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

Report Nos.: 50-250/84-28 and 50-251/84-29

Licensee: Florida Power and Light Company
9250 West Flagler Street
Miami, FL 33101

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

Facility Name: Turkey Point 3 and 4

Inspection Conducted: August 17 - September 27, 1984

Inspector: Virgil L. Brownlee for 10/19/84
T. A. Peebles, Senior Resident Inspector Date Signed

Accompanying Personnel: D. R. Brewer

Approved by: Virgil L. Brownlee for 10/19/84
S. A. Elrod, Section Chief Date Signed
Division of Reactor Projects

SUMMARY

Scope: This routine, unannounced inspection involved 213 inspection hours on site, including 48 hours of backshift, in the areas of licensee action on previous inspection findings, LER followup, annual and monthly surveillance, monthly and refueling maintenance, operational safety, engineering safety features walkdown, plant events, and independent inspection.

Results: Of the eight areas inspected, no violations or deviations were identified in five areas; three violations were identified in three areas (paragraph 8, failure to retain an operating record; paragraph 10, failure to follow the post trip review procedure; paragraph 11, failure to establish an adequate startup procedure).

REPORT DETAILS

1. Licensee Employees Contacted

*K. N. Harris, Vice President-Turkey Point
*J. A. Labarraque, Tech. Dept. Supt. and Acting Plant Manager
C. J. Baker, Plant Manager Nuclear
*D. W. Haase, Chairman Safety Engineer Group
J. P. Mendietta, Service Manager-Nuclear
*D. D. Grandage, Operations Superintendent Nuclear
*J. W. Kappes, Maintenance Superintendent Nuclear
*T. A. Finn, Operations Supervisor
*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, Startup Superintendent
*E. A. Suarez, Technical Department Supervisor
*W. R. Williams, Assistant Superintendent Electrical Maintenance
*J. Arias, Jr., Regulation and Compliance Engineer
*R. F. Englmeier, Corporate QA Manager
*R. H. Reinhardt, Site QC
*F. A. Houtz, Site QC
*J. M. Donis, Site Resident Engineer
*J. M. Mowbray, Site Engineer
*P. J. Baum, Training Supervisor
*V. A. Kaminskas, Reactor Engineer Supervisor
*B. A. Abrishami, IST Supervisor
*R. G. Mende, Reactor Engineer
*R. M. Brown, HP Supervisor
*J. B. Harper, QA Corporate
D. Tomaszewski, Plant Engineer Supervisor
R. E. Garrett, Plant Security Supervisor
J. E. Moaba, Corporate Licensing
G. J. Boissy, PEP Program Manager

Other licensee employees contacted included construction craftsmen, 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.

Exit meetings were held on September 7 and 14, 1984, with the persons noted above. The areas requiring management attention were reviewed, including: failure to establish adequate startup procedures, with the inadequate Estimated Critical Condition (ECC) and 1/M guidance being examples (250/84-28-02 and 251/84-29-03). The licensee acknowledged the findings.

An exit was held with the plant manager on October 2, 1984, to discuss the reactor trip of September 20, 1984, and the finding, which was a failure to follow the Post Trip Review procedure, and failure to evaluate the Reactor Trip properly before returning to power operation (250/84-29-02). The licensee acknowledged the finding.

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, except for a training assessment item which has been revised in scope and schedule. The region has been updated and agrees with the progress.

      The procedures upgrade project is meeting their timetable, but it is recognized that improvement is needed in the final versions. The loss of 120 V AC procedures are being re-revised to incorporate further comments from the plant and the requirements of NRC Bulletin 79-27.

      The site facility upgrade is progressing on schedule with the preliminary fill for the administrative building begun and the relocation of necessary facilities completed for the HP facility.

      The standard Technical Specification upgrade project has completed the final job scope and it was approved. The budget and work schedule were approved. The coordinator is on site and two other site personnel have arrived. They are beginning the individual job task statements.

   b. (Open) IFI 50-250/84-23-07 and 50-251/84-24-07. The re-submittal of several sections of the Inservice Test program and associated Technical Specifications remain outstanding.

   c. (Closed) IFI 250/84-23-06 and 50-251/84-24-06. A problem with containment isolation valve, CV-4-2907, possibly due to inadequate control of construction activities has been resolved. The licensee has assigned construction area coordinators to designated areas to control the work and report to the Plant Supervisor Nuclear (PSN). Further information is in paragraph 7.

4. Unresolved Items*

   Unresolved items were not identified during this inspection.

5. Licensee Event Report (LER) Followup (92700)

   The following LER was reviewed and closed. The inspector verified that: reporting requirements had been met; causes had been identified; corrective actions appeared appropriate; generic applicability had been considered; and the LER forms were complete. A more detailed review was then performed to verify that: the licensee had reviewed the event; corrective action had been taken; no unreviewed safety questions were involved; and violation of regulations or Technical Specification conditions had been identified.

   (Closed) LER 250/84-23. On August 22, 1984, while Unit 3 was at 100% power, a turbine runback occurred due to a dropped shutdown control bank A rod, J3. Initially, a fuse was blown but replacement indicated further problems which were traced to cabling inside containment. The unit operated from 9:03 a.m. to 8:22 p.m. at reduced power, complying with the surveillances and Limiting Conditions of Operation (LCO) of the Technical Specification. A loose cable connector was found and replaced and the unit placed on line at 2:30 p.m. on August 23, 1984.

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

   The inspectors observed Technical Specification 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 IX and the Technical Specification.

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

   Refueling water storage tank level channel calibration
   Residual Heat Removal (RHR) pump IST/operability test
   High Head Safety Injection (HHSI) pump IST/operability test
   Nuclear instrument power range calibration
   Protection channel Tavg-delta T calibration (3 days late)
   Intake Cooling Water IST

*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.

The emergency diesel generator test which was run on the A diesel on September 6, used OP 4304.1 which had not been modified to include the data taker's name and reviewer's name and acceptance criteria as cited on August 3, 1984, in Inspection Report 250/84-23 and 251/84-24. The licensee has since modified the procedure.

The RHR and HHSI pump ISTs were recently modified to include surveillance of support systems. As a result of the tests, one of the RHR pumps and two of the HHSI pumps required increased surveillance as the seal water seals were leaking in excess of recommended. The IST program surveillances and inadequate testing for operability was a violation (250/84-23-02 and 251/84-24-02) and will be followed under these numbers.

The Intake Cooling Water IST was not adequate to evaluate the operability of the system and is an example of a potential violation and is addressed in special Inspection Report 250/84-29 and 251/84-30.

No violations or deviations are addressed in this paragraph.

7. 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:

- 3A main feedwater pump drain valve - replacement
- 4C Intake Cooling Water (ICW) pump - replacement
- A-B-C Auxiliary feedwater pump throttle linkage - replacement
- Emergency Diesel Generator fuel oil leak - repair
- Instrument air, containment isolation valve, manual reach rod - repair
- Charging pumps - repairs
- Pressurizer level indicator, in Unit 4 chg. pump room - replacement
- Shutdown control rod J3 on Unit 3 - repair of cable
- Air line on CV-4-2907, Emergency Containment Cooler 4C, component cooling return - repair

Broken air lines on CV-4-2907 were noted in report 84-23 and 84-24. As the air lines on this valve were found broken on July 24 and on September 5, 1984, the licensee was requested to evaluate the problem. This will be an IFI (251/84-29-04).

(Closed) IFI (250/84-23-06 and 251/84-24-06) This was a followup on the same valve and referenced control of construction forces. The licensee has assigned construction area coordinators to control the work and report to the Plant Supervisor Nuclear (PSN).

No violations or deviations were identified.

8. 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 LCOs 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 water structure, auxiliary, diesel, 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 4 to verify operability and proper valve alignment:

Emergency diesel generators
High head safety injection systems
Containment spray systems
Auxiliary feedwater systems

On August 22, 1984, the inspectors reviewed the Plant Curve Book. It was determined that the charts, graphs and curves that comprise the Plant Curve Book were not being retained as an operating record after supersession. No archival copy of any pages from the book were available in the document control system.

The licensee did not consider the Plant Curve Book to constitute an operating record. However, the various pages of the book are used extensively by control room personnel to determine whether or not existing plant parameters meet Technical Specification requirements. For example, the moderator temperature coefficient (MTC) control graph is used by control room operators to verify, in accordance with TS 3.1-2, that the MTC is not positive prior to increasing reactor power above 70 percent. This curve was violated on June 12, 1984 (LER 251-84-12 applies). The licensee subsequently determined that the MTC graph was overly conservative and created a new graph which was substituted into the Plant Curve Book. The original graph was discarded without retaining an archival copy. Subsequent followup on the LER was impeded because a copy of the original curve could not be found by the licensee.

Numerous curves retained in the Plant Curve Book are used as the principle means of ensuring compliance with Technical Specification requirements and therefore, constitute operating records. Failure to retain an archival copy of the MTC curve constitutes a failure to retain an operating record and is a violation (250/84-28-02 and 251/84-29-03).

9. Engineered Safety Features Walkdown (71710)

The inspector verified operability of the Component Cooling Water and Intake Cooling Water systems 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. The 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, that 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 valve position indication was compared, and remote instrumentation was functional.

Several potential violations of the above criteria were found and are documented in a special Inspection Report 250/84-29 and 251/84-30.

10. Plant Events (93702)

An independent review of the following events were conducted:

On August 22, 1984, while Unit 3 was at 100% power, a turbine runback occurred due to a dropped shutdown control bank A rod, J3. Initially, a fuse was blown but replacement indicated further problems which were traced to cabling inside containment. The unit operated from 9:03 a.m. to 8:22 p.m. at reduced power complying with the surveillances and LCO of the TS. A loose cable connector was found and replaced and the unit placed on line at 2:30 p.m. on August 23, 1984.

On August 29, 1984, at 8:50 p.m. the licensee discovered that the 4B Intake Cooling Water (ICW) strainer had been removed from service since 11:15 a.m. on August 24, 1984. This removed from service, one of two headers of ICW which is only allowed to be removed for 24 hours. This violation of a TS LCO was investigated and is reported in Inspection Reports 250/84-29 and 251/84-30.

On September 14, 1984, an UNUSUAL EVENT was declared when a potential loss of containment integrity was discovered. The event began during the dayshift when a surveillance test was performed which cycled the two series service air containment isolation valve handwheels. One of the handwheels was thought to be slipping and a maintenance request was written. At 8:40 p.m., maintenance personnel informed the PSN that the inboard containment isolation valve, V-4-204, was found in the open position. At 8:55 p.m., an operator verified the valve to be open and closed it; the outboard valve was verified to be closed. This condition is not permitted by the TS and the licensee promptly reported it. The personnel performing the surveillance did not believe the valve to have opened and did not request verification of the valve position.

On September 20, 1984, Unit 4 tripped due to loss of the 4A instrument bus inverter. The inverter deenergized as a result of blown fuses. Loss of inverter 4A resulted in loss of power to vital AC instrument bus 4P07. This resulted in a turbine runback due to receipt of a dropped rod signal from power range nuclear instrument N-42 as the instrument lost power. After a 30% runback, the reactor tripped due to the receipt of a low level signal in "B" steam generator in conjunction with steam flow exceeding feed flow.

When Operations attempted to restore power to the bus from the standby inverter it also deenergized due to blown fuses. It was successfully reenergized after reducing the inverter loads. Fuses were replaced in inverter 4A and the reactor was restarted. Subsequent to the reactor restart, the inspector reviewed Appendix "A" of Off Normal Operating Procedure (ONOP) 0208.1, "Shutdown Resulting From Reactor Trip or Turbine Trip". Appendix "A", entitled "Post Trip Review", is performed by the Shift Technical Advisor and approved by the Plant Supervisor - Nuclear prior to returning the unit to power.

Section 1A of the Post Trip Review requires the listing of certain trip sequence time intervals as identified by the sequence of events computer printout. Two of these items, reactor trip relay dropout and reactor trip breakers open, must have their times compared to verify that the reactor trip breakers opened within 100 milliseconds of the reactor trip relay dropout time. The purpose of this comparison is to verify that the breaker trip mechanism is operating properly. Time differences greater than 100 milliseconds require evaluation to ensure that the minimum permissible time difference of 167 milliseconds will not be exceeded.

The inspector noticed that the time interval between reactor trip relay dropout and reactor trip breakers opening was 230 milliseconds. This discrepancy was not noticed by the Shift Technical Advisor or the Plant Supervisor - Nuclear during the preparation of the Post Trip Review. The reactor was operated at power for several hours before the discrepancy was identified and brought to the licensee's attention.

The licensee performed applicable portions of OP 1004.2, "Reactor Protection System - Periodic Test", and verified that the time from reactor trip relay dropout to reactor trip breaker opening was 80 milliseconds. The licensee believes that the 230 milliseconds time interval which occurred during the reactor trip reflects a time delay inherent in the DDPS computer due to the noncontinuous sampling of the relays.

The failure of the licensee to identify the excessive time delay between relay dropout and trip breaker opening as required by Appendix "A" of ONOP 0208.1, is a violation (251/84-29-02) as the procedure was not followed.

11. Independent Inspection Effort (92706)

During the reporting period the inspector's 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 August 23, 1984, the licensee restarted Unit 3 after a fourteen hour shutdown. The shutdown was necessary to allow the repair of an electrical cable for the control rod drive mechanism (CRDM) associated with control rod J-3. The failed electrical cable had resulted in a loss of power to the CRDM which caused rod J-3 to be fully inserted into the core during power operation.

Prior to the startup, two Estimated Critical Conditions (ECC) calculations were independently performed in accordance with licensee Operating Procedure (OP) 1009.1. One ECC calculation determined that 275 gallons of boric acid solution needed to be added to the reactor coolant system to make the reactor critical with control rod bank D at 100 steps. The second ECC calculation substantially supported the first, requiring a 287 gallon boration to reach criticality with control rod bank D at 100 steps. The calculated boration was accomplished and a reactor startup was commenced.

The reactor did not reach criticality until control rod bank D had been withdrawn to 192 steps. The additional 92 steps of control bank D rod withdrawal indicated that the ECC reactivity balance calculation was in error by approximately 440 PCM. Supervisory personnel monitoring the startup believed the reactivity error resulted from rapidly decreasing Xenon concentration compounded by the effects of operating the reactor with one dropped rod for approximately twelve hours.

Consequently, the 92 step difference between actual critical rod height and predicted critical rod height was not viewed as indicative of a problem. A formal review of the discrepancy was not made. The discrepancy was not recorded in either the Shift Supervisor's or Reactor Operator's log.

Since there is no inherent reason why ECC calculations can not be accurate even during periods of rapid Xenon decay and since the reactor was not operated with the dropped rod long enough to significantly alter fuel or fixed poison concentrations, the inspector reviewed OP 1009.1, "Estimated Critical Conditions," to determine if procedural inadequacies accounted for the inaccurate ECC calculation.

During the course of the review, numerous procedural discrepancies were noted which introduced inaccuracies into the ECC calculation in question and which because of their procedural nature, would introduce errors under any circumstance. A summary of identified discrepancies follows:

a. OP 1009.1 assumes that all reactor shutdowns begin with the reactor at 100% power with all control rod banks fully withdrawn from the core. The ECC worksheet on page 4 of the procedure has pre-printed entries to this effect in the blanks for steps 1.2 and 1.3. Section 8.1 of the procedure states that reference conditions other than 100% power with all rods out may be used if required. However, no guidance is given as to which circumstances require departure from the pre-printed values. Consequently, this option had not been used even when the shutdown occurred from less than 100% power with control bank D partially inserted.

b. OP 1009.1 does not provide for the calculation of Xenon Equivalent Power. All shutdowns are assumed to begin from 100% equilibrium Xenon. This is assumed even for shutdowns following large power changes where Xenon has been driven significantly out of equilibrium.

c. OP 1009.1 assumes that equilibrium samarium has a reactivity of 434 PCM. However, figure 5 of the Plant Curve Book indicates that equilibrium Samarium has a reactivity of 463 PCM. OP 1009.1 allows use of the DDPS computer to obtain a value of maximum samarium. The DDPS computer uses a value of 719 PCM as the maximum Samarium concentration reached following a trip from 100% power. A review of vendor supplied data indicates that the correct value is approximately 940 PCM. Accurate ECC calculations depend on accurately knowing the difference between maximum samarium and equilibrium samarium in the core. The licensee ECC calculations of OP 1009.1 use a maximum-to-equilibrium differential of approximately 255 PCM whereas vendor supplied information indicates the maximum to equilibrium differential can be as large as 325 PCM.

d. OP 1009.1 assumes that the power defect existing in the core prior to shutdown is always the 100% power defect. No correction is made to the 100% power defect when shutdowns occur after operating as less than 100% power.

e. OP 1009.1 uses a constant differential boron rod worth of -10 PCM/PPM when converting boron concentration to units of reactivity. Figure 8 from the Plant Curve book indicates that differential boron worth changes as a function of boron concentrations. The failure to account for changes in boron worth with changing boron concentrations introduce inaccuracies into the ECC calculations.

f. OP 1009.1 allows the use of the DDPS computer to obtain values of xenon and samarian worth without taking into account the accuracy of the computer calculation. When the DDPS system is reset after a computer failure the values initialized for xenon and samarium are always 100% equilibrium values. If the reactor has been operating at less than 100% power the use of these values introduces inaccuracies in the ECC calculation.

g. OP 1009.1 allows the use of DDPS xenon and samarium values even though the computer program which is used to calculate the values has not been updated for several years and does not reflect values associated with the currently installed core.

h. OP 1009.1 does not require verification of the actual boron concentration in the reactor coolant system immediately prior to startup. Consequently, the boration or dilution which was made as a result of the ECC calculation is not checked to see that it succeeded in creating the actual boron concentration required by ECC.

i. Numerous graphs kept in the Plant Curve Book and required for use in OP 1009.1 are not corrected for core age. Some of these graphs shift significantly over core life and failure to account for this shift can reduce the accuracy of the estimated critical condition calculation.

j. OP 1009.1 does not provide any guidance as to the length of time an ECC calculation remains valid. Changing xenon concentration following shutdown can cause actual xenon concentration in the core to differ from the xenon concentration predicted at startup. Consequently, it is important to commence the startup as close to the estimated startup time as possible. Failure to startup at a time reasonably close to the time used in calculating the ECC introduces inaccuracies in the computation.

The above mentioned items, all introduce unnecessary inaccuracies into the ECC calculation process. In addition to these procedural inaccuracies, licensee personnel incorrectly computed samarian reactivity on both of the ECC calculations performed prior to the startup on August 23, 1984. Contrary to the procedural requirements, each individual failed to take into account the increase in samarium concentration that occurred during the fourteen hours immediately following the reactor shutdown. This increase of approximately 38 PCM in samarium reactivity contributed to the 90 step ECC error. This error was not detected by the Plant Supervisor - Nuclear when he reviewed the calculations. Additionally one of the ECC worksheets did not indicate an intended time of criticality as required by step 2.1. This omission was also not detected by the Plant Supervisor - Nuclear.

A review was made of additional recent ECC calculations to determine how closely they predicted actual critical rod height. On April 24, 1984, after a startup on Unit 3, the actual critical rod height was 85 steps below the estimated critical rod height calculated using OP 1009.1. On May 12, 1984, after a startup on Unit 3, the actual critical rod height was 145 steps below the estimated critical rod height calculated using OP 1009.1.

Conversations with staff reactor engineers revealed that the licensee was aware of these inaccurate ECC's. However, the engineers minimized ECC significance since each startup incorporated on inverse count rate (1/M) plot to monitor the approach to criticality.

A review of recent 1/M plots was conducted. A summary of the discrepancies follows:

a. No guidance exists identifying required operator action when the 1/M plot indicates that the reactor will not be critical even with control bank D fully withdrawn.

b. No guidance exists identifying required operator action when the 1/M plot critical rod height prediction significantly differs from that predicted by the ECC calculation.

c. No guidance exists identifying required operator action when the 1/M plot predicts that criticality will occur below the rod insertion limit.

e. No guidance exists identifying time delays associated with subcritical multiplication. Recording 1/M plot data prior to allowing the neutron population through subcritical multiplication can significantly degrade the accuracy of the 1/M plot.

Nine 1/M plots performed in conjunction with reactor startups performed within the past six months were reviewed. Most plots did not accurately predict reactor critical rod heights. Some plots consisted of only two lines and therefore did not contain sufficient data to accurately estimate reactor criticality. One plot, performed in conjunction with a Unit 3 startup on August 23, 1984, indicated that the reactor could not achieve criticality even if bank D rods were fully withdrawn. However, the criticality was reached with bank D at 192 steps. The ECC calculation predicted criticality at 100 steps.

The failure to develop accurate ECC and 1/M plot procedures constitutes a failure to establish adequate startup procedures and is a violation. (290/84-28-01 and 251/84-29-01).