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

Report Nos.: 50-250/87-51 and 50-251/87-51

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: November 23 - December 28, 1987

Inspectors: _____ for 2/2/88
D. R. Brewer, Senior Resident Inspector Date Signed

_____ for 2/2/88
T. F. McElhinney, Resident Inspector Date Signed

_____ for 2/2/88
G. A. Schnebli, Resident Inspector Date Signed

Approved by: _____ 2/3/88
R. V. Crlenjak, Section Chief Date Signed
Division of Reactor Projects

SUMMARY

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

Results: One violation with two examples for failure to meet the requirements of Technical Specification 6.8.1 was identified (250, 251/87-51-01).

REPORT DETAILS

1. Persons Contacted

Licensee Employees

J. S. Odom, Vice President
*C. J. Baker, Plant Manager - Nuclear
*L. W. Pearce, Operations Superintendent
*T. A. Finn, Training Supervisor
J. D. Webb, Operations - Maintenance Coordinator
*W. R. Williams, Assistant Superintendent Planned Maintenance
D. Tomasewski, Instrument and Control (I&C) Department Supervisor
J. C. Strong, Electrical Department Supervisor
*L. W. Bladow, Quality Assurance (QA) Superintendent
E. F. Hayes, Quality Control (QC) Supervisor
*R. J. Earl, 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
*G. Solomon, Regulation and Compliance Engineer
J. Donis, Engineering Department Supervisor
D. E. Meils, Chemistry Supervisor

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

*Attended exit interview on December 31, 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 December 31, 1987. The areas requiring management attention were reviewed. No proprietary information was provided to the inspectors during the reporting period.

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 were identified in this report.

4. 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 250,251/86-39-05. The licensee failed to complete fire protection modifications on several Unit 4 raceways and several common penetration seals by the scheduled completion date. On October 10, 1986, the licensee submitted a letter (L-86-410), which reported the items which would not be completed by the required dates of September 30 and October 1, 1986. On November 3, 1986, the licensee provided an update letter (L-86-451) on the outstanding Unit 4 electrical conduit protection and common unit penetration seal work. All remaining open items of this URI were completed November 12, 1986. This item is closed.

(Closed) URI 250,251/86-33-07. Adequacy of the fuse control program. This unresolved item will be administratively closed and corrective action on violation 250,251/87-39-01, for failure to maintain control of fuses in accordance with approved procedures, will be tracked. This item is closed.

(Closed) IFI 250/84-23-19 and 251/84-24-19. Provide response addressing what reviews are being done to assure that design deficiencies have not occurred in the routing of power to miscellaneous relay rack equipment which may require safety-related power sources. The licensee in letter (L-85-321), dated August 20, 1985, stated that the design deficiency in question was identified during an engineering review of the auxiliary power upgrade ("C-Bus" modification) at Turkey Point. The licensee has also performed other reviews, such as the System Operability Review Program (SORP) and the Appendix R Safe Shutdown circuit review, that included the consideration of power to selected safety-related equipment. This item is closed.

(Closed) IFI 251/84-23-05. Investigate the Spent Fuel Pit Ventilation System. On January 29, 1985, Bechtel Power Corporation provided FPL a response on the spent fuel pit ventilation damper location. Bechtel recommended that a damper be added at the discharge of Unit 4's exhaust fan for ALARA considerations. Additionally, Bechtel states that the existing condition does not constitute an unreviewed safety question or a potential hazard to the public because the existing isolation damper provides a physical boundary to the Spent Fuel Pit air space. This item is closed.

(Closed) IFI 250,251/84-18-06. A number of discrepancies in the Spent Fuel Pool area were identified including physical and procedural discrepancies. In an inter-office correspondence, dated January 14, 1985, the licensee addressed each item. A number of procedures were developed to govern the operation of the Spent Fuel Pit, these procedures include, 3/4-OP-033 - Spent Fuel Pit Cooling System, 3/4-OP-038.1 - Preparation for Refueling Activities, 3/4-OSP-034.1 - Spent Fuel Pit Inlet and Exhaust Damper Operability Test, OP-10400 - Spent Fuel Pit Ventilation Exhaust System - Air Flow Test, 3/4-ONOP-033.3 - Accidents Involving New or

Spent fuel and 3/4-ONOP-067. Inadvertent Release of Radioactive Gas. Additionally, a number of system modifications were performed. They include the following: installed new level indications, replacement of Spent Fuel Pit Overhead Crane Access Door Seal, replaced the air driven transfer cart with an electrically driven transfer cart, and the completion of a modification on Fuel Transfer System for crane dual cable. This item is closed.

(Closed) IFI 250,251/86-30-03. Review the engineering evaluation of noise in the steam flow and pressure signals. The licensee developed a Design Equivalent Engineering Package to redesign the supports for the Main Steam Flow Transmitter Lines (DEEP 87-328), dated October 6, 1987. The packages safety evaluation states that this modification will reduce the tubing vibration problem which appears to be causing high spiking of the reactor protection system. This item is closed.

(Closed) IFI 250/84-22-04. Items (components, system, etc.) of significance which are not included in the Technical Specifications (TS) but are in the the standard TS will be identified and maintained on a priority basis. The licensee has submitted to the NRC its standard TS for approval. Additionally, procedure O-ADM-701, Plant Work Order Preparation, continues prioritization methods for all plant equipment including time limits for starting the work. This item is closed.

5. Onsite Followup and In-Office Review 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 250/86-035. Technical Specification (TS) exceeded when three charging pumps were inoperable. On September 25, 1986, while Unit 3 was at 100% power, with the 3A and 3B charging pumps out of service for maintenance, the 3C charging pump was isolated during the testing of the 3B charging pump. The 3B charging pump failed its acceptance criteria, resulting in the 24 hour Limiting Condition of Operation being exceeded. The cause of the failure was a leaking relief valve on 3C charging pump. The relief valve for 3C charging pump and the suction valves on 3B charging pump were rebuilt. Both pumps were satisfactorily tested and returned to service. This item is closed.

(Closed) LER 250/86-036. Both Emergency Diesel Generators (EDG) Out of Service. The 'B' EDG was out of service for instrument calibrations in preparation for performing its eight hour test run. The 'A' EDG was started to verify its operability and, when completing its test run, it would not stop. An Event Response Team was formed to determine the cause

of the 'A' EDG failure. The cause was determined to be the governor solenoid was out of adjustment. The governor solenoid was adjusted and the 'A' EDG was declared back in service on November 7, 1986. Preventative maintenance procedure O-PMI-023.1, Emergency Diesel Generator Instrumentation Calibration, was revised to include the governor solenoid adjustment. This item is closed.

(Closed) LER 250/86-033. Potential 4160 Volt Bus and Emergency Diesel Generator Lockout. On August 15, 1986, the licensee discovered two types of devices whose postulated failure could adversely affect the ability of safety related equipment to perform their intended safety function. An auxiliary overcurrent relay (174X/TDDO) located on each 4160 volt tie breaker could fail resulting in the opening of the supply breakers, including the EDG breakers. Additionally, failure of auxiliary switches of the EDG supply breaker switch contacts (1-1T or 3-3T) could provide a false input to the EDG failure circuit in the load sequencer, which would block loading of the battery chargers. The licensee performed a Temporary System Alteration to disable the 174/TDDO relays. A training brief was written to alert the operators of the potential failures of the relays. Emergency Operating Procedures 3/4-EOP-E-0, Reactor Trip or Safety Injection, and EOP-ES-0.1, Reactor Trip Response, were revised to ensure that the battery chargers are energized within 30 minutes. This item is closed.

(Closed) LER 250/86-40. Single Failure in the Control Room Ventilation System (CRVS) May Result in Loss of Control Room Ventilation System. The licensee determined that if a loss of power to the motor control center (MCC) 3A occurred and the transfer switch sticks between its two position (MCC 3A and MCC D), no control circuit power will be available and the CRVS air conditioning compressors and air handlers would be disabled. The licensee has taken the following corrective action; O-OSP-200.1, Schedule of Plant Checks and Surveillances, has been revised to verify operability of the transfer switch weekly; quick-connect jumpers have been installed around the control circuitry for the air conditioner compressor units not connected during normal plant operation; Training Brief 186 was issued; the transfer switch cover-plate was modified to increase clearance; and finally, PC/M 87-243, Transfer Switch Upgrade for OP-312A and DP-412A was developed to provide a permanent fix and the schedule for implementation is being tracked in the integrated schedules. This item is closed.

6. Followup of 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 250,251/86-39-01. Inadequate Procedure, two examples. Temporary Operating Procedure (TOP) 233, Functional Test of PC/M 84-209 - Power Mismatch Modification and PC/M 84-211 - Turbine Runback Modification was inadequate in that performance of procedure steps 8.5 and 8.29 resulted in an unplanned actuation of the reactor protection system. The second example was that no plant procedure existed specifying the method by which fire doors shall be controlled. The licensee revised procedure TOP-233 and successfully performed the functional test. The licensee developed procedures, ASP-31, Breaching of Fire Protection System and Fire Rated Assemblies and O-SMM-016.6, Fire Door Inspection, to help control fire doors. Additionally, fire doors have been painted red and signs requiring closure have been placed on them. This item is closed.

(Closed) Violation 251/86-33-05. Failure to follow AP-0103.4, In-Plant Equipment Clearance Order and AP-0103.32, Reactor Cold Shutdown Conditions. The individual involved in failure to follow procedure AP-0103.4 was counseled and shift briefing, to reemphasize full compliance with the equipment clearance procedure, was given to all operators. Procedure AP-0103.32 was revised to clarify the requirements for Residual Heat Removal (RHR) loop operability. This item is closed.

(Closed) Violation 250,251/86-33-04. Drawing 5610-T-D-18B, Revision 1, Steam Break Protection, was not accurate. Drawing 510-T-D-18B was revised on July 22, 1986, to correct the inaccuracies. Additionally, under the Performance Enhancement Program the licensee is updating all drawings under the select system review. This action is tracked by the Integrated Schedule. This item is closed.

(Closed) Violation 250,251/86-33-03. Adequate procedures did not exist to control deluge system valve line-ups, including pressure switch isolation valves. O-OP-016.1, Fire Protection Water System, was revised to incorporate deluge system valve line-ups. Additionally, drawings on the fire protection system were revised to reflect the valve line-ups. This item is closed.

(Closed) Violation 250,251/86-33-02. The licensee failed to take adequate measures to assure that conditions adverse to quality were promptly identified and corrected, in that a Unit 3 pressure transmitter 495 failed for 20 seconds and the actions of ONOP-0208.14, Deviation or Failure of Reactor Protection and Safety-Related Hagan Instrumentation Channels, were not implemented and the licensee failed to evaluate the potential for additional losses of all control room lighting. The licensee completed the following actions; the instrument loop was checked out and the affected transmitter was replaced, additionally, the transmitter was returned to the vender for examination; ONOP-0208.14 was revised to clarify the requirements for channel spiking; the train of control room emergency DC lighting that had failed was repaired by July 22, 1986, and the DC lighting that was out of service for modifications was returned to service July 19, 1986; and finally, shift briefings were held in October 1986. This item is closed.

(Closed) Violation 250/85-24-03. Failure to perform the accumulator boron concentration analysis prior to heating up above 200 degrees F. This violation will be administratively closed and the corrective action tracked under the corrective actions for violation 251/87-35-01, which is a repeat of the above violation. This item is closed.

(Closed) Violation 250, 251/87-44-03. Two examples of failure to meet the requirements of TS 6.8.1. Example one, on numerous occasions shift relief turnovers were not documented in the Reactor Operator's logbook and checklists were not properly and thoroughly completed as required by AP-0103.2, Shift Turnover Requirements. Example two, actions were taken under the guidelines of Procedure O-ADM-207, Operation Instructions in the Event of a Situation not Addressed by Procedures, which were not promptly recorded in the Plant Supervisor's logbook. This violation with two examples is administratively closed and will be opened under violation 250, 251/87-51-01. For further detail of the violation refer to inspection report 250, 251/87-44.

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:

4-OSP-075.1	Auxiliary Feedwater Train 1 Operability Verification.
4-OSP-075.2	Auxiliary Feedwater Train 2 Operability Verification.
4-OSP-075.6	Auxiliary Feedwater Train 1 Backup Nitrogen Test.
4-OSP-075.7	Auxiliary Feedwater Train 2 Backup Nitrogen Test.
0-OSP-075.11	Auxiliary Feedwater Inservice Test.

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. Unit 3 and 4 Condensate Storage Tank (CST) Holddown Nuts

On November 5, 1987, with both units in Mode 5, a member of the Technical Department identified loose nuts on Unit 4 CST anchor bolts. Subsequently, the same condition was found to exist with the Unit 3 CST. Plant Work Order (PWO) WA873091022 was immediately generated to document the problem and questioned CST seismic structural integrity and operability. Non-Conformance Report (NCR) 87-0240 was written on November 12, 1987, to identify the problem to engineering for an as-found evaluation and corrective action. Engineering provided an initial assessment of operability on November 16, 1987, which stated: "The condensate storage tanks are not required to be operable when the units are in Mode 5. Therefore, there is no operability concern. Disposition NCR prior to leaving Mode 5." The disposition contained in the NCR required measurements of the existing condition for later evaluation and the immediate corrective actions required to repair the nonconforming condition. The NCR was also entered on the QC NCR-Open Item Status Report with an action due date of November 28, 1987, and a note to complete prior to Mode 4.

Unit 4 entered Mode 4 at 1315 on November 29, 1987, then Mode 3 at 0430 on November 30, 1987. At 0001 on December 1, 1987, it was discovered that the corrective actions for the NCR had not been accomplished. Engineering was immediately contacted for short term corrective action to determine the CST operable since the unit had left Mode 5 without implementation of the requirements of the first disposition of the NCR. Engineering provided additional corrective actions required to ensure operability until an in depth analysis was conducted to determine operability in the as-found condition. These actions required that all anchor nuts be torqued to a snug tight fit, then each anchor bolt/nut could be worked one at a time to return them to their original design configuration without affecting tank operability.

The corrective actions were completed and the data requested by the initial disposition on the NCR was returned to engineering to allow for later determination of tank operability in the as-found condition. The licensee's Technical Department formally requested the Engineering Department, by memorandum dated December 10, 1987, to expedite the as-found evaluation to ensure compliance with Technical Specification 3.19.1.1 and 3.0.4 were met. In addition, 10 CFR Part 50.72 requires a licensee to notify the NRC if they discover a condition while the reactor is shutdown, that, had it been found while the reactor was in operation, would have resulted in the nuclear power plant being in a condition outside its design basis. The engineering evaluation was completed on December 23, 1987, and determined that both CSTs were operable in their as-found condition.

The inspectors conducted several discussions with responsible licensee personnel in Maintenance, Operations, Engineering and Quality Control to determine the root cause and corrective actions to prevent mode changes without all work being completed. The discussions indicated that this was an isolated occurrence and although the licensee has a program to track NCRs, versus Mode required by, it may not have been sufficient as the problem did occur. When this issue was identified by the licensee, QC immediately modified the format of the NCR-Open Item Status Report to include a separate column for the required Mode versus a note as previously discussed. In addition, all General Operating Procedures (GOPs) which direct Mode Changes, will be modified to include a step that ensures all maintenance, testing, NCRs, etc., are complete and verified prior to the Mode change. The inspectors consider that these changes will prevent recurrence of this issue.

b. Unit 4 Intake Cooling Water Pump (ICW) Troubleshooting

At 1535 on December 18, 1987, with 4A and 4C ICW pumps in service, Unit 4 experienced the loss of the 4C pump. Several attempts to start the 4B pump failed. At that point both the 4B and 4C pumps were declared inoperable and the licensee commenced a unit shutdown in accordance with Technical Specification 3.0.1., although the 4A ICW pump was capable of supplying all loads. The ICW system at Turkey Point consists of 3 pumps that can feed two ICW headers. The 4A ICW pump is powered from the 4A 4160 volt safety-related bus. The 4B and 4C ICW pumps are powered from the 4B 4160 volt safety-related bus. At that time both the A and B emergency diesel generators (EDGs) were operable and capable of supplying reliable onsite emergency power to their respective busses. At this time NRC Region II was contacted regarding the application of discretionary enforcement to allow the licensee 24 hours to return the 4B and 4C ICW pumps to service. This 24 hour extension was granted with the stipulation that upon any evidence of common mode failure the unit would be immediately shutdown.

The current Technical Specifications require all three ICW pumps to be operable and allow only one ICW pump to be inoperable for 24 hours. The specifications do not allow any credit for ICW electrical trains. Additionally, they do not provide an action statement for this condition so a unit shutdown commenced approximately one hour after declaring the 4B and 4C ICW pumps out of service. The Standard Technical Specifications (STS), including the FPL submitted version under the Performance Enhancement Program, would allow 72 hours of operation with one ICW train out of service.

An extensive, coordinate effort was undertaken by the licensee to restore the two inoperable pumps to service. Investigation into the failure of the 4C ICW pump revealed the upper coupling had broken. The material used for this coupling (17-4 pH) had previously been identified by the licensee's engineering department as susceptible to corrosion failure and were being replaced on a per pump basis with couplings made of Nitronic 50 which is much less susceptible to this type of failure. The previous failure which identified this problem was documented in LER 251/87-004. A review of documentation determined all the couplings on the running 4A ICW pump had previously been replaced with the new material. The investigation into the failure of the 4B ICW pump revealed that the bronze bushing at the stuffing box had bound to the shaft. Evaluation into this failure is still ongoing. The upper coupling on the 4C ICW pump was replaced with the improved coupling material. The 4C ICW pump was satisfactorily tested and placed back in service early in the morning of December 19, 1987, thus relieving the requirement for discretionary enforcement. The 4B ICW pump was replaced with a spare pump. This pump was then satisfactorily tested on the afternoon of December 19, 1987, and placed back in service prior to exceeding the current Technical Specifications action statement and the 24 hour extension allowed by the discretionary enforcement.

No violations or deviations were identified within the areas inspected.

9. Engineered Safety Features Walkdown (71710)

The inspectors performed an inspection designed to verify the operability of the Unit 3 and 4 Component Cooling Water System. This was accomplished by performing a complete walkdown of all accessible equipment. The following criteria were used, as appropriate, during this inspection:

a. Systems lineup procedures match plant drawings and as built configuration.

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

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

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

e. Valves in the flow paths are in 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 was compared and remote instrumentation was functional.

g. Major system components are properly labeled.

The inspectors reviewed procedures 3,4-OSP-030.3, entitled Component Cooling Water System Flowpath Verification, revisions dated June 18, 1987 and September 18, 1987, for Units 3 and 4, respectively. The inspectors also reviewed the applicable operating diagram 5610-T-E-4512, sheet 1, revision 53.

The inspectors did not note any adverse conditions during the walkdown. During this past outage the licensee has performed numerous maintenance activities on the Unit 3 and 4 CCW system which includes eddy current testing and plugging of defective tubes in the CCW heat exchangers, and replacement of CCW surge lines that were experiencing external corrosion. The licensee also made two enhancements to the CCW system which includes; the change of corrosion inhibitors in the CCW from chromates to molybdates for environmental considerations and the change in operating procedures which allows the running of one CCW pump instead of two with the plant in Mode 5. This change is designed to decrease flow induced vibrations which could damage the CCW heat exchanger tubes.

No violations or deviations were identified within the areas inspected.

10. Facility Modifications (37701)

The inspectors reviewed the facility modification to install the Reactor Vessel Level Monitoring System (RVLMS), which satisfies the requirements of NUREG-0737, Item II.F.2, Inadequate Core Cooling Instrumentation System (ICCI). The NRC found the licensee's submittals for the design and installation of RVLMS to be acceptable in a letter and attached safety evaluation dated January 28, 1985.

The installation, functional testing, and calibration of the RVLMS portion of the ICCI system were completed by December 16, 1983, for Turkey Point Unit 3 and May 25, 1984, for Unit 4. The plant specific Technical Specifications for the RVLMS was incorporated in Change 155 (Amendment 125 for Unit 3 and Amendment 119 for Unit 4) on July 28, 1987.

The inspectors reviewed Plant Changes and Modifications packages (PC/M) 81-162 and 81-167 which installed the ICCI system instrumentation on Units 3 and 4, respectively. The PC/Ms were implemented in accordance with the licensee's Administrative Procedure 0190.15, including changes

to the original PC/M. The documents received the proper level of review and contained the required safety evaluations. FSAR evaluations were conducted and the FSAR was updated with Revision 4, dated July 1986, to reflect the new system. The following procedures were reviewed to ensure the new surveillance requirements were included:

a. O-PMI-041.3 Incore Thermocouples (Excore Thermocouples) QSPDS Calibration Procedure.

b. 3/4-OSP-204 Accident Monitoring Instrumentation Channel Checks.

c. AP-0190.16 Scheduling and Surveillance of Periodic Tests and Checks Required By Technical Specifications.

NUREG-0737, Item II.F.2 is closed.

No violations or deviations were identified within the areas inspected.

11. 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
Intake Cooling Water Structure
Control Room Vertical Panels and Safeguard Racks
Condensate Storage Tanks
Auxiliary Feedwater
Component Cooling Water
4160 Volt Buses and 480 Volt Load and Motor Control Centers
Units 3 and 4 Main Steam Platform

a. Unit 3 Return to Service

Unit 3 was returned to service on December 22, 1987, following an outage that began on September 25, 1987, in order to repair the 3B Reactor Coolant Pump, spray valve PCV 455B, and seal table leaks. On December 25, 1987, the operators were experiencing difficulties controlling Reactor Coolant System (RCS) pressure due to pressurizer spray valve PCV-455A not fully closing. The operations staff decided to commence unit shutdown in order to repair PCV 455A. The turbine was tripped and the reactor was taken subcritical with all the control rods manually taken to the bottom position. Upon entering the source range operation, Source Range Nuclear Instrument (SRNI) N-31 actuated a high flux level trip and a subcritical reactor trip occurred. Initial investigation by the licensee indicates that N-31 reenergized spontaneously above permissive P-6 during unit startup on December 22, 1987. This spontaneous reenergization of the SRNI is an ongoing industry wide problem. The licensee is awaiting modification from Westinghouse to replace the solid state circuitry with the relay type circuitry in order to preclude this problem from recurring. The Off Normal Operating Procedure (ONOP)-059.3, entitled Nuclear Instrumentation Malfunction, does not address removing a SRNI from service above the P-6 setpoint. Therefore, N-31 was removed on clearance C311083 by removing the fuses. Removing the fuses causes the SRNI to fail safe (i.e., tripped high). When the power level decreased below the P-6 setpoint (10-10 amps on both Intermediate Range Nuclear Instruments) the Source Range High Flux Trip automatically reset, and caused the reactor trip. The licensee is planning to include steps to ONOP 059.3 to ensure that the SRNI is placed in bypass position when taken out of service. The licensee subsequently repaired PCV 455A by adjusting the bench spring pressure from approximately 3-5 psi to 13.5 psi. This new setting is identical to the setting found for PCV 455B which is seating properly. Unit 3 was started up and returned to service on December 27, 1987.

b. Unit 4 Returned to Service

Unit 4 was returned to service on December 4, 1987, following an outage that began on October 12, 1987, due to Hurricane Floyd. The licensee decided to remain shutdown in order to complete repairs on equipment including the 4B RCP, PORV-456, and retorquing of the conoseals, along with modification to the Residual Heat Removal Pumps recirculation lines. On December 13, 1987, with the unit at 100% power, the RCO noticed an unexplained increase in the RCS as he was performing 4-OSP-046.1, entitled RCS Leakrate Determination. The RCO referred to ONOP-2608.2, entitled CVCS Malfunction of Boron Concentration Control System, and secured the primary water. The inventory increase discontinued. The licensee suspected valve 114A, which supplies primary water to the blender, was not fully seated. The valve was cycled and primary water was realigned. No further increase in RCS inventory was observed. The primary water inleakage

was calculated to be approximately 2.3 gpm (135 gallons total) which added an estimated +20 pcm of reactivity to the system. The RCO stepped in rods 8 steps (D-223 to D-215) to compensate for positive reactivity addition. The operators were unable to duplicate the dilution incident. The plant was set back to normal parameters and a second leak rate calculation was performed. The leak rate was at .02 gpm which indicates that the inleakage had discontinued. On that same day, Unit 4's load was reduced to 100 MWE due to the Northeast (NE) intercept valve going closed. Crud had built up in the control oil orifice which caused the intercept valve to close partially. Maintenance personnel made adjustments to the NE intercept valve orifice which opened the valve. The unit was returned to full power later that same day. On December 16, 1987, at 0920 hours the NE intercept valve went to mid-position due to control oil orifice being clogged with crud. While attempting to fully open the valve by adjusting control oil orifice differential pressure (dp) the valve went full open and the #4 control valve went from 80% to 20% open. This caused a swing in load of 80 MWE and the plant was stabilized at 660 MWE. The licensee developed a Temporary Procedure (TP)-417 in order to make adjustments to the control oil for the NE intercept valve without causing severe load swings. The adjustment was made such that the NE intercept valve was fully open with the #4 control valve controlling in the correct position. Later that same day the RCO noticed a loss of approximately 20 MWE due to the #4 control valve going partially closed. The operations superintendent recommended stabilizing the unit at 680 MWE and 93% reactor power until a fix could be implemented. The turbine vendor (Westinghouse) established new adjustments for the smothering and control oil orifices in order to prevent the intercept and control valves from drifting closed. The unit experienced a loss of two ICW pumps on December 18, 1987, which caused the operators to decrease power to 49%. This event is discussed further in paragraphs 8 and 13. The unit was returned to 100% power on December 20, 1987, after the vendor recommended adjustments to the smothering and control oil orifices were performed.

No violations or deviations were identified.

12. 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.

The US/IAEA Safeguards Agreement has been in force since July 31, 1980. The commitments by the U.S. in this treaty, which carries the force of law, are defined in the Code of Federal Regulations, the treaty itself, and the site-specific Facility Attachments. 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. 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.

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 December 9, 1987, with Unit 3 in Mode 5, an automatic containment and control room ventilation isolation occurred. PRMS Channel R-19 was out of service and Plant Work Order (PWO) number 7397 was generated. While I&C department personnel were disassembling PRMS-19 drawer, breaker 3P08-19 tripped. This deenergized PRMS rack number 66 which contains PRMS-12. This causes control room and containment ventilation isolation. A strand of wire was found in PRMS-19 drawer on the power cable, which is believed to be the cause of a ground fault that tripped breaker 3P08-19. The remaining PRMS drawers and panel number 66 were cleaned and the entire rack was placed back in service.

On December 9, 1987, with Unit 3 in Mode 5 and Unit 4 in Mode 1, 100% power, a site visitor alarmed the explosive detector while attempting to enter the plant. The visitor backed out of the explosive detector and produced a .38 caliber bullet from a pocket and gave the bullet to the security guard. The visitor entered the site and the bullet was returned as the visitor exited the site.

On December 15, 1987, with Unit 3 in Mode 5, an automatic control room ventilation isolation occurred. I&C personnel were removing jumpers that were installed during repairs to PRMS 11 and 12, when all the leads were

momentarily removed from the terminals. This deenergized the relay coil which caused the control room ventilation isolation. PRMS 11 and 12 were subsequently returned to service and the control room and containment isolation was reset.

On December 17, 1987, with Unit 3 in Mode 4, an automatic control room ventilation isolation occurred and valves RCV-609, CV-2819, CV-2826 closed. The Reactor Control Operator (RCO) was performing the monthly surveillance test on PRMS-19 and when the high alarm setpoint was initiated, breaker 3P08-19 tripped which deenergized the entire PRMS rack. Deenergization of PRMS 11 and 12 results in the ESF actuation of the control room and containment ventilation. The cause was determined to be an apparent short in the relay circuit. The PRMS rack was subsequently reenergized and the control room/containment ventilation isolation was reset.

On December 17, 1987, with Unit 3 in Mode 4 and Unit 4 in Mode 1, at 93% power, a visitor attempted to enter the plant at the security gate. Upon emptying his pockets prior to entering the explosive/metal detector, security personnel noticed a .357 magnum bullet in his personal belongings. The Security Supervisor denied access into the plant and the visitor departed.

On December 18, 1987, with Unit 4 in Mode 1, at 94% power, the licensee declared an Unusual Event due to the loss of 4B and 4C ICW pumps. The 4C ICW pump coupling sheared during continuous operation and the 4B ICW pump could not be started by the operators. The licensee formed an Event Response Team (ERT) and commenced unit shutdown in accordance with Technical Specification 3.0.1. Upon receiving discretionary enforcement from the NRC, the licensee was able to maintain reactor power around 50% for the 24 hour extension to repair the 4B and 4C ICW pumps. The licensee terminated the Unusual Event at 8:24 p.m., and proceeded with repairs on the pumps. Both pumps were placed back in service on December 19, 1987, prior to exceeding the 24 hour time frame. The unit was returned to full power. This event is discussed further in paragraph 8.b.

On December 18, 1987, with Unit 3 in Mode 3 and Unit 4 in Mode 1, at 53% power, the Emergency Notification System (ENS) telephone became inoperable during followup notification to the NRC operations center regarding termination of the Unit 4 Unusual Event. The notification was completed via commercial telephone. The ENS telephone was repaired and placed back in service on December 19, 1987.

On December 21, 1987, with Unit 3 in Mode 3 and Unit 4 in Mode 1, at 100% power, a contractor vehicle was found to contain a box of .12 gauge shotgun shells while being searched prior to entering the site. Security denied access to the vehicle and the driver.

On December 25, 1987, with Unit 3 in Mode 3, a subcritical reactor trip occurred. The root cause was evaluated and the unit was restarted on December 27, 1987. This event is discussed further in paragraph 11.

11. Temporary Instruction (TI) 2500/26 (25026)

This TI is provided to ensure that fasteners selected by the licensee in response to NRC Bulletin 87-02, entitled Fastener Testing To Determine Conformance With Applicable Material Specifications, are representative of installed fasteners and that suspect fasteners are selected for testing. On December 7, 1987, the inspectors witnessed the sampling process of the fasteners which were selected for the testing. The licensee obtained a printout of Stores Usage in order to facilitate the sampling process. The safety related bolting (purchased QL-1) was split almost evenly between ASME SA 193 and ASME SA 307. A total of eleven safety related fasteners with their associated nuts were selected. The sizes were determined by the actual usage over the past 12 months. This sample consisted for four ASME SA 193, Grade B-7 and seven ASME SA 307, Grade B fasteners. The associated nuts sample consisted of the following: Eight ASME SA 194, Grade 2H; two ASME Section III, Class 2 and one ASME SA 307, Grade B. The non safety-related sample (commercial grade) was based solely on the actual usage in the plant. A total of ten fasteners were selected along with the associated nuts. All of the fasteners selected were Grade 5 steel. The nuts selected did not have any material specification documented except for one which was ASME A 194, Grade 2H. The licensee's sample technique included taking four of each fastener and nut selected. One fastener and its associated nut were tagged and sealed in a plastic bag for shipment to the testing laboratory. The remaining three samples of fasteners and nuts were placed in a plastic bag and were to be locked in a safe. These extra samples were to be kept if the original sample was damaged or lost enroute or at the testing laboratory. During the licensee's sampling process a number of non-safety related (commercial grade) cap screws were identified to have manufacturers marks which were suspect. The marks in question include: KS, J, M, FM and A. The licensee's sample which was selected for testing included screws with KS and M manufacturers markings. Review of the licensee's sampling process indicates that licensee selected a representative sample of fasteners and nuts used at the plant and that the fasteners were properly tagged for shipment to the testing laboratory. The results of the testing are expected to be received by the week of January 4, 1988. This TI will remain open pending the completion of the fastener testing and the review of the licensee's receipt inspection program. The inspectors will also review the maintenance/warehouse procedures for issue and control of safety and non-safety related fasteners.