U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.: 50-321 and 50-366
License Nos.: DPR-57 and NPF-5

Report Nos.: 50-321/96-07 and 50-366/96-07

Licensee: Georgia Power Company
P.O. Box 1295
Birmingham, AL 35201

Facility Name: Hatch Units 1 and 2
Location: Baxley, Georgia
Dates: May 12, 1996 - June 22, 1996

Inspectors: FOR R.W. Wright 7/19/96
B. L. Holbrook, Sr. Resident Inspector Date Signed
E. F. Christnot, Resident Inspector
G. W. Salyers, Emergency Preparedness Inspector
(Sections P3.2, P4.1-P4.4)

Accompanying Inspector: J. A. Canady

Approved by: Pierce H. Skinner 7/19/96
Pierce H. Skinner, Chief, Date Signed
Project Branch 2
Division of Reactor Projects

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EXECUTIVE SUMMARY

Plant Hatch, Units 1 and 2
NRC Inspection Report 50-321/96-07, 50-366/96-07

This integrated inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covers a 6-week period of resident inspection; in addition, it includes the results of an announced inspection by a regional based inspector.

Operations

Operators acted conservatively during the Unit 1 transient and manual scram that occurred on May 26. The decision to insert a manual scram and initiate the Reactor Core Isolation Cooling System following the loss of feedwater indicated sound judgement on the part of shift operators and supervision. This was a positive observation for operations performance during this transient (Section 01.2).

The event review and root cause determination for the feedwater transient was very through and identified the most probable cause of the failure. Improvements can be made in the engineering design review process that could have possibly prevented the transient (Section 01.2).

Operator activities to verify proper component operation following setpoint changes made to motor control center breaker instantaneous trip devices were well documented and controlled with supervisory and management oversight. The devices tested worked properly and all acceptance criteria were met (Section 02.1).

The inspectors reviewed the Hatch Strike Contingency Plan and did not identify any significant deficiencies. The plan appeared to be thorough and comprehensive. The assumptions used for plan development were appropriate and reasonable. The number of qualified personnel identified for duty were appropriate for proper and safe operation of the site. The staffing and operational plan met the requirements of both units Technical Specifications and Final Safety Analysis Reports (Section 6.1).

Maintenance

Maintenance and inspection activities observed on the 1B Emergency Diesel Generator were satisfactory. The inspectors concluded that licensee management attention to the generator stator winding spacer problem was appropriate. Engineering and maintenance performance to inspect, monitor and correct the deficiencies were satisfactory. The long term corrective action for generator replacement demonstrated a management commitment to safe and reliable equipment (Section M1.2).

The observed maintenance activities to replace some Unit 2 Station Service battery cells were completed in a thorough and professional manner with engineering and supervisory oversight. The licensee's present plan to test, monitor and replace cells that may potentially develop a short was appropriate (Section M1.3).

The inspectors determined that supervisors were present, procedures were used and work was performed in a professional manner during the repair of the Unit 1 B Reactor Feed Pump check valve. No deficiencies were observed (Section M1.4).

An unresolved Item (50-321,366/96-07-01: Determine Safety Significance and Testing Requirements for Unit 1 and Unit 2 Containment Isolation Status Panel), was documented. The inspectors identified that these panels were not part of the routine testing program and some indications were not verified for proper operation (Section M3.1).

The inspectors identified that during routine surveillance activities the crew demonstrated good communications, coordination, and procedure usage. The pre-job briefs were considered good (Section (M3.1).

Engineering

A violation (50-321,366/96-07-02: Failure to Conduct Testing Following Molded Case Circuit Breaker Instantaneous Trip Setpoint Changes), was identified (Section E2.1).

An inspector Followup Item (50-366/96-07-03, Degradation and Replacement of Unit 2 Station Battery 2B Due to Buildup of Cell Sediment), was identified. The long term plan for battery replacement was reasonable. The inspectors will observe licensee activities to monitor battery performance and conduct battery replacement (Section E2.2).

The inspectors concluded that the licensee's refueling methodology, with respect to heat load removal capability, was bound by the Updated Final Safety Analysis Report (UFSAR). The description in the UFSAR of the maximum heat load condition was consistent with what the licensee does during a normal routine refueling outage (Section E2.3).

The inspectors concluded that work activities for control rod blade disposal were conducted and controlled in accordance with plant procedures. Health Physics personnel were present and provided adequate work coverage. The inspectors observed that the

Refuel Floor Coordinator was present on the refuel floor, and was attentive to work activities (Section E2.4).

Plant Support

In general, Health Physics practices and radiological controls were satisfactory. General housekeeping could be improved (Section R1.1).

The inspectors observed activities in the Operations Support Center during a recent emergency preparedness exercise and concluded that Health Physics instruments were available, calibrated and well maintained. Supplies were readily available and procedures were the current revision. Improvements could be made in the command and control function of the Operations Support Center (Section P2.1).

The inspectors concluded that licensee's use of the Emergency Action Levels in classifying events during the recent emergency preparedness exercise was satisfactory. The inspectors concluded that the Senior Reactor Operator was aware that some dose calculations were incorrect (Section P3.2).

The inspectors concluded that during the recent emergency preparedness exercise accountability was satisfactorily performed in the required time. The licensee identified areas in which the process could be improved (Section P4.1).

The inspectors concluded that during the recent emergency preparedness exercise the licensee's internal communication was adequate but substantial communication improvements can be made (Section P4.2).

The inspectors determined that during the recent emergency preparedness exercise the licensee maintained a computerized dose assessment system and its staff could perform dose assessment. Improvements can be made in drill control (Section P4.3).

The overall performance of the licensee during an annual emergency preparedness exercise was adequate. Substantial improvements can be made. Notification of offsite authorities was identified as an Exercise Weakness. Inspector Followup Item 50-321,366/96-07-04: Exercise Weakness for Failure to Make Adequate Notifications to State, Local, and Federal Authorities was documented (Section P4.4).

All observed work activities for fire header leaks were performed in a controlled manner with engineering and supervisory oversight. The leak repairs appear to be appropriately prioritized to maintain a satisfactory fire protection system. The overall efforts to control underground fire main leaks were reasonable (Section F2.1).

Report Details

Summary of Plant Status

Unit 1 began the report period at 60% Rated Thermal Power (RTP) after a power reduction due to a problem with the Reactor Feedwater Pump Turbine (RFPT) 1B check valve. The check valve was repaired and 100% RTP was achieved on May 13. Power was then reduced to about 75% RTP due to the closing of a turbine control valve. Repairs to the valve were completed and 100% RTP was achieved the same day. On May 25 power was reduced to about 18% RTP to repair an Electro Hydraulic Control (EHC) leak. On May 26, the reactor was manually scrammed following the trip of both RFPTs. Repairs were completed on the RFPT control system. A reactor startup was commenced on May 27, and 100% RTP was reached on May 28. The unit operated at 100% RTP for the remainder of the report period except for routine testing activities.

Unit 2 operated at 100% RTP throughout the report period except for routine testing activities.

I. Operations

O1 Conduct of Operations

O1.1 General Comments (71707)

Using Inspection Procedure 71707, the inspectors conducted frequent reviews of ongoing plant operations. In general, the conduct of operations was professional and safety-conscious; specific events and noteworthy observation are detailed in the section below.

O1.2 Unit 1 Power Reduction and Manual Scram

a. Scope of Inspection (92901) (93702)

On May 26, the inspector responded to the site to provide prompt assessment, inspection and review licensee activities following a manual scram from about 18% Rated Thermal Power (RTP). The scram was initiated due to a loss of Feedwater (FW) following a dual Reactor Feedwater Pump Turbine (RFPT) trip.

b. Observations and Findings

Due to an Electro Hydraulic Contro (EHC) fluid leak on valve 1N32F021, Turbine Extraction Relay Dump Valve, the licensee began reducing power on May 25 to remove the main turbine from service and initiate repairs. Power was reduced to about 60% RTP and RFPT 1B was removed from service and placed in standby. RFPT 1A supplied water to the reactor.

Reactor power was then reduced to about 18% RTP and the main turbine was removed from service. At about 5:14 a.m. both RFPTs tripped. The thrust bearing wear trip alarm actuated for both pumps. The operators attempt to reset the RFPTs was unsuccessful. A manual scram was initiated prior to reaching the automatic low level scram setpoint. Reactor Core Isolation Cooling (RCIC) was initiated to restore and maintain reactor level. The lowest reactor water level indicated was about 128 inches above the top of active fuel.

The inspector discussed the event with licensee personnel, reviewed recorder data, and reviewed operator log entries. The inspector verified that the unit was stable and all control rods were inserted. Observed operator activities indicated an awareness of the overall plant status. A review of Emergency Operating Procedures (EOPs) used was conducted. The licensee's notification to the NRC Headquarters Operation Office, due to an Engineered Safety Features (ESF) and Reactor Protection System (RPS) actuation, was correct and timely.

The licensee identified the initiating problem as a current limiter in a circuit board for a common power supply for the recently-installed General Electric Mark 5 RFPT control system. This failure resulted in an instantaneous trip of both RFPTs that was not resettable. The licensee discovered that installed test leads were causing a grounding condition when physically moved. This was initially thought to have resulted in a blown fuse which caused the current limiter failure. As part of the immediate corrective actions the test leads were removed. Additional analysis and troubleshooting indicated that a defective capacitor in the current limiter was the most likely cause. The inspector was informed that the Unit 1 test leads would be reinstalled at the next opportunity.

The inspector noted that the RFPTs on both units have the new RFPT control system and unit 2 RFPTs were susceptible to a similar failure.

The licensee later determined a less than adequate design contributed to the problem. The RFPTs protection logic was not intended to be affected by a single failure of the circuit board. As part of the licensees long term corrective actions a design change to correct the problem will be completed during the next refueling outages. Refueling outages were scheduled during the Fall and Spring of 1997 for Unit 1 and Unit 2, respectfully.

c. Conclusions

Based on the observations, reviews and discussions the inspectors concluded that the operators acted conservatively in accordance with EOPs. The decision to insert a manual scram and initiate the

RCIC systems indicated sound judgement on the part of shift operators and supervision. Overall operator performance was good.

The event review and root cause determination by the Event Review Team was very thorough and identified the most probable cause of the failure. Improvements can be made in the engineering design review process that could have possibly prevented this transient.

O2 Operational Status of Facilities and Equipment

O2.1 Operation of Motor Control Center, Motor Operated Valves and Electrical Loads

a. Inspection Scope (71707) (92901)

The inspectors used Inspection Procedure 71707 and 92901 to observe and assess operator activities involved with cycling selected Motor Control Center (MCC) Motor Operated Valves (MOV) and testing of electrical loads.

b. Observations and Findings

These activities were performed to verify proper component operation following setpoint changes made to MCC breaker instantaneous trip devices. (Reference paragraph E2.1). Commencing on May 30, and completing on June 3, 1996, operations personnel tested MCC electrical loads using applicable portions of plant procedures. Among the procedure usage observed were the following:

- 34SV-E11-002-1S, RHR Valve Operability - Unit 1
- 34SV-E21-002-2S, Core Spray Valve Operability - Unit 2
- 34SV-E41-001-2S, HPCI Valve Operability - Unit 2

c. Conclusions

The results of these activities were well documented and controlled with supervisory and management oversight. The devices tested worked properly and all acceptance criteria were met. The inspectors did not identify any concerns involving these activities.

06 Operations Organization and Administration

06.1 Strike Contingency Plan

a. Inspection Scope (92709)

The inspectors used Inspection Procedure 92709 to review Hatch strike contingency plans. Among the items reviewed were the following:

- Requirements for minimum onsite shift staffing for various departments including plant management, operations, maintenance, security, chemistry and radiation protection, surveillance and calibrations, and administration

- Access to the plant for personnel, radwaste shipments, delivery of support goods, medical care and ambulance services, local agencies and local fire departments.

- Number of qualified personnel to implement the emergency plan and availability of Emergency Notification System (ENS) and emergency communication equipment.

b. Observations and Findings

The inspectors reviewed the licensee document titled, Plant E. I. Hatch Contingency Plan for IBEW Local 84 Strike. The document contained assumptions used for plan development and detailed plans in the areas of operations, maintenance, plant administration, departmental roles, corporate support staffing and additional support personnel. The plan indicated shift schedule for various departments and support groups. The inspector discussed the plan with various licensee management and supervisory personnel. On June 26, a meeting was held between NRC management and licensee management in the Region II office to discuss the plan. The inspectors also reviewed the following to ensure specific requirements were met:

- Units 1 and 2 Technical Specifications (TS) Section 5.0, Administrative Controls

- Unit 1 Final Safety Analysis Report (FSAR), Section 1.9, Plant Management; Section 9.3, Solid Radwaste System; Chapter 13, Conduct of Operations; and Appendix D, Quality Assurance Program

- Unit 2 FSAR, Section 11.5, Solid Radwaste System; Chapter 13, Conduct of operations; and Section 17.2, Quality Assurance During the Operations Phase

- 10 CFR 50.54, Conditions of Licenses, paragraph (m)(2)

c. Conclusions

The inspector's review of the Hatch Contingency Plan did not identify any significant deficiencies. The plan appeared to be thorough and comprehensive. The assumptions used for plan development were appropriate and reasonable. The number of qualified personnel identified for duty was appropriate for proper and safe operation of the site. The staffing and operational plan met the TS and FSAR requirements.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments

a. Inspection Scope (62703)

The inspectors observed all or portions of the following work activities:

- Maintenance Work Order (MWO) 1-96-0496: Replace Standby Lube Oil Pump on EDG 1B
- MWO 1-96-0912: Inspect EDG 1B Stator Windings
- MWO 2-96-0044: Replace Cells 16 and 108 on SS Battery 2B
- MWO 2-96-0407: Perform 1 and 5 Year PMs on Security EDG

b. Observations and Findings

Maintenance activities were observed and reviewed during the reporting period to verify that work was performed by qualified personnel and that procedures adequately described work that was not within the skill of the trade. Activities, procedures, and work request were examined to verify authorization to begin work, provisions for fire hazards, cleanliness, exposure control, proper return of equipment to service, and that Technical Specification (TS) requirements were met.

In addition, see the specific discussions of maintenance and surveillances observed under sections M1.2, M1.3, M1.4, and M3.1, below.

M1.2 Maintenance on 1B Emergency Diesel Generator (EDG)

The inspector observed partial performance of maintenance on the 1B EDG under MWOs 1-96-496 and 1-96-912, and reviewed licensee documentation of the work activities.

b. Observation and Findings

Licensee personnel and technicians from the Forrest Park, GA, maintenance support shop implemented the work instructions contained in MWOs 1-96-496 and 912. The work activities involved the replacement of the EDG standby lube oil pump and inspection of the stator windings. The pump was replaced due to an oil leak at the shaft seal. The inspection of the stator by use of fiber optics was an ongoing activity and has been previcusly discussed in NRC Inspection Reports 50-321,366/94-27, 95-08, 96-02, 96-04 and 96-06.

The inspectors were informed that the inspection of the 1B EDG stator indicated that the end turn spacers, both the original and the recently installed replacement spacers, were still coming loose. The licensee had determined, as previously documented, that the movement of the original spacers was due to the stator end turns flexing during EDG operation. The movement of the recently installed spacers was due to release of some of the bonding material for the new spacers. The loose spacers were removed. The licensee stated that the spacers will be replaced at the next available opportunity. The licensees safety evaluation, as previously documented, determined that the spacer problems did not present any immediate operability concern.

c. Conclusion

The inspectors concluded that licensee management's attention to the EDG spacer problem was appropriate. Engineering and maintenance performance to inspect, monitor and correct the deficiencies were satisfactory. The long-term corrective action for EDG replacement demonstrate a management commitment to safe and reliable equipment. Maintenance activities conducted to replace the standby oil pump were conducted in accordance with approved procedures and with appropriate supervision.

M1.3 Replacement of Cells in Unit 2 Station Service (SS) Battery

a. Inspection Scope (62703)

The inspector observed maintenance activities during the replacement of Unit 2 SS Battery cells 16 and 108 performed under MWO 2-96-044. The inspectors also reviewed licensee documentation of completed work activities.

b. Observations and Findings

Licensee personnel identified that 42 of the 120 cells in the 2B SS battery (safety related) contained excessive amounts of sponge like sediment and deposits. The licensee stated that the sediment appeared to be the result of active plate material being shed due to plate impurities introduced during the manufacturing process.

The licensee, battery vendor, and architect/engineer did not believe the condition caused an immediate operability concern. To correct the problem the licensee implemented a plan to immediately replace some cells and to eventually replace all cells.

The inspectors discussed the problem with licensee management. The inspectors were informed that the 2B battery was purchased in 1993. The battery capacity test performed during the fall of 1995 indicated that the battery capacity was about 114%.

The inspectors observed portions of the cell replacement activities. The work was performed using engineering procedure 42EN-R42-001-OS, Individual Cell Isolation and Load Testing, Revision 1. The activity involved the disconnecting and jumpering out of the old cell; ensuring that the new cell is in the charged condition prior to installation; by using proper lifting technique and equipment, removing the old cell and installing the new cell; ensuring that new cell has the proper voltage and orientation; and removing the jumper and connecting the new cell into the battery.

The inspector reviewed TS 3.8.4, DC Sources - Operating, Limiting Conditions for Operations (LCO) 3.8.4, Required Action Statement (RAS) C.1 and observed that the completion time for RAS C.1 was two hours. The inspector observed that the changeout of the cells were accomplished within the required TS action time.

The inspectors reviewed section 8.3 of the Unit 2 Final Safety Analysis Report (FSAR) and did not identify discrepancies between the FSAR and plant practices and procedures.

c. Conclusions on Conduct of Maintenance

The observed maintenance activities were completed thoroughly and professionally with engineering and supervisory oversight. Concerns were not identified as a result of the observed activities. The inspectors concluded that the licensees present plan to test, monitor and replace cells that may potentially develop a short was appropriate.

M1.4 Check Valve Repair on the 1B Reactor Feed Pump

a. Inspection Scope (61726)

Reactor power was reduced on Unit 1 to repair a leaking discharge check valve on the B Reactor Feed Pump. The inspectors observed work activities in progress and reviewed the licensee's documentation for the completed work.

b. Observations and Findings

The leaking check valve was identified by the licensee during a walkdown of the system. The leak was about 40 drops per minute.

Maintenance personnel torqued the pressure seal bolts and the leak appeared to stop. Later the leak started again and was worse than before. The valve required disassembly. A small steam cut was machined smooth and the valve was reassembled, torqued and placed in service. The leak was repaired.

The inspectors observed work activities, reviewed MWO 1-96-1746, Inspect/Replace Pressure Seal on B RFPT Discharge Check Valve, and reviewed applicable maintenance procedures for the work performed.

c. Conclusions

The inspectors observed that supervisors were present, procedures were used and work was performed in a professional manner. No deficiencies were observed.

M3.1 Surveillance Observations

a. Inspection Scope (61726)

The inspectors observed all or portions of the following Unit 1 and Unit 2 surveillance activities:

- 42SV-G31-002-1S, RWCU Isolation LSFT
- 34SV-E41-002-2S, HPCI Pump Operability

b. Observations and Findings

The Reactor Water Clean-Up (RWCU) Isolation Logic System Functional Test (LSFT) was a revised procedure in that this LSFT was previously performed during refueling outages. The revised procedure was written to allow performance of the surveillance with the unit online. The procedure required that the two RWCU pumps be disabled from starting. This was accomplished by disconnecting motor leads and/or removal of thermal overloads. However, the pump motor starters would close and give light indications in the Control Room (CR).

The individual system isolation valves operated when required. The valve position indicating lights at the system control board indicated properly. However, the inspectors observed that the corresponding lights on the CR containment isolation status panel did not indicate properly. This problem was corrected by bulb replacement. However, the inspector observed that a check of the indicating lights on the CR containment isolation status panel, on the vertical section of panel 1H11P601, was not part of the procedure requirements. Engineering personnel involved with the LSFT did not readily know whether or not the isolation status panel indicating lights for the RWCU were ever tested or verified. The inspectors were not aware of any time that the containment isolation status panel did not accurately reflect system configuration.

The inspectors observed that the isolation status panel was not addressed in TS. However, the inspectors observed that operators routinely use the containment isolation status panel to verify system configuration. Additionally, plant procedures, such as the abnormal procedure for a reactor scram, indicated this panel could be used for system status verification.

The inspectors discussed this observation with operations and engineering management. The inspectors were informed by management personnel that the containment isolation status panel was more of an operator aid than a system panel. They were not aware of any requirement for panel testing or verification that the indications operated correctly. They were not sure to what extent or if the panel was ever tested.

The inspectors also observed a Unit 2 High Pressure Injection Coolant (HPCI) operability surveillance and inservice testing (IST). Operations personnel with the support of maintenance, health physics (HP), and engineering performed the test on June 18. The inspectors attended the pre-job brief; observed portions of the surveillance; and reviewed the IST, TS, and operator logs. The calibration dates of the timing instruments used for response time testing were current.

c. Conclusions

The inspectors concluded that RWCU Isolation LSFT was conducted in accordance with procedures by knowledgeable personnel. Coordination between operations and engineering personnel was good. The inspectors concluded that since the operators are directed by plant procedures and routinely use the system isolation status panel to verify system configuration, a reasonable expectation was that the panel be tested or verified to ensure that system configuration is correctly indicated.

The inspectors will conduct a more detailed review to determine the safety significance and testing requirements of the containment isolation status panel. This item is identified as Unresolved Item (URI) 50-321,366/96-07-01: Determine Safety Significance and Testing Requirements for Unit 1 and Unit 2 Containment Isolation Status Panel, pending additional review by the inspectors of this issue.

During the performance of the HPCI/IST Pump Operability Test, coordination between operations, maintenance, HP, and engineering personnel was good. Personnel involved in the performance of the surveillance complied with procedural requirements. Necessary data entries were made in the appropriate control room logs. The pre-job brief was thorough, pre-planned and effective in coordinating the duties of the groups involved in the performance of the surveillance.

M8 Miscellaneous Maintenance Issues (92700) (92902)

M8.1 (Closed) Licensee Event Report (LER) 50-366/95-10: Component Failure Results in Engineering Safety Feature System Actuation. The inspectors reviewed the licensee's documentation dated January 2, 1996. A partial Primary Containment Isolation System (PCIS) signal was received from the main steam line high radiation monitors. The problem was caused by a failed soldered connection internal to the hydrogen flow rate monitor that resulted in an increased hydrogen injection flow rate. The hydrogen monitor was replaced and the system returned to operable status. Based upon the inspectors' review of the licensee actions, this item is closed.

III. Engineering

E1 Conduct of Engineering

On-site engineering activities were reviewed to determine their effectiveness in preventing, identifying, and resolving safety issues, events, and problems.

E2 Engineering Support of Facilities and Equipment

E2.1 Trip Setpoint Changes on Circuit Breakers

a. Inspection Scope (92903)

The inspectors performed followup activities associated with the adjustments of instantaneous trip setpoints on molded-case circuit breakers (MCCB). The inspectors observed that, on May 21, the breaker for B loop Residual Heat Removal (RHR) torus cooling valve, 2E11F024B, tripped when the control switch was placed to OPEN. The system was started in preparation for a routine High Pressure Control Injection (HPCI) monthly operability surveillance.

b. Observations and Findings

NRC Inspection Report (IR) 50-321, 366/96-06 documented a violation for NRC identified deficiencies involving incorrect setpoints for MCCBs. Following the NRC-identified deficiency, the licensee initiated a review of existing MCCB instantaneous trip setpoints. They identified a total 577 MCCBs that required trips to be verified and adjusted if necessary. They also established acceptance criteria and developed a methodology for setpoint calculations. An additional licensee goal was to establish an instantaneous trip setpoint index.

The licensee determined that some setpoints should be adjusted while others were determined to be satisfactory. Engineering and maintenance personnel issued a series of documents to start a setpoint adjustment program for both units. The program began during the week of May 13, 1996.

During maintenance trouble shooting activities, following the trip of the circuit breaker for valve 2E11F-24B, on May 21, maintenance personnel determined that the trip setpoint was set low. The trip setpoint had been adjusted from a HIGH setting to a value of 8, on May 14. Maintenance personnel adjusted the trip setpoint back to the original HIGH setting. Subsequent operability test demonstrated the valve functioned properly.

Following the breaker trip problem, engineering personnel halted their setpoint change activities to reassess the setpoint calculations. They determined that some assumptions used for trip setpoint calculations for some breakers were not valid. The trip setpoints for selected breakers would require re-adjustment.

The inspectors discussed the problem with the licensee management, operations, maintenance and engineering personnel. Licensee management discussed their breaker trip setpoint assumptions, methodology and overall plan with NRC management in a conference call on May 29.

The licensee modified the MCCB trip setting action plan. The inspectors reviewed the licensee's modified action plan and discussed the plan with licensee management and engineering personnel.

Engineering personnel determined that the operating characteristics for valve 2E11F24B, as well as other breakers that required re-adjustment, were outside the generic assumptions used to determine trip settings for a group of similar valves and motors. An inaccurate value for locked rotor amps was used to determine the MCCB setting for RHR valve 2E11F024B. The trip setpoints identified as being incorrect were reset and the loads were tested by June 28, 1996, to verify proper operation. All acceptance criteria were met and no deficiencies were identified.

Eight additional breakers were identified where the breaker trip settings were lowered but could not be tested due to existing plant conditions. The breaker trip setpoints were returned to their original as-found settings. They will be cycled for testing the next available opportunity. Because none of these valves were tested after their trip settings were lowered, they were all susceptible to the same failure as valve 2E11F024B until their trip setpoints were returned to original setting.

The eight breakers that were not tested and were subsequently returned to their original as-found trip setpoint condition were:

- Unit 1 Plant Service Water (PSW) valves P41-F049 and F050;

- Reactor Building Closed Cooling Water (RBCCW) valve P42-F051;

- Unit 2 RHR valve E11-F009;

- RHR valve E11-F015B;

- Reactor Water Clean-up (RWCU) valve G31-F001;

- RBCCW valves P42-F051 and F052;

The inspectors reviewed licensee documentation and procedures that controlled the work activities. The documents provided information and direction for determining the proper setpoints, adjusting the setpoints and recording the as found/as left data. Documents reviewed included:

- Criteria for Determining Recommended Instantaneous Trip Setting for MCCBs

- MWOs 1-96-1771 and 2-96-627

- Engineering Instructions -- MCC Checklist for 2R24-S012, 600/208V MCC 2B Div 2

- Procedure 52PM-R24-001-OS: Allis Chalmers Low Voltage MCC Inspection

- 50AC-MNT-001-OS: Maintenance Program

The inspectors determined that the work activities were controlled by the MWO process, using engineering instructions. The licensee did not treat the setpoint changes as a design change or plant modification. Procedure 50AC-MNT-001-OS: Maintenance Program, Revision 24, was used during the activity and contained instructions to identify, on block 32 of the MWO, specific testing requirements. In this case, the MWOs did not identify any requirement for functional testing and no testing was performed after the setpoint changes.

The engineering instructions provided did not identify testing requirements or contain acceptance criteria to verify that the end devices would perform as designed after the MCCB instantaneous trip set points were adjusted.

Procedure 52PM-R24-001-OS was also used during the activity and contained provisions for testing instantaneous trip setpoints in steps 7.4.7.16 - 7.4.7.21. However, engineering personnel determined that testing was not necessary and these steps were marked as not applicable.

The inspectors questioned licensee management with respect to the lack of any functional testing. They were informed that the lack of testing was discussed, especially with two licensed operators who questioned why testing was not required. Engineering personnel informed the inspectors that tests were not being performed because the setpoints were being adjusted from calculations derived from a methodology and standard that was nationally recognized. Engineering stated that testing would not serve any beneficial function.

c. Conclusions

The inspectors concluded that the failure to conduct testing activities following MCCB setpoint adjustments (particularly lowering the setpoints) was not a conservative decision. Between May 9 and 21, a total of 295 instantaneous trip setpoints on circuit breakers, located on both units, were reviewed. The set points on 72 of these breakers were decreased without the issuance of either a design change document and/or a post modification/maintenance test.

Procedure 50AC-MNT-001-0S: Maintenance Program, was not fully implemented in that the MWO did not identify testing requirements.

10 CFR 50, Appendix B, Criterion V states, in part, that activities affecting quality shall be prescribed by documented instructions, procedures or drawings. Criterion V also states, in part, that the instructions, procedures or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.

In this case, work activities were completed and the work instructions did not contain appropriate acceptance criteria for determining that important activities were satisfactorily accomplished.

The inspectors concluded that functional testing following work activities of RHR valve 2E11F024B may have provided early identification of the valve failure that occurred during normal plant activities on May 21. However, the inspectors concluded that the TS required Action Statement for the RHR valve 2E11F024B was not exceeded. In this case, the safety significance was minimal. However, during other plant conditions, the failure to conduct testing following breaker trip setpoint changes could present a more safety-significant problem.

The failure to identify and complete testing requirements following MCCB instantaneous trip setpoints changes was a significant oversight. This problem is identified as VIO 50-321, 366/96-07-02: Failure to Conduct Testing Following Molded Case Circuit Breaker Instantaneous Trip Setpoint Changes.

E2.2 Sediment in Unit 2 Station Service (SS) Battery 2B

a. Inspection Scope (92903)

The inspectors discussed and reviewed with engineering personnel, the licensee's assessment and replacement plan for the 2B SS battery. Excessive sediment buildup led to degradation of the battery which required the replacement of two cells.

b. Observations and Findings

The inspectors were informed by the licensee that, due to the sediment buildup in the jars in Unit 2 SS Battery 2B, plans were being made to replace all 120 cells. The expected time of arrival of the new cells was the end of July, 1996. The licensee planned for battery cell replacement at the next refueling outage (Spring 1997). However, engineering personnel were monitoring (cell voltage check) 30 cells on a rotating basis so that the total battery was assessed each month. Additionally, weekly visual inspections of the cells were conducted. Based upon the cell voltage checks and visual inspections, the licensee was prepared to change out specific cells as necessary. The licensee indicated they could replace the total battery with the unit in operation, if required.

Engineering personnel stated that the excessive sediment appears to be the result of active plate material being shed due to contamination or impurities introduced into the plate material at the factory. Discussions with licensee personnel indicated that the plates may not have been cured properly after fabrication and this could also be the cause of the sediment buildup. The inspectors observed that, as of the first week of June, 1996, a total of 42 cells had sediment.

The inspectors observed the sediment and noted that the color of the sediment matched the color of the cells positive plates, indicating possible plate dissolution. The sediment buildup could possibly result in a battery short circuit. However, the inspectors were not aware of any previous battery cell or battery short circuit problem.

The inspectors' review of Unit 2 TS 3.8.4, indicated that operability or performance of individual battery cells is not a TS requirement. Surveillance Requirement 3.8.4.1 required that terminal voltage for each half of the SS batteries be equal to or greater than 125 volts while on float charge. The inspectors' and licensee's observation of battery performance indicated that battery voltage was usually above 130 volts, meaning that several cells could fail and the battery would remain operable.

The inspectors were informed that cell 16 was shipped to the manufacturer for failure analysis. The manufacture determined that the cell sediment was due to a manufacturing defect during the plate curing process. The licensee informed the inspectors that the vendor determined that a 10 CFR Part 21 report was not required.

The inspectors observed that during the change out of two cells (see paragraph M1.3) chainfalls and lifting straps were used to move and properly position the new cells. Unlike Unit 1, Unit 2 does not have a permanently installed seismic qualified cell lifting crane and trolley device. The inspectors were informed that a special temporary lifting device would be fabricated to facilitate cell movement.

c. Conclusions

The inspectors did not consider the sediment to be an immediate operability concern. The exact rate of sediment buildup of the battery cells was not determined. However, the licensee is closely monitoring the problem. The sediment has been observed only in the Unit 2 SS Battery 2B cells. No sediment was observed in the other Unit 2 or Unit 1 cells. The inspectors will continue to monitor licensee activities for the assessment of battery performance and replacement. This item is identified as Inspector Followup Item 50-366/96-07-03, Degradation and Replacement of Unit 2 Station Battery 2B Due to Buildup of Cell Sediment.

E2.3 Spent Fuel Pool System Review and Inspection

a. Inspection Scope (37551) (40500)

The inspectors reviewed the Spent Fuel Pool and associated cooling systems.

b. Observations and Findings

The spent fuel pool cooling (SFPC) systems for Units 1 and 2 are of slightly different designs, and are described in Updated Final Safety Analysis Report (UFSAR) Section 10.4.3 for Hatch Unit 1 and Section 9.1.3 for Unit 2, and applicable system operating procedures. The inspectors also reviewed the adequacy of the original heat load design assumptions for the SFPC systems relative to the current operating practice.

Since 1993, the licensee had been primarily using the newly-installed Decay Heat Removal (DHR) system for fuel pool cooling after at least 72 hours following a reactor shutdown. The DHR can be used along with the SFPC system of either unit, if necessary. This system is primarily operated during refueling outages to provide decay heat removal from either Unit 1 or Unit 2 spent fuel pools. Use of the DHR system allows the RHR systems and/or the SFPC system to be taken out of service for inspections, repairs, and/or modifications during outages. The DHR system has been sized to handle a heat load of 40 Million British Thermal Units per hour. Complete loss of the DHR system would not change the conclusions nor the evaluation stated in the UFSAR regarding the adequacy of the SFPC system.

The inspectors reviewed the licensee Document Change Request 96-05 and its associated 10 CFR 50.59 evaluation, dated March 14, 1996, that approved a Unit 1 and Unit 2 UFSAR change. The changes clarified the differences documented in IR 50-321,366/95-23, paragraph 4.e. The changes also clarified the UFSARs regarding what constitutes a normal refueling strategy with respect to how much of the core is offloaded into the spent fuel pool during refueling outages. The change clarified that either a partial or a full-core offload can be performed and that neither one of these is necessarily the normal mode of operation. The UFSAR changes will be submitted at the next scheduled update. The inspectors concluded that the wording in the UFSAR Document Change Request accurately described the licensee's refueling and fuel pool cooling methodology and practice.

c. Conclusions

The inspectors concluded that the licensee's refueling methodology, with respect to heat load removal capability, was bound by the UFSAR. The description in the UFSAR of the maximum heat load condition was consistent with what the licensee does during a normal routine refueling outage.

E2.4 Control Rod Blade Removal from Spent Fuel Pool

a. Inspection Scope (92903)

The inspectors observed the activities of a contract company preparing control rod blades (CRB) for offsite shipment. This activity was controlled by engineering personnel and included the removal of stellite balls and blade shearing, compaction, and placement in cask liners.

b. Observations and Findings

Fuel pool gates were removed and the inspectors verified that the personnel passage beneath the transfer canal was locked as, required by procedure. The locking of the passage was one of the corrective actions implemented due to a spill of stellite balls from a bucket onto the floor of the transfer canal during a similar operation, as discussed in IR 50-321,399/94-31. A small piece of metal was discovered during a dose measurement of the CRB that read approximately 20K rem. The small piece, smaller than a dime, was retrieved and disposed of with the compacted CRB. It was suspected that the sliver of metal came from the CRB cutting operation since it was discovered in the vicinity of the operation.

c. Conclusions

The inspector concluded that work activities were conducted and controlled in accordance with plant procedures. Health Physics personnel were present and provided adequate work coverage. The inspectors observed that Refuel Floor Coordinator was present on the refuel floor, and was attentive to refuel floor work activities.

E8 Miscellaneous Engineering Issues (92700) (92903)

E8.1 (Closed) Licensee Event Report (LER) 50-366/95-08: Reactor Vessel Inventory Loss Results in an Unplanned Engineering Safety Feature System Actuation. The inspectors reviewed the licensee LER documentation dated November 28, 1995. This event was discussed in IR 50-321,366/95-26 as an apparent violation and was the subject of a predecisional enforcement conference conducted in the NRC Region II office on December 28, 1995. The conference resulted in the issuance of a Severity Level IV Violation 50-366/95-26-01. Inspection Report 50-321, 366/95-27 discussed the completion of some of the corrective actions documented in IR 95-26. These actions involved the Unit 1 Remote Shutdown Panel (RSDP) and the operation of the Unit 1 Reactor Core Isolation Cooling (RCIC) from the RSDP. Based on the issuance of the violation and the inspectors review of licensee's corrective actions, this LER is closed.

IV Plant Support

R1 Radiological Protection and Chemistry Controls

R1.1 Observation of Routine Radiological Controls

a. Inspection Scope (71750)

General Health Physics (HP) activities were observed during the report period. This included locked high radiation area doors, proper radiological postings, and personnel frisking upon exiting the Radiological Control Area (RCA). The inspectors make frequent tours of the RCA and discussed radiological controls with HP technicians and HP management.

b. Observations and Findings

During the general tours of the plant the inspectors observed that most areas that had been contaminated during the recent refueling outage were now clean and free of contamination. Barriers and stanchions had been removed and properly stored. However, storage areas were not well-organized and some areas contained trash and dirt. The inspectors observed that dosimetry was properly worn, good frisking techniques were used to and HP assistance for personnel frisking was available when required.

c. Conclusions

In general, HP practices and radiological controls were satisfactory. General housekeeping could be improved.

P1 Conduct of Emergency Preparedness (EP) Activities

a. General Comments

The licensee conducted an off hours, annual EP exercise on May 22, 1996. The exercise started at 7:10 p.m. with a fire in a 4160 Volt-Alternating Current (VAC) switchgear room. The scenario progressed to an Alert at 7:20 p.m., a Site Area Emergency at 8:03 p.m., and a General Emergency at 8:48 p.m.. Offsite authorities did not participate in this licensee-only annual exercise.

P2 Status of Operations Support Center (OSC) Health Physics (HP) Equipment and Resources

P2.1 Status of OSC Equipment

a. Inspection Scope (82301)

The resident inspectors inspected HP instruments and equipment located within the OSC facility for current calibration and battery condition. The availability of required resources was verified.

b. Observations and Findings

The inspectors observed that HP instruments located on tables for current usage and those located within storage cabinets contained a current calibration sticker, were in good condition, and their batteries were properly charged. Other items, such as procedures and supplies, were available. Also, radios and telephones were verified to be operating properly. The inspectors were informed that one fax machine had experienced problems. The inspectors observed that cables from the junction boxes to the communications equipment were placed on the floor and could pose a tripping hazard.

The inspectors observed that adequate resources were available to man the required positions, including field work activities. The inspectors observed that the noise level from drill participants was extremely high. Many of the side discussions were of a personal nature and not related to the drill activities. Command and control to curtail these activities was not implemented. In one instance, the inspectors observed that 19 minutes was required to dispatch a two-person field team from the time a phone request was received. This time was considered excessive. The lack of command and control could be a contributor to this excessive time.

c. Conclusion

The inspectors concluded that HP instruments were available, calibrated and well-maintained. Supplies were readily available and procedures were the current revision. The inspectors concluded that improvements could be made in the command and control function of the OSC.

P3 EP Procedures and Documentation

P3.2 Use Of The Emergency Implementing Procedures

a. Inspection Scope (82301)

The inspectors evaluated the licensee's use of the Emergency Action Levels (EALs) in properly classifying simulated events.

b. Observations and Findings

In the initial classification, the Senior Reactor Operator (SRO), who was the Emergency Director (ED), wanted to declare a Site Area Emergency (SAE), based on the Emergency Action Level (EAL) "fire continuing for greater than 10 minutes and compromising the function of a safe shut down system", rather than declaring an Alert, using the EAL "fire defeating redundant safety system train." The controller directed the ED to declare an Alert. The two EALs were not that clearly distinguishable. Either declaration would have satisfactorily classified the event.

There were no unplanned releases at the beginning of the exercise that would have resulted in a measurable dose at the site boundary. However, as part of their emergency procedure, the Simulator Control Room performed a dose calculation when the initial declaration was made. The calculation results indicated 49 mrem/hr at the site boundary. An indication of 49 mr/hr should have been Classified as an Alert. 50 mr/hr would have been an SAE. The Simulator Control Room did not declared an Alert based upon the result of the calculation or acknowledge that the calculation was in error. During discussions between the licensee and the inspectors, the licensee acknowledged that the dose calculation did not represent the simulated situation.

c. Conclusion

The inspectors concluded that licensee's use of the EALs in classifying the events during the exercise was satisfactory. The inspectors concluded that the SRO was aware that the dose calculation of 49 mrem/hr was incorrect.

P4.1 <u>Personnel Accountability During The Exercise</u>

a. <u>Inspection Scope (82301)</u>

The inspectors observed this portion of the exercise to determine that accountability could be accomplished within the required 30 minutes and to evaluate the licensee accountability process and proficiency.

b. <u>Observations and Findings</u>

Personnel accountability was initiated at the SAE declaration and the results were reported to the ED within the required 30 minutes. The initial report indicated that 64 people were missing. Security promptly started looking for the missing personnel. Security's methodology in locating the missing personnel was inefficient. Security officers personally checked the staff's badges and compared the person's badge to the list of missing personnel rather than communicating the list of missing personnel to facility managers. At the end of the exercise, there were still four personnel unaccounted for.

c. <u>Conclusion</u>

The inspectors concluded that accountability was satisfactorily performed in the required time. The licensee identified areas in which the process could be improved.

P4.2 <u>Communications During The Exercise</u>

a. <u>Inspection Scope (82301)</u>

The inspectors observed this area to determine the flow of information within a facility and from facility to facility within the Emergency Response Organization (ERO). This observation included communication equipment as well personnel communication.

b. <u>Observations and Findings</u>

Delayed and missed communications were noted between the Simulator Control Room, Technical Support Center (TSC), and Emergency Operations Facility (EOF). During the exercise, the EOF asked the Simulator Control Room to give priority to communicating with the EOF. Consequently, communications with other facilities were delayed or the facilities did not receive information. An example of a delay observed by the inspectors was the declaration of the SAE. A SAE was declared at 8:03 p.m., but TSC was not informed of the declaration until 8:20 p.m.

The high noise level in the TSC hampered communications. The inspectors noted that page system announcements in the TSC were bearly audible. Because of the high noise in the TSC, the announcements coming over the page system went unnoticed. The inspectors observed that after the SAE was declared at 8:03 p.m., a page announcement for the evacuation of all nonessential personnel was made at 8:12 p.m.. Personnel in the TSC did not receive the message and nonessential TSC personnel were not evacuated. At 9:05 p.m., 53 minutes later, the TSC manager recommended to the ED that all nonessential TSC personnel be evacuated.

Delays were experienced in fax transmissions. The licensee's fax machines transmit to an onsite computer. The computer then faxes the information to the appropriate locations. The increased number of faxes through the fax/computer interface caused a noticeable delay in receipt of information at the receiving locations.

The Simulator Control Room personnel could have been more descriptive in their communications with the TSC. The inspectors observed that the Simulator Control Room's communication lacked detail in specifics of the problems that they were having. As an example, the Simulator Control Room did not specify whether the Standby Liquid Control System problem was caused by a bad switch, power supply, or component failure.

Dose projections were performed from the EOF, once the EOF was activated. Notifications were released from the TSC, but the dose projection information was computed in the EOF and phoned to the TSC for entry on the notification form. This arrangement contributed to some confusion in notification. Two General Emergency notification forms had to be voided. One voided notification form, which contained the worst case Protective Action Recommendation (PAR), was incorrectly conveyed to the NRC by the TSC communicator (simulated).

The TSC turned the ED position over to the EOF while a notification form was being prepared. Consequently, the TSC notification form was voided. This delayed crucial data from being transmitted to the offsite agencies.

c. Conclusion

The inspectors concluded the licensee's internal communications were adequate but substantial communications improvements can be made.

P4.3 Communicating Dose Assessment To Offsite Agencies

a. Inspection Scope (82301)

This area was observed by the inspectors to determine whether the licensee could perform dose assessment.

b. Observations and Findings

The licensee's procedure defined significant core damage as a reading of 4.8 E+5 counts per minute on the Drywell Wide Range Monitors. Dose assessment personnel in the TSC misread the indication on the Dry Well Wide Range Monitors of 64,000 as 6.4 E+5 counts per minute and recommended a PAR based on significant core damage. The error was identified and corrected.

A drill control problem surfaced because the responders used Safety Parameter Display System (SPDS) No. 1, which was displaying actual plant data instead of SPDS No.2 which contained simulated data. Consequently, the wrong wind direction was entered on the notification form. This resulted in making PARs for the wrong offsite sectors.

The inspectors observed that the State of Georgia did not receive a complete set of radiological release information until approximately 1 hour after the declaration of the General Emergency.

There were no unplanned releases at the beginning of the exercise. As part of the emergency procedure, a dose calculation was performed when the initial declaration was made. EIP-18, Prompt Offsite Dose Assessment, Data Entry Sheet 2, instructed the user to use either Main Stack Narrow Range or Main Stack KAMAN data in computing offsite dose assessments. In the initial dose assessment calculation, the Simulator Control Room operator incorrectly used data from both the Main Stack Narrow Range and Main Stack KAMAN. The calculation estimated 49 mr/hr at the site boundary which was incorrect for the simulated plant conditions. Later in performing the dose assessment calculation for the SAE, the Simulator Control Room operator used the correct monitors.

c. Conclusion

The inspectors determined that the licensee maintained a computerized dose assessment system and the staff could perform dose assessment. Improvements can be made in drill control.

P4.4 Offsite Notifications During the Exercise

a. Inspection Scope (82301)

The inspectors observed licensee notifications to State, local, and Federal authorities for timeliness and message content.

b. Observations and Findings

EIP-073-03, Offsite Emergency Notifications, required that follow-up notifications to State and local authorities to be made periodically during an Alert or higher emergency classification, and that significant events which potentially impact offsite emergency actions be reported as soon as practicable.

The notification messages were not updated to accurately reflect significant changes in plant conditions and system status. Significant errors in the notifications were not corrected in a reasonable amount of time.

Message No. 2, which reported the declaration of a SAE, was transmitted from the Simulator Control Room. The message recommended to the State and local authorities that non-essential personnel be evacuated. The evacuation was appropriate for onsite personnel, but should not have been an offsite recommendation.

Message No.3, during the SAE, was transmitted from the Simulator Control Room and provided contradictory information to offsite authorities. The form identified that no release was occurring. However, Block 12 of the form contained release rate values for noble gases and iodine. Block 13 contained estimated dose values at the site boundary, 2, 5, and 10 miles. This message continued to incorrectly recommend offsite authorities evacuate non-essential personnel.

Message No.4, which reported the declaration of a General Emergency, indicated that an elevated release started at 8:30 p.m. The form indicated that the release magnitude information was not available. However, the notification form listed a dose projection time as 8:12 p.m. and listed dose values for the site boundary, 2, 5, and 10 miles.

Due to the drill control problem noted previously, the wind direction was incorrectly listed on Message No.4 as from 332 degrees instead of from 147 degrees, the stability class was incorrectly listed as G instead of D and the form contained protective action recommendations for the wrong offsite sectors. Although the licensee noted the error in meteorological data within a few minutes, the error was not corrected for 45 minutes - an excessive length of time.

c. <u>Conclusion</u>

The inspectors concluded that there were numerous examples of different deficiencies in the licensee's notifications. Licensee performance in this area was identified as an Exercise Weakness. This is identified as Inspector Followup Item 50-321,366/96-07-04: Failure to Make Adequate Notifications to State, Local, and Federal Authorities.

F2.1 Fire Main Underground Leaks

a. Inspection Scope (92904)

The inspectors conducted a review of fire header leaks. The review included observation of licensee work activities and documentation for repair of underground fire main water leaks located both inside and outside the protected area.

b. Observations and Findings

In addition to observing work activities involved in repairing several fire main leaks, the inspectors reviewed 18 Maintenance Work Orders (MWOs) for previous leak repairs. The review period was from June of 1994 to the present. Of the MWOs reviewed, work activities for 13 were completed. The five remaining MWOs were ready to be worked. Two MWOs were for leaks within the protected area. The inspector verified that the leaks were properly isolated. The completed work activities observed were performed under the following MWOs:

1-94-0012, Fire Main Leak Near Fire Equipment Bldg

1-94-0116, Fire Main Leak West Side of Bldg 10

1-94-5913, Fire Main Leak South Side of Security Bldg

1-95-2426, Leak East of Unit 1 Condensate Storage Tank

1-95-4353, Fire Main Leak North of Computer Bldg

The inspectors noted from the observations and documentation reviews that the failures were due to flange leaks and ruptured pipes. As a corrective action to help minimize the leaks, the licensee rerouted warehouse truck traffic to prevent traveling over the main fire header. The inspectors observed that some work delays occurred and most were due to a lack of replacement parts. The inspectors did not consider the work delays to be excessive. The inspectors' review of the overall high pressure fire system availability did not identify significant deficiencies for protection capability. In the cases where the leaks were identified as exceeding the jockey fire pump capability, the leaks were properly isolated to provide maximum protection capability. The inspectors verified that compensatory actions were implemented and the requirements of the Fire Protection Program (FPP) were met.

The inspectors discussed the problems with engineering and FPP personnel. Engineering indicated that piping age and possibly some vehicle traffic was the cause of the leaks. There were no extensive plans for piping replacement but rather to repair and replace leaking piping as necessary.

c. Conclusions

All observed work activities were performed in a controlled manner with engineering and supervisory oversight. The leak repairs appear to be appropriately prioritized to maintain a satisfactory fire protection system. The overall efforts to control underground fire main leaks were reasonable.

V. Management Meetings and Other Areas

X. Review of UFSAR Commitments

A recent discovery of a licensee operating its facility in a manner contrary to the Updated Final Safety Analysis Report (UFSAR) description highlighted the need for a special focused review that compares plant practices, procedures and/or parameters to the UFSAR description. While performing the inspections discussed in this report, the inspectors reviewed the applicable portions of the UFSAR that related to the areas inspected. The inspectors verified that the UFSAR wording was consistent with the observed plant practices, procedures, and /or parameters.

X.1 Exit Meeting Summary

The inspectors presented the inspection results to members of the licensee management at the conclusion of the inspection on July 8, 1996. An interim exit was conducted on May 23, 1996. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

X.2 Other NRC Personnel On Site

On May 20-21, 1996, Mr. P. H. Skinner, Chief Reactor Projects Branch 2, visited the site. Mr. Skinner met with the resident staff and discussed plant status and generic issues. He toured the plant, attended licensee management plant status meetings, and reviewed miscellaneous plant documents.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

Anderson, J., Unit Superintendent
Betsill, J., Operations Manager
Coggin, C., Engineering Support Manager
Curtis, S., Operations Support Superintendent
Davis, D., Plant Administration Manager
Fornel, P., Performance Team Manager

Fraser, O., Safety Audit and Engineering Review Supervisor
Hammonds, J., Regulatory Compliance Supervisor
Kirkley, W., Health Physics and Chemistry Manager
Lewis, J., Training and Emergency Preparedness Manager
Moore, C., Assistant General Manager - Plant Support
Roberts, P., Outages and Planning Manager
Sumner, H., General Manager - Nuclear Plant
Thompson, J., Nuclear Security Manager
Tipps, S., Nuclear Safety and Compliance Manager
Wells, P., Assistant General Manager - Operations

INSPECTION PROCEDURES USED

IP 61726: Surveillance Observations
IP 62703: Maintenance Observations
IP 71707: Plant Operations
IP 71750: Plant Support Activities
IP 82301: Operational Status Of The Emergency Preparedness Program
IP 92700: Onsite Follow-up of Written Reports of Nonroutine Events at Power Reactor Facilities
IP 92709: Licensee Strike Contingency Plan
IP 92901: Followup - Operations
IP 92902: Followup - Maintenance/Surveillance
IP 92903: Followup - Followup Engineering
IP 92904: Followup - Plant Support
IP 93702: Prompt Onsite Response to Events at Operating Power Reactors

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-321,366/96-07-01	URI	Determine Safety Significance and Testing Requirements for Unit 1 and Unit 2 Containment Isolation Status Panel (Paragraph M3.1).
50-321,366/96-07-02	VIO	Failure to Conduct Testing Following Molded Case Circuit Breaker Instantaneous Trip Setpoint Changes (Paragraph E2.1).
50-366/96-07-03	IFI	Degradation and Replacement of Unit 2 Station Battery 2B Due to Buildup of Cell Sediment (Paragraph E2.2).

50-321,366/96-07-04	IFI	Exercise Weakness - Failure to Make Adequate Notifications to State, Local, and Federal Authorities. (Paragraph P4.4).

<u>Closed</u>

50-366/95-08	LER	Reactor Vessel Inventory Loss Results in an Unplanned Engineering Safety Feature System Actuation (Paragraph E8.1).
50-366/95-10	LER	Component Failure Results in Engineering Safety Feature System Actuation (Paragraph M8.1).