U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 88-05

Docket No. 50-333

License No. DPR-59

Licensee: Power Authority of the State of New York
P.O. Box 41
Lycoming, New York 13093

Facility: J.A. FitzPatrick Nuclear Power Plant

Location: Scriba, New York

Dates: March 8, 1988 - April 30, 1988

Inspectors: A.J. Luptak, Senior Resident Inspector
R.A. Plasse, Jr., Resident Inspector
R.A. Laura, Reactor Engineer

Approved by: Jon R Johnson 5/17/88
J.R. Johnson, Chief, Reactor Projects Section 2C, DRP — Date

Inspection Summary:

Areas Inspected:

Routine and reactive inspection during day and backshift hours of Licensee Event Report review, operational safety verification, surveillance observations, maintenance observations, followup of previous inspection findings, Bulletin followup, followup of a Part 21 report, and review of periodic and special reports. This involved a total of 229 inspection hours which included 3 hours of backshift on March 25, and 22 hours of weekend/holiday inspection coverage on March 13, 20, 27 and April 9 and 10, 1988.

Results:

No violations were identified during the inspection period. Inconsistences were noted in conducting surveillance testing as described in section 12. Section 4 details an unclear Technical Specification concerning the acceptance criteria for Containment Integrated Leak Rate Tests.

DETAILS

1. Persons Contacted

During this inspection period, the inspector interviewed or held discussions with operators, technicians, maintenance, contractor, engineering, administrative, and supervisory personnel.

2. Summary of Plant Activities

The inspection period began with the plant at full power. There was a power reduction to 86% on March 12 to support a rod sequence exchange. An additional day at this power level occurred due to repairs to the B Traversing Incore Probe machine. On March 14, power was reduced to 66% to troubleshoot a ground caused by the B Reactor Feed Pump turbine. The plant attempted to return to full power on March 15, but was delayed due to a B Reactor Feed Pump speed control problem. On March 16, power was again reduced to 66% to allow for B Reactor Feed Pump turbine repairs. Repairs were completed and the plant returned to full power on March 18. The plant remained at full power throughout the remainder of the inspection period.

3. Review of Plant Events (71707, 92700)

On March 10, the licensee declared the High Pressure Coolant Injection System (HPCI) inoperable when the HPCI turbine failed to start during routine surveillance testing. The turbine did not start because the HPCI turbine steam supply valve 23MOV14 failed to open after an automatic initiation signal was generated as part of the logic system functional test. The licensee determined that the motor of 23MOV14 had failed, due to excessive peak motor current required for unseating the valve in the open position. Investigation of the high current revealed that the valve stem and stem nut threads did not have adequate lubrication, preventing smooth valve operation.

Maintenance records revealed that 23MOV14 was repacked on February 24 and retested satisfactorily on February 25. Review of the "Valve Repacking" procedure MP-59.9 revealed that it did not include instructions to determine if the valve stem and stem nuts were adequately lubricated. 23MOV14 was repacked to correct excessive packing leakage. It is believed that the packing leakage washed away the existing lubrication. The licensee replaced the failed motor, relubricated the valve stem and stem nut threads, and revised the maintenance procedure to require inspection, cleaning and lubrication of valve stem and stem nuts during repacking of manual and power operated valves to prevent a reoccurrence.

On March 10 while performing Automatic Depressurization System (ADS) surveillance testing as required by technical specifications due to HPCI being inoperable, a Reactor Core Isolation Cooling System (RCIC) isolation occurred due to personnel error. The isolation was the result of the technician inputting a signal into the RCIC high steam flow trip unit 13 MTU 283 instead of the reactor vessel low level trip unit 2-3 MTU 283A as required for the ADS testing. The cause of the isolation was determined, the isolation signal reset, and the RCIC system restored to a standby lineup within 5 minutes.

On April 18 the A Reactor Protection System Motor Generator (RPS MG) tripped causing a loss of the A RPS bus. This loss resulted in a half scram, half group one isolation signal and caused the isolation of the Reactor Water Cleanup, Containment Atmosphere Dilution, and Reactor Building Ventilation systems and started the A Standby Gas Treatment system. The A RPS power supply was transferred to alternate and all systems restored to normal. The RWCU system remained secured to perform scheduled maintenance.

The A RPS MG A tripped when the MG drive motor protective relay for motor overcurrent or high temperature deenergized due to an open circuit in the relay coil. The MG drive motor did not experience an abnormally high current or temperature condition. Following repair of the RPS MG by replacing the failed relay, loads were transferred from the alternate power source to the MG (12 hours after failure of the relay).

The relay coil failure is considered to be a random or age-related failure. The relay coil is normally energized and had been in service for approximately 13 years. A computer search of the Nuclear Plant Reliability Data (NPRD) system did not indicate frequent or unusual problems associated with relays of the same manufacture and type as the failed relay.

No violations were identified during this review.

4. Previous Inspection Findings (92701)

(Closed) INSPECTOR FOLLOWUP ITEM (80-15-07): Review of licensee followup on failure of the Standby Gas Treatment System caused by water in a discharge line common to both trains. The inspector reviewed the licensee's followup report (JTS-88-0102) and found it to be adequate. The licensee is implementing quarterly preventive maintenance to check the drain line for evidence of excessive silt/corrosion buildup. This item is considered closed.

(Closed) INSPECTOR FOLLOWUP ITEM (82-19-06): Review results of licensee investigation as to why the B Standby Liquid Control continuity circuit did not provide a loss of continuity indication with explosive valve fired. Results of licensee investigation (including input from the vendor) reveals that the cable running from the control room to the squib valve tends to act as a "capacitor" and stores a small amount of charge large enough to prevent the circuit from reaching the 3 millamp threshold. The corrective action will be to modify the loss of continuity indication circuit by installing a diode. The diode will prevent build-up of stored charge in the squib cable. Plant modification F1-87-155, which will modify the circuit, is in the conceptual stage of design and is not yet available for review. The modification is currently scheduled for completion during the August 1988 refueling outage. This item is closed.

(Closed) INSPECTOR FOLLOWUP ITEM (85-20-07): Marginal Emergency Lighting conditions. The inspector reviewed emergency lighting installation and determined that the current installation meets the requirements of 10 CFR 50, App. R, Section III.J. This item is closed.

(Closed) INSPECTOR FOLLOWUP ITEM (83-06-02): Correct discrepancies in Core Spray System Drawings and valve lineups. The licensee's corrective actions were verified. This item is closed.

(Closed) UNRESOLVED ITEM (82-04-02): Evaluation of Primary Containment Integrated Leak Rate Test Acceptance criteria. During the 1982 inspection, the inspector noted that the licensee was using a higher acceptance criteria of 0.5%/day versus a value of .375%/day which was used during the preoperational and 1978 PCILRT. During the 1982 inspection, it was noted that discrepancies existed in this area in the Technical Specification and FSAR. The inspector reviewed F-ST-39F, Type "A" Test (60 psig), Primary Containment Integrated Leakage Rate Test, and noted the licensee has been using an acceptance criterion of less than .375%/day since 1985. This item is closed.

However, the inspector raised concerns regarding the Technical Specification in this area. TS 4.7.A.2.a(8) states that the leakage rate acceptance criteria at peak pressure shall be less than 0.75 (La) and not greater than Ld, which is 0.5 weight percent of contained air per 24 hours at the test pressure. Within the plant's TS, the terms Ld and La are not defined. 10 CFR 50, Appendix J, section II defines La as the maximum allowable leakage rate at peak pressure as specified for preoperational tests in the Technical Specifications or associated bases, and as specified for periodic tests in the operating license. Ld is defined as the design leakage rate at peak pressure as specified in Technical Specifications or associated bases.

In the bases section for Technical Specifications 4.7.A several statements are made.

a. "The design basis accident leakage rate is 0.5 percent/day at a pressure of 45 psig."

b. "The design basis loss-of-coolant accident was evaluated in FSAR, Section 14.6, incorporating the primary containment maximum allowable accident leak rate of 1.5 percent/day". This statement was used to determine the offsite dose calculation to meet 10 CFR 100 requirements.

c. "The maximum allowable test leak rate at the peak pressure of 45 psig (Pa) is 0.5 weight percent per day (Lam)."

10 CFR, Appendix J, III, 5.b, under acceptance criteria, states the leakage rate Lam (measured leakage rate at peak pressure) shall be less than 0.75 La. Based on the inspector review, it appears the licensee is procedurally meeting 10 CFR, App. J, requirements, however, the TS and bases are unclear.

The licensee began a proposed TS amendment (PTS 84-1?) in 1984 to investigate this issue. During a recent discussion with the licensee concerning long standing TS problems this item was noted as still being open. The licensee stated that more effort will be placed in resolving these outstanding issues.

5. <u>Licensee Event Report (LER) Review</u> (90712)

The inspector reviewed LERs to verify that the details of the events were clearly reported. The inspector determined that each report was adequate to assess the event, the cause appeared accurate and was supported by details, corrective actions appeared appropriate to correct the cause, and generic applicability to other plants was not in question.

During this inspection period, the following LERs were reviewed:

LER 88-01, High Pressure Coolant Injection (HPCI) was made inoperable when steam supply valve 23-MOV-14 failed to open during surveillance testing. Followup of this event is discussed in section 3.

LER 88-02, Reactor Core Isolation Cooling (RCIC) was isolated for approximately one minute while High Pressure Coolant Injection (HPCI) was also inoperable due an error during surveillance testing. Followup of this event is discussed in sections 3 and 12.

No violations were identified.

6. <u>Emergency Notification System Reports</u> (92700)

The inspector reviewed the following events which were reported to the NRC via the Emergency Notification System as required by 10 CFR 50.72. The review included a determination that the reporting requirements were met, that appropriate corrective actions had been taken, and that the event had been evaluated for possible generic implications.

The following reports were reviewed:

Event Date	Subject
March 10, 1988	High Pressure Coolant Injection (HPCI) declared inoperable when the HPCI turbine failed to start during routine surveillance testing.
March 10, 1988	While performing Automatic Depressurization System (ADS) surveillance testing as required by Technical Specifications due to HPCI being inoperable, a Reactor Core Isolation Cooling System (RCIC) isolation occurred due to personnel error.
April 18, 1988	A Reactor Protection System Motor Generator (RPS MG) tripped causing a loss of the A RPS bus. This loss resulted in a half scram, half group one isolation signal; causing isolation of Reactor Water Cleanup, Containment Atmosphere Dilution, and Reactor Building Ventilation Systems and starting of the A Standby Gas Treatment systems.

No violations were identified.

7. Operational Safety Verification (71707)

a. Control Room Observations

Daily the inspector verified selected plant parameters and equipment availability to ensure compliance with Technical Specifications limiting conditions for operation. Selected lit annunciators were discussed with control room operators to verify that the reasons for them were understood and corrective action, if required, was being taken. The inspector observed shift turnovers biweekly to ensure proper control room and shift manning. The inspector directly observed the operations listed below to ensure adherence to approved procedures:

-- Routine Power Operations.

-- Ascension to full power after completion of B Reactor Feed Pump Turbine maintenance.

-- Issuance of Radiation Work Permits and Work Request/Event/Deficiency forms.

No violations were identified.

b. Shift Logs and Operating Records

Selected shift logs and operating records were reviewed to obtain information on plant problems and operations, detect changes and trends in performance, detect possible conflicts with Technical Specifications or regulatory requirements, determine that records are being maintained and reviewed as required, and assess the effectiveness of the communications provided by the logs.

No violations were identified.

c. Plant Tours

During the inspection period, the inspector made tours of control rooms and accessible plant areas to monitor station activities and to make an independent assessment of equipment status, radiological conditions, safety and adherence to regulatory requirements.

No violations were identified.

d. Tagout Verification

The inspector reviewed the following safety-related protective tagout records (PTRs) to verify that breakers, switches and/or valves were in the required positions.

-- PTR 880763 on the Containment Atmosphere Dilution System.
-- PTR 880852 on the Control Rod Hydraulic System.
-- PTR 880848 on the High Pressure Coolant Injection System.

No violations were identified.

e. Emergency System Operability

The inspector verified operability of the following systems by ensuring that each accessible valve in the primary flow path was in the correct position, by confirming that power supplies and breakers were properly aligned for components that must activate upon an initiation signal, and by visual inspection of the major components which might prevent fulfillment of their functional requirements:

-- High Pressure Coolant Injection System.

-- Reactor Core Isolation Cooling System.

-- A Core Spray System.

No violations were identified.

8. Surveillance Observations (61726)

The inspector observed portions of the surveillance procedures listed below to verify that the test instrumentation was properly calibrated, approved procedures were used, the work was performed by qualified personnel, limiting conditions for operations were met, and the system was correctly restored following the testing.

-- F-ST-4E, HPCI Logic System Functional Test, Rev. 23, dated December 22, 1987, performed March 11, 1988.

-- F-ST-1D, MSIVs, Main Team Line Drain Valves and Reactor Water Sample Valves Logic Functional Test, Rev. 19, dated December 30, 1987, performed March 24, 1988.

-- F-ST-100C, Reactor Protection System and Primary Containment Rev. Isolation System Instrument Functional Test/Calibration (ATTS), Rev. 7, dated January 8, 1988, performed March 29, 1988.

-- F-ST-15G, Pressure Suppression Chamber, Reactor Building Vacuum Breaker Differential Simulated Auto Actuation and Setpoint Test (IST), Rev. 5, dated September 30, 1987, performed April 8, 1988.

The inspector also witnessed all aspects of the following surveillance test to verify that the surveillance procedure conformed to specification requirements and had been properly approved, limiting conditions for operation for removing equipment from service were met, testing was performed by qualified personnel, test results met technical specification requirements, the surveillance test documentation was reviewed, and equipment was properly restored to service following the test:

-- F-ST-4B, HPCI Flow Rate/HPCI Pump Operability/HPCI Valve Operability Tests (IST), Rev. 32, dated January 8, 1988, performed March 22, 1988.

No violations were identified.

9. Maintenance Observations (62703)

a. The inspector observed portions of various safety-related maintenance activities to determine that redundant components were operable, that these activities did not violate the limiting conditions for operation, that required administrative approvals and tagouts were obtained prior to initiating the work, that approved procedures were used or the activity was within the "skills of the trade," that appropriate radiological controls were properly implemented, that ignition/fire prevention controls were properly implemented, and that equipment was properly tested prior to returning it to service.

b. During this inspection period, the following activities were observed:

-- WR 71/56334, Troubleshoot ground on 'B' 125v battery.
-- WR 31/57080, Troubleshoot 'B' Reactor Feed Pump Control Oil System.
-- WR 08/52273, Remove irradiated material from fuel pool.
-- WR 03/57935, Troubleshoot B Control Rod Drive Hydraulic Pump.

No violations were identified.

10. Licensee Action on NRC Bulletins (92701)

The inspector reviewed licensee records relating to the NRC Bulletin identified below to verify that the NRC Bulletin was received and reviewed for applicability; written responses were provided if required and the corrective action taken was adequate.

BU 88-01, Defects in Westinghouse Circuit Breakers. The purpose of this bulletin dated February 5, 1988 was to provide information on Westinghouse series DS circuit breakers and safety concerns associated with their use and to request additional inspections if the licensee was utilizing this type breaker in Class 1E service.

The licensee review determined that none of the subject Westinghouse DS series breakers are in Class 1E service at the plant. The inspector has no further questions and no concerns were identified.

11. 10 CFR Part 21, Followup of Degradation of Aluminum Backed Insulation (92701)

Generic Letter 85-22 and Regulatory Guide 1.82 addressed the concern of Loss of Coolant Accident (LOCA) generated debris clogging suction strainers for the ECCS pumps. A recent 10 CFR, Part 21 report was submitted by Pennsylvania Power an Light Company (PP&L) concerning the use of a fiberglass based, aluminum foil laminated cover over insulation blankets, in the drywell of the Susquehanna, Unit 2 site. PP&L found that the aluminum was delaminating and that during a LOCA, large quantities of loose aluminum could become debris. Insulation debris could then clog suction strainers for the ECCS pumps.

FitzPatrick has had an insulation replacement program in place for several years for the following reasons: removal of all asbestos insulation, replacement of deteriorating insulation, reduction of the drywell heat load, and installation of insulation which was easier to remove and reinstall for ISI inspections.

To date, approximately 60-70% of the insulation in the drywell has been replaced with insulation which meets Generic Letter 85-22 and Regulatory Guide 1.82 requirements. After completion of a review of the ALARA video record of the drywell, the licensee has determined approximately 10% of the remaining insulation has aluminum foil. The remaining insulation is being replaced on an as needed basis each refueling outage. The aluminum foil backed fiberglass cloth used at FitzPatrick is similar in design to that used at Susquehanna, but was manufactured by a different vendor. In addition, FitzPatrick has not encountered a delamination problem as described by Susquehanna. The licensee has initiated a work request to identify all fiberglass insulation covers with aluminum foil laminate in the drywell. The resident inspector will followup this drywell inspection when it is performed.

No discrepancies were identified.

12. Assurance of Quality

This section is included to provide an assessment of worker activities and management oversight and effectiveness in ensuring activities are conducted in a manner which assures quality.

The inspectors observed numerous surveillance tests performed by operations and I&C personnel over the course of the inspection period. All tests were directed from the control room by approved procedures. In general, the plant procedures are well written and give clear, concise directions. Monitoring has shown tests to be well conducted with watchstanders being attentive to duties and following directions with additional copies of approved procedures.

However, the inspector has noted exceptions to this conduct of testing which requires management attention. There are no specific requirements that all watchstanders have a copy of the procedures being performed, or for personnel to repeat orders back to the control room operator directing the test. An I&C technician caused a Reactor Core Isolation Cooling (RCIC) isolation (LER 88-02) when he inserted a test signal to the wrong trip unit. The step being performed in the test was clear and was directed by the control room operator verbatim. The I&C technician was inattentive to the order given and also did not have a copy of the procedure to verify he was to manipulate the correct trip unit prescribed by the procedure.

This concern has been discussed with the licensee. The inspector will continue to monitor onsite management's activities in this area.

In addition, attention is needed to resolve numerous longstanding Technical Specification concerns, one of which is described in section 4.

13. Review of Periodic and Special Reports (90713)

Upon receipt, the inspector reviewed periodic and special reports. The review included the following: inclusion of information required by the NRC; test results and/or supporting information consistent with design predictions and performance specifications; planned corrective action for resolution of problems, and the reportablility and validity of report information. The following periodic reports were reviewed:

-- February 1988, Operating Status Report, dated March 9, 1988.
-- March 1988, Operating Status Report, dated April 8, 1988.

No unacceptable conditions were noted.

14. Exit Interview (30703)

At periodic intervals during the course of this inspection, meetings were held with senior facility management to discuss inspection scope and findings. In addition, at the end of the period, the inspector met with licensee representatives and summarized the scope and findings of the inspection as they are described in this report.

Based on the NRC Region I review of this report and discussions held with NYPA representatives during the exit meeting, it was determined that this report does not contain information subject to 10 CFR 2.790 restrictions.