U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 88-01

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: January 12, 1987 - March 7, 1988

Inspectors: A. Luptak, Senior Resident Inspector
R. Plasse, Jr., Resident Inspector
H. Gray, Lead Reactor Engineer

Approved by: Joseph Kolaczyk acting 3/25/88
J. Johnson, Chief, Reactor Date
Projects Section 2C, DRP

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, Engineered Safety Feature system walkdown, licensee reorganization, Technical Specification clarity concerns, Fitness for Duty policy, Generic Letter 84-11 followup, and review of periodic and special reports. This involved a total of 280 inspection hours which included 12 hours of backshift on January 23, and February 15, 23, 1988, and 29 hours of weekend/holiday inspection coverage on January 16, 18, 23, 24, 1988, and February 14, 27, 1988.

Results:

One violation was identified involving failure to maintain Low Pressure Coolant Injection Manual Isolation Valve locked open as required (section 4.c). Areas of concern were noted in the improper hanging of tags during replacement of valves associated with High Pressure Coolant Injection (HPCI) high level trip instruments (section 4.d). Several concerns were identified with respect to the adequacy of technical specification requirements (sections 10 and 11). It was determined that the licensee has met the requirements of Generic Letter 84-11 concerning inspections of austenitic stainless steel piping welds susceptible to intergranular stress corrosion cracking (IGSCC) (section 12). Their efforts in this area have been well planned and extensive.

DETAILS

1. Summary of Plant Activities

The inspection period began with the plant in cold shutdown with a 2 week maintenance outage in progress. Major work accomplished during this outage involved: replacement of 16 control rod drive mechanisms, inspection of the torus coating with no significant findings, recirculation scoop tube modifications, and preventive maintenance on electrical equipment.

On January 23, 1988, the licensee began a plant startup. During an inspection of the drywell at 500 psig reactor pressure, leakage from a Reactor Water Cleanup System (RWCU) weld was found. A plant shutdown was conducted to support RWCU weld repair. Upon satisfactory completion of RWCU weld repair, a reactor startup was conducted on January 26.

Full power operation was achieved January 31, 1988 and was continued throughout the inspection period.

2. Previous Inspection Findings (92701)

(Closed) INSPECTION FOLLOWUP ITEM (78-05-07): The basis for Technical Specification (TS) 1.1.B is unclear. This item concerns the fact that the TS basis assumes a static head of 4.56 psi will ensure a bundle flow of 28×10^3 lbs/hr. The licensee is reviewing this issue and will clarify the basis during the next reload analysis to be completed August 1988. This item is being tracked by the licensee, is considered to be of minimal safety significance, and is considered closed.

(Closed) INSPECTOR FOLLOWUP ITEM (78-05-09): Technical Specification (TS) 4.6 G.1 conflicts with its basis. TS 4.6.G.1 requires flow between the two recirculation loops to be within 15% when the pumps are at the same speed. The basis for TS 4.6.G.1 stipulates the flow should be within 10%. The inspector reviewed ST-23C, Jet Pump Operability Test for Two Loop Operation, which requires the flows to be within 10%. The licensee has been enforcing the more conservative value by this procedure and plans to submit a proposed amendment change to change the TS requirement to 10% in April 1988. This item is closed.

(Closed) UNRESOLVED ITEM (78-13-05): Calibration program to determine applicability of RTD calibrations. The licensee has collected data which includes all the RTDs in use presently at the site. Many of these RTDs, and in particular instruments which supply safety system functions, are being checked for operability and accuracy. The licensee is in the process of generating a generic procedure to check all RTDs for operability and accuracy. This item is closed.

(Closed) UNRESOLVED ITEM (79-04-07): Environmental qualifications of stem mounted limit switches (SMLS). The licensee was scheduled to replace the Main Steam Isolation Valve (MSIV) SMLS and review the qualification of reactor water sample valves SMLS. The inspector verified the MSIV SMLS were replaced with qualified SMLS. He also verified that the reactor water sample SMLS were required to be qualified under Regulatory Guide 1.97 which the licensee is required to complete during the refueling outage scheduled for August 1988. The reactor water sample SMLS will be replaced with qualified SMLS during the outage. This item is closed.

(Closed) UNRESOLVED ITEM (81-02-04): Reevaluation of meeting NUREG 0578, Item 2.1.4, Containment Isolation Provisions. The inspector verified that the licensee had installed modification F1-85-92, which supplied automatic containment isolation valves for the Transversing Incore Probe purge line and Recirculation Pump Mini-Purge System. In addition, the inspector reviewed proposed modification F1-87-68, which will add automatic containment isolation valves in the Reactor Water Cleanup system. This modification will be installed during the refueling outage scheduled to begin in August 1988. This item is closed.

(Closed) UNRESOLVED ITEM (81-18-01): Evaluate the independence of the drywell atmospheric radiation monitor. The inspector reviewed the design package for modification F1-87-45 which will install a redundant supply and return line for the drywell atmospheric radiation analyzers. This modification will be installed during the refueling outage scheduled to begin in August 1988. This item is closed.

(Closed) INSPECTION FOLLOWUP ITEM (83-BU-02): Stress corrosion cracking in stainless steel piping. The licensee program for intergranular stress corrosion cracking detection and mitigation was reviewed as discussed in section 12. This item is closed.

3. Licensee Event Report (LER) Review (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 87-11-1, is a supplemental report involving electrical penetrations which were not properly sealed as fire barriers. This LER reported the results of mechanical penetration inspection and corrective actions which were described in the initial LER.

LER 87-21 reported a Reactor Water Cleanup system isolation due to high area temperature. Followup of this event was discussed in Inspection Report 50-333/87-26.

No violations were identified.

4. 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.

-- Plant startups on January 23 and January 26, 1988.

-- 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.

On January 23, 1988, during a drywell tour while the reactor was critical and a plant startup was in progress, the inspector found the handwheel for 10-RHR-81A, the Low Pressure Coolant Injection Manual Isolation Valve, on the grating beneath the valve with the

lock and chain used to secure the valve hanging from the railing. Operation Procedure 13 and Surveillance Test 40G requires 10-RHR-81A to be locked open.

The inspector notified the shift supervisor and repairs were made to the valve handwheel and the valve was locked open. In addition, the three other valves required to be locked inside the drywell were checked and verified to be locked.

A licensee critique of the occurrence concluded that the valve handwheel was loose based upon operators initially having to push the handwheel on to begin valve operation. Once valve stroking was begun, no additional problems were encountered. No work request was initiated to repair the loose handwheel. The individuals which opened the valve in restoring from the Protective Tagout were unclear as to whether they had locked the valve or not. The required position listed on the tagout restoration called for the valve to be open but did not stipulate it to be locked open. In addition, the licensee identified the need to emphasize the independence of the second person verification and restoration of system required lineup or use of special condition tags if the situation warrants.

This is the second instance in three months when the inspector discovered a valve not locked as required. As documented in Inspection Report 87-22, October 21 - November 30, 1987, a non-safety related manual isolation valve on the Control Rod Drive Hydraulic system was not locked as required. A Notice of Violation was not issued for that instance since it appeared to be an isolated case and was of minimal safety significance.

The failure to lock open 10-RHR-81A is a violation of Technical Specification 6.8 which requires procedures and policies be implemented to provide for the control of equipment (333/88-01-01).

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 880017, on Hydraulic Control Unit 22-23.

-- PTR 880119, 880166 on A Residual Heat Removal System.

-- PTR 880228, on High Pressure Coolant Injection System Reactor High Water Level Switches.

During a walkdown of PTR 880228 on LIS 101D, Reactor water level instrumentation for HPCI high level trip function, the inspector found the PTR to be attached to the wrong valves. The PTR was hung immediately following installation of new valves for the instrument, and no labels had been attached to the valves. The inspector determined by tracing the instrument lines that the valve which the PTR listed as the low pressure isolation was in fact the high pressure isolation and vice versa. Both valves were shut as required. The inspector findings were verified by an Instrument and Control Technician. The inspector discussed his findings with the shift supervisor who indicated that the PTR was going to be cleared as soon as personnel were available to do so.

The inspector verified that the valves were correctly labeled. This appears to be an isolated instance and of minimal safety significance.

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:

-- Standby Gas Treatment System.

-- B Core Spray System.

-- Emergency Diesel Generator Fuel Oil and Air Start Systems.

During the two week maintenance outage, the inspector also visually inspected components which are normally inaccessible, verified the positions of normally inaccessible valves for various systems prior to startup and verified proper valve alignment for the shutdown condition.

No violations were identified (other than the unlocked LPCI valve described in section 4.c above).

5. 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 operation were met, and the system was correctly restored following the testing.

-- F-IMP-71.18A, Reactor Projection System (RPS) Type HFA Relay Post Relay Work Trip Logic Verification , Rev. 3, dated March 19, 1986, performed January 13, 1988.

-- F-ST-3J, Core Spray Subsystem Logic Functional Test, Rev. 15, dated December 22, 1987, performed February 10, 1988.

-- F-ST-4B, HPCI Flow Rate/HPCI Pump Operability/HPCI Valve Operability Tests (IST), Rev. 32, dated January 8, 1988, performed January 23, 1988, and February 17, 1988.

-- F-ST-24A, RCIC Pump and Valve Operability/Flow Rate Test (IST), Rev. 22, dated November 8, 1986, performed February 23, 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-24A, RCIC Pump and Valve Operability/Flow Rate Test (IST), Rev. 22, dated November 8, 1986, performed January 23, 1988.

No violations were identified.

6. Maintenance Observations (62703)

a. The inspector observed portions of various safety-related maintenance activities to verify 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 71/54573, Install modification to 2nd level undervoltage alarm circuit.

-- WR 93/55771, Inspect and repair A and C Emergency Diesel Generator Emergency Service Water Heat Exchangers.

-- WR 00/55874, Underwater inspection of torus.

-- WR 71/05093, Inspection and preventive maintenance of 10500 bus.

-- WR 03/55735, Replace Control Rod Drive 22-23.

-- WR 13/051534, Repair Reactor Core Isolation Cooling Steam Inlet Valve.

No violations were identified.

7. Engineered Safety Feature (ESF) System Walkdown (71710)

The inspector verified the operability of the selected ESF system by performing a complete walkdown of accessible portions of the system to confirm that system lineup procedures match plant drawings and the as-built configuration, to identify equipment conditions that might degrade performance, to determine that instrumentation is calibrated and functioning, and to verify that valves are properly positioned and locked as appropriate:

-- High Pressure Coolant Injection System

During the review, the inspector noted several discrepancies between the plant operating procedure drawing and the as-built drawing. These discrepancies were mainly with labeling or relative location of instrumentation. The licensee reviewed the inspectors comments and changed the drawings as appropriate. The inspector does not consider the discrepancies found to be significant however, since the operators extensively use the operating procedure drawings, they should accurately reflect the as-built drawings. In addition, the inspector is concerned with the errors noted in the as-built drawings. The inspector will continue to review the licensee's drawing controls during future inspections.

No violations were identified.

8. Reactor Water Cleanup System Leak (71707, 93702)

On January 24, 1988, during a plant startup after the completion of a two week maintenance outage, a leak was discovered on the Reactor Water Clean Up (RWCU) System while conducting a walkdown inside the drywell at 500 psig. The leak was found on a pipe-to-elbow weld of the six inch RCWU suction line. The leak was located on an isolable section of piping between the containment isolation valves. A plant shutdown was conducted to investigate and repair the leaking weld.

Using ultrasonic testing, the circumferential crack was found to be about 3/8 of an inch on the surface of the weld and approximately 2 inches at the root of the weld. The section of the piping is carbon

steel and therefore not susceptible to intergranualar stress corrosion cracking. A sample of the weld containing the crack was removed and sent to an engineering laboratory for analysis to determine the cause.

A review of radiography film performed during initial plant construction revealed that a repair had been made to this weld during initial construction, however, post repair radiography showed no abnormalities. The licensee conducted ultrasonic testing of 7 other welds on the RWCU system considering similar configurations and high stress loading. No additional abnormal indications were found. When considering stress loading, the licensee hypothesized stresses which might have resulted in pipe support snubbers being locked up. Additionally, the licensee checked the system snubbers to ensure they were operating freely.

The plant returned to operation on January 26, 1988, after removing the cracked portion of the weld and making repairs to the weld. Based on the initial laboratory report the cause of the cracks was an elevated temperature corrosion fatigue failure. The laboratory also concluded the cyclic stresses believed to be responsible for the crack were apparently thermally induced stresses, with possible contributions from pressure fluctuations and/or vibratory loading at the elbow. In addition, corrosion on the crack surface could have increased the stress level at the crack tip. The laboratory will be sending the detailed analysis results to the licensee within 2-3 weeks.

No violations were identified.

9. High Pressure Coolant Injection Inservice Testing Concerns (71707, 61726, 61725)

On February 17, a ground was identified on Reactor Core Isolation Cooling (RCIC), 13-MOV-131, Turbine Stem Inlet Motor Operator. Prior to declaring the system inoperable and beginning the repair, the licensee performed surveillance tests to verify operability of High Pressure Coolant Injection System (HPCI) as required by Technical Specifications (TS), section 3.5.3. Upon further review of the surveillance test results by the inspector, it was noted that the IST limit for the HPCI pump differential pressure was in the required action range. Discussion of the results with the ISI engineer revealed that the test data was consistent with results received on January 27, 1988, during the last performance of the test. At that time, it was determined that maintenance performed during a previous outage invalidated the previous baseline and that a new baseline should be established. However, a procedural change to incorporate new baseline data was not made. A procedure change was made on February 17, 1988, to incorporate the new baseline data and RCIC repairs were authorized.

Upon further review of the IST program and discussion with the licensee over the handling of the HPCI pump situation, it was determined that additional NRC inspection was necessary in this area. An IST inspection was performed by NRC region-based inspectors February 22, 1988 to February 26, 1988, and is documented in Inspection Report 50-33/88-04. This report will address the HPCI IST concerns described above. It should also be noted that the licensee determined that an internal audit was necessary. The QA audit/appraisal of the IST program was planned to begin on February 22, 1988, however it was deferred pending completion of the NRC inspection.

10. Review of Technical Specifications (TS) for Primary Containment Isolation Instrumentation (71707)

During a review of Technical Specifications concerning the isolation functions of the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) Systems, the inspector raised a concern of how the instrumentation (associated with these isolations) are described in TSs.

The isolations associated with HPCI and RCIC such as high steam line flow, high area temperature, low steam line pressure, and turbine exhaust high pressure, are part of the Primary Containment and Reactor Vessel Isolation Control System as described in the Final Safety Analysis Report, section 7.3.4.8.

However, they are not listed in Technical Specification 3.2.A, protective instrumentation which supply primary containment isolation functions. They are contained under TS 3.5.B which is instrumentation which initiates and controls core and containment cooling systems. The issue of concern is, in accordance with TS 3.5.B, the instrumentation associated with the HPCI and RCIC isolation functions are required to be operable when the system it controls is required to be operable. However, providing a primary containment isolation function, these instruments are required to be operable when primary containment is required as stipulated in TS 3.5.A.

The inspector is not aware of any instances where these instruments were inoperable and would have been required to be operable for containment isolation. The licensee agrees that the TS is not correct as written but feels that they have considered the instrumentation as part of the containment isolation system and have required it to be operable when appropriate. The licensee is reviewing the issue for a possible TS amendment change.

No violations were identified.

11. Technical Specification (TS) Requirements for ECCS Systems During Outages (71707)

During this two week maintenance outage, the inspector raised questions concerning the Technical Specification requirements for Emergency Core Cooling System (ECCS) during cold shutdown condition.

Technical Specification 3.5.A, Core Spray System and Low Pressure Coolant Injection (LPCI) Mode of the Residual Heat Removal (RHR) system, requires low pressure ECCS to be operable whenever irradiated fuel is in the vessel and prior to reactor startup from cold condition with the exception that one system (one core spray system or one Low Pressure Coolant Injection System) may be inoperable for up to seven days. If one system is inoperable for greater than seven days or two systems are inoperable, TS 3.5.A.6 requires the reactor be placed in the cold condition within 24 hours. Cold condition is defined as reactor coolant temperature less than or equal to 212 degrees F.

Technical Specification 3.5.F, Minimum Emergency Core and Containment Cooling System Availability, states when irradiated fuel is in the vessel and reactor is in the cold condition, all LPCI, core spray and containment cooling subsystems may be inoperable provided no work is being done which has the potential for draining the reactor vessel.

During the outage at various times, the licensee made the low pressure ECCS inoperable, but maintained at least one core spray system with its associated normal and emergency power supply operable. Also, control rod drive replacements were being conducted which the licensee does consider as having the potential for draining the reactor vessel.

Plant policy and practice has been to maintain one operable core spray system when doing control rod drive replacements. This policy has not been formally documented.

The plant's TS do not cover what ECCS systems are required when doing work which has the potential for draining the vessel. Standard TS require one core spray system and one RHR pump with an operable flow path while in cold shutdown. Also, standard TS allow ECCS to be inoperable provided the vessel head is removed, the cavity flooded, the spent fuel gates removed and the required water level is maintained over irradiated fuel.

Based on the availability of one core spray system, and other non-ECCS systems to inject into the vessel, no technical safety concerns exist. However, administratively, the requirements for ECCS systems during this condition should be more clearly defined. The licensee is conducting a review of this TS and preparing a TS interpretation in accordance with plant procedures. Review of this interpretation will be made by the Plant Operations Review Committee to determine if the interpretation is sufficient or an amendment is required.

No violations were identified.

12. Generic Letter 84-11 - Inspection of Boiling Water Reactor Piping (25589)

Generic Letter 84-11 for inspections of austenitic stainless steel piping welds susceptible to intergranular stress corrosion cracking (IGSCC) was issued on April 19, 1984. Generic Letter 84-11 is applicable to the IGSCC susceptible piping 4" and over in diameter and in systems operating over 200 degrees F that are part of, or connected to, the reactor coolant pressure boundary. Temporary instruction TI 2515/89 summarizes NRC inspection actions to verify licensee completion of activities required by GL 84-11.

An announced overview inspection of IGSCC detection and mitigation activities for the time period of 1983 to 1988 was conducted to confirm that the requirements of Generic Letter 84-11 have been met or revised as applicable. The inspector determined that significant effort has been directed toward detection and mitigation of IGSCC during the 1984, 1985 and 1987 outages by the licensee.

FitzPatrick plant modifications F1-83-059 and F1-84-067 provided for Induction Heating Stress Improvement (IHSI) of all but two stainless steel piping reactor coolant pressure boundary welds subject to IGSCC. The final report on IHSI implementation by GE dated November 1984 indicates completion of ultrasonic examination (UT) on welds after IHSI. The two welds (28-50 and 28-108) not receiving IHSI had a whip restraint interference problem but were UT examined during the 1985 and 1987 outages and the presence of IGSCC in these welds was not detected.

The installation of equipment and procedures to provide for hydrogen injection into the reactor coolant water to lower its oxygen content and thereby reduce the driving force for IGSCC growth is planned for the 1988 outage. The reactor water chemistry is measured for comparison against the Boiling Water Reactor Owner's Group (BWROG) guidlelines for conductivity, chloride, sulfate and silicon. The inspector reviewed the water chemistry data for December 1987, noting evidence of effective control of water chemistry.

The inspector reviewed portions of outage plans, outage reports and NRC safety evaluations directed toward detection, evaluation and mitigation of IGSCC for the 1984, 1985 and 1987 outages. The documentation provides the basis for concluding the licensee has met the requirements of GL 84-11 or provided a suitable alternative. Ultrasonic Testing (UT) records for three pipe sizes (12", 22" and 28") were reviewed to establish conformance of the ultrasonic examination technique to procedural requirements. The UT sheets for the seven welds with IGSCC (12-4, 12-61, 28-48, 28-53, 28-56, 28-112 and 28-113) were reviewed. It was noted that these welds were evaluated using 1987 data for possible IGSCC flaw growth and for flaw

acceptance margin in accordance with the ASME Code Section XI, IWB 3640. No IGSCC crack growth is predicted by this evaluation for these post IHSI treated welds.

The record of qualification by demonstration at EPRI to show capability to detect IGSCC of six UT examiners were reviewed. The use of Level 1 UT examiners was shown to be limited to equipment setup and work under the direct supervision of Level II or Level III personnel.

For leak detection, the technical specification (3.6.D.1) limits the increase in a 24 hour period to 2 gpm with a total of 5 gpm maximum unidentified leak rate permitted. The technical specification (3.6.D.5) requires restoration of an inoperable equipment sump monitor or floor drain sump monitor within 24 hours or requires initiation of an orderly shutdown.

Part E of attachment 1 to GL 84-11 requires that a visual examination for leakage be performed of reactor coolant piping (RCP) during each plant outage in which the containment is deinerted. FitzPatrick plant procedures do not require RCP visual examination for leakage at each outage when the containment is deinerted. Visual examination of the RCP for leakage at 500 and 1000 psi pressure has been performed during plant startup from refuel outages. The NUREG 0313, revision 2 and the Generic Letter 88-01 issued January 25, 1988 do not require visual examination of RCP during all outages where the containment is deinerted such that the FitzPatrick practice in this area is consistent with current NRC IGSCC guidelines.

However, consideration should be given to evaluation of unidentified leak rate changes to decide on a case by case basis if visual inspection of RCS piping including small diameter attachments is advisable during brief shutdowns where the containment is deinerted.

No violations were identified.

13. Fitness for Duty Policy (92701)

On January 1, 1988, a local resident commented that regular and contract employee's at FitzPatrick were consuming alcoholic beverages during lunch and onsite. Upon followup by the resident inspector, the individual said that they only know of drinking during lunch period and knew of no drinking onsite. The issue was brought to licensee's site management attention. The licensee does not prohibit employees from consuming alcohol offsite during lunch. The licensee stated that security and supervisors are trained in recognizing intoxication and would deny access or remove personnel from site if intoxicated. The licensee does not believe there is a drinking problem at the FitzPatrick plant, but will continue to monitor this and is sensitive to the issue.

No problems were identified in this area.

The inspector also examined records, and data relating to the experience associated with the licensee's fitness for duty program. Information was provided to the Region as requested in RI TI 88-01.

14. New York Power Authority Reorganization (30702)

On January 19, 1988, The New York Power Authority announced a reorganization. The major impact of the reorganization was to restructure the corporate engineering group to ensure all engineering associated with the company's nuclear facilities is under the Nuclear Generation Department. This also added a new field engineering group at the FitzPatrick plant which reports to the corporate office to assist in the engineering of plant modifications.

The inspector will review implementation of the reorganization during future routine inspections of the facility.

15. Management Meeting (30702)

On January 29, 1988, a meeting was held at the NRC Region I Office, King of Prussia, PA, at the licensee's request to discuss plant performance and programs, future plans and a recent reorganization.

The licensee discussed how the reorganization is planned to benefit FitzPatrick in the area of engineering support. The licensee presented department improvements made over the past year and planned improvements for the coming year. Detailed discussions were conducted on more significant programs and programs of interest to the NRC. In addition, the plant goals for 1988 were presented. General areas discussed are as follows:

- Operations
- Radiological Controls
- Maintenance and Modifications
- Technical Services
- Instrument and Controls
- Planning
- Training
- Security
- Scram Reductions
- IGSCC

Meeting participants were:

New York Power Authority

J. Brons, Executive Vice President, Nuclear Generation
R. Converse, Resident Manager, FitzPatrick
W. Fernandez, Superintendent of Power, FitzPatrick

Nuclear Regulatory Commission

W. Russel, Regional Administrator
W. Kane, Director, Division of Reactor Projects
S. Collins, Deputy Director, Division of Reactor Projects
W. Johnston, Acting Director, Division of Reactor Safety
E. Wenzinger, Chief, Reactor Projects, Branch 2, DRP
J. Johnson, Chief, Reactor Projects Section 2C, DRP
A. Luptak, Senior Resident Inspector, FitzPatrick
P. Eapen, Chief, Test Program Section, DRS
N. Blumberg, Chief, Operations Program Section, DRS

16. Assurance of Quality (71707)

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

As noted in paragraph 7, several discrepancies were found between the plant operating procedure drawing and the as-built system drawing for the HPCI system. During maintenance which replaced valves in a reactor water level instrument line, tags were installed on the wrong valves (section 4.d). In addition, during a drywell inspection a valve was found with its handwheel not installed, when the valve was required by procedure to be locked open. These items indicate a need for increased first line supervisory oversight to assure activities are being properly performed and in particular the area of protective tagging.

In addition, several concerns were addressed with respect to technical specification requirements as noted in paragraph 10 and 11. These are other examples where effort is needed to ensure plant activities are meeting requirements to assure overall quality.

The inspector noted that licensee management conducted a successful 2-week maintenance outage. The outage was preplanned and performed with extensive management involvement with the ongoing maintenance and resolving any discrepancies expeditiously. This resulted in successful completion of work which included some longstanding plant deficiencies. In addition, management personnel were utilized on shift during the plant startup which provided effective oversight.

A review of the licensee program to detect and mitigate intergranular stress corrosion cracking found it to be well planned with significant effort placed in this area.

17. 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:

-- January 1988, Operating Status Report, dated February 10, 1988.

No unacceptable conditions were noted.

17. 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.