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

Report Nos.: 50-250/92-16 and 50-251/92-16

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 Units 3 and 4

Inspection Conducted: June 27 through July 24, 1992

Inspectors: R. C. Butcher, Senior Resident Inspector — Date Signed 8/17/92

R. Butcher for G. A. Schnebli, Resident Inspector — Date Signed 8/17/92

R. Butcher for L. Trocine, Resident Inspector — Date Signed 8/17/92

R. Butcher for J. J. Lenahan, Reactor Inspector — Date Signed 8/17/92

Accompanying Personnel: M. T. Janus, Reactor Engineer

Approved by: M V Sinkule for K. D. Landis, Chief
Reactor Projects Section 2B
Division of Reactor Projects — Date Signed 8/18/92

SUMMARY

Scope:

This routine resident inspector inspection involved direct inspection at the site in the areas of an NRC Bulletin, monthly surveillance observations, monthly maintenance observations, operational safety, verification of plant records, TMI action items, preparations for refueling, and plant events. Backshift inspections were performed on June 30, July 1, 9, 10, and 20 thru 23, 1992.

Results:

Within the scope of this inspection, the inspectors determined that the licensee continued to demonstrate satisfactory performance to ensure safe

plant operations. One violation, one non-cited violation, and one weakness were identified.

50-250,251/92-16-01, Violation - Failure to follow a procedure resulting in a spill of flush water and residual spent resin in the radwaste building (paragraph 11.b).

50-250,251/92-16-02, Non-Cited Violation - Failure to install seal table moveable cart in accordance with design drawing requirements (paragraph 5.a).

Weakness - Adequacy of the radwaste building drainage system in preventing any future spills from reaching the environment (paragraph 11.b).

REPORT DETAILS

1. Persons Contacted

Licensee Employees

* T. V. Abbatiello, Site Quality Manager
R. J. Earl, Quality Assurance Supervisor
R. J. Gianfrencesco, Support Services Supervisor
E. F. Hayes, Instrumentation and Controls Maintenance Supervisor
R. G. Heisterman, Mechanical Maintenance Supervisor
D. E. Jernigan, Technical Manager
H. H. Johnson, Operations Supervisor
* V. A. Kaminskas, Operations Manager
* J. E. Knorr, Regulatory Compliance Analyst
R. S. Kundalkar, Engineering Manager
J. D. Lindsay, Health Physics Supervisor
G. L. Marsh, Reactor Engineering Supervisor
* L. W. Pearce, Plant General Manager
M. O. Pearce, Electrical Maintenance Supervisor
* T. F. Plunkett, Site Vice President
* D. R. Powell, Services Manager
R. N. Steinke, Chemistry Supervisor
* F. R. Timmons, Security Supervisor
* M. B. Wayland, Maintenance Manager
J. D. Webb, Outage Manager (Acting)
* E. J. Weinkam, Licensing Manager

Other licensee employees contacted included construction craftsman, engineers, technicians, operators, mechanics, and electricians.

NRC Resident Inspectors

R. C. Butcher, Senior Resident Inspector
* G. A. Schnebli, Resident Inspector
L. Trocine, Resident Inspector

Accompanying NRC Inspector

* M. T. Janus, Reactor Engineer

Other NRC Inspectors

J. J. Lenahan, Reactor Inspector

* Attended exit interview on July 24, 1992

Note: An alphabetical tabulation of acronyms used in this report is listed in the last paragraph in this report.

2. Other NRC Inspections Performed During This Period

Report No.	Dates	Area Inspected
50-250,251/92-17	July 20 thru 24, 1992	Emergency Preparedness

3. Plant Status

Unit 3

At the beginning of this reporting period, Unit 3 was operating at 87% power in order to extend the unit run time until the beginning of the August 24, 1992, refueling outage. Unit 3 had been on line since May 12, 1992. The following evolutions occurred on this unit during this assessment period:

- On June 28, 1992, at 7:35 p.m., a load reduction was commenced in order to facilitate the cleaning of the 3A north and south waterboxes. At 8:10 p.m., reactor power reached 58%.

- On June 29, 1992, at 4:50 a.m., a power ascension was commenced. At 8:55 a.m., reactor power reached 87%, and extended reduced power operation at this power level was recommenced in order to extend the unit run time until the beginning of the August 24, 1992, refueling outage.

- On July 4, 1992, at 5:00 p.m., a power reduction to 55% was commenced to clean the TPCW heat exchangers. At 6:30 p.m. reactor power reached 55%.

- On July 5, 1992, at 5:30 p.m., a load increase to 87% power was commenced. At 7:15 p.m. reactor power reached 87%.

- On July 9, 1992, at 9:45 a.m., a load increase was commenced on Unit 3 due to FPL system load demands. At 2:05 p.m. the Unit 3 reactor power reached 100%.

- On July 10, 1992, at 7:30 p.m., a load reduction to 87% power was commenced. At 10:00 p.m., reactor power was stabilized at 88% for turbine vibration considerations.

Unit 4

At the beginning of this reporting period, Unit 4 was operating at 100% power and had been on line since May 7, 1992. The following evolutions occurred on this unit during this assessment period:

- On July 2, 1992, at 6:30 p.m., a load reduction to approximately 50% power was commenced in order to facilitate cleaning of the 4A TPCW heat exchanger. At 8:30 p.m., reactor power was stabilized at 50%.

- On July 3, 1992, at 1:05 p.m., power ascension was commenced. At 6:35 p.m., 100% reactor power was re-achieved.

- On July 22, 1992 at 6:30 a.m., a load reduction to 88% was commenced to induce Xenon oscillations for O-OP-059.3. At approximately 7:00 a.m., reactor power was stabilized at 88%.

- On July 22, 1992, at 2:10 p.m., a power increase to 100% was commenced. At 3:30 p.m., reactor power reached 100%.

4. Followup on Items of Noncompliance (92702)

A review was conducted of the following noncompliances to assure that corrective actions were adequately implemented and resulted in conformance with regulatory requirements. Verification of corrective action was achieved through record reviews, observation, and discussions with licensee personnel. Licensee correspondence was evaluated to ensure the responses were timely and corrective actions were implemented within the time periods specified in the reply.

(Closed) VIO 50-250,251/92-03-01, Failure To Maintain Safety Related 4 KV Switchgear In A Seismically Qualified Configuration As Required By 10 CFR 50, Appendix A, Criterion 2.

The licensee responded to this violation in letter L-92-082 dated April 8, 1992. The inspectors reviewed the corrective actions required to correct this violation and found them to be adequate. This violation is closed.

5. Followup on Inspector Followup Items (92701)

Actions taken by the licensee on the items listed below were verified by the inspectors.

a. (Closed) Unresolved Item 250,251/89-28-01, Adequacy of Licensee's Evaluation of Seal Table Interaction per IEN 85-45.

During an inspection conducted June 13 -16, 1989, documented in NRC Inspection Report numbers 50-250,251/89-28, the inspector noted that the moveable frame portion of the flux mapping system, which was positioned over the seal table, was restrained by an attachment to only one anchor bracket. The vendor design drawings, Teleflux numbers 44308 and 44317 showed that the moveable frame should be restrained during normal operations by attaching the frame assembly to two floor mounted anchor brackets. The licensee issued nonconformance report number 89-0239 and 89-0240 to document and disposition those problems. The inspector questioned the acceptability of attaching the moveable frame using only one attachment. In response to the inspector's questions, the licensee directed Westinghouse to perform an analysis for the moveable cart installation restrained by only one of the two

anchor brackets. Westinghouse previously recommended that this analysis be performed in a letter dated June 7, 1985, to Mr. C. M. Wethy, Vice President, Turkey Point. The inspector reviewed Westinghouse letter serial number FPL 89-749, dated June 19, 1989, which summarizes the results of the Westinghouse engineering evaluation. Westinghouse determined in the engineering evaluation (EQT-89-023) that no failure would have occurred and that the stresses in the seal interface tubes were less than 60 percent of the material yield strength. Westinghouse concluded that the tubes connecting the moveable cart to the seal table and the transfer mechanisms, as well as the one holddown anchor bracket, would have sufficient structural integrity so that there would be no loss of reactor coolant pressure boundary. However, the licensee and Westinghouse decided it was desirable to reduce the load on the tubing and restore the original safety margin by adding two additional lateral restraints to the moveable cart. The requirements for these restraints are stated in Westinghouse letter, serial number FPL-89-762, dated June 26, 1989. The additional restraints were installed by the licensee under PC/M 89-409 (Unit 3) and PC/M 89-410 (Unit 4). The inspector reviewed the system turnover documentation for PC/M 89-409 and 89-410, and verified that the work had been completed. Although the design evaluation indicated failure of the tubing probably would not have occurred, failure to restrain the moveable cart in accordance with the installation requirements shown on the Teleflux drawings is an example of a failure to follow drawing requirements. Criterion V of 10 CFR 50 Appendix B, requires that activities affecting quality be prescribed by documented instructions, procedures or drawings, and shall be accomplished in accordance with the instructions, procedures, or drawings. The failure of the licensee to install the moveable transfer cart in accordance with the requirements shown on the Teleflux drawings were identified to the licensee as a violation. Since the licensee has completed corrective action to correct the violation, and the criteria specified in Section VII.B of the NRC Enforcement policy were satisfied, this violation is not being cited. This item will be tracked as NCV 50-250,251/92-16-02, Failure to Install Seal Table Moveable Cart in Accordance with Design Drawing Requirements. This item is considered closed.

b. NRC BULLETIN 92-01: FAILURE OF THERMO-LAG 330 FIRE BARRIER SYSTEM TO MAINTAIN CABLING TRAYS AND SMALL CONDUITS FREE FROM FIRE DAMAGE.

NRC Bulletin 92-01 was issued on June 24, 1992, and was delivered to the licensee on June 25, 1992, through the local resident inspector's office. The Bulletin required all holders of operating licenses to take the following immediate actions:

- For those plants that use either 1- or 3-hour pre-formed Thermo-Lag 330 panels and conduit shapes, identify the areas of the plant which have Thermo-Lag 330 fire barrier material

installed and determine the plant areas which use this material for protecting either small diameter conduit or wide trays (widths greater than 14 inches) that provide safe shutdown capability.

- In those plant areas in which Thermo-Lag fire barriers are used to protect wide cable trays, small conduits, or both, the licensee should implement, in accordance with plant procedures, the appropriate compensatory measures, such as fire watches, consistent with those which would be implemented by either the plant technical specifications or the operating license for an inoperable fire barrier.

The facility utilizes Thermo-Lag 330 throughout the plant (approximately 14,000 linear feet) and the licensee's TS 3/4.7.9.a. for fire rated assemblies states:

> With one or more of the above required fire rated assemblies and/or sealing devices inoperable, within 1 hour either establish a continuous fire watch on at least one side of the affected assembly, or verify the OPERABILITY of fire detectors on at least one side of the inoperable assembly and establish an hourly fire watch patrol.

As an interim compensatory action the licensee implemented an hourly roving fire watch in areas containing Thermo-Lag with operable fire detectors and a 30 minute roving fire watch in areas containing Thermo-Lag with no fire detection system or inoperable fire detectors. These compensatory actions would remain in effect until the licensee determined the applicability of the Bulletin to the facility.

Due to the licensee's extensive use of Thermo-Lag in areas without fire detection systems, mainly in the outdoor areas of the turbine building and the RCA, the licensee's initial evaluation determined continuous fire watches would be required in about 20 different locations. This would require approximately 100 additional personnel to perform this function. Since this many additional people would impact the licensee, two conference calls were conducted on June 26, 1992, between the licensee, RII, and NRR. The purpose of the call was to provide an acceptable understanding of the requirements for a continuous fire watch for this situation and to better identify the conduit and cable tray sizes of concern addressed in the Bulletin. In was agreed upon between the licensee and NRC that a roving watch conducting rounds approximately every 10 minutes would be adequate for areas containing Thermo-Lag without fire detection systems available for the situations discussed in the Bulletin. In addition, NRR stated that conduit sizes of 4 inches and greater and cable tray 14 inches and less were acceptable and that the requirements of the Bulletin were not applicable for those applications.

In order to document the corrective actions taken to address the concerns stated in the Bulletin, the licensee made a 10 CFR 50.72 voluntary report at 4:30 p.m. on June 26, 1992, stating the following:

> NRC Bulletin No. 92-01 discusses the failure in fire endurance testing of Thermo-Lag 330 fire barrier system. The testing called into question, in specific test configurations, the ability of Thermo-Lag 330 to maintain cabling in wide cable trays and small conduits free from fire damage. The bulletin determines that conduit and cable trays with Thermo-Lag in certain configurations be considered inoperable and that compensation be provided as required by Technical Specifications. FPL is in the process of determining the applicability of the Bulletin test configuration of Thermo-Lag to Turkey Point. Additionally, Turkey Point is identifying the areas that need to have compensatory actions applied in the interim. Turkey Point has implemented a 30 minute roving fire watch in those areas that do not have fire detection but do have Thermo-Lag. The testing performed on the Thermo-Lag that was discussed in the Bulletin resulted in failure at some time less than 1 hour but greater than 30 minutes. Based upon this information, a roving fire watch provides detection during that interval. After the identification of those areas that meet the criteria of the Bulletin appropriate compensatory actions will be instituted until Appendix R qualification can be reverified for the Thermo-Lag protected electrical equipment as applied at Turkey Point. Fire watch zones of a limited area will be designated for a continuous fire watch. The fire watch zone will either be traversed by the fire watch continuously to watch for fire in the area or monitored by remote detection equipment. This compensatory action results in each area being observed at least once each 10 to 15 minutes. In accordance with the actions requested by Bulletin 92-01, within 30 days FPL will notify the NRC of the compensatory actions being taken to assure fire barrier operability. FPL will keep the NRC informed of the corrective action to be taken to allow Turkey Point to discontinue compensatory action due to the consequences of the lack of qualification of Thermo-Lag as required by Bulletin 92-01.

Another conference call between the licensee, RII, and NRR was conducted at 9:00 a.m. on July 1, 1992, to discuss acceptable fire detection methods. The licensee proposed to install CCTVs in all locations containing Thermo-Lag without a fire detection system. The CCTVs would be monitored

continuously by an individual at a central location. This installation would provide a means of fire detection in the areas containing Thermo-Lag thus reducing the requirement for a 10 to 15 minute roving fire watch to a 1 hour roving fire watch as specified in the licensee's TS. The system described by the licensee would require approximately 40 CCTVs and 10 video monitors. Each monitor would be connected to 4 CCTVs and would scan between them at 20 second intervals. The NRC agreed that installation of the CCTV system to provide a means of fire detection was an acceptable interim corrective action until a long term resolution to the Thermo-Lag issue was identified and evaluated.

The licensee installed the CCTV fire detection system in accordance with TSA 03-92-16-03 to continuously monitor the areas of the plant where Thermo-Lag is installed without a fire detection system. Installation and testing of the system was completed on July 14, 1992, and the system was declared operational. The resident inspectors observed the system in operation in daylight and nighttime hours and considered the system would provide on acceptable means of fire detection in those areas previously not covered by a fire detection system.

The licensee made another voluntary 10 CFR 50.72 report at 11:10 a.m. on July 14, 1992, stating the following:

> NRC Bulletin No. 92-01 discussed the failure in fire endurance testing of Thermo-Lag 330 fire barrier system. The testing called into question, in specific test configurations, the ability of Thermo-Lag 330 to maintain cabling in wide cable trays and small conduits free from fire damage. The bulletin declared that conduit and cable trays with Thermo-Lag in certain configurations be considered inoperable and that compensation be provided as required by Technical Specifications. Florida Power and Light (FPL) determined the applicability of the Bulletin test configurations of Thermo-Lag to Turkey Point. Additionally, FPL identified the areas that needed to have compensatory actions applied.
>
> Turkey Point has implemented a watch zone based fire watch compensatory action, as discussed in a June 26, 1992, 10 CFR 50.72 notification, in those areas that do not have fire detection but do have Thermo-Lag in the configurations identified in Bulletin No. 92-01. This compensatory action resulted in each affected area being observed at least once each 10 or 15 minutes.

Beginning on July 14, 1992, remote detection by closed circuit television was activated and will provide detection in those areas determined to require compensation for inoperable Thermo-Lag. The closed circuit television system is presently comprised of 39 cameras and 10 monitors. Each monitor cycles through 4 camera displays approximately every 80 seconds. Therefore each camera display will be in view for approximately 20 seconds. A camera, cable or monitor failure can be detected from the monitor location. The 10 monitors will be viewed by a dedicated operator.

The closed circuit television system, which provides detection, as well as an hourly fire watch patrol, will be used in lieu of a continuous fire watch as provided in Technical Specification 3.7.9, Fire Rated Assemblies. The camera configuration is subject to change based upon the experience gained during use of the system and further evaluation of the Thermo-Lag qualification situation.

In accordance with the actions requested by Bulletin 92-01, FPL will notify the NRC by letter of the compensatory actions being taken or modifications being made to assure fire barrier operability.

The resident inspectors will continue to follow the licensee's long term solution to the Thermo-Lag issue.

6. Onsite Followup and In-Office Review of Written Reports of Nonroutine Events and 10 CFR Part 21 Reviews (90712/90713/92700)

The Licensee Event Reports and/or 10 CFR Part 21 Reports discussed below were reviewed. The inspectors verified that reporting requirements had been met, root cause analysis was performed, corrective actions appeared appropriate, and generic applicability had been considered. Additionally, the inspectors verified the licensee had reviewed each event, corrective actions were implemented, responsibility for corrective actions not fully completed was clearly assigned, safety questions had been evaluated and resolved, and violations of regulations or TS conditions had been identified. When applicable, the criteria of 10 CFR Part 2, Appendix C, were applied.

(Closed) LER 50-251/92-01, Technical Specification 3.0.3 Entry-Malfunction of Rod Cluster Control Position Indication.

The root cause of this event is that rod position indication varies with temperature. This temperature related variation was first identified at Turkey Point in 1972. In 1972 Westinghouse stated that temperature variations in indicators are inherent to the design of the system. Generally, the temperature dependent variation is corrected within the

one hour thermal/soak time allowed by TS 3.1.3.2. However, in this instance, after the one hour soak time two rods still indicated misalignment requiring entry into TS 3.0.3. The licensee took prompt corrective action to correct the misalignment which included performing incore flux maps and aligning the RPIs to indicate properly. In addition, the licensee has an ongoing task team to study this industry problem for plants of this vintage. The residents will continue to follow the efforts of this task team. This LER is closed.

7. Monthly Surveillance Observations (61726)

The inspectors observed TS required surveillance testing and verified that the test procedures conformed to the requirements of the TSs; testing was performed in accordance with adequate procedures; test instrumentation was calibrated; limiting conditions for operation were met; test results met acceptance criteria requirements and were reviewed by personnel other than the individual directing the test; deficiencies were identified, as appropriate, and were properly reviewed and resolved by management personnel; and system restoration was adequate. For completed tests, the inspectors verified testing frequencies were met and tests were performed by qualified individuals.

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

- 4-OSP-206.3, Inservice Valve Testing Hot Standby to Cold Shutdown, for charging pump discharge valve HCV-121 following maintenance;
- OP 1604.1, Full Length RCC - Periodic Exercise, for Unit 3;
- TP-846, Monitoring Rod Control for Rod Drop Cause; and
- EP-AD-007, Emergency Response Facilities and Equipment Surveillances, quarterly communication test of the FTS-2000 phone system.

The inspectors determined that the above testing activities were performed in a satisfactory manner and met the requirements of the TSs. Violations or deviations were not identified.

8. Monthly Maintenance Observations (62703)

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

The following items were considered during this review, as appropriate: LCOs were met while components or systems were removed from service; approvals were obtained prior to initiating work; activities were accomplished using approved procedures and were inspected as applicable; procedures used were adequate to control the activity; troubleshooting

activities were controlled and repair records accurately reflected the maintenance performed; functional testing and/or calibrations were performed prior to returning components or systems to service; QC records were maintained; activities were accomplished by qualified personnel; parts and materials used were properly certified; radiological controls were properly implemented; QC hold points were established and observed where required; fire prevention controls were implemented; outside contractor force activities were controlled in accordance with the approved QA program; and housekeeping was actively pursued.

The inspectors witnessed/reviewed portions of the following maintenance activities in progress:

- Overhaul and repair of the B AFW pump and turbine,
- replacement of the lube oil storage tank degraded piping, and
- non-code repair of 4A MSIV leaking drain line (see item b. below).

For those maintenance activities observed, the inspectors determined that the activities were conducted in a satisfactory manner and that the work was properly performed in accordance with approved maintenance work orders.

a. By memo dated June 23, 1992, the resident inspectors were requested to solicit EDG unavailability data from the licensee to help the NRC assess the safety significance of EDG unavailability information nationwide. The NRC desired EDG OOS time related to the reactor status (at power or shut down), OOS duration, and the reason for being OOS (scheduled PM, corrective maintenance or testing). This data was requested to be provided by July 9, 1992. The data requested was not readily available in the format noted above, however the licensee did provide for each EDG its 12 month average unavailability data from the first quarter of 1988 to present. The licensee is accumulating the data in the requested format but estimates it will be available approximately four weeks later than requested. The residents will forward the EDG unavailability data when available.

b. On June 16, 1992, a small steam leak was discovered on a 1/2 inch main steam drain line on the 4A MSIV body on the upstream side of the valve. This leak is the result of a through wall pinhole located at the toe of a weld on a pipe elbow. The leak location is unisolable from the 4A Steam generator and is subjected to full steam header pressure during plant operation. This discrepant condition was documented on NCR N-92-0125. The initial engineering assessment of operability provided the following evaluation:

> The discrepant 1/2" nominal diameter pipe weld to the 3/4" ninety degree elbow is part of the steam drain pipe from the

MSIV POV-4-2604 to steam trap ST-4-1. The subject piping is classified as Safety Related, Quality Group B per 5610-T-E-4061, Sheet 1, Rev. 72. The pin hole leak represents a minor reduction in plant efficiency but does not adversely affect the operability or reliability of POV-4-2604.

Additionally, the containment isolation scheme is defined as a closed system inside containment for this penetration. In the unlikely event of a steam generator tube rupture, the off site dose that would result from the pin hole leak is bounded by "Safety Evaluation for Operation with the Main Steam Sampling Valves Open for Post-Accident Monitoring", JPN-PTN-SEIJ-89-125. This analysis concluded that even with a passive failure of the non-seismic portion of the sample line tubing (3/8" diameter which is much larger than the pin hole), the results of calculation JPE-LR-87-033, Rev. 1, "Dose Calculation for S. G. Tube Rupture" are still acceptable and are within the bounds of the FSAR analysis.

Therefore, based on the above discussion, no operability concern exists.

The main steam system at this location has a design pressure of 1085 psig and a design temperature of 600 degrees F. Normal operating conditions are approximately 745 psig at 513 degrees F. As the piping system is a Class B pressure boundary, the normal Furmanite sealant encapsulation techniques can not be employed. However, Generic Letter 90-05 provides guidance for circumstances when a licensee can be granted approval of a non-code repair. The licensee is required to evaluate the leakage conditions and determine whether performing a code repair is practical. If the affected section of piping can be isolated for completing a code repair within the time period permitted by the limiting condition for operation without a plant shutdown, the licensee is required to perform a code repair.

The licensee determined that a code repair was not practical in that it would require a plant shutdown and cooldown with undue and unnecessary stress on facility systems and components in order to perform a code repair on the unisolable portion of the leaking pipe. Generic Letter 90-05 requires that temporary non-code repairs of code Class 1 and 2 piping must have load bearing capability similar to that provided by engineered weld overlays or engineered mechanical clamps. The non-code repair must be designed to meet the load-bearing requirements of the piping, assuming the flaw is completely through the wall for the circumference of the pipe at the location of the flaw. Additionally, the NRC must grant relief in order to perform temporary non-code repairs on a case by case basis. Engineering Evaluation JPN-PTN-SEMS-92-032, "Evaluation to Support Relief Request for Engineered Mechanical Clamp on Main Steam Drain Line Piping Elbow" was prepared to support a plant relief request

submitted to the NRC in FPL Letter L-92-185, dated June 26, 1992. A revised letter, L-92-206, was issued by the licensee on July 9, 1992, and a meeting was conducted at the NRC Headquarters Office between the licensee and NRR on July 9 and 10, 1992, concerning this issue. On July 10, 1992, FPL received verbal approval from the NRC to Proceed with the non-code repair of the MSIV drain line. The approval was granted with the understanding that the leak in question would be repaired in accordance with the code for Class 2 piping on the first occasion that Unit 4 reached Mode 5. Also, the non-code repair would be inspected by an engineer at least once per day until the code repair was completed. The resident inspectors will follow up on the installation of the mechanical clamp assembly.

Violations or deviations were not identified.

9. Operational Safety Verification (71707)

The inspectors observed control room operations, reviewed applicable logs, conducted discussions with control room operators, observed shift turnovers, and monitored instrumentation. The inspectors verified proper valve/switch alignment of selected emergency systems, verified maintenance work orders had been submitted as required, and verified 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. The implementation of radiological controls and plant housekeeping/cleanliness conditions were also 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/structures to verify proper valve/switch alignment:

- A and B emergency diesel generators,
- control room vertical panels and safeguards racks,
- intake cooling water structure,
- 4160-volt buses and 480-volt load and motor control centers,
- Unit 3 and 4 feedwater platforms,
- Unit 3 and 4 condensate storage tank area,
- auxiliary feedwater area,

- Unit 3 and 4 main steam platforms, and
- auxiliary building.

The licensee routinely performs QA/QC audits/surveillances of activities required under its QA program and as requested by management. To assess the effectiveness of these licensee audits, the inspectors examined the status, scope, and findings of the following audit reports:

Audit Number	Number of Findings	Type of Audit
QAO-PTN-92-025	0	TSs 6.1 and 6.2
QAO-PTN-92-028	0	Radwaste Handling & Shipment

No additional NRC followup action is required.

As a result of routine plant tours and various operational observations, the inspectors determined that the general plant and system material conditions were satisfactorily maintained, the plant security program was effective, and the overall performance of plant operations was good. Violations or deviations were not identified.

10. TI 2515/115, Verification of Plant Records

The licensee's efforts to conduct an independent review of the accuracy of plant records was previously discussed in IR 50-250,251/92-13. The licensee's findings indicated that some discrepancies in log keeping had occurred. These discrepancies were documented in QA's May Performance Monitoring Audit QAO-PTN-92-023. Based on the licensees' findings and Information Notice 92-30, Falsification of Plant Records, the licensee has initiated the following self-monitoring programs to ensure accurate plant records are being obtained.

- QA issued a change to surveillance checklist QCS-8, Surveillance of Plant Surveillance Testing, (which is performed quarterly) to verify, among other things, a TS required surveillance completed within the previous two weeks was properly performed and to verify by security computer printout that personnel entries into the required areas for completion of the TS surveillance were made.

- QA issued surveillance checklist QCS-17, Surveillance of Operation Log Taking and Watch Standing Activities, (which is performed quarterly) to verify operations logs were properly performed and to verify by security computer printout (or other means) that personnel entries into the required areas were made.

- Operations department issued procedure ODI-CO-008, Operator Rounds Verification, to provide for periodic review of operator logs to assure that all logs are factual. The log review is to be accomplished six times in each six month period and the reviewed

logs are to have been recorded within the previous 48 hours. Verification of personnel entries into required areas will be by security computer read out, posted guard logs, or HP records.

By implementation of the above self-monitoring programs, the licensee has an operations and a QA overview of operator log accuracy. QA also periodically samples other plant organizations for log/surveillance activity accuracy. This completes TI 2515/115.

Violations or deviations were not identified.

11. Plant Events (93702)

The following plant events were reviewed to determine facility status and the need for further followup action. Plant parameters were evaluated during transient response. The significance of the event was evaluated along with the performance of the appropriate safety systems and the actions taken by the licensee. The inspectors verified that required notifications were made to the NRC. Evaluations were performed relative to the need for additional NRC response to the event. Additionally, the following issues were examined, as appropriate: details regarding the cause of the event; event chronology; safety system performance; licensee compliance with approved procedures; radiological consequences, if any; and proposed corrective actions.

a. On June 28, 1992, at 8:00 p.m., the licensee's security personnel received a report from the Air Force that an aircraft from Homestead Air Force Base had jettisoned two 750-gallon fuel tanks and two inert practice bombs west of the plant. The aircraft was experiencing an emergency situation which required the above action. On June 29, 1992, the Air Force found the jettisoned fuel tanks in an open area approximately five miles west of the plant. Licensee personnel contacted Homestead Air Force personnel regarding flights near the Turkey Point facility and were provided a copy of a Flight Information Publication dated June 25, 1992, for area planning. This document contained a caution for Homestead Air Force Base that stated, "Do not overfly Turkey Point Nuclear/Conventional Power Plant." No further action is considered necessary.

b. At 8:30 p.m. on July 9, 1992, approximately 15 minutes after the completion of a resin transfer from the SRST to the primary HIC, the control room was notified of water flowing out of the south cask room in the radwaste building. At 2:00 a.m. on July 10, 1992, the licensee identified the source of the leakage to be the flush line between the primary and secondary HICs. One end of the flush line had been connected to the fill/divert valve; and an open end of a second identical hose, which was loosely coiled on the floor with the remainder of the flush line, had been connected to the secondary HIC leaving the open end of the flush line from the fill/divert valve unterminated. In addition, the continuity of this portion of the system was not verified prior to the resin

transfer from the SRST to the primary HIC and the flush of residual spent resin to the secondary HIC via the fill/divert valve. As a result, approximately 1 cubic foot of residual spent resin and 100 to 250 gallons of flush water leaked onto the floor of the south cask room between the primary and secondary HICs at a rate of approximately 50 gpm. The liquid overfilled the drainage troughs and flowed onto the floor and into the truck bay of the radwaste building. The leakage was successfully contained within the radwaste building.

As a result of this event, approximately 4500 square feet of floor space in the radwaste building was contaminated. Although the average activity in the contaminated areas ranged from 5,000 dpm to 10,000 dpm, a few small areas were identified to be greater than 100,000 dpm. Cleanup of the contaminated areas is in progress.

TS 6.8.1 requires that written procedures be established, implemented, and maintained covering the activities referenced in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978. Paragraph 7 of Regulatory Guide 1.33, Appendix A, Revision 2, February 1978, recommends that implementing procedures be written for the control of radioactivity for limiting materials released to the environment and for limiting personnel exposure. Item b(1) of this paragraph specifies the need for procedures pertaining the solid waste system including the handling of spent resins and filter sludge. Paragraph 8.1.2.13 of procedure O-HPS-042.5, Transfer and Dewatering Bead Resin in RADLOCK High Integrity Containers, requires that the flush line be connected to the fill/divert valve and routed to the secondary HIC. Paragraph 8.1.2.14 of this procedure requires that the open end of the flush line be connected to the standpipe on the fill container of the secondary HIC. Contrary to these requirements, on July 9, 1992, procedure O-HPS-042.5 was not followed in that the flush line was not connected to the secondary container prior to the transfer of spent resin from the SRST to the primary HIC and prior to the collection of residual spent resin and flush water in the secondary HIC. This failure to follow a procedure constitutes a violation and will be tracked as VIO 50-250,251/92-16-01, failure to follow a procedure resulting in a spill of flush water and residual spent resin in the radwaste building. In addition to this violation, the NRC is concerned that the radwaste building drainage system may not be adequate to prevent any future spills from reaching the environment and considers this a weakness.

One violation and one weakness were identified.

12. TMI Item II.F.2.4., Install Additional Instrumentation for Detection of Inadequate Core Cooling

By letter from the NRC to FPL dated January 28, 1985, NUREG-0737, Item II.F.2, Inadequate Core Cooling Instrumentation (ICCI) System, was

approved based on FPL's submittals in response to GL 82-28. Previously, IR 50-250,251/82-06 and IR 50-250,351/85-30 addressed the status of TMI Action Item II.F.2.3B. TI 2515/065, TMI Action Plan Requirement Followup, Appendix A, Enclosure 1, identifies that Item II.F.2.3 has been renumbered to Item II.F.2.4. TMI Action Item II.F.2.4, Instrumentation for Detection of Inadequate Core Cooling, consists of the following instrumentation at TPNP;

- Subcooling Margin Monitor,
- RVLMS, and
- CETs.

NRC's approval of FPL's submittal, as incorporated in the safety evaluation, was based on the following actions being accomplished:

- upgrading of the vital inverters,
- interconnecting the QSPDS to the SAS,
- implementation of revised procedures for the ICCI, and
- TS changes for the RVLMS.

PC/M No. 83-117, Replacement of 120V Vital AC Plant Inverters - Units 3 and 4, was completed in July, 1987. This modification replaced the existing vital inverters with a new vital inverter system designed to automatically transfer to an alternate power source upon loss of inverter AC output or during overload conditions. The alternate source equipment was installed under PC/Ms 84-205 and 84-202 for Unit 3 and Unit 4 respectively. The previous 120 volt vital AC inverter system did not have the capability to automatically transfer to an alternate supply. The inspectors verified the noted PC/Ms had been accomplished and the QSPDS is powered from the new 120V vital AC inverters.

PCM 81-162 and PCM 81-167, for Units 3 and 4 respectively, provided for the fiber optic cable connection between the QSPDS and ERDADS. (SAS has become known as SPDS which is part of ERDADS). These PCMs have been completed.

The inspectors reviewed the licensee's procedures and verified that core subcooling based on core exit thermocouples and RVLMS were incorporated into the EOPs. Also, a special announced EOP team inspection was conducted in February, 1990 (IR 50-250,251/89-53) and the EOPs were found to be satisfactory and in conformance with the Westinghouse Owner's Group Emergency Response Guidelines. Also, during emergency exercise drills, the QSPDS has been utilized to obtain ICCS data.

By letter from the NRC to FPL dated July 28, 1987, the NRC issued Amendment No. 125 and Amendment No. 119 to facility operating licenses DPR-31 and DPR-41 respectively. These amendments incorporated plant

specific TSs for the RVLMS. The licensee converted to standard TS in 1991 and the inspector verified that the new TSs accurately reflect the RVLMS as previously approved.

This completes TMI Action Item II.F.2.4.

NUREG-1435, Volume 1, Status of Safety Issues at Licensed Power Plants (TMI Action Plan Requirements), listed two TMI Action Items as open and requiring verification. Those two items were II.E.4.1.2, Dedicated Hydrogen Penetrations, and II.F.2.4, Additional Instrumentation for Detection of Inadequate Core Cooling. IR 50-250,251/91-32 closed TMI Items II.E.4.1.2 and II.E.4.1.3. Based on the closure of TMI Action Item II.F.2.4 above, this completes all the open TMI Action Items that require verification. Therefore, TI 2515/065, TMI Action Plan Requirement Followup, is complete and considered closed.

Violations or deviations were not identified.

13. Preparation For Refueling (60705)

On July 8 and 10, 1992, the first shipment of new fuel assemblies for the Unit 3 August 24 refueling outage was received onsite and off-loaded into the new fuel room. The inspector attended the pre-receipt briefing held by Reactor Engineering which covered the revisions to O-OP-040.1, Handling New Fuel Shipping Containers and New Fuel Assemblies, as well as, the schedule of activities for the upcoming weeks of fuel shipment. The inspectors witnessed portions of the off-loading of the new fuel shipping containers from the truck and the subsequent transfer of the new fuel from the shipping containers to the new fuel room storage racks. The following groups were present and provided coverage during this off-loading process: Operations, Quality Assurance, Health Physics, Security, and Reactor Engineering. The receipt and handling of the new fuel assemblies was performed in accordance with O-OP-040.1, Handling New Fuel Shipping Containers and New Fuel Assemblies and O-OSP-040.11, Receipt of New Fuel. Operations personnel controlled the new fuel movement process, while Reactor Engineering and Health Physics performed inspections of the shipping containers and new fuel, once the containers were opened, and recorded the results as per the above controlling procedures. This recorded data was then compared to the shipping data for agreement. All evolutions in this receipt process were well handled and coordinated by the various personnel involved.

On July 17 and 20, 1992, the inspectors witnessed portions of the off-loading process for the second shipment of new fuel received on site. This process included the transfer of the new fuel elements from the shipping containers to the new fuel room storage racks and the subsequent transfer on the 20th from the new fuel storage racks to interim storage in the spent fuel pool for the August RFO. This final transfer to the SFP was done in accordance with the same procedures used for the receipt and transfer of the new fuel, O-OP-040.1, Handling New Fuel Shipping Containers and New Fuel Assemblies and O-OSP-040.11, Receipt of New Fuel. The process was supported by Operations, Quality

Assurance, Health Physics, Security, and Reactor Engineering personnel, who handled and coordinated these receipt/transfer evolutions in a proficient manner.

Violations or deviations were not identified.

14. Unit 3 Containment Building Tendon Surveillance (61700)

A regional specialist inspector examined procedures and quality records and observed work activities covering the Unit 3 20-year containment building tendon surveillance inspection. Acceptance criteria utilized by the inspectors appear in TS 3/4.6.1.6.

a. Review of Tendon Surveillance Procedures

The inspector examined the following procedures which control the tendon surveillance activities:

(1) FPL procedure number 0-SMM-051.2, Containment Tendon Inspection.

(2) Bechtel procedure number 21701-539-CP-1, Technical Requirements for Tendon Surveillance for Containment Structure Post-Tensioning System - Unit 3.

The inspector verified the procedures specified precautions and limitations, predicted lower limit prestress forces, normalizing factors, tendon wire inspection and testing requirements, grease sampling and testing requirements, stressing ram and gage calibration requirements, and witness/inspection hold points.

b. Observation of Tendon Surveillance Activities

The inspector witnessed the stressing operation for determination of the lift-off forces in horizontal tendon numbers 53H51 and 53H53. These measurements were made in accordance with TS 4.6.1.6.1.a which requires checking the lift-off forces in adjacent tendons when the selected surveillance tendon had a measured lift-off force of less than the predicted prestress lower limit value. The selected surveillance tendon, number 53H52, had a measured lift-off value 3.5% below the predicted lower limit. The measured prestress forces in tendons 53H52 and 52H53 were 4.7% and 3.6%, respectively, below the predicted lower limit values. The measured lift-off values were corrected using the appropriate normalizing factors per procedural and Regulatory Guide 1.35.1 criteria. The licensee entered a 15 day LCO per TS 3.6.1.6.a since more than 2 tendons had a lift-off force less than the predicted lower limit. The inspector witnessed stressing operations for restressing of tendon numbers 53H52 and 53H53 to above the predicted lower limit values. The tendons were restressed to approximately 102% of the predicted lower limit value and the LCO was cleared approximately 24 hours after it was

entered. The inspector examined the anchorage assemblies on tendons 53H51, 53H52, and 53H53. No problems were identified. The inspector also witnessed sampling of the tendon corrosion protection materials (grease) from tendon 53H51 and 53H53 prior to performance of the tendon lift-off measurements. The grease will be tested to determine the chemical properties of the grease per the requirement of TS 4.6.1.6.1.e.(2).

c. Review of Tendon Surveillance Quality Records

The inspector examined the records documenting tendon lift-off forces in tendon numbers 13H32, 13H33, 13H34, 64H60, 64H61, and, 64H62. The measured forces in these tendons averaged 96% of the predicted lower limit for the tendons. The licensee entered 15 day LCOs on June 15, 1992 for the low lift-off forces measured in tendons 13H32, 13H33, 13H34 and on June 19, 1992 for the low lift-off forces measured in tendon 64H60, 64H 61, and 64H62. The licensee restressed these six tendons to a stress level above the 20 year surveillance predicted lower limit value and cleared each LCO within the 15 day limit. TS 3.6.1.6.c also requires the licensee to submit a Special Report within 30 days to the NRC to address the low lift-off value. The licensee submitted the special report to the NRC in a letter dated July 10, 1992. The inspector reviewed the report which addressed the low lift-off forces measured in tendons 13H33, 64H61, and 35H52 and their adjacent tendons. The measured lift-off forces in the 4 vertical and 4 dome surveillance tendons exceeded the predicted lower limit values. The licensee concluded that the level of prestress force available in the hoop tendons is sufficient to maintain containment integrity. The licensee committed to submit a more detailed report to NRC by September 15, 1992, following completion of the Unit 3 tendon surveillance. The inspector also reviewed the following documents pertaining to the Unit 3 tendon surveillance.

(1) Records documenting lift-off forces measured in vertical tendons 12V22, 34V11, 45V29 and 56V21. The measured prestress values were above the predicted lower limits for all four vertical tendons.

(2) Nonconformance Report Number N-92-0112, 0117, 0123, 0128, and 0136.

(3) Calculation number C-SJ-539-02, Determination of Lift-off Force for 20th year Tendon Surveillance.

d. Conclusion

Subsequent to completion of the onsite review by the regional specialist inspector and submittal of the July 10, 1992 Special Report, the licensee completed the Unit 3 tendon surveillances. One of the two remaining hoop tendons exceeded the predicted lower

limit while the other hoop tendon and its two adjacent tendons were below the predicted lower limit. These three tendons were restressed to a force above the predicted lower limit. A review of FSAR Section 5.1.4 disclosed that the measured prestress force in the hoop tendons are greater than the minimum value required for design basis accident condition internal containment pressure of 49.5 psig. This pressure is specified in the Basis for TS 3/4.6.1.6. Although the prestress (lift-off) force measured in the hoop tendons were below the predicted lower limits for the 20 year surveillance, none of the measured prestress forces were below the minimum design limits. The prestress force levels measured in the horizontal tendon are not an indication of abnormal degradation of the structural integrity of the Unit 3 containment, and thus are not a concern for continued operation at this time. However, the licensee plans to conduct a detailed study to determine the reason for the apparent accelerated loss of prestress force in the horizontal tendons. The results of this study will be submitted to the NRC in the September 15, 1992 Special Report. The licensee plans to begin the tendon surveillance inspection on the Unit 4 containment in August, 1992.

The inspector determined that the licensee's Unit 3 tendon surveillance inspections were performed in a satisfactory manner and complied with the requirements of the Technical Specifications.

Violations or deviations were not identified.

15. Exit Interview (30703)

The inspection scope and findings were summarized during management interviews held throughout the reporting period with the Plant General Manager and selected members of his staff. An exit meeting was conducted on July 24, 1992. The areas requiring management attention were reviewed. The licensee did not identify as proprietary any of the materials provided to or reviewed by the inspectors during this inspection. Dissenting comments were not received from the licensee. The inspectors had the following findings:

Item Number	Description and Reference
50-250,251/92-16-01	VIO - Failure to follow a procedure resulting in a spill of flush water and residual spent resin in the radwaste building (paragraph 11.b).
50-250,251/92-16-02	NCV - Failure to install seal table moveable cart in accordance with design drawing requirements (paragraph 5.a).

Weakness	Adequacy of the radwaste building drainage system in preventing any future spills from reaching the environment (paragraph 11.b).

16. Acronyms and Abbreviations

AC	Alternating Current
AD	Administrative Directive
AFW	Auxiliary Feedwater
CCTV	Closed Circuit Television
CET	Core Exit Thermocouple
CFR	Code of Federal Regulations
CO	Conduct of Operations
dpm	Disintegrations Per Minute
EDG	Emergency Diesel Generator
EOP	Emergency Operations Procedure
EP	Emergency Procedure
ERDADS	Emergency Response Data Acquisition Display System
F	Fahrenheit
FPL	Florida Power & Light
FSAR	Final Safety Analysis Report
FTS	Federal Telecommunications System
GL	Generic Letter
HCV	Hydraulic Control Valve
HIC	High Integrity Container
HP	Health Physics
HPS	Health Physics Surveillance
ICCI	Inadequate Core Cooling Instrumentation
ICCS	Inadequate Core Cooling System
IR	Inspection Report
KV	Kilovolt
LCO	Limiting Condition for Operation
LER	Licensee Event Report
MSIV	Main Steam Isolation Valve
NCR	Non-conformance Report
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
ODI	Operations Department Instruction
OOS	Out of Service
OP	Operating Procedure
OSP	Operations Surveillance Procedure
PC/M	Plant Change/Modification
PM	Preventive Maintenance
POV	Power Operated Valve
psig	pounds per square inch gage
PTN	Plant Turkey Nuclear
QA	Quality Assurance
QAO	Quality Assurance Organization
QC	Quality Control
QCS	Quality Control Surveillance
QSPDS	Qualified Safety Parameter Display System

RII	Region II
RCA	Radiation Control Area
RCC	Rod Control Cluster
RFO	Refueling Outage
RPI	Rod Position Indication
RVLMS	Reactor Vessel Level Monitoring System
SAS	Safety Assessment System
SFP	Spent Fuel Pit
SG	Steam Generator
SPDS	Safety Parameter Display System
SRST	Spent Resin Storage Tank
ST	Steam Trap
TI	Temporary Instruction
TMI	Three Mile Island
TP	Temporary Procedure
TPCW	Turbine Plant Cooling Water
TS	Technical Specification
TSA	Temporary System Alteration
V	Volt
VIO	Violation