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

Report Nos.: 50-250/86-41 and 50-251/86-41

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: October 14 - November 3, 1986

Inspectors: _____ 11/25/86
for D. R. Brewer, Senior Resident Inspector Date Signed

_____ 11/25/86
for K. W. Van Dyne, Resident Inspector Date Signed

_____ 11/25/86
for J. B. MacDonald, Resident Inspector Date Signed

Approved by: _____ 11/25/86
for S. A. Elrod, Section Chief Date Signed
Division of Reactor Projects

SUMMARY

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

Results: Violations - Failure to meet the requirements of Technical Specification (TS) 3.3.3 and TS 6.8.1 in that otherwise adequate procedures were not properly implemented (paragraph 6).

REPORT DETAILS

1. Persons Contacted

Licensee Employees

*C. M. Wethy, Vice President - Turkey Point
*C. J. Baker, Plant Manager-Nuclear - Turkey Point
F. H. Southworth, Senior Technical Advisor
E. Preast, Site Engineering Manager (SEM)
*D. D. Grandage, Operations Superintendent and Acting Plant Manager
T. A. Finn, Operations Supervisor
J. Crockford, Operations Enhancement Coordinator
J. Webb, Operations - Maintenance Coordinator
*J. W. Kappes, Maintenance Superintendent - Nuclear
R. A. Longtemps, Mechanical Maintenance Department Supervisor
D. Tomasewski, Instrument and Control (IC) Department Supervisor
J. C. Strong, Electrical Department Supervisor
W. Bladow, Quality Assurance (QA) Superintendent
M. J. Crisler, Quality Control (QC) Supervisor
J. A. Labarraque, Technical Department Supervisor
R. G. Mende, Reactor Engineering Supervisor
*J. Arias, Regulation and Compliance Supervisor
*R. Hart, Regulation and Compliance Engineer
W. C. Miller, Training Supervisor
P. W. Hughes, Health Physics Supervisor
G. Solomon, Regulation and Compliance Engineer
J. Donis, Engineering Department Supervisor
J. J. Zudans, Nuclear Engineering, Human Factors Performance
D. W. Haase, Safety Engineering Group (SEG) Chairman
M. J. Powell, Training Department
*B. A. Abrishami, System Performance Supervisor
*J. Anderson, Quality Assurance Supervisor
*R. H. Reinhardt, Quality Control Supervisor

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

*Attended exit interview on November 5, 1986.

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 November 5, 1986. The areas requiring management attention were reviewed.

One violation was identified:

Failure to meet the requirements of TS 3.3.3 and TS 6.8.1 in that, while operating in a mode in which containment integrity was required, 11 valves inside containment, including two containment isolation valves, had to be declared out of service because no acceptance testing was accomplished after maintenance had been performed on the valves (paragraph 6) (251/86-41-01).

Two Unresolved Items (URI) were identified:

Evaluate the adequacy of the licensee's procedure for calculating estimated critical conditions (ECC) (paragraph 7) (250,251/86-41-03).

Evaluate the results of ongoing reviews of deficiencies in the Radiation Controlled Area Training (RCAT) Program (paragraph 8) (250,251/86-41-02).

The licensee acknowledged the inspector's comments and did not identify as proprietary any of the materials provided to or reviewed by the inspectors during this inspection.

3. 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. Two unresolved items identified during this inspection are discussed in paragraphs 7 and 8.

4. Inspection and Enforcement Bulletin (IEB) Followup (92703)

(Open) IEB 85-03. The inspectors have reviewed the initial licensee response and subsequent correspondence regarding IEB 85-03, Motor Operated Common Mode Failures During Plant Transients Due To Improper Switch Settings. The purpose of this bulletin is to request licensees to develop and implement a program to ensure that switch settings on safety related high pressure coolant injection and emergency feedwater system motor operated valves are selected, set and maintained correctly to accommodate the maximum differential pressures (DP) expected on these valves during both normal and abnormal events within the design basis. The bulletin required licensees to address five actions: Review and document the design basis for operation of each valve, including maximum DP expected during opening and closing of the valve for both normal and abnormal events as defined in the design basis: Based on these results, establish the correct switch settings: Change individual valve settings, as appropriate: Prepare or revise procedures to ensure that correct switch settings are determined and maintained throughout the life of the plant: Submit a written report to the NRC within 180 days of the date of the bulletin that; reports the results of the review and contains the program schedule to accomplish the above actions. The licensee should complete the program as soon as practical and within two years of the date of the bulletin: Provide a written report on completion of the program that includes, verification of completion, summary of the findings of valve operability prior to any adjustments as a result of the bulletin and a summary of the data.

The NRC considered the licensee's initial response, letter L-86-226 dated May 28, 1986, inadequate because it did not contain the description of the program to assure valve operability. A phone conversation between Region II and FPL staff concerning the initial bulletin response was held on September 9, 1986. At that time, the following information was requested:

a. How will the torque switch settings be determined and what will be the responsible discipline?
b. How are the DP values determined?
c. How will the testing be accomplished?
d. What is the justification if full DP testing is not performed?

On October 10, 1986, the licensee informed Region II by letter L-86-408 that the completion date of October 17, 1986, stated in letter L-86-226, for establishing actual torque switch settings that will be calculated and maintained in a control document would be delayed until January 30, 1987. The necessary vendor thrust calculations had not been received by the licensee. The inspectors will continue to track the progress of the program established in accordance with IEB 85-03.

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

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

- Nuclear Chemistry Procedure NC-23A, Operation of the Post Accident Sampling System (PASS) For Reactor Coolant (RCS) During Non-Emergency Conditions
- Preoperational Test POP 0800.110, AFW Pumps RPM Monitoring (A pump)
- Surveillance Procedure 4-OSP-075.1, AFW Train 1 Operability Verification.

No violations or deviations were identified within the areas inspected.

6. 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 trouble-shooting 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. On October 25, 1986, with Unit 3 at 100% power and Unit 4 in Mode 4 (Hot Shutdown) the B emergency diesel generator (EDG) trouble alarm annunciated and the ready to start light went out. The cause of the alarm was determined to be low lube oil temperature. The local low lube oil temperature alarm annunciates at 100°F. The actual lube oil temperature was approximately 99°F. The nuclear watch engineer (NWE) was dispatched to the B EDG and found no visible signs of equipment damage or lube oil leaks and reported lube oil temperatures to be holding almost steady just below the 100°F alarm setpoint. The plant supervisor nuclear (PSN), on shift, reviewed the EDG logic diagrams and determined that the low lube oil temperature alarm would not prevent the EDG from auto starting. Based on this information the B EDG was considered to be operable and remained in service. Five hours later the B EDG was declared out of service to perform troubleshooting and repairs. The apparent cause of the low lube oil temperature alarm was the failure of the immersion heater contractor coil caused by a blown fuse. Both were replaced, the heater was tested satisfactorily and the B EDG was declared back in service.

   On October 27, 1986, again the B EDG trouble alarm annunciated as a result of low lube oil temperature. The EDG was declared out of service. Troubleshooting revealed a nick in the insulation of the wiring for the immersion heater, causing an intermittant ground in the circuit. The damaged wire was replaced, the heater was tested satisfactorily and the B EDG was declared back in service.

Of particular concern to the inspectors was that at the time of the low lube oil temperature alarm on October 25, the PSN made the determination that the B EDG should still be considered in service based on the fact that the alarm did not block the auto start logic. The low lube oil temperature alarm is to alert the operator that a condition is being approached in which the EDG may not be able to perform its intended emergency function of attaining rated speed and load in less than 15 seconds as a result of an insufficiently warmed lube oil system and engine. The fact that the auto start logic was not blocked by the alarm was not an indication that the EDG was actually in an operable condition. Furthermore, the auto start logic is unaffected by many parameters which do affect EDG operability.

Subsequent to the decision to keep the EDG in service, documentation from the vendor was produced by licensing personnel which stated that the lube oil temperature switch can be set as low as 85°F and the EDG can still be considered in service. The central issue in this matter is the philosophy that equipment, especially safety-related components, in Nuclear Power plants must be proven operable to be considered in service. If not proven operable, either by surveillance testing that remains unchallenged by component failure or by adequate engineering justification, they must be considered out of service. The inspectors expressed the concern to plant management that PSNs should seek the assistance of other plant disciplines when equipment operability is concerned. As a result, in part, of these discussions, Operations memo PTN-OPS-86-210 was written to all on shift licensed senior reactor operators directing them to contact the Operations Supervisor, Operations Superintendent and Licensing prior to making a decision on equipment operability.

b. On October 27, 1986, while Unit 4 was in Mode 3 (hot standby) in preparation for reactor startup, operating personnel discovered that the required post-maintenance testing for numerous valves inside containment had not been performed. The valves, primarily from the in-service inspection and testing (IST) Program, included two containment isolation valves, which were subsequently declared out of service. The licensee commenced an immediate unit cooldown per TS 3.3.3 and made preparations to test all affected valves. In addition, Motor Operated Valve (MOV)-751, which is in the RHR pump suction line, required plant depressurization to less than 525 psig for testing.

Technical Specification (TS) 3.3.3 requires that the containment isolation valves for Phase A containment isolation, Phase B containment isolation and containment ventilation isolation shall be operable with the isolation times of each power operated or automatic valves within the limits established for testing in accordance with Section XI of ASME Boiler and Pressure Vessel code and applicable Addenda as required by 10 CFR 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i).

Article IWV-3200 of ASME Boiler and Pressure Vessel code Section XI, entitled Valve Replacement, Repair, and Maintenance, specifies that when a valve has undergone maintenance which could affect its performance it shall be tested to demonstrate that the performance parameters which could be affected by the maintenance are within acceptable limits prior to the time it is returned to service. Footnote 1 further defines adjustment of stem packing as maintenance that could affect valve performance parameters.

TS 6.8.1 requires that written procedures and administrative policies shall be implemented that meet or exceed the requirements of section 5.1 and 5.3 of ANSI N18.7-1972 and Appendix A of USNRC Regulatory Guide 1.33.

Administrative Procedure (AP) 0190.28, Post Maintenance Testing, requires, in part, that:

(1) Prior to commencing work on valves in the IST program post-maintenance testing shall be determined and listed on the appropriate test sheet.

(2) Prior to commencing work on valves in the IST program the IST Coordinator or Shift Technical Advisor (STA) shall be notified to verify the adequacy of post-maintenance testing and to ensure proper test planning and scheduling. The IST Coordinator or STA shall also complete the appropriate sections of the periodic test procedure to perform testing as specified.

(3) IST valves contained in the IST Program list that are underlined require Plant Supervisor-Nuclear and/or IST Coordinator review prior to performing any maintenance, including packing adjustment, when performed under any conditions other than cold or refueling shutdown.

Operating Procedure (OP) 0209.1, Valve Exercising Procedure, requires, in part, that:

(1) Whenever an IST valve is to be replaced, repaired, or undergo maintenance which could affect its performance it shall be tested. The test shall demonstrate that the performance parameters which could be affected by the replacement, repair, or maintenance are within acceptable limits. The test shall be performed prior to returning the valve to service.

(2) The Plant Supervisor-Nuclear shall ensure that a completed OP 0209.1 Appendix P test sheet showing the valve has passed all of its required testing necessary to prove valve operability before returning the valve to service.

On October 23, 1986, Unit 4 was shutdown to locate and repair a suspected condenser tube leak. Operating personnel maintained the unit in mode 4 (hot shutdown). While in this condition, maintenance personnel entered containment to identify and repair valve packing leaks. Eleven valves, which were determined to have excessive packing gland leakage, were subsequently adjusted to eliminate the leakage. All but one were IST valves and two of the remaining ten valves were containment isolation valves. Contrary to the above requirements, the PSN did sign the plant work orders to authorize maintenance to begin, however, the affected IST valves apparently received no review by the PSN and/or IST Coordinator prior to performing packing adjustments even though Unit 4 was not in cold or refueling shutdown. Neither the IST Coordinator nor the STA were notified prior to commencing work on valves in the IST program so that they may verify the adequacy of post-maintenance testing and ensure proper test planning and scheduling. Because operating personnel were apparently unaware that containment isolation valves, as well as eight other IST valves, had undergone maintenance and should have been declared out of service, the requirement of TS 3.3.3 was exceeded and none of the associated LCOs were met. The above discrepancies, together, constitute a violation (251/86-41-01).

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 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:

Emergency Diesel Generators
Auxiliary Feedwater
Control Room Vertical Panels and Safeguards Racks
Unit 4 Containment Spray System
Unit 3 and Unit 4 Component Cooling Water Systems

On October 28, 1986, while performing a reactor startup, operating personnel realized that criticality would be reached prior to control rods being withdrawn above their insertion limit. The rod insertion limit for control rods in bank C is 107 steps at hot zero power operation and estimates derived from the 1/m plot predicted criticality with C bank at 100 steps. The licensee suspected an error in the primary chemistry analysis method in which boron concentration is determined. Control rods were reinserted, estimated critical conditions (ECC) were checked and boron concentration was verified with no apparent discrepancies. Subsequently, another startup was commenced and again, resulted in a predicted critical rod position that deviated excessively from the ECC. On the third attempt criticality was achieved 730 pcm earlier than calculated from the ECC. The inspectors are concerned with the apparent inaccuracies in calculating ECCs and are following the licensee's investigation into the matter. Consequently, the evaluation of OP 1009.1, Estimated Critical Conditions, to accurately determine ECCs constitutes an unresolved item (250,251/86-41-03).

No violations or deviations were identified in the areas inspected.

8. Radiation Protection: Training and Qualifications (83523)

On October 15, 1986, a Plant Turkey Point (PTP) Training Department instructor improperly graded the Radiation Controlled Area (RCA) requalification test of a non-licensed operator, allowing the test score to be recorded as a passing score when, in reality, the operator had failed the test. Due to the serious nature of this occurrence, an immediate independent review of the Radiation Controlled Area Training (RCAT) Program was initiated by QA personnel at the Plant Manager's request.

The RCAT consists of three sub-tests:

a. General Employee Training (GET) - plant orientation - 50 multiple choice questions

b. INPO - general health physics practices - 50 multiple choice questions

c. Site Specific Training (SST) - PTP health physics practices - 14 multiple choice questions and 6 short answer questions

The passing score for each section is 80%.

Initial QA review of 129 tests with scores of 80% revealed that 54 had been graded improperly, and when regraded, the correct scores were failing scores of less than 80%. Unescorted access to the RCA was immediately denied to the 54 individuals involved and a complete QA review of all current RCAT tests at PTP was initiated. A total of 2815 RCAT tests were regraded at PTP, and 294 individuals were denied unescorted RCA access as a result of having one or more sub-test scores downgraded from a passing score to a failing score.

Similar review actions were taken at St. Lucie Plant to determine if RCAT programmatic deficiencies existed there as well.

FPL's Security personnel also have been conducting an independent investigation of grading practices within the RCAT program. The results of the investigation revealed the following:

a. The instructor involved in the October 15, 1986 event stated that he had passed individuals who had actually failed the test on more than one occasion.

b. Instructors would routinely screen answer sheets to detect potential failures and return the test to the testee to allow the individual to rethink certain answers.

c. No evidence presently exists that indicates these practices exist outside the RCAT program.

d. There is no indication that PTP plant management expected or instructed the grading improprieties, or were aware of the practices.

The results of the QA test regrades, statistical analysis, investigation, root cause and proposed corrective actions are summarized in FPL letter PTN-TRNG-86-617.

The inspectors have audited a sample population of the RCAT tests involved in the regrading process. Specific areas of concern are: actual QA independence, the accuracy of the regrading, trends which may indicate programmatic and/or mechanical deficiencies and indications that possible improper instructor assistance had been given to enable an examinee to improve his score from failing to passing. Due to the significance of this issue and the need to properly and completely review the data compiled, conclusions, corrective actions and implementation of the corrective actions this issue will be identified as an unresolved item (250,251/86-41-02).

No violations or deviations were identified in the areas inspected.

9. Engineered Safety Features Walkdown (71710)

The inspectors verified operability of the Unit 4 Containment Spray (CS) System by performing a complete walkdown of all accessible equipment. The following criteria were used, as appropriate, during the walkdown:

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

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

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

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

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

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

Several concerns were noted to the licensee management:

a. Details 1 and 2 of drawing 5610-T-E-4510 sheet 1 of 2 do not accurately represent the field condition.

b. Valve label plates were found on the floor.

c. Tygon tubing attached to valve 4-945-C, apparently used as a drain line, was still attached and contained some fluid.

d. Various items, including equipment such as a builder's square and metal rods and trash, were strewn about in the CS room.

e. A Quality Control sticker from valve 3-896T was found on the floor.

f. In general, the Unit 4 CS System room was dirty. Trash and loose articles were found in the radiation controlled area. There appeared to be several boric acid leaks from various valves and piping.

No violations or deviations were identified in the areas inspected.

10. Plant Events (93702)

An independent review was conducted of the following events.

On October 23, 1986, Unit 4 was shutdown to locate and repair a suspected condenser tube leak. All steam generator conductivity meters in the control room had pegged high off scale and chemical analysis verified excessive conductivity and chloride levels in the condenser and the steam generators, indicative of gross tube leakage. Unit 4 was shutdown while an investigation commenced to determine the root cause of the condenser inleakage. Based on the chloride concentration and conductivity levels in the secondary system the licensee calculated that there was approximately 600 gallons of circulating water inleakage. Because of the severity of the chloride contamination, an extensive clean up of the condensate and feedwater systems was initiated. This was hampered by a shortage of water meeting the quality standards for the secondary system. The water treatment plant onsite which

provides such water was in service only intermittently and resin type ion exchangers were trucked to the site to provide adequate water. Several leak detection methods were used in an attempt to locate the leak, including the use of a helium sniffer, soaping of the water box tube sheet, and hydrostatic testing of the hotwell side of the condenser tubes. All were unsuccessful in determining the source of inleakage. On October 31, 1986, the unit was returned to service and the licensee continues to investigate other possible flowpaths of inleakage to the condenser.