U.S. NUCLEAR REGULATORY COMMISSION

Region I

Report No.: 96-02

Docket No.: 50-333

License No.: DPR-59

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

Facility: James A. FitzPatrick Nuclear Power Plant

Location: Scriba, New York

Dates: February 18, 1996 through April 6, 1996

Inspectors: G. Hunegs, Senior Resident Inspector
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Approved by: C J Cowgill III ______ 5/3/96
Curtis J. Cowgill, Chief — Date
Projects Branch No. 2
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INSPECTION SUMMARY: Routine NRC inspection of plant operations, maintenance, engineering, plant support, and quality assurance/safety verification.

RESULTS: See Executive Summary

EXECUTIVE SUMMARY

James A. FitzPatrick Nuclear Power Plant

Inspection Report No. 50-333/96-02

Plant Operations: On February 21, 1996, a reactor shutdown was commenced due to control rod scram time test failure to position 46. Scram insertion time limits to positions 38, 24, and 04 were not exceeded. The slower scram time to position 46, (five percent insertion) was caused by the scram solenoid pilot valve (SSPV) diaphragms adhering to the valve seat and retarding the start of rod motion. During the shutdown, FitzPatrick replaced the SSPV diaphragms with new material and has implemented industry guidelines to continue to monitor the rod control system. FitzPatrick was aggressive in followup of the problem and took conservative actions to shutdown to replace the SSPV diaphragms. During the shutdown, a loss of electrohydraulic control occurred and operators inserted a manual scram. Operators demonstrated conservative decision-making by inserting a manual scram upon the loss of EHC. Operating crew actions were characterized by good procedure adherence and a methodical approach to plant cooldown.

The startup activities from March 4 to 7 were characterized by clear operator communications, attentive management oversight, and effective control by shift supervision. Shift turnovers were performed in a controlled manner and crew and pre-evolution briefings were good.

Prior to the shutdown on February 21, 1996 safety/relief valves (SRVs) D, E, and H had indications of pilot valve and/or main seat leakage. The pilot assemblies for D and H and the E SRV main body were replaced during the forced outage and currently SRVs do not have indications of leakage. FitzPatrick closely monitored SRV performance through daily torus heat up rate calculations, observations of SRV tailpipe temperature, and discussions at the daily plant leadership team meetings. The inspectors concluded that FitzPatrick was sensitive to industry problems related to SRV leakage.

Maintenance: Several failures associated with safety/relief valve operation were identified during the shutdown. The causes of these failures were related to poor foreign material controls during maintenance and issues related to solenoid operated valve reassembly both by the vendor and the licensee. A high degree of management involvement during the trouble shooting, maintenance and testing activities was noted. The issue was aggressively pursued by station personnel and the licensee assessments were self critical. However, automatic depressurization system previously conducted on the maintenance activities introduced a potential failure which affected the satisfactory operation of safety related components and represented an NRC violation.

Engineering: A back-fill capability provided from the control rod drive system for each reactor vessel instrumentation reference leg was installed in November, 1993. Based on the inspector's review of the modification package and the safety evaluation, the inspector concluded that the safety evaluation provided a thorough review of the safety implications of the backfill modification.

Plant Support: General housekeeping condition of the drywell was found to be good. FitzPatrick's actions to investigate the source of oil were thorough and corrective actions following the discovery of the drained snubber were appropriate.

Safety Assessment/Quality Verification: Licensee Event Reports (LERs) 96-003 and 96-004 were well written, concise, accurate, and properly submitted.

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DETAILS

1.0 SUMMARY OF FACILITY ACTIVITIES

1.1 NYPA Activities

The plant was at 100 percent power at the beginning of the inspection period. On February 21, operators commenced a power reduction to facilitate repairs to the "A" reactor feed pump. On February 22 at 3:33 a.m. a reactor shutdown was commenced due to control rod scram time test failure. During the shutdown, a major electro-hydraulic control system leak occurred and the plant was manually scrammed from 7 percent power. During the forced outage, the scram solenoid pilot valve diaphragms were replaced, a leaking high pressure coolant injection system steam supply valve was repaired, safety/relief valve (SRV) maintenance was conducted, and the "B" recirculation pump seal was replaced. The reactor was taken critical on March 5 at 2:58 a.m.. The plant returned to full power on March 12, and remained there until the end of the inspection period.

Effective March 11, 1996, H. Salmon, the former Site Executive Officer, assumed the new position of Vice President of Nuclear Operations. He will direct operations and supporting activities at both FitzPatrick and Indian Point 3 Nuclear Power Plants. M. Colomb, the former General Manager, Operations, was named to the new position of Plant Manager. He will be responsible for day-to-day operations and support functions and will report to H. Salmon.

1.2 NRC Activities

The resident inspectors conducted routine and reactive inspection activities in several areas including: plant operations (Section 2.0), maintenance (Section 3.0), surveillance (Section 4.0), engineering (Section 5.0) and plant support (Section 6.0). The inspection activities during this report period included inspection during normal, backshift and weekend hours by the resident staff.

2.0 PLANT OPERATIONS (Inspection Procedures 71707,93702,92901,62703)

2.1 Operational Safety Verification

The inspectors observed plant operation and verified that the facility was operated safely and in accordance with procedures and regulatory requirements. Regular tours were conducted of the plant with focus on safety related structures and systems, operations, radiological controls and security. Additionally, the operability of engineered safety features, other safety related systems and on-site and off-site power sources was verified. No safety concerns were identified as a result of these tours.

- Control room
- secondary containment building
- radiological control point
- electrical switchgear rooms
- emergency core cooling system pump rooms
- security access point

- protected area fence
- intake structure
- diesel generator rooms

2.2 Followup of Events Occurring During Inspection Period

2.2.1 Degraded Scram Times

NRC Information Notice 96-07, Slow Five Percent Scram Insertion Times Caused By Viton Diaphragms in Scram Solenoid Pilot Valves, was issued on January 26, 1996 to alert licensees to the industry problem. Because of the industry concern with delayed scram times, FitzPatrick elected to increase the frequency of scram time testing and performed scram time testing on February 21, 1996. On February 21 at 3:33 a.m. a reactor shutdown was commenced due to control rod scram time test failure. Review of control rod scram time test data indicated that the technical specification (TS) scram insertion time limits to position 46 were exceeded for 18 rods. The slower scram time was caused by the scram solenoid pilot valve (SSPV) diaphragms adhering to the valve seat and retarding the start of rod motion. During the shutdown, FitzPatrick replaced the SSPV diaphragms with new material and has implemented industry guidelines to continue to monitor the rod control system. The inspectors observed maintenance activities associated with replacing the diaphragms and noted that activities were well conducted.

Background:

NRC Information Notice 94-71, Degradation of Scram Solenoid Pilot Valve Pressure and Exhaust Diaphragms, was issued in October 1994 to advise licensees of the potential failure of SSPV diaphragms made of Buna-N due to premature hardening and subsequent failure. On a reactor protection system scram signal, the scram solenoid pilot valves reposition to vent the scram air header, allowing the scram valves to open. This could result in reactor control rod scam times greater than allowed by technical specifications during surveillance testing. FitzPatrick replaced all 274 SSPVs on each control rod drive (CRD) hydraulic control unit with new SSPVs containing Viton elastomers during the last refueling outage in 1994-5. Viton is a trade name for a fluoroelastomer which was selected by the Boiling Water Reactor Owner's Group (BWROG) as a replacement for the Buna-N. The SSPVs that were removed contained Buna-N elastomers. The elastomers were changes to Viton due to the environmental qualification concerns with the Buna-N elastomers. Buna-N SSPVs can only be qualified for 3-5 years, while the Viton SSPVs were initially qualified for 15 years.

Recent industry problems have surfaced with respect to slow five percent scram insertion times caused by Viton diaphragms in scram solenoid pilot valves. These slow times resulted from adherence of the exhaust Viton diaphragm to the brass valve seat in the scram solenoid pilot valves. Several BWR plants have noted the trend toward slower scram insertion times to notch 46 after about 6 months in service.

The inspectors reviewed previous control rod scram time tests conducted in June, September, and December, 1995. The test results showed a small increase of 3.16 percent. FitzPatrick did not consider that the increase reflected a sustained trend and noted that these changes were less than the increases seen at other plants.

On February 7, 1996, FitzPatrick prepared an operability assessment to evaluate the issues associated with the control rod drive system with delayed scram times. The safety evaluation noted that for some plant transients the slower scram speed can influence the peak power and consequently, the fuel thermal limits, but based on the FitzPatrick scram time test data, the control rods would be able to meet the required time for insertion.

On February 21, 1996, during scram time testing of 20 control rods, 18 control rods exceeded the TS limit to position 46 (5 percent insertion with an average time of 0.373 seconds. The TS limit to position 46 is 0.338 sec. Averaging in the 20 control rods with the previously measured times does not increase the core average time greater than the Technical Specification limit, but the licensee assumed that all control rod times would increase if tested and therefore initiated a shutdown. As previously stated, earlier quarterly tests of 10 percent of control rods showed some degradation, but were within limits. The review of test data indicated that TS scram insertion time limits to positions 38, 24 and 04 were not exceeded.

FitzPatrick has implemented the BWROG Regulatory Response Group (RRG) interim recommendations which were issued on February 16, 1996. The guidance is as follows:

1) Perform control rod scram time testing of at least 5% of control rods every 60 days. 2) Every 120 days perform control rod scram time testing of at least 10 percent of the number of Viton SSPV control rods in the core and 3) perform a functional test of the alternate rod insertion logic and valves once per cycle.

The RRG recommended that any rod in the sample with a 5 percent scram insertion time greater than 0.49 second should be declared inoperable. The options available to the industry are to replace the exhaust Viton diaphragms with new Viton diaphragms to restore the performance of the SSPV for the short term.

The inspectors monitored FitzPatrick's response to the industry issue, and they concluded that FitzPatrick was aggressive in followup of the problem and took conservative actions to shutdown to replace the SSPV diaphragms. Accelerated scram time testing will continue to be conducted to monitor the performance of the Viton diaphragms.

2.2.2 Plant Shutdown

On February 22 at 2:06 p.m., operators inserted a manual scram from 7 percent power due to oscillations on the number 1 turbine bypass valve. All control rods went full in and equipment operated as designed. The oscillations were caused by a major electrohydraulic control (EHC) system leak. Investigation revealed a cracked stainless steel section of pipe. Visual and metallographic examinations, as well as scanning electron microscopy revealed that the cracking exhibited characteristics of fatigue. Corrective actions, in part, included replacing supply tubing to all four turbine bypass valves with flexible hoses and additional walkdowns and monitoring were performed.

The inspectors noted that the high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) systems were started in the pressure control mode to maintain cooldown rate. This mode of operation is described in Operating Procedure (OP)-15, High Pressure Coolant Injection, and OP-19, Reactor Core Isolation Cooling as a mode of operation described as the reactor vessel pressure (RPV) control mode in which turbine steam flow is used to control RPV pressure.

Use of HPCI and RCIC in the pressure control mode as a depressurization method is not described in the FSAR. Primarily, the discussion in FSAR section 6.4.1 for HPCI centers on its primary purpose, which is to ensure that the reactor is adequately cooled to limit fuel clad temperature in the event of a small break in the reactor coolant system piping. The RCIC system is similarly described in FSAR section 4.7.

FitzPatrick has evaluated the operation of HPCI and RCIC in the pressure control mode, which is described in the BWROG Emergency Procedures Guidelines, and is in the process of updating the FSAR to reflect this mode of operation.

The inspectors reviewed the evaluation and considered this inconsistency to be minor.

The inspectors observed that operators demonstrated conservative decision-making by inserting a manual scram upon the loss of EHC. Operating crew actions were characterized by good procedure adherence and a methodical approach to plant cooldown.

2.2.3 Plant Startup Activities

The inspectors observed startup activities in the control room during the period of March 4 through March 7. The startup was delayed by the failure of the "G" SRV to cycle during ST-22B, Manual Safety/Relief Valve Operation (see paragraph 3.1.1). The startup was characterized by clear operator communications, attentive management oversight, and effective control by shift supervision. Operators used appropriate procedures and control rod pull sheets. Shift turnovers were performed in a controlled manner and crew and pre-evolution briefings were good. The inspectors concluded that the overall startup was performed effectively.

2.2.4 Safety/Relief Valve Leakage

Prior to the shutdown on February 27, 1996, SRVs D, E, and H had indication of pilot valve and/or main seat leakage. FitzPatrick developed a method for estimating SRV leakage based on torus heat-up rate. Based on this method, the most recent total SRV leakage was estimated to be 100 lbs/hr. The licensee planned on a plant shutdown prior to exceeding 600 lbs/hr. The pilot assemblies for D and H and the E SRV main body were replaced during the forced outage and currently SRVs do not have an indication of leakage.

The inspectors monitored FitzPatrick's performance related to SRV leakage. FitzPatrick closely tracked SRV performance through daily torus heat up rate calculations and observations of SRV tailpipe temperature. In addition, the inspectors noted that SRV performance was discussed at the daily plant leadership team meetings. The inspectors concluded that FitzPatrick was sensitive to industry problems related to SRVs.

2.2.5 HPCI Steam Admission Valve Leakage Update (NRC Inspection Report 50-333/95-21)

Following the September forced outage, the periodicity at which the high pressure coolant injection system (HPCI) drain pot alarm actuation had increased to 12 times per shift. The licensee determined the increase in alarm frequency was the result of increased steam leakage past 23-MOV-14, the HPCI turbine steam inlet valve. During the recent forced outage, the inspectors monitored repair activities performed by the maintenance staff on the 23-MOV-14 valve. The valve is a double disc motor operated gate valve which isolates the reactor steam from the relatively ambient temperature and pressure conditions of the HPCI turbine when in the standby line up. The leakage causes a concern because of the detrimental effects to the turbine lube oil systems, potential water hammer damage to the exhaust line and adverse effects on turbine back pressure. The maintenance staff disassembled the valve and machined the valve seats in an attempt to increase the seating force of the valve disk by reducing the seating surface area. Following the repairs the leakage was reduced to an alarm rate of three to four times a shift, however the leakage has since increased to approximately once an hour. The licensee is currently planning additional corrective maintenance for the valve during the next refueling outage. The inspectors concluded that the licensee is taking reasonable measures to continue to monitor the issue.

3.0 MAINTENANCE (62703,61726,92902)

3.1 Maintenance Observation

Maintenance activities were observed during this inspection period on safety-related activities to verify that these activities were being conducted in accordance with approved procedures, technical specifications, and appropriate industrial codes and standards. Observation of activities and review of records included verifying required administrative authorizations and tagouts were obtained, procedures were adequate, certified parts and materials were used, test equipment was calibrated, radiological requirements were implemented, system prints and wire removal documentation were used and quality control hold points were established. Work observed was performed safely and in accordance with proper procedures. The inspectors noted that an appropriate level of supervisory attention was given to the work depending on its priority and difficulty.

- WR 94-04324 Replace 70MOD-105 Damper Actuator per Modification M1-92-394 in the control room ventilation system.

- WR 96-01126 Replace 03TK-125 Accumulator Tank (HCU-06-15) per Maintenance Procedure (MP)-3.10, HCU Water Accumulator (03TK-125) and Nitrogen Cylinder (03TK-128) Maintenance.

- WR 94-00348, 94-00355, 95-00730, 734, 739, 748, 749 Replacement of nitrogen tubing and fittings on various solenoid operated valves for the safety relief valves. (see section 3.1.1)

- 95-06137-01 HPCI Turbine Steam Inlet Isolation Valve

- 96-00503-00 Reactor Water Recirculation Pump B Seal

- 95-0698 Setpoint Change for "E" SRV high temperature alarm

With the exception of the solenoid operated valve failures described below, no concerns were identified during inspector review of the above activities.

3.1.1 Solenoid Operated Valve Failures

On February 26, 1996, during a drywell tour by a plant employee, the solenoid operated valves (SOVs) to the "G" safety/relief valve (SRV) were found to be leaking nitrogen. While troubleshooting, foreign particulate material was obtained from the ports of the SOV block assembly. During subsequent nitrogen flushing of the SOVs, the licensee discovered that two of the eleven sets of SOVs failed to operate correctly from the control room and that two other SOVs failed to operate correctly from the remote automatic depressurization panel outside the control room. The failures were characterized by two modes; failure to cycle completely and failure to reposition closed following deenergization of the solenoid. The licensee replaced all solenoid valves with new or rebuilt SOVs during the forced outage. Following repair activities, an additional failure was discovered when the "G" SRV failed to cycle during start-up testing.

The safety objective of the pressure relief system (11 safety/relief valves) is to prevent overpressurization of the reactor coolant system in order to prevent failure of the reactor coolant system (RCS). The pressure relief system also provides automatic depressurization for small breaks in the reactor coolant system so that the low pressure coolant injection and the core spray systems can inject water to protect the fuel barrier. Manual operation of the SRVs is provided from the control room as well as the reactor building for controlling system pressure following reactor scrams and subsequent plant transients. In the automatic depressurization system (ADS) and manual modes, pressurized nitrogen, controlled by solenoid operated valves, positions the pilot operated safety relief valves (SRV) open and reduces reactor coolant pressure. As stated in the FSAR, the pressure relief system provides for manual depressurization at a remote panel located outside the control room in the highly unlikely event the control room were to become uninhabitable.

Technical Specifications (TS) section 3.6.E, Safety/Relief Valves, require at least 9 of the 11 safety/relief valves to be operable in the safety mode. As only 1 valve failed to cycle open from the control room and one valve did not cycle shut from the control room, the licensee had at least 9 of 11 safety/relief valves operable in the safety mode. TS section 3.5.D, Automatic Depressurization system, also requires that the automatic depressurization system be in service with at least 5 of the 7 ADS valves operable. Of the 2 SRVs which failed to cycle open, only one was an ADS valve which failed to cycle from the reactor building. As the valve cycled properly from the control room, all 7 of the ADS valves were considered operable. In these instances where the SOVs failed to reseat, the ADS function would have been accomplished, opening the SRVs and depressurizing the RCS. The following is a summary of the SOV failures identified:

- The E2 SOV did not open from the remote panel in the reactor building as a result of a loose stem locking nut.

- The L1 SOV did not open from the control room panel as a result of a loose stem locking nut. This SRV is not an ADS valve and therefore would not get an ADS signal to open, but the SOV failure would not have allowed depressurization from the control room. The operators had 10 other SRVs available. The SRV would have functioned from the remote panel in the reactor building.

- The H2 SOV failed to cycle completely shut following opening from the reactor building as a result of foreign material intrusion. This may have prevented the SRV from reseating. The ADS function would not have been inhibited.

- The J1 SOV failed to cycle completely shut following opening from the control room as a result of foreign material.

- The G1 SOV failed to open from the control room during post work testing due to excess Locktite.

The licensee's review of the event determined several reasons for the above listed failures including loose stem locking nuts; debris induced failures; and excessive use of Locktite thread locking compound. Each of these issues is addressed below:

Loose Stem Locking Nut

The effect of the loose stem locking nut is significant because of the adverse affect on the valve seat to disk clearances. The valve stem is threaded into the coil plunger which positions the valve when the coil is energized. A loose stem locking nut may allow the stem to back out of the plunger, reducing the stroke distance, and effectively reduce the valve disk to seat clearance to a point which would restrict the flow of nitrogen through the valve. This restriction would prevent the opening of the SRV. The licensee postulated that the stems were loose because of inadequate torquing and lack of Locktite thread locking compound. Two SOVs failed to cycle because of the loose stem locking nut. Another SOV also had this condition, but not severe enough to prevent operation. These valves were last rebuilt by a vendor at another vendor's test facility. The licensee reported this issue under 10 CFR Part 21 via a 10 CFR 50.73 notification on March 28, 1996.

Excessive Locktite

The licensee determined that excessive use of Locktite thread locking compound by maintenance personnel during the recent rebuild of the Gl SOV was the cause for the valve failure. The licensee found residual Locktite compound between the solenoid plunger and bonnet tube prevented the solenoid from overcoming the spring force required to reposition the valve open.

Foreign Material

In two cases the SOVs repositioned open, but would not go closed following repositioning of the control switch. The licensee attributed these failures to foreign material internal to the SOVs, which was introduced during maintenance activities during the last refueling outage. In addition to these two cases, three other SOVs were found to have some quantity of foreign material in them. The debris was characterized as small metal filings. The licensee determined that the most likely source of the foreign material was debris from maintenance cutting operations. The SOVs had new fittings and supply tubing replaced during the 1994/95 refueling outage.

The inspector reviewed eight different work requests (WR) which involved work on the SOVs during the last refueling outage and had the following observations with regard to foreign material and work control:

Two WRs gave directions to have the system blown down prior to reassembly. These procedure steps were subsequently annotated as NR (not required) by maintenance, central planning, and quality assurance personnel.

Four WRs referenced installation specification IS-S-01, Instrument/Tubing Installation, Revision 4, which provides instructions for the routing of tubing and the fabrication, installation, and inspection of tubing supports at FitzPatrick. Step J.1.3 of the procedure states,"flushing of tubing shall be in accordance with the requirements of the applicable installation procedure." The installation procedures in these instances did not specify any flushing requirements but did request maintenance workers to complete appendix E of IS-S-01. The appendix titled Seismic Tubing Inspection Report, listed several inspection attributes, including "Flushing Complete". These blocks were annotated as N/A (not applicable) by the maintenance staff. The inspection attribute also referenced the previously discussed step J.1.3, which implies that flushing shall be done in accordance with the installation procedure.

All eight WRs included a procedural step referencing the foreign material exclusion (FME) procedure, AP 5.06, for maintenance staff to record the methods used to clean the system internals following debris producing activities on valves. Methods included vacuuming internals; wiping with a tack cloth; and wiping using an approved solvent or demineralized water. Six of the WRs were annotated NR (not required).

Conclusion

Through interviews with station personnel and review of procedures, the inspector determined that there was no formal guidance on what maintenance activities require flushing following work on plant components and systems. Procedures direct that the work instructions have flushing requirements in them, but do not give guidance as to when post maintenance flushing is required.

The inspectors noted strong management oversight and involvement during the trouble shooting, maintenance and testing activities. The issue was aggressively pursued by station personnel and the licensee assessments were self critical. Nonetheless, the maintenance activities previously conducted with respect to the ADS introduced a potential failure which could have affected the satisfactory operation of safety related components.

Technical Specifications 6.8(A)1 require that written procedures and administrative policies shall be established, implemented and maintained that meet or exceed the requirements and recommendations of Section 5 of American National Standards Institute (ANSI) 18.7-1972 "Facility Administrative Policies and Procedures." Section 5 of ANSI 18.7-1972 requires, in part, that facility rules and instructions shall be established pertaining to maintenance that can affect the performance of safety-related equipment and that maintenance that can affect the performance of safety-related equipment shall be properly preplanned and performed in accordance with written procedures which conform to applicable codes, standards, specifications, and criteria. Administrative Procedure (AP) 5.06, "System Internal Cleanliness and Foreign Material Exclusion" establishes requirements for maintaining system and component cleanliness during modification, maintenance, operations and refueling activities. These requirements include administrative controls and techniques established to define practices while working to minimize the introduction of foreign material into systems. Requirements for maintaining system and component cleanliness were not met in that foreign material was found in the pneumatic supply lines and pilot solenoid valves for the safety/relief valves which revealed that previous system maintenance was not properly performed. The inspectors concluded that requirements to maintain system cleanliness were not met which affected the pressure relief system and represent a violation. **(VIO 96-02-01)**

The inspector concluded that the lack of guidance on flushing of systems following maintenance is a weakness.

4.0 SURVEILLANCE

Surveillance activities observed and reviewed emphasized inspection of safety-related activities. Observations of activities and review of records included verifying that required administrative approval was obtained, procedural precautions and limitations were observed, review of test data was accurate and timely, surveillance tests conformed to the technical specification requirements, calibrated test equipment used, radiological controls were observed, and required surveillance frequencies were met. Surveillance activities observed were performed safely and in accordance with proper procedures.

- ST-22B Manual Safety Relief Valve Operation and Valve Monitoring System Functional Test
- ST-3P Core Spray Flow Rate and Valve Inservice Test
- ST-24A, Reactor Core Isolation Cooling Pump and Motor Operated Valve Operability Test
- RAP 7.4.1 Scram Time Test

No concerns were identified during inspector review of the above activities.

5.0 ENGINEERING (37551)

5.1 Temporary Instruction (TI) 2515/128, Rev. 1, Plant Hardware Modifications to Reactor Vessel Water Level Instrumentation (NRC Bulletin 93-03)

The objective of the TI was to verify and evaluate the licensee's implementation of hardware modifications to the reactor vessel water level instrumentation in response to NRC Bulletin 93-03, Resolution Of The Issues Related To Reactor Vessel Water Level Instrumentation In Boiling Water Reactors (BWRs) and to evaluate the licensee's performance implementing the requirements of 10 CFR 50.59 with respect to this design modification.

The BWR reactor vessel water level system is described in Final Safety Analysis Report (FSAR) section 7.8.5.2. The BWR reactor vessel level instrumentation determines the water level in the reactor vessel by measuring the differential pressure between a constant reference leg and a water column connected to the reactor vessel below the water level. The water in the reference leg is maintained at a constant level by condensing steam in a condensate pot that is connected to the steam space above the water level in the reactor vessel. During operation, gases are produced by the radiolytic decomposition of water and can become dissolved in the water in the reference leg. During an event that depressurizes the reference leg, the dissolved gases can be released and displace some of the water in the reference leg. The resulting reduced inventory in the reference leg can result in erroneous indications of high level on vessel water level instruments. Functions associated with the reactor vessel water level instrumentation are critical to the safety and operation of BWRs.

To ameliorate this condition, a back-fill capability for each reactor vessel instrumentation reference leg was installed under modification F1-93-075 in November, 1993. Back-fill is provided from the control rod drive (CRD) system through one tap installed on the CRD system branching out to the five reference column back-fill modules. The back-fill modules contain filters, metering valves, flow monitoring equipment and flow indication. Manually operated metering valves are used to adjust flow rate. The outlet of each backfill module is routed to the reference column.

The inspector reviewed nuclear safety evaluation, JAF-SE-93-072, Reactor Vessel Water Level Backfill Modification and reviewed the modification package F1-93-075. The following attributes of the safety evaluation were observed:

- Metering valves provide manual flow control from the CRD header to the reference legs and are set to a position to provide 0.5 gallons per hour which is the flow requirement determined to be sufficient and have minimal effect on level error.

- Primary containment isolation is provided by the existing excess flow valves and manual isolation valves.

- The licensee has administrative controls in place to prevent misoperation of the manual isolation valve in the reference leg. Isolation of the reference leg due to manual closing of the containment isolation root valve is controlled by a valve line-up completed with a second party check.

- Operating Procedure (OP) 27A, Reactor Water Level Reference Leg Backfill System, includes procedure steps for placing the system in service, flow rate surveillance, changing flow rate when necessary, taking the system out of service and provides specifications for allowable outage time.

- Valves associated with the reactor vessel water level instrumentation system are included in the licensees inservice testing (IST) program.

- The safety evaluation appropriately addressed anticipated transients and their effects on the system.

Based on the inspectors review of the modification package and the safety evaluation, the inspector concluded that the safety evaluation provided a thorough review of the safety implications of the backfill modification. Based on this review, the inspection requirements for TI 2515/128 are considered complete.

6.0 PLANT SUPPORT (71707,40500,92904)

6.1 Radiological Controls

Radiological protection activities were observed on a periodic basis. The activities observed included radiological work practices, radiation surveys and compliance with radiological procedures and requirements. Activities conducted in this area were determined to be acceptable.

6.2 Security

Implementation of portions of the security plan were observed. Areas observed included access point search equipment operation, condition of physical barriers, site access control, security force staffing, and response to system alarms and degraded conditions. These areas of program implementation were determined to be adequate.

6.3 Housekeeping

The inspectors assessed the control of plant housekeeping in safety related areas. They also examined these areas for potential missile hazards such as gas cylinders that could damage safety significant equipment. No concerns were identified.

6.3.1 Drywell Tour

On February 26, 1996, the inspector toured the drywell to walk down portions of various safety related systems not normally accessible during plant operation. The inspector noted a small sheen of oil on one of the "B" reactor water recirculation (RWR) pump foundations. This observation was given to the licensee's drywell coordinator who subsequently determined that the oil had drained out of an RWR pump suction line support (snubber). The appropriate limiting condition for operation (LCO) was entered and the snubber was replaced. An operability evaluation and system walkdown performed by the licensee determined that the piping system was operable and the integrity of the piping was not affected. The inspector noted that the drywell floor, as well as the downcomers, were clear of tools and debris.

The general housekeeping condition of the drywell was found to be good. FitzPatrick's actions to investigate the source of oil was thorough, and corrective actions following the discovery of the drained snubber were appropriate.

6.4 Emergency Preparedness

6.4.1 Practice Emergency Preparedness Drill

On March 28, the FitzPatrick staff conducted a practice emergency preparedness drill. Other than initial telephone notification, there was no participation by offsite emergency response agencies, and the drill was conducted principally for training benefit. The drill progressed to a General Emergency Action Level per FitzPatrick's Emergency Preparedness Plan. The inspectors observed portions of the drill from the simulator, technical support center, and emergency offsite facility. No problems were identified by the inspectors.

6.4.2 Relocation of the Joint News Center

A new Joint News Center (JNC) is under construction and will be located next to the Emergency Operations Facility (EOF) at the Oswego County Airport near Fulton, NY. FitzPatrick is temporarily using the alternate Joint News Center (JNC) at Niagara Mohawk's Corporate headquarters in Syracuse for the period of April 1 to May 6, 1996 for any declared emergencies at the facilities pending completion of the new JNC.

7.0 SAFETY ASSESSMENT/QUALITY VERIFICATION (40500,37551,92700,90712)

7.1 Review of Licensee Event Reports

The inspectors reviewed the following Licensee Event Reports (LERs) and found them to be well written, concise, accurate, and properly submitted for NRC staff review within the guidelines of 10 CFR 50.73:

- LER 96-003, "Plant Shutdown Due to Degraded Control Rod Scram Times and Manual Scram Due to Leak in the Main Turbine Electro-Hydraulic Control System"
- LER 96-004, "Multiple Safety Relief Valve Pilot Solenoid Failures Due to Foreign Materials, Vendor Deficiencies and Procedural Errors"

8.0 MANAGEMENT MEETINGS (30702,71707)

8.1 Review of UFSAR Commitments

A recent discovery of a licensee operating their facility in a manner contrary to the Updated Final Safety Analysis Report (UFSAR) description highlighted the need for a special focused review that compares plant practices, procedures and/or parameters to the UFSAR descriptions. While performing the inspections discussed in this report, the inspectors reviewed the applicable portions of the UFSAR that related to the areas inspected. The following inconsistency was noted between the wording of the UFSAR and the plant practices, procedures and/or parameters observed by the inspectors:

> Use of HPCI and RCIC in the pressure control mode as a depressurization method is not described in the FSAR. (section 2.2.2)

This inconsistency was considered to be minor and no additional documentation is required in this inspection report.

8.2 Exit Meetings

Periodic meetings were held with station management to discuss inspection findings. Following the inspection an exit meeting was held on April 12, 1996, to discuss the inspection findings and observations. FitzPatrick did not object to the findings or observations discussed at the exit meeting. No proprietary information was covered within the scope of the inspection report. No written material regarding the inspection findings was given to FitzPatrick during the inspection period.