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

Report Nos.: 50-250/84-34 and 50-251/84-35

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: September 23 - October 20, 1984

Inspectors: SAElrod 11/27/84
T. A. Peebles, Senior Resident Inspector Date Signed

SAElrod 11/27/84
D. R. Brewer, Resident Inspector Date Signed

Approved by: SAElrod 11/27/84
Stephen A. Elrod, Section Chief Date Signed
Division of Reactor Projects

SUMMARY

Scope: This routine, unannounced inspection entailed 277 inspector-hours at the site, including 42 hours of backshift, in the areas of licensee action on previous inspection findings, annual and monthly surveillance, annual and monthly maintenance, operational safety, Engineered Safety Features walkdown, plant events, and independent inspection.

Results: Of the seven areas inspected no violations or deviations were identified in five areas; four violations were identified in two areas (failure to establish maintenance procedures, paragraph 6; failure to properly implement a temporary procedure change, paragraph 6; failure to implement maintenance procedures, paragraph 6; and failure to establish adequate instructions or drawings, paragraph 8).

REPORT DETAILS

1. Licensee Employees Contacted

K. N. Harris, Vice President-Turkey Point
*C. J. Baker, Plant Manager-Nuclear
D. W. Haase, Chairman Safety Engineer, Group
J. P. Mendietta, Service Manager-Nuclear
*D. D. Grandage, Operations Supt. - Nuclear
*T. Young, Project Site Manager
J. W. Kappes, Maintenance Supt. - Nuclear
T. A. Finn, Operations Supervisor
J. A. Labarraque, Tech. Dept. Superintendent
P. W. Hughes, Health Physics Supervisor
W. C. Miller, Training Supervisor
*M. J. Crisler, Quality Control Supervisor
*K. L. Jones, Site QA Superintendent
L. C. Huenniger, Start-up Superintendent
W. R. Williams, Asst. Supt. Elect. Maint.
*R. A. Longtemps, Asst. Supt. Mechanical Maint.
*H. Arias, Jr. Regulation and Compliance Eng.
J. M. Donis, Site Eng. Supervisor
J. M. Mobray, Site Mechanical Eng.
*P. J. Baum, Training Supervisor
V. A. Kamiskas, Reactor Eng. Supervisor
*B. A. Abrishami, IST Supervisor
R. G. Mende, Reactor Engineer
*R. M. Brown, HP Supervisor
D. Tomaszewski, Plant Eng. Supervisor
R. E. Garrett, Plant Security Supervisor
J. E. Moaba, Corporate Licensing
G. J. Boissy, PEP Program Manager

*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.

The exit meeting was held on October 19, 1984, with the persons noted above. The areas requiring management attention were reviewed. The items which were potential violations were: failure to establish maintenance procedures, five examples (250/84-34-01 & 251/84-35-01): failure to properly implement a temporary change as required by Tech. Spec. 6.8.3 (250/84-34-02 & 251/84-35-02): failure to properly implement procedures, three examples (250/84-34-03 & 251/84-35-03); and failure to implement the requirements of 10 CFR 50, Appendix B, Criterion V, for an activity affecting quality (251/84-35-04). The items which the inspectors were to follow-up were: correction of auxiliary feedwater system valve labels and drawings (250/84-34-05 and 251/84-84-34-05); hoisting and rigging program (250/84-34-06 & 251/84-35-06); Health Physics discrepancies in RWP requirements (250/84-34-07 & 251/84-35-07); simultaneous push button requirement in reactor protection testing (250/84-34-08 & 251/84-35-08); operator headsets for surveillance (250/85-34-09 & 251/84-35-09); and possible graphite corrosion of component cooling water heat exchanger (250/84-34-10 and 251/84-35-10).

3. Licensee Action on Previous Inspection Findings (92702)

   a. Monthly update of Performance Enhancement Program (PEP)

      The PEP was reviewed to determine if commitments were being met. Status was discussed with the PEP Manager and with other management. Major areas are on schedule. The only notable exception is that an additional personnel slot for the Services Dept. has not yet been filled. The region has been updated and agrees with the progress.

      Several new projects have been added to PEP as a result of commitments from an enforcement conference held on September 19, 1984, and subsequent letters from the licensee. These resulted in a Confirmation of Concurrence letter being sent to the licensee on October 11, 1984. Inspection of these items will be conducted during PEP inspections and under the routine program.

      The procedures upgrade project is meeting their timetable, but it is recognized that improvement is needed in the final review process.

      The site facility upgrade is progressing on schedule; however, permits have not been obtained for the filling of a mangrove area to accommodate the simulator. Fill for the administrative building is almost to the stage where driving of the pilings could begin, but a contractor has not been chosen. The HP facility is ready to be started, but a contractor has not been chosen. Other construction permits have not been obtained and may begin to impact the schedule before the end of the year.

      The Standard Technical Specification upgrade project has moved into their new quarters. The coordinator is on site and four other site personnel have arrived. They have begun writing the individual job task statements.

   b. The re-submittal of several sections of the Inservice Test (IST) program and associated Technical Specifications (TS) remains outstanding. (IFI 84-23-07 & 84-24-07 OPEN). The Plant Nuclear Safety Committee recently (PNSC) approved a change to the IST submittal to incorporate several outstanding items.

4. Unresolved Items

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

No unresolved items were identified.

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

The inspectors observed Technical Specification (TS) required surveillance testing and verified that testing was performed in accordance with adequate procedures; that test instrumentation was calibrated; that limiting conditions for operation 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 that any deficiencies identified during the testing 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 Inservice Test (IST) program for pumps and valves was reviewed for adequacy against ASME Section XI and the TS.

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

Reactor Protection System - periodic test
Safeguard relay rack train A, B and emergency load sequencer timer - periodic test
Reactor trip and reactor trip bypass breakers - inspection and maintenance.

The reactor protection system periodic test was performed on October 11, 1984, in accordance with operating procedure (OP) 1004.2. The procedure specifies an approved method of testing the reactor protection system reactor trip and permissive matrices, in compliance with Technical Specification table 4.1-1, item 24.

During the performance of the procedure, the operators had to stop and repeat occasional steps because they were not sure they were performed successfully. The lack of certainty resulted from a combination of poor communications and haste in completing the task. The poor communications resulted from poor transmission quality over the portable walkie-talkies. Operator's occasionally experienced difficulty understanding received transmissions. Numerous transmissions had to be repeated before both transmitter and recipient were confident that procedural steps were completed satisfactorily. Contributing to the communications difficulties was the unnecessary urgency on the part of the operators to complete the procedures. Occasionally the operators at the protection racks moved ahead of the control room operator and consequently they were not sure for which step the control room operator was acknowledging. Each time this occurred, the step in question was repeated.

While performing section 8.24.2 of the procedure an apparently erroneous trip signal was received. Step 4 requires that pushbuttons S5 and S2 be depressed simultaneously and that the operators verify that no trip signal was received. The control room operator received a trip signal. The step was repeated and the control room operator did not receive a trip signal. The operators at the protection rack decided that the control room operator must have been mistaken in his original response. Step 7 of the procedure requires pushbuttons S6 and S3 to be simultaneously depressed and that no trip signal be received. Again the control room operator received the improper trip indication. The step was repeated successfully. The operators were satisfied that procedural steps 4 and 7 were not indicative of a problem with the protection matrix. The operators could not explain why the system responded as it did. The Plant Supervisor-Nuclear had not been informed of the apparent discrepancy. One operator believed that the improper indications were received because the pushbuttons were not depressed "simultaneously". Discussions with the operator pushing the buttons revealed that they were depressed at as close to the same time as humanly possible. During a retest, it was discovered that if S2 was depressed before S5 then the erroneous trip signal occurred. However, if S5 was depressed before S2 the proper "no trip" resulted. Since the reason why could not be ascertained, the instrumentation and control department was requested to research the matter. This is an inspector followup item (250/84-34-08 and 251/84-35-08).

The inspector observed the performance of OP-4004.2, "Safeguard Relay Rack Train A, B and Emergency Load Sequencer Timer - Periodic Test on Unit 3". Again steps of the procedure had to be repeated because the control room operator could not maintain the pace of the operators at the test racks. The inspector brought the matter to the attention of the Plant Supervisor - Nuclear who had not noticed that the hasty performance of the procedure was causing confusion.

Additional confusion was generated due to the necessity of the control room operator to shout his responses to the operator at the test racks. The shouting was distracting and could easily be heard on the Unit 4 side of the control room. The plant Supervisor - Nuclear indicated that the installed intercom system could not be used because the phone cords were too short. New, longer cords and headsets are due to be installed within 30 days. This is an inspector followup item (250/84-34-09 and 251/84-35-09).

As a result of these surveillance observations the inspector concluded that additional supervisory involvement in the performance of these procedures is warranted.

A detailed discussion of the problems discovered during the performance of Maintenance Procedure (MP) 0707.10, "Reactor Trip and Reactor Trip Bypass Breakers - Inspection and Maintenance" is included in paragraph 9, "Plant Events".

6. Monthly and Refueling Maintenance Observations (62703)

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

The following items were considered during this review: limiting conditions for operations were met while components or systems were removed from service; approvals were obtained prior to initiating the work; activities were accomplished using approved procedures and were inspected as applicable; functional testing and/or calibrations were performed prior to returning components or systems to service; quality control records were maintained; activities were accomplished by qualified personnel; parts and materials used were properly certified; radiological controls were implemented; fire prevention controls were implemented; and housecleaning was actively pursued.

The following maintenance activities were observed and/or reviewed:

3C & 4C Component cooling water heat exchanger cleaning
4C component cooling water heat exchanger tube plugging
4B intake cooling water strainer cleaning
4B residual heat removal pump repair

The component cooling water (CCW) heat exchangers were inspected and cleaned to ensure that tube fouling was not contributing to the reduced flow observed during system performance testing. Numerous discrepancies were identified by the inspectors during the maintenance activities on the 3C and 4C CCW heat exchangers. The discrepancies are itemized below.

a. Technical Specification 6.8.1 requires that written procedures be established that meet or exceed the requirements and recommendations of Appendix "A" of USNRC Regulatory Guide 1.33. The Regulatory Guide recommends, in Appendix "A", that procedures be developed for draining and refilling heat exchangers. The licensee has not developed procedures for draining and refilling the CCW heat exchangers.

b. The CCW heat exchanger are safety related and are essential for the cooldown of the plant following the design basis accident. It is essential that the integrity of the heat exchangers be reliably established. The licensee has not developed procedures for performing CCW heat exchanger tube leak tests. Following the cleaning of the 3C CCW heat exchanger, the licensee performed a tube test without benefit of a procedure. No tube leaks were identified. Several hours later the inspector determined, by visual observation, that the tubes were leaking. Additional leak testing by the licensee resulted in the plugging of five additional tubes. Acceptance criteria for allowable leakage has not been established.

c. The licensee does not have a procedure identifying precautions to be taken and methods to be followed to plug leaking CCW heat exchanger tubes. Consequently the five tubes plugged in the 3C CCW heat exchanger were not independently verified to be correctly installed in the proper locations.

d. The licensee does not have a procedure identifying precautions to be taken and methods to be followed to properly hydro-blast the CCW heat exchangers. Consequently the inspector determined through visual observation, that at least two tubes were missed during the general hydro-blast cleaning of the 4C CCW heat exchanger. The tubes were encrusted with sediment on one end such that they appeared to have been intentionally plugged with phenolic plugs. However, when the inspector touched the sediment it crumbled, revealing that the contract employee who performed the hydro-blast cleaning had misidentified the tube.

e. The licensee does not have a procedure identifying the proper method of performing general leak tests following the reinstallation of CCW heat exchanger end bell covers. Acceptance criteria for the tests have not been established.

The failure to establish procedures identifying the proper method of draining and filling the heat exchangers, plugging leaking heat exchanger tubes, testing for tube leakage, testing for end bell cover leakage and hydro-blast cleaning constitutes a failure to establish adequate maintenance procedures and is a violation (250/84-34-01 and 251/84-35-01).

The inspector identified additional discrepancies concerning the CCW heat exchangers which merit licensee attention. Inspection and Enforcement Information Notice 84-71, "Graphite Corrosion of Cast Iron in Salt Water" identifies a potentially significant problem pertaining to graphite corrosion of cast iron in salt or brackish water. Visual inspections of the CCW heat exchanger water boxes indicate that corrosion, possibly graphite, is definitely occurring. The possible graphite corrosion of the CCW heat exchangers is an inspector followup item (250-84-34-10 and 251-84-35-10). Visual observation of the vent and drain connections of the end bells of the CCW heat exchangers indicates that the piping junctions are plugged with corrosion products. These connections must remain clear to allow proper venting and draining of the water boxes.

On September 28, 1984, the inspector observed the cleaning of intake cooling water (ICW) basket strainer 4B. The basket strainer was being cleaned to reduce the indicated differential pressure across the strainer assembly. The licensee had not developed a formal procedure for cleaning the strainer. Cleaning the strainer required that the ICW inlet supply valve 4-324 be tagged shut. The top of the strainer assembly was removed to allow access to the basket, a large amount of leakage was apparent through the shut valve. The maintenance technician indicated that the inlet valves to the 3A, 3B, 4A and 4B basket strainers all leaked excessively. The condition has persisted for several years. The licensee does not specify the method to be used to clean the basket strainers. Maintenance technicians generally use scrub brushes to scrape fouling material from the basket strainer. Flushing water is supplied by the excessive leakage past the inlet isolation valve. The two backflush valves provided to generate reverse flow across the basket strainer are not used. Consequently the cleaning process is impeded because material scrubbed from the strainer is pushed back into the strainer by the flow of the water leaking past the inlet isolation valves. A drain valve (4-326) on the bottom of the basket strainer is used to remove flushing leakage water during the cleaning process. During the cleaning of the 4B basket strainer this valve was opened and subsequently reshut. No independent verification of the shutting of this valve (4-326) was performed. Independent verification of the valves operated during the cleaning process was performed and documented on the clearance form. Drain valve 4-326 was not included in the clearance and so its operation was not independently verified. The failure to have an adequate procedure for cleaning the ICW strainers is an additional example of potential violation 250/84-34-01 and 251/84-35-01.

On October 2, 1984, the inspector observed the repair of the 4B residual heat removal (RHR) pump. The RHR pump seal was leaking and repairs entailed the replacement of the mechanical seal in accordance with maintenance procedure (MP) 3207.2. During the removal of the pump, the inspector observed the use of a chain fall which had not been weight tested. A discussion of the licensee weight testing program is included in this report under paragraph 10, "Independent Inspection Effort."

Several discrepancies were noted in the implementation of MP-3207.2," Residual Heat Removal Pump - Disassembly, Repair, Seal Replacement and Assembly. The discrepancies together constitute a failure to have an adequate procedure for the repair of the RHR pump.

MP-3207.2 contains the following statement:

NOTE: Due to varying maintenance conditions and requirements, some steps in this procedure may be deleted for a specific activity. All steps deleted shall be marked "NA" to indicated Not Applicable for this activity. When the procedure is completed, a comment shall be entered listing all steps deleted and an explanation of why they were deleted.

An evaluation of MP-32-7.2 revealed that it did not contain any guidance identifying which steps of the procedure could be deleted. Guidance upon which to base the decision to delete a step of the procedure was nonexistent. Supervisory involvement in the decision to delete a step was not required. Consequently the personnel performing the procedure could delete safety related steps without the concurrence of the Plant Nuclear Safety Committee.

The maintenance technicians performing the repair chose to delete steps 9.14 and 9.15 and 9.16. These steps require the measurement of the impeller wear areas, measurement of the casing wear rings and comparison of the two measurements to ensure that the required tolerances were met. Immediately following step 9.16, a QC hold point was established to verify the acceptability of the data. The QC holdpoint was also deleted since there were no measurements to be verified.

Technical Specification (TS) 6.8.1 requires that written procedures be established implemented and maintained that meet the requirements and recommendations of Appendix "A" to USNRC Regulatory Guide 1.33. Appendix "A" of Regulatory Guide 1.33 requires that maintenance which can effect the performance of safety-related systems should be performed in accordance with written procedures. TS 6.8.3 allows temporary changes to procedures required by TS 6.8.1 provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant management staff, at least one of who holds a Senior Operators License on the Unit affected.

c. The change is documented, reviewed by the PNSC and approved by the Plant manager-Nuclear within fourteen days of the implementation.

The decision to delete several steps from MP-3207.2 constitutes a temporary change to the procedure. The change was never approved by two members of the plant management staff nor was it ever reviewed by the PNSC or approved by the Plant Manager-Nuclear. Consequently the temporary change constitutes a violation of TS 6.8.3 (250/84-34-02 and 251/84-35-02).

The performance of MP 3297.2 required the removal of safety-realted snubbers. prior to beginning MP-3207.2 and on-the-spot-change (OTSC) was generated to add steps to the procedure which would ensure that snubbers removed during the maintenance activity would be identified and reinstalled prior to returning the 4B RHR pump to service. One part of the OTSC required the following addition to step 9.5:

> "Verify all interferences (including snubbers) have been removed and record which items removed in remarks section."

Personnel performing MP-3207.2 did not add this statement to step 9.5. They did not make any entries in the remarks section of the procedure documenting which snubbers had been removed. This constitutes a failure to implement the OTSC and consequently a failure to implement MP-3207.2.

In May 1984, the licensee established MP-0707.33, "Snubber Removal and Replacement. This procedure provides detailed instructions for the removal and replacement of mechanical snubbers. MP-3207.2 did not reference MP-0707.33 and consequently the RHR snubbers were removed and reinstalled without using the procedure. This constitutes a failure to implement MP-0707.33.

Administrative Procedure (AP) 0190.10, "Cleaning of Nuclear Safety-Related Systems and Components: states that all openings in nuclear safety-related systems or components shall be protected from outside contaminants except when necessary to carry out required operations. During the performance of MP-3207.2 a component cooling water (CCW) pipe flange was disassembled and left with a pipe end open to the environment. The flange was not protected against foreign material intrusion. this constitutes a failure to implement AP-0190.10. A similar occurrence, involving potential contaminations of CCW piping removed from a high head safety injection pump is documented in inspection report 250/84-22 and 251/84-23.

The failure to implement the snubber documentation requirement of MP-3207.2, the failure to implement the snubber removal and installation requirements of MP-0707.33 and the failure to implement the cleanliness requirements of AP-0190.10 are three examples which together constitute a failure to meet the implementation requirements of Technical Specification 6.8.1 (250/84-34-03 and 251/84-35-03).

7. Operational and 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, reviewed tagout records, verified compliance with Technical Specification (TS) Limiting Conditions of Operation (LCO) and verified return to service of affected components.

The inspectors by observation and direct interviews verified that the physical security plan was being implemented in accordance with the station security plan.

The inspectors verified that maintenance work orders had been submitted as required and that followup and prioritization of work was on-going.

The inspectors observed plant housekeeping/cleanliness conditions and verified implementation of radiation protection control.

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 Units 3 and 4 to verify operability and proper valve alignment:

Emergency Diesel Generators
Component Cooling Water
Intake Cooling Water

Containment Spray
Charging and Letdown
High Head Safety Injection
Containment penetrations
Emergency Containment Coolers

8. Engineered Safety Features Walkdown (71710)

The inspector verified operability of the Auxiliary Feedwater Systems on 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 equipment conditions were satisfactory and items that might degrade performance were identified and evaluated (e.g. hangers and supports were operable, housekeeping, etc, was adequate);

c. with assistance from licensee personnel the interior of the breakers and electrical or instrumentation cabinets were inspected for debris, loose material, jumpers, evidence of rodents, etc.;

d. that instrumentation was properly valved in and functioning and calibration dates were appropriate;

e. that valves were in proper position, breaker alignment was correct, power was available, and valves were locked as required; and

f. local and remote position indication was compared, and remote instrumentation was functional.

During the AFW system walkdown, the inspector identified three nitrogen connections to the Unit 4, Train A, AFW supply header. The connections, located downstream of valves 4-141, 4-241 and 4-341, were not shown on any plant drawing and were not listed as part of Operating Procedure (OP) 7300.3, "Auxiliary Feedwater System - Operating Instructions". The connections were determined to be part of an infrequently used nitrogen purge and capping system which also supplies the Unit 4 moisture separator reheaters and feedwater heaters. The three connections to train A of the AFW system could be used to place a nitrogen cap on the three Unit 4 steam generators. The nitrogen system was operational. Numerous plant personnel contacted did not realize the nitrogen capping system was still connected to the AFW system. Most personnel believed it had been removed many years ago. Apparently the nitrogen capping connections were removed on Unit 3 but not on Unit 4. No drawing of the nitrogen system existed as a plant controlled document. However, in May 1984, a hand drawn sketch of the system was made. As of October 20, 1984, this sketch had not been used to create a drawing for use by plant personnel.

The nitrogen capping system apparently operates at 125 psi. The AFW system is a high pressure system which operates at up to 1005 psi. The three valves isolating the nitrogen supply system from the Unit 4 AFW system were found to be shut. The status of the valves during the preceding years can not be verified, however, the licensee believes that they have not been opened. The valves have now been danger tagged shut to prevent the possible overpressurization of the nitrogen piping during AFW system operation.

10 CFR 50, Appendix "B", quality assurance criterion V, requires that activities quality be precribed by documented instructions, procedures or drawings and be accomplished in accordance with those same instructions, procedures or drawings. The licensee Quality Assurance Topical Section 5.2 Revision 3 and Quality Procedures 5.1 Revision 3 and 6.6 Revision 4 implement these requirements.

Contrary to the above, the licensee created a condition adverse to quality by not establishing procedures, instructions or drawings describing approved methods of interfacing the nitrogen capping system with the Unit 4 AFW system. This is a violation affecting Unit 4 only (251/84-35-04).

Numerous other discrepancies were noted during the walkdown of the AFW system they are itemized below.

a. Two drain valves exist in the train "B" header on Unit 4 which are not identified on controlled drawing 5610-T-E-4062 Revision 25, Sheet 3 of 5. The drain valves are not identified in OP-7300.3 which is the procedure used to ensure what the AFW system is lined up for proper operation.

b. Numerous check valves, drain valves and orifices have no identification tags.

c. Drain valve 179 is incorrectly tagged as valve 190 and Drain valve 379 is incorrectly tagged as valve 192. The licensee is determining if valves 179, 279 and 379 are misidentified on drawing 5610-T-E-4062, Revision 25, sheet 3 of 5.

d. Valve 479 is not labeled on drawing 5610-T-E-4062, revision 25, sheet 3 of 5.

e. Valve 378 is missing its handwheel.

f. Valve AFFI 4-003 has a bent stem.

g. Valves 177, 277 and 377 are locked open but this fact is not identified on drawing 5610-T-E-4062, revision 25, sheet 3 of 5.

h. Additional isolation valves exist for pressure instruments 1429, 1430 and 1431 which are not shown on drawing 5610-T-E, Revision 25, sheet 3 of 5.

Correction of these discrepancies will be tracked as a followup item. (250/84-34-05 and 251/84-35-05).

9. Plant Events (93712)

An independent review of the following events was conducted:

Between October 3 and October 8, 1984, while performing maintenance procedure (MP) 0707.10, "Reactor Trip and Reactor Trip Bypass Breakers - Inspection and Maintenance" two unusual problems were discovered which affected the breaker manual closing mechanism. Four breakers on Unit 4 and one breaker on Unit 3 were found to have cracked braze joints on the manual closing mechanism support bracket. The discrepancies were evaluated by the manufacturer (Westinghouse) and the Plant Nuclear Safety Committee (PNSC) and both organizations determined that the braze joint cracks could not cause a problem which would prevent the breaker from opening. The braze joints will be repaired when Westinghouse provides a braze specification and a repair procedure per the agreement documented in a Westinghouse letter dated October 5, 1984.

The second discrepancy involved a failed bearing located on the manual closing mechanism for each of three separate breakers. The Unit 4B reactor trip bypass breaker, 4B reactor trip breaker and the Unit 3B reactor trip bypass breaker were affected. The problem was discovered during manual testing in accordance with MP 0707.10. After repeatedly cycling the breaker it unexpectedly could not be reclosed. A search for the problem revealed that a bearing on the manual opening mechanism had come off its shaft. The bearing was not required to be examined by MP-0707.10. Each reactor trip and trip bypass breaker was examined to evaluate the bearings. One was found to be frozen to its shaft. Another was found to be ready to fall apart. The three damaged bearings were replaced. The manufacturer, Westinghouse, has visited the site and obtained samples of the damaged bearings. An evaluation is in progress. The principle concern is that the failure can result in tiny parts from the bearing falling into other breaker components and preventing automatic opening when required. The licensee replaced all the bearings in each trip and trip bypass breaker closing mechanism as a precautionary measure. Additional actions may be taken when the Westinghouse evaluation is completed.

On October 9, 1984, the Unit 4A inverter de-energized for no apparent reason. The loss of various vital instrument channels resulted. A reactor trip signal was received from the loss of power to the source and intermediate range nuclear instruments. The reactor was not critical at the time. All control rods were fully inserted. Primary temperature was 180°F and pressure was 380 psig. The primary system was solid. PORV-4-455C opened, as designed, on loss of power while operating in the overpressure mitigating system mode. Reactor pressure quickly dropped to 50 psig requiring all reactor coolant pumps to be secured. Decay heat removal via the RHR system was implemented and the unit was stabilized.

An effort was made to restore instrument power by switching to the "AS" spare inverter which can be shared between Unit 3 Buss A and Unit 4 Buss A. When the manual bus transfer was made the spare inverter output breakers tripped and the inverter blew internal fuses. At essentially the same time, a spike was received on the Unit 3A inverter which resulted in a 70% reactor runback due to a voltage fluctuation or power range nuclear instrument N-42.

The "A" spare inverter was eventually energized feeding Unit 4 by closing its input breakers with all output breakers open and then sequentially restoring the vital loads. The 4A inverter had failed in a similar manner on September 20, 1984. The inverter has received numerous repairs since April, 1984. As a result of the problems of September 20, a technical representative identified an arcing diode n a buffer amplifier. The diode was replaced and 4A inverter successfully run on a test load. The 4A inverter was placed in service on October 8, 1984, apparently without the specific approval of the elect ical maintenance department. It functioned for seven hours before failing, as described, on October 9, 1984.

Additional troubleshooting has been performed. On October 9, a grounded buffer amplifier card was found and replaced. On October 12, an improperly wired current limiting circuit was identified and corrected. Special testing procedures have been developed and implemented to identify problem areas. Temporary procedures have been developed to enable shifting to the spare inverter without causing the spare inverter to trip. The licensee, in conjunction with the manufacturer's technical representatives is continuing the evaluation.

10. Independent Inspection Effort (92706)

During the reporting period, the inspectors routinely attended meetings with licensee management and shift turnovers between shift supervisors, shift foremen and licensed operators. These meetings and discussions provided a daily status of plant operating and testing activities in progress as well as a discussion of significant problems or incidents.

On October 2, 1984, the inspector observed the hoisting of the 4B residual heat removal pump with a chain fall which had not been recently weight tested. A review was made of the licensee heavy load handling program to determine its adequacy.

The Florida Power and Light Company Quality Assurance Program is designed to meet the requirements of the Regulatory Guides and industry standards itemized in Appendix "C" of the Topical Quality Assurance Report. One of the standard included in Appendix "C", is ANSI N45.2.2-1972, "Packaging, Shipping, Receiving, Storage and Handling of Items for Nuclear Power Plant." Section 7.4 of ANSI N45.2.2 itemizes requirements for inspection of equipment and rigging. This section requires that an inspection program be established for handling equipment and rigging, including methods of identifying acceptable and nonconforming items. Periodic inspections are required to be supplemented with special visual and non-destructive examinations and dynamic load tests.

On page 16 of Appendix "C" of the Topical Quality Assurance Report, the licensee qualifies its commitment to ANSI N45.2.2-1972. The qualification states, in part:

> "In lieu of having a program of periodic, documented inspections of rigging and handling equipment, FPL's practice is to have the individual user determine the equipments' acceptability prior to each use. This prior-to-use inspection is exactly the same as that required during periodic inspections, and uses criteria identified in ANSI N45.2.2-1972, paragraph 7.4. This practice also precludes the need for a system to indicate the acceptability of rigging and handling equipment.

The inspector concluded that chain falls used to hoist safety-related equipment are to be inspected prior-to-use.

The prior-to-use inspection is to be exactly the same as the periodic inspection identified in section 7.4 of ANSI N45.2.2-1972. The inspector determined that the licensee does not have a procedure or instruction itemizing periodic inspection requirements of section 7.4 of ANSI N45.2.2-1972. Consequently no guidance exists for an individual to determine that he has made all required prior-to-use inspections and verifications. One of the required verifications, listed in section 7.4.2, requires that hoisting equipment that does not meet manufacturers' specifications shall not be used. The licensee does not verify, either periodically (as per ANSI N45.2.2) or prior-to-use (as per Appendix "C" of the Topical Quality Assurance Report) that chain falls conform with manufacturer's standards. A realistic assessment of conformance would require special visual inspections, non-destructive examinations and dynamic load tests which are too extensive to be performed prior to each use of a given chain fall.

Conversations with the licensee revealed that the licensee was aware of the need to improve the handling equipment testing and inspection program. The licensee is developing an improved program. The inspector noted that ANSI/ASME N45.2.15-1981, "Hoisting, Rigging, and Transporting of Items for Nuclear Power Plants" could be used by the licensee in developing their program. The adequacy of the hoisting and rigging program is an inspector followup item (250/84-34-06 and 251/84-35-06).

Between October 1 and October 4, 1984, the inspector observed health physics practices associated with the repair of the 4B residual heat removal pump. Some poor practices were observed which prompted a review of the radiation work permit (RWP) program as it relates to health physics technicians.

Operating procedure (OP) 11550.2, "Radiation Rules of Practice", requires in Section 8.3.1 that all persons working with radioactive material where contamination of the person is possible shall wear protective clothing appropriate to the work involved, as stated in the RWP. During the repair of the 4B RHR pump the applicable RWP required full anti-contamination clothing, beta glasses and a respirator during grinding. A maintenance technician grinding on a pump shaft weld was conforming to these requirements. However, a health physics (HP) technician standing next to the maintenance technician was wearing no respirator, no hood and a lab coat instead of anti-contamination coveralls. When asked about the obvious discrepancy, the HP technician said that he was allowed to assign himself less restrictive clothing requirements per OP 11550.2 section 8.3.3, which states:

> "For jobs requiring a Radiation Work Permit (RWP), the protective clothing requirements for the job shall be specified on the RWP. Personnel entering RWP area to perform observation and inspection activities only, may wear less than the RWP clothing requirements if so directed by Health Physics."

A conversation with the Health Physics Supervisor confirmed the HP technicians routinely provide HP support to maintenance technicians without conforming to the protective dress requirements of the maintenance RWP. He also stated that the HP support is provided under a standing RWP (#2) which is general in nature and allows access to areas to perform surveillances. The inspectors concerns are follows:

a. Apparently any HP technicians can deviate from the RWP protective dress requirements without consulting supervisory personnel. This includes the decision not to wear a respirator in an area of potential airborne contamination.

b. HP technicians directly supporting a maintenance repair task do not consider themselves as subject to the RWP requirements of that task. They assume that their support function of decontamination and cleanup constitutes only an area surveillance and therefore they consider themselves only subject to the requirements of RWP #2.

These concerns constitute an inspector followup item (250/84-34-07 and 251/84-35-07).

During the repair of the 4B RHR pump, a large cement access cover was removed above the pump housing. This access area is used by the intake area crane to remove the pump. Removing the cement slab constitutes a breach of the RHR pump room locked high radiation area boundary. Any individual could enter the locked high radiation area in a matter of seconds by climbing down readily accessible supports. During the 4B RHR pump repair, the cement slab was removed for several days. The access hole was only posted with warning signs and tape. No physical locked barrier existed to prevent entry. Then informed of this discrepancy the Health Physics Supervisor responded that the temporary access did not require a locked barrier because it was not a normal entrance to the locked high radiation area. The inspector requested that a HP technician to be posted at the access to prevent entry until a locked physical barricade could be installed. Surveys subsequently performed by the licensee and confirmed by the inspector revealed that the radiation levels existing in the RHR pump room during the time the cement slab was removed did not require the area to be locked. Evidently, the area is only required to be locked during certain evolutions such as resin discharge. Since the area did not meet the requirements of a locked high radiation area during the period in question, this occurrence does not constitute a violation.

During the HP support of the 4B RHR pump repair. the inspector observed the following discrepancies which constitute poor HP practices:

a. The inspector informed HP supervisory personnel that a large storage can of contaminated material was overturned inside the work area. Corrective action was not taken until 4 hours later when the inspector again questioned supervisory personnel about the matter.

b. A sheet of protective material used to catch debris from grinding on the pump was contaminated to 100,000 cpm. After grinding was completed, the material and its loose contamination were left overnight exposed to the atmosphere. Cleanup did not take place until the next day. The windy outdoors environment could have spread the contamination outside the work boundary.

c. The work area was excessively cluttered with previously contaminated items such as a quantity of ladders, mops, piping and a pipe cutting machine. The items were being stored in the area until decontamination could be arranged. The clutter hindered the ability to work on the pump.