UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W., SUITE 2900
ATLANTA, GEORGIA 30323-0199

Report No.: 50-400/94-12

Licensee: Carolina Power and Light Company
P. O. Box 1551
Raleigh, NC 27602

Docket No.: 50-400 Licensee No.: NPF-63

Facility Name: Harris 1

Inspection Conducted: May 7 - June 3, 1994

Inspectors: D. Roberts for 6/16/94
J. Tedrow, Senior Resident Inspector Date Signed

Other Inspectors: D. Roberts, Resident Inspector
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Approved by: H. Christensen 6/17/94
H. Christensen, Section Chief Date Signed
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SUMMARY

Scope:

This routine inspection was conducted by two resident inspectors in the areas of plant operations, plant startup from refueling, outage activities, maintenance observation, surveillance observation, review of nonconformance reports, design changes and modifications, plant housekeeping, radiological controls, security, fire protection, emergency preparedness, 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:

Two violations were identified: Failure to properly review and approve a change to modification acceptance testing, paragraph 4.b; Failure to promptly correct a deficiency with an emergency plan implementing procedure, paragraph 5.e.

The licensee made excellent progress reducing the number of temporary modifications. In addition, an auxiliary operator demonstrated excellent attention to detail by identifying a deficiency with a main steam isolation valve and thereby avoiding a potential plant transient, paragraph 2.a(1).

The addition of detailed checklists for power ascension flux map reviews and the requirement of having the Plant Nuclear Safety Committee review any noted anomalies was considered to be a strength, paragraph 2.b.

The documentation and routing of post maintenance testing requirements was cumbersome, paragraph 3.a. The performance of a rod drop timing test was deficient, paragraph 3.c(2).

The turnover review performed for a plant modification was weak, paragraph 4.b.

night order book; equipment inoperable record; active clearance log; grounding device log; temporary modification log; chemistry daily reports; shift turnover checklist; and selected radwaste logs. In addition, the inspector independently verified clearance order tagouts.

The inspectors found the logs to be readable, well organized, and provided sufficient information on plant status and events. Clearance tagouts were found to be properly implemented. The inspector noted that the number of temporary modifications had been reduced from approximately 40 to 3. The licensee made excellent progress during the refueling outage in reducing the number of temporary modifications.

On May 26 an auxiliary operator noticed an air leak from a solenoid fitting on Main Steam Isolation Valve (MSIV) 1MS-84. A work request was generated and the leak was repaired. The operator's attention to detail during the performance of his rounds was exemplary considering the location of the leak and the general noise level near the equipment. The identification and repair of the air leak avoided a potential inadvertent closure of the MSIV. This type of scenario on the plant simulator typically produced a safety injection signal on low steam line pressure. Performance of operator rounds was considered strong in this case.

No violations or deviations were identified.

(2) Facility Tours and Observations

Throughout the inspection period, facility tours were conducted to observe operations, surveillance, and maintenance activities in progress. Some of these observations were conducted during backshifts. Also, during this inspection period, licensee meetings were attended by the inspectors to observe planning and management activities. The facility tours and observations encompassed the following areas: security perimeter fence; control room; emergency diesel generator building; Reactor Auxiliary Building (RAB); waste processing building; turbine building; fuel handling building; emergency service water building; battery rooms; electrical switchgear rooms; and the technical support center.

During these tours, observations were made on monitoring instrumentation which included equipment operating status, area atmospheric and liquid radiation monitors, electrical system lineup, reactor operating parameters, and auxiliary equipment operating parameters. Indicated parameters were verified to be in accordance with the TS for the current operational mode. The inspectors also 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. The licensee's performance in these areas was satisfactory.

No violations or deviations were identified.

b. Plant Startup From Refueling (71711)

The inspector observed unit startup activities following refueling to verify that plant systems were properly returned to service and that the startup was conducted in accordance with approved procedures and in compliance with the TS.

The plant began diluting to criticality on May 10 and entered the startup condition (Mode 2) at 1:51 p.m. Criticality was achieved at 7:08 p.m. on May 10. The inspectors observed the approach to criticality and verified that criticality occurred within the limits calculated by licensee personnel. Power ascension testing and activities were also observed. Implementation of portions of the following procedures was verified and/or data reviewed:

- GP-004 Reactor Startup
- GP-005 Power Operation
- EPT-069 Initial Criticality
- PLP-626 Power Ascension Testing Program After a Refueling Outage
- EPT-093 Turbine First Stage Pressure Data

The inspectors reviewed the 30 percent power flux map performed in accordance with procedure FMP-101, Incore Flux Mapping. A detailed review checklist was recently added to procedure PLP-626 to ensure an adequate review of the flux maps were performed. This checklist was developed from lessons learned from the licensee's other facility. The checklist required comparison of different core regions and individual fuel assemblies for differences between predicted relative power estimates and those actually measured by the incore detectors. If the checklist criteria was not satisfied, then additional evaluation by engineering personnel was required followed by a Plant Nuclear Safety Committee (PNSC) review prior to power escalation.

The 30 percent flux map indicated that seven assemblies had slightly excessive differences between predicted and actual

relative power. Analysis determined that the differences were from an expected actual radial flux tilt along with modeling inaccuracies for individual assemblies. No fuel anomalies were identified and the PNSC authorized a power increase to 50 percent. Subsequent flux maps at higher power levels produced satisfactory results as well. The inspector considered the implementation of this comprehensive checklist and review of flux map anomalies to be a strength of the power ascension testing program.

No violations or deviations were identified.

c. Outage Activities (71707)

Major activities performed during this scheduled refueling outage included replacement of AFW piping inside of containment, 100 percent Eddy Current testing of the steam generator U-tubes, elimination of the Resistance Temperature Detector (RTD) bypass manifold, replacement of 480 VAC LK-16 breakers on the "A" busses, rebuild of an MSIV, Motor Operated Valve (MOV) testing, emergency diesel generator maintenance, and reduction of hot leg Reactor Coolant System (RCS) temperature. Many of these activities were addressed in NRC Inspection Report 50-400/94-10.

Licensee personnel replaced the remaining sections of AFW carbon steel piping which were not replaced during the previous refueling outage in October 1992 (reference LER 92-14) with chrome-moly piping. This was done because erosion/corrosion measurements indicated pipe wall thinning was occurring. The new material is expected to be more resistant to erosion. Presently, the only AFW carbon steel piping located inside of containment is that which was replaced in kind during the October 1992 outage and has experienced only one cycle of operation. Licensee personnel obtained as-installed wall thicknesses for this piping which will enable them to establish an accurate erosion rate for predicting a suitable period for future pipe replacement. The licensee's actions were considered to be appropriate.

No violations or deviations were identified.

3. Maintenance

a. Maintenance Observation (62703)

The inspector observed/reviewed maintenance activities to verify that correct equipment clearances were in effect; work requests and fire prevention work permits were issued and TS requirements were being followed. Maintenance was observed and work packages were reviewed for the following maintenance activities:

- Adjust nuclear instrumentation overpower reactor trip setpoints to 25 percent and 50 percent to support power ascension in accordance with procedure MST-I0070,

Calibration of NIS Power Range Overpower Trip High Range Bistables.

- Replace rotating element in "A" Charging/Safety Injection Pump (CSIP) in accordance with procedure CM-M0019 Pacific Charging/Safety Injection Pump Size Two and One-Half Inches RL Type IJ Disassembly and Maintenance, and inspect per procedure CM-M0021, Westinghouse High-Speed Gear Drives Type SU-10 For Charging/Safety Injection Pump, Disassembly, Maintenance and Assembly.

- Repair boric acid leak in letdown flow control valve 1CS-2.

- Replace ramp generator on TDAFW pump control circuit in accordance with procedure PIC-I0825, Terry Turbine Governor Control System Calibration (Auxiliary Feedwater Pump).

- Repair steam leaks from steam generator secondary manways in accordance with procedure CM-M0008, Removal and Replacement of the Steam Generator Secondary Manway Covers.

- Repair bonnet leak on check valve 1AF-68 in accordance with procedure CM-M0069, Anchor-Darling Check Valves Disassembly and Maintenance.

- Replace torque switch for motor operated valve 1AF-55 in accordance with procedure PM-I0043, Motor Operated Valve Testing and Calibration.

- Troubleshoot rod control system for control rod F-4 which dropped during low power physics testing.

In general, the performance of work was satisfactory with proper documentation of removed components and independent verification of the reinstallation. The inspector reviewed post maintenance testing documentation for the above work. The licensee specifies three types of testing from three different organizations. The maintenance planning organization determines appropriate functional verification testing such as valve stroking, visual inspections, motor current checks etc. The Inservice Inspection (ISI) group determines any ASME Section XI required testing and the operations group identifies appropriate surveillance tests to declare TS components operable. The inspector determined that the appropriate testing was performed. However, the licensee's documentation and routing of post-modification test requirements through three organizations is cumbersome.

During MOV testing on 1AF-55, the torque switch failed and the valve was driven into the seat by the motor operator. The inspector reviewed the work package to repair the torque switch and a generic engineering evaluation which addressed any valve components which had exceeded thrust limits. Licensee personnel

performed calculation HNP-C/EQ-1227 and identified that the valve yoke clamp component was overstressed due to the torque switch failure. This component was subsequently inspected with satisfactory results.

In conjunction with the MOV testing program, the new torque switch setting established for this valve exceeded previous maximum thrust values. New short term thrust limits were calculated for this valve and several other AFW system valves (1AF-74, 1AF-93, 1AF-137, 1AF-143, and 1AF-149) by engineering personnel via PCR-7284, Extended Thrust Ratings for AFW MOVs. The short term allowable thrust will only be allowed during the next cycle of operation and will be removed after the next refueling outage during which the valves will be upgraded/qualified for the higher thrust.

Inspector Followup Item (400/94-12-01): Review the licensee's activities to upgrade/qualify AFW MOVs for higher thrust values.

No violations or deviations were identified.

b. Surveillance Observation (61726)

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 was utilized; and TS requirements were followed. The following tests were observed and/or data reviewed:

- OST-1004 Power Range Heat Balance Daily Interval
- OST-1039 Calculation of Quadrant Power Tilt Ratio
- OST-1075 Turbine Mechanical Overspeed Trip Test 18 month Interval
- OST-1508 ISI Operability Test for 1CS-167, 1CS-294, 1CS-775, 1CS-776 Quarterly Interval
- MST-I0635 Self Heating Test of Installed Resistance Thermometers
- MST-I0636 In-Situ Response Time Testing of Installed RTDs using AMS Model ERT-1
- EST-709 Reactor Coolant System Flow Determination by Calorimetric
- EST-710 Hot Channel Factor Tests
- EST-724 Shutdown and Control Rod Drop Test Using Computer

In general, the performance of these procedures was found to be satisfactory with proper use of calibrated test equipment, necessary communications established, notification/authorization of control room personnel, and knowledgeable personnel having performed the tasks.

c. Review of Nonconformance Reports (71707)

Adverse Condition Feedback Reports (ACFR) were reviewed to verify the following: TS were complied with, corrective actions and generic items were identified and items were reported as required by 10 CFR 50.73.

(1) ACFR 94-1819 reported that regular work was being scheduled during the same time interval as work listed on the outage schedule without the same risk reviews performed in conjunction with the generation of the outage schedule. The inspector discussed this item with outage personnel and work planning personnel and was informed that the transition between outage work control and on-line work control was being evaluated to address a specific time for relinquishing control and for evaluating the risk involved with any planned work.

Inspector Followup Item (400/94-12-02): Review the licensee's transition of work control following startup from an outage.

(2) ACFR 94-1925 reported a discrepancy associated with control rod drop time testing required by TS 4.1.3.4. Specifically, the TS required that individual shutdown and control rod drop times from the fully withdrawn position shall be less than or equal to 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry. The TS further stated that the rod drop time of shutdown and control rods shall be demonstrated through measurement prior to reactor criticality for all rods following each removal of the reactor vessel head and at least once per 18 months.

On May 10, 1994, during startup from the refueling outage, licensee personnel performed procedure EST-724 which was intended to satisfy the surveillance requirement of TS 4.1.3.4. The acceptance criteria stated in the test procedure was consistent with the requirements of the TS. The test was setup to record voltage signals associated with the opening of the reactor trip breakers and control rod travel to the bottom of the core. The voltage signals, produced by an auxiliary contact on the reactor trip breaker and by rod position indication coils, were sampled by a data acquisition system and analyzed to determine the drop time of each individual Rod Cluster Control Assembly (RCCA).

Prior to reactor criticality on May 10, 1994, licensee personnel performing the test reviewed drop time data for the rods, which ranged from 1.424 seconds for rod F-8 in control bank C, to 1.713 seconds for rod B-6 in control bank A. The test was signed off as satisfactory at 3:00 a.m. on May 10 and reactor criticality was achieved at 7:08 p.m. that evening.

On May 20, licensee personnel reviewed plots of output voltage versus time for each rod and discovered a discrepancy with the May 10 test. It was discovered that the signal corresponding to the opening of the reactor trip breakers (beginning of decay of stationary gripper coil voltage) was unretrievable in the plots provided for all 52 of the RCCAs. Essentially, the data recorded in procedure EST-724 to satisfy the acceptance criteria had been determined by measurements of rod drop time from beginning of rod motion to dash pot entry. Licensee personnel immediately performed an operability determination for the RCCAs and determined them to be operable. This determination was based on several factors including an evaluation of previous data to determine the amount of time that elapsed between beginning of decay of stationary gripper coil voltage to beginning of rod motion. A review of historical data indicated that the longest recorded time for gripper coil voltage decay was 0.063 seconds and relatively constant for the various rods. Licensee personnel decided to include a constant value of 0.15 seconds to the rod drop times for conservatism to account for the unmeasured coil decay time. The 0.15 second value was used based upon evaluations of reactor trip response times by the nuclear steam supply system vendor. This practice was considered acceptable by licensee personnel since the parameter in question, gripper coil voltage decay, would not have changed over the previous cycle unless maintenance had been performed on inductive parts of the rod control system.

Licensee personnel trended all previous hot rod drop testing with the May 10 test results and found no deviations. The vendor who measured the rod drop times provided information to validate the test results by analyzing data taken for another plant. This analysis was performed by comparing data using the start of stationary gripper voltage decay with data obtained using the data acquisition system which looked at just the start of rod motion. This comparison showed less than one percent deviation between the two methods. The results tended to support the licensee's overall conclusion that the loss of reactor trip breaker opening time was not significant to the overall rod drop times.

The inspectors discussed this issue with NRC headquarter's personnel who concurred in the inspectors conclusion that a safety issue did not exist and that the licensee's actions were logical and acceptable. The licensee's initial actions regarding the conduct and review of the May 10 test were considered to be weak.

No violations or deviations were identified.

d. Review of Special Reports (90713)

The licensee issued a special report dated April 26, 1994, regarding the results of the recent inservice inspection conducted on the three steam generators during RFO-5. The licensee completed eddy current testing on 100 percent of the tubes in the three steam generators. The report, required by TS 4.4.5.5.a, documented the number of new tubes plugged during the outage as 0, 3 and 0 for the "A", "B", and "C" steam generators, respectively. The total number of tubes plugged for the "A", "B", and "C" steam generators was 15, 7, and 13, respectively. The number of tubes plugged in each steam generator is small when compared to other plants of similar age. The plant has an operating and chemistry profile similar to that of other facilities.

The licensee continues to look at ways to extend the operating life of the steam generators. During this refueling outage an RCS hot-leg reduction modification was implemented specifically for that purpose.

e. Licensee Action on Previously Identified Inspection Findings (92902)

(1) (Closed) Inspector Followup Item 400/93-07-02: Follow the licensee's actions to develop a leak test for check valve 1CS-167.

The inspector reviewed procedure OST-1508, ISI Operability Test for 1CS-167, 1CS-294, 1CS-776 Quarterly Interval. This procedure was recently changed to incorporate new seat leakage and backseat testing requirements for check valve 1CS-167. Attachment 2 to the procedure contained a leakrate acceptance criteria (specified in percent change of volume control tank level) of 0.018 percent per minute. This corresponded to a leak rate of approximately 0.25 gallons per minute which supported the assumptions contained in PCR-6351, Design Basis Evaluation of Alternate Miniflow Return Line and Valve 1CS-167. Additionally, the inspector reviewed data obtained during the most recent performance of procedure OST-1508 on April 24, 1994, and verified that the data was within the specified acceptance criteria.

(2) (Closed) Inspector Followup Item 400/93-08-04: Follow the licensee's activities to retest the "B" CSIP, determine the root cause of its failure, and assess any generic implications.

The inspector reviewed the data from test procedure EPT-404, Charging Safety Injection Pumps Runout Testing and Pump Curve Verification Test. The test was performed on April 23 and 24, 1994 to determine whether or not the new "A" and "B" CSIPs met current TS limits on pump runout flow. The "A" pump was tested because the rotating element was replaced in April 1994 following damage received during a pump teardown inspection (reference NRC Inspection Report 50-400/94-10). The test data for the "A" and "B" pumps (684.4 and 684.2 gpm, respectively) barely met the pump runout limit of 685 gpm. The licensee performed an engineering evaluation (PCR 7274, CSIP Runout Technical Specification Compliance) to evaluate instrument inaccuracies and the effects of increased diverted flow from the safety injection lines. The preoperational tests performed on the "A" and "B" pumps prior to initial plant startup used a diverted flow of 60 gpm through the normal charging flowpath. Due to plant conditions and TS limits on RCS controlled leakage, diverted flow for the April 23 and 24 tests was limited to 31 gpm. The engineering evaluation demonstrated that had the "A" and "B" CSIPs been tested using an amount of diverted flow consistent with that used during preoperational testing, the two pumps would have met the acceptance criteria by a greater margin.

The inspectors reviewed a licensee memorandum dated October 28, 1993, which discussed planned actions resulting from a root cause analysis of the "B" pump's 1993 shaft failure. The analysis, performed by the plant's nuclear steam supply system vendor, discussed various possible contributors to the shaft failure without identifying a sole root cause. These included possible pump and driver misalignment, improper operation (frequent pump starts), material defects and installation defects. Potential gas entrainment in the fluid passing through the pump was also discussed. During the recently completed refueling outage for the "A" pump, licensee personnel performed pump and driver alignments, non-destructive examinations on locknut threads, and inspections for evidence of cavitation due to gas entrainment. The inspection produced no evidence of gas binding for the "A" CSIP.

4. Engineering

Design Changes and Modifications (37828)

Plant Change Requests (PCR) involving the 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 approved acceptance criteria or deviations were resolved in an acceptable manner, and that appropriate drawings and facility procedures were revised as necessary. In addition, PCR's documenting engineering evaluations were also reviewed. The following modifications and/or testing in progress was observed:

- PCR-5696 EDG 1A-SA and 1B-SB Maintenance Mode Circuitry
- PCR-6105 EDG Start Time Recorder
- PCR-6502 Startup Feedwater Source
- PCR-6959 Addition of Seismic Supports on Blowdown System
- PCR-7274 CSIP Runout Technical Specification Compliance
- PCR-7306 Evaluation of Rod Drop Time Response Data for RFO-5

In general the licensee's control and implementation of these modifications was satisfactory.

a. The startup feedwater source modification involved providing automatic opening signals to the AFW flow control valves so that they could be throttled or closed during startup operations when low feedwater rates are demanded (reference NRC Inspection Report 50-400/91-13, violation 400/91-13-01). The valves were provided with opening signals from safety injection, loss of offsite power, AMSAC, low-low steam generator level, and loss of the main feedwater pump actuation circuits. The modification also installed manual isolation valves on the discharge of the motor driven AFW pumps to facilitate testing. Vendor testing of the pumps was also performed to verify that pump operation at reduced flowrates did not degrade pump performance. In conjunction with the reduced feedwater flowrates used during startup, minimum flow requirements established to prevent AFW check valve degradation were replaced with requirements to perform visual check valve inspections during refueling outages.

   Implementation work packages for the modification were reviewed as well as post-modification testing. Installation of the manual isolation valves was observed as was the subsequent system hydrostatic test.

b. During this inspection period the inspector reviewed the post modification testing documentation associated with PCR-6105. In the testing requirements section of the modification package, functional tests and calibration checks were specified to be performed for the affected generator control cabinet transducers and output loops. The inspector verified that this testing had been accomplished for the "A" EDG. The inspector verified that the calibration checks for the "B" EDG frequency, voltage, and watts transducers had not been performed. The work request which was generated to perform this testing had been changed to delete the required testing. The modification had been turned over to Operations' control on April 22, 1994.

The inspector questioned licensee personnel about this matter. The system engineer monitored the performance of the acceptance testing. The results of the acceptance testing for the "A" EDG control circuitry indicated that the modification had little affect on the circuitry and the system engineer therefore believed that only a functional test was needed to verify proper operation. Based upon this conclusion, the system engineer deleted the loop calibration procedures from the work request.

Procedure MOD-204, Modification Implementation, section 5.7.7 provides guidance on making revisions to modification acceptance tests and requires that appropriate reviews be conducted by engineering and affected organizations. Furthermore, section 5.8 contains directions for performing turnover of the completed modification to Operations' control. As part of this turnover review, the modification testing listed must be verified to be satisfactorily completed. The turnover review failed to identify the acceptance testing discrepancy because only the completed status of the work request was verified and not the details of what testing had actually been performed.

The change to the acceptance testing as listed on the work request without a concurrent revision to the acceptance testing listed in the modification package and accompanying reviews by engineering and affected organizations, was contrary to the requirements of procedure MOD-204 and is considered to be a violation of TS 6.8.1.a.

Violation (400/94-12-03): Failure to properly review and approve a change to modification acceptance testing.

The failure of licensee personnel to identify this testing discrepancy prior to turnover of the modification to Operations' control indicates a weakness in this process.

The inspector discussed the technical aspects of the acceptance testing with the system engineer and considered that even though the change to the testing was improper, the functional testing

actually performed was sufficient to verify proper system operation.

5. Plant Support

a. Plant Housekeeping Conditions (71707) - 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.

b. Radiological Protection Program (71707) - Radiation protection control activities were observed routinely to verify that these activities were in conformance with the facility policies and procedures, and in compliance with regulatory requirements. The inspectors also reviewed selected radiation work permits to verify that controls were adequate.

During a walkdown of the RAB on May 25, 1994, the inspectors identified examples of inadequate radiological postings in the "A" and "B" CSIP rooms. Specifically, the "A" and "B" CSIPs were posted as High Contamination Areas (HCA) requiring only gloves as a minimum for entry. The inspector noted that this posting of minimum dress was not in accordance with the licensee's program procedure PLP-511, Radiation Control and Protection Program, section 5.7.4.4 which specified full protective clothing for entry into HCAs. The posted boundaries extended to the baseplates for each pump and encompassed the pumps and speed increasers. In the case of the "B" pump, the boundary extended beyond the speed increaser and to the motor coupling. The inspector verified that the contamination levels around the pumps did warrant the HCA postings. Although worker activities in the radiologically controlled area are dictated by Radiation Work Permits (RWPs), some RWPs which allow access to HCAs instructed workers to dress as posted for the area. No personnel contaminations resulted from the posting deficiency.

In NRC Inspection Report 50-400/94-10 the licensee was issued a Non-Cited Violation (NCV 400/94-10-03) for a similar posting deficiency. In response to this, ACFR 94-1962 was generated on May 23 to revise the program procedure to provide general guidance and remove the specific minimum dress requirements. An advance change to procedure PLP-511 was subsequently initiated and approved on June 1, 1994. Since licensee personnel were in the process of implementing corrective actions for the NCV to revise program requirements, the example mentioned above will not be cited.

c. Security Control (71707) - The performance of various shifts of the security force was observed in the conduct of daily activities which included: protected and vital area access controls; searching of personnel, packages, and vehicles; badge issuance and retrieval; escorting of visitors; patrols; and compensatory posts.

In addition, the inspector observed the operational status of closed circuit television monitors, the intrusion detection system in the central and secondary alarm stations, protected area lighting, protected and vital area barrier integrity, and the security organization interface with operations and maintenance.

d. Fire Protection (71707) - 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.

e. Emergency Preparedness (71707) - The inspectors reviewed ACFR 94-348 which documented a deficiency in shift manning levels in support of the licensee's emergency plan. Specifically, the ACFR stated that procedure PEP-102, Site Emergency Coordinator - Control Room, listed replacements for the nuclear shift supervisor when acting as the Site Emergency Coordinator (SEC). Replacement personnel listed included the roving Senior Control Operator (SCO), Shift Supervisor Designee (SSD), and the control room SCO. The ACFR stated that the roving SCO position and SSD positions were not always manned and that the control room SCO did not receive training on the performance of a dual role as SEC and his usual position as operations leader. This deficiency was identified by licensee personnel on January 22, 1994, during a review of the emergency procedures. On March 29 the ACFR was canceled based upon a licensee memorandum which stated that shift staffing levels were sufficient to implement the emergency plan and procedures. Therefore licensee compliance personnel did not believe a deficient condition existed and no corrective action was taken.

On May 24, 1994, the inspector discussed shift manning with licensee personnel and learned that the roving SCO position had not been manned for the last two years and wasn't planned to be staffed in the future. Procedure PEP-102 had not been changed to reflect this. Failure to take prompt corrective action and fix this identified deficiency is contrary to the requirements of 10 CFR Part 50, Appendix B, Criterion XVI and is considered to be a violation.

Violation (400/94-12-04): Failure to promptly correct a deficiency with an emergency plan implementing procedure.

The inspectors found plant housekeeping and material condition of components to be satisfactory. With the exceptions noted above, the licensee's adherence to radiological controls, security controls, fire protection requirements, emergency preparedness requirements and TS requirements in these areas was satisfactory.

f. Licensee Action on Previously Identified Inspection Findings (92904)

(Closed) Violation 400/94-06-01: Failure to properly control vehicles inside the protected area.

The inspector reviewed and verified completion of the corrective actions listed in the licensee's response letter, dated May 11, 1994. Many of the corrective actions were completed prior to the March 5 incident which had been attributed to personnel error. The only actions taken since March 5 were to discipline the responsible individual and to continue making announcements on the plant television information network and in the site news publication.

6. Exit Interview (30703)

The inspectors met with licensee representatives (denoted in paragraph 1) at the conclusion of the inspection on June 3, 1994. During this meeting, the inspectors summarized the scope and findings of the inspection as they are detailed in this report, with particular emphasis on the violations addressed below. 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. No dissenting comments from the licensee were received.

Item Number	Description and Reference
400/94-12-01	Inspector Followup Item: Review the licensee's activities to upgrade/qualify AFW MOVs for higher thrust values, paragraph 3.a.
400/94-12-02	Inspector Followup Item: Review the licensee's transition of work control following startup from an outage, paragraph 3.c(1).
400/94-12-03	Violation: Failure to properly review and approve a change to modification acceptance testing; paragraph 4.b.
400/94-12-04	Violation: Failure to promptly correct a deficiency with an emergency plan implementing procedure, paragraph 5.e.

7. Acronyms and Initialisms

ACFR - Adverse Condition Feedback Report
AFW - Auxiliary Feedwater
AMSAC - ATWAS Mitigation System Actuation Circuitry
ASME - American Society of Mechanical Engineers
ATWS - Anticipated Transient Without Scram

CFR	-	Code of Federal Regulations
CSIP	-	Charging Safety Injection Pump
CVCS	-	Chemical and Volume Control System
EDG	-	Emergency Diesel Generator
GPM	-	Gallons per Minute
HCA	-	High Contamination Area
ISI	-	Inservice Inspection
LER	-	Licensee Event Report
MOV	-	Motor Operated Valve
MSIV	-	Main Steam Isolation Valve
NCV	-	Non-Cited Violation
NRC	-	Nuclear Regulatory Commission
PCR	-	Plant Change Request
PNSC	-	Plant Nuclear Safety Committee
RAB	-	Reactor Auxiliary Building
RCCA	-	Rod Cluster Control Assembly
RCS	-	Reactor Coolant System
RTD	-	Resistance Temperature Detector
RWP	-	Radiation Work Permit
SCO	-	Senior Control Operator
SSD	-	Shift Supervisor Designee
TDAFW	-	Turbine Driven Auxiliary Feedwater
TS	-	Technical Specification
VAC	-	Volts Alternating Current