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

Report Nos.: 50-250/87-39 and 50-251/87-39

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: August 24 - September 21, 1987

Inspectors: [signature] 10/23/87
D. R. Brewer, Senior Resident Inspector — Date Signed

[signature] 10/23/87
J. B. MacDonald, Resident Inspector — Date Signed

Approved by: [signature] 10/23/87
Bruce Wilson, Section Chief — Date Signed
Division of Reactor Projects

SUMMARY

Scope: This routine, unannounced inspection entailed direct inspection at the site, including backshift inspection, in the areas of annual and monthly surveillance, maintenance observations and reviews, engineered safety features, operational safety, plant events, and refueling startup.

Results: Two violations were identified: Failure to implement maintenance procedures for the control of fuses in the safeguards protection circuitry (paragraph 8); and the failure to meet the requirements of 10 CFR 50, Appendix B, Criterion V, two examples; in that a required post maintenance test was not performed on flow transmitter 3-475 and a check valve other than that specified on a construction drawing was installed in a safety-related system (paragraph 8).

REPORT DETAILS

1. Persons Contacted

Licensee Employees

J. S. Odom, Vice President
*C. J. Baker, Plant Manager-Nuclear
*F. H. Southworth, Maintenance Superintendent
D. A. Chaney, Site Engineering Manager (SEM)
*D. D. Grandage, Operations Superintendent
T. A. Finn, Training Supervisor
J. D. Webb, Operations - Maintenance Coordinator
D. H. Taylor, Operations System Enhancement Coordinator
J. W. Kappes, Performance Enhancement Coordinator
R. A. Longtemps, Mechanical Maintenance Department Supervisor
*D. Tomasewski, Instrument and Control (I&C) Department Supervisor
*J. C. Strong, Electrical Department Supervisor
*W. Bladow, Quality Assurance (QA) Superintendent
R. E. Lee, Quality Control Inspector
E. F. Hayes, Quality Control (QC) Supervisor
*J. A. Labarraque, Technical Department Supervisor
R. G. Mende, Operations Supervisor
J. Arias, Regulation and Compliance Supervisor
R. D. Hart, Regulation and Compliance Engineer
*W. C. Miller, Senior Technical Advisor
V. Kaminskas, Reactor Engineering Supervisor
P. W. Hughes, Health Physics Supervisor
*G. Solomon, Regulation and Compliance Engineer
*J. Donis, Engineering Department Supervisor
*D. E. Meils, Chemistry Supervisor
*D. W. Jones, Procedure Upgrade Program Supervisor
*S. D. Ferrell, Licensing Engineer
*A. G. Abbott, Startup Administrative Coordinator

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

*Attended exit interview on September 21, 1987.

2. Exit Interview

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 September 21, 1987. The areas requiring management attention were reviewed. The licensee acknowledged the findings without exception. No proprietary information was provided to the inspectors during the reporting period.

Two violations were identified:

Failure to meet the requirements of Technical Specification (TS) 6.8.1, in that fuses were not controlled as required by maintenance procedures (paragraph 8, 250, 251/87-39-01).

Failure to meet the requirements of 10 CFR 50, Appendix B, Criterion V, two examples, in that a required post maintenance test was not performed and an incorrect check valve was installed in a safety-related nitrogen system (paragraph 8, 250, 251/87-39-02).

3. Unresolved Items (URI)

Unresolved items are matters about which more information is required to determine whether they are acceptable or may involve violations of requirements or deviations from commitments. No unresolved items are identified in this report.

4. Followup on Items of Noncompliance (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) Violation 250, 251/85-13-02. Failure to meet the requirements of 10 CFR 50, Appendix B, Criterion XIII, Handling, Storage and Shipping - three examples. FPL response letter dated March 14, 1986, was found to be acceptable (per NRC letter dated April 11, 1986). The inspector reviewed Operating Procedure 16009.11, On Site Unpacking, Inspection, and Manual Loading of Hafnium Vessel Flux Depression Assemblies, and Maintenance Procedure 0736, Heavy Load Handling which addresses the concerns raised in this violation. Violation 250, 251/85-13-02 is closed.

(Closed) Violation 251, 251/85-13-03. Failure to meet the requirements of 10 CFR 50, Appendix B, Criterion XV, Control of Nonconforming Materials. FPL response letter dated March 14, 1986, was found to be acceptable (per NRC letter dated April 11, 1986). The inspector reviewed the Q.C. Inspection Checklists developed to resolve this issue. Violation 250, 251/85-13-03 is closed.

(Closed) Violation 250, 251/86-10-01. Failure to meet T.S. 3.8.1, three examples; OP 3400.1, PLS Book, ONOP 9608.1. FPL response letter dated April 16, 1986, was found to be acceptable (per NRC letter dated June 5, 1986). Part B of this violation was addressed in Inspection Report 250, 251/86-35. The inspector reviewed the procedure changes made to ONOP

0208.1 (Annunciator List - Panel H - Safety injection and Auxiliary), ONOP 9608.1 (125 V DC System Location of Grounds), and OP 3400.1 (Intake Cooling Water) which resolved the remaining issues. Violation 250, 251/86-10-01 is closed.

(Closed) Violation 251/83-32-04. Procedure 4-GOP-503, Cold Shutdown to Hot Standby, revision dated August 6, 1987, requires as a prerequisite, that the emergency containment cooling and filtering system be in standby in accordance with procedure 4-OP-055, Emergency Containment Cooling and Filtering, prior to exceeding 350F. 4-OP-055, revision dated October 28, 1986, properly identifies by number and position the valves affected by PCM 80-96. Violation 251/83-32-04 is closed.

(Closed) Violation 251/83-32-05. Failure to Follow procedures - three examples. Licensee correspondence to Region II, L-84-176, dated July 13, 1984, addressed the continuing concern for procedural noncompliances and summarized efforts taken to minimize recurrences. The licensee implemented procedure 0-ADM-201, Upgrade Operations Procedure Usage, revision dated July 18, 1987, which delineates the exceptions, clarifications, interpretations and requirements for verbatim compliance to procedures. Violation 251/83-32-05 is closed.

5. Followup on Unresolved Items (URIs), Inspector Followup Items (IFIs), Inspection and Enforcement Information Notices (IENs), IE Bulletins (IEBs) (information only), IE Circulars (IECs), and NRC Requests (92701)

(Closed) URI 251/83-32-06. Concerning labeling discrepancies related to PCM 80-96 effecting isolation valves in the Emergency Containment Filter system. The licensee has established a comprehensive component labeling and tagging program. Procedure AP-0103.34, Component Labeling/Tagging most recent revision dated May 19, 1987, outlines responsibilities and requirements for identifying and correcting labeling deficiencies. Since the program has been implemented, only a very few minor labeling discrepancies have been observed. URI 251/83-32-06 is closed.

(Closed) IFI 250, 251/85-30-06. Failure of purchasing department to properly expedite the purchase of repair parts. The inspector reviewed FPL memorandum from O. Arredondo to J. W. Kappes, dated January 6, 1986, which discusses this issue. The purchasing department had implemented a program for monitoring every purchase order processed to ensure delays are minimized. IFI 250, 251/85-30-06 is closed.

(Closed) IFI 250, 251/86-05-02. Improve AP 109.3, On the Spot Change (OTSC) procedure to minimize nonessential OTSC changes. AP 109.3 was modified June 5, 1986 and June 18, 1987 to address the concerns of this issue. IFI 250, 251/86-05-02 is closed.

(Closed) IFI 250/83-09-04. Concerning the need for including the surveillance schedule checklists in an approved procedure. The licensee implemented a single comprehensive operating surveillance procedure

0-OSP-200.1, Schedule of Plant Checks and Surveillances, which incorporated all surveillance, schedule and departmental responsibilities. In addition, 0-OSP-200.2, Plant Startup Surveillances, and Operating Procedure 204.2 contain the applicable check lists. IFI 250/83-09-04 is closed.

(Closed) IFI 250/84-23-09 and 251/84-24-09. Concerning the issue that local gauges used only for local indication are not calibrated on a routine schedule unless they are part of a remote indication calibration loop. The licensee implemented Administrative Procedure 0190.26, Calibration Control of Installed Nuclear Safety Related Instrumentation and Control Equipment, which resolves this issue. This procedure requires that each responsible department determine a calibration frequency based on manufacturer's recommendations, operating experience and Technical Specification requirements and establish a schedule to accomplish these calibrations. IFI 250/84-23-09 and 251/84-24-09 are closed.

(Closed) IFI 250/87-35-06. Determine how the wrong check valve was installed in the train 1 AFW nitrogen back up system. The inspectors have determined that construction personnel failed to ensure that the right check valve was installed. Violation 250/87-39-02 was issued as a result. This item is discussed in detail in paragraph 8. IFI 250/87-35-06 is closed.

(Closed) IFI 251/83-39-10. Concerning testing of the Auxiliary Feed Water (AFW) pumps. From December 5, 1983 to December 16, 1983, several undocumented AFW pump surveillance test runs were performed. Violation 250, 251/84-04-01 was issued as a result, and encompasses the concerns of this IFI. The corrective actions to the violation were reviewed and the violation was closed in paragraph 3 of inspection report 250,251/87-27. IFI 251/83-39-10 is closed.

(Closed) URI 250, 251/87-35-04, Changes to process sheets subsequent to Plant Nuclear Safety Committee (PNSC) approval. The inspectors expressed to the licensee the concern that intent changes to Plant Change Modification (PCM) process sheets after the PCM has been PNSC approved, may be contrary to requirements of TS 6.5.1.6.d. The inspectors were informed by the licensee that a previous plant QA finding was issued relative to this issue. The corrective actions to the QA finding were reviewed and appear to be adequate to preclude recurrence. Since this was initially a licensee identified item and corrective actions were implemented, 10 CFR Part 2, Criterion C guidelines for mitigation of potential enforcement action apply. This item is discussed in greater detail in paragraph 8. URI 250, 251/87-35-04 is closed.

6. Onsite Followup and In-Office Review of Written Reports Of Nonroutine Events (92700/92712)

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.

(Closed) LER 250-86-005. Failure of Main Steam Isolation Valve (MSIV) 3C to close during Unit 3 startup. The inspector reviewed the change made to the Main Steam Isolation Valve Closure Test Procedures (3-OSP-072 and 4-OSP-072) to require testing of both air supply paths to each MSIV prior to performing the timing portion of the test. LER 250-86-005 is closed.

(Closed) LER 250-86-028. Potential for the loss of the minimum flow for the safety injection pumps. Subsequent to the issuance of this LER, IE Bulletin 86-03 (Potential Failure of Multiple ECCS Pumps Due to Single Failure of Air-Operated Valve in Minimum Flow Recirculation Line) was issued. The actions required by this LER will be followed in the closeout of IEB 86-03. LER 250-86-028 is closed.

(Closed) LER 250-86-009. Potential concern existed associated with the CCW flow through the RHR heat exchangers. The licensee completed all flow balance testing and evaluations in February and March, portions of which were witnessed by the NRC and discussed in Inspection Report 50-250, 251/87-07. NRC staff members audited the results and informed FPL, in a letter from D. G. McDonald to C. O. Woody dated February 5, 1987, that the NRC agreed the CCW system is balanced and the safety-related components are being cooled per their design requirements. LER 250-86-009 is closed.

(Closed) LER 250/87-12, Emergency Diesel Generator (EDG) Auto Start Due to Personnel Error. On May 7, 1987, the B EDG auto started as a result of personnel error in the performance of restoration of a clearance order. Violation 250, 251/87-22-01 was subsequently issued. The corrective actions to the event will be tracked via the violation. LER 250/87-12 is closed.

(Closed) LER 250/87-13, Missed Technical Specification (TS) Surveillance of Spent Fuel Pool (SFP) Exhaust Monitor. On May 14, 1987, it was discovered that the TS surveillance due by April 18, 1987 for the SFP exhaust monitor had not been performed. Violation 250, 251/87-27-01 was subsequently issued. The corrective actions to this event will be tracked via the violation. LER 250/87-13 is closed.

(Closed) LER 250/87-14, Containment Spray (CS) System Design Discrepancy. On May 19, 1987, with both units in outages, it was determined that CS pump discharge flow orifices required by design had never been installed. PCM 87-194 (Unit 3) and PCM 87-177 were generated and the orifices were installed. A complete review of the PCMs is documented in paragraph fourteen of inspection report 250, 251/87-35. LER 250/87-14 is closed.

(Closed) LER 250/87-17, Loss of All Boric Acid Flowpaths. Several times from May 28, 1987 to June 3, 1987, all boric acid flowpaths were lost to each unit. During troubleshooting activities configuration control of the system was lost. A Region II team inspection investigated the events. The inspection findings are documented in inspection report 250,251/87-28. Two violations were identified 250,251/87-28-01 and 250,251/87-28-02. Due to the significance of this event these violations are presently under consideration for escalated enforcement action. The corrective actions of this event will be tracked via the above violations. LER 250/87-17 is closed.

(Closed) LER 250/87-18, Switchover Time Safety Injection Phase to Recirculation Phase. Westinghouse (W) letter FPL-87-578, dated March 11, 1987, documented that evaluations performed by W could not demonstrate that a ten minute interruption of ECCS flow during switchover from the injection to the recirculation phase to be acceptable. The analysis concluded that a maximum of two minutes could be tolerated without potentially exceeding peak cladding temperature requirements. The licensee performed safety evaluation JPE-LR-87-017. The evaluation concluded that Emergency Operating Procedures (EOP) needed to be revised to ensure that switchover from the injection phase to the recirculation phase occur within fifty seconds. The inspectors reviewed 3/4-EOP-ES-1.3, Transfer to Cold Leg Recirculation, revision dated June 16, 1987 and May 26, 1987 respectfully and 3/4-EOP-1.4, Transfer to Hot Leg Recirculation, revision dated May 26, 1987 and June 16, 1987 respectfully, to ensure that the procedural changes had been properly implemented. The licensee is reviewing this event with respect to 10 CFR 21 requirements and also is studying potential ECCS modifications to improve the switchover process. LER 250/87-18 is closed.

(Closed) LER 250/87-20, ICW/CCW Heat Exchanger Concerns. On June 16, 1987, upon review of CCW heat exchanger performance test data the inspectors brought to the attention of the licensee that on December 11, 1986 for approximately seventeen hours Unit 3 had operated outside the design basis of the CCW system. The scenario is documented in paragraph nine of inspection report 250,251/87-27. Unresolved Item 250,251/87-27-02 was subsequently generated. The corrective actions of this LER will be tracked via the unresolved item. LER 250/87-20 is closed.

7. Monthly and Annual Surveillance Observation (61726/61700)

The inspectors observed TS required surveillance testing and verified: that the test procedure conformed to the requirements of the TS, that testing was performed in accordance with adequate procedures, that test instrumentation was calibrated, that limiting conditions for operation (LCO) were met, that test results met acceptance criteria requirements and were reviewed by personnel other than the individual directing the test, that 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:

Auxiliary Feedwater System Surveillance Tests 3/4-OSP-75.1 and 75.2
Auxiliary Feedwater System Post Modification Tests 800.116 & 800.163
Boric Acid Storage Tank Chemistry Sampling Procedure NC-2
Auxiliary Feedwater System Nitrogen Test 3-OSP-075.7

During the recently completed Unit 3 refueling outage the Auxiliary Feedwater (AFW) system was modified by installing an improved steam line check valve design and increasing the size of the backup nitrogen supply system. Those portions of the post modification tests involving operation of the steam driven pumps can not be performed until after unit startup (mode 2) because insufficient heat is available in modes 3, 4, 5 and 6. Similarly, during cold shutdowns, monthly surveillances which would normally be due during the outage are not required to be performed, as specified in TS 4.10. The post shutdown testing requirement is met by performing the surveillances during startup subsequent to cold shutdown, as specified in TS 4.10.1. Consequently, though the AFW system is required to be operable in modes 1, 2 and 3, it is not tested until after passing through mode 3 into mode 2.

Immediately following startup, the Unit 3 reactor core physics tests were performed. It is appropriate that these tests be performed immediately after the initial criticality, since the reactor core was altered during the refueling outage. The Unit 3 low power physics tests were completed at 4:00am on September 6, 1987.

AFW system post modification test, Preoperational Procedure (POP) 0800.163 was started at 10:14am on September 6, 1987 and was terminated without completion at 1:11pm due to the failure of flow control valve 3-2831 to close when required. Train 2 of the AFW system was declared out of service, placing the unit in a 72 hour limiting condition for operation (LCO). The valve was repaired at 10:50pm and declared back in service subsequent to successful stroke testing in accordance with procedure 3-OSP-75.10.

At 3:45am on September 7, 1987, the C AFW pump was taken out of service because a white deposit in the governor control oil sight glass was thought to indicate water intrusion into the oil. POP 0800.163 had not yet been restarted. Another post modification test procedure, POP 0800.116, Unit 3 AFW Nitrogen Backup System Test, had not been started. Additionally, monthly surveillance testing required to be performed in startup (mode 2) subsequent to exiting cold shutdown had not yet been performed.

Plant Work orders were written at approximately 3:30am on September 7, 1987 for both the B and C AFW pumps because the governor control oil appeared cloudy, a condition often indicative of water contamination. Only the C AFW pump was declared out of service. Its oil was white and opaque while the B governor oil was light white and remained transparent.

TS 3.18.1.3 specifies that, with a single AFW pump inoperable, within 4 hours, the operability of the two independent AFW trains shall be verified. If both trains are determined to be operable, a 30 day LCO would exist. If only one of two trains were operable, a 72 hour LCO would exist. If neither train were operable, a plant shutdown followed by a cooldown would be required. As specified in TS basis B3.18-1, the verification of operable trains is accomplished by verifying that the trains are aligned as specified in the allowable options of TS Table 3.18-1 and determining that the trains have been successfully tested within the last surveillance interval.

When the C AFW pump was removed from service, the Unit 3 surveillances and post maintenance tests required to be performed, subsequent to entering mode 2, had not yet been performed. POP 0800.163 had been initiated and terminated while only partially complete because a train 2 flow control valve failed open. The only successful Unit 3 system tests which existed were performed before March 1987, prior to the beginning of the refueling outage. These tests provided little insight into system operability because system modifications had been performed during the outage.

Licensee personnel specified that a 30 day LCO had been entered when the C AFW pump was removed from service. Discussions with the licensee revealed that the operators felt the trains could be considered operable since no Unit 3 surveillance requirements had been exceeded. Additionally, the common A and B AFW pumps had been successfully tested, to Unit 4 only, in accordance with the Unit 4 monthly surveillance program.

While it is true that no Unit 3 surveillance was overdue, an operability concern existed because no surveillance existed which could be considered valid. The 4 hour verification time period allowed in TS 3.18.3 could have been used to perform the Unit 3 monthly surveillances (on one or both trains) to provide meaningful test results upon which to base an operability verification. No attempt was made to implement either the routine surveillances of 3-OSP-075.1 and 3-OSP-075.2 or to complete POP 0800.163 and POP 0800.116 within 4 hours of removing the C pump from service. Consequently, an unnecessary lack of system operational data existed.

Additionally, the B AFW train operability was suspect due to the cloudy white appearance of the governor oil in the B AFW pump. Both the B and the C AFW pumps received essentially the same Plant Work Order (PWO). The C pump defect statement (PWO WA872500330) was, "White deposit bottom of oil sight glass, possible water". The PWO statement of seriousness specified, "could render pump inoperable". The B pump defect statement (PWO WA872500323) was, "B AFW governor oil is cloudy (water)", and the statement of seriousness was, "could render pump inoperable due to governor problems". Although both B and C pump oil problems were observed, only the C pump was addressed as an operability concern necessitating removal from service. The B pump was not removed from service because visual observation of the oil revealed a less severe cloudy white condition than seen on the C pump.

A concern existed as to the operability of the B AFW pump as documented on PWO WA872500323. Additionally, no valid surveillance data existed for the Unit 3 AFW system. Nevertheless, operators assumed that both AFW train 1, supplied by the A pump, and AFW train 2, supplied by the B pump, were operable in declaring a 30 day LCO subsequent to removing the C pump from service. Had the B pump also been considered inoperable, the allowable LCO would have been limited to 72 hours.

The C AFW pump remained out of service for approximately 18 hours during which time the oil was changed and the governor was vented. Pump test runs continued to cause the oil to turn cloudy. During this time no additional AFW system testing was performed.

A sample of the C pump governor oil was sent to a laboratory for analysis. Test results were not immediately available. A small sample of oil was centrifuged in the chemistry laboratory on site and no water separated from the oil. It was determined that the white cloudy oil was caused by air entrainment and not by water. The C pump oil remained cloudy and was returned to service September 7, 1987 at 9:42 p.m. based on a verbal vendor technical representative's statement that a small amount of air entrainment did not constitute an operability concern. The B pump remained aligned to train 2 and was considered operable.

Post modification testing was reinitiated at 1:40am on September 8, 1987. All required post modification testing was completed at 7:00 a.m. on September 8. Required surveillance testing was begun at 2:00 p.m. on September 8 but was not completed until September 10 at 5:36am. The test completion delay was caused by a train 2 flow control valve oscillation which resulted in nitrogen system usage in excess of the design rate. Additionally, a pressure regulator was leaking. This also contributed to the excessive train 2 nitrogen usage and the failure to meet the acceptance criteria of surveillance procedure 3-OSP-075.7, AFW System Train 2 Backup Nitrogen Test.

On September 10, 1987, a telephone report from the oil analysis laboratory on the C AFW pump governor oil confirmed that 4% water contamination was present. The C AFW pump was declared out of service at 12:45 p.m. The B pump remained in service. Its cloudy appearance was slight and appeared improved. The C pump received special governor oil flushes under Temporary Procedure 382. Additionally the governor oil cooler, a water cooled heat exchanger was replaced. Testing did not identify any coolant leaks. Repair efforts are documented in PWO 6194. The C Pump was returned to service on September 20, 1987.

Between September 7-10 the licensee was questioned about the operability of the B AFW pump. The B pump, like the C pump, was initially thought to suffer from air entrainment in the oil. When the C pump laboratory oil results identified 4% water in its oil, the licensee did not immediately sample the B pump oil which displayed a similar though less severe

symptom. The B pump oil was eventually sampled on September 14, 1987. Test results, provided on September 15 showed that there was no water in the oil.

NRC inspector concerns are summarized below:

1) Post maintenance AFW system testing was not implemented expeditiously. Additionally, a portion of post maintenance test POP 0800.116 could have been performed in modes 4 or 5 but implementation was delayed until mode 2. This caused less information to be known about system operability when the plant passed through mode 3 and entered mode 2.

2) The C AFW pump was removed from service without a review of any objective evidence of operability. A 4 hour time period allowed for performing an operability verification was not used to perform those surveillances or post modification tests designed to demonstrate proper system function.

3) The first root cause analysis for the cloudy oil condition in the C AFW pump was erroneous and was based on centrifuge results which were of unknown accuracy. Consequently, the pump was returned to service while moisture continued to contaminate the oil.

4) Indications of water contamination of the B AFW pump were not promptly evaluated. A sample of the oil was not analyzed for water content until 7 days following the observed condition which was 4 days following the verification that the similar condition on the C pump was caused by moisture intrusion.

No violations or deviations were identified within the areas inspected.

8. Maintenance Observations (62703/62700)

Station maintenance activities of 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; that approvals were obtained prior to initiating work; that activities were accomplished using approved procedures and were inspected as applicable; that procedures used were adequate to control the activity; that troubleshooting activities were controlled and repair records accurately reflected the maintenance performed; that functional testing and/or calibrations were performed prior to returning components or systems to service; that QC records were maintained; that activities were accomplished by qualified personnel; that parts and materials used were properly certified; that radiological controls were properly implemented;

that QC hold points were established and observed where required; that fire prevention controls were implemented; that outside contractor force activities were controlled in accordance with the approved QA program; and that housekeeping was actively pursued.

a. Repairs to Flow Transmitter FT-475

On September 12, 1987, Unit 3 steam flow transmitter FT-475 was observed to be indicating zero steam flow while the reactor was operating at approximately 25% power. The transmitter, which indicates steam flow from the 3A steam generator was declared out of service and maintenance troubleshooting was begun. It was subsequently determined that the flow transmitter was incorrectly wired such that it could not perform as designed.

The I&C department and the Startup Department had both performed calibration checks on FT-475 during the second week of May 1987, subsequent to the performance of electrical cable splice replacements inside the containment building. The flow transmitter was observed to perform correctly. In an attempt to identify when the wiring error was made, the licensee reviewed previously closed work orders. It was determined that Nonconformance Report (NCR) C-655-87 was issued on May 24, 1987, requiring the replacement of an additional electrical wire splice in the cable for FT-475. The splice was replaced on May 29, 1987 and the NCR was closed on June 11, 1987.

The NCR specified that the splice be repaired in accordance with procedure 5610-E-1593/87-093, Revision 0, entitled Acceptance Criteria and Installation Details for Raychem Splices. Section 8.0 of the procedure specifies, in part, that post maintenance functional tests shall be performed on instrument loops receiving wire splice repairs or replacements. Although previous splice replacements were followed by the required post maintenance tests, the splice installed on May 29, 1987 was not followed by the required test. Since the Unit 3 reactor was in a refueling outage, the wiring discrepancy could not be observed through observation of the installed steam flow meters. The discrepancy could not be visually identified until sufficient steam flow existed subsequent to placing the Unit 3 turbine generator on line on September 12, 1987.

The failure to implement a required post maintenance test is contrary to the requirements of 10 CFR 50, Appendix B, Criterion V and Florida Power and Light Topical Quality Assurance Report 5.0, Revision 6, and is one example of violation 250/87-39-02 which is applicable to Unit 3 only.

The licensee is reviewing corrective actions to prevent recurrence of similar problems. Under consideration is the use of a form which requires that the specific type of post maintenance test be specified. This will prompt the review of post maintenance needs and

minimize the potential for failing to implement the testing portion of a larger repair procedure. Corrective actions will be more completely reviewed subsequent to the licensee's response to the Notice of Violation.

b. Auxiliary Feedwater Nitrogen Check Valve Replacement

Inspection Report 250,251/87-35 documented an NRC concern that a check valve with an incorrect seating design pressure was installed in the Unit 3 Train 1 Auxiliary Feedwater (AFW) Nitrogen Backup System during the implementation of Plant Change Modification (PCM) 85-175. This concern was discussed with the licensee and documented as IFI 250/87-35-06.

Subsequent review of the issue revealed that check valve 293 was required to have a 0.33 psi differential seating pressure in accordance with the specifications of drawing 5610-J-558, revision 2, contained in the PCM package. A check valve with a 10 psi differential seating pressure was found to be installed. The valve was subsequently replaced by members of the I&C Department under Plant Work Order (PWO) 308766.

The failure to install a nitrogen check valve of the type required by drawing 5610-J-558, revision 2 is a violation of 10 CFR 50, Appendix B, Criterion V, which requires that activities affecting quality be accomplished in accordance with appropriate instructions, procedures and drawings. This violation is a second example of violation 250/87-39-02 which is applicable to Unit 3 only.

Licensee letter PTN-JPM-87-900, dated September 16, 1987 summarizes the licensee's review of the discrepancy. It was noted that both the 10 psi and 0.33 psi rated check valves are externally identical except for a 1/8 x 1/2 inch adhesive label. The small size of the label made verification of the pressure rating difficult. The system had received a detailed post installation physical inspection, as evidenced by the number and type of discrepancies documented in Deficiency Report DR-655-87. Some spare 10 psi rated valves were removed from the warehouse although they were not needed in the field. This contributed to the potential for erroneous installation.

The licensee's preliminary conclusion is that the discrepancy occurred through inadvertent human error. As a corrective action, the licensee plans to remove and inspect each check valve in the AFW system. Replacement valves will have either permanent metal bands etched with the valve pressure identification or will be etched by plant personnel in a readily identifiable location. This effort is necessary because some adhesive labels have fallen off the valves subsequent to installation, making in place verification of pressure rating impossible.

c. Changes to Construction Process Sheets

In Inspection Report 250,251/87-35, an NRC concern was identified relative to the control of changes to construction process sheets (URI 250,251/87-35-04). Specifically, it appeared that Section 6.6.9 of Administrative Site Procedure (ASP) 2, entitled Preparation of Site Procedures/Process Sheets, Revision 4, conflicted with the requirements of TS 6.5.1.6.d in that it allowed modifications to be performed to the facility which had not been reviewed as proposals by the Plant Nuclear Safety Committee (PNSC). This concern was identified on August 24, 1987.

A similar concern had been identified by a licensee Quality Assurance Inspector and Corrective Action Request (CAR) 87-026 was issued on June 19, 1987. On July 9, 1987 the Project Site Manager responded to the CAR as documented in letter PTN-JPM-87-834, specifying that appropriate changes would be made to ASP-2 by September 15, 1987 to correct the problem. An Additional Commitment was made in letter PTN-JPM-87-851, dated August 3, 1987 to implement, effective immediately, the use of a Change of Intent Checklist to determine whether a proposed process sheet change could be made without PNSC review. The corrective actions were determined to be acceptable by the Quality Assurance Superintendent on August 19, 1987.

Additional NRC review of this issue has confirmed that ASP-2 was deficient in that it allowed changes to be made to process sheets which constituted facility modifications without prior PNSC approval. The failure to control process sheet changes such that TS 6.5.1.6.d requirements were not implemented would normally result in the issuance of a Notice of Violation by the NRC. However, this discrepancy was clearly identified by the licensee prior to being independently identified by an NRC inspector. Additionally, the licensee's corrective actions were reviewed and evaluated to be appropriate and timely. The NRC encourages and supports licensee initiatives for self-identification and correction of problems. Consequently, this issue has been reviewed against the mitigation criteria specified in 10 CFR 2, Appendix C, Section V and a determination has been made that, due to the licensee's actions, no Notice of Violation will be issued, and URI 250,251/87-35-04 is closed.

d. Safeguards Circuitry Fuse Discrepancies

On September 13, 1987, Unit 3 experienced an automatic reactor trip and safety injection, as discussed in paragraph 13. A contributing factor to the event was that a fuse in the 3C main steam line flow transmitter 3-FT-494 circuitry failed. further investigation revealed that the fuse was slightly undersized. Procedure MP 0707.17 required an 0.375A fuse, the fuse that failed was an 0.25A fuse. The licensee initiated an inspection of all the fuses in the Unit 3 and

Unit 4 ESF protection racks. Typical as found discrepancies observed were fast blow fuses installed instead of slow blow fuses and amperage rating discrepancies similar in magnitude to that of 3-FT-494. 5.0A fuses were found in several circuits which actually required fused rated at 0.375A or less.

The licensee believes that a controlled activity such as a special test, PCM or PWO was performed in the 1976 - 1980 time frame changing all the ESF protection circuitry from 5.0A fuses to the specific fuses required by current procedures. The licensee is currently reviewing all quality records from this period to verify this belief. Westinghouse has performed an initial safety evaluation as to the impact of the 5.0A fuses in lieu of the required smaller fuses on the operability of the ESF protection circuits. W concluded that the fuses should be replaced immediately but that the 5.0A fuses did not affect plant safety. The W position is documented in correspondence SECL-87-479. All circuits were restored to the as designed condition specified by procedures:

MP 0707.16,	Hagan Summators repair and calibration, revision dated August 16,1985.
MP 0707.17,	Hagan single and dual comparators , repairs and calibration, revision dated November 6, 1985
MP 0707.18,	Hagan M/A controller calibration and repair, revision dated April 3, 1986.
MP 0707.19,	Hagan M/A control station calibration and repair, revision dated August 16, 1985.
MP 0707.20,	Hagan Isolators repairs and calibration, revision dated August 16, 1985.
MP 0707.25,	Hagan 40 and 45 volt loop power supplies repair and calculation, revision dated November 13, 1985.
MP 0733,	Hagan's MV/I amplifier repairs and calibration, revision dated February 3, 1987.

Engineering is evaluating the as found condition of all circuitry impacted by this fuse inspection. NCRs 87-0216 and 87-0221 were generated for Unit 3. The Unit 4 fuse inspection is still in progress, when completed all discrepancies will be evaluated by engineering via an NCR. Failure to maintain control of fuses in accordance with approved procedures is a violation (250, 251/87-39-01).

9. 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
Auxiliary Feedwater
Control Room Vertical Panels and Safeguards Racks
Intake Cooling Water Structure
4160 Volt Buses and 480 Volt Load and Motor Control Centers
Component Cooling Water
Main Steam Isolation Valve Control
Unit 4 Feedwater
Boric Acid Storage Tanks

a. Misinterpretation of a Technical Specification LCO

On August 21, 1987, while reviewing nuclear chemistry summary sheets, a Senior Reactor Operator (SRO) noticed that the boron concentration of the C Boric Acid Storage Tank (BAST) was listed as greater than that allowed by TS 3.6.b.3. The recorded concentration was 23,100 parts per million (ppm) and the allowed range is 20,000 to 22,500 ppm. An additional sample was taken and the boron concentration was measured as 22,800 ppm. The tank, which was the sole supply of boric acid to the Unit 4 boric acid transfer pumps, was declared out of service.

TS 3.6.b requires that two boric acid transfer pumps be operable and that at least 3,080 gallons of boron solution be contained in the boric acid storage tanks. System piping shall be operable to the extent of establishing one flow path from the boric acid tanks to the reactor coolant system. Although one of the two boric acid transfer pumps may be out of service for 24 hours, no allowable out of service time exists subsequent to the supply tank becoming unavailable.

The SRO misinterpreted this requirement by failing to realize that the out of service BAST required the plant to enter TS 3.0.1 which specifies, in part, that when an LCO is not met within one hour action shall be initiated to place the unit in at least hot standby (mode 3) within the next 6 hours. The SRO consulted an interim book of proposed TS kept in the control room for reference purposes pending approval for use by the NRC. The interim book permits licensee personnel to become familiar with nonbinding specifications prior to their incorporation in the license.

The SRO correctly determined that the loss of the C BAST required the plant to enter a 72 hour LCO in accordance with the interim specifications. Therefore, no plan was developed for a plant shutdown because ample out of service time was thought to exist. The failure to enter TS 3.0.1, in accordance with the approved TS, was not identified by other supervisory personnel contacted by the SRO in the course of restoring the tank to service. The discrepancy was brought to management's attention by NRC inspectors.

Although the SRO believed that 72 hours existed in which to establish a boron supply of the correct concentration, he chose to pursue corrective action immediately. An alternate alignment was established and tested within six hours of the discovery of the problem. Consequently, the shutdown time limit specified in TS 3.0.1 was not exceeded.

Some licensee managers were not aware that the interim TS were less restrictive (in some areas) than the approved TS. Additionally, the potential existed for licensed personnel to consult the interim specifications without fully reviewing the approved TS requirements. Therefore, corrective actions designed to assure that all personnel follow the approved TS have been implemented. Training summaries specify that the approved TS must be followed. The interim specifications are informational in nature and do not constitute regulatory requirements. Licensee management prefers to maintain the interim book in the control room to facilitate awareness of the pending requirements.

b. Chemistry Sampling Concerns

A review was conducted of the chemistry results taken on August 21, 1987 for the C BAST. An out of specification sample (23,100 ppm) was obtained at approximately 1:55 am. Chemistry Department guidance did not require that the result be reported to the Operations Department until after a confirmatory sample verified the unacceptable condition. A confirmatory sample was taken at approximately 3:55 a.m.

This sample indicated that the BAST boron concentration was within specification (22,100 ppm). Consequently, no report of a possible problem was made to Control Room personnel. Twelve hours later only the out of specification sample result was sent to the control room on the chemistry summary sheet. This result was identified as unacceptable. The SRO was informed of the subsequent confirmatory sample result. However, as a conservative measure, at 4:30 pm, the SRO requested that an additional confirmatory sample be taken. The sample result was received at 7:40 pm and it indicated that the C BAST was out of specification (22,800 ppm). The tank was declared out of service as discussed in paragraph 9.b.

Subsequent licensee review of this issue, documented in correspondence PTN-NC-87-126, dated August 26, 1987, resulted in a licensee determination that the first sample results were erroneous. This conclusion was supported by the fact that no additions had been made to the C BAST since it was last sampled at 22,200 ppm. Additionally, a trainee, under the direct supervision of a qualified Chemistry Technician, had obtained and analyzed the sample. The qualified Chemist performed the confirmatory sample analysis and obtained a satisfactory result.

The licensee considers the third boron sample result (22,800 ppm) to be suspect because adequate tank mixing may not have occurred during recirculation. Additional backup samples were not taken prior to diluting the tank. The tank dilution was expedited because of the TS operability concern.

The inability of Chemistry Department Technicians to accurately sample the C BAST on 2 out 3 occasions is an NRC concern. A review of Chemistry Department procedures revealed that lack of specific administrative procedures may have contributed to the poor sample results. No written instructions existed specifying the required recirculation time for the BAST. A one hour recirculation time was used as a non-mandatory rule of thumb. No basis existed which validated or justified this time frame. Additionally, the Chemistry Department had not developed administrative procedures covering department policies.

The licensee issued correspondence PTN-NC-87-124, dated August 25, 1987, addressing BAST sampling and analysis. The recirculation time was mandated as 60 minutes in duration. The sample results were required to be analyzed immediately. During the analysis the BAST must remain in recirculation to facilitate a timely re-analysis should the initial results be out of specification. BAST concentrations found to be out of specification are required to be reported to control room personnel immediately, as well as to the Chemistry Supervisor. Each out of specification result is required to be followed immediately by confirmatory samples.

Although not specified in PTN-NC-87-124, discussions with the Chemistry Supervisor revealed that the corrective actions apply to all chemistry samples performed to meet TS surveillance requirements. This generic interpretation appears appropriate.

Additionally, the licensee is pursuing the upgrade of chemistry procedures. Five additional departmental administrative procedures will be developed by early 1988. These include procedures 0-ADM-650, Chemistry Department Policy Procedures, and 0-ADM-653, Reporting Abnormal Chemistry Values.

10. Engineered Safety Features Walkdown (71710)

The inspectors performed an inspection designed to verify the operability of the Main Steam Isolation Valve Nitrogen system by performing a complete walkdown of all accessible equipment. The following criteria were used, as appropriate, during the walkdown:

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

b. Equipment conditions were satisfactory and items that might degrade performance were identified and evaluated (e.g. hangers and supports were operable, housekeeping was adequate).

c. Instrumentation was properly valved in and functioning and that calibration dates were not exceeded.

d. Valves were in proper position, breaker alignment was correct, power was available, and valves were locked/lockwired as required.

e. Local and remote position indication was compared and remote instrumentation was functional.

f. Breakers and instrumentation cabinets were inspected to verify that they were free of damage and interference.

Minor discrepancies were identified in procedure 3-OP-065.2, AFW and MSIV Backup Nitrogen Gas Supply System, revision dated June 18, 1987. However, the items had been previously identified and corrected by licensee personnel by use of a temporary procedure change. The inspectors reviewed on the spot change (OTSC) 5321 to ensure that discrepancies in 3-OP-065.2 were identified and properly corrected. The PNSC subsequently approved the OTSC corrections and a revised procedure was issued on September 23, 1987. Drawing 5610-M-339, sheet 2, revision 2 was used during the walkdown. No drawing discrepancies were identified.

The inspectors noted that a wrench which is permanently maintained at each of the MSIV nitrogen stations to facilitate nitrogen bottle replacement was missing from the 3A and 3B MSIV station. The system engineer committed to replacing the wrenches.

11. Summary of International Atomic Energy Agency (IAEA) Activities

In fulfillment of the Safeguards Agreement between the United States and the IAEA, the IAEA selected, on July 19, 1985, Turkey Point Unit 4 for participation in its international safeguards inspection program. A major portion of this program requires the continuous surveillance of the fuel inventory through camera monitoring and seal wire placement. The surveillance program ensures that the fuel inventory does not change between physical audits. On April 10, 1987, the Commission issued Amendment 117 to the Facility Operating Licence No. DPR-41 for the Turkey Point Plant, Unit 4. The amendment adds License Condition 3.J regarding implementation of the IAEA Safeguards program for Unit 4.

The NRC inspectors verified, during routine tours of the Unit 4 Spent Fuel Pool (SFP) and the accessible portions of the containment building, that seal wires were in place and intact and that surveillance cameras were operable. Seal wires are placed by IAEA inspectors on the containment equipment access hatch and the reactor vessel head seismic restraints, if accessible. Only the seal wires on the equipment hatch can be observed from outside the containment building. The containment building is not normally entered during power operation. Two surveillance cameras are installed in the Unit 4 SFP. The SFP area is always accessible through locked and alarmed doors.

IAEA inspectors are scheduled to visit the site for routine equipment checks on September 24, 1987.

12. Plant Startup from Refueling (71711)

An inspection was conducted to ensure that selected systems which were modified, disturbed or tested during the Unit 3 outage were returned to service prior to startup. Major modifications were made to the AFW, MSIV, and Containment Spray (CS) systems. Inspection report 250, 251/87-35, paragraph 14, discusses, in detail the implementation of the licensee design change and modification process, and specifically reviewed the AFW and CS PCMs.

a. AFW and MSIV Backup Nitrogen System

The inspectors reviewed procedure 3-OP-065.2, AFW and MSIV Backup Nitrogen Gas Supply, revision dated June 18,1987, to ensure that the operating diagrams referenced and the body of the procedure had incorporated the PCM. The inspectors noted two component number and one position discrepancies. The procedure was performed on September 4, 1987, for the MSIV system, the discrepancies noted by the inspectors were identified and corrected in the procedure by use of on the spot change (OTSC) 5321. The procedure was performed on September 8, 1987, for the AFW system. The inspectors have performed several system walkdowns to ensure proper valve line up, nitrogen pressure readings and seismic supports as well as general system condition, labeling and housecleaning.

b. Containment Spray

A flow restricting orifice was installed at the discharge of each Unit 3 and Unit 4 containment spray pumps. The inspectors reviewed drawing 5610-T-E-4510, Revision 71, to ensure that the orifice modification had been incorporated. An ESF walkdown was performed on the CS system on August 24, 1987, and is documented in inspection report 250, 251/87-35, paragraph 11. The following procedures for Units 3 and 4 were verified to have been updated to reflect changes in system operation. The changes incorporated a new net positive suction head (NPSH) requirement to place one pump in pull-to-lock after the low level alarm is reached.

EOP-E-1	Loss of Reactor or Secondary Coolant
EOP-ECA-2.1	Uncontrolled Depressurization of All Steam Generators
EOP-ECA-3.1	Steam Generator Tube Rupture - Subcooled Recovery Desired
EOP-ECA-0.0	Loss of All AC Power
EOP-FR-Z.1	Response to High Containment Pressure
EOP-ECA-0.2	Loss of all AC Power Recovery with Safety Injection Required

13. 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. The licensee plans to issue LERs on each event within 30 days following the date of occurrence.

On September 13, 1987, Unit 3 experienced and automatic reactor trip and safety injection (SI). The trip occurred during performance of OP-8001.4, revision dated October 14, 1986, Turbine Generator - Overspeed Trip Test. Unit 3 was initially at approximately 30% power with the A Steam Generator (SG) Flow Transmitter FT-475 out of service (paragraph 9). In order to perform the test the generator must be taken off line and turbine load and reactor power must be reduced to below the P-10 set point. The generator was taken off line and turbine load reduced to below the P-10 setpoint, but reactor power was above the P-10 setpoint. Operation personnel inserted control rods and borated to reduce power. In doing so average reactor coolant temperature (Tave) decreased to 530F, below the Low Tave Safety Injection (SI) setpoint of 543F. With Tave recovering, but still below 543F, a decision was made to continue with the turbine overspeed test. By taking manual control of the turbine governor, the turbine speed was increased from the normal operating speed of 1800 rpm to the overspeed trip setpoint of 1975 rpm. The turbine tripped as designed and the turbine stop valve slammed closed. The rapid closure of the turbine stop valves caused steam line vibrations. The C SG flow transmitter FT-494 instrument tubing sensed the vibration and FT-494 began spiking repeatedly. The rapid cycling of the FT-494 circuitry caused a fuse to fail, causing FT-494 to fail high, generating a high steam flow signal. With the A SG Flow transmitter FT-475 already out of service and Tave still below 543F the logic for automatic safety injection on high steam flow on 2 out of 3 generators, in conjunction with 2 out of 3 low Tave was completed. Safety injection actuated, the reactor tripped and a phase A isolation occurred. All equipment performed as designed and the unit was stabilized. A review of this event resulted in violation 250, 251/87-39-01 and example one of violation 250/87-39-02, as described in paragraph 8.

On September 15, 1987, Unit 4 experienced and automatic containment and control room ventilation isolation. Just prior to the event the sampling paper on Process Radiation Monitor System (PRMS) detector R-11 had been found to be gathering prior to the pick up spool. Shortly after the paper was cleared R-11 tripped. The cause of the R-11 trip was that the backing up of the sampling paper created an accumulation of particles that were picked up by the detector when the paper was cleared. The containment atmosphere was sampled and an RCS leakage rate calculation was performed as precautionary measures. No abnormal conditions were observed.

On September 19, 1987, Unit 3 experienced an RCS leakage rate in excess of 10 gallons per minute (gpm). Pressurizer spray control valve 3-PCV-455B did not respond when operations personnel attempted to modulate it. In order to facilitate troubleshooting on 3-PCV-455B, operations personnel attempted to close the upstream manual isolation valve 3-573. When valve 3-573 was closing the packing began to leak, allowing reactor coolant to spill to the containment sump. The maximum leakage calculated was 11.7 gpm. Valve 3-573 was backseated and the leakage rate was immediately reduced to approximately 1-2 gpm. Subsequent leakage was measured at 0.4 gpm. The licensee plans to continue opeation with valve 3-573 backseated until an outage of sufficient duration to perform repairs. The valve is being administatively controlled in the open position on its backseat. Valve 3-PCV-455B was repaired and returned to service.