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

Report Nos.: 50-250/85-20 and 50-251/85-20

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: May 20 - June 10, 1985

Inspectors: [signature] 7/3/85
T. A. Peebles, Senior Resident Inspector — Date Signed

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D. R. Brewer, Resident Inspector — Date Signed

Approved by: [signature] 7/3/85
Stephen A. Elrod, Section Chief — Date Signed
Division of Reactor Projects

SUMMARY

Scope: This routine, unannounced inspection entailed 126 direct inspection-hours at the site, including 21 hours of backshift, in the areas of licensee action on previous inspection findings, IE Bulletin (IEB) followup, annual and monthly surveillance, maintenance observations and reviews, operational safety verification, engineered safety features (ESF) walkdown, plant events, refueling operations and independent inspection.

Results: Of the nine areas inspected, no violations were identified in seven areas and two violations were identified in two areas: failure to implement some aspects of a post maintenance, preoperational procedure, paragraph 7; and failure to establish an adequate procedure for loss of a vital electrical bus, paragraph 12.

REPORT DETAILS

1. Persons Contacted

C. M. Wethy, Vice President-Turkey Point
*C. J. Baker, Plant Manager-Nuclear
J. P. Mendieta, Services Manager-Nuclear
*D. D. Grandage, Operations Superintendent-Nuclear
T. A. Finn, Operations Supervisor
*K. L. Jones, Technical Department Supervisor
B. A. Abrishami, Inservice Testing (IST) Supervisor
H. E. Hartman, Inservice Inspection (ISI) Supervisor
D. Tomaszewski, Plant Engineering Supervisor
E. A. Suarez, Technical Department Engineer
D. A. Chaney, Corporate Licensing Supervisor
*J. Arias, Regulation and Compliance Supervisor
R. L. Teuteberg, Regulation and Compliance Engineer
*R. Hart, Regulation and Compliance Engineer
J. W. Kappes, Maintenance Superintendent-Nuclear
W. R. Williams, Assistant Superintendent, Electrical Maintenance
F. H. Southworth, Electrical Department Engineer
*R. A. Longtemps, Assistant Superintendent, Mechanical Maintenance
E. F. Hayes, Assistant Superintendnet, Instrument and Control (I&C) Maintenance
V. A. Kaminskas, Reactor Engineering Supervisor
R. G. Mende, Reactor Engineer
R. E. Garrett, Plant Security Supervisor
P. W. Hughes, Health Physics Supervisor
R. M. Brown, Assistant Health Physics Supervisor
W. C. Miller, Training Supervisor
P. J. Baum, Assistant Training Supervisor
J. M. Donis, Site Engineering Supervisor
J. M. Mobray, Site Mechanical Engineer
*L. C. Huenniger, Start-up Superintendent
*H. T. Young, Project Site Manager
*H. J. Crisler, Quality Control (QC) Supervisor
R. H. Reinhardt, QC Inspector
*R. J. Acosta, Quality Assurance (QA) Superintendent
*W. Bladow, QA Supervisor
*L. E. Norris, QA Engineer
*T. P. Coste, Backfit QA Supervisor
J. E. Moaba, Performance Enhancement Program (PEP) Manager
*D. W. Hasse, Safety Engineering Group Chairman
*G. M. Vaux, Safety Engineering Group
T. C. Grozan, Licensing Group
P. Pace, Licensing Engineer
B. C. LaPira, Fire Protection Supervisor
*C. D. Tyson, System Protection Specialist

Other licensee employees contacted included construction craftsmen, engineers, technicians, control room operators (CRO), mechanics, electricians and security force members.

*Attended Exit Interview

2. Exit Interview

The inspection scope and findings were summarized during management interviews held throughout the reporting period with the Plant Manager-Nuclear and selected members of his staff.

The exit meeting was held on June 10, 1985, with the persons noted in paragraph 1. The areas requiring management attention were reviewed.

The items which were identified as potential violations were: failure to establish an adequate procedure as required by Technical Specification (TS) 6.8.1 for loss of a vital power supply (250,251/85-20-01); and failure to properly implement a procedure required by TS 6.8.1 for post maintenance testing (250,251/85-20-02).

One unresolved item was identified: blocking safety injection (SI) when temperature decreases below 543 degrees F following a reactor trip with no intention of performing a plant cooldown (250,251/85-20-03). This item was originally presented as an inspector followup item (IFI) but was changed to an unresolved item on June 11, 1985, due to possible safety significance. The licensee was informed of the change.

Another exit was held with the Plant Manager-Nuclear on June 14, 1985, to discuss an additional unresolved item; licensee personnel may not be adequately familiar with the requirements of TS (250,251/85-20-04).

The licensee acknowledged the findings. The licensee did not identify as proprietary any of the materials provided to or reviewed by the inspectors during this inspection.

3. Licensee Action on Previous Inspection Findings (92702)

   a. Monthly update of Performance Enhancement Program

      The PEP was reviewed to determine if commitments were being met. Status was discussed with the PEP Manager and with other members of management. A meeting was held in the Region II Office on May 23, 1985, to discuss the current status of the enhancement program. The agenda included a status overview of the building program, procedure

upgrade program, TS program, management improvement program, quality assurance and quality control programs, and general discussion of progress. No significant problem areas were identified. The schedule for the PEP continues to be met within acceptable limits and all modifications have been cleared by the Region.

On May 31, 1985, the site was visited by USNRC Commissioner James K. Asselstine. In addition to a discussion of the PEP status, the meeting agenda included an overview of training facility improvement, maintenance program status and philosophy, plant modifications and a plant tour. The tour included the radiological controlled area and the Unit 3 containment building, both inside and outside the biological shield wall.

b. Previous Inspection Findings

(Closed) Violation 250/84-11-03. The emergency diesel generator (EDG) surveillance procedure has been reviewed by the inspectors on numerous occasions. Compliance with the procedure has improved. Any discrepancies which occur are promptly referred to the Technical Department for evaluation and resolution. Corrective actions itemized in Inspection Report 250,251/84-11 are complete and are satisfactory.

(Closed) IFI 250,251/84-14-06. The inverter transfer switches are controlled by the shift supervisor (Plant Supervisor-Nuclear [PSN]) during normal plant operations. Use of the transfer switches to facilitate maintenance is controlled as directed by the appropriate plant work order. Emergency use of the transfer switches is controlled by applicable instrument bus off-normal operating procedures. Consequently, there have not been repeated instances of operating the wrong inverter transfer switches. This item is closed.

(Open) IFI 250,251/84-18-06. The facility operating license is subject to 10 CFR 50.59. 10 CFR 50.59 allows the holder of a license to make changes to the facility as described in the safety analysis report without prior commission approval unless it involves a change to the TSs or is an unreviewed safety question. Records of determination must be kept and a report sent to the NRC annually. The items listed in Inspection Report 250,251/84-18 should be evaluated to determine which require 10 CFR 50.59 reviews.

(Closed) IFI 250/84-34-06 and 251/84-35-06. Additional problems in the area of hoisting and rigging resulted in the issuance of Violation 250,251/85-13-02. The adequacy of the hoisting and rigging program will be evaluated as part of the corrective action for the violation.

4. Unresolved Items

Unresolved items are matters about which more information is required to determine whether they are acceptable or may involve violations or deviations.

Two unresolved items were identified and are discussed in paragraph 12.

5. IE Bulletin Followup (92703)

(Open) IEB 79-27

The inspector reviewed the requirements of IEB 79-27, Loss of Non-Class IE Instrumentation and Control Power System Bus During Operation, which was issued on November 30, 1979. Previous reviews are documented in Inspection Reports 250,251/84-11 and 250,251/84-14. Based on previous reviews an Unresolved Item (250/84-14-02) was created to determine the safety significance of the licensee's lack of procedures covering loss of vital instrument buses.

A Violation (250/84-29-03 and 251/84-30-04) was issued due to the unavailability of a procedure, Loss of 120 Volt AC Vital Bus, in September 1984. On September 20, 1984, a Unit 4 reactor trip occurred, and the standby inverter could not be placed in service. The post trip review did not address the procedural inadequacy and the Plant Nuclear Safety Committee did not review the lack of procedural guidance as a potential safety hazard.

The original licensee response to the IEB, FPL L-80-71, failed to address the intent of the bulletin. IEB 79-27 requires that the licensee prepare emergency procedures that will be used by the CRO upon loss of power to each class 1E and non-class 1E bus supplying power to safety-related and non-safety-related instrument and control systems. The emergency procedures required to be developed included, but not exclusively, procedures required to achieve a cold shutdown condition during losses of vital power supplies.

Supplemental information was submitted on June 6, 1980, in response to a request for clarification by the USNRC Region II Office (FPL L-80-173). This letter addressed annunciator procedures, bus redundancy, restoration of power and availability of redundant indications. Obtaining cold shutdown was addressed by stating that the reactor trip procedure and normal hot-to-cold shutdown procedures were available for use.

Loss of 120 volt vital instrument panel procedures were developed in the fall of 1984, approximately five years after IEB 79-27 was issued. On June 6, 1985, a reactor trip occurred due to the loss of the normal inverter power supply to instrument panel 4P06. Attempts to reenergize panel 4P06 from the spare inverter were initially unsuccessful, resulting in a partial loss of vital instrument power for approximately 50 minutes. During this event, Off Normal Operating Procedure (ONOP) 4-ONOP-003.6 was found to be inadequate in that it did not address several significant losses of control

functions. A detailed discussion of this event is contained in paragraph 12. See also Violation 250, 251/85-20-01.

As a result of the June 6, 1985, loss of instrument power and the associated procedural inadequacy, the licensee plans to reevaluate each loss of 120 volt vital instrument panel procedure for technical adequacy. This IEB remains open pending completion of the reviews and determination of any appropriate non-vital instrument procedures which need to be developed.

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

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

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

Integrated leak rate test (see Report 250/85-19)
Installation of thermocouples and control rod drive electrical cables
Emergency diesel generator operability test
Auxiliary feedwater pump operability test
Local leak rate testing (LLRT) of residual heat removal (RHR) valves MOV-750 and MOV-751

Within the area no violations or deviations were identified.

7. Maintenance Observations (62703 & 62700)

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

The following items were considered during this review, as appropriate: LCOs were met while components or systems were removed from service; approvals were obtained prior to initiating the work; activities were accomplished using approved procedures and were inspected as applicable; procedures used were adequate to control the activity; troubleshooting activities were controlled and the repair record accurately reflected what actually took place; functional testing and/or calibrations were performed prior to returning components or systems to service; quality control records were maintained; activities were accomplished by qualified personnel; parts and materials used were properly certified; radiological controls were implemented; QC holdpoi ts were established where required and were

observed; fire prevention controls were implemented; outside contractor force activities were controlled in accordance with the approved QA program; and housekeeping was actively pursued.

The following maintenance activities were observed and/or reviewed:

Installation of new cable for the A diesel generator
Repair of the 4C inverter
Repair of the 4A inverter
Repair of the B auxiliary feedwater trip and throttle valves
Unit 4 refueling transfer cart repairs
Removal of source and intermediate range nuclear instruments

The inspectors observed preoperational testing of the A EDG following the rerouting of the output cables. The cables were rerouted to provide additional separation as required by 10 CFR 50, Appendix R, modifications.

During the performance of Preoperational Procedure 0800.55, Diesel Generator A Breaker 4AA20 Control Rerouted Cable Preoperational Test, some procedural steps were not properly implemented. Section 9.0, entitled Instructions, step 9.1, requires that permission to perform the test be granted by the PSN. Step 5.2 of the procedure requires that communications be established between the control room and the technicians testing the equipment. Contrary to these requirements, on June 4, 1985, technicians began performing Preoperational Procedure 0800.55 without obtaining permission from the PSN and without establishing communications with the control room; consequently, neither the PSN nor the control room operators were aware the preoperational procedure was in progress. At this time, the A EDG was out-of-service on a clearance; therefore, no loss of additional equipment was experienced by starting the procedure without authorization.

The failure to properly implement Preoperational Procedure 0800.55 is a Violation (250,251/85-20-02).

A review of the procedure revealed that it required a continuity check to be performed on four wires of cable A-4AA20C. The continuity check necessitated removing the four wires from their terminals and subsequently relanding them. The procedure did not require that the relanded wires be verified by a QC inspector. Members of the Relay Department were aware of the importance of the QC inspection but they do not normally document the results of their QC inspection. Most of the work done by the Relay Department occurs in the switchyard and is not safety-related. The licensee is developing guidelines for incorporating the differing QC requirements of the Relay and Maintenance Departments into procedures. The Relay Department will utilize the more stringent QC documentation requirements of the Maintenance Department while working on safety-related equipment. An on-the-spot-change was written into Preoperational Procedure 0800.55 which incorporated a QC holdpoint to verify and document the relanding of the wires.

8. 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, reviewed tagout records, verified compliance with TS limiting conditions of operation and verified the return to service of affected components.

The inspectors, by observation and direct interviews, verified that the physical security plan was being implemented in accordance with the station security plan.

The inspectors verified that maintenance work orders had been submitted as required and that followup and prioritization of work was accomplished.

The inspectors observed plant housekeeping/cleanliness conditions and verified implementation of radiation protection control.

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 and/or components on Unit 3 and Unit 4 to verify operability and proper valve/switch alignment:

Refueling Water Storage
Emergency Diesel Generators
Auxiliary Feedwater Pumps
Component Cooling Water
4160 Volt and 480 Volt Switchgear
Radiological Waste Building
Control Room Vertical Panels
Nuclear Instrumentation Drawers
High Head Safety Injection
Fire Suppression System
Cable Spreading Room

Numerous loudspeakers were noted to be inoperable during general site tours. The licensee has not made significant improvements in this area. However, the plant has submitted a request for technical assistance to Power Plant Engineering to address the problem. A program for short term improvement has not been implemented.

9. Engineered Safety Features Walkdown (71710)

The inspectors verified the operability of the Unit 3 and Unit 4 common Auxiliary Feedwater System by performing a complete walkdown of the accessible portion of the system. The following specifics were reviewed and/or observed as appropriate:

a. that the licensee's system lineup procedures matched plant drawings and the as-built configuration;

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

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

d. that 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; and

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

The following discrepancies were identified and initial licensee corrective actions have been taken: some lights in the auxiliary feedwater pump area were inoperable, and the loudspeaker closest to the pumps was inaudible.

10. Plant Events (93702)

An independent review was conducted of the following events.

a. On May 20, 1985, at 12:35 a.m., the water level was lowered in the refueling cavity, spent fuel pool and fuel transfer canal to facilitate repairs to the fuel transfer carriage. The review of this event is discussed in paragraphs 11 and 12.

b. On May 20, 1985, at 11:15 a.m., an I&C technician noticed a light out on comparator PC-3-405B directly above the pressurizer comparator he was working on per plant work order 8734 and procedure 14007.14. The light and its attached fuse were removed and the subsequent loss of power caused RHR valve MOV-751 to fail closed. This resulted in a three minute loss of RHR cooling since the suction flowpath was isolated and the 3B RHR pump shutdown. The light/fuse combination was immediately reinstalled, the valve was reopened and the pump restarted. Additional information is being obtained relating to this event. Routine followup is in progress.

c. On May 30, 1985, at 6:11 p.m., a Unit 4 reactor trip occurred due to the loss of the A spare (AS) inverter power supply to vital instrument panel 4P07. An attempt was made to transfer panel 4P07 to the 4A (normal) inverter but the 4A inverter also failed. A blown fuse was replaced in the 4A inverter and additional troubleshooting did not reveal any other problems. Power was restored to 4P07 at 6:52 p.m. The licensee plans to perform preventive maintenance inspections on all inverters in June 1985. The inverters are scheduled for replacement

beginning in June 1985. The replacement schedule is being determined and may extend into 1986.

d. On June 6, 1985, at 11:21 a.m., a Unit 4 reactor trip occurred due to the loss of the 4C inverter power supply to vital instrument panel 4P06. An attempt was made to transfer panel 4P06 to the CS (spare) inverter but the CS inverter also failed. Both inverters had blown fuses. A ground in nuclear instrument N41 caused the blown fuse in the 4C inverter. Filter circuits 303 and 304 were replaced under plant work order 7206. Additional information concerning this event is discussed in paragraph 12.

e. On June 8, 1985, at 12:25 p.m., the primary relief tank was drained to the containment sump to lower the level and cool the tank. The containment activity alarm was received and consequently a containment ventilation isolation occurred due to the R-11 containment radiation monitor. The control room ventilation system also isolated as designed. Containment integrity was established at the time of the event and no release to the environment occurred. The primary relief tank was drained to the sump instead of being pumped down with the reactor coolant drain pumps because the waste gas compressors were not fully operational. A rise in R-11 activity was expected but reaching the ventilation isolation setpoint was unexpected. The warm relief tank water resulted in more vaporization than expected and contributed to the activity level.

f. On June 10, 1985, at 1:44 p.m., a shutdown of Unit 4 was commenced because both EDGs were out-of-service. The B EDG was isolated while receiving Appendix R cable separation modifications. The A EDG was receiving power for its auxiliary equipment via a temporary electrical supply from the 4A motor control center (MCC). The 3A MCC was unexpectedly deenergized when construction workers using high pressure water allowed the water the enter the MCC cabinets and short out the bus. The 3A MCC is the normal power supply to the A EDG auxiliary equipment. A temporary electrical supply from the 4A MCC ensured that the auxiliaries were available to perform their functions but may not have met electrical cable separation requirements. As a precautionary measure, the A EDG was declared out-of-service and a load reduction, preparatory to shutdown, was begun. The ability to start and load the A EDG was not lost. At 4:25 p.m., the 3A MCC was restored to service and the normal electrical supply to the A EDG was restored. The unit was returned to full power.

No violations or deviations were identified in this area.

11. Refueling Activities

Refueling activities were begun prior to successfully completing all local leak rate testing on the RHR system. After loading 12 fuel elements in the vessel it was determined that RHR valves 750 and 751 had not passed their local leak rate tests. Additional testing could only be performed by

shutting these valves; consequently, RHR cooling would not be available as required by TS 3.4.1.f. However, when moving fuel in the vicinity of the reactor pressure vessel hot legs, TS 3.4.1.f allows RHR flow to be secured provided that the reactor coolant does not exceed 160 degrees F. Additionally, if one RHR loop is not in operation the LCO requires that all operations involving an increase in the reactor decay heat load or a reduction in boron concentration must be suspended. The licensee wrote a special temporary procedure to meet the requirements of the LCO such that the LLRT could be performed without removing the 12 fuel elements from the core. Additional valve testing was unsatisfactory and subsequent maintenance efforts required all fuel elements to be returned to the spent fuel pool (SFP).

During the performance of the LLRT on RHR valves MOV-750 and MOV-751, the use of instrument air in the containment while containment integrity was established caused a slight (approximately 1 psig) pressure increase. The fuel transfer canal isolation valve was open; consequently, water was slowly pushed from the reactor cavity to the SFP. The licensee identified the slowly rising SFP level and shut the transfer canal isolation valve. The SFP level had increased approximately two inches.

On May 19, 1985, the chain on the fuel transfer cart drive motor broke. To repair the cart the transfer canal had to be drained, which is usually performed by pumping from the SFP to the refueling water storage tank (RWST) via the lower SFP suction. The attempt to pump down the SFP was unsuccessful because discharge valve 798B was stuck open. This caused recirculation of the SFP water with little flow to the RWST. The decision was made to pump down the SFP by lowering the water level of the reactor cavity via the RHR system. Water level in the reactor cavity was reduced to 21 feet above the reactor vessel flange. At the time there were 12 fuel elements in the reactor vessel. Only one RHR coolant loop was in service because the Unit 3A RHR pump motor was being overhauled. Approximately 55 minutes after reducing the water level, on-shift personnel realized that TS 3.4.1.g required that two RHR coolant loops shall be operable when the water level is less than 23 feet above the reactor vessel flange. The TS states that, should less than two RHR coolant loops be operable, action to return two loops to operation shall be taken as soon as possible. The water level was restored within 25 minutes.

The intent of TS 3.4.1.g is to have two operable coolant loops when the water level is less than 23 feet above the flange. If, after reducing the water level, a loop becomes inoperable it should be restored as soon as possible. The advisability of reducing the water level knowing that only one loop is available, to perform other than essential time-critical maintenance, has not been established. This issue is discussed in paragraph 12 as an unresolved item.

12. Independent Inspection (92706)

During the report period the inspectors routinely attended meetings with licensee management and monitored shift turnovers between PSN, shift foremen,

Nuclear Watch Engineers and licensed operators. These meetings provided a daily status of plant operating and testing activities in progress as well as a discussion of significant problems or incidents. Based on these discussions, the inspectors reviewed potential problem areas to independently assess: their importance to safety, the proposed solutions, improvement and progress and adequacy of corrective actions. The inspector's reviews of these matters were not restricted to the defined inspection program. Independent inspection efforts were conducted in the following areas:

Loss of electrical bus procedures
Use of the safety injection system block switch
Auxiliary feedwater pump electric overspeed cycling
Inservice testing of containment integrity valves
Estimated critical conditions calculations
Licensee knowledge of Technical Specifications

On May 30, 1985, the Unit 4 reactor tripped due to a failed instrument power supply. The spare inverter also failed and consequently vital instrument bus 4P07 remained without power for approximately 40 minutes. During the resulting transient the PSN directed that the SI system be blocked. Average temperature was greater than 543 degrees F and pressure remained above 2000 psig. Under these conditions, the block switch would not have been enabled if operated but equipment failure allowed the enabling. Use of the block switch precluded automatic (if it had been needed) SI actuation due to high steam line flow in conjunction with low average temperature. Other automatic actuations were not affected. No circumstances developed which required the use of the system.

Plant procedures specify that the SI system is to be blocked only when performing a reactor plant cooldown. The Technical Specifications require that the plant be placed in cold shutdown if the high steam line flow in conjunction with low average temperature SI signal is unavailable. Cold shutdown must be achieved within 30 hours. On May 30, 1985, this SI signal was made unavailable by use of the SI block switch. A preplanned plant cooldown was not already in progress and no plan was in progress to take the plant to cold shutdown. The signal was blocked for approximately one hour at a time when the PSN anticipated he would receive a high steam line flow signal due to use of the atmospheric steam valves.

The use of the SI block switch to prevent the automatic initiation of SI while not in cold shutdown may have safety significance and is an Unresolved Item (250, 251/85-20-03).

On June 6, 1985, the Unit 4 reactor tripped due to the loss of vital instrument panel 4P06 and the 4C inverter. Attempts to switch to the CS (spare) inverter were unsuccessful. Numerous vital indication and control circuits remained without power for approximately 50 minutes.

On-shift personnel implemented Off Normal Operating Procedure, 4-ONOP-003.6, Loss of 120 Volt Vital Instrument Panel 4P06, which specifies immediate and supplementary operator actions necessary to cope with the loss of vital

instrument power. The procedure was inadequate because it did not address the following significant items:

a. The loss of all pressurizer heaters and a method of restoring the heaters to services;

b. The loss of pressurizer spray valve control and a method of securing undesired spray actuation; and

c. A method of restoring letdown flow.

A review was performed of procedure 3-ONOP-003.6 which addresses the loss of vital instrument bus 3P06 on Unit 3. It also failed to address these areas. Inadequate procedures for controlling the plant during a loss of vital instrument power remains a concern. Additional discussion is found in paragraph 5.

The failure to develop comprehensive procedures that address all significant aspects of losses of vital instrument buses is a Violation (250,251/85-20-01).

Paragraph 11 addresses an occasion when the reactor vessel cavity water level was reduced to less than 23 feet above the reactor vessel flange and two residual heat removal coolant loops were not operable. This action was contrary to the intent of TS 3.4.1.g and was done without realizing that a limiting condition for operation was being entered.

Inspection Report 250, 251/85-13 document an additional example of on-shift licensed personnel failing to be cognizant of TS requirements while altering a safety-related system to allow maintenance activities. Additional failures to comply with TS requirements are documented by Licensee Event Reports (LER) as follows:

January 11, 1985, LER 250-85-01, failure to comply with the sampling requirements of TS 3.9, Table 3.9-3, Item D, for tritium grab samples in the Unit 3 spent fuel pool;

March 12, 1985, LER 250-85-08, failure to comply with the surveillance requirements of TS 4.1, Table 4.1-1, for the System Level Particulate, Iodine and Noble Gas (SPING) effluent monitors;

April 1, 1985, LER 250-85-09, removal of mechanical shock arrestor (snubber) without performing evaluations as required by TS 3.13.3; and

April 6, and April 8, 1985, LER 250-85-11, removal of additional snubbers without performing evaluations required by TS 3.13.3.

The familiarity of licensee personnel with the requirements of TS is an Unresolved Item (250, 251/85-20-04).