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

Report Nos.: 50-250/87-35 and 50-251/87-35

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: July 20 - August 24, 1987

Inspectors: HO Christensen for 9/11/87
D. R. Brewer, Senior Resident Inspector Date Signed

HO Christensen for 9/11/87
J. B. Macdonald, Resident Inspector Date Signed

Approved by: HO Christensen for 9/11/87
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, engineered safety features, operational safety, plant events, and plant procedures.

Results: Three violations, one unresolved item and two inspector followup items were identified.

REPORT DETAILS

1. Persons Contacted

Licensee Employees

*J. S. Odom, Vice President
*C. J. Baker, Plant Manager-Nuclear
*F. H. Southworth, Maintenance Superintendent
D. A. Chaney, Site Engineering Manager (SEM)
D. D. Grandage, Operations Superintendent
*T. A. Finn, Training Supervisor
*J. D. Webb, Operations - Maintenance Coordinator
D. H. Taylor, Operations System Enhancement Coordinator
J. W. Kappes, Performance Enhancement Coordinator
R. A. Longtemps, Mechanical Maintenance Department Supervisor
D. Tomasewski, Instrument and Control (I&C) Department Supervisor
J. C. Strong, Electrical Department Supervisor
*W. Bladow, Quality Assurance (QA) Superintendent
R. E. Lee, Quality Control Inspector
E. F. Hayes, Quality Control (QC) Supervisor
*J. A. Labarraque, Technical Department Supervisor
R. G. Mende, Operations Supervisor
*J. Arias, Regulation and Compliance Supervisor
*R. D. Hart, Regulation and Compliance Engineer
W. C. Miller, Senior Technical Advisor
V. Kaminskas, Reactor Engineering Supervisor
P. W. Hughes, Health Physics Supervisor
G. Solomon, Regulation and Compliance Engineer
J. Donis, Engineering Department Supervisor
W. Pike, Safety Engineering Group Engineer
*F. Irizarry, Administrative Supervisor
V. B. Wager, Licensing Engineer
G. Marsh, Reactor Engineer
*P. L. Pace, Licensing Supervisor, Corporate
*H. H. Jabali, Assistant Chief Engineer, Juno Project Engineering
*H. J. Dager, Vice President, Engineering
*H. T. Young, Site Project Manager

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

NRC Personnel

M. Scott, Project Engineer, DRP-1C

*Attended exit interview on August 25, 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 exit meeting was conducted on August 25, 1987. The areas requiring management attention were reviewed. The licensee acknowledged the findings without exception. No proprietary information was provided to the inspectors during the reporting period.

Three violations were identified:

Failure to meet the requirements of Technical Specification (TS) 4.1, Table 4.1-2 (Sheet 2 of 3), Item 10, accumulator boron concentration, in that, a satisfactory sample was not obtained from the 4C accumulator prior to heatup above 200F (paragraph 8) (251/87-35-01).

Two examples of failure to meet the requirements of TS 6.8.1, in that, valve 3-40-856 was not properly controlled (locked closed) as required by approved administrative procedure and a compensatory continuous firewatch, required by administrative and temporary procedures, was found asleep (paragraph 10) (250,251/87-35-02).

Failure to meet the requirements of TS 3.7.2.b, in that, the A Emergency Diesel Generator (EDG) was out of service for greater than twenty-four hours without the remaining B EDG being tested to prove operability (paragraph 10) (250,251/87-35-03).

3. Unresolved Items (URI)

Unresolved items are matters about which more information is required to determine whether they are acceptable or may involve violations of requirements or deviations from commitments. One unresolved item was identified during this inspection period.

Determine the adequacy of Administrative Site Procedure (ASP)-2, revision 4, section 6.6.9. The procedure allows process sheets and process sheet revisions to be implemented without prior Plant Nuclear Safety Committee (PNSC) approval, which may be contrary to the requirements of TS 6.5.1.6.d. (paragraph 14) (250,251/87-35-04).

4. Inspector Followup Items (IFI)

Track licensee development of a mechanism to mark superseded drawings kept in document control ready use files such that they are not issued for use to plant personnel (paragraph 14) (250,251/87-35-05).

Evaluate the circumstances surrounding the apparent installation of an incorrect check valve in the Unit 3 Auxiliary Feedwater (AFW) nitrogen system (check valve 293). Evaluate the circumstances surrounding the

removal of spacers between the Current Pneumatic (I/P) module for AFW valve FCV 3-2832 (paragraph 14) (250/87-35-06).

5. 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 250/83-41-07 and 251/83-40-07. Post modification System Restoration Without Complete As-Built Packages - 3 examples. FPL response letter dated March 19, 1984 was found to be acceptable (per NRC letter dated August 17, 1984). The thrust of the violation dealt with programmatic problems associated with turnover and testing of Plant Change/Modifications (PC/M). The inspector reviewed the following procedures for missing aspects indicated in the originating report details section:

ASP-21	Turkey Point Plant "Startup", Revision 3.
ASP-11	Turkey Point Plant "Construction Turnover".
AP-0190.15	Plant Changes and Modifications (PC/M).

The procedures contained the aspects referred to in the subject report. The specific fixes for this violation appear to be adequate. The design control program has been selectively reviewed by the NRC since issuance of this violation. Design control activities at the site which include broader programmatic implementation of the above violation's corrective action are encompassed in other NRC documents such as Confirmatory Order EA 86-20. Violation 250/83-41-07 and 251/83-40-07 is closed.

(Closed) Violation 250/83-41-01 and 251/83-40-01. Failure to Compensate Intermediate Range Nuclear Instrumentation Adequately. FPL response letter dated March 19, 1984 was found to be acceptable (per NRC letter dated August 17, 1984). The inspector reviewed procedure MP 12207.1 dated February 10, 1987, Intermediate Range Nuclear Instrumentation Compensating Voltage Adjustment, and found that the necessary changes had been made. Review of the licensee 's documentation indicated that the procedure has not had problems in this specific area since the original violation. Violation 250/83-41-01 and 251/83-40-01 is closed.

(Closed) Violation 250/84-29-01 and 251/84-30-02. 10 CFR 50.59 Evaluation Not Made. The original item in the inspection report discussed Intake and Component Cooling Water (ICW/CCW) system changes which placed the systems outside of the Final Safety Analysis Report parameters for which there was no safety evaluation performed (10 CFR 50.59). The NRC letter of May 13,

1985 found the licensee's written response to the violation acceptable. Subsequent events of a similar nature have occurred in the implementation phase of this violation that have been tracked by the NRC. This violation is administratively closed and tracked under Unresolved Item 250,251/87-27-02. Violation 250/84-29-01 and 251/84-30-02 is closed.

(Closed) Violation 250/84-29-02 and 251/84-30-03. Technical Specification Operability Not Shown. The NRC found the licensee's written response to the violation acceptable (May 13, 1985). The heat exchangers associated with ICW/CCW system still demonstrate degradation problems and the implementation problems of this violation are being tracked by the NRC as URI 250,251/87-27-02. Based on this tracking, the subject 1984 violation is administratively closed. Sub-sections (3) and (4) under the 1984 violation are acceptable under implementation. Violation 250/84-29-02 and 251/84-30-03 is closed.

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

(Closed) IFI 250/83-41-02 and 251/83-40-02. Failure to Implement Proper Maintenance and Housekeeping in Accordance with Quality Procedure 2.12, Revision 0. At the time this item was open, oil soaked lagging from the Unit 3 reactor coolant Pump (RCP) lubrication oil system had resulted in a fire. The inspector reviewed the following procedures:

AP 0103.11	Housekeeping
0-PME-061.1	Reactor Coolant Pump Oil Collection System, April 12, 1985.
0-PME-041.2	RCP Motor Oil Fill and Drain, February 24, 1987.
AP-090.73	Quality Control Inspection and Surveillance Program, March 10, 1987.

The above procedures contained adequate instructions to prevent the reoccurrence of the above mentioned fire. The PME procedures have been reviewed by the Procedure Upgrade Program. Additionally, Maintenance procedures such as MP-2147.2 (Charging Pump Disassembly, Repair and Assembly) have clean up instructions in their texts and/or places for initials in the text as a means of indicating clean up has been accomplished. IFI 250/83-41-02 and 251/83-40-02 is closed.

(Closed) IFI 250/83-41-03 and 251/83-40-03. Inadequate pre-startup valve lineup for Safety Injection and Containment Spray Systems. The cause of this item was that the licensee did not have a formal valve lineup for vent and drain valves within the systems. The inspector reviewed the following procedures:

3-OP-062	Safety Injection, June 18, 1987.
4-OP-068	Containment Spray System, January 29, 1987.

Both of the system lineup procedures contain valve alignment position, check, and verification for vents and drains. IFI 250/83-41-03 and 251/83-40-03 is closed.

(Closed) IFI 250/83-41-04 and 251/83-40-04. Inconsistency Between AP-0103.5 and OP-4103.1. Previous procedures conflicted over whether valve 837 was locked, and if Unit 4 Safety Injection (SI) pump suction and discharge valves had the correct prefix on their identification tags. Procedure 0-ADM-205 dated July 18, 1987, which has replaced AP-0103.5 currently shows valve 837 as locked closed. Procedure 34-OP-62 which replaced OP 4103.1 currently indicates the valve as locked closed. At present, the Unit 4 SI pump suction and discharge valves are correctly labeled. IFI 250/83-41-04 and 251/83-40-04 is closed.

(Closed) IFI 250/83-41-05 and 251/83-40-05. Fire Protection Controls During Welding. At the time this item was open, during a work evolution, a firewatch did not have a charged fire extinguisher. The inspector reviewed the following documents:

MP-15537.5	Fire Protection Equipment Surveillance of May 7, 1987.
0-ADM-013.4	Special Interim Fire Watch Duties and Training for Appendix "R" Modification of June 9, 1987.
0-ADM-013	Fire Watch Requirements and Duties of November 20, 1986.
AP-0190.67	Welding, Cutting, Grinding, and Open Flame Work Safety Procedure of March 3, 1987.

The review of the above procedures indicated that there are at least two checks for extinguisher conditions prior to work being performed. IFI 250/83-41-05 and 251/83-40-05 is closed.

(Closed) IFI 250/84-23-15 and 251/84-24-15. Evaluate Licensee Ability to Deal With Real Time Procedure Change Requirements. A real time support group which is a subgroup under the Procedure Upgrade Program has been formed. Additional personnel have been hired to support the real time procedure effort. IFI 250/84-23-15 and 251/84-24-15 is closed.

A review was conducted of the following items to assure that the licensee completed adequate applicability reviews, made appropriate distributions and if required, implemented adequate and timely corrective actions.

(Open) URI 250,251/86-18-13. Licensee to Provide Loss of DC Procedure. This item remains open as the last licensee action of NRC Bulletin 79-27, which is discussed later in this paragraph.

(Closed) 79-Bu-27, Loss of Non-Class IE Instrumentation and Control Power System Bus During Operation. The inspector reviewed the following FPL documentation:

Letter Number	Date	Recipient
L-80-71	March 3, 1980	NRC
L-80-173	June 6, 1980	NRC
PTP-RE-85-124	July 9, 1985	Site Licensing
JPE-PTRO-87-926	May 20, 1987	Technical Department (site)

The 1980 FPL letters were the licensee's response to the bulletin.

The inspectors reviewed the following NRC IE Inspection Reports which dealt with aspects of the bulletin:

Number	Item/Section
a. 251/84-14	Item 02
b. 250/84-29 and	Item 03
251/84-30	Item 04
c. 250,251/85-20	Section 5
d. 250,251/86-18	Section 11, Item 13
e. 250,251/87-07	Section 2
f. 250,251/87-10	Section 4.C
g. 250,251/87-33	Section 5

Reports a,b,c,f, and g dealt with AC vital bus portion of the bulletin. Reports d and e above dealt with Loss of DC Power. Report d (section 11, item 13) which is an unresolved item that has yet to be closed contains the final known action required of the licensee under the bulletin 79-27. For Administrative purposed bulletin 79-27 is closed. URI 250,251/86-18-13 of NRC Inspection report 250,251/86-18 will track the remaining action of the bulletin.

(Closed) 84-Bu-02, The licensee's response to IE Bulletin No. 84-02, "Failure of General Electric Type HFA Relays in Use in Class 1E Safety Systems," was reviewed and evaluated during this inspection period by the Plant Systems Section at Region II. Justification for closing-out this IE Bulletin is as follows (Item Number correspond to action items in the bulletin):

1a. The licensee stated in his response, dated July 20, 1984, that he would replace the relays identified in the bulletin as being a potential safety problem. Review of the appropriate work orders indicates that the problem relays were replaced with qualified relays within the stipulated time frame. The licensee confirmed that all HFA relays mounted in the safety-related 4160 volt switchgear were replaced. We note here for the record that the HFA relay located in the compartment for the condensate pump is actually the bus clearing relay mentioned in the licensee's response.

1b and c. The licensee stated in his response, dated July 20, 1984, that he did not have any normally energized HFA relays with nylon or lexon coil spools installed in safety-related applications. Therefore, the functional test and visual inspection were not

required to be performed. Also, it was not necessary to provide a basis for continued operation.

1d. The stipulated report was provided within the required time period.

2 and 3 These items were not applicable to Turkey Point.

In consideration of the above facts, open item 84-BU-02 is closed for Units 3 and 4.

(Closed) 85-BU-02, The licensee's response to IE Bulletin No. 85-02, "Undervoltage Trip Attachments of Westinghouse DB-50 Type Reactor Trip Breakers" was reviewed by Region II inspectors according to the guidelines in Temporary Instruction 2515/72. Justification for closing this IE Bulletin is as follows (Item Number correspond to action items in the bulletin):

1. The licensee stated in his response, dated December 9, 1985, that he performed the required test on the undervoltage trip device within the specified time frame.

2. The licensee stated in his response, dated December 9, 1985, that the appropriate test procedure was revised to include the conducting of a force margin test on the undervoltage trip devices. Request for Procedure Change, OTSC No. 3734, was reviewed. This document revised TOP206, "Reactor Protection System-Periodic Test (Unit 4 Only)", and confirms the licensee response on this item.

3. The licensee stated in his response, dated December 9, 1985, that the specified written instructions were issued. Review of "Training Brief #94" confirms the response.

4. The required report was submitted within the stipulated time period.

This bulletin was applicable to only Unit 4. In consideration of the above facts, open item 85-BU-02 is closed for Units 3 and 4 and, T2515/72 is closed for Unit 4.

7. Onsite Followup and In-Office Review of Written Reports Of Nonroutine Events (92700/92712)

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 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.

(Closed) LER 250/85-035 Technical Specification Containment Sump Level Indication. The inspector reviewed documentation for work performed on the sump level indicators LT-6308A and B. The documentation included the component test sheets and work orders. LER 250/85-035 is closed.

8. 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:

Nuclear Plant Operator Logsheets, 4-OSP-201.3
Auxiliary Feedwater System Flowpath Verification, 4-OSP-075.5
Safety Injection Pumps Inservice Test, 0-OSP-062.2

On June 27, 1987, with Unit 4 in Mode 5 and the Reactor Coolant System (RCS) temperature 190F, the licensee was making preparations to heatup to greater than 200F. TS require specific surveillances to be satisfactorily performed prior to attaining an RCS temperature of 200F. Specifically TS 4.1, Table 4.1-2, Item 10, requires that the boron concentration of each of the three cold leg accumulators be sampled and verified to be 1950 ppm or greater prior to RCS heat up above 200F. The 4A and 4B accumulators were satisfactorily sampled and had boron concentrations of 2215 ppm and 2240 ppm, respectively. The 4C accumulator was found to be empty, therefore a sample could not be taken and compliance to requirements of TS 4.1, Table 4.1-2, Item 10 could not be obtained. Concurrently, TS 3.15.1, Overpressure Mitigation System (OMS), requires that with RCS pressure boundary integrity established, the valves required to fill the accumulators must be closed with power removed, until the RCS temperature is greater than 380F. In order to comply with the requirements of TS 4.1, Table 4.1-2, Item 10 and TS 3.15.1 the licensee should have placed Unit 4 in a lesser condition of operation by reducing RCS temperature, depressurizing the RCS and then filling the 4C accumulator.

Contrary to the above, the licensee did not comply with TS 4.1, Table 4.1-2, item 10. The required sample was not taken prior to exceeding 200F. The licensee delayed filling and sampling the 4C accumulator until temperature was increased above 380F, when TS 3.15.1 restrictions on valve MOV-869 were no longer applicable. The NRC was not informed of this decision.

This discrepancy was identified by NRC inspectors during routine log reviews on July 23, 1987. The licensee was informed that the decision not to implement the TS required surveillance constituted a violation of TS 4.1, Table 4.1-2, Item 10. (251/87-35-01).

This violation is similar to violation 250/85-24-03, issued on July 30, 1985. On June 22, 1985 a Unit 3 plant heatup was in progress in accordance with Operating Procedure (OP) 0202.1, dated April 12, 1985, entitled Reactor Startup - Cold Conditions to Hot Shutdown Conditions. Section 3.12.13 of the procedure required that the boron concentration in each accumulator be verified to be at least 1950 ppm prior to exceeding an RCS temperature of 200F. An on-the-spot-change (OTSC) was approved to move this requirement to another section of OP 0202.1 which was not performed until after the required sampling temperature of Technical Specification 4.1, Table 4.1-2, item 10 was exceeded.

FPL responded to this violation in letters L-85-342, dated August 29, 1985 and L-86-31, dated January 24, 1986. The licensee stated in part:

> "At the time of the incident, the accumulators were drained and preparations were being made for a reactor coolant system heatup to greater than 200F. A conflict in sampling criteria vs. equipment availability criteria was misinterpreted thus allowing OTSC to be made to Operating Procedure (OP) 0202.1, "Reactor Startup - Cold Shutdown to Hot Shutdown Conditions", that moved the accumulator sampling to a later step in the procedure. Another factor contributing to this misinterpretation was the Overpressure Mitigating System (OMS) Technical Specification which does not allow opening of motor operated valve (MOV)-869 with reactor coolant system temperature below 380F. MOV-869 is opened when filling the accumulators.
>
> The OTSC was cancelled in order to re-establish the TS requirement back into the procedure at the proper sequence."

In both instances, when unable to obtain a required accumulator sample, licensee personnel chose to omit, rather than meet, the requirement. The decision not to fill the accumulators was influenced by a licensee interpretation of TS 3.15.1 which precluded the opening of accumulator fill valve MOV-869.

9. 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.

a. Steam Generator (SG) Feedwater Regulating Valve Concerns

On July 28, 1987, Unit 4 was required to reduce power to approximately 15% to facilitate repairs to the 4A SG feedwater regulating valve, FCV-4-478. The two piece coupling which secures the valve actuator to the valve stem was observed to be shifting, allowing valve stem movement independent of actuator motion. The coupling for FCV-4-478 was replaced with a coupling from a Unit 3 SG feedwater regulating valve. NCR-87-187 was generated to address this issue. The recommended disposition of the non-conformance report (NCR) was to install a bolt into a spare connection port in the coupling to maintain proper orientation. On August 5, 1987, Unit 4 was required to reduce power again to approximately 15% due to a similar problem with the coupling of the 4C SG feedwater regulating valve. Troubleshooting revealed that during the previous re-assembly of the valves, the coupling halves for the 4B and 4C SG feedwater regulating valves had been interchanged. Due to inexact machining the interchanged coupling valves didn't provide full thread engagement, allowing independent actuator and stem motion. The couplings were restored to their appropriate valves and set screws were installed and the unit was returned to 100% power.

An Event Response Team (ERT), (FPL correspondence PTN-TECH-87-534), was formed to address the feedwater regulating valve problems. The long term corrective action is to replace the existing couplings with couplings designed by Fisher Valve, during the next outage of sufficient duration.

b. Feedwater System Piping Thickness Concerns

The licensee established an initial program to perform ultra sonic wall thickness inspections on selected fittings in the feedwater system in response to the Surry pipe rupture event (IEN 86-106). Engineering correspondence JPE-PTPM-87-616, dated April 27, 1987, delineated the inspection program and acceptance criteria. The acceptance criteria was that the measured thickness must be greater

than Tm + 0.075 inches, where Tm is the minimum wall thickness required to withstand internal pressure, or an NCR must be generated. The following NCRs were generated:

*87-0103	(C-0535-87)	87-0195	*C-0643-87
*87-0104	(C-0500-87	87-0198	*C-0644-87
87-0108		*C-0510-87	*C-0647-87
*87-0110	(C-0555-87)	*C-0541-87	
*87-0118		*C-0542-87	

The (*) indicates NCRs which resulted in at least one fitting being replaced. More specific information and a more formalized long term program will be submitted to the NRC in the licensee response to IE Bulletin 87-01: Thinning of Pipe Walls in Nuclear Power Plants, issued July 9,1987. Licensee responses are requested within sixty days from the receipt of the bulletin.

The following PWOs and procedures were reviewed in detail for specific support to these and other maintenance activies:

PWO 6845 - Repair of 4-FCV-478
PWO 6844 - Repair of 4-FCV-488
PWO 6870 - Repair of 4-FCV-498
PWO 6146 - Calibration of the A EDG air start pressure indicator #
PWO 0263 - A EDG fuel oil filter change and strainer cleaning #
PWO 6152 - A EDG skid tank level indicator troubleshooting #
0-GMI-102.1, Troubleshooting and Repair Guidelines, dated March 17, 1987
see violation 250,251/87-35-03

10. 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
Intake Cooling Water Structure
4160 Volt Buses and 480 Volt Load and Motor Control Centers
Fire Protection Deluge Valves

a. Fire Watch Found Asleep on Duty

AP-15500, entitled Fire Protection Program, revision dated July 9, 1987, section 9.4.1 requires in part that backup suppression be established as compensatory action during fire protection impairment of automatic fire suppression systems such as the halon system. Section 9.5.3 specifies, in part, that the posting of continuous firewatch is an acceptable compensatory action when the halon system is impaired. TP-347, entitled DC Equipment and Inverter Rooms Supplemental Cooling Monitoring and Standby Condition, revision dated June 25, 1987, section 5.1.1, requires that anytime door 108A-1 is maintained open, a continuous firewatch shall be established to close the door (108A-1) within 60 seconds of sounding of the Halon Activation Alarm.

On July 29, 1987, during a routine tour performed by a member of the Quality Assurance Department, a continuous fire watch posted in accordance with AP-15500 and TP-347 was found to be asleep. The continuous fire watch post was established in early June, 1987 because fire door 108A-1 was required to be open as specified in evaluation JPE-LR-87-020, Revision 2, entitled Turkey Point Units 3 and 4 Safety Evaluation and Justification for Continued Operation With Loss of Heating, Ventilating, and Air Conditioning (HVAC) to DC Equipment and Inverter Rooms. TP-347 was established to implement the requirement of evaluation JPE-LR-87-020.

The failure of the fire watch to remain awake and attentive to his responsibilities could have precluded the effective operation of the inverter room halon system. The failure to have a continuously alert fire watch constitutes an inadequate implementation of AP-15500 and TP-347 and is an example of violation (250,251/87-35-02).

Although this discrepancy was identified by the licensee and promptly compensated for, a violation is being issued because of past occurrences of personnel sleeping on duty. Inspection Reports 250,251/87-05 and 250,251/87-11 document 7 occasions when security personnel were found to be sleeping. The licensee identified 6 of the 7 examples. Escalated enforcement action was taken on April 21, 1987, when a Severity Level III violation was issued. The fire watch

was an employee of the same contractor which supplies security personnel but he had no security responsibilities. However, his task was important to the proper functioning of the halon fire suppression system. Discussions with licensee management revealed that this problem is viewed seriously by FPL management. Corrective actions are being developed and will be reviewed subsequent to the licensee's response to the Notice of Violation.

b. Valve Lock Found Incorrectly Attached

On August 6, 1987, during a routine plant tour, the inspectors noticed that instrument air valve 3-40-856 was not locked as required by procedure 0-ADM-205, Administrative Control of Valves, Locks, and Switches, revision dated July 18, 1987. The valve was closed, as required, and the lock was closed, However, the lock wire was not properly engaged through the valve handwheel and around the instrument air pipe. Consequently, the valve could be opened while the lock remained engaged. The licensee promptly corrected the discrepancy.

TS 6.8.1 requires, in part, that procedures be established and implemented which meet or exceed the recommendations of Appendix A of USNRC Regulatory Guide (RG) 1.33. RG 1.33, Appendix A, item 1.c specifies that procedures should be developed to control equipment through locking and tagging. The failure to maintain valve 3-40-856 locked is an example of violation (250/87-35-02).

c. Emergency Diesel Generator Operability Test Requirement

On July 30, 1987, at 4:45 p.m., the A EDG was taken out of service for routine preventive maintenance. Approximately twenty hours later at 12:00 noon on July 31, 1987, the maintenance was completed and the post maintenance operability test run was commenced. Rated speed was attained but the "A" EDG tripped due to high crankcase pressure. Root cause troubleshooting and analysis were commenced.

Operations personnel were advised by Regulatory Compliance to be prepared to run the B EDG if the A EDG was not returned to service by 4:00 p.m. At 3:37 p.m., approximately twenty three hours after being taken out of service, the repairs to the A EDG were completed and the post maintenance operability test run was recommended. The A EDG was successfully run, the procedures and accompanying data sheets were reviewed as satisfactory and the A EDG was returned to service at 7:35 p.m., twenty six hours and fifty minutes after being removed from service.

TS 3.7.2.b states in part that power operation may continue if one diesel generator is out of service provided the remaining diesel generator is tested daily and its associated engineered safety features are operable.

Contrary to the above, the licensee failed to comply with the requirements of TS 3.7.2.b, in that on July 30-31, 1987, the A EDG was out of service for greater than 24 hours (26 hrs., 50 min.) without verifying the operability of the B EDG. This is a violation of TS 3.7.2.b (250,251/87-35-03).

The licensee took prompt corrective actions to prevent recurrence of this event. FPL correspondence PTN-TECH-87-539 was generated stating in part that the more conservative interim TS 3.8.1.1, applicable to EDG operability, be implemented immediately. Operations personnel were instructed to meet the following guidance in order to achieve regulatory compliance.

(1) If an EDG is discovered to be inoperable, test the remaining EDG for operability within 2 hours of the discovery and at least once per 24 hours thereafter;

(2) If an EDG is to be taken out of service for preplanned maintenance or other modifications, test the remaining EDG for operability prior to taking the EDG out of service for the maintenance and test the remaining EDG at least once per 24 hours thereafter. This will ensure that the units are not placed in a position where both EDGs are out of service at the same time.

d. Component Cooling Water (CCW) Heat Exchanger Cleaning Concerns

During review of the Plant Supervisor-Nuclear (PSN) and Shift Technical Advisor (STA) logs of July 23, 1987, the inspectors identified an apparent discrepancy in the CCW heat exchanger cleaning requirements. The PSN log and equipment out of service (EOOS) log documented that the 4C CCW heat exchanger was declared out of service for cleaning at 8:00 p.m. on July 23, 1987. The ICW inlet temperature/date and CCW heat exchangers in service curve, in the PSN log indicated that plant conditions existed such that a valve watch operator was required to be posted at TCV-2201 prior to removing a heat exchanger from service. The STA log conflicts with the PSN log entries, in that it documents the 4C CCW heat exchanger as being declared out of service and a valve watch operator posted at TCV-2201 at 10:00 p.m. on July 23, 1987. The inspectors' concern was that had the 4C CCW heat exchanger been removed from service at 8:00 p. m. and the valve watch operator not been posted until 10:00 p.m. that for this 2 hour period the licensee would have met the requirements of safety evaluation/JCO JPE-6-85-38, Revision 3. More importantly in this scenario the Unit 4 CCW system would not have been able to accept and dissipate the MHA heat load of 12.0 x 10E6 BTU/HR.

The inspectors immediately brought this concern to licensee management, who in turn initiated an investigation. Personnel on shift July 23, 1987 were contacted and time sheets were reviewed. The licensee concluded that the STA log entries were in error. The

4C CCW heat exchanger was in fact removed from service at 8:00 p.m. on July 23, 1987, but a reactor operator trainee was properly posted as the valve watch operator at TCV-2201 at 7:30 pm.

The inspectors independently reviewed the time sheets of the trainee and interviewed him to ensure that he in fact had been properly posted and informed of his responsibilities as the valve watch. The inspectors concern that although not documented in quality records that the requirements of Safety Evaluation/JCO JPE-L-85-38, Revision 3 were satisfied.

The licensee initiated corrective actions to improve programmatic controls. FPL correspondence PTN-TECH-87-462 delineated improved measures to ensure that CCW heat exchanger design basis heat removal capability is maintained throughout the cleaning process.

11. Engineered Safety Features Walkdown (71710)

The inspectors performed an inspection designed to verify the operability of the Containment Spray (CS) system by performing a complete walkdown of all accessible equipment. The following criteria were used, as appropriate, during the walkdown:

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

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

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

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

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

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

Conditions that were noted and brought to the attention of licensee include:

Unit 3

Valve 3-890A - boric acid on a flange stud
Valve 3-891A - boric acid on the packing flange - PWO dated 5/5/87
Slight boric acid buildup on the 3B Containment Spray (CS) pump

Containment isolation valve (CIV) tags not hung on Unit 3 CS CIVs:
3-880A&B and 3-890A&B
PWOs outstanding on valves;
3-844B - since 6/23/86
3-880B - since 6/23/86
3-891B - since 5/5/87

Unit 4

Boric acid buildup on the 4A CS pump - PWO dated 7/5/87
Clearance order 4-86-8-053 hung on valve 4-496T

12. 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. The licensee plans to issue LERs on each event within 30 days following the date of occurrence.

On July 28, 1987, while Unit 4 was at 100% power, it was discovered that the 4A Steam Generator (SG) feedwater flow control valve actuator was not properly coupled to the valve stem. Automatic closure on the valve was not affected, but it was determined that the valve would not have remained properly seated after receipt of feedwater isolation. In order to facilitate repairs, a power reduction to approximately 15% power was required. An ERT was formed in response to this event. Refer to paragraph 9.

On July 29, 1987, while Unit 4 was at 100% power, a voltage spike occurred in the process radiation monitor system (PRMS) rack and resulted in PRMS 4-R-11 tripping. Containment ventilation isolation and control room ventilation isolation occurred as designed. The voltage spike occurred when an I&C technician increased the PRMS R-15 high voltage supply from 1000 volts to 1500 volts. PRMS R-11 was reset.

On July 31, 1987, while Unit 3 was in Mode 5, the reactor trip breakers opened due to Nuclear Instrumentation System (NIS) channel N-36 (spiking high) high flux trip. Trip breakers were reset and tripped manually to verify SOE printout received. N-36 high flux trip did not printout due to the channel being inhibited. N-36 was placed in level trip bypass and taken out of service under clearance 3-87-7-156. A suspected bad cable/connector was the probable cause of trip.

On July 31, 1987, while Unit 3 was in Mode 5, a class B fire occurred at 1:20 pm, east of Unit 3 Turbine Plant Cooling Water (TPCW) pumps, due to a hydrogen gas leak near a construction activity involving grinding and welding. The leak was identified on a fitting between drain valve 3-4617U and PI-3-1059E. It was located under a construction work area, covered by hot work permit #7-528, for Control Work Order (CWO) A-433. The fire team responded and using CO2 extinguishers put out the fire. A water hose was also used to cooldown the piping. The hydrogen supply from the gas house was isolated. The leak was repaired by I&C and checks for other leaks-identified and repaired. Fire was reported out at 1:30 pm.

13. Summary of International Atomic Energy Agency (IAEA) Activities

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 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.

14. Design, Design Changes, and Modifications (37700)

An inspection was conducted to ascertain the licensee's methods of assuring that design changes and modifications meet the review and approval criteria specified in the Technical Specifications and 10 CFR 50.59. Previous reviews of design changes were documented in NRC Inspection Report 250,251/87-14, paragraphs 8a through 8c. In a May 21, 1987 letter the licensee was informed that the emergency diesel generator sequencer wiring discrepancies described in paragraphs 8a and 8b of the report were under consideration for escalated enforcement action. In a July 21, 1987 letter the NRC concluded that the wiring discrepancies represented a Severity Level IV violation. The sequencer wiring errors were found because the licensee conducted an expanded wiring evaluation after identifying and correcting a protection relay discrepancy. The sequencer wiring discrepancies were licensee identified, were promptly corrected and could not reasonably be expected to have been prevented by the corrective action from a previous violation. Consequently, pursuant to the provisions of 10 CFR Part 2, Appendix C, no notice of violation was issued for the item.

Additional Plant Change Modifications (PC/Ms) were evaluated in the areas of containment spray, auxiliary feedwater and main steam isolation. Specifically, reviews were conducted of the following:

PC/M 87-194 Unit 3 Containment Spray Restricting Orifice
PC/M 87-177 Unit 4 Containment Spray Restricting Orifice
PC/M 85-175 Unit 3 AFW Nitrogen Station Additions and Relocation
PC/M 85-176 Unit 4 AFW Nitrogen Station Additions and Relocation

Also reviewed were the following implementing administrative procedures:

Administrative Procedure 0190.15, Plant Changes and Modifications
Administrative Procedure 0-ADM-503, Control and use of Temporary System Alterations
Administrative Site Procedure 11, Construction Turnover
Administrative Site Procedure 2, Preparation of Site Procedures and Process Sheets

Each of the PC/M packages contained a written safety evaluation which concluded that change could be implemented without prior NRC approval under the provisions of 10 CFR 50.59. Each modification resulted in the respective system, as described in the Final Safety Analysis Report (FSAR), being changed to provide increased reliability. None of the PC/Ms required changes to the facility Technical Specifications.

a. Containment Spray Restricting Orifice

PC/M 87-177 and PC/M 87-194 were developed to install a flow restricting orifice at the discharge of the Unit 3 and 4 containment spray pumps. In May 1987, the licensee determined that the absence of flow orifices resulted in the potential for insufficient net positive suction head (NPSH) and pump runout. LER 250/87-14 documented this concern. The PC/Ms provided analyses to justify orifice design and installation. Additionally, the low level alarm setpoint of each refueling water storage tank (RWST) was reset from 33 feet to 40 feet to assure adequate NPSH at the containment pump suction piping.

The safety evaluations for PC/M 87-177 and PC/M 87-194 were reviewed by the inspectors and found to meet the requirements of 10 CFR 50.59. In addition to the licensee's safety evaluation the PC/Ms contained a safety evaluation performed by Westinghouse (SECL-87-223, Revision 2) which concluded that, in no case would the orifice installation or RWST level alarm setpoint change exceed any system design parameter or regulatory limit.

The safety evaluations were comprehensive in scope taking into account the following areas:

Containment Spray Pump Performance
Removal of Iodine from the Containment Atmosphere
Recirculation Loop Leakage
Large Break Loss of Coolant Accident Analysis
Small Break Loss of Coolant Accident Analysis
Post Accident Longterm Cooling
Containment Integrity Effects (Loss of Coolant and Main Steam Break)
Emergency Diesel Generator Loading Changes

In May 1987, when the need for installed orifices was identified, both Unit 3 and 4 were already shutdown for unrelated maintenance items. PC/M 87-177 and PC/M 87-194 were implemented prior to each unit being returned to power. PC/M 87-177 was initially reviewed and approved by the Plant Nuclear Safety Committee (PNSC) on May 19, 1987. Supplement 1 was approved on June 6, 1987 and Supplement 2 was approved on June 17, 1987. PC/M 87-194 was reviewed and approved by the PNSC on June 17, 1987.

Drawings essential to plant operation subsequent to orifice installation were promptly issued. Drawing 5610-M-470-46/87-194 was created to show orifice design characteristics. Drawing 5610-T-E-4510, Revision 71 was issued June 5, 1987 documenting the new system configuration for Unit 4. Revision 73 to the drawing was issued on July 15, 1987, documenting the Unit 3 configuration. The drawings were updated prior to the units returning to power.

Plant procedure changes were reviewed to ensure that the PC/Ms were incorporated into operating instructions. Each PC/M package contained a list of procedures which were affected by the change in the RWST low level alarm setpoint. Numerous Emergency Operating Procedures (EOPs) were affected by the alarm setpoint change because the switch from post LOCA injection to recirculation is initiated upon receipt of the low RWST level alarm. The following procedures for Units 3 and 4 were verified to have been updated to reflect changes in system operation. The changes incorporated a new NPSH requirement to place 1 pump in pull-to-lock after the low level alarm is reached.

EOP-E-1	Loss of Reactor or Secondary Coolant
EOP-ECA-2.1	Uncontrolled Depressurization of All Steam Generators
EOP-ECA-3.1	Steam Generator Tube Rupture - Subcooled Recovery Desired
EOP-ECA-0.0	Loss of All AC Power
EOP-FR-Z.1	Response to High Containment Pressure
EOP-ECA-0.2	Loss of All AC Power Recovery with Safety Injection Required

Spot checks were performed on 18 procedures each for Units 3 and 4 to verify that the EOPs correctly reflected a RWST low level alarm at 40 feet (155,000 gallons) instead of the previously acceptable 33 feet

(115,000 gallons). The new calibration setpoints were verified to be included in instrument and Control maintenance procedure 3/4-PMI-062.1, RWST Level Instrumentation Channels L-3/4-6583 A/B Calibration. All procedures were revised prior to returning the respective units to power. All procedure revisions were approved by the PNSC prior to implementation.

Process sheet 87-170 for PC/M 87-194 was reviewed to verify that the modification was performed in accordance with approved procedures. The administration of process sheets is controlled by Administrative Site Procedure (ASP) 2. The process sheet was initially approved by the PNSC on June 23, 1987. Revision 1 to the process sheet was issued on July 6, 1987 to cut, remove and reinstall sections of downstream pipe to reduce cold spring pipe misalignment. This change necessitated the removal and subsequent reinstallation of safety related piping and valves and consequently constitutes a modification to the facility. Technical Specification 6.5.1.6.d requires that the PNSC review all proposed changes or modifications to plant systems or equipment that affect nuclear safety.

All nuclear safety related process sheets are required to be reviewed by the PNSC as specified in ASP 2, section 5.6.2. However, section 6.6.9 of ASP 2 specifies that work controlled by a process sheet (or a revision to a process sheet) may proceed when the process sheet has been approved by the Field Engineer, Quality Control, Quality Assurance and the Construction Supervisor, but before PNSC approval, when so directed by the Project Site Manager or his designee.

If prior PNSC approval for a process sheet is not sought, ASP 2 requires that the proposed work be discussed with a member of the Technical Department Staff, the Nuclear Plant Supervisor and the Project Field Engineer. A memorandum is prepared and the Technical Department staff representative, Nuclear Plant Supervisor and Area Construction Supervisor sign the form. The proposed process sheet or revision to a process sheet is routed to the Quality Control Department for review and approval by the Plant Manager Nuclear. Authorization for work to proceed is valid for seven working days from the date of the memorandum.

ASP 2, Revision 4, Section 6.6.9 appears to conflict with the requirements of TS 6.5.1.6.d in that it allows modifications to be performed to the facility which have not been reviewed as proposals by the PNSC. This concern was discussed with the Site Project Manager on August 24, 1987. The inspectors were informed that a similar finding had been made by the Quality Assurance Department relative to this issue. The Site Project Manager had responded to the finding and corrective action was being implemented. The adequacy of section 6.6.9 of ASP 2 will remain an unresolved item pending a review of the Quality Assurance finding, the Site Projects

Managers response to that finding and an evaluation of the provisions of 10 CFR Part 2, Appendix C with respect to violation issuance (URI 250,251/87-35-04).

b. AFW Nitrogen Station Additions and Relocation

PC/Ms 85-175 and 85-176 provide for the relocation of an existing backup nitrogen bottle station for AFW control valve instrument air and adds a new, redundant bottle station and associated tubing and instruments. The nitrogen bottle stations supply backup nitrogen to each train of the control valves independently, ensuring a more reliable supply of AFW to the steam generators. PC/M 85-176 was installed during the Unit 4 refueling outage which ended on September 1, 1986. PC/N 85-175 was installed in the Spring of 1987 during a Unit 3 refueling outage. Both PC/Ms were approved by the PNSC prior to implementation. The written safety evaluation contained in section 7 of the PC/M packages was reviewed and found to be acceptable with respect to the requirements of 10 CFR 50.59.

The safety evaluations were comprehensive in scope, taking into account the following areas:

Seismic Structural Requirements
Missile Protection
Emergency Lighting
Health Physics Doserates During Post Accident Operation
Design Basis Description and System Function
Accident Consequences and Probability

The modification packages contained a comprehensive list of drawings and documents affected by the change. Drawing 5610-M-339 revision 23 was issued on July 8, 1986, subsequent to the installation of PC/M 85-176 (Unit 4). Revision 29 was issued on June 19, 1987, reflecting the correct configuration of Unit 3. Drawings were updated prior to the respective units entering mode 3 operation.

In preparation for a walkdown of the Unit 3 and 4 nitrogen trains, a copy of the applicable drawing (5610-M-339) was obtained from document control personnel. The copy provided was revision 25. Several field discrepancies were identified between the as-constructed Unit 4 nitrogen system and revision 25 of 5610-M-339. Subsequently, it was determined that revision 29 was the current revision of 5610-M-339 and revision 25 had been superseded. Revision 29 did not contain the field discrepancies found in revision 25.

The inspector expressed a concern that a superseded drawing revision was issued from document control. It was determined that the licensee did not have a method to identify and mark superseded file drawings. Apparently, revision 25 was issued from the corporate engineering office as an aperture card. The processing of the card

from a blueprint can take several weeks. Blueprint drawings are immediately available in the document control office because they are processed on site. Consequently, the aperture card revision of a drawing may be superseded by a blueprint revision. No requirement exists specifying that when a new revision is received in blueprint form the aperture card that it supersedes be marked appropriately or removed from the files. This creates the possibility that a superseded revision to the drawing could be issued.

This potential was discussed with document control personnel. It was determined that, through experience, most document control personnel were aware that the aperture card revision of a drawing may be out of date. Consequently, it is common practice from the aperture card revision to be checked against the latest blueprint revision and to issue for use the more current of the two. The licensee is evaluating the advisability of implementing a marking system such that all versions of superseded drawings that are retained in ready use files are clearly marked as to their status. Resolution of this issue will be tracked as an Inspector Followup Item (250,251/87-35-05).

During a walkdown of Unit 3 AFW nitrogen train 1, a Plant Work Order (PWO-308766) deficiency tag was observed on pressure regulator PC-1706. A downstream gauge indicated that the regulator was supplying 105 psi instead of the required 79 to 81 psi. Repairs were begun on August 21, 1987. It was determined that the regulator was operating properly. However, check valve 293 was determined to be improperly seated, allowing 105 psi instrument air to pressurize the nitrogen piping adjacent to the regulator. Further evaluation, which is not yet completed, indicated that an incorrect check valve was installed during PC/M 85-175. Drawing 5610-J-558, revision 2, indicates that the correct check valve be designed to close with 0.33 psi differential. A check valve designed with 10 psi differential was observed to be installed. Followup on this issue is in progress.

Additionally, during the walkdown of the Unit 3 train 2 AFW nitrogen system a discrepancy was identified in the mounting of the current to pressure (I/P) conversion module for flow control valve FCV-3-2832. This valve provided AFW flow control to the B steam generator. Apparently, the I/P module was removed and remounted during nitrogen tubing replacement associated with PC/M 85-175. When remounted, spacers were not reused between the I/P module and the module support plate. LER 250-86-31 documents an occasion when water accumulated between the back of the I/P module and its support plate causing moisture to enter the I/P module case vent. This resulted in solenoid valve failure and subsequent valve inoperability as water entered the sensing lines. Corrective action was specified in the LER, which was sent to the NRC on September 2, 1986 as an attachment to licensee letter L-86-355. Corrective action item 3 specified that the I/P module cases for all AFW flow control valves were remounted

to create a space between the module and its support plate. The space permits any incidental water (such as rain water since the valves are located outside) to drip down the support plate and past the I/P module without accumulating behind the module. This precludes a repeat valve failure.

The failure of the licensee to maintain spacers behind the I/P module case for AFW FCV 3-2832 appears to be a deviation from commitments specified in LER 250-86-31.

The inspector expressed concern over the installation of the incorrect check valve and removal of the I/P module case spacers. These two issues will receive additional inspector followup and evaluation (IFI 250/87-35-06).