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

Report Nos.: 50-250/86-10 and 50-251/86-10

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

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

Facility Name: Turkey Point 3 and 4

Inspection Conducted: February 10 - March 10, 1986

Inspectors: _____ 16 APR 86
T. A. Peebles, Senior Resident Inspector Date Signed

_____ 16 APR 86
D. R. Brewer, Resident Inspector Date Signed

_____ 16 APR 86
S. Guenther, Project Engineer Date Signed

Approved by: _____ 16 APR 86
Stephen A. Elrod, Section Chief Date Signed
Division of Reactor Projects

SUMMARY

Scope: This routine, unannounced inspection entailed 221 direct inspection hours at the site, including 36 hours of backshift inspection, in the areas of licensee action on previous inspection findings, annual and monthly surveillance, maintenance observations and reviews, operational safety, engineered safety features walkdown, independent inspection, and plant events.

Results: Violation - Failure to meet the requirements of Technical Specification (TS) 6.8.1., three examples, (paragraphs 3 and 7).

REPORT DETAILS

1. Persons Contacted

Licensee Employees Contacted

*C. M. Wethy, Vice President - Turkey Point
*C. J. Baker, Plant Manager - Nuclear
E. Preast, Site Engineering Manager
D. W. Hasse, Safety Engineering Group Chairman
*D. D. Grandage, Operations Superintendent - Nuclear
T. A. Finn, Operations Supervisor
J. Crockford, Assistant Operations Supervisor
J. Webb, Operations/ Maintenance Coordinator
*B. A. Abrishami, Acting Technical Department Supervisor
D. A. Chaney, Corporate Licensing
*J. Arias, Regulation and Compliance Supervisor
*R. Hart, Regulation and Compliance Engineer
*J. W. Kappes, Maintenance Superintendent - Nuclear
O. E. Suero, Electrical Maintenance Supervisor
R. A. Longtemps, Mechanical Maintenance Supervisor
E. F. Hayes, Instrument and Control (I&C) Maintenance Supervisor
R. G. Mende, Reactor Engineering Supervisor
R. E. Garrett, Plant Security Supervisor
P. W. Hughes, Health Physics Supervisor
W. C. Miller, Training Supervisor
J. M. Donis, Site Engineering Supervisor
J. M. Mowbray, Site Mechanical Engineer
*R. H. Reinhardt, Acting Quality Control (QC) Supervisor
*L. W. Bladow, Quality Assurance (QA) Superintendent
*J. A. Labarraque, Performance Enhancement Program (PEP) Manager

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

*Attended exit interview

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 March 13, 1986. The areas requiring management attention were reviewed.

One violation was identified:

Failure to meet the requirements of TS 6.8.1, three examples, in that: (1) Operating Procedure (OP) 3400.1 did not address instrument air requirements for intake cooling water (ICW) discharge check valve operability (paragraph 7); (2) failure to implement Administrative

Procedure (AP) 0140.2 in that the Precautions, Limitations and Setpoint Document was not updated (paragraph 3); and (3) Off-Normal Operating Procedure (ONOP) 9608.1 was used for a purpose other than that for which it was intended (paragraph 7) (250,251/86-10-01).

Four unresolved items (UNRs) were identified:

Resolve NRC concerns over the adequacy of the licensee's ICW system design. ICW system control valves (CV) 2202 and 2201 were determined to fail in the "as is" position upon loss of instrument air. During an accident response, the ICW system could be configured in a lineup different from any analyzed in the Final Safety Analysis Report (FSAR) (paragraph 10) (250,251/86-10-03).

Resolve NRC concerns that improper throttling setpoints for component cooling water (CCW) valves 748A and 748B might have, under certain accident conditions, rendered the CCW system unable to supply adequate cooling water to safety-related components (paragraph 5) (250,251/86-10-04).

Resolve NRC concerns that Unit 3 circulating water pump (CWP) lube water system piping might have been improperly supported, such that a seismic event could have sheared the lube water/ICW pipe joint, resulting in a loss of ICW flow greater than previously analyzed (paragraph 8) (250,251/86-10-05).

Resolve NRC concerns that nonseismically qualified support frames for ICW pump motor heater control boxes could adversely affect pump operation during a seismic event (paragraph 8) (250,251/86-10-06).

One inspector follow-up item (IFI) was identified:

Reinspect the Unit 3 and Unit 4 intake water structure to verify correction of ICW system discrepancies (paragraph 8) (250,251/86-10-07).

The licensee did not identify as proprietary any of the materials provided to or reviewed by the inspectors during this inspection. The licensee acknowledged the findings without dissenting comments.

3. Licensee Action on Previous Inspection Findings (92702)

a. Performance Enhancement Program (PEP) Summary

On February 28, 1986, the licensee held a dedication and ground breaking ceremony for the simulator and training facility. Construction of the facility is scheduled for completion in September 1986. The simulator is scheduled to be completely installed and operable in June 1987.

On March 7, 1986, the licensee conducted the quarterly PEP update briefing for NRC management. The meeting was held at the Turkey Point

site in the nearly completed administration building. Presentations were made in the areas of management initiatives, operations performance, maintenance performance, material management (spare parts) enhancements, training, and PEP schedule commitments. The meeting was beneficial in providing summaries of progress in these areas.

b. Previously Identified Items

(Closed) Licensee Event Report (LER) 250/84-28 - This event involved the improper installation of a power range nuclear instrument drawer because of a failure to establish an adequate maintenance procedure. This procedural inadequacy and the resultant failure to maintain the required level of instrument redundancy were cited as violations (250/84-39-03 and 250/84-39-02 respectively) which are addressed and closed separately. The corrective measures taken for those violations are considered adequate to close this LER.

(Closed) Violation 250/84-39-02 - This violation involved a failure to maintain the minimum degree of redundancy required by the TS for the overtemperature and overpower delta-T protective circuits. OP 12304.2, Power Range Nuclear Instrumentation Periodic Channel Functional Test, has been revised to provide operator guidance on removing a channel from service when acceptance criteria cannot be met.

(Closed) Violation 250/84-39-03 - This violation involved the installation of a Unit 3 power range nuclear instrument drawer without the benefit of an approved procedure. The corrective actions for this violation were previously reviewed in Inspection Report 250,251/85-06 and found to be incomplete.

The licensee has since revised Maintenance Procedure (MP) 12307.3, "Quarterly and Standard Calibration of the Nuclear Power Range Instrumentation, Axial Flux Deviation Process Instrumentation to OPSP and OTSP, and Nuclear Power Range Axial Flux Deviation Alarm," to include testing of both leads which interconnect the power range nuclear instrument and the overpower and overtemperature delta-T protective circuits. The licensee has also developed a new procedure, Troubleshooting and Repair Guidelines (O-GMI-102.1), which governs the performance of general maintenance, troubleshooting, and repair on installed safety-related systems. This procedure provides direction on the proper labeling of any electrical leads which are disconnected during troubleshooting or maintenance and requires that disconnected leads be independently verified upon reconnection.

(Closed) Violation 250,251/85-06-01 - This violation involved a failure to adequately calibrate the power range nuclear instruments, contrary to TS 4.1. MP 12307.3 has been revised to require testing of the entire instrument channel. Overtemperature and overpower delta-T protective circuit inputs from the nuclear instrumentation system are now verified for proper voltage and polarity.

(Closed) LER 250/84-35 - This event involved the failure of a power operated relief valve (PORV) block valve to close upon demand. The malfunction was caused by a failed block valve torque switch. The switch was replaced under Plant Work Order (PWO) 4429-63; post-maintenance testing was satisfactorily completed on December 30, 1984.

(Closed) LER 250/84-34 - This event involved the failure of a reactor coolant system (RCS) letdown isolation valve, CV-3-204, to close upon demand. The malfunction was caused by a failed solenoid valve which continuously applied air to the CV-3-204 diaphragm thereby preventing it from closing. The solenoid valve was replaced under PWO 6290-63; post-maintenance testing was satisfactorily completed on December 14, 1984.

(Closed) LER 250/84-31 - This event involved the failure of a containment isolation valve, CV-3-855, to close upon demand. The malfunction was caused by a failed ASCO solenoid valve which continuously applied air to the CV-3-855 diaphragm thereby preventing it from closing. The solenoid valve was replaced under PWO 6251-63. As stated in paragraph 9 of Inspection Report 250,251/86-05, the licensee has agreed to send future failed ASCO solenoid valves to the vendor for troubleshooting to determine the cause of failure.

(Closed) LER 250/84-18 - This event involved a notification by Bechtel Power Corporation and power plant engineering that a 10 CFR Part 21 deficiency existed in the control circuitry for safety-related pressure controllers PC-600 and PC-601. The cause of the deficiency was a failure to identify the control circuitry as safety-related and was documented in Inspection Report 250/84-23, 251/84-24 as a further example of violation 250,251/84-09-05.

The pressure controllers in question were provided with separate and redundant safety-grade power supplies under Plant Change/Modifications (PC/Ms) 84-132 and 84-133 for Units 3 and 4, respectively. The cause of the deficiency will continue to be tracked as violation 250,251/84-09-05.

(Open) LER 250/84-27 - The event involved the discovery of four types of material discrepancies (cracked insulating link, under-voltage trip attachment coil tape damage, manual closing mechanism bracket cracked welds, and manual closing mechanism failed bearing) while performing periodic maintenance on reactor trip/bypass breakers. All of these discrepancies, with the exception of the cracked welds on the manual closing mechanism brackets, have been corrected under various PWOs. In a letter to the licensee dated October 5, 1984, Westinghouse stated that the cracked welds would not compromise the normal (electrical) opening and closing functions of the breakers. Westinghouse further stated that they would forward a repair procedure to FPL. The licensee failed to followup on this vendor commitment and consequently had not, as of March 7, 1986, repaired the cracked welds. The licensee has

reinitiated contact with Westinghouse and will pursue repair of the cracked welds. This event will remain open pending completion of those repairs.

(Open) LER 250/84-29 - This event involved a turbine runback caused by an operator inadvertently taking the dropped rod mode switch out of bypass while performing OP 12304.2, Power Range Nuclear Instrumentation Periodic Channel Functional Test, on channel N-43. The resultant rapid decrease in reactor power, as sensed by the other power range nuclear instruments, caused a second turbine runback because of what appeared to the instruments as a dropped control rod. The licensee tested the governor runback motor and found that it was running at 2440 revolutions per minute (rpm) in the lower direction instead of the rated 1500 rpm. This resulted in turbine runbacks at a rate in excess of the design 200 percent per minute and of a magnitude in excess of the design 30 percent.

As a corrective measure the licensee replaced the turbine governor runback motor with one which was tested to run at 1080 rpm thereby reducing generator load at a rate of 158 percent per minute. In order to achieve the design 30 percent turbine runback, the licensee adjusted the time delay relay setpoint, which governs the time period for which the runback motor runs, from 9 seconds to 11.5 seconds.

Administrative Procedure (AP) 0140.2 establishes the controls to ensure proper plant review and approval of proposed setpoint changes. It requires that the Technical Department review and process a Setpoint Change Worksheet (Appendix "A" to AP 0140.2), prepare a safety evaluation, obtain appropriate approvals, and notify the appropriate implementing department of the setpoint change. AP 0140.2 further assigns responsibility to the I&C Department for updating the Precautions, Limitations and Setpoints (PLS) Document when a setpoint change is implemented.

A review of the PLS Document revealed that the Unit 3 setpoints related to rod drop protection had never been updated to reflect the turbine runback modification implemented in December 1984.

This discrepancy constitutes a failure to properly implement AP 0140.2, Changing Setpoints, dated April 13, 1984, and is contrary to the requirements of TS 6.8.1. It is one of three examples of violation 250,251/86-10-01.

(Closed) LER 250/84-33 - This event involved a reactor trip caused by a turbine trip at 100 percent power. The turbine tripped as a result of an exciter failure which caused a loss of generator field and a generator trip. The exciter was removed and replaced with one procured from another facility. The failed exciter was shipped to Westinghouse for inspection and refurbishment and is being installed in Unit 4 during the current refueling outage. The licensee retained the services of a contractor to perform a failure analysis; all of the contractor's

recommendations have been implemented to preclude a recurrence of this failure.

4. Unresolved Items

An unresolved item is a matter about which more information is required to determine whether it is acceptable or may involve a violation or deviation. Four unresolved items were identified during this inspection and are summarized in paragraph 2.

5. 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 inspector verified that testing frequencies were met and tests were performed by qualified individuals.

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

- Special Test 86-01, Unit 4 CCW System Flow Test
- Special Test 86-01, Unit 3 CCW System Flow Test
- Special Test 86-03, Unit 3 CCW System Residual Heat Removal Heat Exchanger Throttling Valve Adjustment Test

As a result of findings identified in NRC Inspection Reports 250,251/85-32 and 250,251/85-40, FPL committed to perform a two phase Select System Review. The purpose of the review is to assure that concerns expressed in Inspection Report 250,251/85-40 regarding the operation and function of the auxiliary feedwater system are addressed relative to other important plant systems. The Phase 1 assessment was completed on January 31, 1986. The Phase 2 assessment is in progress and is scheduled for review completion in December 1986. Modifications resulting from the assessments are scheduled to be complete for Unit 3 in the spring of 1987 and for Unit 4 in the fall of 1987. The NRC Region II management staff is evaluating the appropriateness of the corrective action schedule.

The Phase 1 safety system review identified a significant concern relative to the adequacy of CCW system flow to meet heat removal requirements during post accident recirculation cooling with only one operable CCW pump. Single pump operation during the recirculation phase must be anticipated due to the design basis assumption that a failure will incapacitate one of the two normally available pumps.

CCW system valves 748A and 748B for each Unit were throttled to 30 percent open in 1972 to preclude exceeding the maximum desired CCW flow through the

residual heat removal (RHR) system heat exchangers. The throttling was done at the recommendation of the nuclear steam supply system (NSSS) vendor. The licensee believes that an RHR heat exchanger upgrade was performed on each heat exchanger in the 1974 time frame. The upgrade eliminated the need to restrict flow to 4000 gallons per minute (gpm) through each heat exchanger. However, the licensee did not return valves 748A and 748B to their previous positions; the valves remained throttled to 30 percent open.

In mid January 1986, the licensee calculated that the 30 percent throttled position of these valves would not allow 4000 gpm flow through the RHR heat exchangers during the recirculation phase following the occurrence of the maximum hypothetical accident (MHA), assuming one CCW pump operation. The valves were not immediately realigned because the licensee was not sure that 4000 gpm was the minimum flow assumed to exist in the design analysis. The licensee presented its finding in the Safety System Review Phase 1 Report on January 31, 1986.

Between mid January and late February the licensee researched the design basis for the CCW system with the help of the NSSS vendor and the on-site engineering contractor. The NSSS vendor was asked to determine whether the 4000 gpm figure could be reduced through additional engineering calculations. The vendor responded that a reanalysis was possible but it could not be performed in a short period of time.

On February 24, 1986, Unit 3 CCW valves 748A and 748B were fully opened to lessen the concern that insufficient CCW might be available during the MHA, single CCW pump scenario. The Unit 4 valve positions were not of immediate concern because the Unit was shutdown in a refueling outage.

On March 4, 1986, Special Test 86-01, Unit 4 CCW System Flow Test, was performed to evaluate the CCW flow through 1 CCW pump, 1 RHR heat exchanger, 3 CCW heat exchangers and both CCW headers. Various system alignments and throttle positions were tested in an effort to provide assurance that design flow rates were achievable following the MHA.

Following preliminary reviews of the test results it was determined that operation with valves 748A and 748B fully open resulted in CCW flows to each of three emergency containment coolers (ECCs) of less than the required 2000 gpm flow assumed in the FSAR. Additionally, the test results confirmed that with the valves throttled to the 30 percent open position, insufficient CCW flow passed through the RHR heat exchangers.

At 9:15 p.m. on March 4, 1986, the licensee declared all three Unit 3 ECCs out of service and began to shutdown the reactor.

The NRC is continuing to evaluate the status of the licenee's CCW system. Special tests performed on March 3 and March 8 indicate that, as a result of throttling CCW valves 748A and 748B in 1972, the CCW system might not have been capable of supplying adequate cooling water to the RHR heat exchangers. After the valves were restored to the full open position the CCW system might not have been capable of supplying adequate cooling water to the ECCs.

These issues constitute an unresolved item pending licensee and NRC review of final test data to ascertain the capabilities of the CCW system with valves 748A and 748B throttled and fully open (250,251/86-10-04).

6. Maintenance Observations (62703/62700)

Station maintenance activities on safety-related systems and components were observed and reviewed to ascertain that they were conducted in accordance with approved procedures, regulatory guides, industry codes and standards and in conformance with the 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 trouble-shooting activities were controlled and repair records accurately reflected what took place; 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.

The following maintenance activities were observed and/or reviewed:

- AFW Pump "C" turbine replacement (PWO 046450)
- AFW pump "B" Trip and Throttle valve repair (PWO 5540)
- Unit 3 Source Range Nuclear Instrument Repair
- ICW Discharge Check Valve 4-321 Disassembly and Inspection
- Unit 3 Lube Water Pipe Support Installation
- ICW Pump "3A" Motor Cable Inspection

7. 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 follow-up 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 on Unit 3 and Unit 4 to verify operability and proper valve/switch alignment:

- Emergency Diesel Generators
- Auxiliary Feedwater
- 4160 Volt and 480 Volt Switchgear
- Control Room Vertical Panels and Safeguards Racks
- Intake Cooling Water
- Unit 3 Residual Heat Removal System

Three areas of noncompliance were identified during this inspection in the area of plant operations.

a. Use of a procedure for other than its intended purpose

On February 11, 1986, the Unit 3 reactor tripped due to a personnel error. Control room operators received an annunciator alarm indicating that a containment isolation cabinet fuse failure had occurred.

The operators did not realize that this alarm was caused when construction personnel, working near the 4D DC load center, bumped open breaker 4D0139. The workers had removed a protective panel in front of the load center to facilitate pulling new electrical cables. Removal of the cover was not essential and the task could have proceeded with the cover in place.

Inadvertently opening breaker 4D0139 resulted in a loss of power to one channel of containment high and high-high pressure bistables. The bistables failed in the alarmed condition as designed. The annunciator procedure for the fuse failure target required that the operators check the affected instrument rack to determine the source of the problem and that the DC breaker list be consulted to identify the appropriate power supply.

The control room operators chose to use Off-Normal Operating Procedure (ONOP) 9608.1 to identify the problem source because it provides guidance as to allowed breaker manipulations while the DC breaker list does not. However, the operators failed to realize that the guidance on allowed breaker manipulations found in ONOP 9608.1 was based on the assumption that the breaker being manipulated would provide the only deenergized circuit.

Consequently, the operators used ONOP 9608.1 for a purpose other than that for which it was intended. The operators opened breaker 3D2311 as allowed by the procedure, which deenergized a second channel of containment high pressure and in turn, actuated the engineered safety features logic which caused a reactor trip. Due to the particular breakers which were opened, only Train "A" of the engineered safety features actuated. The inspector verified that all systems responded as required for the circumstances.

Technical Specification (TS) 6.8.1 requires that written procedures and administrative policies be established, implemented and maintained that meet or exceed the requirements and recommendations of sections 5.1 and 5.3 of ANSI N18.7-1972 and Appendix A of USNRC Regulatory Guide 1.33.

Section 5.1.6.1 of ANSI N18.7-1972 requires that maintenance which can affect the performance of safety-related equipment shall be properly preplanned and performed in accordance with written procedures, documented instructions or drawings appropriate to the circumstances.

Contrary to the above, on February 11, 1986, maintenance troubleshooting was performed which was not properly preplanned and relied upon a procedure which was not appropriate to the circumstances, in that ONOP 9608.1, dated October 16, 1985, entitled 125 Volt DC System - Location Of Grounds, was used for a purpose other than that for which it was intended. Consequently, operations personnel inadvertently initiated a reactor trip by implementing a procedural step, normally acceptable during ground isolation proceedings, solely to identify the source of a perceived fuse failure in the reactor protection system. The step was inappropriate to the circumstances because no 125 volt DC ground existed and a reactor protection system fault existed which invalidated the procedural guidance.

This discrepancy constitutes one of three examples of the licensee's failure to meet the requirements of TS 6.8.1. The three examples jointly constitute violation 250,251/86-10-01.

b. Failure to establish an adequate operating procedure for the ICW system

The ICW system is an essential subsystem of the shutdown cooling system and the emergency core cooling system. Operating Procedure 3400.1, Intake Cooling Water System - Normal Operation, provides instructions for ICW system operation and alignment.

Technical Specification 6.8.1 requires that written procedures and administrative policies be established, implemented and maintained that meet or exceed the requirements and recommendations of sections 5.1 and 5.3 of ANSI N18.7-1972 and Appendix A of USNRC Regulatory Guide 1.33.

Appendix A of USNRC Regulatory Guide 1.33 states that written procedures should be established for the shutdown cooling system and the emergency core cooling system.

FPL inter-office correspondence PTN-TECH-85-754, ICW Pump Discharge Check Valves, dated November 7, 1985, states that instrument air to the ICW system pump discharge check valve closing cylinders is necessary for continued operation of the ICW system.

FPL inter-office correspondence JPE-PTPM-85-1409, dated December 16, 1985, postulates that the check valve air closing cylinders enhance valve operation by overcoming minor rust/friction binding to reduce check valve slam. The document states that air closing cylinders are not considered essential to ICW system operability provided that operation without instrument air available is kept negligibly short.

The instrument air discrepancies specified in paragraph 8, as part of IFI 250,251/86-10-07, contribute to the potential for degraded performance of the ICW check valve closing cylinders and thus, if undetected, could impact ICW system operability.

Contrary to the above, as of March 10, 1986, Operating Procedure 3400.1, dated August 7, 1985, was inadequate, in that it failed to provide any guidance in the form of requirements and limitations for the operation of the Units 3 and 4 ICW pump discharge check valves (3/4-311, 321, 331) with respect to the availability of instrument air to the check valve closing cylinders.

This discrepancy constitutes one of three examples of the licensee's failure to meet the requirements of TS 6.8.1. The three examples jointly constitute violation 250,251/86-10-01.

8. Engineered Safety Features Walkdown (71710)

The inspector verified operability of the intake cooling water system for Units 3 and 4 by performing a complete walkdown of the accessible portion of the system. The following specifics were reviewed and/or observed as appropriate:

a. that the licensee's system lineup procedures matched plant drawings and the as-built configuration;

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

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

d. that valves were in proper position and were locked/lockwired as required;

e. that local and remote position indications were functional and in agreement;

f. that breakers and instrumentation cabinets were free of damage and interference; and

g. that breaker alignment was correct and power was available.

During the walkdown two unresolved items and several maintenance discrepancies were brought to the attention of the licensee. The Unit 3 reactor was in cold shutdown and the Unit 4 reactor was in refueling shutdown during the system inspection. Repairs and refurbishment were in progress for the Unit 4 system as part of the normal refueling outage. Maintenance items directly attributable to Unit 4 outage work were not considered as problem areas.

The inspectors observed a portion of the Unit 3 piping which supplies the circulating water pumps with bearing lubrication from the A and B ICW headers. The stainless steel piping showed signs of degradation at several weld joints. One weld joint was leaking. Similar piping for Unit 4 was in the process of being replaced with polyvinyl chloride (PVC) piping.

The piping was determined to be class "D", non-safety-related. However, it was not properly supported in that several supports were not made up. A long stretch of horizontal pipe appeared to have no restriction to movement in the axial direction. The lube water pipe is connected to the ICW headers via an orifice, check valve and manual isolation valve.

Degraded welds on the lube water pipe, indicative of poor maintenance practices, did not constitute a safety concern because the orifice near the lube water/ICW interface would restrict lost ICW flow to a relatively small amount. However, the inspectors were concerned that, in the absence of axial restraints, the lube water pipe could be susceptible to shearing at a point between the orifice and the ICW header. The consequences of a pipe shear in this area could be severe because the escaping ICW would not be limited by the downstream orifice.

The adequacy of the existing lube water pipe supports in preventing a pipe shear between the orifice and the ICW header is an unresolved item pending completion of a licensee evaluation of the pipe stresses which would be experienced during a seismic event (250,251/86-10-05).

Additionally, it was noted that supports for the electrical junction boxes for each ICW pump did not appear to be seismically installed. These junction boxes are located on metal supports immediately adjacent to the associated ICW pump. For each Unit 3 and Unit 4 ICW pump, a horizontal crossbar between two vertical supports was clamped to the cable shield containing the

main motor power cables. The motor heater junction box, electrically connecting the motor heater to its power supply did not appear to be seismically mounted.

The inspector raised the concern that, during a seismic event, the metal supports could give way causing the motor heater junction box to pull away from the pump. Additionally, it seemed possible that, during a seismic event, the horizontal crossbar could pull on the cable shield for the main motor cables.

The adequacy of the metal supports adjacent to each ICW pump is an unresolved item pending the completion of a seismic analysis by the licensee and evaluation of the data by the NRC (250,251/86-10-06).

Additional maintenance discrepancies were noted and brought to the attention of the licensee for correction. These items constitute an IFI and will be reinspected at a later date to verify completion of corrective actions (250,251/86-10-07).

- Instrument air piping for the ICW pump discharge check valves was not adequately supported, in that clips and supports were severely corroded.

- Two pressure gages on air regulators were degraded, in that one did not show any pressure indication and one suffered from water intrusion.

- One threaded air pipe was loose, increasing the possibility of air leakage.

- One air vent plug protective cover was broken such that foreign material could have entered the air closing cylinder.

- Corrosion was apparent at the base of the cable shield for the Unit "3A" ICW pump main motor cables.

- ICW pump discharge pressure gages did not appear to be calibrated.

- The "3A" ICW pump motor was missing several bolts for motor cover plates.

- A support next to each Unit 3 and Unit 4 ICW motor junction box was so loose that it could impact the motor when shaken by hand.

- A 20 foot section of spare 3 inch pipe was wedged between the "3A" and "3B" ICW system headers.

- Several valve tags were illegible because they had been taped over while painting was in progress. The tape had not been removed.

- The "4A" ICW pump discharge check valve was partially stuck, in that a leak from one side of the check valve caused a buildup of corrosion products resulting in partial binding of the check mechanism. After

being notified of this discrepancy, the licensee removed and inspected the check valve. The corrosion was not severe enough to prevent the valve from fulfilling its intended function. Following reinstallation, no binding was evident.

The licensee began to address these items and unresolved items 250,251/86-10-05 and 250,251/86-10-06 following identification. Several of the items were previously known to exist but had not yet received appropriate attention.

9. Independent Inspection

During the report period the inspectors routinely attended meetings with licensee management and monitored shift turnovers between shift supervisors, shift foremen and licensed operators. These meetings included daily discussions of plant operating and testing activities as well as discussions of significant problems or incidents. As a result, the inspectors reviewed potential problem areas to independently assess their importance to safety, the adequacy of proposed solutions and corrective actions, and improvements in progress. The inspector's reviews of these matters were not limited to the defined inspection program. Independent inspection efforts were conducted in the following areas:

- ICW System Valve 2202 Single Failure determination
- CCW System Flow Balance Testing
- Threaded Fastener Overheating
- ICW System Records Review

No violations or deviations were identified.

10. Plant Events (93702)

An independent review was conducted of the following event.

On February 13, 1986, Power Plant Engineering informed the Turkey Point staff that the ICW system was susceptible to failure because two valves, CV-2201 and CV-2202, might not perform as required. Each valve is operated automatically using the instrument air system. However, the instrument air system is not safety-related and is assumed to fail during accident scenarios.

CV-2201 must shut on receipt of a safety injection signal to isolate the turbine plant cooling water (TPCW) heat exchangers from the ICW system. It was determined that failures could occur (such as loss of instrument air) which would result in CV-2201 remaining open during an accident. Consequently, accident scenarios exist in which CV-2201 is required to close but can not.

The worst case scenario occurs when a single active failure results in only one operable ICW pump. A reduction in cooling water flow occurs due to only one working pump and that pump's cooling effectiveness is further reduced because water is being directed away from the CCW heat exchangers to the

TPCW heat exchangers through failed valve CV-2201. Consequently, the ICW system will provide less than the design flow requirements of the CCW heat exchangers. As of February 13, 1986, the degree to which this condition could adversely affect the safety function of the ICW system had not been analyzed. The analysis was not performed because it was believed that CV-2201 would shut as required under all circumstances.

CV-2202, the CCW heat exchanger outlet control valve, was also identified by Power Plant Engineering as being able to fail in a manner previously not identified. The valve is required to remain open during an accident to allow ICW to remove heat from the CCW heat exchangers. It was determined that the valve would fail closed on loss of instrument air. This created the potential for a complete loss of the capability to remove accident heat loads.

The immediate impact of the CV-2202 problem was mitigated by previously approved plant operating procedures which required the bypass valve around CV-2202 to be kept open. Consequently, even though CV-2202 could fail closed ICW would reach the CCW heat exchangers through the bypass.

On February 13, 1986, the Unit 3 reactor was operating at 100 percent power. The Unit 4 reactor was in refueling shutdown. At 8:11 p.m. the licensee notified the USNRC operations center of the significant event under 10 CFR 50.72 b.2.(iii)(B). This item refers to any event or condition that alone could have prevented the fulfillment of the safety function of structures or systems that are needed to remove residual heat. License corrective actions included the preparation of procedure changes directing the Nuclear Turbine Operator to isolate CV-2201 if the accident of concern should occur and conducting shift briefings to explain the problem to on-shift personnel.

On February 14, 1986, following a review of the licensee's actions, it was determined by the NRC Region II staff that the failure of CV-2201 to close when needed placed Unit 3 outside any limiting conditions for operation allowed by the TS. In such a circumstance, the reactor should have been shut down under the guidance of TS 3.0.1. The licensee began a power reduction at 12:27 p.m.

On February 14, 1986, at 11:55 a.m., the licensee stationed a dedicated watchstander at Unit 3 CV-2201. Timing tests were conducted to determine the length of time required to isolate the valve. Calculations were performed to verify that the time required to shut the valve was not longer than the time available prior to exceeding system design heat load parameters. The operator was provided with written instructions and was in constant radio communication with the control room. At 4:30 p.m. on February 14, 1986, the Plant Nuclear Safety Committee approved the compensatory measures which had been implemented and the Unit 3 load reduction was terminated. The reactor was returned to 100 percent power at 5:37 p.m.

The FSAR states, in section 9.1.1, that each of the auxiliary cooling systems which serve an emergency function provide sufficient capability in the emergency operational mode to accommodate any single failure of an

active component and still function in a manner to avoid undue risk to the health and safety of plant personnel and the public.

Additionally, General Design Criterion (GDC) 52, as stated on page 9.1-2 of the FSAR, specifies:

> Where an active heat removal system is needed under accident conditions to prevent exceeding containment design pressure this system shall perform its design function, assuming failure of any single active component.

The licensee has determined that valves CV-2201 and CV-2202 for both Units 3 and 4 presently exist in substantially the same configuration and design as existed when the plants were constructed. The possible construction and operation of the ICW system in a manner that allowed a single active failure to preclude the removal of accident heat loads is a matter receiving additional NRC review and analysis. Additionally, the NRC is evaluating the length of time required by the licensee to evaluate the potential for valves CV-2201 and CV-2202 to fail in manners previous not thought possible. The original licensee review of the valves' failure characteristics was begun in the fall of 1984 but was not completed until February 13, 1986. The adequacy of the licensee's response to the identification of the problem prior to receiving direction from the NRC Region II staff is also under review. These items together constitute an unresolved item (250,251/86-10-03).

Within this area no violations or deviations were identified.