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

APR 19 1988

Report Nos.: 50-250/88-05, 50-251/88-05

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

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

Facility Name: Turkey Point

Inspection Conducted: March 21-25, 1988

Inspector: John B Kahle for P. G. Stoddart — 4/12/88 Date Signed

Approved by: John B Kahle — J. B. Kahle, Section Chief, Division of Radiation Safety and Safeguards — 4/12/88 Date Signed

SUMMARY

Scope: This routine, unannounced inspection was conducted in the areas of liquid and gaseous waste processing, liquid and gaseous effluents, effluent monitoring, post accident sampling, confirmatory measurements and environmental monitoring.

Results: No violations or deviations were identified.

REPORT DETAILS

1. Persons Contacted

   Licensee Employees

   *T. Abatiello, Supervising Engineer, QA
   *J. Arias, Compliance Supervisor
   *C. Baker, Plant Manager
   *A. Dyches, QA Engineer
   *R. Earl, Quality Control Supervisor
   *A. Gould, Staff Specialist, Chemistry and Waste Management
   *P. Hughes, Health Physics Supervisor
   *J. Kappes, Maintenance Superintendent
   *M. Kule, Project Engineer
   *J. Labarraque, Technical Department Superintendent
   *D. Meils, Chemistry Supervisor
   K. Remington, Chemist
   *G. Salamon, Compliance Specialist
   R. Spooner, Quality Assurance, Corporate
   R. Steinke, Chemist
   D. Tomaszewski, I&E Supervisor

   Other licensee employees contacted included engineers, technicians, and office personnel.

   Nuclear Regulatory Commission

   *D. Brewer, Senior Resident Inspector
   T. McElhinney, Resident Inspector
   G. Schnebli, Resident Inspector

   *Attended exit interview

2. Exit Interview

   The inspection scope and findings were summarized on March 25, 1988, with those persons indicated in Paragraph 1 above. The inspector described the areas inspected and discussed in detail the inspection findings listed below. No dissenting comments were received from the licensee. Licensee investigation and remedial measures in the April 1987 event of contamination of the demineralized water system appeared adequate. Licensee actions in replacing several principal components of the post accident sampling system were timely and demonstrated licensee initiative in problem solving. The inspector's review of the Semiannual Radiological Effluent Reports for 1987 indicated these to be acceptable. Licensee laboratory determination of NRC-supplied "spiked" samples for radioactivity content was adequate. Licensee's audit programs in the inspector's review areas were adequate. Licensee's cross-check program for comparison of counting laboratory results with those of a vendor

laboratory was acceptable. The licensee did not identify as proprietary any of the material provided to or reviewed by the inspector during this inspection.

3. Licensee Action on Previous Enforcement Matters (84723)

(Closed) Unresolved Item (URI) 50-250, 251/87-18-01: Root cause evaluation of demineralized water system contamination incident of April 1987 and of corrective actions taken.

During the general time frame of the demineralized water system (DWS) contamination problem of April 19-20, 1987, the normal Units 3/4 demineralized water processing system was out of service and demineralized water was being supplied to the Units 3/4 DWS from the adjoining fossil plant (Units 1/2). Prior to April 19, 1987, when the contamination was discovered, the available DWS pressure was lower than normal because of limited flow and pumping capacity at the fossil plant.

The licensee found that the Unit 3 spent fuel pool (SFP) could not be brought up to the desired level using the DWS because of low water pressure. As an alternative, the licensee elected to use water from the primary water storage tank (PWST) to fill the SFP. To accomplish this, it was necessary to use a section of SFP water fill line normally used by the DWS and to open isolation valve No. 821, located between the SFP and check valve No. 513 (provided to prevent backflow into the DWS).

With isolation valve No. 821 open, only one check valve, No. 513, prevented water from the PWST from entering the DWS; further, at such time as the filling of the SFP was discontinued and until isolation valve No. 821 was closed, the full static head pressure of the SFP was applied to check valve No. 513. On the DWS side of check valve No. 513, DWS pressure was lower than normal and below either the pressure of the fill line from the PWST or the static head of the SFP.

During the licensee's investigation, it was determined that check valve No. 513 was defective and leaked under pressure. With the DWS at a pressure lower than either the static head pressure of the SFP or the pressure of the PWST water during filling of the SFP, with isolation valve No. 821 open, and with check valve No. 513 leaking, a path existed for leakage into the DWS from both the PWST and the SFP for the entire period that isolation valve No. 821 was open.

The licensee's investigation team concluded, based on analyses of samples from the DWS, that the Unit 3 SFP was the probable source of the radioactive contamination of the DWS, and that the leaking check valve No. 513 was the probable point of entry of the contaminant into the DWS.

While no contamination of the DWS was detected until more than 24 hours following the filling of the SFP from the PWST, it was considered probable that the containment was confined during that time to a section of the DWS piping until the DWS was used on the night of April 18, to fill the demineralized water storage tank (DWST) and the laboratory water tank. The presence of contaminants in the demineralized water system was first discovered shortly after midnight on April 19.

The licensee's report on the investigation indicated the root cause to be the leaking check valve No. 513, with contributing causes being the indeterminate time during which isolation valve No. 821 may have been left open following completion of the operation filling the SFP with water from the PWST, the low pressure of the DWS, and failure to test check valve No. 513 on a regular basis; these were among eleven recommendations for remedial measures which have since been implemented.

The inspector's review of the licensee investigation reports and remedial measures indicated that the licensee' evaluation of the root cause and the licensee's program to prevent a recurrence appeared to be adequate. This matter is considered closed.

4. Audits (84723, 84724, 84725)

The inspector reviewed licensee corporate audit reports on plant activities and discussed the reports with licensee personnel.

Audit QAO-PTN-87-863 (September 14, 1987) covered the Post Accident Sampling System (PASS) with respect to the plant Technical Specifications and with respect to the implementing PASS procedures. The audit report indicated that the PASS was essentially inoperable with the exception of the boron analysis section and that many of the PASS procedures were inadequate. In response to the audit findings, the licensee undertook a major PASS update project which included an engineering contract with Westinghouse. By March 1, 1988, all major inoperative components of the PASS had either been replaced or repaired with the exception of the chloride monitor. For chloride analysis, the licensee demonstrated the interim capability for grab sampling with analysis to be performed at an offsite or onsite facility within the prescribed 24-hour period; at the time of the inspection, the licensee was considering the use of an inline chromatograph for remote analysis of chloride in primary coolant. At the time of the inspection, licensee test records indicated the PASS to be operational with the exception that grab sampling for chloride analysis was utilized in place of inline analysis.

Audit QAO-PTN-88-036 (January 20, 1988) evaluated PASS operation during an emergency preparedness drill, with emphasis on grab sampling for chloride, transport of the sample, and analysis. Sampling was successful; however, while moving the sample cart, one wheel became jammed in a crack in the concrete floor and the cart could not be moved by one person. In response to the audit findings, the licensee revamped sample transfer procedures, reviewed pre-established sample cart routes, and made simulated runs using the sample cart to assure that the cart could be moved to its proper destination by one person. The licensee's response to the corporate audit findings projected that corrective actions would be completed by July 1988, however, a licensee representative informed the inspector that completion was anticipated during April 1988.

The site Quality Assurance group conducted activities of an audit nature under reports titled "Performance Monitoring Summary" or "PMONS." The "PMONS" program appeared to be a useful augmentation of the audit program. The activities monitored under PMONS were typically of a discrete nature, such as calibration of process radiation monitoring channel R-14 (PMON-PTN-87-030, March 9, 1987), flushing of the contaminated demineralizer water system (PMON-PTN-87-082, June 3, 1987), and exchange of plant vent and Unit 3 spent fuel pit particulate and iodine filter cartridges (PMON-PTN-87-046, March 26, 1987).

The audits and performance monitoring summaries appeared to be thorough and of adequate depth. Findings were promptly resolved during the audit period. In discussions with licensee personnel, the inspector concluded that there was a high level of management commitment to fully address and resolve the findings.

No violations or deviations were identified.

5. Semiannual Radioactive Effluent Release Reports (84723, 84724)

The inspector reviewed the licensee's Semiannual Radioactive Effluent Release Reports for January to June 1987, and July to December 1987, and discussed the reports with licensee representatives. The following is a summary of releases for CY 1987:

Liquids (Units 3 and 4 Combined)

Fission and Activation Products	7.5 E-01 Ci
Tritium	5.4 E+02 Ci
Dissolved Gases	8.2 E+00 Ci
Gross Alpha	None detected*

*Based on lower limit of detection of less than 1 E-09 uCi/ml.

Gases (Units 3 and 4 Combined)

Fission and Activation Products	1.7 E+03 Ci
Iodine	2.3 E-02 Ci
Particulates	2.1 E-03 Ci
Tritium	8.2 E+02 Ci

The licensee's calculated radiation doses in the environment attributable to plant releases for each unit were less than 1% of the limits established in the plant Technical Specifications.

In discussions between the inspector and licensee representatives, the inspector noted that dissolved gas activity in liquid effluents was higher than in previous reports. The licensee representatives stated that the

increase was attributable to analyzing all liquid batch releases for dissolved gas instead of the minimum Technical Specification requirement for analysis of one sample per month. Based on data for the fourth quarter (CY-1987), the only time period in which dissolved gas analysis was done on all releases, the increase appeared to be approximately a factor of ten. The impact of the calculated increase in offsite doses resulting from the increase in dissolved gases was negligible.

The reports were submitted within the required schedule, appeared to be full responsive to Regulatory Guide 1.21 guidance, and were considered to be acceptable.

No violations or deviations were identified.

6. Radioactive Gaseous Effluent Treatment Systems (84724)

Radioactive gaseous effluent treatment systems were provided in the plant design for the purpose of minimizing the environmental effects of potential plant radioactive gaseous effluents. Appendix I to 10 CFR Part 50 requires the licensee to provide for treatment of radioactive gaseous effluents to radiation exposure in offsite areas to levels as low as reasonably achievable (ALARA).

The inspector discussed plant operating experience of the gaseous effluent treatment systems with licensee representatives, reviewed relevant procedures and records, and toured the gaseous effluent treatment system areas accompanied by licensee personnel.

The inspector discussed maintenance, operation and testing of high efficiency particulate air (HEPA) filters and carbon adsorbers with licensee representatives. Required tests were performed during refueling outages. Technical Specification 4.7 requires leak testing and carbon sample analysis of the following systems: emergency containment filter system; post accident containment vent system; and the Control Room ventilation (emergency internal cleanup) system. The Technical Support Center emergency ventilation system filters, while not a Technical Specification requirement, was also tested to meet the guidance of Regulatory Guide 1.52.

The inspector reviewed the test documents covering the most recent tests for the four systems. The in-place leak tests for the HEPA filters and carbon adsorbers were conducted by a qualified vendor and results were within specified limits. The laboratory analysis of carbon from the adsorber bed of the Technical Support Center emergency ventilation system did not meet Technical Specification requirements for methyl iodide retention. New charcoal was installed and the system was satisfactorily retested on February 29, 1988. Methyl iodide retention tests of the carbon in the other three systems were satisfactory.

In reviewing the plant Technical Specifications, the inspector noted that Amendment 103, to Table 3.9-4, dated 1983, concerning hydrogen and oxygen

monitors in the explosive gas monitoring system, stated " ... Until modifications to the gas analyzing equipment are complete, the following action will be taken ..." The inspector inquired into the current status of the item and discussed the matter with licensee personnel. The inspector reviewed a memorandum dated March 21, 1988, in which it was noted that a vendor representative was to be onsite in early April 1988 for the purpose of resolving the operability issues of the online hydrogen and oxygen monitoring system. The inspector was informed that subsequent to the date of the memorandum, a firm date of April 5, 1988, had been established for the visit of the vendor representative. Following that visit, it was anticipated that a schedule or date could be established to close-out this matter.

The licensee's waste gas process system utilized six decay tanks. At the time of the inspection, with both units at full power, approximately 15 tanks per month were being released to the atmosphere, with decay based on two days of fill time and ten days radioactive decay. For the period of 1974 through 1986, the licensee's average annual of fission product and activation gases was 5.8 E+03 Ci, which closely approximated the average for all PWRs in NRC Region II for the same time period. Calculated offsite radiation doses attributable to gaseous effluent releases were less than or equal to 2.2 E-03 mRem for inhalation and less than or equal to 5.1 E-03 mRad for noble gas air dose. The doses were less than 1% of 10 CFR 50, Appendix I limits and were therefore considered to be ALARA.

No violations or deviations were identified.

7. Radioactive Gaseous Effluent Monitoring, Sampling and Analysis (84724)

The inspector discussed radioactive gaseous effluent monitoring, sampling and analysis with licensee personnel.

High range effluent monitors (NUREG-0737 Item II.F.1, Attachment 1) had been recently calibrated against samples from the waste gas decay tanks in order to demonstrate capability to adequately respond to nuclides other than Xe-133. Prior to making calibration runs against radioactive gases, monitors were electronically checked out, aligned and calibrated with electronic pulse generators. The monitoring sampling path was then modified to provide a closed recirculating pathway. Gas samples were obtained from the volume control tank gas space or from one of the waste gas decay tanks after approximately one week of decay to minimize the effects of plateout of the particulate decay products of the short-lived noble gases (e.g., Kr-88, Xe-138). The gas samples were then injected into the monitor recirculating system and the monitor readout values recorded when equilibrium was obtained. Representative samples were then withdrawn and analyzed on MCAs which were calibrated against standards traceable to NBS. Analysis results were then correlated to monitor readings. Gas concentrations established in the monitor were on the order of 1 E-03 uCi/ml to 1 E-05 uCi/ml.

Correlations between monitor readouts and laboratory analyses were within less than 25%. These values were considered acceptable and were within the criteria and guidance of NUREG-0737, Item II.F.1, Attachment 1, and of the NRR letter dated August 11, 1982, providing additional guidance for calibration and surveillance requirements.

In recent years, the licensee had experienced a number of failures of the radiation monitor on the main condenser air ejector discharge. The cause had been considered by the licensee to be moisture-related. In discussions between the licensee and the vendor it was the opinion of the vendor representative that moisture may not have been the only problem in that most of the detectors involved had been in continuous service for three to four years which was in excess of the expected two-year useful life of the detectors. The vendor representative recommended detector replacement at scheduled two year intervals to forestall age-related failures and this recommendation was implemented; this appeared to resolve the problem but the fix had not been in place long enough, at the time of the inspection, to fully assess the long-term benefits.

The licensee had also experienced a series of failures in process and air monitors located in the primary coolant equipment areas. It was determined that ammonia fumes from the all-volatile treatment process were entering radiation monitor equipment housings and causing corrosion-related failures. The licensee replaced many of the electronic component boards and ducted clean air into the monitor housings to prevent ammonia intrusion. As in the case of the condenser air ejector monitor fix, the licensee at the time of the inspection had not accumulated sufficient time on the systems to determine if the clean air pressurization of the monitor equipment housings had totally resolved the problem.

With the exception of the high-range gaseous effluent monitors (NUREG-0737 items), the process and effluent radiation monitoring system, as well as the area radiation monitoring system, had been in place for over fifteen years and had encountered many apparently age-related problems. The licensee initiated a program for replacement of the principal component circuit boards and other electronic circuit components on a compatible-replacement basis. At the time of the inspection, essentially all of the aging-sensitive components had been replaced through a vendor who supplied components to replace components originally supplied by the original equipment vendor, Tracerlab Company, which is no longer in business. The replacement components incorporated state-of-the-art improvements as far as practicable and were expected to improve system performance and reliability.

No violations or deviations were identified.

8. Liquid Radioactive Waste Processing (84723)

The inspector and licensee representatives discussed the operation and performance of the liquid radioactive waste processing system. Liquid

radioactive waste from the miscellaneous waste process streams was treated through filtration and demineralizer trains by a vendor, DURATEK, under contract with the licensee. The processed liquid, after treatment, typically had residual radioactive concentrations on the order of 1 E-06 uCi/ml prior to discharge to the recirculating cooling canal system.

The total plant discharge of liquid fission and activation product activity to the site circulating cooling water canal system for CY 1987 was 0.75 curies. Calculated maximum radiation dose to an individual in the environment, based on a theoretical individual on the shoreline of the site cooling canal, was approximately 1% of the annual Technical Specification limit and was less than the ALARA limits of Appendix I to 10 CFR Part 50 and within the limits of 40 CFR 190.

No violations or deviations were identified.

9. Post Accident Sampling System (PASS) (84723, 84724)

The inspector discussed the current status of the PASS with licensee representatives and toured the PASS installation with licensee representatives and system operators. In QAO Audit QAO-PTN-87-863 (October 1987), the PASS had been found to be inoperable with the exception of the Boronometer. Since that time, the licensee had replaced many of the non-functioning components. As of February 29, 1988, only the PASS chloride in-line monitor remained out-of-service. At the time of the inspection, the licensee was considering the procurement and installation of an in-line ion chromatograph for the determination of chloride in reactor coolant. With the existing ion-specific chloride meter inoperable, the licensee retained the capability for grab sampling of primary coolant for transport to either an onsite or offsite laboratory for chloride analysis within the prescribed 24-hour period for plants utilizing seawater cooling.

A QA audit (QAO-PTN-88-036) was conducted during an emergency preparedness drill on January 20, 1988. As part of the demonstration of operability of alternate sampling procedures for the PASS, the audit team observed the transporting of the shielded cask containing a PASS liquid sample from the PASS location to the counting laboratory. At one point, a wheel of the cask cart became lodged in a space between two sections of floor in such a manner that the technician was unable to move the cask cart by himself. The audit finding concluded that the time limitation for sampling and analysis had not been met and recommended review of the transport methods and procedures and a review of the proposed transport routes.

The licensee performed the review and determined that several of the routes originally laid out had been made impassable for the cask cart as the result of plant equipment changes or modifications. The licensee committed to complete the review and necessary changes or modifications needed to fully implement the PASS alternate sampling procedure by July

1988; however, the licensee anticipated completing the process in early April 1988.

No violations or deviations were identified.

10. Collection of Collocated TLD Data (80721)

The inspector requested the licensee to provide TLD data for the collocated TLD stations in the vicinity of the plant. The requested material was provided on March 23, 1988. The TLD data was provided in units of uR/hr. The material supplied to the inspector was forwarded to the Dosimetry Specialist at RI.

No violations or deviations were identified.

11. Confirmatory Measurements (84725)

The inspector discussed the licensee's quarterly "blind sample" QA cross-check program with licensee staff members. The program utilized a vendor who participated in a cross-check program with the National Bureau of Standards. The licensee's results on tritium in the second quarter of 1987 were in disagreement with the vendor's analysis at approximately the same time as the licensee's tritium results on the NRC-supplied samples were in disagreement (samples sent May 1987). The licensee reviewed procedures and took necessary corrective action. A second set of NRC supplies samples was analyzed with satisfactory agreement in November 1987.

The licensee's count room QA practices were audited by plant QA and reported in QAO-PTN-87-842. Calibration procedures were reviewed to verify that NBS-traceable standards were used. The auditors observed that calibration records, efficiency checks, data trending and other control charts were utilized. The audit report verified that calibration and control program was satisfactory. The only adverse finding was in the area of chloride and fluoride analyses, where operation of a new ion chromatograph had not been fully proceduralized.

As part of the NRC Confirmatory Measurements Program, spiked liquid samples were sent, on November 5, 1987, to the licensee's facility for selected radiochemical analyses. The licensee's analytical results were received by NRC by letter dated January 11, 1988. The following comparison of results is presented in Table 1. The acceptance criteria for the comparisons are listed in Table 2.

In the inspector's review of these data, comparative results for tritium (H-3), strontium-90, and iron-55 were in agreement. Strontium-89 was not compared due to the short half-life of strontium-89 and the age of the spike material. The result of the above condition was analytical uncertainty values for strontium-89 of 50% or more, making valid comparisons impractical. These data should be reviewed in detail by cognizant licensee staff members for any significant trends in the data among successive years in which samples have been analyzed for the licensee's facility. Any biases noted may be indicative of a programmatic weakness and licensee efforts should be expended in determining reasons for such biases.

These results were discussed with licensee personnel during this inspection.

No violations or deviations were identified.

TABLE 2

Criteria for Comparing Analytical Measurements

This enclosure provides criteria for comparing results of capability tests and verification measurements. The criteria are based on an empirical relationship which combines prior experience and the accuracy needs of this program.

In these criteria, the judgement limits denoting agreement or disagreement between licensee and NRC results are viable. This variability is a function of the NRC's value relative to its associated uncertainty, referred to in this program as "Resolution"[1] increases, the range of acceptable differences between the NRC and licensee values should be more restrictive. Conversely, poorer agreement between NRC and licensee values must be considered acceptable as the resolution decreases.

For comparison purposes, a ratio[2] of the licensee value to the NRC value for each individual nuclide is computed. This ratio is then evaluated for agreement based on the calculated resolution. The corresponding resolution and calculated ratios which denote agreement are listed in the Table below. Values outside of the agreement ratios for a selected nuclide are considered in disagreement.

[1]Resolution = $\frac{\text{NRC Reference Value for a Particular Nuclide}}{\text{Associated Uncertainty for the Value}}$

[2]Comparison Ratio = $\frac{\text{Licensee Value}}{\text{NRC Reference Value}}$

TABLE

Confirmatory Measurements Acceptance Criteria
Resolutions vs. Comparison Ratio

Resolution	Comparison Ratio for Agreement
< 4	0.4 - 2.5
4 - 7	0.5 - 0.2
8 - 15	0.6 - 1.66
16 - 50	0.75 - 1.33
51 - 200	0.80 - 1.25
>200	0.85 - 1.18

TABLE 1

CONFIRMATORY MEASUREMENT COMPARISONS FOR H-3, SR-90, AND FE-55 ANALYSES
FOR TURKEY POINT NUCLEAR PLANT ON JANUARY 11, 1988

(uCi/ml) &lution	Licensee Reso (Licensee/NRC)	NRC Comparison		Ratio Isotope	(uCi/ml)
H-3	2.18 E-05	2.38 E-05	48	0.92	Agreement
Fe-55	1.61 E-05	1.46 E-05	49	1.10	Agreement
Sr-90	2.87 E-06	2.95 E-06	25	0.97	Agreement