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

Report Nos.: 50-250/87-14 and 50-251/87-14

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

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

Facility Name: Turkey Point 3 and 4

Inspection Conducted: March 9 - April 27, 1987

Inspectors: _____ 5/20/87
D. R. Brewer, Senior Resident Inspector Date Signed

_____ 5/20/87
K. W. Van Dyne, Resident Inspector Date Signed

_____ 5/20/87
J. B. Macdonald, Resident Inspector Date Signed

_____ 5/20/87
P. J. Fillion, Reactor Inspector Date Signed

Approved by: _____ 5/20/87
B. Wilson, Section Chief Date Signed
Division of Reactor Projects

SUMMARY

Scope: This routine, unannounced inspection entailed direct inspection at the site, including backshift inspection, in the areas of annual and monthly surveillance, maintenance observations and reviews, operational safety, plant events, previously existing issues reviews, and International Atomic Energy Agency (IAEA) program reviews.

Results: Violations - Failure to meet the requirements of 10 CFR 50, Appendix B, Criterion III, Failure to meet the requirements of TS 3.10.1, 3.10.2, 3.10.6, and 6.2.2.e; Failure to meet the requirements of TS 6.8.1; and failure to meet the requirements of 10 CFR 50, Appendix B, Criterion VI.

REPORT DETAILS

1. Persons Contacted

Licensee Employees

*C. M. Wethy, Vice President - Turkey Point
*C. J. Baker, Plant Manager-Nuclear - Turkey Point
*F. H. Southworth, Maintenance Superintendent - Nuclear
D. A. Chaney, Site Engineering Manager (SEM)
*D. D. Grandage, Operations Superintendent and Acting Plant Manager
T. A. Finn, Operations Supervisor
J. Webb, Operations - Maintenance Coordinator
J. W. Kappes, Performance Enhancement Coordinator
R. A. Longtemps, Mechanical Maintenance Department Supervisor
D. Tomasewski, Instrument and Control (IC) Department Supervisor
J. C. Strong, Electrical Department Supervisor
*W. Bladow, Quality Assurance (QA) Superintendent
R. E. Lee, Quality Control Inspector
*M. J. Crisler, Quality Control (QC) Supervisor
*J. A. Labarraque, Technical Department Supervisor
R. G. Mende, Reactor Engineering Supervisor
*J. Arias, Regulation and Compliance Supervisor
R. Hart, Regulation and Compliance Engineer
*W. C. Miller, Training Supervisor
P. W. Hughes, Health Physics Supervisor
*G. Solomon, Regulation and Compliance Engineer
*J. Donis, Engineering Department Supervisor
J. J. Zudans, Nuclear Engineering, Human Factors Performance
R. L. Wade, Engineering Department
W. J. Pike, Safety Engineering Group Engineer
*D. E. Meils, Chemistry Department Supervisor
*D. W. Haase, Safety Engineering Group Chairman
*J. P. Mendieta, Services Manager, Nuclear

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

*Attended exit interview on April 27, 1987.

2. Exit Interview

The inspection scope and findings were summarized during management interviews held throughout the reporting period with the Plant Manager-Nuclear and selected members of his staff. An interim exit meeting was conducted on April 27, 1987. The areas requiring management attention were reviewed. The licensee acknowledged the findings and requested that the violations be evaluated for mitigation as described in 10 CFR 2, Appendix C.

Four violations were identified:

- Failure to meet the requirements of 10 CFR 50, Appendix B, Criterion III, in that design inputs were not correctly translated into design output documents (paragraph 8).

- Failure to meet the requirements of TS 3.10.1, 3.10.2, 3.10.6 and 6.2.2.e prior to the lifting of the Unit 4 reactor vessel upper internals (paragraph 9).

- Failure to meet the requirements of TS 6.8.1, in that Operations personnel failed to follow procedure for relocating secondary sources within the Unit 3 Spent Fuel Pool (SFP) (paragraph 9).

- Failure to meet the requirements of 10 CFR 50, Appendix B, Criterion VI, in that an electrical wiring diagram did not reflect as built conditions (paragraph 8).

3. 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 that the responses were timely and that corrective actions were implemented within the time periods specified in the reply.

(Closed) Violation 250/84-23-03, 251/84-24-03, Failure to originate records. This violation documents a failure to properly document the disposition of deficiencies identified during the 'A' Emergency Diesel Generator operability testing per OP 4304.1 on July 26, 1984. 0-OSP-023.1 has replaced OP 4304.1 and incorporates corrective action based on this violation.

(Closed) Violation 250, 251/85-20-02 Preoperational Procedure 0800.55 Was Not Properly Implemented. The licensee acknowledged the violation and had added QC hold points to Preoperational Procedure 0800.55 to ensure procedure compliance. In addition, departmental guidelines were developed (AP 0109.1) to give guidance for developing future preoperational procedures which require the support of expertise not available within the startup department. This item is closed.

(Closed) Violation 250, 251/85-20-01, Failure to Establish Adequate Procedures for Loss of Vital Instrument Power. The licensee has upgraded procedures 4-ONOP-003.6 and 3-ONOP-003.6, Loss of 120V Vital Instrument Panels, to include the actions necessary to cope with the failures identified in the violation. The licensee also issued Training Briefs 71 and 78 to identify several items of importance to operations and to provide additional information for selector switch alignments. Training on these procedures was conducted for the operating crews. This item is closed.

(Closed) Violation 250, 251/86-09-04, Failure to Conduct An Independent Fire Protection and Loss Prevention Inspection and Audit for 1985. The licensee revised the policy for fire protection audits, placing full scheduling and coordinating responsibility under the Quality Assurance Department. Procedure QI 18 QAD 3 was updated to reflect this change. An audit of the Fire Protection and Loss Prevention Program was completed on June 11, 1986. This item is closed.

(Closed) Violation, 250, 251/86-09-05, Excessive Time Between Shift Fire Brigade Drills. Procedure AP 15500, Fire Protection Program, Section 8.1.3 and 10 CFR 50 Appendix R Section III.1.3.b requires fire brigade drills to be performed at regular intervals not to exceed three months for each shift fire brigade team. The three month interval was incorrectly interpreted to mean one in a quarter. This interpretation allowed a drill to be held at the beginning of one quarter and at the end of the subsequent quarter which could exceed the three month interval. The licensee has revised AP 15500 to perform fire drills at regular intervals not to exceed three months for each shift fire brigade. This item is closed.

4. Followup on Unresolved Items (URIs), Inspector Followup Items (IFIs), Inspection and Enforcement Informaiton Notices (IENs), IE Bulletins (IEBs) (information only), IE Circulars (IEC), and NRC Requests (92701)

A review was conducted of the following items to assure that adequate applicability reviews were performed, appropriate distribution was made, and if required, adequate and timely corrective actions were taken or planned.

(Open) URI 250,251/86-33-07, Adequacy of Fuse Control Program Pending Completion of Fuse Evaluations. The licensee initiated a fuse inspection and replacement program after discovering, in January 1986, that the 3C Main Steam Isolation Valve control circuit had improperly sized fuses. This program was subsequently given higher priority when, in June 1986, two of three Emergency Containment Filters were found to have improperly sized fuses. Upon completion of the inspection and if necessary fuse replacements, results were to be consolidated and evaluated for possible safety significance. A review of the licensee's actions has revealed that numerous fuses were replaced in the February to July 1986 time frame, but no evaluation of the as-found condition had been performed. This unresolved item remains open pending completion of the licensee's evaluation of the safety significance of the as-found condition.

(Closed) URI 250, 251/86-09-02, Verify Surveillance Testing of Electric Fire Pump. From September 1, 1985 to December 31, 1985, the electric fire pump surveillance was performed twice. As determined by a search of records in Document Control, the first surveillance was done on September 3, 1985, and the next was on December 3, 1985. No records could be found to verify a surveillance was performed during the months of October or November. The licensee reported this problem through Licensee

Event Report (LER) 250/86-20. This LER was reviewed and disposition as closed in inspection report 250, 251/86-39. The above report also identified other missed surveillances which are being tracked under violation 250, 251/86-39-02. This item is closed.

(Closed) IFI 250,251/86-44-04, Repair and Replacement Programs. This item identified the fact that the licensee did not have a unified repair and replacement program in place. The licensee has issued Administrative Procedure AP-0190.89, ASME Section XI Repair/Replacement, which adequately addresses the inspectors concerns. This item is closed.

(Closed) IFI 250,251/84-09-01, Circuit Breaker Material Deficiencies. This item identified material deficiencies within the inner workings of several 3C and 4C bus breakers. Disassembly and inspection of these breakers was completed in March, 1984, with the assistance of the Vendor Technical Representative. Inspection results are documented in Plant Work Orders 7352 and 7353. This item is closed.

(Closed) IFI 250/84-23-13 and 251/84-24-13, Review of Special Allowances for A Single Component Cooling Water (CCW) Train Failure. The licensee had Bechtel Power Corporation conduct a CCW flow balancing safety assessment for both units. The results of the assessment were submitted to NRR for review. NRC letter dated February 5, 1987, to Mr. C. O. Woody, identifies that the NRC staff members, both Regional and Headquarters witnessed the tests; have audited the results; and agree that the CCW system is balanced and the safety-related components are being cooled per their design requirements. This item is closed.

(Closed) IFI 250/84-23-11 and 251/84-24-11, Discrepancies Between the Data Used to Install Magnetrol LS-3-1584 A, Under Plant Change/Modification (PCM) 80-100. The control room curve book which is used by the control room operators has been upgraded to a control document. This item was identified as IFI 250/84-23-12 and 251/84-24-12 and was closed in inspection report 250, 251/87-10. The drawing on the water storage tank in the control room curve book is used as a reference and is not intended to show specific instrument nozzle locations. Therefore, the drawing does not need to be updated. The curve utilizes two scales: one for the operations people, and one for the I&C technicians. The I&C curve book is also a controlled document. There is a difference in the bottom level tank indication for the I&C scale to the actual bottom of the tank used by the operatins scale. Since the control room cureve book is now a control document and the drawing is not required, this item is closed.

(Closed) IFI 250/84-23-14 and 251/84-24-14, Evaluation of Corrective Action for Repeated Pressure Relief Tank (PRT) Alarms. The licensee performed a thermodynamic analysis of the PRT to determine if raising the upper limit of the operating bands for temperature and pressure would effect the ability of the PRT to perfrom its design function. This function as defined in FSAR chapter four is being able to condense and

cool a discharge of pressurizer steam equivalent to 110% of the volume above the full power setpoint. The analysis concluded that the existing greenbands were extremely conservative. Therefore, an increase in the temperature and pressure alarm setpoints to 250°F and 30 psig from 125°F and 10 psig, respectively, would not decrease its effectiveness as a quench tank, and would still provide a considerable margin to the maximum acceptable tank pressure and temperature. The inspector reviewed the updated Operating Procedure OP 041.3, Pressurizer Relief Tank, and Administrative Procedure 0140.2, Changing Setpoints, to verify that the setpoint limits for the PRT have been changed. This item is closed.

(Closed) IFI 250, 251/86-18-09, Combine Scaffold Control Procedures. The control of scaffolding by a single procedure, combining ASP-29 and AP 0103.11, was determined not to be feasible. However, ASP-29 was changed to require that the Plant Supervisor-Nuclear approved scaffolding which will be used in excess of thirty days. This item is closed.

(Closed) IFI 250, 251/86-24-02, Failure to Adequately Correct Deficiency in EOP. Attachment E of 3/4-EOP-E-0, Reactor Trip or Safety Injections, was writtern to provide operator guidance for a loss of offsite power without a safety injection and which would not potentially overload the EDG. Attachment E has been deleted with the subsequent implementation of PCMs to prevent EDG overloading. This item is closed.

(Closed) IEC 250,251/80-CI-04, Securing of Threaded Locking Devices on Safety-Related Equipment. This circular is directly applicable to both units at Turkey Point. LER 250/79-31 and LER 251/80-07 describe instances where Main Steam Check Valve (MSCV) disc stud nuts separated from the studs. A new type of locking device was installed on the MSCV stud nuts. Subsequent inspections revealed, however, that this new locking device was inadequate to prevent disengagement of the MSCV stud nuts. In December 1980, on Unit 4 and in October 1985, on Unit 3 the MSCV stud nuts were welded to the studs as a positive means of assuring nut retention. A safety evaluation completed on January 30, 1987, determined that although previous nut retention methods were inadequate to prevent nut disengagement and constituted a substantial safety hazard, welding of the MSCV stud nuts to the studs adequately addressed the operability of the MSCVs. Inspection of a Unit 3 MSCV in the current refueling outage has revealed no stud nut operability concerns and supports engineering's January 30, 1987, safety evaluation. Additionally, inspection of the Main Steam Isolation Valves have revealed no operability concerns related to locking devices. Maintenance procedures for other safety-related equipment, including the High Head Safety Injection System, were reviewed to ensure securing of locking devices is adequately covered. This circular is closed.

(Closed) IEC 250,251/80-CI-12, Valve-Shaft-To-Actuator Key May Fall Out of Place When Mounted Below Horizontal Axis. This circular identifies a discrepant condition involving valve shaft actuator connections. Licensee review of keyway connections for butterfly valves at Turkey Point

indicates that because keyway connections are oriented above the horizontal they are not susceptible to the failure mode discussed in the circular. This circular is closed.

(Closed) IEC 250,251/80-CI-16, Operational Deficiencies in Rosemount Model 510DU Trip Units and Model 1152 Pressure Transmitters. This circular identifies common mode failures for Rosemount Model 510DU Trip Units and Model 1152 Pressure Transmitters which could result in a failure to automatically activate a safety function. Licensee review indicates that no Rosemount Model 510DU Trip Units nor Model 1152 Pressure Transmitters are installed in any safety-related nor quality - related applications. This circular is considered closed.

(Closed) IEC 250, 251/79-CI-23, Motor Starters and Conductors Failed to Operate. The Gould, Inc., Electrical Products Group, reported to the NRC of potential defects in certain NEMA size 3, starters and conductors. The licensee conducted a review and check of all their starters and conductors and none were found. Therefore, the Circular is not applicable. This item is closed.

(Closed) IEC 250, 251/79-CI-10, Pipefitting Manufactured From Unacceptable Material. Liberty Equipment and Supply Company of Kennewick, Washington and Tube Turns Division, Chermtron Corporation reported that some potentially defective pipe fittings were manufactured from material with a high carbon content beyond the range allowed by ASTM A-234 WPB. The licensee reviewed their records and purchase orders and found no purchase orders were written for Liberty Equipment and Supply Company or Tube Turn Division. This item is closed.

(Closed) IEC 250, 251/80-CI-22, Confirmation of Employee Qualification. The IE Circular referenced to two unrelated instances where individuals' qualifications were not properly reviewed by personnel prior to employment. The licensee confirmed that their personnel office contacts all previous employers of new hires and receives a copy of their college transcript. This item is closed.

(Closed) IEC 250, 251/80-CI-07, Problems with HPCI Turbine Oil System. Although this Circular was addressed to BWRs, the licensee verified that they have similar turbine lube oil systems. They also confirmed that they had no water leakage. Turbine lube oil samples are collected quarterly and sent to the power resources lab for analysis. No problems were identified. This item is closed.

(Closed) IEC 250, 251/80-CI-05, Emergency Diesel Generator (EDG) Lubricating Oil Addition and Onsite Supply. The licensee reviewed the recommended actions in the IE Circular and concluded that they have an oil inventory maintained by stores, minimum ten drums (50 gallons) and maximum twenty drums (100 gallons). Using maximum leak rate, they concluded that this is more than enough oil to support a seven day run. The manufacturer's recommended procedure is used to address adding lubricating oil and type of oil to be used. This item is closed.

5. Unresolved Items

Unresolved items are matters about which more information is required to determine whether they are acceptable or may involve violations or deviations. No unresolved item were identified during this inspection period.

6. Onsite Followup of Written Reports Of Nonroutine Events (92700)

The Licensee Event Reports (LERs) discussed below were reviewed and closed. 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 that the licensee had reviewed the 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.

LER 250/87-05	LER 251/86-05	LER 251/86-22	LER 251/87-05
LER 250/87-08	LER 251/86-06	LER 251/86-26	LER 251/85-29
LER 251/86-11	LER 251/86-27	LER 251/86-04	LER 251/86-12
LER 251/86-30			

The LERs listed above were generated as a result of containment and control room ventilation automatic isolations caused by spurious high radiation spikes of the Process Radiation Monitor System (PRMS). The PRMS power supply drawers are overly sensitive to any electrical perturbations which could occur during I&C troubleshooting, calibration evolutions and/or from the use of two way radios in the vicinity of the drawers. The power to the drawers is supplied in series, making every drawer susceptible to a single electrical disturbance. The licensee is currently in the process of replacing the PRMS drawers. The new drawers are less susceptible to external electrical interference. The series power supplies have been modified to make each drawer electrically independent thereby greatly reducing the problems previously incurred during previous troubleshooting and maintenance activities. The licensee plans to have completed the PRMS modifications to both units prior to restart from the present dual unit outage. The inspectors will address the status of the replacement effort in Inspection Report 250, 251/87-22. The LERs listed above are closed.

LER 250/86-39
LER 250/87-09
LER 250/87-10

The LERs listed above were generated as a result of automatic and manual reactor trips caused by continuing problems with the turbine governor control oil system. Dirt and other impurities had accumulated in the system. The impurities deposited on the restricting orifices within the

governor and auxiliary governor piping, changing the geometry of the orifices and affecting their pressure controlling functions. The system becomes erratic and unreliable when the governor pressure drops below the normal operating pressures of 26-30 psig. When the dirt in the orifice breaks free the pressure increases rapidly from below the normal range (approximately 24 psig) to its normal pressure of slightly greater than 27 psig. The auxiliary governor interprets the sudden pressure increase (approximately 3% per second) as a sudden increase in turbine speed and it immediately starts dumping control oil. The turbine control oil, lube oil and seal oil piping system were completely flushed and filtered during the current Unit 3 refueling outage. Based on the total system clean and flush, the LERs listed above are closed.

LER 250/79-31
LER 251/80-07

The LERs listed above were generated as a result of deficiencies described in IEC 80-04. The LER corrective actions were reviewed with regard to the recommended actions of the circular. IEC 80-04 was addressed and closed in paragraph 5. The LERs listed above are closed.

(CLOSED) LER 251/86-24, Unit Shutdown Due to Missed Post Maintenance Testing of Containment Isolation Valves. Violation 251/86-41-01, issued in Inspection Report 250, 251/86-41, paragraph 6 delineates the event and subsequent violation cited as a result of the event. The licensee response to the Notice Of Violation will be tracked to ensure that all corrective actions related to this event are identified and properly implemented. LER 251/86-24 is closed.

(CLOSED) LER 250/86-32, Reactor Trip Due to Lightning. The inspectors reviewed the LER and documented the event in Inspection Report 250, 251/86-35, paragraph 11. The inspectors concur with the licensee's conclusions. No further action is required. LER 250/86-32 is closed.

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

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

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

B EDG 18 Month Overhaul
B EDG 8 Hour Load Test and Full Load Rejection
B EDG Lockout Relay Testing
EDG Fuel Oil Storage Tank Cleaning and Inspection O-PMM-022.4
Reactor Coolant System Leak Rate Calculations
Control Room Ventilation System Automatic Power Transfer Testing

On January 16, 1987, the NRC issued IE Notice 87-04, entitled Diesel Generator Fails Test Because of Degraded Fuel, documenting the failure of an EDG due to fuel line fouling in a strainer between the day tank and the engine. The interior surface of the day tanks was found to have coatings of sludge which could easily be removed. The licensee is evaluating this IE Notice for improvements which could enhance fuel oil and tank cleanliness.

The EDG fuel oil storage tank had not been cleaned during the past 15 years. On January 28, 1987, the EDG fuel oil storage tank was declared out of service when a periodic sample revealed unacceptable levels of water and sediment. Subsequently, backup samples verified that water and sediment levels, except for the extreme lower level, were acceptable. Licensee Event Report 87-007 for Unit 3 was issued on February 27, 1987 documenting the discrepancy. Short term corrective actions included a partial tank flush. Long term corrective actions included a licensee decision to evaluate the use of a chemical additive to minimize organic growth and to develop a procedure to clean the tank.

The storage tank was cleaned between March 27 and April 1, 1987 in accordance with preventive maintenance procedure O-PMM-022.4, Diesel Oil Storage Tank Cleaning, revision dated March 26, 1987. However, the day tanks were not cleaned. The licensee is evaluating the need to clean the day tanks and skid tanks. These tanks also have not been cleaned during the past 15 years.

8. Maintenance Observations (62703/62700)

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

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

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

The following maintenance activities were observed and/or reviewed:

Unit 3 and 4 Spent Fuel Pool Bridge Crane Rail Inspection
Relay Work Order (RWO) 87-0019, Repair of the B Diesel Loss of Field Relay
RWO 87-0021, Inspection and test of the A Diesel Relays
RWO 87-0020, A EDG Protection Relay Voltage Checks
Plant Work Orders (PWO) 5764, Test and Repair Control Room Ventilation Transfer Switch
Diesel Generator Fuel Oil Storage Tank Cleaning

a. Emergency Diesel Generator Wiring Discrepancy

On March 27, while both units were in refueling shutdown (mode 6), personnel from the Relay Department were performing periodic testing to verify the operability and correct calibration of several of the B Emergency Diesel Generator (EDG) protective relays. The loss of field excitation relay-140 was determined to be inoperable in that its activation did not cause the actuation of the generator lockout relay-86. Troubleshooting was performed under Relay Work Order (RWO) 87-0019. It was determined that connection wiring drawing 5610-M-16-73/83-155 sheet 1 of 4, Revision 2, did not reflect the exact wiring of B EDG control panel 4C12. Missing from the drawing was a wire that should have connected relay-127/159 stud 11 to relay-151-A stud 1. The absence of the wire created an open circuit and prevented operation of the loss of field excitation relay input to the B EDG lockout relay-86. Additionally, undervoltage relay-127 and overvoltage relay-159 were not operable. These two relays provided inputs to a control room annunciator designed to alert the operator to abnormal voltage conditions.

The loss of field excitation relay became disconnected in EDG panel 4C12 because the actual routing of wires from point to point inside the panel did not match connection diagram 5610-M-16-73/83-155 sheet 1 of 4. This drawing was used to develop Plant Change Modification (PCM) 83-155 which was implemented on May 7, 1986. The purpose of the PCM was to install isolation switches necessary to meet 10 CFR 50, Appendix R (fire protection alternate shutdown) requirements.

Consequently, PCM 83-155 was developed incorrectly and the discrepancy could not be identified from drawing 5610-M-16-73/83-155 sheet 1 of 4. The error in the PCM caused the loss of field relay, the undervoltage relay and the overvoltage relay to be disabled. Since the implementation of PCM 83-155 should not have affected these relays, a functional check of the relays was not included in post modification testing and went unnoticed until the periodic relay test was next due.

The licensee identified the discrepancy during routine surveillance testing on March 27, 1987 and the wiring error was promptly corrected. Corrective action was completed as documented in Non-Conformance Report (NCR) 87-0095. Change Request Notice (CRN) E-4380 was issued to justify and approve the wiring addition and to initiate a revision to the drawing.

Testing of the A EDG lockout relay was performed on March 27 to determine whether a similar relay problem existed. The tests were performed under RWO 87-0020 and 87-0021. No discrepancies were identified. However, the B EDG remained in overhaul (out of service) while A EDG lockout relay-86 was activated. During eight lockout relay actuations, each of approximately 3 seconds duration, the A EDG was not capable of automatic starting and the B EDG was also inoperable. This condition is discussed in paragraph 9B.

The failure of to have an accurate drawing of safety-related wiring in EDG panel 4C12 is a violation of 10 CFR 50, Appendix B, Criterion VI (250,251/87-14-04).

b. 3B Sequencer Wiring Discrepancies

To determine whether the implementation of PCM 83-155 had inadvertently affected other relays and since the PCM included work on wiring of the 3B electrical load sequencer, a 3B sequencer inspection was performed by the licensee.

The inspection revealed two wiring problem areas: (1) affecting the operation of the 3B containment spray (CS) pump during a Unit 3 design basis accident (DBA); and (2) affecting the automatic start of the Unit 4B and 4C intake cooling water (ICW) pumps and the 4B component cooling water (CCW) pump during a Unit 3 DBA.

The DBA for either Units 3 or 4 assumes a loss of coolant accident (LOCA) on one unit in conjunction with a dual unit loss of offsite power (LOOP). The redundant design of the Engineered Safety Features (ESF) precludes loss of system function for the CS, CCW, and ICW systems due to any single failure such as the failure of a pump or the loss of either EDG.

(1) Given a large break LOCA in conjunction with a LOOP, normal safeguards operation results in the 3A and 3B sequencers starting their respective CS pumps between 17 and 23 seconds after the start of the A and B EDGs. If the LOCA results from a smaller break, then elevated containment pressure may not exist when the sequencer reaches the 17 to 23 second period. Consequently, by design, the start of the CS pumps is enabled and will occur automatically when elevated containment pressure is detected.

During the 3B sequencer inspection the licensee identified a disconnected wire affecting the operation of the 3B CS pump. The wiring error affected only the smaller break LOCA/LOOP scenario. It altered the start logic of the 3B CS pump such that automatic start on elevated containment pressure achieved subsequent to the 23 second time period was precluded. The remote manual start capability of the pump from the control room remained operable. No PCM has been identified affecting the circuit of concern. The licensee continues to review maintenance records to determine when the wire was inadvertently disconnected. No maintenance, which would by design alter the CS wiring in the 3B sequencer, has been recently performed.

This discrepancy resulted in a loss of redundancy in the CS system. If, during the accident of concern, the 3A CS pump failed or the A EDG failed, then all system function would be lost until a Control Room Operator diagnosed the failure and manually started the 3B pump as required by the Emergency Operating Procedures.

Previous ESF actuation testing did not identify the wiring discrepancy. The ESF testing performed by the licensee simulated a large break LOCA in conjunction with an immediate LOOP. Test signals provided an instantaneous, simulated elevated containment pressure such that the 3B CS pump automatically started during the 17 to 23 second time period. Testing has not been performed to simulate the smaller break LOCA scenario. Thus, the automatic start circuitry subsequent to the sequencer completing its timed start sequence has not been tested.

(2) An additional wiring discrepancy, consisting of two jumpers which should not have remained in the circuitry, resulted in the 4B CCW pump, the 4B ICW pump and the 4C ICW pump being incapable of starting during a Unit 3 DBA. Both automatic and manual remote (control room) start capability were disabled.

The ICW and CCW wiring error occurred during the implementation of PCM 79-145 which was completed on May 13, 1984. This PCM modified the automatic power transfer circuit for motor control center D. The root cause of the error was that Process Sheet

84-019, which gave detailed instructions to the craftsman making wiring changes, did not instruct the craftsman to remove the two wires. Electrical Wiring Diagrams, from which the Process Sheets are developed, clearly indicated that the wires needed to be removed to provide proper circuit operation.

This discrepancy resulted in a loss of redundancy in the ICW and CCW systems. If, during a Unit 3 DBA (LOCA and dual unit LOOP), the 4A CCW pump and the 4C CCW pump, or the 4A ICW pump, or the A EDG failed, then all Unit 4 ICW or CCW system function would be lost.

Post-modification testing and periodic surveillance testing, although implemented as required, did not reveal the ICW and CCW wiring deficiencies because, although testing duplicated a Unit 3 DBA, it did not simulate a simultaneous LOOP on Unit 4. During testing, offsite power remained available and, as per design, the Unit 4 ICW and CCW pumps were not stripped and did not load on the EDGs. Consequently, no opportunity existed to test the 3B sequencer's ability to trip and reload the Unit 4 pumps. The licensee is developing, for implementation in May 1987, a dual unit LOOP test which will test the operability of this circuitry.

The failure to maintain adequate control of design changes affecting the ICW, CCW and CS systems is a violation of 10 CFR 50, Appendix B, Criterion III (250,251/87-14-01).

c. Additional Wiring Verifications Performed

As a result of identifying the B EDG wiring discrepancy, the 3B sequencer was completely inspected. Subsequent to the identification of wiring discrepancies in the 3B sequencer the licensee has performed (or plans to perform) additional inspections as follows:

(1) A 100 percent check of the internal wiring in the 3A, 3B and 4A sequencer panels has been completed;

(2) A 100 percent check of the internal wiring in the 4160 volt bus 3B breaker compartments has been completed;

(3) A 100 percent check of the internal wiring in EDG panel 4C12 has been completed; and

(4) A 100 percent check of the internal wiring in sequencer panel 4B will be completed prior to the restart of either unit.

Checks of those items completed above did not reveal any functional errors other than those mentioned in this report. The specific discrepancies described in this report have been corrected. The

licensee has committed to develop and implement a more comprehensive ESF test which will include all functional circuitry in the sequencers. The test will be performed by simulating a dual unit LOOP. The DBA will be simulated for each unit. Testing modifications necessary to verify the proper operation of ESF circuitry outside the large break LOCA timeline are under review.

d. SFP Bridge Crane Rail Corrosion

The SFP bridge crane was declared out of service in February due to the identification, in November 1986, of corrosion damage to the crane rails, rail hold-down clips, and anchor bolts. Contractor personnel performing a crane evaluation at the licensee's request provided information relative to the degraded nature of the crane in late November 1986. A confirming inspection was performed by the plant staff on December 30, 1986. On January 14, 1987 a safety evaluation was written justifying the delay of compensatory measures for 30 days due to the low probability of a seismic event occurring in south Florida. A one week extension of this delay was approved by the Engineering Department in February 1987 to accommodate a planned SFP fuel inventory as discussed in paragraph 11.

The evaluation of the degraded condition revealed that the corrosion was caused by the long term effects of borated water on the rails, clips and anchor bolts. Heavily borated water is maintained in the SFP to cool expended fuel and maintain a large subcritical margin. The borated solution in the water is corrosive to ferritic metals. However, at the relatively low temperatures in the SFP, the corrosive effects are very slow. The degraded condition and potential loss of seismicity resulted from long-term effects which were avoidable. Water from the SFP was allowed to stand in the rail troughs for extended periods of time and small amounts of water dripped into the trough during fuel movement evolutions. Insufficient drainage and poor housekeeping practices allowed standing water to contact the ferritic metal for prolonged periods. Additional discrepancies identified by the licensee included improper rail hold-down bolt thread engagement and hold-down clip rail overlap.

The repair schedule has been delayed until September 1987, during which time the crane will be unavailable for use. The delay results from the licensee's desire to coordinate rail repairs with a planned installation of a new crane. The existing crane was immobilized in late February by using wire tie-downs and "C" clamps. This action is to prevent it from falling into the SFP during an unlikely seismic event. The licensee does not anticipate the use of the Unit 4 SFP bridge crane prior to September 1987. However, the crane will have to be repaired before any fuel can be transferred from the Unit 4 reactor to the SFP. The next planned fuel movement will occur during the Unit 4 refueling outage scheduled for March 1988. The licensee documented this condition in a letter (L-87-96) issued on March 2, 1987.

e. Raychem Splice Safety Evaluation

On April 12, 1987, the licensee issued safety evaluation JPE-LR-87-016, Revision 0, to justify the operation of Unit 4 until the next refueling outage, tentatively scheduled for March 1988. The evaluation was written in response to IE Notice 86-53, dated June 26, 1986, which indicated that Raychem splice installations at several plants did not comply with the manufacturer's installation requirements. The licensee has identified Raychem splice problems similar to those described in the Notice.

JPE-LR-87-016 is being reviewed by the NRC. A meeting with the licensee has been scheduled for May 4, 1987 to discuss, in part, pertinent issues relative to the environmental qualification of the splices and the status of the upgrade program described in JPE-LR-87-016.

9. Operational Safety Verification (71707)

The inspectors observed control room operations, reviewed applicable logs, conducted discussions with control room operators, observed shift turnovers and confirmed operability of instrumentation. The inspectors verified the operability of selected emergency systems, verified that maintenance work orders had been submitted as required and that followup and prioritization of work was accomplished. The inspectors reviewed tagout records, verified compliance with TS LCOs and verified the return to service of affected components.

By observation and direct interviews, verification was made that the physical security plan was being implemented.

Plant housekeeping/cleanliness conditions and implementation of radiological controls were observed.

Tours of the intake structure and diesel, auxiliary, control and turbine buildings were conducted to observe plant equipment conditions including potential fire hazards, fluid leaks and excessive vibrations.

The inspectors walked down accessible portions of the following safety related systems to verify operability and proper valve/switch alignment:

A and B Emergency Diesel Generators
Auxiliary Feedwater
Control Room Vertical Panels and Safeguards Racks
Spent Fuel Pool (Units 3 and 4)
Intake Cooling Water Structure

a. Unit 4 Core Alterations

On April 9, 1987, while Unit 4 was in Mode 6, core alterations commenced prior to satisfying all requirements to ensure containment integrity. Specifically, the reactor vessel upper internals were partially lifted while purge valves in the containment ventilation system were jumpered open. The following narrative summarizes the activities which led to this event.

The Unit 4 reactor vessel head was lifted on April 4, 1987, to facilitate inspection/evaluation of the reactor coolant system material degradation due to the buildup of boric acid as discussed later in this report. As a result of the vessel head detensioning, the International Atomic Energy Agency (IAEA) requested that a fuel inventory be performed in accordance with the IAEA safeguards agreement. This inventory required lifting of the reactor vessel head and removal of the upper internals.

Temporary Procedure (TP)-327, Unit 4 - Precautions and Limitations For Head Lift and Upper Internals Removal, provides the prerequisites and required related system status for lifting the head and removing the upper internals. TP-327, in step 5.7, requires verification of containment integrity. Step 5.10 requires the containment vent and purge system, including the radiation monitors which initiate isolation to be tested and verified to be operable. Maintenance Procedure (MP) 1407.9, Reactor Vessel - Removal of Upper Internals, provides the instructions for removal of the upper internals. MP 1407.9, in step 9.12, requires that the Plant Supervisor - Nuclear (PS-N) be notified that the upper internals lifting rig is ready to lifted over the upper internals.

On April 8, 1987, the PS-N was notified by maintenance personnel that the upper internals lifting rig was ready to be lifted over the upper internals. Subsequently, the polar crane malfunctioned resulting in approximately a one day delay in lifting the upper internals. It was erroneously assumed at this point, by operations, that maintenance personnel would renotify the PS-N of the intention to raise the lifting rig over the upper internals. On April 9, 1987, Area Radiation Monitor System (ARMS) channel 5, which monitors the U-4 containment operating deck, alarmed in the control room. Upon investigation, operations personnel discovered that the upper internals were being lifted. The lift was immediately stopped by the PS-N. Electrical jumpers on the containment purge valves were removed and the valves were verified to have closed. The result was that containment integrity had not been established prior to lifting the reactor vessel upper internals. The following Technical Specifications apply during core alterations: 3.10.1 requires, in part, that each containment building penetration providing direct

access to the outside atmosphere shall be either closed or capable of being closed by an operable automatic containment ventilation isolation valve; 3.10.2 requires the containment ventilation system to be operable, but if the system is inoperable each penetration providing direct access to outside atmosphere must be closed; and 3.10.6 requires direct communications to be maintained between the control room and the refueling station; 6.2.2.e requires that all core alterations be directly supervised by a licensed SRO or an SRO Limited to Fuel Handling who has no other concurrent responsibilities.

The primary basis for the TS in Section 3.10 is to restrict the release of radioactivity from the containment atmosphere to the environment. This event is significant in that, had there been a radiological release to the containment while performing core alterations it would also have been released to the environment. In fact, had PRMS channel 5 not yet been verified operable per TP-327 step 5.13 the internals lift may not have been known by operations personnel until after completion of the lift.

The apparent cause of the event can be attributed to the performance of MP 1407.9 and TP-327 in parallel. Technically. the prerequisites for lifting the upper internals per TP-327 may be performed at any time during the performance of MP 1407.9 up to the point of the actual upper internals lift in step 9.29. Although MP 1407.9 provides the instructions for removing the vessel upper internals, it contains no requirement to ensure all prerequisites appropriate to perform core alterations are complete. The inadequacy of MP 1407.9 to ensure proper completion of TS required prerequisites is a violation (251/87-14-02) which applies to Unit 4 only. Corrective action necessary to prevent recurrence was implemented immediately following the event. It consisted of an additional step in MP 1407.9 for Q.C. to verify the appropriate prerequisites, including setting containment integrity, prior to removing the upper internals.

b. A EDG Lockout Relay Testing

Testing of the A EDG lockout relay was performed on March 27 to complete periodic requirements and to determine whether relay problems similar to those discussed in paragraph 8 existed. The testing was performed as a part of RWO 87-0020 and RWO 87-0021. No discrepancies were identified relative to relay operability. However, the B EDG remained in overhaul while the A EDG lockout was activated for short periods of time as required by RWO 87-0021. During 8 actuations, each of approximately 3 seconds duration, the A EDG was not capable of automatic starting and the B EDG was also inoperable. A loss of offsite power would have resulted in a blackout. This condition was mitigated by the short time period involved, the shutdown condition of both units, and the speed with

which the A EDG could have been returned to service. Additionally, having determined that the B EDG had defective wiring, it was necessary to establish the operability of similar relays in the A EDG. Also, the Turkey Point Technical Specifications do not address the operability of EDGs in cold or refueling shutdown, so the actions taken by the licensee did not violate diesel operability requirements.

However, the licensee performed the testing under a work order which did not address precautions and limitations relevant to a loss of offsite power. Although the work order received prior Quality Control review, it did not require that the time period for which the EDG was locked out be minimized. Nor did it require that the lockout be reset at the completion of testing. Additionally, it did not address procedures or mechanisms for restoring immediate operability in the event that emergency power was required while both diesels were inoperable.

Concerns relative to this issue were discussed with licensee management at the exit meeting on April 27, 1987. The licensee stated that, in the above circumstance, written procedural precautions were not considered essential due to the knowledge level of the personnel performing the testing and the unlikely prospect of a loss of offsite power.

c. Unit 4 Conoseal Leak

On March 11, 1987, Unit 4 reactor was manually shutdown as required by Technical Specification due to a leak in the containment personnel hatch inner door. Unit 4 was maintained in Mode 3, Hot Standby, until March 13, while repairs to the personnel hatch door were in progress. On March 13, 1987, site engineering was informed by Westinghouse (W) that potential corrosion rates from a known leak at the port instrumentation column assembly (conoseal) on the Unit 4 reactor vessel head penetration as evaluated in Safety Evaluation JPE-M-86-077 were inaccurate and may actually be double those previously evaluated. Unit 4 was immediately taken to Mode 5, Cold Shutdown, to assess the leak and the extent of boric acid contamination and subsequent surrounding corrosion areas.

On March 16, 1987, NRC personnel entered the Unit 4 containment to observe the conoseal, the reactor vessel, and other areas affected by the conoseal leak. Subsequently, an NRC Augmented Inspection Team (AIT) was formed and dispatched to the site. Initial observations and followup concerns are contained in the AIT report (251/87-16).

On April 27, 1987 Florida Power and Light issued a report to the NRC via letter L-87-186. The subject of the report was instrumentation port column assembly leakage at the Unit 4 reactor vessel head. This

report provides background information related to the conoseal leakage, identifies the components affected by the leakage, identifies the potential leakage mechanisms and describes subsequent corrective actions.

d. Unit 3 Fuel Shuffle

On April 16, 1987, Operations personnel began the process of relocating the two Unit 3 regenerative secondary sources to new fuel assemblies. The work was being performed in the Unit 3 spent fuel pit. Operating Procedure 16900.16 (October 25, 1985) Rod Cluster Control Tool, provides the operators with the instruction to properly accomplish the relocation. The first source was relocated without incident. However the second source was not fully withdrawn from its existing fuel assembly and was damaged when the operators attempted to move it horizontally to the new fuel assembly location. Step 8.2.2 of OP 16900.16 ,states, in part, that when raising the source to:

> Release the joystick lock and move it to position "up with the rod". This will raise the gripper with rod attached until the up travel limit is reached. The "up" light will go on and the motor will stop. Now the RCC assembly is fully withdrawn in the RCC change tool.

Contrary to the instruction of step 8.2.2, the operator in control of the motor operated winch relied upon the visual judgement of another reactor operator, rather than the mechanical indicators available, to determine when the source was fully withdrawn. Although the travel limit switch had not been reached, the "up" light had not gone on and the motor had not automatically stopped, the operator stopped the upward vertical lift of the source. The source was not been fully withdrawn from the original fuel assembly. Subsequently, when the source was moved horizontally it impacted the the original assembly and was bowed. The operator immediately stopped the movement and attempted to straighten the source but it had been creased and permanently damaged. The SFP was evacuated, at the direction of the PSN, and Off Normal Operating Procedure 3-ONOP-033.3, Accidents Involving New or Spent Fuel was performed to ensure that the integrity of the source had not been breached. Failure to follow the instructions of procedure OP 16900.16 constitutes a violation (250/87-14-03).

FPL has performed a 50.59 safety evaluation, which justifies Unit 3 start up with only one regenerative secondary source. The evaluation referenced Westinghouse letter titled Turkey Point Secondary Source (April 22, 1987), which stated that removal of one secondary source assembly from the Unit 3 core would not affect the design parameters used in the safety evaluation.

e. A EDG Testing

On April 9, 1987, OP 4304.3, EDG - Eight Hour Full Load Test And Load Rejection was performed on the B EDG. The inspectors reviewed the completed procedure and noticed that the first several steps of the procedure which required the test cocks to be opened and the engine to be manually barred over were not performed. Justification, as noted by the APSN in the remarks section, was that the B EDG had just completed the 12 hour break-in run following an overhaul. The inspectors were concerned that operations personnel may not possess the expertise required to allow these steps not to be performed. After several interviews with varying levels of plant management, it was learned that the vendor technical representative and technical department system engineer were present and had instructed operations personnel to omit the prerequisite steps based on the results of the just completed break-in run. The inspectors expressed to plant management, at the monthly exit interview, the need to fully and accurately justify and document all departures from verbatim compliance to approved procedures. The licensee concurred with this position.

10. 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.

On March 10, 1987, during leak rate testing of the Unit 4 personnel hatch, the overall airlock leakage rate did not meet the TS acceptance criteria for overall containment leak rate. The plant was shut down as required by the Technical Specifications. The cause of the high personnel hatch leak rate was that an equalizing valve was not fully closed due to a misadjusted mechanical linkage. LER 251-87-007 was issued on April 9, 1987, addressing the corrective actions which have been implemented.

On April 19, 1987, while both units were in refueling shutdown, a temporary loss of control room ventilation occurred due to the failure of an automatic power transfer switch to shift upon loss of the 3A motor control center. Proper operation of transfer switch was precluded by physical interference from a cover panel. With the cover plate removed the transfer switch operated properly. The cover plate is being modified to ensure that adequate clearance exists between the plate and switch contacts. LER 250-86-40 and Inspection Report 250,251/86-45 provide

additional information relative to the susceptibility of the control room ventilation system to single failure. On December 18, 1986, the licensee determined that the single failure potential constituted a substantial safety hazard and was reportable under the requirements of 10 CFR 21. A modification to the transfer switch is being developed and is scheduled for installation in accordance with the Integrated Schedule. Short term corrective actions, in the form of weekly transfer switch tests, have been implemented and have been reviewed by the inspectors as described in Inspection Report 250,251/87-10. A method of testing the transfer switch in a manner which also verifies the absence of coverplate interference is being developed.

11. International Atomic Energy Agency (IAEA) Surveillance Program

In fulfillment of the Safeguards Agreement between the United States and the IAEA, the IAEA selected, on July 19, 1985, Turkey Point Unit 4 for participation in its international safeguards inspection program. A major portion of this program requires the continuous surveillance of the fuel inventory through camera monitoring and seal wire placement. The surveillance program ensures that the fuel inventory does not change between physical audits.

The US/IAEA Safeguards Agreement has been in force since July 31, 1980. The commitments by the U.S. in this treaty, which carries the force of law, are defined in the Code of Federal Regulations, the treaty itself, and the site-specific Facility Attachments. On April 10, 1987, the Commission issued Amendment 117 to the Facility Operating Licence No. DPR-41 for the Turkey Point Plant, Unit 4. The amendment adds License Condition 3.J regarding implementation of the IAEA Safeguards program for Unit 4.

The NRC inspectors verified, during routine tours of the Unit 4 Spent Fuel Pool (SFP) and the accessible portions of the containment building, that seal wires were in place and intact and that surveillance cameras were operable. Seal wires are placed by IAEA inspectors on the containment equipment access hatch, the missile shields and the reactor vessel head seismic restraints. Only the seal wires on the equipment hatch can be observed from outside the containment building. The containment building is not normally entered during power operation. Two surveillance cameras are installed in the Unit 4 SFP. The SFP area is always accessible through locked and alarmed doors.

On March 13, 1987, the licensee determined that a conoseal leak had caused corrosive damage to several reactor vessel head closure studs. Repair work required opening of the Unit 4 equipment hatch and removal of the missile shields. On March 15 at 5:50 p.m. the licensee informed the NRC Office of Nuclear Material Safety and Safeguards (NMSS) of the need to remove the seal wires in the applicable areas. The four seal wires on the equipment hatch were cut at 7:00 p.m. on March 15. The two seal wires on the missile shields were removed at 1:00 p.m. on March 16. A required written report was issued as licensee letter L-87-135 on March 18, 1987,

entitled IAEA Reportable Occurrence 87-02. The continuous surveillance of the reactor was being maintained by three seal wires on the reactor vessel head seismic restraints.

Subsequently, on March 21, the licensee decided that a comprehensive evaluation and repair of affected equipment required the removal of the reactor head. NMSS was notified via telephone on March 21 at 7:30 p.m. An additional, confirmatory telephone call was made on March 22 at 10:30 a.m. The seals were removed at 12:00 p.m. on March 22, 1987. The NRC Senior Resident Inspector and NMSS were promptly informed of the removal. After the seals were removed, the IAEA expressed a desire to re-establish the physical inventory of the Unit 4 reactor fuel as required by the Safeguards Program. Although the conoseal repairs did not require the removal of the upper internals, it was decided that removal would be performed to facilitate the inventory. The physical inventory was satisfactorily completed on April 9, 1987. Additionally, a surveillance camera has been installed in the containment to provide photographic records verifying that no changes to the fuel inventory are being made while the head remains removed.

The camera will remain in operation until IAEA inspectors reinstall seal wires on the reactor vessel head seismic restraints and the missile shields. This is expected to occur on April 30, 1987. The seal wires will be reinstalled on the equipment hatch on May 6, 1987. Schedules are subject to change due to uncertainties inherent in the repair of the conoseal. Coordination of IAEA site visits is being arranged between the licensee and NMSS. The camera in the containment building will be removed by IAEA inspectors prior to power operation.

On March 30, 1987, two IAEA inspectors visited the site, accompanied by an NMSS staff member, to perform a book inventory of Unit 4 fuel and to replace surveillance camera film. No discrepancies were identified by the IAEA inspectors.

On February 13, 1987, a limited physical inventory of the Unit 4 SFP was performed by IAEA inspectors. A complete inventory was planned using the ION I and Fork Detector but it could not be implemented because the bridge crane was out of service. The SFP bridge crane was declared out of service in February due to the identification, in November 1986, of severe corrosion damage to the crane rails, rail hold-down clips, and anchor bolts. The repair schedule is not complete. However, the crane will not be available for use until September 1987, at the earliest. In the interim, the crane has been immobilized using wire tie-downs to prevent it from falling into the SFP during a seismic event.

On March 2, 1987, the licensee issued letter L-87-96, documenting this problem. In this letter the licensee requested that the IAEA inventory of the SFP using the ION I and Fork detectors be deferred until late 1987.