U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket No:	50-400
License No:	NPF-63
Report No:	50-400/98-07
Licensee:	Carolina Power & Light (CP&L)
Facility:	Shearon Harris Nuclear Power Plant, Unit 1
Location:	5413 Shearon Harris Road New Hill, NC 27562
Dates:	July 5 - August 15, 1998
Inspectors:	J. Brady, Senior Resident Inspector R. Hagar, Resident Inspector in training F. Jape, Senior Project Engineer (Sections O8.1, O8.2, M1.1, M2.1, E8.1, E8.2) J. Coley, Reactor Inspector (Sections M7.1, M8.1, M8.2, M8.3)
Approved by:	J. Zeiler, Acting Chief, Projects Branch 4 Division of Reactor Projects

Enclosure

EXECUTIVE SUMMARY

Shearon Harris Nuclear Power Plant, Unit 1
NRC Inspection Report 50-400/98-07

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 announced inspections by two regional inspectors.

Operations

- Operations performance during the period was in accordance with required procedures (Section O1.1).
- While completing an emergency diesel generator semiannual operability test, non-licensed operator performance was effective and consistent with both procedural requirements and management expectations (Section O4.1).
- Self-assessment activities were identifying issues for improvement (Section O7.1).

Maintenance

- Maintenance activities were performed by well qualified and experienced personnel. The work was satisfactorily completed using approved procedures. Maintenance personnel were very knowledgeable and skillful and performed their work in a professional manner (Section M1.1).
- Surveillance testing was adequately conducted. Operations and maintenance personnel exhibited knowledge of their assigned tasks, were professional, and accomplished the activities cautiously and with plant safety in mind (Section M2.1).
- Maintenance on the containment spray check valves has been conducted in accordance with applicable procedures, and has identified no adverse conditions associated with those valves (Section M3.1).
- The licensee's Maintenance Rule periodic assessment met the requirements delineated in the licensee's Maintenance Rule program procedure and the Maintenance Rule. Corrective actions sampled were appropriately implemented (Section M7.1).

Engineering

- Several cancelled Engineering Service Requests that were reviewed were rejected for valid technical and/or administrative reasons. No significant issues were dropped when these Engineering Service Requests were rejected (Section E1.1).

Plant Support

- The control of contamination and dose for the site was good and was attributable to good teamwork between the various departments (Section R1.1).
- The performance of Security and Safeguards activities was in accordance with applicable procedures (Section S1.1).
- Fire Protection activities were being adequately conducted in accordance with procedures (Section F1.1).

Report Details

Summary of Plant Status

Unit 1 began this inspection period at 100 percent power, and remained at 100 percent power throughout the period.

I. Operations

01 Conduct of Operations

01.1 General Comments

a. Inspection Scope (71707)

The inspectors conducted frequent reviews of ongoing plant operations including control room tours, shift turnovers, and observation of operations surveillance activities.

b. Observations and Findings

In general, the conduct of operations was professional and safety-conscious. Routine activities were adequately performed in accordance with procedures. Operations shift crews were appropriately sensitive to plant equipment conditions and maintained a questioning attitude in relation to unexpected equipment responses. Alarm response procedures were appropriately consulted when alarms existed.

c. Conclusions

Operations performance during the period was in accordance with required procedures.

02 Operational Status of Facilities and Equipment

02.1 General Comments (71707)

The inspectors conducted frequent tours of the facility to verify equipment condition, housekeeping, and proper use of clearances. The inspectors found equipment condition and housekeeping acceptable and that clearances were being used in accordance with procedures.

02.2 Engineered Safety Feature (ESF) System Walkdowns (71707)

The inspectors walked down accessible portions of the following ESF systems:

- Diesel Generator Building Ventilation
- Reactor Auxiliary Building Switchgear Room Ventilation

- Reactor Auxiliary Building Electrical Equipment Protection Room Ventilation

Equipment operability, material condition, and housekeeping were acceptable in all cases. No discrepancies were noted.

03 Operations Procedures and Documentation

03.1 General Comments (71707)

The inspectors conducted frequent reviews of operations logs and procedure usage, and found that procedures were appropriately followed.

04 Operator Knowledge and Performance

04.1 General Comments

a. Inspection Scope (71707)

The inspectors observed non-licensed operator performance during the performance of procedure OST-1085, 1A-SA Diesel Generator Operability Test Semiannual Interval Modes 1-6, on July 29, 1998.

b. Observations and Findings

The operators participated in the pre-job briefing, and used effective communications. The operators completed all procedure steps in turn, and used a placekeeping method that was both effective and consistent with management expectations. The operators were knowledgeable, and satisfactorily answered questions posed by the inspectors. The test was completed satisfactorily.

c. Conclusions

While completing a semiannual emergency diesel generator operability test, non-licensed operator performance was effective and consistent with both procedural requirements and management expectations.

06 Operations Organization and Administration

06.1 General Comments (71707)

The inspectors frequently observed control room staffing, and found that it consistently met Technical Specification (TS) requirements.

07 Quality Assurance in Operations

07.1 General Comments

a. Inspection Scope (40500, 71707)

During the inspection period, the inspectors reviewed multiple licensee quality assurance activities, including:

- Condition Reports;
- Nuclear Assessment Section Audits on HNP Document Control (HNAS 98-075), Spent Fuel Shipping and Special Nuclear Material Accountability Program (HNAS 98-116), Biennial Procedure Review (HNAS 98-121), and Harris E&E Center Laboratories (RNAS 98-060); and
- Plant Nuclear Safety Committee (PNSC) meeting on July 22, 1998.

b. Observations and Findings

The inspectors' review of Condition Reports, Nuclear Assessment Section Audits, and PNSC meeting, indicated that the site was appropriately focussed on the identification and correction of problems.

The document control audit (HNAS 98-075) identified several cases where follow-up on corrective actions taken as a result of these self assessments had not been timely, comprehensive, or well documented. While reviewing this assessment, the inspectors observed that for one of the identified issues, the assessment report did not itself describe comprehensive corrective actions. The identified issue was that a program was not in compliance with a procedure, and for this issue, the assessment described "developing a plan to bring the program into compliance" as one corrective action. However, the assessment did not describe implementation of that plan and assessment of its effectiveness as additional corrective actions. Another corrective action described for the same issue was to "develop performance indicators" to monitor the program's health, but the assessment did not describe any steps to ensure that management will review and properly interpret those indicators, and subsequently initiate and implement appropriate action to improve program performance. The inspectors considered those additional actions as essential for truly bringing the program into compliance and maintaining it there. The lead assessor-in-training for this assessment acknowledged these comments, and responded that:

- The phrase "developing a plan to bring the program into compliance" was meant to include more than just developing a plan; it was meant to include all activities that would be required to bring the program into compliance and maintain it there.

- The Nuclear Assessment Section will monitor the development and implementation of the performance indicators, to ensure that the indicators are valid, and that management properly interprets them and subsequently initiates and implements appropriate actions to improve program performance.

c. Conclusions

Self-assessment activities were identifying issues for improvement.

O8 Miscellaneous Operations Issues (92700, 92901)

O8.1 (Closed) Violation (VIO) 50-400/97-09-02: Failure to Properly Check Main Control Room Chart Recorder.

The inspectors verified the corrective actions described in the licensee's response, dated November 5, 1997, were completed. However, the corrective actions were determined to be ineffective as evidenced by an additional, similar occurrence. Discussion of the repeat occurrence was reported in NRC Inspection Report 50-400/98-01, which resulted in the issuance of VIO 50-400/98-01-01. Corrective actions for VIO 50-400/98-01-01 have not been completed by the licensee; therefore, this item remains open. VIO 50-400/97-09-02 is closed.

O8.2 (Closed) Licensee Event Report (LER) 50-400/97-02-00 and 50-400/97-02-01: Inoperable Main Feedwater Isolation Valves.

This event was discussed in NRC Inspection Report 50-400/97-03 and 50-400/97-12 resulting in the issuance of Non-Cited Violation (NCV) 50-400/97-03-02. The previous referenced inspection reports discussed the completion and verification of the licensee's corrective actions. The LER was kept open pending receipt and review of additional corrective actions as described in Supplement 1 of the LER. The inspectors verified completion of the remaining corrective actions. LERs 50-400/97-002-00, -01, are closed.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments

a. Inspection Scope (62707)

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

- WR/JO 98-ABMF-1 — Reactor Coolant Pump Cartridge Seal Rebuild
- WR/JO 97-AGQH-1 — Repair of Fire Wrap in Reactor Auxiliary Building
- WR/JO 98-AFBN-1 — Repair of 1A-SA Safety Bus Primary Undervoltage Relays
- WR/JO 98-AFBZ-1 — Repair of 1B-SB Safety Bus Secondary Undervoltage Relay

b. Observations and Findings

The inspectors found the work performed under these activities to be professional and thorough. All work observed was performed with the work package present and in active use. Technicians were experienced and knowledgeable of their assigned tasks. The inspectors frequently observed supervisors and system engineers monitoring job progress, and quality control personnel were present whenever required by procedure. Peer-checking and self checking techniques were being used. When applicable, appropriate radiation control measures were in place.

WR/JO 98-ABMF-1 required particularly skillful workmen. Discussion with the two workmen revealed them to be very knowledgeable and experienced at this task. The work was complicated by the need to keep all components wet for contamination and radiation control, and clean since the seal forms a reactor coolant system boundary.

During the time the work was in progress, health physics personnel were present to ensure proper control. Air sampling and smears were taken throughout the observation period. Radiation levels were monitored and the Health Physics (HP) technician's suggestions to control contamination were welcomed by the workmen.

This work was performed in a designated room due to the radiation levels and the need for cleanliness control. The seal package was refurbished in preparation for the next refueling outage. The post maintenance test was not completed at this time. It is required to be done within 30 days of the outage, which is currently scheduled for October, 1998. The post maintenance test is a leak-off pressure test of the seals.

WR/JO 97-AGQH-1 involved a cosmetic repair of the fire wrap outer blanket. A periodic, scheduled inspection of the fire wrap revealed several tears in the Siltemp blanket cover. A work ticket was initiated as specified by section 7 of the inspection procedure, FPT-3560 F, Revision (Rev.) 3. The tears did not cause the fire barrier to be inoperable. Integrity was maintained since there was no direct path for flames or hot gases to travel to the cables and the alumina silica blanket was in place. Therefore, the repair was considered to be cosmetic.

The workman involved with the repair was very knowledgeable and experienced. The completed work was independently inspected and accepted by a Quality Control inspector. The criteria for accepting the repairs were specified in CMP-009, One Hour Fire Wrap Process Control, Rev. 8. This activity involved climbing a ladder. Personnel safety was considered and utilized during the fire wrap repair.

c. Conclusions

The maintenance activities observed by the inspectors were performed by well qualified and experienced workmen. The work was satisfactorily completed using approved procedures. The craftsmen were very knowledgeable and skillful and performed their work in a professional manner.

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 Surveillance Observation

a. Inspection Scope (61726)

The inspectors observed all or portions of the following surveillance tests:

- OST-1092 1B-SB RHR Pump Operability Quarterly Interval Modes 1-2-3, Rev. 7.
- OST-1011 Auxiliary Feedwater Pump 1A-SA Operability Test, Rev. 8.
- MST-110196 Containment Wide Range Pressure Loop Calibration, Rev. 4.
- OST-1124 6.9 KV Emergency Bus Undervoltage Trip Actuating Device Operational Test Monthly Interval, Rev. 12.

b. Observations and Findings

These surveillances were performed in a professional manner by experienced, qualified personnel. The procedures for these tests were present and in use during performance of the testing. Pre-job briefings were held with the appropriate personnel prior to doing the work. Issues discussed included reviews of the surveillance test objective(s), Technical Specification requirements, communications, and acceptance criteria.

Upon completion of testing, restoration and alignment of plant components were completed and independently verified. Overall, these tests were well done and met the acceptance criteria.

c. Conclusions

The surveillance testing was adequately conducted. The operations and the maintenance personnel exhibited knowledge of the tasks, were professional, and accomplished the activities cautiously and with plant safety in mind.

M3 Maintenance Procedures and Documentation

M3.1 Containment Spray Check Valve Testing and Maintenance

a. Inspection Scope (62707)

The inspectors reviewed the testing requirements and recent maintenance history for check valves in the containment spray system.

b. Observations and Findings

The inspectors found that the check valves in the containment spray system were all in either ASME Code Class 2 or ASME Code Class 3. Consequently, because the inservice inspection (ISI) program imposes special inspection requirements only on valves in ASME Code Class 1, the subject check valves are not subject to special ISI requirements. Records of maintenance performed on these check valves indicate that the maintenance had been performed in accordance with applicable procedures, and the inservice testing program (IST). The maintenance and inspection activities have identified no adverse conditions associated with the valves.

c. Conclusions

Maintenance on the containment spray check valves has been conducted in accordance with applicable procedures, and has identified no adverse conditions associated with those valves.

M7 Quality Assurance in Maintenance Activities

M7.1 Maintenance Rule Periodic Assessment

a. Inspection Scope (62706)

Paragraph (a)(3) of the Maintenance Rule requires that performance and condition monitoring, associated goals and preventive maintenance activities for systems, structures and components (SCC's) be evaluated taking into account, where practical, industry-wide operating experience. This evaluation was required to be performed at least one time during each refueling cycle, not to exceed 24 months between evaluations. The NRC Maintenance Rule baseline inspection of Harris was conducted July 21-25, 1997. At that time the licensee had not completed the first periodic assessment since the Maintenance Rule did not take effect until July 10, 1996. On October 16, 1997, the licensee completed the assessment. This inspection was conducted to verify the effectiveness of the periodic assessment and of corrective actions taken.

b. Observations and Findings

The Harris Nuclear Plant (HNP) was operating in cycle 7 at the time the Maintenance Rule was implemented on July 10, 1996. Cycle 8 operation

began in June, 1997. Therefore, the first (a)(3) assessment, covers the period from July 10, 1996, until the beginning of cycle 8 operation. CP&L's Procedure No. ADM-NGGC-0101, Maintenance Rule Program, Rev. 9, which implements the Maintenance Rule for HNP, provides the details that should be addressed as part of the (a)(3) assessment in section 9.11, Periodic Assessment. This procedure states, in part that, "most of the items covered by this periodic assessment are on-going activities." Therefore, in addition to the items addressed in section 9.11, the evaluation took credit for a self-assessment (HESS 96-26) conducted on December 2-5, 1996, and resulting corrective actions, to satisfy the 10 CFR 50.65 (a)(3) requirements.

To verify the adequacy of the assessments, the inspectors reviewed completed documentation and held discussions with applicable engineers responsible for a sample of the eleven key procedural areas of the periodic assessments. Findings identified during the performance of the periodic assessment resulted in four CRs being initiated (97-04633, 97-04640, 97-04643 and 97-046440). One hundred and forty-nine findings were also identified by the licensee during the Maintenance Rule self-assessment performed December 2-5, 1996. As a result of this self-assessment, a considerable upgrade of the Maintenance Rule program occurred. The upgrade resulted in scoping at the function versus system level. This approach significantly expanded performance monitoring and resulted in considerably more involvement of the system engineer in the development of monitoring for his/her assigned systems. Therefore, in addition to the program upgrade, the system engineer became more knowledgeable of Maintenance Rule requirements. Corrective actions taken by the licensee for the above findings were reviewed and found to be satisfactory. The inspectors also reviewed changes made to Work Coordination Manual Procedure No. WCM-001, On-Line Maintenance Risk Management, Rev. 5, and held discussions with the superintendent of the Probabiliatic Safety Assessment (PSA) Unit concerning these changes.

c. Conclusions

The licensee's Maintenance Rule periodic assessment met all of the requirements delineated in the licensee's Maintenance Rule program procedure and the Maintenance Rule. Corrective actions sampled were appropriately implemented. As a result of findings identified during the self assessment, a significant upgrade in the Maintenance Rule program occurred, resulting in scoping at the function versus system level.

M8 Miscellaneous Maintenance Issues (92700, 92902)

M8.1 (Closed) VIO 50-400/96-09-02: Inadequate Maintenance Rule Scoping.

The licensee's Letter of Response, dated December 9, 1996, and Supplemental Responses, dated February 28, 1997, and July 30, 1997, were reviewed and found to be acceptable. This item dealt with the discovery by an inspector that the boric acid filter isolation valve, 1CS-559, was not scoped in the Maintenance Rule program. Subsequent investigation by

the licensee discovered that some specific components in maintenance rule systems were loaded in the Equipment Data Base System (EDBS) under other non-Maintenance Rule systems.

However, based on this violation and the preliminary results of initial NRC Maintenance Rule baseline inspections at other plants, Harris Nuclear Plant conducted a Maintenance Rule self assessment. The assessment was completed on December 5, 1996. Results of this assessment indicated other scoping items which needed further review to ensure full compliance with the Maintenance Rule. Additional time was requested by the licensee in letters of supplemental response, dated February 28, 1997, and July 30,1997, to complete their reviews, take the necessary corrective action, and to present their findings to the expert panel for approval.

The inspectors reviewed documentation of corrective actions taken by the licensee to correct the above findings. In addition, a list detailing the date and expert panel meeting which approved the new system scoping was reviewed. The corrective actions taken by the licensee were considered sufficient to correct the reported discrepancies and to prevent similar re-occurrences. This item is considered closed.

M8.2 (Closed) LER 50-400/97-24-00: Solid State Protection System (P10 and P14 Permissive Input) Testing Deficiency.

On November 18, 1997, with the plant at approximately 100 percent power, HNP engineering personnel determined that an operating experience item (OE#8636) was applicable to the HNP solid state protection system. The OE entry described a deficiency during testing of solid state protection system (SSPS) universal logic boards in a memory configuration, that would not allow the detection of an internal card, subcomponent failure. The potential card failure scenario could affect the feedwater isolation signal from safety injection inputs and P-14 permissive inputs, and the source range de-energization from the P-10 permissive input.

Based on the above information, previous tests performed to satisfy the requirements of Reactor Trip Instrumentation TS 4.3.1.1, item 21 (P-10), and Engineering Safety Feature Actuation System Instrumentation TS 4.3.2.1, item 5 (P-14), were inadequate in light of the identified testing anomaly. The failure to adequately test each of the inputs for P-10 and P-14 permissive signals was caused by inadequate surveillance procedures. The SSPS test scheme utilized in HNP surveillance procedures was designed and provided by the SSPS vendor. Following the evaluation of this condition, the HNP control room staff entered TS 4.0.3 and completed testing to verify the operability of the SSPS P-10 and P-14 permissive signals. This testing was accomplished on November 19, 1997, after revising the bi-monthly surveillance procedures (MST-I001 and MST-I0320) which were in error. The 18-month SSPS Actuation Logic and Master Relay Surveillance Procedures (MST-I0072 and MST-I0073) were also revised on June 4, 1998, to resolve the P-10 and P-14 SSPS logic testing anomaly prior to their next use, which is currently scheduled for Refueling Outage 8 (October 1998). The licensee

determined that there was no actual safety consequence as a result of the identified testing anomaly.

The above corrective actions were reviewed and found to be acceptable by the inspectors. However, failure to adequately test the inputs for P-10 and P-14 permissive signals is a violation of TS 4.3.1.1, item 21 (P-10), and TS 4.3.2.1, item 5 (P-14). This non-repetitive, licensee-identified and corrected violation is being treated as a non-cited violation, consistent with Section VII.B.1 of the NRC Enforcement Policy, and will be reported as NCV 50-400/98-07-01, SSPS P-10 and P-14 Permissive Input Testing Deficiency.

M8.3 (Closed) LER 50-400/98-02-00 & -01: Solid State Protection System (P-11 permissive overlap) Testing Deficiency.

On January 21, 1998, with the plant at approximately 100 percent power, HNP engineering personnel determined that Westinghouse Nuclear Safety Advisory Letter (NSAL) 97-011 was applicable to the HNP SSPS. This NSAL notified the industry that current SSPS design did not allow for complete overlap testing of the P-11 permissive function at power. Specifically, the monthly Analog Channel Operational Test (ACOT) required by TS 4.3.2.1, Table 4.3-2, item 10a, for the P-11 function could not be adequately performed at power.

NSAL 97-011 explained that the capability to test the P-11 permissive from the process protection system is limited to the setpoint and does not include the overlap to the SSPS logic input relay since the bistable test switch is opened prior to testing the channel. Opening the bistable test switch is required by the SSPS system design to satisfy the test logic for safety injection, since the SSPS input relay is de-energized above 2000 psig pressurizer pressure. Opening of the bistable test switch does not allow the input relay to change state during bistable setpoint verification; and therefore, complete overlap testing for the SSPS input relay is not accomplished.

The above testing discrepancy was caused by an inadequate review of the initial Technical Specification for consistency with other permissive signals and the capability for testing SSPS permissive signals at power.

NSAL 97-011 recommended that the P-11 function could be tested by a procedure revision which would bypass the test logic to permit the P-11 bistable test switch to remain in the normal position so that the logic input relay may be cycled during channel testing. The NSAL provided an example of how to implement this bypass by modifying jumpers on the NCT card in the pressurizer pressure circuit. The purpose of this bypass was to permit the logic input relay to become energized at power to test the P-11 function.

HNP revised the operational test procedure to lift the pressurizer pressure transmitter input lead to fail the pressure signal low during testing. This accomplished the same result as the NSAL recommendation to permit testing of the P-11 function, but was accomplished without a

plant modification. This testing in bypass was administratively controlled to be performed only if the pressurizer pressure is above the P-11 setpoint. Since actual pressure is above P-11 and safety injection is unblocked at the start of bypass testing, the safety function to automatically unblock safety injection when pressure increases above P-11 is met for this bypass testing. 10 CFR 50.59 safety evaluation reviews for the procedure and FSAR changes determined that this change does not involve an unreviewed safety question.

MST-I0122, MST-I0123, and MST-I0124, Rev. 6, which incorporated enhancements to P-11 permissive testing were reviewed by the inspectors. Testing for the revised procedures was completed on June 30, 1998, July 14, 1998 and July 6, 1998, respectively. However, failure to adequately complete the overlap testing of the P-11 permissive function at power, specifically, the monthly ACOT required by TS 4.3.2.1, Table 4.3-2, item 10a for the P-11 function, is a violation of the preceding Technical Specifications. This non-repetitive, licensee-identified and corrected violation is being treated as a non-cited violation, consistent with Section VII.B.1 of the NRC Enforcement Policy, and will be reported as NCV 50-400/98-07-02, P-11 permissive Overlap Testing Deficiency.

The above violation is non-repetitive even though LER 50-400/97-24 was submitted to NRC on December 18, 1997. This LER also reported a SSPS testing deficiency, but was specifically related to a logic card failure scenario. Therefore, the corrective actions completed for LER 50-400/97-24 would not have been expected to identify the P-11 overlap testing issue.

III. Engineering

E1 Conduct of Engineering

E1.1 Engineering Service Requests (37551)

The inspectors examined the circumstances associated with several rejected Engineering Service Requests (ESRs), to determine whether any significant issues had been dropped when the ESRs were rejected.

The inspectors determined that the ESRs were rejected for valid technical and/or administrative reasons and no significant issues were dropped.

E8 Miscellaneous Engineering Issues (92903)

E8.1 (Closed) VIO 50-400/97-12-05: Failure to Establish and Implement Engineering Procedures.

The corrective actions presented in the licensee's response, dated January 28, 1998, and accepted by the NRC on February 19, 1998, were verified as completed. VIO 50-400/97-12-05 is closed.

E8.2 (Open) VIO 50-400/97-13-02: Inadequate Corrective Actions for Preheater Bypass Valve Air System Design Deficiency.

The immediate corrective actions, described in the licensee's response, dated March 19, 1998, and accepted by the NRC on March 30, 1998, have been verified as completed by the inspectors. However, long range plans call for a plant modification to resolve a problem of a small air leak in the instrument air supply line that could disable a preheater bypass valve but not be detected in the main control room. Installation of this modification is planned for the next refueling outage (RFO 8), currently scheduled for October 1998. Therefore, this item will remain open.

E8.3 (Open) Unresolved Item (URI) 50-400/98-06-04: USQ Determination Related to Cycle 6 Reload.

This URI was identified during an inspection of licensee procedures for compliance with 10 CFR 50.59, and remained open to enable completion of an assessment of the adequacy of the licensee's Unreviewed Safety Question (USQ) Determination for the cycle 6 core reload.

NRC Inspection Report 50-400/98-06 noted that the licensee's analysis had indicated that the calculated dose consequences for certain accidents with the cycle 6 core had increased over the corresponding consequences with the cycle 5 core, and that, despite that increase, the licensee had determined that installing the cycle 6 core did not involve a USQ. This determination was documented in an unnumbered safety evaluation that was titled "NRM&SA HNP Cycle 6 Reload 50.59 Safety Review," dated April 24, 1994. That document concluded that because the dose consequences associated with the cycle 6 core were below the dose limits described in Chapter 15 of NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, use of the cycle 6 core did not constitute a USQ. The licensee's Plant Nuclear Safety Committee (PNSC) approved that safety evaluation in meeting 94-26. The licensee subsequently installed the cycle 6 core during refueling outage 5 (RFO 5), between March 19, 1994, and May 12, 1994, and inserted the increased dose results into the FSAR via Amendment 45.

However, because the calculated dose consequences for certain accidents with the cycle 6 core had increased over the corresponding consequences with the cycle 5 core, use of the cycle 6 core involved a USQ, which required obtaining prior Commission approval prior to startup from the refueling outage.

The licensee apparently first identified an issue with the cycle 6 dose consequences during a training session that was being held as part of the corrective actions for a separate issue. That issue had been reported in LER 95-006, and had been identified as a condition outside the design basis of the plant. During training that was conducted in February, 1996, feedback from participants in that class apparently prompted licensee management to reconsider the USQ determination associated with the cycle 6 reload. In a memo dated March 8, 1996, the

Regulatory Affairs Manager described a "plan for resolving an issue of 'increased consequences' from our cycle 6 Main Steam Line Break analysis," and the licensee subsequently directed the cycle 6 fuel vendor to reanalyze the main steam line break accident. The vendor completed the analyses, and the licensee inserted revised dose results (which showed no increase in consequences when compared to the calculated doses for cycle 5) into the FSAR via Amendment 48.

The licensee's actions thus indicate that the licensee had identified an adverse condition (the increased calculated doses) and had planned and implemented action to correct that condition.

The inspectors questioned the licensee about these activities in August 1998, and requested a copy of the condition report generated for the above issue. The licensee was unable to produce evidence that the condition had been documented as an adverse condition in accordance with procedure AP-615, Condition Reporting, for the 1996-1998 time frame or under new procedure CAP-NGGC-0001, Corrective Action Management. As a result of that questioning, the licensee initiated the following two CRs:

- CR 98-02092 documented the condition that "On 4/24/94 a 10 CFR 50.59 evaluation incorrectly determined that cycle 6 reload was not an unreviewed safety question."
- CR 98-02093 documented the condition that "In 1996 HNP identified an incorrect 10 CFR 50.59 evaluation and failed to generate a CR. Had a CR been generated, the reportability determination would have determined a condition outside design basis and an NRC report was required." This CR further states that "Since that time HNP has reevaluated the cycle 6 MSLB analysis such that dose consequences have not increased. Therefore, HNP is within its design basis currently (and NRC reporting is no longer required)."

This URI remains open pending NRC review of the collective safety and regulatory significance of these issues.

IV. Plant Support

R1 Radiological Protection and Chemistry (RP&C) Controls

R1.1 General Comments

a. Inspection Scope (71750)

The inspectors observed radiological controls during the conduct of tours and observation of maintenance activities.

b. Observations and Findings

The inspectors found radiological controls to be acceptable. The general approach to the control of contamination and dose for the site was good. Teamwork between the various departments continued to be a major contributor to the good control of dose.

c. Conclusions

The control of contamination and dose for the site was good and was attributable to good teamwork between the various departments.

S1 Conduct of Security and Safeguards Activities

S1.1 General Comments

a. Inspection Scope (71750)

The inspectors observed security and safeguards activities during the conduct of tours and observations of maintenance activities.

b. Observations and Findings

The inspectors found the performance of these activities was in accordance with procedures. Compensatory measures were posted when necessary and properly conducted.

c. Conclusions

The performance of Security and Safeguards activities was in accordance with applicable procedures.

F1 Control of Fire Protection Activities

F1.1 General Comments

a. Inspection Scope (71750)

The inspectors observed fire protection equipment and activities during the conduct of tours and observation of maintenance activities.

b. Observations and Findings

The inspectors found the fire protection activities properly conducted in accordance with procedures.

c. Conclusions

Fire Protection activities were being adequately conducted in accordance with procedures.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at the conclusion of the inspection on August 19, 1998. The licensee acknowledged the findings presented.

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

PARTIAL LIST OF PERSONS CONTACTED

Licensee

D. Batton, Superintendent, On-Line Scheduling
D. Braund, Superintendent, Security
B. Clark, General Manager, Harris Plant
A. Cockerill, Superintendent, I&C Electrical Systems
J. Collins, Manager, Maintenance
J. Cook, Manager, Outage and Scheduling
J. Donahue, Director Site Operations, Harris Plant
J. Eads, Supervisor, Licensing and Regulatory Programs
M. Keef, Manager, Training
G. Kline, Manager, Harris Engineering Support Services
R. Moore, Manager, Operations
K. Neuschaefer, Superintendent, Radiation Protection
W. Peavyhouse, Superintendent, Design Control
J. Scarola, Vice President, Harris Plant
S. Sewell, Superintendent, Mechanical Systems
D. Tibbitts, Manager, Nuclear Assessment
C. VanDenburgh, Manager, Regulatory Affairs

NRC

S. Flanders, Harris Project Manager, NRR
M. Ernstes, Acting Chief, Reactor Projects Branch 4

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 40500:	Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems
IP 61726:	Surveillance Observations
IP 62706:	Maintenance Rule
IP 62707:	Maintenance Observation
IP 71707:	Plant Operations
IP 71750:	Plant Support Activities
IP 92700:	Onsite Followup of Events
IP 92901:	Followup - Plant Operations
IP 92902:	Followup - Maintenance
IP 92903:	Followup - Engineering

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-400/98-07-01	NCV	Solid State Protection System P-10 and P-14 Permissive Input Testing Deficiency (Section M8.2).
50-400/98-07-02	NCV	P-11 Permissive Overlap Testing Deficiency (Section M8.3).

Closed

50-400/97-09-02	VIO	Failure to Properly Check Main Control Room Chart Recorder (Section O8.1).
50-400/97-02-00	LER	Inoperable Main Feedwater Isolation Valves (Section O8.2).
50-400/97-02-01	LER	Inoperable Main Feedwater Isolation Valves (Section O8.2).
50-400/98-07-01	NCV	Solid State Protection System P-10 and P-14 Permissive Input Testing Deficiency (Section M8.2).
50-400/98-07-02	NCV	P-11 Permissive Overlap Testing Deficiency (Section M8.3).
50-400/96-09-02	VIO	Inadequate Maintenance Rule Scoping (Section M8.1).
50-400/97-24-00	LER	Solid State Protection System (P-10 and P-14 Permissive Input) Testing Deficiency (Section M8.2).
50-400/98-02-00	LER	Solid State Protection System (P-11 Permissive overlap) Testing Deficiency (Section M8.3).

50-400/98-02-01	LER	Solid State Protection System (P-11 Permissive Overlap) Testing Deficiency (Section M8.3).
50-400/97-12-05	VIO	Failure to Establish and Implement Engineering Procedures (Section E8.1).

Discussed

50-400/97-13-02	VIO	Inadequate Corrective Actions for Preheater Bypass Valve Air System Design Deficiency (Section E8.2).
50-400/98-06-04	URI	USQ Determination Related to Cycle 6 Reload (Section E8.3).