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

Report No.: 50-302/85-41

Licensee: Florida Power Corporation
3201 34th Street, South
St. Petersburg, FL 33733

Docket No.: 50-302 License No.: DPR-72

Facility Name: Crystal River 3

Inspection Conducted: September 26 - October 23, 1985

Inspector: R L Carroll Jr. for 11/14/85
T. F. Stetka, Senior Resident Inspector Date Signed

Accompanying Personnel: J. E. Tedrow, Resident Inspector

Approved by: V. W. Panciera 11/18/85
V. W. Panciera, Section Chief Date Signed
Project Section 2B
Division of Reactor Projects

SUMMARY

Scope: This routine inspection involved 169 inspector-hours on site by two resident inspectors in the areas of plant operations, security, radiological controls, licensee event reports, nonconforming operations reports, TMI task action plan (NUREG 0737) followup, and licensee action on previous inspection items. Numerous facility tours were conducted and facility operations observed. Some of these tours and observations were conducted on backshifts.

Results: Three violations were identified: (Failure to post a high radiation area, paragraph 5.b.(5)(a); Failure to verify the accuracy of two independent estimated critical positions prior to reactor startup, paragraph 5.a; and Failure to adhere to the requirements specified on a RWP for protective clothing, paragraph 5.b.(5)(b)).

REPORT DETAILS

1. Persons Contacted

Licensee Employees

W. Bandhauer, Nuclear Safety Supervisor
*G. Boldt, Nuclear Plant Operations Manager
*P. Breedlove, Nuclear Records Management Supervisor
R. Brown, Nuclear Electrical/I&C Supervisor
J. Buckner, Nuclear Security and Special Projects Superintendent
*J. Bufe, Nuclear Compliance Specialist
*M. Craven, Nuclear Security Officer
*D. Green, Nuclear Licensing Specialist
E. Howard, Director, Site Nuclear Operations
*W. Johnson, Nuclear Plant Engineering Superintendent
*K. Lancaster, Manager, Site Nuclear Quality Assurance
*P. McKee, Nuclear Plant Manager
*C. McLane, Building Serviceman
*R. Pinney, Senior Nuclear Engineer
V. Roppel, Nuclear Plant Engineering and Technical Service Manager
E. Simpson, Director, Nuclear Operations Eng. and Licensing
*P. Skramstad, Nuclear Chem/Rad Protection Superintendent
*D. Smith, Nuclear Maintenance Superintendent
*E. Standard, Nuclear Mechanic
K. Vogel, Nuclear Senior Electrical/I&C Engineer
*K. Wilson, Supervisor, Site Nuclear Licensing
*R. Wittman, Nuclear Operations Superintendent
G. Moore, Chairman, Nuclear General Review Committee

Other personnel contacted included office, operations, engineering, maintenance, chem/rad and corporate personnel.

*Attended exit interview

2. Exit Interview

The inspector met with licensee representatives (denoted in paragraph 1) at the conclusion of the inspection on October 23, 1985. During this meeting, the inspector summarized the scope and findings of the inspection as they are detailed in this report with particular emphasis on the violations, unresolved item, and inspector followup items. The licensee representatives acknowledged the inspector's comments and 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 Items

(Closed) Violation (302/85-29-02): The inspector verified that a Short Term Instruction (STI 85-36) was issued to ensure that licensed personnel are familiar with the requirements of the Offsite Dose Calculation Manual (ODCM). In addition, the inspector verified that the placards were installed near the Reactor Building purge exhaust fan control switches to provide information to operators prior to securing the purge fans. Discussions with licensee representatives also indicate that controlled copies of the ODCM are now being maintained in the control room.

(Closed) Violation (302/85-29-04): The procedure SP-220 that was completed on July 5, 1985, was reviewed to assure satisfactory performance. A memorandum, dated July 8, 1985, and sent to shift supervisors to remind them of the requirements for making a voluntary entry into an action statement, was also reviewed. Based upon these reviews, this item is considered to be closed.

(Closed) Violation (302/85-29-05): The inspector verified that the licensee completed the corrective actions as follows:

- A letter written on July 13, 1985, provided guidance to maintenance personnel clarifying the intent and scope of procedure CP-113;

- An investigation to determine if any jumpers were improperly installed was conducted and completed on October 18, 1985;

- Procedure CP-113 was revised as revision 44 on August 26, 1985, to clarify the intent of section 5.4 of the procedure that addresses jumper control; and,

- Procedure MP-108B was revised to delete the 2"-6" requirement. The procedure now requires that the water level be below the reactor vessel flange.

Based upon this review, this item is considered to be closed.

(Open) Inspector Followup Item (302/84-30-04): During the last outage (Refuel V), the licensee replaced a substantial portion of the nuclear services seawater (RW) system piping with piping that has been relined with a superior coating substance in accordance with modification (MAR) 83-06-27-01. This work was observed by the inspectors as documented in NRC Inspection Report 50-302/85-26. The licensee intends to add a refueling outage visual inspection to the Preventative Maintenance (PM) program to ensure that pipe integrity is monitored. This item remains open pending the addition of this visual inspection program to the PM program.

(Closed) Inspector Followup Item (302/85-37-01): The licensee has clarified procedure OP-103 and relabeled the level column in figure 7.10 to read "Dipstick".

(Closed) Inspector Followup Item (302/85-37-04): The inspector has reviewed the licensee's post trip evaluation for the reactor trip which occurred at 6:44 p.m., on August 20, 1985. The inspector reviewed the plant's parameters and condition and has no further questions on this item.

(Closed) Unresolved Item (302/85-26-08): The licensee has revised the Station Battery Service Test, SP-523, to reflect the higher discharge rates.

(Closed) Inspector Followup Item (302/85-33-07): The licensee has incorporated the guidelines discussed in Information Notice (IEN) 85-58 in the reactor trip breaker maintenance procedure PM-118. A receipt inspection is done on all breakers received from the General Electric Service Shop in Atlanta, Georgia. This receipt inspection includes the performance of PM-118 before the breaker is accepted.

(Closed) Inspector Followup Item (302/85-19-03): The inspector has reviewed the work package to install high temperature and radiation resistant control cables for valves RCV-11 and RCV-13. These cables were installed in accordance with modification (MAR) 81-11-23. The licensee has also performed an equipment qualification modification on all safety related valves in hostile environments. The inspector considers this action sufficient and has no further questions on this item.

(Closed) Unresolved Item (302/85-37-02): Evaluation of this item by NRC Region II has determined that the licensee's compensatory measures were adequate. The licensee will install an alarm system for the door in question and in the interim will maintain a security guard post for this door.

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. A new unresolved item is identified in paragraph 8.c of this report.

5. Review of Plant Operations

The plant started this inspection period in power operation (Mode 1). While reducing power for turbine generator repairs on September 27, 1985, a control rod group dropped into the reactor core causing the plant to be shutdown to the hot standby mode (Mode 3) (see paragraph 8.a for details of this event). Following repairs to the control rod auxiliary power supply on October 1, the reactor was restarted at 5:13 a.m. and the plant reentered Mode 1 at 6:38 a.m. On October 2, the plant was shut down to Mode III for repairs to a control rod drive position indicator. By 1:50 p.m., on October 2, this repair was completed and the reactor was again taken critical. Power operation was resumed at 2:45 p.m., on October 2.

On October 9, at 9:34 a.m., the reactor was manually tripped from approximately 96% of full power due to inadvertent closure of two main steam isolation valves (see paragraph 8.b for details of this event). Following repairs to the main steam isolation portion of the Emergency Feedwater Initiation and Control (EFIC) system, the reactor was restarted at 6:47 p.m. and the plant entered Mode 1 at 7:57 p.m., on October 9, where it remained for the duration of this inspection period.

a. Shift Logs and Facility Records

The inspector reviewed records and discussed various entries with operations personnel to verify compliance with the Technical Specifications (TSs) and the licensee's administrative procedures.

The following records were reviewed:

Shift Supervisor's Log; Reactor Operator's Log; Equipment Out-Of-Service Log; Shift Relief Checklist; Auxiliary Building Operator's Log; Active Clearance Log; Short Term Instructions (STIs); Selected Chemistry/Radiation Protection Logs; and Completed Operation Procedures (OPs).

In addition to these record reviews, the inspector independently verified clearance order tagouts.

On October 10, during a review of the data completed on October 9 for procedure OP-210, Reactor Startup, the inspector noted that the two Estimated Critical Position (ECP) calculations required by section 6.2.1 resulted in final reactivity values of $-0.55\%\ \Delta k/k$ and $-0.24\%\ \Delta k/k$. This provided a difference between these values of $0.31\%\ \Delta k/k$. In addition to requiring these two separate and independent calculations, step 6.2.1 also specifies that the two ECP calculations be in agreement within $\pm 0.1\%\ \Delta k/k$ prior to proceeding with the reactor startup. No other ECP calculations were performed and a reactor startup was made based upon these results.

Failure to adhere to the requirements of procedure OP-210 is contrary to the procedure adherence requirements of TS 6.8.1.a and is considered to be a violation.

Violation (302/85-41-01): Failure to adhere to the requirements of procedure OP-210 regarding ECP calculation agreement.

b. Facility Tours and Observations

Throughout the inspection period, facility tours were conducted to observe operations and maintenance activities in progress. Some operations and maintenance activity observations were conducted during backshifts. Also, during this inspection period, licensee meetings were attended by the inspector to observe planning and management activities. The meeting attendance included the Nuclear Generator Review Committee (NGRC) meeting (the Offsite Review Committee) held on October 8, 1985.

The facility tours and observations encompassed the following areas: security perimeter fence; control room; emergency diesel generator room; auxiliary building; intermediate building; battery rooms; and electrical switchgear rooms.

During these tours, the following observations were made:

(1) Monitoring Instrumentation - The following instrumentation was observed to verify that indicated parameters were in accordance with TSs for the current operational mode:

Equipment operating status; area, atmospheric, and liquid radiation monitors; electrical system lineup; reactor operating parameters; and auxiliary equipment operating parameters.

No violations or deviations were identified.

(2) Safety Systems Walkdown - The inspector conducted a walkdown of the nuclear services and decay heat seawater, containment hydrogen monitoring (WS) system, and accessible areas of the core flood system to verify that the system's drawings and procedures correctly reflect "as-built" plant conditions.

As a result of the walkdown on the WS system, there appears to be a discrepancy between the valve labeling and the flow diagram. See paragraph 7.i of this report for details.

No violations or deviations were identified.

(3) Shift Staffing - The inspector verified that operating shift staffing was in accordance with TS requirements and that control room operations were being conducted in an orderly and professional manner. In addition, the inspector observed shift turnovers on various occasions to verify the continuity of plant status, operational problems, and other pertinent plant information during these turnovers.

No violations or deviations were identified.

(4) Plant Housekeeping Conditions - Storage of material and components, and cleanliness conditions of various areas throughout the facility were observed to determine whether safety and/or fire hazards existed.

No violations or deviations were identified.

(5) Radiation Areas - Radiation Control Areas (RCAs) were observed to verify proper identification and implementation. These observations included selected licensee conducted surveys, review of step-off pad conditions, disposal of contaminated clothing, and area postings. Area postings were independently verified for

accuracy through the use of the inspector's own radiation monitoring instrument. The inspector also reviewed selected radiation work permits and observed personnel use of protective clothing, respirators, and personnel monitoring devices to assure that the licensee's radiation monitoring policies were being followed.

As a result of these reviews, the following items were identified:

(a) On October 16, 1985, while checking area postings, the inspector identified two solid radwaste storage drums (55 gallon drums) within the radwaste storage area that had indicated dose rates of 300 millirem per hour (mrem/hr) to 500 mrem/hr on contact, and approximately 100 to 150 mrem/hr at 18 inches from the drums. These drums were not posted as a high radiation area.

When licensee personnel were notified of this finding, they responded with their own instrumentation. Upon verifying the inspector's findings, the area was properly barricaded and posted as a high radiation area.

Technical Specification (TS) 6.12.1.a requires all radiation areas that have dose rates greater than 100 mrem/hr and less than 1000 mrem/hr to be barricaded and conspicuously posted as a high radiation area. Failure to adhere to the requirements of TS 6.12.1.a is considered to be a violation.

Violation (302/85-41-02): Failure to barricade and post a high radiation area as required by TS 6.12.1.a.

(b) On October 7, while observing troubleshooting on the "A" High Pressure Injection Pump (MUP-1A), the inspector noticed an electrician working inside the contaminated area around the pump and not wearing anti-contamination protective clothing on his head or rubber shoe covers on his feet. The inspector contacted the health physics office to determine the correct protective clothing to be worn for this work and reviewed the radiation work permit established for this type of work (S85-0399). As a result of this review, it was determined that a hood and rubber shoe covers were required to be worn inside the contaminated area. When health physics technicians were made aware of this finding by the inspector, the technicians instructed the electrician to wear the required protective clothing.

Chemistry and Radiation Protection Procedure RSP-101, Basic Radiological Safety Information and Instructions for "Radiation Workers", step 3.1.3.4 requires that the requirements established on RWPs be observed and adhered to.

Failure to adhere to the requirements of procedure RSP-101 is contrary to the requirements of TS 6.8.1.a and is considered to be a violation

Violation (302/85-41-03): Failure to adhere to the requirements of procedure RSP-101 to wear protective clothing established on a RWP.

(6) Security Control - Security controls were observed to verify that security barriers are intact, guard forces are on duty, and access to the protected area (PA) is controlled in accordance with the facility security plan. Personnel within the PA were observed to ensure proper display of badges and that personnel requiring escort were properly escorted. Personnel within vital areas were observed to ensure proper authorization for the area.

No violations or deviations were identified.

(7) Fire Protection - Fire protection activities, staffing and equipment were observed to verify that fire brigade staffing was appropriate and that fire alarms, extinguishing equipment, actuating controls, fire fighting equipment, emergency equipment, and fire barriers were operable.

No violations or deviations were identified.

(8) Surveillance - Surveillance tests were observed to verify that approved procedures were being used; qualified personnel were conducting the tests; tests were adequate to verify equipment operability; calibrated equipment, as required, were utilized; and TS requirements were followed.

The following tests were observed and/or data reviewed:

- SP-130, Engineered Safeguards Monthly Functional Tests;
- SP-146, EFIC Monthly Functional Test;
- SP-163A, Waste Gas H_2/O_2 Analyzer Channel Calibration;
- SP-317, RC System Water Inventory Balance;
- SP-333, Control Rod Exercises;
- SP-335, Radiation Monitoring Instrumentation Functional Test;
- SP-421, Reactivity Balance Calculations; and,
- SP-422, RC System Heatup and Cooldown Surveillance.

No violations or deviations were identified.

(9) Maintenance Activities - The inspector observed maintenance activities to verify that correct equipment clearances were in effect; work requests and fire prevention work permits, as required, were issued and being followed; quality control personnel were available for inspection activities as required; and TS requirements were being followed.

Maintenance was observed and work packages were reviewed for the following maintenance activities:

- Troubleshooting the control rod drive system;

- Control rod position indication relay replacement for control rod number 4 in rod group number 3;

- Electrical checks of the control rod power train in accordance with procedure PM-126;

- Troubleshooting of oil leaks on the "A" High Pressure Indication Pump (MUP-1A) in accordance with procedure MP-531;

- Reactor trip breaker inspection and testing in accordance with procedure PM-118;

- Troubleshooting the presence of a half trip signal on the Emergency Feedwater Initiation and Control (EFIC) system's main steam line isolation channel in accordance with procedure MP-531;

- Replacement of the "B" Decay Heat Closed Cycle Cooling Heat Exchanger (DCHE-1B) channel heads in accordance with procedures MP-122, PM-112 and modification MAR 83-05-10-01;

- Troubleshooting of reactor coolant system flow indication oscillations in the "B" channel of the reactor protection system in accordance with procedure MP-531;

- Replacement of the sightglass on the "A" Emergency Diesel Generator (EDG-3A) governor; and,

- Troubleshooting the EDG-3A drop target annunciator panel.

While observing reactor trip breaker maintenance and testing which is performed in accordance with procedure PM-118, the inspector noticed that step 7.4.3.2 of the procedure, which recorded the "as left" pickup and dropout voltage settings applied to the undervoltage trip device, did not require verification by Quality Control (QC) personnel although a QC inspector did verify the "as found" pickup and dropout voltages. The inspector discussed this omission with licensee personnel who then agreed that the "as left" voltages should be verified by a QC inspector. Step 7.4.3.2 of the procedure was performed and verified by a QC inspector and the licensee plans to revise procedure PM-118 to require this verification. This revision to PM-118 will be reviewed during future inspections. This item will be tracked along with previously identified concerns on procedure PM-118 identified in inspector followup item (302/85-07-04).

As a result of the work package for troubleshooting the EDG-3A drop target annunciator panel, the inspector noticed that this work was not classified as safety related even though electrical circuit drawings appear to classify this system as safety related. See paragraph 8.c of this report for details.

(10) Radioactive Waste Controls - Solid waste compacting and selected liquid and gaseous waste releases were observed to verify that approved procedures were utilized, that appropriate release approvals were obtained, and that required surveys were taken.

No violations or deviations were identified.

(11) Pipe Hangers and Seismic Restraints - Several pipe hangers and seismic restraints (snubbers) on safety-related systems were observed to ensure that fluid levels were adequate and no leakage was evident, that restraint settings were appropriate, and that anchoring points were not binding.

No violations or deviations were identified.

6. Review of Licensee Event Reports and Nonconforming Operations Reports

a. Licensee Event Reports (LERs) were reviewed for potential generic impact, to detect trends, and to determine whether corrected actions appeared appropriate. Events, which were reported immediately, were reviewed as they occurred to determine if the TSs were satisfied.

LERs 85-08, 85-14, 85-16, and 85-17 were reviewed in accordance with current NRC enforcement policy. LERs 85-08, 85-16, and 85-17 are closed. LER 85-14 will remain open for the following reason:

LER 85-14 reported two successive actuations of the Emergency Feedwater Initiation and Control System (EFIC). As part of the corrective action to prevent recurrence of this type of event, the licensee plans to provide a detailed review and discussion of the entire event to all operators via the Operator's Study Book. This item will remain open pending completion of this corrective action.

b. The inspector reviewed non-comforming operations reports (NCORs) to verify the following: compliance with the TS, corrective actions as identified in the reports or during subsequent reviews have been accomplished or are being pursued for completion, generic items are identified and reported as required by 10 CFR Part 21, and items are reported as required by TS.

All NCORs were reviewed in accordance with the current NRC Enforcement Policy.

No violations or deviations were identified.

7. NUREG 0737 - TMI Task Action Plan Item Review

The following TMI Task Action Plan items were reviewed and applicable installations verified to determine the status of completion:

a. I.D.2., Plant Safety Parameter Display System (SPDS): The licensee has completed installation of the SPDS in accordance with their commitment stated in a letter to the NRC dated June 19, 1985.

To verify implementation of this item, the inspector reviewed the completed modification package (MAR 81-06-38) which included system test procedures T/P-1, T/P-2, and T/P-3 and examined system installation. Due to problems with eight computer points (Nos. 227, 228, 232, 235, 236, 237, 240, and 247), test procedure T/P-3 has not been completed. The licensee is actively pursuing resolution to these problems. The licensee's status on this item is considered to be complete. Further progress on resolution of the computer point problems will be tracked as an Inspector Followup Item.

Inspector Followup Item (302/85-41-04): Review the license's progress to resolve the SPDS computer point problems. (Item I.D.2.)

b. II.B.1., Reactor Coolant System (RCS) Vents: The inspectors verified installation of the upper "J" leg and pressurizer vents in accordance with modification (MAR) 80-04-73 as discussed in NRC inspection report 50-302/83-18. During this reporting period, the inspector verified that the system was periodically tested and that operation of the vents was covered by procedures. The system is being tested in accordance with procedure SP-171, Reactor Coolant High Point Vents Functional Test, and is covered operationally by procedure EP-290, Inadequate Core Cooling.

By letter dated July 10, 1985, from the NRC to FPC, the licensee was granted a permanent exemption for the installation of the reactor vessel head vent. Based upon this letter and NRC reviews of other aspects of this item, the licensee's commitments for item II.B.1. are considered to be complete and this item is closed.

c. II.B.2., Plant Shielding (Equipment Qualification): All shielding and valve modifications were reviewed and examined by the NRC as documented in NRC Report 50-302/83-24.

In a letter dated May 5, 1980, the subject of which included the staff's evaluation of the implementation of "Category A Lessons Learned" requirements, the NRC stated that "The licensee will complete this review (for equipment qualification) for electrical equipment as part of its response to IE Bulletin 79-01B". This position was confirmed in a June 1, 1980 letter from FPC to NRC. This letter referred to NUREG 0578, item 2.1.6.B, which was involved with the qualification of electrical equipment. The licensee responded to IEB 79-01B on October 31, 1980.

During the time period from 1980 to the present, the requirements of IE Bulletin 79-01B were superseded by a new ruling designated as 10 CFR 50.49. During the period of March 4-8, 1985, an NRC inspection was conducted (NRC Inspection Report 50-302/85-09) to review the pre-implementation of the 10 CFR 50.49 rule. This inspection identified eight open items that were to be completed by the end of Refuel V (which ended in August 1985). Further activities concerning equipment qualification will be tracked in accordance with the open items of NRC Report 50-302/85-09 and action on item II.B.2 is considered to be complete.

d. II.E.1.1., Auxiliary Feedwater System Evaluation: Modifications to the emergency feedwater (EFW) system have been completed except for the installation of a new EFW tank which FPC has committed to complete, in a letter dated June 26, 1984, by the end of their next refueling outage (Refuel VI). These modifications were reviewed and verified complete by system walkdowns as follows:

- Replacement of the turbine driven EFW pump (EFP-2) steam admission valve ASV-5 and addition of a new "in parallel" steam admission valve ASV-204 in accordance with modifications (MARs) 85-04-02-01 and 80-11-48-01 respectively. These MARs and their completed installations were verified complete as documented in NRC Inspection Report 50-302/85-29;

- Removal of EFW pump recirculation return line valve CDV-104 and removal of the internals of EFW system suction valve (CDV-103) from the condensate storage tank (CST) in accordance with MAR 83-04-31-01; and,

- Electrical disconnection of valves EFV-3,4,7, and 8 in accordance with temporary MAR T82-07-34-01. This temporary MAR was made a permanent modification by MAR 77-07-01-03A. (Note: This modification was required because valves EFV-3,4,7, and 8 were electrically supplied by non-safety-related motor control centers.) The inspector verified the electrical disconnection of valves EFV-4 and 8 at both the motor control center and the valve operators.

With regard to the installation of missile shields for EFP-2, the licensee has determined through analysis that missile shields are not required. The results of this analysis was transmitted to the NRC in a letter dated August 8, 1984.

The licensee has not completed their Probability Risk Analysis (PRA). In a letter to the NRC dated October 16, 1985, the licensee stated that while their PRA was in its final stage, they intended to wait for the issuance of the NRC generic letter concerning the reliability of the auxiliary feedwater systems before they would provide their submittal.

As a result of this review, it appears that two issues, building of the EFW tank and issuance of the PRA, remain incomplete. These items will be reviewed during subsequent inspections and this item remains open.

e. II.E.1.2., Auxiliary Feedwater System Initiation and Flow: The licensee was required to install a safety-related automatic initiation and flow indication system for the emergency feedwater (EFW) system. This installation was to be followed by Technical Specification (TS) submittals to ensure that this system is included in the license conditions.

The licensee has installed and tested the Emergency Feedwater Initiation and Control (EFIC) system in accordance with modification (MAR) 80-10-66. Installation and testing of the EFIC was reviewed and observed by the inspectors as documented in NRC Inspection Report 50-302/85-33. The licensee also submitted a TS change request and Amendment number 78 was issued to add the EFIC system to the TSs.

Based upon this review, the licensee's actions appear to be complete and this item is considered to be closed.

f. II.F.1.1., Accident Monitoring - Noble Gas Monitor: The licensee has completed installation of the modifications required to satisfy this item. These modifications were reviewed and verified complete by system walkdowns as follows:

- Addition of Main Steam Line (MSL) monitors in accordance with modification (MAR) 80-05-78. This MAR was verified complete as documented in NRC Inspection Report 50-302/81-15;

- Addition of mid and high range monitors for the reactor building purge monitor (RMA-1) and the auxiliary building ventilation monitor (RMA-2) in accordance with MAR 81-04-66-01. In addition to the reviews conducted during this inspection, operation of this system was observed during a post-implementation inspection as documented in NRC Inspection Report 50-302/84-07.

The licensee has been unable to calibrate the mid and high range monitors due to the unavailability of calibration gases and procedures. This was reported to the NRC in Special Report 85-03 dated August 21, 1985, as required by TS 3.3.3.9, action statement 29. This calibration is expected to be completed by November 20, 1985. Activity on this item will continue to be tracked by an existing Inspector Followup Item (302/85-05-03).

During the system walkdowns, the inspector noted that two valves, WSV-95 and WSV-96 appear to be labeled incorrectly and appear incorrectly labeled on the system flow diagram (FD 302-694). This finding was discussed with licensee personnel. The licensee will examine the system and drawings and make applicable changes.

This is considered to be an Inspector Followup Item. This item will be reviewed in combination with the item identified in paragraph 7(i) following.

Based upon this review, the licensee's activities appear to be complete and this item is considered to be closed.

g. II.F.1.2, Accident Monitoring - Iodine/Particulate Sampling: The licensee has completed modification (MAR) 81-04-66-03 that provided the enhanced iodine and particulate sampling capability. This MAR has been reviewed and the system walked down to verify completion. Operation of this system was also observed during a post-implementation inspection as documented in NRC Inspection Report 50-302/84-07.

Based upon this review, the licensee's actions appear to be complete and this item is considered to be closed.

h. II.F.1.3, Accident Monitoring - Containment High Range Monitor: The licensee has completed modification (MAR) 81-04-68 that increased the range of the reactor building radiation monitor. This MAR has been reviewed and a partial system walkdown conducted to verify completion.

Based on this review, the licensee's actions appear to be complete and this item is considered to be closed.

i. II.F.1.6, Accident Monitoring - Containment Hydrogen: The licensee has completed modification (MAR) 79-11-70-02 that added the capability to monitor post accident reactor building hydrogen levels. This MAR and is associated testing has been reviewed and the system walked down to verify completion.

During a walkdown of this system, the inspector noted that there appeared to be a discrepancy between the system installation and the flow diagram of the system (FD 302-693). This discrepancy appears to be limited to labeling of the following valves.

- WSV-26 through WSV-31;

- WSV-38 and 39;

- WSV-47 and 48; and,

- WSV-659 through 662.

This information was presented to licensee personnel for their review. This finding will be tracked as an inspector followup item in combination with that identified in paragraph 7(f).

Inspector Followup Item (302/85-41-05): Review the licensee's progress to resolve drawing discrepancies for flow diagrams FD 302-694 and FD 302-693.

As the result of further correspondence between FPC and the NRC, the licensee has agreed to modify their monitoring system to allow more frequent testing. This modification, MAR 85-09-09-01, has been issued and work is expected to start in about three weeks. The licensee has committed to have this MAR completed by December 31, 1985. The inspector reviewed this MAR and will continue to track the installation as an inspector followup item.

Inspector Followup Item (302/85-41-06): Review the licensee's progress to install the hydrogen monitoring testing MAR 85-09-09-01.

Based upon this review, the licensee's actions appear to be complete and this item is considered to be closed.

j. II.F.2., Instrumentation for Detection of Inadequate Core Cooling (ICC):

The licensee has completed installation of the modifications (MARs) required to satisfy this item. These modifications were reviewed and verified complete by appropriate system walkdowns as follows:

- Removal of the center control rod drive mechanism (CRDM) in accordance with MAR 83-03-04-12;

- Interconnections between the main control board and the ICC instrument panel in accordance with MAR 80-10-66-17, field change notices (FCNs) 5 and 8; and,

- Testing of the completed system in accordance with MAR 83-03-04-11, test procedures (T/Ps) 1, 2, and 3.

The licensee could not obtain satisfactory completion of T/P-3 for the void trending system due to the fact that incorrect initial calibration values had been set into certain instrumentation modules. Therefore, while the ICC system is installed, it is not fully operable.

By letter dated September 6, 1983, the NRC required the licensee to install the ICC system but also directed that the system not be made operational pending resolution by the NRC of instrument usage.

The licensee has met their commitment to install the instrumentation and therefore action on this item is considered to be complete. Operability of the instrumentation, i.e., satisfactory completion of T/P-3, will be tracked as an inspector followup item.

Inspector Followup Item (302/85-41-07): Review the licensee's progress to complete T/P-3 of MAR 83-03-04-11 for the ICC system.

k. II.K.3.5, Automatic Trip of the Reactor Coolant Pumps (RCPs): In generic letter 83-10 dated February 8, 1983, the NRC determined that automatic tripping of RCPs was not necessary and that plant specific procedures directing RCP trips was adequate. In response to that letter, FPC in a letter to the NRC dated February 27, 1985, provided setpoints for use by the operators to trip the RCPs. These setpoints were incorporated into procedure AP-380, Engineered Safeguards System Actuation, as revision 4 dated July 30, 1985.

The inspector verified the inclusion of these setpoints into AP-380 and considers licensee action on this item to be complete.

l. III.A.1.2., Upgrade Emergency Support Facilities: The licensee has completed construction and has implemented use of the Technical Support Center (TSC) and the Emergency Offsite Facility (EOF). In addition, the event recall system has been installed and is fully operable.

The inspectors have observed use of the TSC and the recall system. Inspection of these facilities remains to be completed by the NRC Region II Emergency Preparedness section. This item will remain open pending completion of this inspection.

m. III.A.2.4&5, Emergency Preparedness - Installation and Implementation of the Offsite Dose Computer: The licensee has completed installation of the new meteorological data acquisition system and has implemented its usage. The licensee has had problems with this new system and is continuing to resolve these problems. A discussion of the problems with the new system is documented in NRC Inspection Report 50-302/84-18. Further activities to resolve these problems will be tracked in accordance with Inspector Followup Item (302/84-18-06).

The licensee has decided to not use the Emergency Dose Assessment System (EDAS) they had previously committed to develop. In a letter to the NRC dated December 21, 1984, the licensee informed the NRC of their desire to utilize their existing systems to meet this requirement because of developmental problems with the EDAS. The present system will utilize existing computers and emergency plan procedures EM-204(A), EM-204(B), and EM-204(C).

The inspector has reviewed these procedures and determined that the procedures provide the stated dose calculations. These procedures were also previously reviewed by the NRC as documented in NRC Inspection Reports 50-302/84-18, 50-302/84-28, and 50-302/85-02. As a result of these reviews, one procedure, EM-204(C), has an outstanding Inspector Followup Item (302/84-18-04) pending. Further activity in this area will be tracked by this followup item.

Based upon these reviews, the licensee's activities on this item appears to be complete. Outstanding specific items will continue to be tracked by the identified Inspector Followup Items.

n. III.D.3.4, Control Room Habitability: The licensee has completed the installation and has the control room toxic gas monitoring system operable. Verification of this installation is documented in NRC Inspection Report 50-302/84-09 and operation and testing of this system is periodically observed by the inspectors.

In their submittal to the NRC dated January 30, 1981, the licensee committed to provide, in addition to the toxic gas monitors, a system that would allow pressurizing of the control room. This submittal was approved by the NRC in a Safety Evaluation Report (SER) dated February 17, 1982. During further correspondence between the NRC and FPC regarding this issue, the licensee was requested to provide implementation dates. In the licensee's response that provided these dates, the control room pressurizing issue was overlooked.

During discussions with licensee personnel, it has been determined that the licensee is not intending to pressurize the control room but will instead go to a zone isolation concept. The licensee has let a contract to Gilbert Associates Incorporated (GAI) to re-analyze a study of control room habitability during accident conditions. This re-analysis will be done utilizing the existing control room ventilation system. Based upon the results of this study, modifications to the control room ventilation system, if required, will be made.

The licensee expects to complete this study by March, 1986. Any required modifications will be completed by the end of the next refueling (Refuel VI).

This item will remain open pending completion of the required control room pressurizing modifications.

8. Design, Design Changes and Modifications

Installation of new or modified systems were reviewed to verify that the changes were reviewed and approved in accordance with 10 CFR 50.59, that the changes were performed in accordance with technically adequate and approved procedures, that subsequent testing and test results met acceptance criteria or deviations were resolved in an acceptable manner, and that appropriate drawings and facility procedures were revised as necessary. This review included selected observations of modifications and/or testing in progress.

Modifications reviewed during this inspection period are delineated in paragraph 7 of this report.

9. Nonroutine Event Followup

a. At 11:45 p.m., on September 27, 1985, after a power reduction to approximately 13% of full power to conduct turbine generator repairs, control rod group number 7 dropped into the reactor core. Control rod number 1 in rod group number 7 (rod 1-7) was being powered from the control rod drive auxiliary power supply to align this control rod with the rest of the control rods in group number 7. The dropping of the control rod group occurred while transferring rod 1-7 from the auxiliary power supply back to its normal group power supply. The addition of the negative reactivity resulting from the control rod group insertion placed the reactor in a shutdown condition. The plant was placed in the hot standby (Mode 3) condition at 12:20 a.m., on September 28, to effect repairs to the control rod drive system.

   The cause for the dropped control rod group is attributed to a faulty jogging motor in the auxiliary power supply's programmer control assembly. This faulty motor created a voltage transient which interrupted power to the entire group 7 control rods allowing the entire group to lose power momentarily and drop into the core. Repairs to the control rod drive auxiliary power supply were completed and the reactor was taken critical at 5:13 a.m., on October 1, followed by resumption of power operation at 6:38 a.m.

   The inspector observed the activities of troubleshooting the cause for the dropped rod group and observed the plant startup following repairs.

   No violations or deviations were identified.

b. At 9:34 a.m., on October 9, 1985, the reactor was manually tripped due to the inadvertent closure of two Main Steam Isolation Valves (MSIVs), MSV-413 and MSV-414, for the "B" Once Through Steam Generator (OTSG). Troubleshooting on the Emergency Feedwater Initiation and Control (EFIC) system was in progress at the time the MSIVs closed.

   The EFIC system requires two out of four main steam isolation channels to trip to cause the MSIVs to close. Troubleshooting on the EFIC system was in progress to determine the cause for a half trip signal being present on the main steam isolation channel for the B OTSG. The B EFIC cabinet was deenergized to replace a main steam isolation trip module believed to be the cause of the problem. Unknown to the troubleshooting technicians, this module replacement did not correct the cause of the problem and the half trip signal was still present. When an EFIC cabinet is reenergized all trip modules assume trip status until reset by the technicians. Upon reenergizing the B EFIC cabinet, the necessary two trip signals were present (one from the uncorrected half trip problem and one from the recently energized B EFIC channel) and MSV-413 and 414 closed. The inspectors arrived in the control room shortly after the reactor trip had occurred and verified the status of safety systems, plant status, and plant parameters. The inspectors also observed the troubleshooting and maintenance on the EFIC system and reviewed the licensee's post trip evaluation.

No violations or deviations were identified.

c. At 12:35 p.m., on October 8, 1985, the inspectors were informed of an inadvertent start of the "A" Emergency Diesel Generator (EDG-3A). As part of the review for this event, the inspector reviewed a work package for troubleshooting the EDG-3A drop target annunciator panel which was in progress at the time the diesel started. After discussions with the electricians performing the work, the inspector learned that when an attempt was made to deenergize the drop target panel by opening a breaker on the Engineered Safeguards Diesel Generator D. C. Panel 3A (DPDP-6A), the wrong breaker was inadvertently opened which deenergized two emergency diesel generator air start solenoid valves, EGV-36 and EGV-37. This allowed the admission of air to the diesel and it started. The inspector noted that the trouble-shooting being performed was not classified as safety related.

The inspector reviewed the Safety Listing to determine the safety classification of the EDG-3A drop target annunciator panel. The Safety Listing references Electrical Circuit Schedule Drawings (E-212 series) to determine the safety classification of specific electrical circuits. The inspector reviewed drawing E-212-027 and based on the information from this drawing it appears that the drop target panel should be classified as safety related. The inspector contacted licensee representatives to obtain more precise information. The licensee is presently researching this item to determine the safety classification of this panel. This item will be considered unresolved pending the inspector's review of this safety determination.

Unresolved Item (302/85-41-08): Review the licensee's determination of the safety classification of the EDG-3A drop target annunciator panel.

While reading the Safety Listing, the inspector was unable to find DPDP-6A listed. The inspector discussed this omission with licensee representatives and it was determined that DPDP-6A should be classified as safety related and included in the Safety Listing. It appears that a previous revision to the Safety Listing had incorrectly omitted this panel from the current listing. The licensee plans to revise the Safety Listing to include DPDP-6A. The inspectors have previously identified their concerns over the adequacy of the Safety Listing to properly classify systems in NRC inspection report 50-302/85-29 (Unresolved Item 302/85-29-01). This item appears to be another example of safety listing inadequacy.