U.S. NUCLEAR REGULATORY COMMISSION

Region I

Report No.: 50-333/89-08

Docket No.: 50-333

License No.: DPR-59

Licensee: New York Power Authority
Post Office Box 41
Lycoming, New York 13093

Facility: James A. FitzPatrick Nuclear Power Plant

Location: Scriba, New York

Dates: June 4, 1989 through July 29, 1989

Inspectors: W. Schmidt, Senior Resident Inspector
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Approved by: William A Cook 8/10/89
Acting Chief, Reactor Projects Section No. 1B Date
Division of Reactor Projects

Inspection Summary:

This inspection report discusses routine and reactive inspections during day and backshift hours of plant activities including; plant operations, security, surveillance and maintenance, engineering and technical support, and radiation protection. This report period encompassed a total of 202 hours of direct inspection effort. Of that total, 36 were backshift hours while 14 were deep backshift hours which were conducted on 6/21, 7/8, 7/21, 7/22, 7/23 and 7/29.

Inspection Results:

No violations or deviations were identified. Plant operations are discussed in Section 1. 2) Security is discussed in section 2, including response to a bomb threat and to the radioactive contamination of lemonade and subsequent internal contamination of an individual. 3) Surveillance and Maintenance activities are discussed in Section 3, including; surveillance and maintenance observations, previously open item closure and updating, and one new unresolved item dealing

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with the operability of the PASS. 4) An Emergency Preparedness drill is discussed in Section 4. 5) Engineering and Technical Support areas are discussed in Section 5 including; resolution of 4160 V electrical interrupting rating issue, a new unresolved item dealing with H2 water chemistry, review of ATWS TI 2500/20, and update and closure of previously open items. Radiological protection is discussed in Section 6, including an unplanned radiation exposure to two workers.

The inspector will track the following issues in a subsequent inspection report: 1) Resolution of containment venting and pressurization procedural issues (see Section 1.a). 2) Review and update surveillance tests to meet new standards (see Section 3.1). 3) Review emergency preparedness corrective actions before the September 1989 drill (see Section 4). 4) Detailed review of the licensee's JCO regarding interrupting capacity ratings of 4.16KV electrical switchgear, 5) Resolution of SLC walkdown and procedure concerns (see Section 5.h.1). 6) Resolution of ARI valve wiring protection issue (see Section 5.h.2).

DETAILS

1. Operations (71707, 93702)

During this reporting period, the unit operated at 100% of rated power except for a load reduction to approximately 55% on July 8, to support rebrushing the recirculation pump motor generator sets and repair an oil leak on the C circulation pump.

The inspector reviewed the operability of the following systems during the inspection period:

- emergency diesel generators
- high pressure coolant injection

a. On July 17, the inspector observed that operators were adding nitrogen (N2) to the drywell while venting the torus through the standby gas treatment system (SBGT). This was being done to reduce the Oxygen (O2) concentration in the containment. The limit on O2 in the Technical Specifications (TS) is less than 4% by volume. Prior to the purging operation, the concentration had not exceeded 3% by volume.

The inspector reviewed Operating Procedure (OP)-37 and observed that there was a specific procedure for reducing O2 concentration in the containment. This involves allowing N2 to flow to the drywell. There are no specific instructions on how to control drywell pressure and drywell to torus differential pressure. There are two methods which could be used in other sections of the procedure. 1) The drywell can be vented through SBGT as drywell pressure increases to near 2 psid. 2) The torus can be vented using a procedure to maintain drywell to torus differential pressure greater than 1.7 psid. In the first case, the pressure would be relieved, resulting in a feed and bleed of the drywell. In the second case, the increased pressure would cause the drywell atmosphere to force the water out of the drywell to torus down comers and bubble up through the torus water and be vented to SBGT.

The inspector verified that the licensee had chosen the second alternative. Although there is no specific procedure to perform this evaluation, the valve line up was proper and drywell to torus differential pressure was properly maintained. The inspector discussed the lack of a specific procedure with the Operations Superintendent who committed to review the situation.

1.1 Safety Assessment

The plant continues to be operated safely.

2. Security (71707, 92702)

On July 8, the licensee made an ENS report of a telephone bomb threat. The caller said he had placed a bomb in the reactor building. He then identified himself as a current licensee employee. The licensee security force implemented the bomb threat contingency procedure, including a search of the accessible areas of the plant and the protected area. No bomb nor evidence of wrongdoing was found.

The licensee and the Oswego County Deputy Sheriff completed a full investigation, including an interview of the employee whose name was used by the caller. During the investigation, the employee was suspended from duties with pay. Based on a review of the employee's good record and the investigation, the licensee concluded that the employee was not responsible for the call. He was reinstated on July 10, 1989. The licensee's actions appear to be appropriate and the inspector has no further questions regarding this matter.

2.1 Safety Assessment

The licensee's security department did a proper and thorough investigation into the bomb threat and into the incident surrounding the radioactive contamination of a worker's lemonade (see Section 6.c).

3. Surveillance and Maintenance Observations (61726, 92702, 93702)

a. On June 14, the inspector observed preventive maintenance being performed on one of the two 115 KV off-site emergency transformers. The inspector verified that the licensee performed the surveillance requirements necessary prior to deenergizing the transformer. The maintenance was properly controlled. The inspector had no questions.

b. On June 21, the inspector monitored performance of Instrument Surveillance Procedure (ISP)-3-7, Reactor High Water Level (HPCI) Instrument Functional Test Calibration. During performance of this surveillance procedure in December 1987, a reactor trip occurred because of a false reactor low level signal. This was caused by not fully closing an instrument isolation valve before valving in the test rig used during the level switch calibration. The inspector discussed this event with the technicians performing ISP-3-7. The technicians exhibited proper controls and communications during test performance. The inspector had no further questions.

c. The licensee has been recently correcting various deficiencies identified in the post accident sampling system (PASS). After completion of maintenance on the system, the licensee attempted to obtain a reactor coolant liquid sample on July 3. During the conduct of the test, the PASS system became inoperable due to nitrogen lines in the system becoming filled with reactor coolant while attempting to obtain the sample. The root cause of the incident appears to be to inadvertent cross-tieing of nitrogen and reactor water sample lines during maintenance. The inspectors continue to investigate the PASS system deficiencies identified to determine whether the licensee's corrective actions have been adequate. This item remains unresolved pending completion of this review and determination of how the sample lines were crossconnected (UNR 50-333/89-08-01).

d. (Open) Unresolved Item 89-07-02; LER 89-07: Surveillance of Fire Barrier Penetrations Missed Due to Misinterpretation of Barriers Requiring Surveillance. This LER documents the program weaknesses and corrective actions associated with preventing continued problems in the area of fire barrier penetration identification and surveillances. This area was discussed in inspection report 50-333/89-07. The licensee's short term corrective actions appear adequate. This item remains open pending completion of the licensee's long-term program improvements scheduled for completion in the fall of 1989.

e. LER 89-08: High Pressure Coolant Injection System and Reactor Core Isolation Cooling System Made Inoperable Due to Procedure Deficiency Causing a Missed Surveillance. This event occurred on May 17, 1989, and was described in inspection report 89-07. Based on the review of this LER, the inspectors had no further questions.

f. LER 89-09: Containment Drywell to Suppression Chamber Differential Pressure and Suppression Chamber Level Out of Limits Due to Personnel Error. This event occurred on May 18, and was described in inspection report 89-07. The licensee's corrective actions appear adequate. The inspectors will continue to monitor the conduct of surveillance testing during routine inspections.

g. (Closed) Unresolved Item (50-333/87-03-02): The identification of missing bolts and nuts from the nonstructural parts of the refueling bridge. The inspector toured this area of the spent fuel pool. The licensee was at the time removing spent LPRMs and other stored material. Health physics personnel increased monitoring of this area to ensure that personnel did not pick up radioactive "hot particles" on their protective clothing. An examination of accessible areas of the bridge did not identify the presence of loose parts or material that could inadvertently enter the spent fuel pool.

After the initial identification of this problem, an examination by the licensee was performed on January 30, 1987 (Work Request No. 52849) and again on August 9, 1988 (Preventive Maintenance Work Request No. 06732).

Additionally, a review of the material control log for the spent fuel pool bridge was conducted. The log was well maintained during the fuel pool cleanup activities during the four month cleanup effort. This item has been properly addressed by the licensee. This item is closed.

h. (Closed) Unresolved Item (50-333/89-29-02): Lack of Pressure and/or Flow Requirements in Surveillance Tests - As of June 6, 1989, the licensee incorporated pressure and/or flow requirements in the monthly pump operability surveillance testing for residual heat removal (RHR), RHR service water (RHRSW), core spray (CS), high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) systems. These actions were considered acceptable to the inspector. This item is closed.

i. (Closed) Violation (50-333/89-03-03): Failure to Perform Surveillance Testing at TS Required Discharge Pressures - In their response dated June 6, 1989, the licensee agreed with the violation. As corrective actions, the surveillance tests for HPCI and RCIC have been revised to ensure that the pumps can produce the required flow rates at discharge pressures corresponding to a reactor pressure of 1120 psig, as required by the Technical Specifications. Further, a requirement that the HPCI injection valve goes fully open within thirty seconds of an actuation signal has been added to the quarterly full flow rate and semiannual subsystem logic functional test. These changes were reviewed by the inspector and found to be satisfactory. This item is closed.

3.1 Safety Assessment

The licensee incorporated the necessary testing requirements for centrifugal pumps to ensure that they are operable.

The licensee incorporated the necessary acceptance criteria per TS and system design for HPCI and RCIC.

The licensee committed in their response to violation 50-333/89-03-03 to complete a document that will establish a formal surveillance program which sets standards for the types of criteria that must be established to determine system/component operability. Administrative Procedure (AP) 4.5 is being revised to include these criteria. Once this action has been completed, the licensee will review and update surveillance tests as needed. The licensee currently believes that the criteria will be developed by the end of August, 1989, with the test procedure review to follow. This type action is seen as proper and will be reviewed in a subsequent inspection report.

4. Emergency Preparedness (71707)

On July 12, 1989, the licensee conducted a practice emergency drill. The inspector observed the conduct of the drill in the control room, the technical support center (TSC) and the emergency operations facility (EOF). After completion of the drill, the inspector discussed his observations with the Emergency Preparedness Coordinator. It was determined that the licensee had made similar observations during their drill critique and were taking appropriate correction actions.

4.1 Safety Assessment

The licensee's program appears satisfactory based on the drills observed. The inspectors plan to review the licensee's corrective actions from this drill, as well as the previous drill performed on April 26, prior to the annual exercise planned for September 6.

5. Engineering and Technical Support (37700, 92702)

a. On July 14, 1989, the licensee reported via the ENS, that they had identified that the electrical short-circuit interrupting rating of the emergency switchgear was inadequate. This was discovered during an electrical study conducted as a result of an NRC safety system functional inspection (SSFI), documented in inspection report 89-80. The study was conducted by Stone and Webster to identify the momentary and interrupting short circuit currents for normal and emergency electrical conditions.

It was determined that the most limiting condition for the switchgear was during normal operations with plant loads supplied from the main generator paralleled with a train of emergency diesel generators (EDGs). If a three phase bolted electrical fault occurred, momentary and interrupting short circuit duties on certain 4.16 KV safety and non-safety busses would exceed the ratings specified by the manufacturer. This leads to the potential for switchgear failure and damage to safety-related components. This condition could only occur during monthly one hour surveillance testing runs of the EDGs.

On July 27, the licensee's Plant Operations Review Committee (PORC) reviewed and approved a Justification for Continued Operation (JCO). This JCO allows the EDGs to be run in parallel with the main generator for the one hour surveillance test. The JCO states that the chance of a three phase or phase to phase bolted short is extremely small and thus should not prohibit continued safe operation of the facility.

Based on the above review and the fact that only one division of electrical power would potentially be lost in a worst case faulted condition, the inspector, based on a preliminary review, found this JCO acceptable. The licensee voluntarily submitted the JCO documentation to the NRC for information. This documentation will be reviewed in detail to confirm the adequacy of the licensee's conclusions.

b. On July 23, the inboard reactor water recirculation system sample valve, 02-SOV-39, a primary containment isolation valve, failed closed. This valve provides flow to the Crack Arrest Verification (CAV) system. CAV provides data from representative material specimens to allow for evaluation of crack growth in the recirculation piping while using Hydrogen Water Chemistry (HWC). With 02-SOV-39 failed closed, representative data for crack growth of the recirculation piping material is not possible. Other sample points cannot be used (i.e. Reactor Water Cleanup) due to significant differences in chemistry conditions. Without having the ability to obtain proper chemistry data, the licensee has shutdown the CAV system. The licensee plans on controlling HWC injection by utilizing the main steam line radiation monitoring readings and the feedwater hydrogen concentration. The inspectors will review this HWC control method and determine if the actual recirculation piping chemistry conditions are being maintained in the condition necessary to prevent crack growth. This item remains unresolved (UNR 50-333/89-08-02).

c. (Closed) Generic Letters 83-02 and 83-36 (Multi-plant action items B072 and B083, respectively): On May 31, 1989, Amendment 130 was issued to the Technical Specifications. This amendment incorporated the changes required to the TS by NUREG 0737. This item is closed.

d. TI 2500/27 I&E Bulletin 87-02 - Fastener Testing to Determine Conformance with Applicable Material Specifications

(Closed) Unresolved Item 87-00-02: The inspector previously reviewed laboratory test reports and found them detailed and addressing all areas of concern. In addition, the inspector verified by observation in the warehouse that fasteners identified as faulty were segregated from stock. This review was documented in inspection report 50-333/89-07. During that inspection, the inspector requested additional documentation of the licensee's actions. The information was not available at that time. The inspector has received the documentation from the licensee and has reviewed the licensee's actions to the Bulletin response. The licensee has upgraded random testing of selected vendors of safety-related fasteners. The inspector had no further questions. This item is closed.

e. (Closed) Unresolved Item 88-23-03: This item involves the installation of two double disc gate valves, in the A and B trains of containment spray, which did not have internal bypass flow capabilities. The licensee discussed this installation in LER 88-13-00. The inspector completed his review of this event by reviewing the LER and the subject procurement information.

The procurement records show that the valves were purchased without an external or internal bypass. The NYPA procurement contract requisition CR-87-99 implemented FitzPatrick procurement specifications 87-01-1, which requires in step C.2.1.18 that an external bypass be installed for valves designed to operate at greater than 200 psi and 200F. The licensee's engineering staff questioned Anchor Darling (AD) concerning the lack of a bypass on these valves. AD said that the valve would function properly without a bypass.

The inspector found the safety significance of this event small since when the valve was to be open, the RHR pumps would be running. This would have minimized the differential pressure to 240 psid on the downstream disc, which was the "as-tested" condition by the manufacturer. This item is closed.

f. (Closed) Deviation 88-23-02: In response to inspection report 89-03, dated June 21, 1989, the licensee discussed further actions taken to resolve the deficiencies in monitoring of the 115 KV off-site power supplies. These actions included holding a meeting with Niagara Mohawk Power Corporation (NMPC), the owner of Nine Mile Point Unit 1 and of the transmission line network The meeting was used to discuss the necessary control and flow of information between the two companies. Mutually agreed upon procedures have been established which govern the control and flow of information regarding the status of the 115 KV lines to FitzPatrick. The actions taken are appropriate and resolve the deviation. This item is closed.

g. (Closed) Unresolved Item 88-29-01: As of June 14, 1989, the licensee submitted the twelve proposed changes to Technical Specifications (TS), as committed to in their March 20, 1989 response to inspection report 50-333/88-24. The submittal appears adequate, although some additional information has been requested by the NRC staff during their preliminary review of several submittals.

In an effort to make the TS more readable, the licensee has submitted several of these amendments in a retyped format. The licensee currently has the entire TS on a word processing system, in this revised format, in the White Plains Office. While this is seen as an improvement, the licensee must use proper verification techniques on these retyped pages to ensure that items, other than those which are requested for amendment, not be inadvertantly changed. The licensee plans to submit the retyped TS in its entirety, prior to beginning the TS improvement program.

h. (Closed) TI 2500/20, Rev. 1 Anticipated Transient Without Scram (ATWS) Rule 10 CFR 50.62

1. Standby Liquid Control System (SLC)

The ATWS rule, 10 CFR 50.62C(4) requires all BWRs to modify SLC to provide the equivalent reactivity control capacity of an 86 gpm injection rate of 13% sodium pentaborate solution. This is based on a reactor vessel 251 inches in diameter.

The licensee's modification F1-85-055 addressed the ATWS concerns by:

a. A change of solution in the SLC tank to an enriched sodium pentaborate solution, one with a higher atom percent of the B10 boron isotope. The licensee changed the SLC solution to an enriched 34.7 atom percent B10 boron concentration to provide neutron absorption equivalent to 86 gpm of 13 wt. percent sodium pentaborate solution with natural boron, while using the existing redundant 50 gpm pumps. During the changeover of solution, the existing boron solution was replaced and the tank flushed. The new enrichment was verified by the inspector during a review of September 28, 1988 and March 30, 1989, sample chemistry results documented in Control System Sampling and Analysis procedures. These procedures included monthly and once per operating cycle surveillance to meet the Technical Specification (TS) requirements (T.S. 3.4.C, 4.4.C.1 and 4.4.C.3b.).

The inspector reviewed two pump flow tests performed per procedure No. F-ST6A, "Standby Liquid Control Pump Functional Test Procedure (IST)" which documented the following pump capacity values:

Test Date	Pump A	Pump B
10/3/88	53.9 gpm	55 gpm
10/23/88	53.9 gpm	55 gpm

b. Replacement and changing of setpoints for the SLC tank level transmitter.

The replacement of the SLC tank level instrument was subject to a safety evaluation and found acceptable by the licensee. A review of SLC tank level instrument calibration ISP82 noted that the procedure had included new changes in the level setpoints up to 90.6% ,equivalent to an acceptable value of 109.8 (108.8 - 110.8) in inches of water, and a low level of 86.1%, equivalent to 104.8 (103.8 - 105.8) inches of water. No anomalies were identified.

c. Changes to setpoints of the SLC tank heater and heat tracing system.

The inspector walked down the system and reviewed procedures regarding the operation of the system. He noted during his walkdown that the SLC tank can be breached due to an unserviced foot valve at the tank top. This opening is used by operations personnel to reverify levels. He also noted that two caps on the suction line diaphragms were missing. Further, the operating log indicated that the change in operating levels of SLC was not reflected in the operations round sheets. The documentation noted an acceptable tank level as between 81 and 86%, although levels presently acceptable are 88 to 91%. However, the inspector did note that the actual level on control panel indicator was within TS requirements. These items were brought to the attention of the licensee and will be reviewed in a subsequent report.

In conclusion, the SLC system, as designed, satisfies the items set forth in the ATWS rule. The inspector has no further questions on this area of ATWS modifications. This TI is closed.

2. Alternate Rod Injection System (ARI)

ARI is an additional system to mitigate the consequences of an ATWS event. The system is diverse from the reactor trip system from the sensor output to the final actuation device.

The licensee's modification F1-85-053 used to install this modification used the guidance in BWROG Topical Report NEDE 31096. The ARI system installed five (5) solenoid actuated air dump valves at the scram pilot air header. One three-way valve is installed in series with the existing backup scram valves. Four (4) two-way valves are installed, one each at the East Scram Discharge Volume (SDV) piping, the West SDV piping, and the Hydraulic Control Unit (HCU) subheaders on air bank A and B. On exhibiting a low water level or reactor vessel high pressure that matches the required logic of the ARI circuitry, the solenoids will open and ensure the completion of the scram pilot headers' depressurization.

The inspector walked-down the system to verify the mechanical installation. All five solenoid valves were found to be installed per design and to offer an unrestricted flow path from the scram discharge header when actuated. One concern was identified by the inspector dealing with the electrical conduit supplying the solenoid

valves where the valve wiring joins system wiring. The inspector noted that the protective cover on one conduit protection was installed without its required snubber jacket and the two attachment screws were loose. The licensee has written a work order to examine the connector boxes on the five solenoid installation. Resolution of this item will be reviewed in a subsequent report.

The preoperational test on the ARI system (POT No. 03E) was reviewed to determine if the system response time was properly verified and satisfied. The test data indicated an actuation time of five seconds versus the acceptance criteria of 25-30 seconds. The inspector had no further questions concerning this modification.

i. (Open) Unresolved Item (50-333/88-29-08): Concerns Regarding the Design of the ARI - Two major concerns regarding the design of the ARI system will be resolved during the 1990 outage: 1) concerns regarding the diversity of trip signals due to a possible common mode failure due to signal conditioning equipment in ARI design which is similar to the reactor trip system and; 2) concerns regarding the testability of the system. The present ARI system appears to have the potential to cause unnecessary plant trips during surveillance testing. The licensee is addressing these issues with the purchase of diverse signal devices and changes to the electrical test schemes to allow for testing at power without causing the plant to inadvertently trip.

NRR staff also had stated that the ATWS/RFT trip system logic may not be acceptable until the licensee has demonstrated that the one-out-of-two trip scheme does not have a potential for inadvertent actuation and challenges to other safety systems. This item is described in detail in the Safety Evaluations by NRR in its attachment to the November 18, 1988 letter to NYPA.

These issues will be reviewed after final design changes are completed during the forthcoming 1990 outage. This item on ATWS implementation remains open.

j. (Open) Violation (50-333/88-29-10): On June 15, the licensee submitted a supplement to licensee event report (LER) 88-009-01. This LER provides information which states that the licensee has completed a design evaluation of the crescent area unit cooler requirements and capabilities. Based on this, the licensee has increased the maximum permissible service water inlet temperature to 82 degrees Fahrenheit. This was done after an evaluation of the heat loads in the crescent areas was conducted with the loads running as they would be during use of the EOPs. This item remains open pending further review of the licensee's safety evaluation for allowing the increase in the service water temperature and completion of other commitments being tracked by this item.

k. (Closed) Violation 89-03-01: In their response dated June 21, the licensee agreed with the violation. The corrective actions taken include a revision to Administrative Procedure (AP) 8.2, "Reporting Variations From Normal Plant Operations and 10 CFR 21 requirements." The revision was reviewed and is adequate, if followed, to prevent a failure to make a 10 CFR 50.72 notification when required. This item is closed.

5.1 Safety Assessment

The licensee's actions on the ATWS modifications are adequate. The licensee must continue to pursue installation of the diverse level instruments for the ARI.

6. Radiological Protection (71707)

a. On June 12, two refuel floor workers received unplanned radiation exposures while working around the spent fuel pool. The highest radiation exposure received by either of these workers was 780 millirem to the whole body, which brought this individual's quarterly whole body exposure to 1450 millirems, well within NRC limits. The cause of this exposure was a hot particle which surfaced to the top of the spent fuel pool during their work evolution. When the particle surfaced, it was not readily identified by the workers (the workers were not provided with alarming dosimeters). The area radiation monitors on the refuel floor did not detect the hot particle. The increased radiation field was discovered during a routine HP survey made shortly after the particle surfaced.

The licensee suspended work and secured access to the refueling floor. A full investigation of the event was completed and appropriate corrective actions initiated. The location of the hot particle was identified by completing detailed HP surveys and utilizing remote cameras. The licensee successfully retrieved the particle in a metal container and returned the container to the spent fuel pool. Additional corrective actions, including cleaning the spent fuel pool of any debris which had the potential to float to the surface, were taken. A region based radiation specialist was sent to the site to evaluate the licensee's actions taken as a result of this event. Their findings will be documented in inspection report 89-13.

b. On June 28, the inspector observed the installation of the container head on a cask containing radioactive waste. While the head was being installed, a washer fell and rolled out of the contaminated area. A person outside the area was prompted by the person who dropped the washer to hand it to him. The worker picked up the washer and handed it back into the contaminated area.

The inspector observed that radiation protection personnel saw this occur and took action to have the person who picked up the washer frisk his hand. No contamination was found.

c. On July 14, a contractor radiation protection technician was found to have internal radioactive contamination. This individual was the second shift (evening) supervisor for the spent fuel pool cleanup activities. The contamination was found as the person exited the radiologically controlled area (RCA) by passing through a friskall.

The person was questioned by the licensee as to his activities prior to passing through the friskall. He said he had left the RCA prior to this event to smoke a cigarette and drink some lemonade. A whole body count was performed which showed internal contamination from approximately 1.5 mCi of radioactive material, consisting primarily of Co 60. Radioactive analysis of the lemonade revealed that it contained significant concentrations of radioactive material.

Initially, the supervisor told the licensee's security department that he had no idea why or who would have put the radioactive material into his drink. As a result, the licensee asked the local police to investigate the incident. On July 16, the individual reversed his story, saying that he had taken a radioactive contamination swipe outside the radiologically controlled area (RCA) and put it into the lemonade, shook it, removed the swipe and then drank it. Based on this information, the licensee called off the local police investigation. The individual stated to the licensee that he had done this to "bring conditions to a head". The conditions he was apparently referring to were an apparent strife between the first and second HP shift on the refuel floor.

During subsequent discussions with the licensee, the individual alleged drug use by several contractors who also worked on the refuel floor. Based on this allegation and in an effort to gather further information, the 18 persons who worked on the refuel floor were interviewed and asked to take a drug screening test. Of the 18, 17 took the drug test and one refused and quit. Of the 17 individuals, one positive result was received and the individual was fired. Interviews with these individuals did not uncover any indication of why the incident occurred or who may have been involved.

On July 18, after consulting an attorney, the individual reversed his story and told licensee site security and management that he did not contaminate the lemonade. The NRC Office of Investigations was asked to review this event. The resident inspectors will continue to review the results of the licensee's and NRC investigation.

6.1 Safety Assessment

The actions taken by the licensee once the problems above were discovered were adequate. Concerns still exist in the area of management oversight of activities on the refuel floor. The licensee has started a program to investigate the events that were going on on the refuel floor prior to the lemonade contamination. This investigation is being conducted by the site QA Superintendent and the Vice President - Nuclear Support.

7. NRC Manager Visit

On July 26, James Wiggins, Chief, Branch 1, Division of Reactor Projects, Region I and Robert Capra, Director, Project Directorate I-1, NRR visited the site and talked to licensee management.

8. Exit Interview (30703)

At periodic intervals during the course of this inspection, meetings were held with senior facility management to discuss the 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.