UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30323

Report Nos.: 50-250/89-24 and 50-251/89-24

Licensee: Florida Power and Light Company
9250 West Flagler Street
Miami, FL 33102

Docket Nos.: 50-250 and 50-251 License Nos.: DPR-31 and DPR-41

Facility Name: Turkey Point 3 and 4

Inspection Conducted: April 29, 1989 through May 26, 1989

Inspectors: William K. Poertner 6/28/89
R. C. Butcher, Senior Resident Inspector Date Signed

William K. Poertner 6/28/89
T. F. McElhinney, Resident Inspector Date Signed

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Approved by: R. V. Crlenjak 6/28/89
R. V. Crlenjak, Section Chief Date Signed
Division of Reactor Projects

SUMMARY

Scope:

This routine resident inspector inspection entailed direct inspection at the site in the areas of monthly surveillance observations, monthly maintenance observations, engineered safety features walkdowns, operational safety and plant events.

Results:

One violation with two examples was identified: Failure to follow procedure resulting in an inadvertent drop of Rod M-8, paragraph 5. Failure to follow procedure resulting in a reactor trip during surveillance testing, paragraph 10.

One non-cited violation was identified regarding the use of the wrong calibration curves during RTD calibration, paragraph 10.

Two unresolved items** were identified: Determine the cause of inadequate clearance control, paragraph 5. Resolution of document control discrepancies, paragraph 8.

Two Inspector Followup Items were identified: Followup on concerns with the control and storage of hydrogen on site, paragraph 8. Followup on the resolution to correct the failure of MOV-4-751, paragraph 10.

**Unresolved items are matters about which more information is required to determine whether they are acceptable or may involve violations or deviations.

REPORT DETAILS

1. Persons Contacted

Licensee Employees

*J. W. Anderson, Quality Assurance Supervisor
J. Arias, Assistant to Plant Manager
*L. W. Bladow, Plant Quality Assurance Superintendent
*J. E. Cross, Plant Manager-Nuclear
*R. J. Earl, Quality Control Supervisor
T. A. Finn, Training Supervisor
S. T. Hale, Engineering Project Supervisor
K. N. Harris, Site Vice President
*R. J. Gianfrancesco, Maintenance Superintendent
*V. A. Kaminskas, Reactor Engineering Supervisor
J. A. Labarraque, Senior Technical Advisor
R. G. Mende, Operations Supervisor
L. W. Pearce, Operations Superintendent
*S. Quinn, Acting Radiochemist
*F. H. Southworth, Assistant to Site VP
*R. Steinke, Chemistry Supervisor
J. C. Strong, Mechanical Department Supervisor
*K. Van Dyne, Acting Regulatory and Compliance Supervisor
M. B. Wayland, Electrical Department Supervisor
J. D. Webb, Operations - Maintenance Coordinator

Other licensee employees contacted included construction craftsman, engineers, technicians, operators, mechanics, and electricians.

*Attended exit interview on May 26, 1989.

Note: An Alphabetical Tabulation of acronyms used in this report is listed in paragraph 13.

2. Followup on Previous Inspection Findings (92702)

A review was conducted of the following noncompliances to assure that corrective actions were adequately implemented and resulted in conformance with regulatory requirements. Verification of corrective action was achieved through record reviews, observation and discussions with licensee personnel. Licensee correspondence was evaluated to ensure that the responses were timely and that corrective actions were implemented within the time periods specified in the reply.

(Closed) Unresolved Item 50-251/83-39-04, Failure to remove old seal injection flow transmitters. This concern involved flow transmitters which were not shown on drawing 5610-T-E-4503, Reactor Coolant Pump Details, but were still installed. The licensee generated PC/M 84-104,

RCP Seal Leakoff No. 1 Instrumentation, which removed the flow transmitters. The PC/M was completed and turned over to plant operations on September 5, 1986. This item is closed.

3. Followup on Inspector Followup Items (IFIs)

(Closed) IFI 50-250,251/85-37-03. Control Room Noise Level Increase due to Ceiling Insulation Removal. The licensee installed a carpet in the control room to aid in noise reduction. Engineering recommended a carpet with specifications on weight, height and Noise Reduction Coefficient. However, the carpet installed did not completely meet engineerings specifications. Therefore, a noise reduction effect study was performed in March 1988 by a contractor. The study determined that the carpet reduced the noise level by approximately 3 decibels (db). The noise levels measured during normal plant operation were found slightly below the NRC criterion of 65 db maximum. This item is closed.

(Closed) IFI 50-250,251/85-02-04. Engineering to Evaluate the Monitoring of Loss of Control Voltage at the Emergency Diesel Generators (EDGs). The licensee generated Plant Change/Modification (PC/M) 86-185, Annunciation In Main Control Room On Loss of EDG Control Power, due to concerns addressed in LER 50-250,251/85-02. This PC/M added annunciation in the main control room for the A and B EDGs starting circuitry. In order to reduce the probability of blowing fuses, the non-resistored indicating lights were replaced with resistored indicating lights at the A and B EDG engine panels. The PC/M was implemented on March 14, 1989 and March 21, 1989, for A and B EDGs respectively. This item is closed.

4. Onsite Followup and In-Office Review of Written Reports of Nonroutine Events (92700/90712)

The Licensee Event reports (LERs) discussed below were reviewed and closed. The inspectors verified that reporting requirements had been met, root cause analysis was performed, corrective actions appeared appropriate, and generic applicability had been considered. Additionally, the inspectors verified that the licensee had reviewed each event, corrective actions were implemented, responsibility for corrective actions not fully completed was clearly assigned, safety questions had been evaluated and resolved, and violations of regulations or TS conditions had been identified. When applicable, the criteria of 10 CFR 2, Appendix C, were applied.

(Closed) LER 50-250/89-05, Automatic AFW Pump Actuation Following Attempt to Start Steam Generator Main Feedwater Pump. This event was discussed in detail in Inspection Report 50-250,251/89-06 and the corrective actions required are complete. This LER is closed.

(Closed) LER 50-250/89-04, Reactor Trip Due to Defective Procedure During Steam Generator Protection Channel Testing. The event discussed in this LER was identified previously as Violation 50-250,251/89-06-02 and will be followed through closeout of the violation. This LER is closed.

5. Monthly Surveillance Observations (61726)

The inspectors observed TS required surveillance testing and verified: That the test procedure conformed to the requirements of the TS, testing was performed in accordance with adequate procedures, test instrumentation was calibrated, limiting conditions for operation (LCO) were met, test results met acceptance criteria requirements and were reviewed by personnel other than the individual directing the test, deficiencies were identified, as appropriate, and were properly reviewed and resolved by management personnel and that system restoration was adequate. For completed tests, the inspectors verified that testing frequencies were met and tests were performed by qualified individuals.

The inspectors witnessed/reviewed portions of the following test activities:

- 4-PMI-028.3 RPI Hot Calibration, CRDM Stepping Test, and Rod Drop Test.
- TP-522 Unit 4 Alternate Shutdown Panel Performance Test.
- 4-OSP-075.1 Auxiliary Feedwater Train 1 Operability Verification.

a. On May 7, 1989, the licensee experienced an unexpected drop of Rod M-8. The licensee was performing 4-PMI-028.3, RPI Hot Calibration, CRDM Stepping Test and Rod Drop Test, revision dated 3/30/89, Section 6.3, CRDM Stepping Test for Rod H-12, when Rod M-8 dropped. Investigation by the I&C department found that the moveable coil fuse for rod M-8 was pulled, while stepping rod H-12, which caused rod M-8 to drop unexpectedly. A review of Section 6.3 did not identify any provision for removing the moveable gripper coil fuse for any CRDMs during the performance of this section. However, the next section of the procedure, Section 6.4, Rod Drop Test, did provide for removal of these fuses for the bank under test, which included both rods M-8 and H-12. Discussion with personnel involved in the test indicated that miscommunication between personnel in the Control Room and the Motor Control Center (where the fuses were pulled) caused the personnel to mistakenly proceed into the rod drop section of the procedure which caused the event. TS 6.8.1 requires that written procedures and administrative policies shall be established, implemented and maintained that meet or exceed the requirements and recommendations of Appendix A of USNRC Regulatory Guide 1.33 and Sections 5.1 and 5.3 of ANSI N18.7-1972. Contrary to the above, licensee personnel failed to follow procedure 4-PMI-028.3 by proceeding to Section 6.4 of the procedure, prior to completing Section 6.3, resulting in an inadvertent drop of Rod M-8. This is identified as the first example of Violation 50-250,251/89-24-01.

b. The inspectors witnessed the performance of TP-522, Unit 4 Alternate Shutdown Panel Performance Check, on May 24 and 25, 1989. The test was being performed to ensure the unit could be shutdown from outside the control room utilizing the Alternate Shutdown Panel and personnel at various locations throughout the plant to perform local operation of various components. The test started with T AVG stable between 545 and 550 degrees F, being controlled by control room personnel. The normal shift complement was to remain in the control room throughout the test with instructions to abort the test and resume control of the plant if an abnormal situation developed. An additional shift complement dedicated to the test evacuated the control room and proceeded to their assigned stations. At this time control of the systems and components required to place the plant in a cold shutdown condition was shifted to the Alternate Shutdown Panel and a cooldown was commenced. The initial cooldown was conducted by using "B" Auxiliary Feedwater Pump and dumping steam to the atmosphere. When attempting to place the plant on RHR to cooldown to less than 300 degrees F, MOV-4-751 (RHR suction isolation valve from loop "C" hot leg) would not open. The test was terminated and control of the plant was returned to the control room where conditions were maintained stable until the problem with MOV-4-751 was resolved and testing could be resumed. See section 10, Plant Events, for a discussion on this valve problem on May 23, 1989. Testing was resumed on May 24, 1989, and the plant was cooled down to less than 300 degrees F, using the RHR system. At that time the test was completed satisfactorily and control was returned to the control room. The inspectors consider that the test went very well with all systems and components functioning as required, with the exception of MOV-4-751. Minor deficiencies were noted by the inspectors and the licensee and were documented in PTN-OPS-89-154, dated May 30, 1989, with the appropriate corrective actions to be taken as assigned to the responsible departments.

c. On May 22, 1989, the licensee performed 4-OSP-075.1, Auxiliary Feedwater Train 1 Operability Verification Test, after repairs were made on Flow Control Valves (FCV) 2817 and 2818. The test was unsatisfactory due to FCV-2818 failing the test. The test was reperformed after I&C worked on the valve. During the test the C Steam Generator (SG) did not receive AFW flow. Investigation by the licensee revealed clearance 4-89-5-149 was not released by I&C. This clearance isolated FCV-2818, which feeds the C SG. The licensee was investigating the cause of this event at the end of the inspection period. Therefore, this item will be tracked as Unresolved Item 50-250,251/89-24-04.

6. Engineered Safety Features Walkdown (71710)

The inspectors performed an inspection designed to verify the operability of the Emergency Diesel Generators and the Safety Injection System to support reactor startup. The following criteria were used, as appropriate, during this inspection:

a. System lineup procedures matched plant drawings and as built configuration.

b. Housekeeping was adequate and appropriate levels of cleanliness were being maintained.

c. Valves in the system were correctly installed and did not exhibit signs of gross packing leakage, bent stems, missing handwheels or improper labeling.

d. Hangers and supports were made up properly and aligned correctly.

e. Valves in the flow paths were in the correct position as required by the applicable procedures with power available and valves were locked/lock wired as required.

f. Local and remote position indication were compared and remote instrumentation was functional.

g. Major system components were properly labeled.

No violations or deviations were identified in the areas inspected.

7. Monthly Maintenance Observations (62703)

Station maintenance activities on safety related systems and components were observed and reviewed to ascertain that they were conducted in accordance with approved procedures, regulatory guides, industry codes and standards, and in conformance with TS.

The following items were considered during this review, as appropriate: That LCOs were met while components or systems were removed from service; approvals were obtained prior to initiating work; activities were accomplished using approved procedures and were inspected as applicable; procedures used were adequate to control the activity; troubleshooting activities were controlled and repair records accurately reflected the maintenance performed; functional testing and/or calibrations were performed prior to returning components or systems to service; QC records were maintained; activities were accomplished by qualified personnel; parts and materials used were properly certified; radiological controls were properly implemented; QC hold points were established and observed where required; fire prevention controls were implemented; outside contractor force activities were controlled in accordance with the approved QA program; and housekeeping was actively pursued.

The inspectors witnessed/reviewed portions of the following maintenance activities in progress:

- Troubleshooting LT-4-474 and LT-4-475, "A" Steam Generator Level Channels After Failure.

- Troubleshooting of PORV PCV-4-456, to determine cause of spurious opening.

- Replacement of FCV-4-489 and FCV-4-499, "B" and "C" Steam Generator Feedwater Bypass Valves.

- Troubleshooting MOV-4-751 failure to open. See section 10, Plant Events, dated May 23, 1989.

- Troubleshooting Unit 4 AFW Flow Control Valves.

No violations or deviations were identified in the areas inspected.

8. Operational Safety Verification (71707)

The inspectors observed control room operations, reviewed applicable logs, conducted discussions with control room operators, observed shift turnovers and confirmed operability of instrumentation. The inspectors verified the operability of selected emergency systems, verified that maintenance work orders had been submitted as required and that followup and prioritization of work was accomplished. The inspectors reviewed tagout records, verified compliance with TS LCOs and verified the return to service of affected components.

By observation and direct interviews, verification was made that the physical security plan was being implemented.

Plant housekeeping/cleanliness conditions and implementation of radiological controls were observed.

Tours of the intake structure and diesel, auxiliary, control and turbine buildings were conducted to observe plant equipment conditions including potential fire hazards, fluid leaks and excessive vibrations.

The inspectors walked down accessible portions of the following safety related systems to verify operability and proper valve/switch alignment:

A and B Emergency Diesel Generators
Control Room Vertical Panels and Safeguards Racks
Intake Cooling Water Structure
4160 Volt Buses and 480 Volt Load and Motor Control Centers
Unit 3 and 4 Feedwater Platforms
Unit 3 and 4 Condensate Storage Tank Area
Auxiliary Feedwater Area
Unit 3 and 4 Main Steam Platforms

a. In response to NRC Information Notice 89-44, Hydrogen Storage on the Roof of the Control Room, the resident inspectors were requested to canvass their facilities to determine:

- Distance from the hydrogen storage facility to the nearest safety-related structure or air intake.

- Maximum volume of gaseous or liquid hydrogen stored on site in standard cubic feet or gallons respectively.

The residents obtained the above information from the licensee and forwarded it to the region as requested. Based on the information collected and concerns previously expressed to the licensee, the following concerns with hydrogen storage and control of hydrogen on site will be followed up for resolution as Inspector Followup Item 50-250,251/89-24-02.

(1) Administrative controls or limits have not been established on the amount of hydrogen that could be located in the identified hydrogen storage areas.

(2) The back-up hydrogen storage trailer location is within 5 feet of the Unit 3 RWST and about 55 feet from the Unit 4 RWST. Based on the resident inspectors concerns, the back-up hydrogen storage trailer was previously removed from this location. However, there were no administrative controls to prevent locating the hydrogen trailer in this location.

(3) The Gas House, which stores hydrogen gas cylinders, is within 14 feet of the safety injection pump suction line from the RWST for both Unit 3 and Unit 4. The Gas House is also about 15 feet from the Unit 3 RWST and 45 feet from the Unit 4 RWST.

(4) An administrative limit on the quantity of hydrogen that could be brought on site has not been established.

b. While preparing to inspect controlled documents for completeness and the latest revision, the inspectors determined there was insufficient information available from document control to identify the drawings required at a given location and their latest revision number. Subsequently, document control developed a list of drawings required at controlled drawing stations with the latest revision number. The inspectors checked various procedures and drawings at the TSC, Control Room, and I&C Department to determine if they were complete and up to date. The following results were found:

(1) The following discrepancies were found at the TSC.

- AP 0103.18 and AP 0103.36 were missing.

- OP 0204.2 had revisions dated July 26, 1988 and April 26, 1989 in the file.

- HP 11550.70 was in the file but has been superseded.

(2) The following discrepancies were found in drawing control document No. 10 in the control room.

- Drawing 5610-T-E-4062, sheet 3, had revisions 57 and 58 filed in the book.

- Drawing 5610-T-E-4062/R18, sheet 5, revision 1, and 5610-T-E-4534, sheet 1, revision 39 were missing.

The following obsolete red line drawings were still in the book along with the current drawing.

- 5610-T-E-4501/R74, sheet 1, Rev. 0.

- 5610-T-E-4505/R25, sheet 1, Rev. 0.

- 5610-T-E-4505/R3, sheet 5, Rev. 0.

- 5610-T-E-4512/R68, sheet 1, Rev. 2.

- 5610-T-E-4512/R25, Sheet 2, Rev. 1.

- 5610-T-E-4531/R48, sheet 1, Rev. 0.

- 5610-T-E-4534/R18, sheet 2, Rev.0.

- 5610-T-E-4535/R11, sheet 1, Rev.0.

- Drawing 5610-T-E-4532, sheet 1, had revision 7 and 8 filed in the book.

(3) The inspector examined the back log of unfiled documents in the I&C Department and found documents dating back to February 1989 that had not been filed nor had the cover sheet been returned to document control.

Administrative Procedure (AP) 0190.86, Document Control, dated October 27, 1988, provides instructions for the control of documents (drawings and procedures). AP 0190.86, paragraph 5.5, states the holder of a controlled document is responsible for the proper updating and maintenance of controlled documents in their custody, the prompt return of dated and signed receipt acknowledgement cover letters, and superseded controlled documents. Paragraph 8.1.7 of AP 0190.86 requires the document holder complete the cover letter and return it to Document Control within 30 days along with superseded documents. The findings noted above indicate this is not being accomplished.

Quality Instruction (QI) 6-PTN-1, Document Control, dated March 22, 1988, applies to all controlled documents except drawings. QI6-PTN-2, is for drawing control but has not been issued to date.

QI6-PTN-1, paragraph 5.3.5, states if the document cover letter is not received by document control within 30 days, and if the followup letter is not received by document control within 10 days, document control shall retrieve the holder's document and remove it from distribution list if not updated immediately. This action has not been implemented.

QI6-PTN-1, paragraph 5.4, states document control shall prepare a quarterly status report of controlled documents specifying the latest revision to be sent to each holder. Holders shall verify the currentness of their documents and request updates. This action has not been implemented.

QI6-PTN-1, paragraph 5.7, states that document control shall perform an annual review of all controlled documents (that have their master maintained in document control) and problem areas shall be corrected, documented, and reported to the holders manager. This action has not been implemented.

The Quality Assurance (QA) group has an audit in progress, Audit QAO-PTN-89-988, that covers the areas of document control in which discrepancies were noted by the inspectors. The resolution of Document Control discrepancies will be followed as Unresolved Item 50-250,251/89-24-03, pending completion of the QA audit.

No violations or deviations were identified in the areas inspected.

9. Plant Startup from Refueling (71711) (61709) (61710)

The inspectors witnessed/reviewed selected activities related to the Unit 4 Startup From Refueling Cycle XII. These reviews were performed to verify that the licensee properly restored systems effected during the outage and to ascertain whether plant startup and core physics tests were conducted in accordance with approved plant procedures.

a. The inspectors performed a walkthrough of the Emergency Diesel Generators (EDG) and the Safety Injection System (SIS). The following completed procedures were reviewed to verify that these systems were restored properly:

   - 0-OSP-023.1 Diesel Generator Flowpath Verification.

   - 4-OP-062 Safety Injection.

   The inspectors did not identify any discrepancies in the areas reviewed.

b. The inspectors witnessed the licensee's approach to initial criticality on May 19, 1989. Operating Procedure (OP) 0204.3, Initial Criticality After Refueling, dated March 24, 1989, contained the instructions for achieving initial criticality, establishing the upper limit of the neutron flux level for zero power testing and to verify proper operation of the reactivity computer.

Based on this determination, in order to ensure the reactor was operated below nuclear heating, the test range was established in the next lower decade. The reactivity computer was then checked and calibrated in accordance with Appendix B of the procedure and the acceptance criteria was successfully met for the positive and negative period checks.

Test personnel performed a statistical check of the source range instruments in accordance with Appendix D of OP 0204.3. The licensee used the Chi-Squared method with an acceptance criteria of between 16 and 45.7. The results for Source Range Nuclear Instruments (SRNIs) 31 and 32 were 35.68 and 39.735 respectively.

The testing was commenced by withdrawing the control rods until Control Bank (CB) D was at 160 steps. The RCS was then diluted at 100 gpm until the reactor was approximately 2% shutdown. Thereafter, the dilution rate was decreased to 50 gpm. The Inverse Count Rate Ratio (ICRR) was plotted versus the primary water added until the ICRR was approximately 0.10. The dilution was terminated at this time. Criticality was not achieved during the subsequent mixing, therefore, the operators withdrew control bank D in 15 step increments. The reactor went critical at 3:50 p.m. on May 19, 1989 with D bank at 190 steps and boron concentration at 1520 ppm.

Test personnel next established the upper limit of neutron flux level for all zero power physics testing. Appendix A of the procedure provided instructions for the determination of the nuclear heating range. The results were as follows:

Reactivity Computer	3.41x10-7 amps
Intermediate Range Channel N-35	4.54x10-7 amps
Intermediate Range Channel N-36	4.77x10-7 amps

c. Operating procedure 0204.5, Nuclear Design Check Tests During Startup Sequence After Refueling, dated March 24, 1989, specified the Refueling Outage Test Sequence from initial criticality to full power operation. The inspectors witnessed/reviewed the low power tests which were completed prior to the end of this inspection period. The remaining tests will be reviewed during escalation of power. The tests reviewed included:

- Determination of All Rods Out (ARO) Critical Boron Concentration.

- Determination of the Isothermal Temperature Coefficient.

- Determination of Control Rod Group Worths.

- Determination of Differential Boron Worth.

  (a) The boron endpoint on Bank D was performed in accordance with Appendix A of the procedure. The ARO critical boron concentration was measured at 1538 ppm, which was 34 ppm different from the design value. This met the acceptance criteria of ±50 ppm difference.

  (b) The Isothermal Moderator Temperature Coefficient (ITC) was determined in accordance with Appendix B of the procedure. The measured ITC was -.880 pcm/degrees F which met the acceptance criteria of ±2 pcm/degrees F of the design value -1.7 pcm/degrees F. The Moderator Temperature Coefficient (MTC) was determined to be +.92 pcm/degrees F which met the acceptance criteria of ±5 pcm/degres F.

  (c) The rod worths were determined in accordance with Appendix D, Rod Worth Verification by Rod Swap Method. The design report determined Control Bank (CB) C had the greatest worth, therefore, this bank was the reference bank. The integral worth of CB C was measured as 1329 pcm which met the acceptance criteria of ±10 percent difference from the predicted integral worth. The integral worths of each bank were measured and the acceptance criteria of ±15 percent difference from the predicted integral worth was met. The sum of all the control rod worths was measured as 5861 pcm/degrees F, which met the acceptance criteria of ±10 percent of the predicted value.

  (d) The Hot Zero Power (HZP) Differential Boron Worth was calculated using Control Bank C as recommended by Step 8.7 of the procedure. The differential boron worth was measured at 9.6 pcm/ppm which was the same as the predicted value.

No violations or deviations were identified in the areas inspected.

10. Plant Events (93702)

The following plant events were reviewed to determine facility status and the need for further followup action. Plant parameters were evaluated during transient response. The significance of the event was evaluated along with the performance of the appropriate safety systems and the actions taken by the licensee. The inspectors verified that required notifications were made to the NRC. Evaluations were performed relative to the need for additional NRC response to the event. Additionally, the following issues were examined, as appropriate: details regarding the cause of the event; event chronology; safety system performance; licensee compliance with approved procedures; radiological consequences, if any; and proposed corrective actions.

On May 3, 1989, at 1:45 p.m., with Unit 3 in Mode 5 and Unit 4 in Mode 3, a loss of the Emergency Notification System (ENS) communications was identified. The licensee notified the NRC in accordance with 10 CFR 50.72 (b)(1)(v). The local phone company was contacted concerning the failure and the required communications were re-established at 2:45 p.m.

On May 4, 1989, at 4:50 p.m., with Unit 4 in Mode 3, PORV PCV-4-456 opened momentarily then reclosed causing a 45 psig reduction in RCS pressure. RCS pressure was at 2235 psig which was well below the PORV setpoint of 2335 psig. The RCO closed the PORV block valve (MOV-4-535) and removed power to it as directed by ONOP-1208.1. Subsequent troubleshooting determined the spurious valve actuation was caused by a failure of PC-4-445A, a setpoint comparator in the PORV's control loop. The faulty comparator was replaced and the valve was returned to service. The licensee is currently conducting root cause analysis to determine the cause of the failure.

On May 5, 1989, at 1:52 a.m., with Unit 4 in Mode 3, and with control banks C & D withdrawn, a Reactor Protection System (RPS) trip occurred. While performing 4-SMI-071.4, Steam Generator Protection Set III Analog Channel Test, the Reactor Trip Breakers (RTBs) opened when bistable BS-4-446-1 was placed in the Test position in accordance with step 6.2.3.1 of the procedure. This simulated a reactor power greater than 10% enabling the P7 trips. The RTBs opened due to the presence of the Turbine Stop Valves Closed signal which was introduced due to leads being lifted to the Stop Valve limit switches. All systems functioned as designed. The sequence of events was repeated and the same results were obtained. The procedure was stopped, notification was completed per 10 CFR 50.72 (b)(2)(ii), and an Event Response Team (ERT) was convened to review the incident. Although the turbine stop valves were open as required by procedure 4-SMI-071.4, the lifted leads (4T10 and 4T11) prevented the "Turbine Stop Valve Closed" contacts in the RPS circuitry from closing. This in turn gave the RPS a false signal indicating that the turbine stop valves were closed. When bistable BS-4-446-1 was placed in the tripped position, indicating power greater than 10%, with the stop valves indicating closed, a reactor trip occurred. The ERT formed a task team to determine the root cause for the lifted leads not being relanded. However, the team could find no documentation or information identifying that the leads were lifted. TS 6.8.1 requires that written procedures and administrative policies shall be established, implemented and maintained that meet or exceed the requirements and recommendations of Appendix A of USNRC Regulatory Guide 1.33 and Sections 5.1 and 5.3 of ANSI N18.7-1972. O-GME-102.1, Troubleshooting and Repair Guidelines, step 3.1 requires that a PWO shall be issued prior to commencing work and step 6.2.7 requires that all lifted leads be documented and independently verified. Steps 6.2.9 and 6.3.8 require all lifted leads be reconnected and independently verified. O-ADM-715, Maintenance Procedure Usage, steps 5.5.2 and 5.5.3, provide instructions for independent verification of lifting and relanding

leads. Contrary to the above, leads to the turbine stop valve limit switches were lifted without adequate controls which resulted in a reactor trip during surveillance testing. This is identified as the second example of Violation 50-25,251/89-24-01.

On May 8, 1989, at 6:00 p.m., with Unit 4 in Mode 3, the unit was placed in Technical Specification 3.0.1 due to both the "B" and "C" loop T AVG Resistance Temperature Detectors (RTDs) being out of service. The elements (TE-422 and TE-432) were placed out of service when the I&C department discovered that the wrong calibration data was used to calibrate the RTDs. The Plant Supervisor-Nuclear placed the unit in TS 3.0.1 as required by Table 3.5-2, item 1.5, which requires the High Steam Flow in two-out-of-three steam lines with Low T AVG or Low Steam Line Pressure channels be functional. The bistables for TE-422 and TE-432 were tripped in accordance with ONOP-208.14 and the NRC was notified of the event as required by 10 CFR 50.72(b)(1)(i)(A). The licensee's investigation into the event determined that PC/M 88-234, performed during the current Unit 4 outage, required that the RCS temperature RTDs be replaced. When the new RTDs were received on site, the new response curves for the RTDs to be installed in Unit 4 were obtained and included into procedures to assure the data was readily available when required for testing after installation. During pre-installation testing of the RTDs it was determined that three of them did not meet the acceptance criteria and NCR 89-0250 was initiated to document the deficiency. The NCR was dispositioned by allowing the use of three RTDs that were to be used in Unit 3. However, the calibration curves for the Unit 3 RTDs that were used in place of the defective Unit 4 RTDs were not included into the calibration procedures. Therefore, when the RTDs were calibrated, they were calibrated to the wrong curves. The licensee took prompt corrective action when the discrepancy was identified and corrected the affected procedures. Westinghouse verified that the new curves contained in the updated procedures were for the specific RTDs installed and the circuitry was recalibrated to the new curves. Westinghouse has reviewed the assumptions contained in the Safety Analysis against the calibration data of the installed RTDs. This review concluded that the installation correction values for the RTDs were within the assumptions contained in the Safety Analysis. Therefore, the "B" and "C" loop RPS T AVG indication was not Out-of-Service due to being Out-of-Calibration. Based on this information it was concluded no Technical Specification (TS) violation existed and TS 3.0.1 need not to have been entered. The evolution described above constitutes a violation of TS 6.8.1 in that procedures were not adequately implemented or maintained to ensure that the correct RTD calibration curves were included in the appropriate procedures. It was determined this violation meets the criteria of 10 CFR 2, Appendix C, therefore, no notice of violation will be issued. This item is identified as non-cited violation (NCV) 250,251/89-24-06.

On May 9, 1989, with Unit 4 in Mode 3, the unit experienced a feedwater isolation which was reported to the NRC under 10 CFR 50.72(b)(2)(ii). The unit was in the process of cooling down from Mode 3 to Mode 4 with one level channel (LT-4-475) for the "A" steam generator out of service, with its bistables tripped, due to a level deviation of greater than 10%. During the cooldown an additional channel for the "A" steam generator (LT-4-474) failed high which made up the necessary two out of three logic to cause a feedwater isolation signal for steam generator "A". All systems functioned as required. Subsequent troubleshooting of the level transmitters indicated the sensing lines contained some sludge which was flushed from the lines and the transmitters were successfully recalibrated. The licensee considers the sludge was caused by sludge-lancing of the steam generators during the outage and not performing a flush of lines coming from the steam generator. The licensee stated future sludge-lancing evolutions would require an adequate flush of the lines.

On May 11, 1989, the licensee conducted a cooldown of Unit 4 from Mode 3 to Mode 4, to facilitate repair of FCV-4-489 B and C, steam generator feedwater bypass valves. During operation in Mode 3 the operators noted that seat leakage past the valves was excessive and they felt it would be impossible to control temperature within the narrow band required during the upcoming post refueling low power physics testing. Therefore, the licensee cooled down the unit to less than 350 degrees F and replaced the valves. The valves were replaced due to damaged internal valve body threads which retain the internal throttling cage assembly. These same internal threads were previously repaired by weld buildup and remachining in accordance with NCR 86-083 in March of 1986.

On May 14, 1989, the licensee experienced another loss of Emergency Notification System (ENS) communications. The event occurred at 1:32 a.m., when a 4A Primary Water Pump Motor fault occurred which tripped the "D" MCC Breaker Number 0832, removing incoming power from the MCC. This in turn caused a loss of power to lighting panel Number 33A de-energizing Breaker Number 9, which removed power from the ENS phone. The licensee reported the event in accordance with 10 CFR 50.72(b)(1)(v) upon discovery of the loss of ENS.

On May 23, 1989, during the performance of TP-522, Unit 4 Alternate Shutdown Panel Performance Check, MOV-4-751 (RHR suction isolation valve) failed to open. The licensee made a significant event report in accordance with 10 CFR 50.72(b)(2)(iii)(B). After the valve failed to open due to the breaker tripping on thermal overload, two additional attempts were made with identical results. The licensee then shut MOV-4-750 (the upstream isolation valve to MOV-4-751) to reduce the differential pressure across MOV-4-751. A final attempt was made to open the valve and it again tripped on thermal overload. Maintenance personnel then partially opened the valve manually, which was very hard to operate until the valve disc partially cleared the seats, indicating the valve was

binding during the inital portion of travel. The valve was then MOVATS tested successfully and cycled several times with no problems being identified. The inital indication is that the valve may have been "pressure bound", a phenomenon by which pressure is induced between the discs of the valve. This, in turn, causes the discs to exert greater pressure against the seats. This causes the valve to bind until the discs partially clear the seat area during opening. This would allow the internal pressure between the discs to be relieved thus eliminating the "pressure binding". The licensee is currently performing tests on the valve and discussing the event with the vendor and other utilities to determine the root cause of the failure and possible corrective actions to be taken. The resolution to correct the failure of MOV-4-751 will be tracked as Inspector Followup Item 50-250,251/89-24-05.

11. Management Meeting (94702)

On May 10, 1989, the bi-monthly NRC/FPL Management Meeting was conducted at the site. This meeting was the eleventh in a series of management meetings following issuance of Confirmatory Order 87-85 in October 1987. The meeting was attended by NRC Regional and Headquarters Management and FPL Site and Corporate Management. The topics of discussion included overall plant status, recent operational events, engineering, maintenance, and security initiatives.

12. Exit Interview (30703)

The inspection scope and findings were summarized during management interviews held throughout the reporting period with the Plant Manager - Nuclear and selected members of his staff. An exit meeting was conducted on May 26, 1989. The areas requiring management attention were reviewed. No proprietary information was provided to the inspectors during the reporting period. The inspectors had the following findings:

50-250,251/89-24-01, Violation. Failure to meet the requirements of TS 6.8.1, two examples: Failure to follow procedure resulting in an inadvertent drop of Rod M-8; and failure to follow procedure resulting in a reactor trip during surveillance testing. (paragraph 5 and 10).

50-250,251/89-24-02, Inspector Followup Item. Followup on concerns identified with the storage and control of hydrogen on site (paragraph 8).

50-250,251/89-24-03, Unresolved Item. Resolution of document control discrepancies. (paragraph 8).

50-250,251/89-24-04, Unresolved Item. Determine the cause of inadequate clearance control. (paragraph 5).

50-250,251/89-24-05, Inspector Followup Item. Followup on the resolution to correct the failure of MOV-4-751, (paragraph 10).

50-250,251/89-24-06, non-cited violation with no written notice of violation regarding the use of the wrong RTD calibration curves. (paragraph 10).

13. Acronyms and Abbreviations

ADM	Administrative
ANSI	American National Standards Institute
AP	Administrative Procedures
ARO	all rods out
ASME	American Society of Mechanical Engineers
CB	Control Bank
CCW	Component Cooling Water
CCTV	Closed Circuit Television
CFR	Code of Federal Regulations
CS	Containment Spray
DP	Differential Pressure
ENS	Emergency Notification System
ERT	Event Response Team
FPL	Florida Power & Light
FSAR	Final Safety Analysis Report
HHSI	High Head Safety Injection
ICRR	Inverse Count Rate Ratio
ICW	Intake Cooling Water
IEB	Inspection and Enforcement Bulletin
IFI	Inspector Followup Item
ITC	Isothermal Temperature Coefficient
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LIV	Licensee Identified Violation
LOCA	Loss of Coolant Accident
MP	Maintenance Procedures
MTC	Moderator Temperature Coefficient
NCR	Non-conformance Report
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
ONOP	Off Normal Operating Procedure
OOS	Out of Service
OP	Operating Procedure
OTSC	On the Spot Change
PA	Protected Area
PC/M	Plant Change/Modification
pcm	Percent Millirho
ppm	Parts Per Million
PNSC	Plant Nuclear Safety Committee
PSN	Plant Supervisor Nuclear
PSP	Physical Security Procedures
QA	Quality Assurance

QC	Quality Control
RCO	Reactor Control Operator
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RPS	Reactor Protection System
RTD	Resistance Temperature Detectors
RTB	Reactor Trip Breaker
SRNI	Source Range Nuclear Instrument
SRO	Senior Reactor Operator
T AVG	Average Reactor Coolant Temperature
TS	Technical Specification
TSA	Temporary System Alteration
URI	Unresolved Item