0 Reactor Coolant Spill and RWCU Isolation - Unit 1

6.1 Event Summary

On October 11, 1985, at 10:14 a.m. the reactor water cleanup (RWCU) system on Unit 1 isolated due to high differential flow and a spill of reactor coolant and resin occurred on the 779 foot elevation of the Unit 1 Reactor Building. Troubleshooting on the 'A' filter demineralizer (F/D) effluent control valve (FV-14566A) was in progress at the time of the spill. The spill was contained on the 779 foot level. A precautionary evacuation of non-essential personnel from both reactor buildings was ordered and the Technical Support Center (TSC) was manned. The licensee did not activate the Emergency Plan nor did their Emergency Plan procedures require them to do so. The precise source of the spill was not known. General area radiation levels for the 779 foot elevation were 100-250 mrem/hour, with levels up to 5 rad on contact with the resin. Airborne activity levels remained less than 1 MPC (maximum permissible concentration) until the spill was nearly cleaned up. Airborne levels reached 3.5 MPC for a short period as the resin dried out just prior to completion of the cleanup. The recovery action consisted of 1) isolating the spill source and stabilizing the plant, 2) cleaning up the spill, and 3) accessing the system damage, if any. It was determined that there was no offsite release. The shoes of one individual were contaminated but there was no other personnel contamination. After the gross contamination was cleaned up, Tech Staff performed a system walkdown and observed no physical damage. It was suspected that the spill came from a 150 psig relief valve in the precoat system. On October 12, a test procedure (TP-161-007) was performed to verify system integrity by introducing low pressure water to the low and high pressure portion of the system. In addition, a full system lineup was performed and the system was walked down at full system pressure. No problems were noted which could have led to the spill.

On October 13, at 5:34 a.m., the RWCU system was placed in service on the 'B' F/D. At 11:30 a.m., while attempting to place the 'A' F/D in service, the 150 psig precoat relief valve (PSV-14561) lifted, the RWCU system isolated and a minor spill (about 10 gallons) occurred. There was no radiological hazard associated with this spill since the 'A' F/D had been backwashed.

A few hours later, the RWCU system was restored on the 'B' F/D and the 'A' F/D was left isolated.

6.1 Event Sequence

10/11	5:08 a.m.	'A' RWCU filter demineralizer (FD) taken out of service on permit to work on FD flow control valve (FV-14566A).
	10:14 a.m.	Control room received RWCU hi-flow alarm followed by system isolation. Spill of reactor coolant and resin occurred on 779 foot elevation of Unit 1 Reactor Building.
	10:34 a.m. (Time approx.)	Control room received "High Radiation Reactor Building" alarm due to an alarm on an area radiation monitor (ARM) on 779 foot elevation.
	10:40 a.m.	Personnel evacuated from 779 foot elevation of Unit 1 Reactor Building.
	11:15 a.m.	Evacuation of unnecessary personnel from the Unit 1 Reactor Building and 779 foot elevation of Unit 2 Reactor Building ordered.
	11:10 a.m.	TSC ordered manned.
	11:50 a.m.	Unit 2 Reactor Building elevations 818, 799, and 779 ordered evacuated of non-essential personnel.
	12:05 p.m.	HP reported 100-250 mrem/hr general area radiation levels on 779; contamination levels of 10,000 dpm to 5 rads contact on resin. Airborne levels were about 75 percent of Maximum Permissible Concentration (MPC).
	12:25 p.m.	The shoe of one non-licensed operator was determined to be contaminated. There was no other personnel or clothing contamination.
	12:25 p.m.	ENS call to NRC.
	12:53 p.m.	HP reported that gross contamination (i.e. resin) was cleaned up.
	13:33 p.m.	TSC deactivated, spill cleaned up. No offsite release, RWCU remains isolated.

10/13	6:34 a.m.	RWCU system in service on 'B' F/D. Prior to placing in service, system walkdown, valve lineup and pressurization test were performed to verify system integrity.
	11:32 a.m.	RWCU isolation on high differential flow when placing the 'A' F/D from precoat cycle to hold. The 150# precoat feed relief lifted causing a minor spill. Spill was stopped when operator locally isolated the 'A' F/D.

6.3 Details

On October 11, just prior to the spill, the RWCU system was in service on the 'B' F/D and the 'A' F/D was tagged out to troubleshoot the F/D effluent flow control valve (FV-14566A) under WA S55381. The valve appeared to be bound in the open position. In order to stroke check the valve, an operator pushed the "Cleaning Cycle Start" pushbutton at the F/D control panel and manually stepped through the F/D cleaning cycle sequence by turning a thumbwheel on the programmer in the back of the F/D control panel. Following completion of the cleanup cycle, the operator then pushed the "Precoat Cycle Start" pushbutton and manually stepped through the precoat sequence. Following completion of the precoat cycle (Step 15 of the programmer), the operator turned the ISOLATE/DEISOLATE switch to DEISOLATE, which causes the F/D block valves (HV-14531A and HV-14532A) to open. The operator and other individuals (maintenance mechanics and the Assistant Unit Supervisor) heard flow. They observed that the 'B' F/D flow increased rapidly and the operator attempted to reduce the flow by operating the 'B' F/D effluent flow control valve. The operator then turned the ISOLATE/DEISOLATE switch to ISOLATE, pushed the HOLD pushbutton, and the individuals left the area as water was coming out of the floor drains.

The control room received the RWCU Hi-Flow alarm followed by system isolation and were notified of the leak. A few minutes later, a Reactor Building High Radiation alarm was received which was determined to be due to an area radiation monitor at the sample station on the 779 foot elevation. The Control Room ordered evacuation of unnecessary personnel from 779 foot elevation and later from all Unit 1 Reactor Building. The control room also began vacuum dragging the RWCU system to the condenser to reduce system pressure.

Initial radiological conditions were determined to be 150 mrem/hr general area and it was noted that resin was on the floor. The licensee manned the Technical Support Center (TSC) but did not activate the Emergency Plan (the event did not meet Emergency Plan entry conditions). A precautionary evacuation of unnecessary personnel from the Unit 2 Reactor Building 818, 799 and 779 foot elevations was ordered. Radiation levels increased to 150-250 mrem/hr general area, contamination levels up to 5 rad on contact with resin and airborne levels reached 75% of MPC for affected nuclides.

The isolation of the RWCU system stopped the spill. Health Physics personnel entered and cleaned up the gross contamination. There was no offsite release. One shoe of the operator at the F/D control panel was contaminated, but there were no other personnel contaminations. Following cleanup of the spill, Technical Staff engineers walked down the system and verified that there was no system damage. On October 12, the licensee performed a test procedure to introduce low pressure water from the condensate system to the low and high pressure portions of the system to check for damage. None was found, although leakage past one of the system isolation valves had already repressurized the system. The system was vented at high points, and the 'A' and 'B' F/D were backwashed. On October 13, the 'B' F/D was precoated and placed in service. At 11:32 a.m. October 13, while attempting to place the 'A' F/D in service, a RWCU system high leakage alarm occurred, the system isolated and the precoat relief valve (PSV-14561) lifted causing a minor (10-15 gallon) spill. There was no radiological hazard from this spill since the 'A' F/D had been backwashed. The operator had positioned the ISOLATE/DEISOLATE switch to DEISOLATE following the precoat cycle when this spill occurred. The operator immediately returned the switch to ISOLATE when he heard flow. The 'B' RWCU F/D was returned to service and the 'A' F/D remained isolated.

Plant staff and the Nuclear Safety Assessment Group performed an investigation of the incident. The cause of the October 11 spill was apparently due to a limit switch problem with the 'A' F/D effluent flow control valve (HV-14566A). The full closed limit switch on the HV-145066A energizes a relay if the valve is not fully closed, which causes the F/D influent valve (HV-14506A) to open. The limit switch collar on the HV-14566A was found to be loose and it apparently caused HV-14506A to open. When the operator deisolated the vessel, a flow path existed to allow reactor coolant to reach the 150 psig precoat piping, lifting the 150 psig precoat relief valve. Normally, prior to deisolating the F/D, a backwash cycle would have been completed. However, on October 11, the operator was manually stepping through the programmer which did not allow the backwash to be performed. The 'A' F/D had been backwashed and returned to service on October 10. The activated resin, which spilled on October 11 causing the radiological hazard, had only been in service for one day. Normally, a F/D is inservice for a longer period (nominally six to eight days) prior to backwashing which implies that consequences of a spill could have been more severe.

The minor spill on October 13, was apparently due to gross leakage by HV-14566A. This valve is an air operated globe valve manufactured by VALTEK, and the valve contains soft seats. A leak check of the valve on October 16, identified gross leakage. When the valve was dismantled, the soft seats were missing, accounting for the gross leakage. The valve was repaired.

The programmer design requires that the F/D vessel be deisolated (i.e. the isolation valves, HV-14531A and HV-14532A, opened) prior to returning to hold. The HOLD pushbutton does not receive power until the vessel is deisolated following the precoat sequence, in the existing design. Therefore, the F/D is in precoat with low pressure piping connected to the vessel, when the vessel is deisolated. A failure of one valve (i.e. the F/D influent or effluent valves) which is not designed as an isolation valve, will cause a spill. The licensee modified the programmer circuitry under DCP 85-9054 to remove the block valves and operation of the HV-14506A valve from the programmer. This change ensures that two valves in series will isolate the F/D precoat system from reactor pressure. However, based on discussions with the licensee for Limerick, and Graver (the F/D control system manufacturer), this is a generic problem and will be pursued separately.

Following valve leak checks and repairs to HV-14566A, the 'A' F/D was returned to service on October 22. The Unit 1 RWCU operating procedure OP-161-001 was revised to reflect the modification and to ensure manual isolation valves to low pressure piping (i.e. F/D vent piping and precoat piping) are shut prior to deisolating the F/D vessel. The Unit 2 procedure was also revised to ensure the manual isolation valves are shut, however, the modification has not been completed on Unit 2.

In the inspector's view, the licensee's response to this event, including the immediate response, spill cleanup, and actions to assess RWCU system status and return the system to service, were thorough and conservative. The spill was quickly isolated, the spread of contamination minimized, and the spill was cleaned up with no personnel contamination and very little personnel exposure. Several PORC meetings were held on October 11 and 12 to determine recovery actions, a system walkdown and complete valve lineup were performed, and test procedures prepared to verify system integrity. These steps were taken prior to any attempt to restore the system. The licensee closely monitored reactor coolant chemistry using the post-accident sampling system while the RWCU system was out of service. Although a second, minor spill occurred on October 13, the licensee took prudent actions to prevent this occurrence and there was no hazard from the second spill.

At the time of this report, the licensee's investigations had not been completed and further corrective actions have not been determined. The inspector will review the LER and investigation reports when they are issued. (387/85-31-03)

7.0 Exit Meeting

On November 15, 1985 the inspector discussed the findings of this inspection with station management. 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.