UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
61 FORSYTH STREET, SW, SUITE 23T85
ATLANTA, GEORGIA 30303-8931

April 29, 2005

Carolina Power and Light Company
ATTN: Mr. James Scarola
Vice President - Harris Plant
Shearon Harris Nuclear Power Plant
P. O. Box 165, Mail Code: Zone 1
New Hill, North Carolina 27562-0165

SUBJECT: SHEARON HARRIS NUCLEAR POWER PLANT - NRC INTEGRATED INSPECTION REPORT 05000400/2005002

Dear Mr. Scarola:

On March 31, 2005, the US Nuclear Regulatory Commission (NRC) completed an inspection at your Shearon Harris reactor facility. The enclosed integrated inspection report documents the inspection findings, which were discussed on April 14, 2005, with you and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, one self-revealing finding and one inspector-identified finding of very low safety significance (Green) were identified. These findings were determined to involve violations of NRC requirements. However, because of their very low safety significance and because they have been entered into your Corrective Action Program (CAP), the NRC is treating these issues as non-cited violations, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you contest these non-cited violations, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001, with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Shearon Harris facility.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) components of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Paul E. Fredrickson, Chief
Reactor Projects Branch 4
Division of Reactor Projects

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

Enclosure: NRC Inspection Report 05000400/2005002
w/Attachment: Supplemental Information

cc w/encl: (See page 3)

cc w/encl:
Chris L. Burton, Manager
Performance Evaluation and
Regulatory Affairs CPB 9
Carolina Power & Light Company
Electronic Mail Distribution

Robert J. Duncan II
Director of Site Operations
Carolina Power & Light Company
Shearon Harris Nuclear Power Plant
Electronic Mail Distribution

Benjamin C. Waldrep
Plant General Manager--Harris Plant
Carolina Power & Light Company
Shearon Harris Nuclear Power Plant
Electronic Mail Distribution

Terry C. Morton, Manager
Support Services
Carolina Power & Light Company
Shearon Harris Nuclear Power Plant
Electronic Mail Distribution

David H. Corlett, Supervisor
Licensing/Regulatory Programs
Carolina Power & Light Company
Shearon Harris Nuclear Power Plant
Electronic Mail Distribution

Steven R. Carr
Associate General Counsel - Legal Department
Progress Energy Service Company, LLC
Electronic Mail Distribution

John H. O'Neill, Jr.
Shaw, Pittman, Potts & Trowbridge
2300 N. Street, NW
Washington, DC 20037-1128

Beverly Hall, Acting Director
Division of Radiation Protection
N. C. Department of Environmental
Commerce & Natural Resources
Electronic Mail Distribution

Peggy Force
Assistant Attorney General
State of North Carolina
Electronic Mail Distribution

Public Service Commission
State of South Carolina
P. O. Box 11649
Columbia, SC 29211

Chairman of the North Carolina
Utilities Commission
c/o Sam Watson, Staff Attorney
Electronic Mail Distribution

Robert P. Gruber
Executive Director
Public Staff NCUC
4326 Mail Service Center
Raleigh, NC 27699-4326

Herb Council, Chair
Board of County Commissioners
of Wake County
P. O. Box 550
Raleigh, NC 27602

Tommy Emerson, Chair
Board of County Commissioners
of Chatham County
Electronic Mail Distribution

Distribution w/encl: (See page 4)

Distribution w/encl:
C. Patel, NRR
L. Slack, RII EICS
RIDSNRRDIPMLIPB
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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket No:	50-400
License No:	NPF-63
Report No:	05000400/2005002
Licensee:	Carolina Power and Light Company
Facility:	Shearon Harris Nuclear Power Plant, Unit 1
Location:	5413 Shearon Harris Road New Hill, NC 27562
Dates:	January 1, 2005 - March 31, 2005
Inspectors:	R. Musser, Senior Resident Inspector P. O'Bryan, Resident Inspector R. Aiello, Sr. Operations Engineer, (Section 1R11.1) S. Rose, Sr. Operations Engineer, (Section 1R11.1)
Approved by:	P. Fredrickson, Chief Reactor Projects Branch 4 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000400/2005-002; 01/01/2005 - 03/31/2005; Shearon Harris Nuclear Power Plant, Unit 1; Maintenance Effectiveness, Event Follow-up.

The report covered a three-month period of inspection by the resident inspectors and an announced inspection by two regional operations engineers. Two green non-cited violations (NCVs) were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. Inspector-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. An inspector-identified finding and non-cited violation of 10CFR50, Appendix B, Criterion XVI, "Corrective Action" was identified for failure to promptly correct a condition adverse to quality. The licensee had identified, but did not implement prompt corrective action for a known condition adverse to quality. Specifically, although the design application of specific resistor in the turbine-driven auxiliary feedwater (TDAFW) pump speed control circuitry was determined to be deficient in March 2004, the resistor was not upgraded. The inspectors identified that the licensee did not evaluate this additional information to implement the upgrade sooner. Not reacting to the March 2004 information and not correcting the problem sooner, contributed to the failure of the TDAFW pump on January 5, 2005.

  The finding is more than minor because it affects the Mitigating Systems Cornerstone attribute of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e. core damage). The finding is also associated with the cornerstone attribute of equipment availability and reliability. NRC Inspection Manual Chapter 0609, Appendix A was used to evaluate this finding. Phase 2 and Phase 3 Significance Determination Process analyses determined that this finding is of very low safety significance (Green) because of an exposure time of 3.5 days and that, following the failure of the TDAFW to start, the pump could be started and controlled by plant operators. This finding is related to the cross-cutting area of problem identification and resolution due to the failure to promptly resolve a known condition adverse to quality. (Section 1R12)

Cornerstone: Initiating Events and Barrier Integrity

- Green. A self-revealing finding and non-cited violation of 10CFR50, Appendix B, Criterion III, "Design Control," was identified for the implementation of an inadequate design change, Engineering Service Request (ESR) 97-0233. The inadequate design change resulted in damage to the seats of all three feedwater isolation valves (FWIV). The damaged seats existed for a period of time greater than the allowed inoperability time specified in Technical Specification 3.6.3.

The finding is more than minor because it affects the Barrier Integrity Cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radio nuclide releases caused by accidents or events, and is associated with the cornerstone attribute of design control. The finding also affected the Initiating Events cornerstone attribute of design control due to the increased likelihood of FWIV stem separation and reactor trip at higher reactor power. NRC Inspection Manual Chapter 0609, Appendix A and Appendix H were used to evaluate this finding. A Phase 2 Signficance Determination Process analysis determined that this finding is of very low safety significance (Green) because the upstream feedwater valves were available for feedwater isolation and the loss of the power conversion system was not considered because the valve stems in the A and B FWIV were not degraded. (Section 4OA3)

B. Licensee-Identified Violations

None.

REPORT DETAILS

Summary of Plant Status

The unit began the inspection period at rated thermal power and operated at or near full power for the entire inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R04 Equipment Alignment

a. Inspection Scope

Partial System Walkdowns:

The inspectors performed the following three partial system walkdowns, while the indicated structures, systems and components (SSCs) were out-of-service (OOS) for maintenance and testing:

- ‘A’ emergency service water system with ‘B’ emergency service water system out-of-service on January 14.
- ‘A’ emergency diesel generator with ‘B’ emergency diesel generator out-of-service on January 19.
- ‘B’ emergency diesel generator with ‘A’ emergency diesel generator out-of-service on February 1.

To evaluate the operability of the selected trains or systems under these conditions, the inspectors reviewed valve and power alignments by comparing observed positions of valves, switches, and electrical power breakers to the procedures and drawings listed in the Attachment.

Complete System Walkdown:

The inspectors conducted a detailed review of the alignment and condition of the high head safety injection system. To determine the proper system alignment, the inspectors reviewed the procedures, drawings, and Final Safety Analysis Report (FSAR) sections listed in the Attachment.

The inspectors walked down the system to verify that the existing system alignment was consistent with the correct alignment. Items reviewed during the walkdown included the following:

- Valves are correctly positioned and do not exhibit leakage that would impact the function(s) of any given valve.
- Electrical power is available as required.

- Major system components are correctly labeled, lubricated, cooled, ventilated, etc.
- Hangers and supports are correctly installed and functional.
- Essential support systems are operational.
- Ancillary equipment or debris does not interfere with system performance.
- Tagging clearances are appropriate.
- Valves are locked as required by the licensee's locked valve program.

The inspectors reviewed the documents listed in the Attachment to verify that the ability of the system to perform its function could not be affected by outstanding design issues, temporary modifications, operator workarounds, adverse conditions, and other system-related issues tracked by the Engineering Department.

The inspectors reviewed the following action requests (ARs) associated with this area to verify that the licensee identified and implemented appropriate corrective actions:

- AR #108197, "Gas Void Upstream 1CS-526"
- AR #129123, "CSIP Functional Failure"
- AR #142303, "Gas Voids Found in RWST Suction to CSIP"

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

For the twenty-three areas identified below, the inspectors reviewed the licensee's control of transient combustible material and ignition sources, fire detection and suppression capabilities, fire barriers, and any related compensatory measures, to verify that those items were consistent with FSAR Section 9.5.1, Fire Protection System, and FSAR Appendix 9.5.A, Fire Hazards Analysis. The inspectors walked down accessible portions of each area and reviewed results from related surveillance tests, to verify that conditions in these areas were consistent with descriptions of the applicable FSAR sections.

- 305' level of the reactor auxiliary building including areas 12-A-6-RCC1, 12-A-6-ARP1, 12-A-6-CR, 12-A-6-IRR, and 12-A-6-PICR1 (5 areas).
- 286' level of the reactor auxiliary building including areas 1-A-CSRA, 1-A-CSRB, and 1-A-ACP (3 areas).
- 236' level of the reactor auxiliary building including areas 1-A-3-MP, 1-A-3-COR, and 1-A-3-COME (3 areas).
- 305' level of the reactor auxiliary building including areas 12-A-6-CR1 and 12-A-6-RT1 (2 areas).

- 286' level of the reactor auxiliary building including areas 1-A-BATA, 1-A-BATB, 1-A-SWGRA and 1-A-SWGRB (4 areas).
- The 'A' emergency diesel generator areas including 1-D-1-DGA-RM, 1-D-3-DGA-ES, 1-D-DTA, 1-D-1-DGA-ASU, 1-D-DGA-ER, and 1-D-3-DGA-HVR (6 areas).

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance

a. Inspection Scope

The inspectors verified acceptable performance of the emergency diesel generator jacket water heat exchangers by reviewing licensee test data, condition reports, and procedures. The inspectors verified test results were appropriately categorized against the pre-established acceptance criteria described in Procedure EPT-250, "A Train Emergency Service Water Flow Verification/Balance," and Procedure EPT-251, "B Train Emergency Service Water Flow Verification/Balance." The inspectors also verified that the frequency of testing was sufficient to detect degradation prior to loss of heat removal capability below design basis values. Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

.1 Annual Operating Test Results

a. Inspection Scope

The inspectors reviewed the facility operating history and associated documents in preparation for this inspection. During the week of March 7-11, 2005, the inspectors reviewed documentation, interviewed licensee personnel, and observed the administration of simulator operating tests associated with the licensee's operator requalification program. Each of the activities performed by the inspectors was done to assess the effectiveness of the licensee in implementing requalification requirements identified in 10 CFR 55, "Operators' Licenses." The evaluations were also performed to verify that the licensee effectively implemented operator requalification guidelines established in NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," and Inspection Procedure 71111.11, "Licensed Operator Requalification Program." The inspectors also reviewed and evaluated the licensee's simulation facility for adequacy for use in operator licensing examinations. The inspectors observed two operator crews during the performance of the operating tests. Documentation reviewed included written examinations, job performance measures (JPMs), simulator scenarios, licensee procedures, on-shift records, simulator modification request and performance test records, the feedback process, licensed operator qualification records, remediation plans, watchstanding, and medical records. The records were inspected against the criteria listed in Inspection Procedure 71111.11. Documents reviewed during the inspection are listed in the Attachment.

b. Findings

Introduction. An unresolved item (URI) was identified in that a potential disqualifying condition (solo operation) for a licensed operator existed as stated in the American Nuclear Standards Institute American Nuclear Society (ANSI/ANS) 3.4,1983 standard. This issue is unresolved pending completion of an NRC medical review to evaluate a cardiovascular condition of one licensed operator.

Description. The NRC's requirements related to the conduct and documentation of medical examinations for operators are contained in Subpart C, "Medical Requirements," of 10 CFR Part 55, "Operators' Licenses." Specifically, Section 55.21, "Medical examination," requires every operator to be examined by a physician when he or she first applies for a license. The physician must determine whether the operator meets the requirements of Section 55.33(a)(1), i.e., the operator's medical condition and general health will not adversely affect the performance of assigned operator duties or cause operational errors that endanger public health and safety.

Every time an operator applies for a license pursuant to Section 55.31, "How to apply," or Section 55.57, "Renewal of licenses," an authorized representative of the facility licensee must complete and sign NRC FORM-396, "Certification of Medical Examination by Facility Licensee," attesting, pursuant to Section 55.23, "Certification," that a physician has conducted the required medical examination and determined that the operator's medical condition and general health meet the requirements of Section 55.33(a)(1). The facility licensee must also certify which industry standard (i.e., the 1983 or 1996 version of ANSI/ANS-3.4, "Medical Certification and Monitoring of Personnel Requiring Operator Licenses for Nuclear Power Plants," or other NRC-approved method was used in making the fitness determination.

The ANSI standards describe a number of specific operator health requirements and disqualifying conditions. If an operator's health does not meet the minimum standards, the facility licensee must request a conditional license in accordance with Section 55.23(b) by submitting the appropriate medical evidence with NRC FORM-396. Pursuant to Section 55.33, "Disposition of an initial application," and Section 55.57, as applicable, the NRC will review the license application based on the facility licensee's certification and include any conditions in the license that might be necessary based on the supporting medical evidence.

During a medical records review, the inspectors identified that an operator's record may need to have a "no solo" condition on the individual's operating license to satisfy a potential disqualifying condition due to heart erythema in order to meet the ANSI/ANS 3.4 1983 cardiovascular requirements. The facility licensee was informed that the individual may have had to require an amendment to his/her license that required compliance with a "no solo" condition while performing licensed duties. This issue will be identified as URI 05000400/2005002-01, Potential Disqualifying Condition for a Licensed Operator, pending completion of an NRC medical review of operator's NRC FORM 396 to determine if a license condition is warranted.

.2 Quarterly Training Observation

a. Inspection Scope

On January 25, the inspectors observed licensed-operator performance during requalification simulator training for crew A, to verify that operator performance was consistent with expected operator performance, as described in Exercise Guide EOP-SIM-17.108, "Loss of Instrument Air/Reactor Trip." This training tested the operators' ability to respond to a complete loss of service and instrument air. The inspectors focused on clarity and formality of communication, the use of procedures, alarm response, control board manipulations, group dynamics and supervisory oversight.

The inspectors observed the post-exercise critique to verify that the licensee had identified deficiencies and discrepancies that occurred during the simulator training.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed the two degraded SSC/function performance problems/conditions listed below to verify the licensee's handling of these performance problems/conditions was in accordance with 10CFR50, Appendix B, Criterion XVI, Corrective Action, and 10CFR50.65, Maintenance Rule. Documents reviewed are listed in the Attachment.

- The failure of the turbine driven auxiliary pump control circuitry on January 5.
- Flange leakage on 'A' RHR heat exchanger

The inspectors focused on the following attributes:

- Appropriate work practices,
- Identifying and addressing common cause failures,
- Scoping in accordance with 10 CFR 50.65(b),
- Characterizing reliability issues (performance),
- Charging unavailability (performance),
- Trending key parameters (condition monitoring),
- 10 CFR 50.65(a)(1) or (a)(2) classification and reclassification, and

- Appropriateness of performance criteria for SSCs/functions classified (a)(2) and/or appropriateness and adequacy of goals and corrective actions for SSCs/functions classified (a)(1).

The inspectors reviewed the following ARs associated with this area to verify that the licensee identified and implemented appropriate corrective actions:

- AR #109760, “Mechanical overspeed of the TDAFW pump”
- AR #147194, “TDAFW T&TV tripped shut when opening from MCB”

b. Findings

Introduction. An inspector identified Green NCV of 10CFR50, Appendix B, Criterion XVI, “Corrective Action” was identified for failure to assure that a condition adverse to quality was promptly corrected. During surveillance testing, the turbine-driven auxiliary feedwater (TDAFW) pump oversped and tripped. The overspeed trip was attributed to a failed voltage-dropping resistor in the turbine speed control circuitry. Following multiple previous failures of this component, the licensee included this problem in the CAP and identified that the long-term corrective action for the adverse condition was a design change of the circuitry. However, the inspectors identified that the design change was not implemented in a prompt manner after additional information on the problem was received by the licensee.

Description. On January 5, 2005, during surveillance testing, the TDAFW pump oversped and tripped. The cause of the failure was determined to be an open circuit in the R5 voltage dropping resistor in the TDAFW pump’s speed control circuitry. Similar failures of the R5 resistor, and TDAFW pump overspeed events, had occurred in 1989, 1993, and 2003. In March 2004, industry operating experience indicated that upgrading the R5 resistor to a higher wattage successfully remedied similar repeat failures of Terry Turbine speed control systems at other plants. In March 2004, based on this operating experience information, as well as vendor information and testing results from the Progress Energy E&E Center, the licensee concluded and documented in AR #109760, that a design change to upgrade the R5 resistor was required to address the components susceptibility to failure. As of January 5, 2005 when the relay failed, fourteen months after the 2003 failure, this corrective action had not been implemented. After several schedule delays, the assigned due date for the engineering design change at the time of the most recent failure, was June 2005.

Analysis. The finding is more than minor because it affects the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e. core damage). It is also associated with the cornerstone attribute of equipment availability and reliability. Since the finding is associated with the operability, availability, reliability, or function of a system or train in a mitigating system at power, an evaluation using NRC Inspection Manual Chapter (IMC) 0609, Appendix A, “Determining the Significance of Reactor Inspection Findings for At-Power Situations” was appropriate. The TDAFW pump was last run successfully on December 29, 2004. The TDAFW failure occurred on

January 5, 2005. Therefore, using a t/2 calculation, the exposure time was approximately 3.5 days. Since this is greater than the allowed Technical Specification (TS) outage time for the TDAFW train, a Significance Determination Process (SDP) Phase 2 evaluation was required. Based on the results of the Phase 2 screening, the finding was of potentially greater than very low risk significance (greater than Green) and a Phase 3 evaluation was required. **The Phase 3 evaluation determined that the finding was Green. The risk quantification was performed with the licensee's full scope model and the NRC's computerized model. The evaluation was predicated on an exposure time of 84 hours with the TDAFW pump failing to initially start, but also that it could be recovered by operator actions. The dominant accident sequences involved loss of offsite power, failure of the emergency diesel generators via various means and failure to recover offsite power in a timely manner. There was good agreement between both model results. Due to the low internal events contribution, external events were not required to be analyzed**.

Enforcement. 10CFR50, Appendix B, Criterion XVI requires in part that conditions adverse to quality, such as equipment deficiencies, be promptly identified and corrected. Contrary to the above, the licensee identified, but did not implement prompt corrective action for a known condition adverse to quality. Specifically, although the design application of the R5 resistor in the TDAFW pump speed control circuitry was determined to be deficient in March, 2004, the resistor was not upgraded. The inspectors identified that the licensee did not evaluate this additional information to implement the upgrade sooner. Not reacting to the March 2004 information by correcting the problem sooner contributed to the failure of the TDAFW pump on January 5, 2005. **Because the finding is of very low safety significance and has been entered in the licensee's CAP (AR #147194), this violation is being treated as a non-cited violation (NCV), consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000400/2005002-02**, Failure to Correct a TDAFW Pump Condition Adverse to Quality. This finding is related to the cross-cutting area of problem identification and resolution due to the failure to promptly resolve a known condition adverse to quality.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

The inspectors reviewed the licensee's risk assessments and the risk management actions for the plant configurations associated with the four activities listed below. The inspectors verified that the licensee performed adequate risk assessments, and implemented appropriate risk management actions when required by 10CFR50.65(a)(4). For emergent work, the inspectors also verified that any increase in risk was promptly assessed, and that appropriate risk management actions were promptly implemented.

- 'B' emergency service water outage on January 14 with a tornado watch in effect.
- 'B' emergency diesel generator outage on January 21 with a winter storm advisory in effect.

- Emergent corrective maintenance on the ‘B’ essential service chilled water system on February 25.
- The boric acid totalizer on March 8 with a thunder storm warning in effect.

Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed four operability determinations addressed in the ARs listed below. The inspectors assessed the accuracy of the evaluations, the use and control of any necessary compensatory measures, and compliance with the TS. The inspectors verified that the operability determinations were made as specified by Procedure AP-618, "Operability Determinations." The inspectors compared the justifications made in the determination to the requirements from the TS, the FSAR, and associated design-basis documents, to verify that operability was properly justified and the subject component or system remained available, such that no unrecognized increase in risk occurred:

- AR#146708, “B P-4 Pump Bearing Oil Below Minimum Value When Sampled”
- AR#150114, “Loss of DC Power to B Circuit on A-EDG Anomaly”
- AR#152362, “A and B Train ESCW System Operability Determination”
- AR #151068, “Incorrect Fuse Installed in ARP-2B-SB R2 Location KIB/3095"

Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed the modification described in Engineering Change #59959, “Replacement of TDAFW Turbine Overspeed Resistor R5,” to verify that:

- This modification did not degrade the design bases, licensing bases, and performance capabilities of risk significant SSCs
- Implementing this modification did not place the plant in an unsafe condition, and

- The design, implementation, and testing of this modification satisfied the requirements of 10CFR50, Appendix B.

Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

For the five post-maintenance tests listed below, the inspectors witnessed the test and/or reviewed the test data to verify that test results adequately demonstrated restoration of the affected safety functions described in the FSAR and TS. The tests included the following:

- OST-1073, “1B-SB Emergency Diesel Generator Operability Test Monthly Interval Modes 1 through 6,” and OST-1824, “1B-SB Emergency Diesel Generator Test 18 Month Interval Modes 1 through 6 and Defueled” as the post maintenance test for the ‘B’ emergency diesel generator outage from January 19 through January 21.
- OST-1092, “1B-SB RHR Pump Operability Quarterly Interval, Modes 1-2-3" after maintenance on the ‘B’ RHR pump circuit breaker and valve 1SI-323 on January 26.
- OST-1411, “Auxiliary Feedwater Pump 1X-SAB Operability Test Quarterly Interval Modes 1, 2, 3" after maintenance on the TDAFW governor control circuitry on February 10.
- ORT-1408, “Security Diesel Operability Run Monthly Interval Modes: All” following maintenance on the security diesel fuel oil system on February 25,
- OP-111, “Residual Heat Removal System” following maintenance on the ‘A’ train residual heat removal pump.

The inspectors reviewed the following two ARs, associated with this area to verify that the licensee identified and implemented appropriate corrective actions:

- AR #148638, “EDG 1B-SB AC Meter EI-01EE-6955BIV”
- AR #148699, “B EDG Tripped Following Start From Safety System Outage”

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

For the six surveillance tests identified below, the inspectors witnessed testing and/or reviewed test data, to verify that the SSCs involved in these tests satisfied the requirements described in the TS and the FSAR, and that the tests demonstrated that the SSCs were capable of performing their intended safety functions.

- EPT-033, “Emergency Safeguards Sequence System Test.” and EPT-443, “Emergency Safeguards Sequence Relay Trend and Analysis” on January 10.
- MST-I0207, “Refueling Water Storage Tank Level (L-0993) Operational Test” on February 23.
- *OST-1211, “Auxiliary Feedwater Pump 1A-SA Operability Test, Quarterly Interval Modes 1-4" on February 28.
- *OST-1040, Essential Services Chilled Water Systems Operability, Quarterly Interval Modes 1-6,” for ‘A’ ESCW chiller on March 1.
- OST-1026, “RCS Leakage Evaluation, Computer Calculation, Daily Interval, Modes 1-2-3-4" on March 8.
- ORT-1813, “Remote Shutdown: Test of Additional Components on the ACP 18 Month Interval Modes 1-6 or Defueled” on March 13.

*This procedure included inservice testing requirements.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed the temporary modifications described in Operating Procedure OP-172, “Reactor Auxiliary Building HVAC System”, and Engineering Change EC #60425, “Temporary Manual and Alternate Air Make-up to A & B ESCW Expansion Tanks”, to verify that the modifications did not affect the safety functions of important safety systems, and to verify that the modifications satisfied the requirements of 10CFR50, Appendix B, Criterion III, Design Control. Documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation

a. Inspection Scope

The inspectors observed an emergency preparedness drill conducted on January 27 to verify licensee self-assessment of classification, notification, and protective action recommendation development in accordance with 10CFR50, Appendix E. The inspectors also observed two operations simulator examinations conducted on March 1 and March 8, to verify licensee self-assessment of classification, notification, and protective action recommendation development in accordance with 10CFR50, Appendix E.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems

Routine Review of Action Requests

.1 Routine Review of ARs

To aid in the identification of repetitive equipment failures or specific human performance issues for followup, the inspectors performed frequent screenings of items entered into the licensee's CAP. The review was accomplished by reviewing daily AR reports.

.2 Problem Identification and Resolution Cross-Cutting Aspects

The finding described in Section 1R12 regarding the TDAFW pump overspeed tripping on demand has as its primary cause problem identification and resolution in that the licensee failed to promptly correct a known problem with the TDAFW pump's speed control circuitry.

4OA3 Event Follow-up

.1 (Closed) Licensee Event Report (LER) 05000400/2004-006-00: Manual Actuation of Auxiliary Feedwater Pump.

a. Inspection Scope

The inspectors reviewed the subject LER and Condition Report 143023 to assess the cause and licensee actions taken for the manual actuation of auxiliary feedwater on November 7, 2004. The inspectors reviewed the event to confirm that plant equipment performed as required and that operators took the appropriate actions in response to the event. The inspectors also reviewed the corrective actions to verify that they were appropriate for the event. Documents reviewed are listed in the Attachment.

b. Findings

Introduction. A self-revealing green NCV of 10CFR50, Appendix B, Criterion III, "Design Control," was identified due to an inadequate design change, (ESR 97-0233), which resulted in damage to the seats of all three feedwater isolation valves (FWIV). The damaged seats prevented the valves from isolating containment for greater than the allowed outage time specified in TS 3.6.3.

Description. On November 7, 2004 with reactor power at approximately 4% during power ascension, operators actuated an auxiliary feedwater pump in response to lowering water level in the 'C' steam generator. The plant was shutdown and cooled down, and investigation revealed that the stem of 'C' FWIV was fractured, preventing the valve from opening. Investigation also revealed that the seats on all three FWIV's were damaged, preventing them from performing their containment isolation and feedwater isolation functions.

The licensee's root cause investigation concluded that the root cause of the damage to the FWIV seats and the 'C' FWIV stem failure was inadequate design and design implementation of a modification (ESR 97-0233) to the valves during a refueling outage in April 2000. The licensee also concluded that there was sufficient evidence to indicate that the FWIV seats sustained significant damage as early as the fall of 2001. Although the FWIVs were unable to fulfill their feedwater isolation function, the upstream feedwater isolation valves for each steam generator were available and functional.

Analysis. The inspectors determined that the inadequate design change was more than minor because it affects the Barrier Integrity Cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radio nuclide releases caused by accidents or events. The finding is also associated with the Barrier Integrity Cornerstone attribute of design control. The finding affected the Initiating Events Cornerstone attribute of design control due to the increased likelihood of FWIV separation and reactor trip at higher reactor power. Therefore, a Phase 2 SDP evaluation was required. NRC IMC 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations" was used to determine the change in core damage frequency (CDF). Since the 'A' and 'B' FWIV stems were not damaged, transients associated with the loss of the power conversion system were not considered, and the only core damage sequences affected were those associated with the increased likelihood of a reactor trip. The change in CDF was found to be <10E-7. NRC IMC 0609, Appendix H, "Containment Integrity Significance Determination Process," was also used to determine the significance of the finding with regards to the large, early release frequency (LERF). Per Figure 4.1 of Appendix H, since the change in CDF was <10E-7, the finding is of very low safety significance (Green).

Enforcement. 10CFR50, Appendix B, Criterion III, "Design Control," requires that measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. Contrary to this requirement, an inadequate design (ESR 97-0233) was implemented in April 2000, which resulted in FWIV seat damage and the inability of the FWIV to perform containment isolation functions from 2000 until 2004. However, because of the very low safety significance and because the issue was entered into the CAP (AR #143023), and that the compliance was promptly restored, this finding is being treated as a non-cited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000400/2005002-03, Inadequate Design Results in FWIV's Being Unable to Provide Containment Isolation.

4OA6 Meetings, Including Exit

On April 14, 2005, the resident inspectors presented the inspection results to Mr. Scarola and other members of his staff. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

D. Braund, Superintendent, Security
J. Briggs, HNP, Superintendent, Environmental and Chemical
J. Carney, Supervisor, LOCT
D. Corlett, Supervisor - Licensing/Regulatory Programs
F. Diya, Manager - Engineering
R. Duncan, Director - Site Operations
W. Gurganious, Manager - Nuclear Assessment
C. Kamilaris, Superintendent, OPS Training
E. McCartney, Training Manager
G. Miller, Maintenance Manager
T. Morton, Manager - Support Services
T. Natale, Manager -Outage and Scheduling
T. Pilo, Supervisor - Emergency Preparedness
J. Scarola, Vice President Harris Plant
G. Simmons, Superintendent - Radiation Control
J. Warner, Manager, Shift Operations
E. Wills, Operations Manager
B. Waldrep, General Manager Harris Plant
M. Wallace, Licensing Specialist

NRC personnel

P. Fredrickson, Chief, Reactor Projects Branch 4

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000400/2005002-01	URI	Potential Disqualifying Condition for a Licensed Operator (Section 1R 11).

Opened and Closed

05000400/2005002-02	NCV	Failure to Correct a TDAFW Pump Condition Adverse to Quality (Section 1R12)
05000400/2005002-03	NCV	Inadequate Design Results in FWIV's Being Unable to Provide Containment Isolation (Section 4OA3)

Closed

05000400/2004-006-00	LER	Manual Actuation of Auxiliary Feedwater Pump (Section 4OA3)

LIST OF DOCUMENTS REVIEWED

<u>Section 1R04:Equipment Alignment</u>

<u>Partial System Walkdown</u>
Emergency Service Water system:

Procedure OP-139, “Service Water System”
Drawings 2165-S-0547 and 2165-S-0548, Simplified Flow Diagrams “Circulating and Service Water Systems”

Emergency Diesel Generator system:

Procedure OP-155, “Diesel Generator Emergency Power System”
Drawing 2165-S-0633, sheets 1 through 4, “Simplified Flow Diagram Emergency Diesel Generator Systems

<u>Complete System Walkdown</u>
Procedure OP-110, “Safety Injection System”
System Description 107, “Chemical and Volume Control System”
System Description 110, “High Head Safety Injection System”
Design Basis Document -104, “Safety Injection System”
Drawing 2165-S-1304, “Simplified Flow Diagram Chemical and Volume Control System"
Drawing 2165-S-1305, “Simplified Flow Diagram Chemical and Volume Control System"
Drawing 2165-S-1308, “Simplified Flow Diagram Safety Injection System"
FSAR section 6.3, “Emergency Core Cooling System”
Work order #618141, “Mechanical Seal Replacement B CSIP”
Maintenance Rule Database, systems 2060 and 2080
System 2060 and 2080 Health Reports
Abnormal Operating Procedure - 017, “Loss of Instrument Air”

<u>Section 1R07:Heat Sink Performance</u>

<u>Procedures</u>:
EPT-250, “A Train ESW Flow Verification/Balance”
EPT-251, “B Train ESW Flow Verification/Balance”
PLP-620, “Service Water Program (Generic Letter 89-13)”

<u>Other Documents:</u>
AR #126041, “Low Margin on ESW Flow t 1A-SA EDG JW Cooler”
AR #50768, “Service Water System Performance”
AR #50611, “B EDG Jacket Water Heat Exchanger Fouling”

<u>Section 1R11:Licensed Operator Requalification</u>

AOP-017, “Loss of Instrument Air”
EOP-EPP-004, “Reactor Trip Response”
Adverse Condition Investigation Form CAP-NGGC-0200-3-8

Harris Operations Assessment (H-OP-03-01)
Self Assessment 114694 dated August 2-6, 2004
Scenario DSS-034, Large Break LOCA
Scenario DSS-006, Station Blackout
Badge Access Transaction Reports for Reactivation of Licenses (4)
Licensed Operator Medical Records (22)
Feedback Summaries
Human Performance Errors
2003, 2004 Active Licenses Time on Shift

Remedial Training Records:
Inspectors reviewed four remedial training records, one for a simulator exam failure, and three for simulator passes with remediation.

Written Exams Reviewed:
RO/SRO 2004 LOCT Annual Exam week 1 (04-02-1 RO/SRO)
RO/SRO 2004 LOCT Annual Exam week 4 (04-02-4 RO/SRO)

Simulator Fidelity Documents:
Instructor Log for Open SSRs and Resolved SSRs (Specifically Items 04-0226, 04-0036, 04-0016, 04-0125, 03-0206, 03-0455, 04-0334)
TPP-206, Simulator Program, Rev 6
TAP-409, Conduct of Simulator Training and Evaluation, Rev 3
TAP-412, Simulator Operation and maintenance, Rev 2

Simulator Performance Testing:
Trip of a single Reactor Coolant Pump, TT-05
Power Ramp from 100% to 75% and back to 100% at 45 MW/min, TT-07
Maximum size non-isolable steam break from any single Steam Generator, TT-09
Maximum Design Load Rejection, TT-11
Steady State Test 100%, SST-001
Steady State Test 75%, SST-002
Steady State Test 25%, SST-003
Simulator validation testing for scenarios: 1) DSS-006 and 2) DSS-034

Section 1R12:Maintenance Effectiveness
NUMARC 93-01, “Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants”
ADM-NGGC-0101, “Maintenance Rule Program”

1R13 Maintenance Risk Assessments and Emergent Work Evaluation
WCM-001, On-Line Maintenance Risk Management

1R15 Operability Evaluations
AP-618, “Operability Determinations”
PLP-628, “Plant Fuse Control Program for 1E and Non-1E Applications”

1R17 Permanent Plant Modifications
System Description SD-137, “Auxiliary Feedwater”
Design Basis Document DBD-114, “Auxiliary Feedwater System”
Final Safety Analysis Report Section 10.4.9, Auxiliary Feedwater System

1R23 Temporary Plant Modifications
System Description SD-148, “Essential Services Chilled Water System”
Design Basis Document DBD-114, “Essential and Nonessential Services Chilled Water Systems”
Final Safety Analysis Report Section 9.2.8, Essential Services Chilled Water

4OA3 Event Follow-up
System Description SD-134, “Condensate and Feedwater”
Design Basis Document DBD-112, “Condensate, Main Feedwater, Condensate Polishers, Feedwater Drains and Vents Systems”
Final Safety Analysis Report Section 10.4.7, Condensate and Feedwater System