UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
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July 28, 2003

Carolina Power & 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 05000400/2003003

Dear Mr. Scarola:

On June 28, 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 June 26, 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, the inspectors identified one issue of very low safety significance (Green). This issue was determined to involve a violation of NRC requirements. However, because of its very low safety significance and because it has been entered into your corrective action program, the NRC is treating this issue as a non-cited violation, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you contest this non-cited violation, 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 2.790 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 by G. MacDonald for/

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

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

Enclosure: NRC Inspection Report 05000400/2003003
w/Attachment: Supplemental Information

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket No:	50-400
License No:	NPF-63
Report No:	05000400/2003003
Licensee:	Carolina Power & Light Company
Facility:	Shearon Harris Nuclear Power Plant, Unit 1
Location:	5413 Shearon Harris Road New Hill, NC 27562
Dates:	April 6, 2003 - June 28, 2003
Inspectors:	J. Brady, Senior Resident Inspector R. Hagar, Resident Inspector J. Lenahan, Senior Reactor Inspector (Section 1R08) M. Scott, Senior Reactor Inspector (Section 1R07.2) E. Testa, Senior Health Physicist (Section 2OS1) R. Hamilton, Health Physicist (Sections 2OS2 & 4OA1) K. O'Donohue, Senior Operations Engineer (Section 1R11.2) R. Musser, Senior Resident Inspector, (Surry) (Section 4OA5) M. Maymi, Reactor Inspector (Section 1R07.2)
Approved by:	P. Fredrickson, Chief Reactor Projects Branch 4 Division of Reactor Projects

Enclosure

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SUMMARY OF FINDINGS

IR 05000400/2003-003; 04/06/2003 - 06/28/2003; Shearon Harris Nuclear Power Plant, Unit 1; Personnel Performance During Non-Routine Plant Evolutions and Events.

The report covered a three-month period of inspection by resident inspectors and announced inspections by two regional health physicists, three reactor inspectors, and an operations engineer. One Green non-cited violation (NCV) was 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: Initiating Events

Green. A non-cited violation of Technical Specification 6.8.1 was identified for a failure to establish adequate general operating procedures for reactor trip recovery and hot standby to minimum load (nuclear startup). The general operating procedures did not ensure that the main feedwater regulating valves were shut or isolated prior to operators shutting the reactor trip breakers. Not isolating the main feedwater regulating valves during recovery from a reactor trip resulted in two main feedwater regulating valves opening when the reactor trip breakers were shut. Being in this condition caused a high level in two steam generators and protective signals to trip the main feedwater pumps, isolate the feedwater lines, and start the motor-driven auxiliary feedwater pumps.

The self-revealing issue was greater than minor because it involved a procedural inadequacy that resulted in automatic actuations of equipment related to the mitigating system cornerstone. The issue had very low safety significance because the unit was shutdown and no safety limits were affected. (Section 1R14)

B. Licensee-Identified Violations

None.

REPORT DETAILS

Summary of Plant Status

The plant operated at full power from the beginning of the inspection period until the evening of April 25, when the plant was shut down to begin refueling outage 11. The plant remained shut down until the post-outage reactor startup on the afternoon of May 18. Shortly after startup, the plant automatically tripped from approximately 28 percent power due to a turbine trip. The plant was restarted on May 19, and was manually tripped the next day, May 20, from approximately 15 percent power, following a loss of feedwater flow due to the trip of an operating condensate booster pump. The plant was restarted on May 20, and returned to full power on May 23. On June 7, plant power was reduced to approximately 30 percent, to enable the staff to troubleshoot main condenser problems and returned to full power on June 8. The unit was manually tripped on June 14 when the B main feedwater pump tripped. The plant was returned to full power on June 16, and operated at full power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection

a. Inspection Scope

On May 10, after a tornado watch was declared for the site and prior to the onset of severe weather, the inspectors reviewed actions taken by the licensee in accordance with procedure AP-300, Adverse Weather, to ensure that the adverse weather would neither initiate a plant event nor prevent any system, structure, or component from performing its design function.

On June 12, after a severe thunderstorm warning was declared for the site and prior to the onset of severe weather, the inspectors reviewed actions taken by the licensee in accordance with procedure AP-301, Seasonal Weather Preparations and Monitoring, section 5.2, to ensure that the adverse weather would neither initiate a plant event nor prevent any system, structure, or component from performing its design function.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

a. Inspection Scope

Partial System Walkdowns:

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

- A charging/safety injection pump after that pump returned to service on April 8 following modification activities on April 8
- A & B motor-driven auxiliary feedwater trains, while the turbine-driven train was out of service on April 15
- A residual heat removal system with B residual heat removal system out of service on June 18
- B emergency diesel generator, while the A emergency diesel generator was out of service on June 26

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

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

For the six 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 Final Safety Analysis Report (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 areas in the FSAR. Documents reviewed are listed in the Attachment.

The inspected areas and their corresponding fire zone designations included:

- Switchgear room A (1-A SWGRA)
- Switchgear room B (1-A-SWGRB)
- Main control room (12-A-CR)
- 261' north aisle way motor control centers (1-A-4-COME,1A-35SA & 1A-35SB)
- 261' chiller area (1-A-4-CHLR)
- Containment building (1-C)

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance

.1 Routine Inspection

a. Inspection Scope

During the refueling outage, the inspectors observed the inspection of the A component cooling water heat exchanger and reviewed pictures taken during the inspection of the A emergency services chilled water heat exchanger, to verify that inspection results were appropriately categorized against the pre-established acceptance criteria described in procedure EPT-163, Generic Letter 89-13 Inspections (Raw Water Systems and Local Area Air Handler Inspection and Documentation). The inspectors also verified that the frequency of inspection was sufficient to detect degradation prior to loss of heat removal capability below design basis values.

The inspectors reviewed the corrective actions for AR 50768, initiated during the last refueling outage due to the discovery of significant heat exchanger fouling, to determine whether those actions were effective.

b. Findings

No findings of significance were identified.

.2 Biennial Inspection

a. Inspection Scope

The inspectors selected three types of risk important heat exchangers (HX) and critical systems' components to inspect. Items evaluated were: charging / safety injection pump (CSIP) motor and speed changer coolers; component cooling to service water heat exchanger; and emergency diesel generator HX.

The purpose of the inspection was to verify that: selected heat exchanger test methodology was consistent with accepted industry standards (Electric Power Research Institute Service Water Heat Exchanger Testing Guidelines, TR-107397) or equivalent (NRC Generic Letter 89-13, Service Water System Problems Affecting Safety-Related Equipment); test conditions were appropriately considered; test or inspection criteria were appropriate and met; test frequency was appropriate; as-found results were appropriately dispositioned such that the final condition was acceptable; and, test results considered test instrument inaccuracies and differences. The inspectors walked down: the CSIPs, component cooling HXs, reactor building chillers, and associated valves and piping.

The inspectors reviewed potential common cause problems associated with service water related components and associated repair activities. The problems reviewed included manganese oxide buildup in the internals of the HXs, foreign material in service water, and common instrumentation issues. Major documents reviewed during the inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection

.1 Inservice Inspection (ISI) Activities

a. Inspection Scope

The inspectors reviewed ISI procedures and selected ISI records documenting completed ISI work activities. Nondestructive examination (NDE) procedures reviewed were those covering visual inspection of pipe supports, visual inspection and functional testing of snubbers, performance of liquid penetrant (PT) exams, and performance of magnetic particle (MT) exams. Documents reviewed are listed in the Attachment. Records reviewed were those documenting ISI performed on the following:

- MT inspections performed on steam generator blowdown system piping welds in proximity of containment penetrations to meet commitments in FSAR Section 6.6.8.e, and on the pressurizer.
- PT inspections performed on reactor coolant system, containment spray system, and chemical and volume control system piping welds.
- Visual inspections performed on pipe supports.
- Visual inspections and functional testing performed on snubbers.
- Inspections performed on bolts in upper and lower lateral supports for the steam generators. One hundred percent of the bolts were examined due to fact that loose bolts were identified in the original sample inspected. This was documented in Action Request (AR) 93097.
- AR numbers 92843, 92878, and 93420 which document and disposition mechanical snubber functional test failures.

Qualification and certification records for examiners, and equipment for the above NDE activities were also reviewed. The procedures and records were compared to the requirements in the Technical Specifications (TS) and ASME Boiler and Pressure Vessel Code, Section XI, 1989 Edition, No Addenda. The inspectors reviewed the licensee’s actions as documented in their corrective action program under AR 83058 to address RIS 2003-01, Examination of Dissimilar Metal Welds, Supplement 10 to Appendix VIII of Section XI of the ASME Code.

The inspectors also reviewed Westinghouse Report titled Preliminary Eddy Current Report, RFO 11, dated May 8, 2003, which summarized the examination program and results of the eddy current (ET) inspection performed on the new steam generators installed during RFO 10 after one cycle of operation. The exam methods were compared to the requirements specified in the TS, License Amendments, and applicable industry established performance criteria.

b. Findings

No findings of significance were identified.

.2 Containment Vessel Inspection

a. Inspection Scope

The inspectors walked down the containment building and examined the containment liner. Features examined included the condition of the protective coatings on the liner and the condition of the concrete expansion joint seal at the interface of the liner and interior concrete floor slab. The observations were compared to the requirements specified in the TS, ASME Boiler and Pressure Vessel Code, Article IWE of Section XI, 1992 Edition and 1992 Addenda, and 10 CFR 50.55a.

b. Findings

No findings of significance were identified.

.3 Boric Acid Inspection Program

a. Inspection Scope

The inspectors reviewed licensee procedures which are performed before and after outages to identify boric acid leakage onto various components, evaluate the cause of the leakage, and evaluate the effects of leakage on components. Procedures reviewed were as follows:

- PLP-600, Boric Acid Corrosion Program, Rev. 11
- Work Control Procedure WCM -002, Attachment 6, Process Sheet for Leaks in Borated Systems

The inspectors also reviewed a corporate self-assessment of the boric acid control program which was performed in August, 2002 to examine the effectiveness of the program to address NRC bulletins and generic letters and industry initiatives, and to assure their program will identify and correct boric acid leakage to minimize potential for corrosion damage to SSCs. The inspectors reviewed self-assessment findings to verify that weaknesses and issues identified by the self-assessment were being addressed through the licensee's corrective action program.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

.1 Routine Inspection

a. Inspection Scope

On June 10 and 11, 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 Guides EOP-SIM-17.106 and DSS-017**.** This training tested the operators' ability to deal with feedwater control problems, steam generator tube rupture, emergency classification, and containment isolation problems. The inspectors focused on clarity and formality of communication, 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 identified deficiencies and discrepancies that occurred during the simulator training.

b. Findings

No findings of significance were identified.

.2 Biennial Inspection

a. Inspection Scope

During the week of June 10, 2003, the inspector reviewed documentation, interviewed licensee personnel, and observed the performance of scenario validations with the licensee's operator requalification program. Each of the activities performed by the inspector assessed the effectiveness of the licensee in implementing requalification requirements identified in 10 CFR 55.46, Simulation Facilities. The inspector reviewed and evaluated the adequacy of the licensee's simulation facility for use in operator licensing examinations. Documents reviewed are listed in the Attachment.

The inspector reviewed the simulator test documentation for simulator testing performed as required by ANSI/ANS-3.5, 1998. This included review of transient testing data, core load data, and scenario validation files. The inspector also reviewed the SSRs, Simulator Certification Test Deficiencies Report, and associated procedures.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

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

- Functional failures of the pressurizer power-operated relief valves during refueling outage 10, as described in AR 61057
- Multiple functional failures in the emergency diesel generator starting air system, as described in ARs 63108 and 71959

The inspectors focused on the following:

- 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).

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

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

- Reactor startup and power ascension on May 19 and May 20
- Work for week of June 2 including emergent work for additional cycling of switchyard breakers and a power reduction to 30 percent for condenser water box cleaning.
- Work for the week of June 9, including emergent work associated with repair of a power-supply breaker for the B emergency service water pump.

- Work for the week of June 15, including the reactor startup and power ascension on June 15 and 16.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Nonroutine Plant Evolutions and Events

a. Inspection Scope

During the non-routine evolutions identified below, the inspectors observed plant instruments and operator performance, to verify that the operators performed in accordance with the associated procedures and training.

- Pre-outage plant shutdown on April 25
- Operator recovery from the loss of the A 6.9-kilovolt safety bus on May 11, using procedure AOP-25, Loss of One Emergency AC Bus (6.9 KV) or One Emergency DC Bus (125 V)
- Down power to 30 percent of full power on June 7
- The actuation of auxiliary feedwater during post-trip recovery/prestartup evolutions on June 15

Following the reactor trips identified below, the inspectors observed plant instruments and operator performance, to verify that the operators' response was in accordance with post-trip procedures, plant operating procedures, and related training. Following these trips, the inspectors also observed the post-trip reactor startup and power ascension.

- Automatic trip on May 18
- Manual trip on May 20
- Manual trip on June 14

b. Findings

Introduction

A green non-cited violation of Technical Specification 6.8.1 was identified for a failure to establish adequate general operating procedures for reactor trip recovery and hot standby to minimum load (nuclear startup). The general operating procedures did not ensure that the main feedwater regulating valves were shut or isolated prior to shutting the reactor trip breakers. The result was the over-feeding of the A and C steam generators and the automatic actuation of the auxiliary feedwater system.

Description

On June 14, following a manual reactor trip due to the loss of a main feedwater pump, the feedwater system was not realigned to a normal-shutdown configuration. The operators completed emergency operating procedure EPP-4, Reactor Trip Response, and began restoration of the secondary systems per procedure GP-6, Normal Plant Shutdown From Power Operation To Hot Standby (Mode 1 to Mode 3). At the time of the trip, the main feedwater regulating valves were operating in their automatic-control configuration. Following the trip, the valves were shut, because their automatic-control system was receiving a shut signal from the reactor trip P-4 permissive (average reactor coolant temperature) below the setpoint. When the operators closed the reactor trip breakers in preparation for the startup on June 15, the P-4 signal cleared, the shut signal was removed, and the A and C main feedwater regulating valves opened. Opening these valves allowed the operating feed pump to overfill the A and C steam generators. When level in these steam generators reached the high-level set point, the P-14 permissive caused a trip of the operating feed pump, a trip of the main turbine (while the turbine was not operating), and isolation of the feedwater line. The trip of the last operating feed pump also resulted in an automatic start of the motor-driven auxiliary feedwater pumps. Subsequently, in restoring the system, the operators attempted to restart the main feedwater pump while the high steam generator level permissive conditions were still met, and the main feed pump tripped again. Shortly afterwards the operators were able to restart a main feedwater pump and secure the auxiliary feedwater pumps.

Analysis

The self-revealing issue was greater than minor because it involved a procedure inadequacy that resulted in automatic actuations of equipment in a mitigating system. The issue had very low safety significance because the unit was shutdown and no safety limits were affected.

Enforcement

Technical Specification 6.8.1 required that written procedures be established, implemented and maintained covering the procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978. That appendix recommends General Operating Procedures for Recovery from Reactor Trip (2c) and Hot Standby to Minimum Load (nuclear startup) (2b). Contrary to the above, adequate general operating procedures were not established for reactor trip recovery and hot standby to minimum load (nuclear startup), in that the licensee’s procedures for recovery from reactor trip did not ensure that the feedwater regulating valves were placed in manual and isolated after a reactor trip. Because this failure to establish adequate general operating procedures is of very low safety significance and has been entered into the corrective action program (AR 96171), this violation is being treated as a non-cited violation (NCV), consistent with Section VI.A of the NRC Enforcement Policy: NCV05000400/2003003-01, Failure to establish adequate general operating procedures for reactor trip recovery and hot standby to minimum load (nuclear startup).

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed all five of the operability determinations the licensee generated that warranted selection on the basis of risk insights. 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. In addition, the inspectors compared the justification 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 that the subject component or system remained available, such that no unrecognized increase in risk occurred. The operability determinations reviewed by the inspectors are in the licensee's response to the following ARs:

- AR 85649, B emergency diesel generator tachometer seal
- AR 90589, Emergency Service Water piping wall thickness less than manufacturer's minimum
- AR 92530, Emergency Service Water piping through wall leak on B CSIP oil cooler return line upstream of valve 1SW-172
- ARs 94231 and 94270, Nonconforming potentiometers installed in power range instrumentation amplifiers for N41, N42, N43, and N44
- AR 93739, Containment fan cooler AH-3 (1A-SA and 1B-SB) motor frames coated with resin instead of containment qualified protective coating.

b. Findings

No findings of significance were identified.

1R16 Operator Work-Arounds

a. Inspection Scope

The inspectors reviewed operator work-around 285 (safety injection cold leg accumulator pressure range too narrow, causing an excessive number of operator adjustments), to verify that this work-around did not affect either the functional capability of the related system in responding to an initiating event, or the operators' ability to implement abnormal or emergency operating procedures.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed the portions of Engineering Change (EC) 48993 (which installed pump suction and discharge vents for the A, B, and C charging/safety injection pumps as described in ECs 50958, 50959, and 50960) to verify that:

- the 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.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing

a. Inspection Scope

For the six 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 function(s) described in the FSAR and TS. Documents reviewed are listed in the Attachment. The tests included the following:

- OST-1103, Component Cooling Water ISI Valve Test, Refueling Interval, Mode 5 and 6; following maintenance on motor operated valves 1CC-207, 1CC-208, 1CC-299
- OST-1865, CVCS/SI System Operability Cold Shutdown/Refueling Interval, Mode 4-5-6 or Defueled; following maintenance on motor operated valves 1CS-472, 1SI-86, 1CS-341, 1CS-382, 1CS-423, 1CS-470, 1SI-52, 1SI-238, 1SI-107, 1SI-165, 1SI-291
- OST -1311, Auxiliary Feedwater Valves Remote Position Indication Test 2 year Interval modes 4-6; following maintenance on motor operated valves 1MS-70
- OST-1214, Emergency Service Water System Operability Train A Quarterly Interval Modes 1-2-3-4; following maintenance on motor operated valves 1SW-124, 1SW-121,1SW-231, 1SW-221, 1SW-224
- OST-1411, Auxiliary Feedwater Pump 1X-SAB Operability Test Quarterly Interval Mode 1,2,3; following maintenance on motor operated valves 1AF-143, 1AF-137, 1AF-149, 1MS-72
- OST-1085, 1A-SA Diesel Generator Operability Test Semiannual Interval Modes 1-6; following repair of the voltage regulator

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

The inspectors evaluated the outage activities as described below, to verify that the licensee considered risk in developing outage schedules, adhered to administrative risk-reduction methodologies to control plant configuration, adhered to operating license and TS requirements that maintained defense-in-depth, and developed mitigation strategies for losses of the following key safety functions:

- decay heat removal
- inventory control
- power availability
- reactivity control
- containment

.1 Review of Outage Plan

a. Inspection Scope

Prior to the outage, the inspectors reviewed the licensee's outage risk control plan to verify that the licensee had performed adequate risk assessments, and had implemented appropriate risk management strategies as required by 10CFR50.65(a)(4).

b. Findings

No findings of significance were identified.

.2 Monitoring of Shutdown Activities

a. Inspection Scope

The inspectors observed plant shutdown and portions of the cooldown process on April 26 to verify that TS cooldown restrictions were followed.

b. Findings

No findings of significance were identified.

.3 Licensee Control of Outage Activities

a. Inspection Scope

The inspectors observed the items or activities described below, to verify that the licensee maintained defense-in-depth commensurate with the outage risk control plan for key safety functions and applicable TS when taking equipment out of service. The inspectors reviewed the licensee's responses to emergent work and unexpected conditions, to verify that resulting configuration changes were controlled in accordance with the outage risk control plan, and to verify that control room operators were kept cognizant of plant configuration. Documents reviewed are listed in the Attachment.

- Clearance Activities
- Reactor Coolant System Instrumentation
- Electrical Power
- Decay Heat Removal (DHR)
- Spent Fuel Pool Cooling
- Inventory Control
- Reactivity Control
- Containment Closure

b. Findings

An event that included a loss of decay heat removal occurred on April 28, and was addressed in special inspection 50-400/2003-08. No other findings of significance were identified.

.4 Reduced Inventory and Mid-Loop Conditions

a. Inspection Scope

The inspectors reviewed the licensee's commitments from Generic Letter 88-17, and confirmed by sampling that those commitments are still in place and adequate. Periodically, during the reduced inventory and mid-loop conditions, the inspectors reviewed system lineups to verify that the configuration of the plant systems were in accordance with those commitments. During mid-loop operations, the inspectors observed operator activities, to verify that unexpected conditions or emergent activities did not degrade the operators' ability to maintain required key safety functions.

b. Findings

No findings of significance were identified.

.5 Refueling Activities

a. Inspection Scope

The inspectors observed fuel handling operations during core off-load and reload and other ongoing activities, to verify that those operations and activities were being performed in accordance with technical specifications and approved procedures. Also, the inspectors observed refueling activities, including activities associated with handling new fuel, to verify that the locations of fuel assemblies was tracked from core offload through core reload.

b. Findings

No findings of significance were identified.

.6 Monitoring of Heatup and Startup Activities

a. Inspection Scope

Prior to mode changes and on a sampling basis, the inspectors reviewed system lineups and/or control board indications to verify that TSs, license conditions, and other requirements, commitments, and administrative procedure prerequisites for mode changes were met prior to changing modes or plant configurations. Also, the inspectors periodically reviewed RCS boundary leakage data, and observed the setting of containment integrity, to verify that the RCS and containment boundaries were in place and had integrity when necessary. Prior to reactor startup, the inspectors walked down containment to verify that debris has not been left which could affect performance of the containment sumps. The inspectors reviewed reactor physics testing results to verify that core operating-limit parameters were consistent with the design.

b. Findings

No findings of significance were identified.

.7 Identification and Resolution of Problems.

a. Inspection Scope

Periodically, the inspectors reviewed the outage-related items that had been entered into the licensee's corrective action program, to verify that the licensee had identified problems related to outage activities at an appropriate threshold and had entered them into the corrective action program.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

For the eight surveillance tests identified below, the inspectors witnessed testing and/or reviewed the test data, to verify that the systems, structures, and components involved in these tests satisfied the requirements described in the Technical Specifications, the FSAR, and applicable licensee procedures, and that the tests demonstrated that the SSCs were capable of performing their intended safety functions.

- OST-1829, Motor Driven Auxiliary Feedwater Pumps Actuation Signal on a Trip of Both Main Feedwater Pumps and Time Response Testing of Slave Relay K640, 18 Month Interval
- OST-1859, Remote Shutdown System Operability - Bus Drops Train B, 18 Month Interval
- OST-1824, 1B-SB Emergency Diesel Generator Operability Test 18 Month Interval Modes 1 Through 6 and Defueled

- OST-1858, Remote Shutdown System Operability - Bus Drops Train A 18 Month Interval Modes 5, 6, or Defueled
- OST-1812, Auxiliary Feedwater Isolation: ESF Response Time 18 Month Interval Modes 4-6
- OST-1823, 1A-SA Emergency Diesel Generator Operability Test 18 Month Interval Modes 1 Through 6 and Defueled
- OST-1826, Safety Injection: ESF Response Time, Train B 18 Month Interval on a Staggered Test Basis Mode 5-6 1MS-72
- OST-1085*, 1A-SA Diesel Generator Operability Test Semiannual Interval Modes 1-6

*This procedure included inservice testing requirements.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety (OS) and Public Radiation Safety (PS)

2OS1 Access Control To Radiologically Significant Areas

.1 Access Controls

a. Inspection Scope

During the weeks of April 28 - May 2 and May 12 - 16, the inspectors evaluated licensee guidance for access controls to radiologically significant areas. Selected procedural details for posting, surveying, and access controls to airborne radioactivity, radiation, high radiation area (HRA), locked high radiation area (LHRA) and very high radiation area (VHRA) locations were reviewed and discussed with licensee representatives. The inspectors evaluated nine Radiation Work Permits (RWPs) used for work in radiologically significant areas associated with the Refueling Outage 11 (RFO-11). The selected RWPs were evaluated for adequacy of access controls and specified electronic dosimeter (ED) alarm setpoints against expected work area dose rates and work conditions. Access control procedures for posted LHRA and VHRA locations were reviewed and discussed with radiation protection management, supervision, and technicians.

During facility tours, the inspectors evaluated radiological postings, barricades, and surveys associated with radioactive material storage areas and radiologically significant areas within the reactor, auxiliary, and spent fuel pool buildings. Dose rate measurements were conducted at various building locations, including the SFP area and compared to current radiation survey map data. In addition, the inspectors independently assessed implementation LHRA controls. The inspectors reviewed licensee implementation of inspections to verify the condition of the locked doors and assessed LHRA and VHRA key controls.

During the inspection, the inspectors evaluated the proficiency and knowledge of the radiation workers and radiation protection staff in communicating and applying radiological controls for selected tasks. The inspectors attended a briefing for work activities associated with RWP 770, Reactor Head Bare Metal Inspection. During selected job site reviews and tours within the licensee's radiological control area, the inspectors observed and evaluated radiological worker and radiation protection technician training/skill levels, procedural adherence, and implementation of RWP specified access controls. The inspectors also interviewed management personnel regarding radiological controls associated with reactor head inspection activities.

Radiation protection activities were evaluated against Updated Final Safety Analysis Report (UFSAR) Section 12, Radiation Protection; 10 CFR 19.12; 10 CFR 20, Subparts B, C, F, G, H, and J; and approved procedures. The procedures and records reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

.2 Problem Identification and Resolution

a. Inspection Scope

The inspectors reviewed licensee Corrective Action Program (CAP) Action Request (AR) documents associated with access controls to radiologically significant areas. Eight ARs documented in the Attachment were reviewed and evaluated in detail. The inspectors assessed the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with procedure CAP-NGGC-0200, Corrective Action Program, Revision 7.

b. Findings

No findings of significance were identified.

2OS2 As Low As Reasonably Achievable (ALARA) Planning and Controls

.1 ALARA Planning and Controls

a. Inspection Scope

During the weeks of April 28-May 2 and May 12-16, the inspectors observed and evaluated implementation of the licensee's ALARA program for the RFO-11 outage. The inspectors reviewed ALARA planning, dose estimates, and prescribed ALARA controls for the five outage work tasks expected to incur the maximum collective exposures. Reviewed activities included installation and removal of temporary lead shielding, snubber removal, reactor head inspection, and incore instrument work. Incorporation of planning, established work controls, expected dose rates and dose expenditure into the ALARA pre-job briefings and RWPs for those activities also were reviewed. The inspectors also independently verified that selected jobsite dose rates were consistent with the dose rates recorded on pre-job survey maps for selected containment and auxiliary building work areas and equipment.

Selected elements of the licensee's source term reduction and control program were examined to evaluate the effectiveness of the program in supporting implementation of the ALARA program goals. Reviewed areas included primary chemistry shutdown controls, radiation field monitoring and trending, and temporary shielding.

Trends in individual and collective personnel exposures at the facility were reviewed. Records of year-to-date individual radiation exposures sorted by work groups were examined for significant variations of exposures among workers. Exposure tracking during the RFO-11 outage, and records of exposures to declared pregnant workers incurred from December 2001 through May 2003 as well as associated guidance for controlling such exposures, were also reviewed. Trends in the plant three-year rolling average collective exposure history, outage, non-outage and total annual doses for 1992 through 2001 were reviewed and discussed with licensee representatives.

The licensee's ALARA program implementation and practices were evaluated for consistency with UFSAR Chapter 12, Sections 1-5, Radiation Protection; 10 CFR Part 20 requirements; Regulatory Guide 8.29, Instruction Concerning Risks from Occupational Radiation Exposure, February 1996; and procedural guidance documented in the Attachment.

b. Findings

No findings of significance were identified.

.2 Problem Identification and Resolution

a. Inspection Scope

The inspectors reviewed Action Request (AR) documents and audits related to the ALARA program. The inspectors assessed the licensee's ability to identify, characterize, prioritize, and resolve the identified issues in accordance with CAP-NGGC-0200, Corrective Action Program, Revision 7. Reviewed documents are listed in the Attachment.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

.1 Initiating Events Cornerstone

a. Inspection Scope

For the performance indicators (PIs) listed below, the inspectors sampled licensee submittals for the 2nd through 4th quarters of 2002 and the 1st quarter of 2003. To verify the accuracy of the PI data reported during that period, the inspectors compared the licensee's basis in reporting each data element to the PI definitions and guidance contained in NEI 99-02, Regulatory Assessment Indicator Guideline, Rev. 2.

- Unplanned Scrams per 7000 Critical Hours
- Scrams with Loss of Heat Removal
- Unplanned Power Changes per 7000 Critical Hours

The inspectors reviewed a selection of licensee event reports, operator log entries, daily reports (including the daily CR descriptions), monthly operating reports, and PI data sheets to verify that the licensee had adequately identified the number of scrams and unplanned power changes greater than 20 percent that occurred during the previous four quarters. The inspectors compared this number to the number reported for the PI during the current quarter. The inspectors also reviewed the accuracy of the number of critical hours reported and the licensee's basis for crediting normal heat removal capability for each of the reported reactor scrams. In addition, the inspectors interviewed licensee personnel associated with the PI data collection, evaluation, and distribution.

b. Findings

No findings of significance were identified.

.2 Barrier Integrity Cornerstone

a. Inspection Scope

For the performance indicators (PIs) listed below, the inspectors sampled licensee submittals for the 2nd through 4th quarters of 2002 and the 1st quarter of 2003. To verify the accuracy of the PI data reported during that period, the inspectors compared the licensee's basis in reporting each data element to the PI definitions and guidance contained in NEI 99-02, Regulatory Assessment Indicator Guideline, Rev. 2.

- For the Reactor Coolant System Specific Activity Performance Indicator, the inspectors observed licensee sampling and analysis of reactor coolant system samples, and compared the licensee-reported performance indicator data with records developed by the licensee while analyzing previous samples.

- For the Reactor Coolant System Leak Rate Performance Indicator, the inspectors observed gathering of leak-rate data, and reviewed records of daily measures of RCS identified leakage.

b. Findings

No findings of significance were identified.

.3 Occupational Radiation Safety Cornerstone

a. Inspection Scope

The inspectors reviewed the licensee's records and data generated during Calendar Year (CY) 2002 for the Occupational Exposure Control Effectiveness Performance Indicator (PI). The information reviewed included data reported to the NRC, pertinent corrective action program issues, and selected radiological control program records. The inspectors assessed the licensee's CY 2002 monthly reviews for PI occurrences which were performed pursuant to procedure REG-NGGC-0009, NRC Performance Indicators, Rev. 1. The licensee's disposition of the reviewed issues was evaluated against NEI 99-02, Regulatory Assessment Performance Indicator Guideline, Rev. 2. Specific procedures, records, and Action Request (AR) documents reviewed and evaluated for this PI are listed in the Attachment.

b. Findings

No findings of significance were identified.

.4 Public Radiation Safety Performance Indicator

a. Inspection Scope

The inspectors reviewed the licensee's records and data generated during CY 2002 for the Radiological Effluent Control PI. The information reviewed included data reported to the NRC, pertinent corrective action program issues, and selected radiological effluent control program records. The licensee's disposition of the reviewed issues was evaluated against NEI 99-02, Regulatory Assessment Performance Indicator Guideline, Rev. 2. Specific procedures, records, and ARs reviewed and evaluated for this PI are listed the Attachment.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

.1 Annual Sample Review

a. Inspection Scope

The inspectors selected AR 62940 for detailed review. This AR was associated with the installation of an incorrect part during corrective maintenance on the B emergency diesel generator. The inspectors reviewed this report to verify that the licensee identified the full extent of the issue, performed an appropriate evaluation, and specified and prioritized appropriate corrective actions. The inspectors evaluated the report against the requirements of the licensee's corrective action program as delineated in corporate procedure CAP-NGGC-0200, Corrective Action Program, and 10 CFR 50, Appendix B.

b. Observations and Findings

No findings of significance were identified.

4OA3 Event Follow-up

a. Inspection Scope

Following a loss of shutdown cooling on April 28, the inspectors reviewed plant data and plant records, and interviewed plant operators to determine whether those circumstances constituted a significant operational event as defined in NRC Management Directive 8.3, NRC Incident Investigation Program.

Also for the events described below, the inspectors reviewed the licensee's actions, observed plant parameters for mitigating systems and fission product barriers, evaluated performance of systems and operators, and confirmed proper classification and reporting of the event:

- The automatic reactor trip that occurred on May 18 from 28 percent power due to a turbine trip.

- The manual reactor trip that occurred on May 20 from 15 percent power due to a condensate booster pump trip.

- The reactor trip that occurred on June 14 from 100 percent power due to loss of the B main feedwater pump, including low steam generator levels in the A and C steam generators, which resulted in automatic starting of the motor and turbine-driven auxiliary feedwater pumps.

- The engineered-safeguards-feature actuation of the motor-driven auxiliary feedwater (AFW) pumps that occurred on June 15 while the plant was shut down, due to a high steam generator water level trip of the A main feedwater pump.

b. Findings

The April 28 loss of shutdown cooling event is addressed in NRC Inspection Report 50-400/03-08. The June 15 actuation of auxiliary feedwater is addressed in section 1R14 of this report. With respect to the other events, no findings of significance were identified.

4OA5 Other Activities

.1 (Open) NRC Temporary Instruction 2515/150, Reactor Pressure Vessel Head and Head Penetration Nozzles (NRC Order EA-03-009)

a. Inspection Scope

The inspectors observed activities relative to inspection of the reactor vessel head penetration (VHP) nozzles in response to NRC Bulletins 2001-01, 2002-01, 2002-02 and NRC Order Modifying Licenses dated February 11, 2003. The inspection included review of procedures, assessment personnel training and qualification, and observation and assessment of visual (VT) examinations. Discussions were also held with cognizant licensee personnel. The activities were examined to verify licensee compliance with regulatory requirements and gather information to help the NRC staff identify possible further regulatory positions and generic communications. Specifically, the inspectors reviewed or observed the following:

(1) VT inspection using remote video of approximately 25 percent of the VHP Nozzles and partial inspection (at least one quadrant) of approximately 20 penetrations;

(2) VT inspection of the majority of the outer head surface.

Additionally, the inspectors reviewed the susceptibility ranking calculation, including the basis for head temperature input, and verified appropriate plant specific information was used in the time-at-temperature model for determining RPV head susceptibility ranking.

b. Observations and Findings

1) Verification that the examinations were performed by qualified and knowledgeable personnel.

The inspectors found that visual inspections were being performed in accordance with approved procedures with trained and qualified inspection personnel.

2) Verification that the examinations were performed in accordance with approved procedures.

The bare head remote visual inspection was performed in accordance with procedure EPT-859, “100% Bare Metal Visual Examination of the Reactor Pressure Vessel Head.” The procedure used a sled mounted camera that required numerous passes for each of the 66 nozzles. The entire bare metal surface and circumference of each VHP nozzle was covered with these passes.

Non-destructive examinations of the VHP nozzles were not performed. NRC Order EA-03-009 did not require these inspections to be performed during the May 2003 refueling outage as the plant was in the low PWSCC susceptibility category. These inspections are required to be performed within five years of February 11, 2003. This TI will remain open to assess the licensee's performance of non-destructive examinations of the penetrations during a future refueling outage.

3) Evaluate condition of the reactor vessel head (debris, insulation, dirt, boron from other sources, physical layout, viewing obstructions).

The inspectors noted that no significant examples of insulation, leakage sources, debris, or dirt, impeded the examination. The licensee was able to view 100 % of each of the 65 large nozzles and the reactor head vent nozzle during the visual examinations. Any minor debris masking the penetrations was removed via brushing or through the use of a vacuum prior to final disposition of the penetration.

4) Evaluate ability for small boron deposits, as described in NRC Bulletin 2001-01, to be identified and characterized.

The inspectors observed that the resolution of the video camera was good and capable of detecting any debris or small boron deposits on the bare metal head. There were no obstructions to preclude a 100% visual inspection. Minor boron deposits that were noted were in the vicinity of a previously identified conoseal and a canopy seal weld leak. This boron was isotopically analyzed and determined to be at least 4.5 years old which corresponds to previously identified and repaired leaks on a conoseal and a canopy seal weld. No "popcorn-like" deposits were seen at the head/VHP interface on any penetrations.

5) Determine extent of material deficiencies (associated with the concerns identified in the three bulletins) which were identified that required repair.

No examples of VHP leakage or material deficiencies were identified during the visual examinations.

6) Determine any significant items that could impede effective examinations.

No significant items that could impede the examination process were noted during observation of the visual examinations.

7) Determine the basis for the temperatures used in the susceptibility calculation.

The inspectors reviewed the susceptibility ranking for the plant which included a review of the EDY calculation, HNP-M/MECH-1091, "Effective Degradation Years for the Reactor Vessel Head." The calculation determined that the Harris reactor head had an EDY 2.07 years, placing the plant in the low susceptibility region. The inspectors questioned the basis for using cold leg temperature as the 100 percent power head temperature. The licensee provided the inspector correspondence from the NSSS vendor which supported the basis for this conclusion. The NSSS vendor demonstrated, analytically and through measured upper head fluid temperatures, that plants similar to Harris would have head temperatures at the cold leg temperature with the unit at 100 percent power. The inspectors concluded that the licensee had properly determined the susceptibility ranking of the plant.

8) Verification that visual inspections of the pressure-retaining components above the RPV head were performed.

The inspectors reviewed the visual examination data sheet documenting the visual inspection of the pressure-retaining components above the RPV head performed in accordance with CM-M0070. No evidence of leakage was identified.

4OA6 Meetings, Including Exit

On June 26, 2003, the resident inspectors presented the inspection results to Mr. Jim Scarola and other members of his staff. The inspectors confirmed that proprietary information was neither provided nor examined during the inspection.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

J. Briggs, HNP, Superintendent - Environmental and Chemical
J. Caves, Supervisor - Licensing/Regulatory Programs
F. Diya, Superintendent - Systems Engineering
R. Duncan, Director - Site Operations
W. Gurganious, Manager - Nuclear Assessment
R. Hill, Acting Supervisor - Licensing/Regulatory Programs
T. Hobbs, Operations Manager
A. Khanpour, Manager - Harris Engineering
S. Larson, Quality Control
E. McCartney, Training Manger
G. Miller, Maintenance Manager
T. Morton, Manager - Support Services
T. Natale, Manager - Outage and Scheduling
T. Pilo, Supervisor - Emergency Preparedness
R. Pinella, Boron Corrosion Engineer
K. Rogers, Sr. Specialist, ALARA
J. Scarola, Vice President Harris Plant
G. Simmons, Superintendent - Radiation Control
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
None

Opened and Closed

05000400/2003003-01	NCV	Failure to establish adequate general operating procedures for reactor trip recovery and hot standby to minimum load (nuclear startup) (Section 1R14)

Closed
None

Discussed

TI 2515/150	TI	Reactor Pressure Vessel Head and Head Penetration Nozzles (NRC Order EA-03-009) (Section 4OA5)

LIST OF DOCUMENTS REVIEWED

Section 1R04: Equipment Alignment

A Charging/Safety Injection Pump:
- Procedure OP-107, Chemical and Volume Control System
- Drawing 2165-S-1304, Simplified Flow Diagram Chemical and Volume Control System, sheet 2
- Drawing 2165-S-1305, Simplified Flow Diagram Chemical and Volume Control System

A & B Motor-Driven Auxiliary Feedwater trains:
- Procedure OP-137, Auxiliary Feedwater System, Revision 22
- Drawing 2165-S-0544, Simplified Flow Diagram Feedwater System, Revision 36

A Residual Heat Removal system:
- Procedure OP-111, Residual Heat Removal System, Revision 23
- Drawing 2165-S-1324, Simplified Flow Diagram Residual Heat Removal System, Revision 11

B Emergency Diesel Generator:
- Procedure OP-155, Diesel Generator Emergency Power System, Revision 30
- Drawing 2165-S-0563, Simplified Flow Diagram Diesel Fuel Oil System Unit 1, Revision 8
- Drawing 2165-S-0633S01, Simplified Flow Diagram Emergency Diesel Generator Lube Oil and Air Intake & Exhaust System - Unit 1, Revision 10
- Drawing 2165-S-0633S02, Simplified Flow Diagram Emergency Diesel Generator 1A-SA & 1B-SB Jacket Water System Unit 1, Revision 10
- Drawing 2165-S-0633S03, Simplified Flow Diagram Emergency Diesel Generator 1A-SA & 1B-SB Fuel Oil and Drainage Systems Unit 1, Revision 7
- Drawing 2165-S-0633S04, Simplified Flow Diagram Emergency Diesel Generator 1A-SA & 1B-SB, Starting Air System Unit 1

Section 1R05: Fire Protection

Results from recent completions of the following procedures:

- FPT-3205, “Fire Detector Functional Test Local Fire Detector Panel 5 12 Month Interval”
- FPT-3206, “Fire Detector Functional Test Local Fire Detector Panel 6 12 Month interval”
- FPT-3207, “Fire Detector Functional Test Local Fire Detector Panel 7 12 Month Interval”
- FPT-3151, “Fire Extinguisher Inspection: Auxiliary Building Monthly Interval”
- OPT-3010, “Fire Hose Service Test Various Intervals Modes All”
- FPT-3425, “Fire Damper Inspection 18 Month Interval RAB 286 Elevation”
- FPT-3426, “Fire Damper Inspection 18 Month Interval RAB 236 Elevation and 261 Elevation Modes: All”
- FPT-3550, “Fire Penetration Seal Visual Inspection 18 Month Interval”

Other documents:

- AR 85811, Operations personnel are not familiar with the prescribed access and egress routes described by the safe shutdown analysis
- AR 85136, During the last completion of the fire door surveillance procedure, relatively many fire doors were identified with deficiencies
- AR 83451, Excessive rework is required to maintain 9-foot-tall fire doors

Section 1R07: Heat Sink Performance

Completed Procedures/Inspections

- EPT-163 Att. 1, CSIP Oil and Gear Coolers Inspection, Rev. 11, completed 10/25/01, 10/26/01
- EPT-163 Att. 1, Containment Fan Coolers Inspection, Rev. 11, completed 12/01/01, 10/31/01,10/05/01
- EPT-163 Att. 1, EDG Jacket Water Coolers Inspection, Rev. 11, completed 10/28/01, 10/01/01
- EPT-163 Att. 1, ESCW Condensers (Chillers) Inspection, Rev. 11, completed 11/01/01,10/05/01
- EPT-163 Att. 1, Component Cooling Water Heat Exchangers Inspection, Rev. 11, completed 11/06/01, 10/03/01
- OST-1214, Emergency Service Water System Operability Train A Quarterly Interval Modes 1-2-3-4, Rev. 28, completed 01/04/03
- OST-1215, Emergency Service Water System Operability Train B Quarterly Interval Modes 1-2-3-4, Rev. 29, completed 01/26/03
- EPT-250, A Train ESW Flow Verification/Balance, Rev. 12, completed 04/08/03, 10/15/02
- EPT-251, B Train ESW Flow Verification/Balance, Rev. 11, completed 06/10/02
- Balance of Plant Heat Exchanger Eddy Current Examination Report for Component Cooling Water Heat Exchanger A & Emergency Service Cooling Water Chillers 1A & 1B - Condenser Section, 04-05/97

Procedures

- EPT-163, Generic Letter 89-13 Inspections (Raw Water Systems and Local Area Air Handler Inspection and Documentation), Rev. 11
- OST-1007, CVCS/SI System Operability Train A Quarterly Interval Modes 1-4, Rev.25
- APP-ALB-006, Main Control Board, Annunciator 8-5, Rev. 15

Engineering Service Requests

- ESR 0100026, Install Temperature Probes on CVCS Pumps, Rev. 5
- ESR 9600548, Service Water Setpoint, Rev. 0
- ESR 9500392, CSIP Pumps without ESW Supply, Rev. 0
- ECR 438, Flow Instrumentation for Service Water to CSIP Coolers, 03/22/02
- ECR 1787, Improvements to Service Water Instrument Sensing Line Taps, 02/03/03
- ECR 2459, Flush Capability for ESCW Instrument Taps, 04/10/03
- EC 49074, RFO10 Generic Letter 89-13 Test/Inspection Evaluation, Rev. 0

System Descriptions

- SD-139, Service Water System, Rev. 14
- SD-145, Component Cooling Water System, Rev. 8

Completed Work Orders

- WO 101764-01, Emergency Service Water Pump B Discharge Header Pressure Instrument LP-P-9101B Calibration, 07/09/01

Calculations

- HNP-M/MECH-1011, Pump Degradation Limits for ESW, CCW and ESCW, Rev. 7

Drawings

- 4617D56, Component Cooling Water Heat Exchanger, Rev. 6

Condition Reports

- AR 50768, Service Water System Performance, 10/31/01
- AR 50611, B EDG Jacket Water Heat Exchanger Fouling, 10/01
- AR 50531, Low Service Water Flows to CSIP Gear and Oil Coolers, 10/01
- AR 55801, Controlotron Ultrasonic Flowmeter Failure to Work with Associated CSIP Gear/Oil Cooler Pipe Lines, 02/14/02
- AR 90817, Service Water System Monitoring Capability, 4/17/03
- AR 91099, EPT-250, 251 Provision to N/A Controlotron Steps, 4/21/03
- AR 90921, Classification Criteria Improvement, 4/21/03

Other Documents

- Response to Action Item 94H0749, Deletion of EPT-174, CCW Heat Exchanger Performance Test, from the Service Water Program, 08/03/98
- Drawing #123P112, Sub #5, Component Cooling Water Heat Exchanger Specification Data Sheets
- Service Water Flows to Individual Components Trending (EDG Jacket Water Cooler, CCW Heat Exchanger, Chiller Condenser, Containment Coolers, CSIP Pump Oil and Gear Coolers), 01/99 - 04/03
- EDG 1B-SB Jacket Water Heat Exchanger Performance Trending, RFO9-RFO10 Resistance Factor Across CCW Heat Exchanger Trending, 2001-2003
- RAB Auxiliary Operator Logs, Rev. 15
- Materials Services Section Technical Report, HNP- Corrosion Evaluation of Normal Service Water Corrosion Coupons, 10/16/02, 02/27/03, 06/24/02, 03/28/02

Section 1R08: Inservice Inspection

Procedures

- Procedure NDEP-0201, Liquid Penetrant Examination, visible dye, solvent removable, Rev 24
- Procedure NDEP-0301, Dry Powder Magnetic Particle Examination, Rev. 14
- Procedure NDEP-0611, VT-1 Visual Examination of Nuclear Power Plant Components, Rev. 14
- Procedure NDEP-0613, VT-3 Visual Examination of Nuclear Power Plant Components, Rev. 18
- Procedure MST-M0033, Mechanical snubber Operational (Functional) Test, Rev. 9
- Procedure EST-201, ASME System Pressure Tests, Rev. 14
- Procedure ISI-202, Safety-Related Component Support (Hangers and Snubbers) Examination and Testing Programs, Rev. 15
- Appendix B to Nuclear NDE Manual NGGM-PM-0011, NDE Surface Examination Criteria, Rev. 3

Section 1R11: Licensed Operator Requalification

- Training Administrative Procedure, TAP-500, Rev. 3
- Reactor Operator and Senior Reactor Operator Initial Training Program, TPP-201, Rev. 2
- Simulator Program, TPP-206, Rev. 3
- Conduct Of Simulator Training And Evaluation, TAP-409, Rev.1
- Simulator Operation and Maintenance, TAP-412, Rev.1
- Development Phase, TAP-300, Rev. 2
- Licensed Operator Continuing Training Program, TPP-306, Rev. 12

- TT-001, Manual Reactor Trip, Rev. 5
- TT-002, One Reactor Coolant Pump Trip, Rev. 5
- TT-003, Simultaneous Closure of All Main Steam Isolation Valves, Rev. 6
- TT-004, Simultaneous Trip of All Reactor Coolant Pumps, Rev. 6
- TT-005, One Reactor Coolant Pump Trip, Rev. 5
- TT-010, Slow RCS Depressurization To Saturation Using PORVs and No SI, Rev. 6

- SST-003, Steady State Accuracy and Stability Test, Rev. 7
- SST-001, Steady State Accuracy and Stability Test, Rev. 7

- Core Validation Against Design Data Sheets

- Scenario Validation Package - EOP-SIM-17.108, Loss of Instrument Air / Reactor Trip
- Scenario Validation Package - EOP-SIM-18.4, Current Plant Challenges

Section 1R12: Maintenance Effectiveness

Action Requests

- 61057, Power-Operated Relief Valve Functional Failures During Refueling Outage 10
- 50708, Diaphragm Leaks on All 3 Pressurizer Power-Operated Relief Valves
- 60144, Maintenance Rule Self-Assessment Issue 1: Several Plant Events Were Not Classified as Functional Failures

- 63108, For Systems 5105 and 5112, Several Events Should be Considered Functional Failures
- 71959, Maintenance Rule Functional Failures on System 5112
- 76011, Maintenance Rule Functional Failure on Emergency Diesel Generator 1A-SSN Air Compressor
- 85618, Emergency Diesel Generator 1A-SSN Air Compressor Maintenance Rule Functional Failure
- 88695, Fuse Blown on 1A-SA Diesel Starting Air Compressor

Work Orders

- 299934-01, Replace diaphragm on valve 1RC-114
- 299933-01, Replace diaphragm on valve 1RC-116
- 299931-01, Replace diaphragm on valve 1RC-118

Other Documents

- Maintenance Rule Scoping and Performance Criteria report for system 2050, dated April 8
- Maintenance Rule Event Log report for system 2050, dated April 8
- Maintenance Rule Performance Summary report for system 2050, dated April 8
- Drawing 2165-S-1301, Simplified Flow Diagram Reactor Coolant System Sheet 2, Revision 9
- Engineering Change 49986, Replacement of Copes-Vulcan [Air Operated Valves] Diaphragms with Silicon Rubber Diaphragms, Revision 0
- System Description SD-100.03, Pressurizer and Controls, Revision 9
- Maintenance Rule Scoping and Performance Criteria report for system 5112, dated May 29
- Maintenance Rule Event Log report for system 5112, dated May 29
- Maintenance Rule Performance Summary report for system 5112, dated May 29
- Drawing 2165-S-644S04, Simplified Flow Diagram Emergency Diesel Generator1A-SA & 1B-SB Starting Air System Unit 1, Revision 19

Section 1R15: Operability Evaluations

- AR 93739, Service life 1 protective coatings not applied to motors on air handler 3
- EC 52617, Evaluate the properties of Dow-Corning varnish to provide assurance that the motor frame can accomplish its design function throughout a design basis event

Section 1R17: Permanent Plant Modifications

- Engineering Change 48993 through revision 3
- Engineering Change 50960 for the A CSIP
- Engineering Change 50958 for the B CSIP
- Engineering Change 50959 for the C CSIP
- NRC Inspection Report 50-400/02-06
- System Description SD-107, Chemical and Volume Control System
- System Description SD-110, Safety Injection
- Design Basis Document DBD-104, Safety Injection
- Operating Procedure OP-107, Chemical and Volume Control System
- Drawing 2165-S-1305, Simplified Flow Diagram Chemical and Volume Control System System

Section 1R19: Post Maintenance Testing

Procedures

- OST-1103, Component Cooling Water ISI Valve Test, Refueling Interval, Mode 5 and 6; for motor operated valves 1CC-207, 1CC-208, 1CC-299,
- OST-1865, CVCS/SI System Operability Cold Shutdown/Refueling Interval, Mode 4-5-6 or Defueled; for motor operated valves 1CS-472, 1SI-86, 1CS-341, 1CS-382, 1CS-423, 1CS-470, 1SI-52, 1SI-238, 1SI-107, 1SI-165, 1SI-291
- OST-1106, CVCS/SI System Operability Quarterly Interval, Mode 4-5-6; for motor operated valves 1SI-287,
- OST -1311, Auxiliary Feedwater Valves Remote Position Indication Test 2 year Interval modes 4-6, for maintenance on motor operated valves 1MS-70,
- OST-1214, Emergency Service Water System Operability Train A Quarterly Interval Modes 1-2-3-4, for maintenance on motor operated valves 1SW-124, 1SW-121,1SW-231, 1SW-221, 1SW-224
- OST-1411, Auxiliary Feedwater Pump 1X-SAB Operability Test Quarterly Interval Mode 1,2,3, for maintenance on motor operated valves 1AF-143, 1AF-137, 1AF-149, 1MS-72
- OST-1085, 1A-SA Diesel Generator Operability Test Semiannual Interval Modes 1-6, for repair of the voltage regulator

Section 1R20: Refueling and Outage Activities

Generic Letter 88-17 Documents

- Procedure AOP-020, “Loss of [Reactor Coolant System] Inventory or Residual Heat Removal While Shutdown,” Revision 24
- Procedure AP-013, “Plant Nuclear Safety Committee,” Revision 22
- Engineering Service Request ESR 9500808, “Removable Equipment Hatch Cover Bolting Requirements,” Revision 0
- Engineering Service Request ESR 9800297, “Containment Closure Procedure,” Revision 0
- Procedure GP-008, “Draining the Reactor Coolant System,” Revision 23
- Contingency Plan HNP-C/CONT-1009, “Containment Building Removable Equipment Hatch,” Revision 0
- Procedure OMP-003, “Outage Shutdown Risk Management,” Revision 16
- Procedure OMP-004, “Control of Plant Activities During Reduced Inventory Conditions,” Revision 8
- Procedure OST-1034, “Containment Penetrations Test Weekly Interval During Core Alterations and Movement of Irradiated Fuel Inside Containment,” Revision 10 and
- Procedure OST-1091, “Containment Closure Test Weekly Interval During Core Alterations and Movement of Irradiated Fuel Inside Containment,” Revision 10

Other Procedures

- GP-001, Reactor Coolant System Fill and Vent Mode 5, Revision 23
- GP-002, Normal Plant Heatup from Cold Solid to Hot Subcritical Mode 5 to Mode 3, Revision 31
- GP-004, Reactor Startup (Mode 3 to Mode 2), Revision 31
- GP-005, Power Operation (Mode 2 to Mode 1), Revision 42
- GP-006, Normal Plant Shutdown from Power Operation to Hot Standby (Mode 1 to

Mode 3), Revision 33
- GP-007, Normal Plant Cooldown Mode 3 to Mode 5, Revision 33
- GP-009, Refueling Cavity Fill, Refueling and Drain of the Refueling Cavity Modes 5-6-5, Revision 41

Refueling Outage Contingency Plans

- Crane Boom over Refueling Water Storage Tank Inventory Control Capability at Risk
- ECCS Vend Mod Reliance on Freeze Seal
- Valve 1CC-129 Replacement Reliance on Freeze Seal
- Containment Closure

Section 1R22: Surveillance Testing

Procedures

- OST-1829, Motor Driven AFW Pumps Actuation Signal on a Trip of Both Main Feedwater Pumps and Time response testing of slave relay K640, 18 Month Interval
- OST-1859, Remote Shutdown System Operability - Bus Drops Train B, 18 Month Interval
- OST-1824, 1B-SB Emergency Diesel Generator Operability Test 18 Month Interval Modes 1 Through 6 and Defueled
- OST-1858, Remote Shutdown System Operability - Bus Drops Train A 18 Month Interval Modes 5, 6, or Defueled
- OST-1812, Auxiliary Feedwater Isolation: ESF Response Time 18 Month Interval Modes 4-6
- OST-1823, 1A-SA Emergency Diesel Generator Operability Test 18 Month Interval Modes 1 Through 6 and Defueled
- OST-1826, Safety Injection: ESF Response Time, Train B 18 Month Interval on a Staggered Test Basis Mode 5-6

Section 2OS1: Access Control To Radiologically Significant Areas

Procedures

- Administrative Procedure (AP)-110, Pre-Job Briefings, Revision (Rev.) 8
- AP-504, Administrative Controls for LHRA and VHRA, Rev. 20
- AP-535, Performing Work in the RCA, Rev. 15
- AP-545, Containment Entries, Rev. 24
- Environmental and Radiation Control (ERC) Procedure -003, Radiation Protection - Conduct of Operations, Rev. 16
- HPP-600, Radiation Work Permits, Rev. 17
- HPP-601, Conduct of Radiation Protection Pre-Job Briefing, Rev. 3
- HPP-625, Performance of Radiological Surveys, Rev. 19
- HPP-627, Radiological Controls for Diving Operations, Rev. 7
- HPP-800, Handling Radioactive Material, Rev. 37
- HPP-880, Spent Nuclear Fuel Shipping & Receipt, Rev. 23
- Plant Program Procedure - 511, Radiation Control & Protection, Rev. 17
- DOS-0002, Dosimetry Issuance, Rev. 18
- HPS-0003, Radiological Posting, Labeling & Contamination, Rev. 8
- HPS-0008, Performing Work in the RCA, Rev. 2
- Preventative Maintenance (PM) - 0002, Radiation Control and Protection Manual,

Rev. 33

Radiation Work Permit (RWP) Documents

- RWP 769, RFO-11 Reactor Headwork/Refueling
- RWP 770, RFO-11 Reactor Head Bare Metal Inspection
- RWP 761, RFO-11 S/G Manway Removal/Installation
- RWP 764, RFO-11 S/G Nozzle Dam Support
- RWP 771, RFO-11 Seal Table
- RWP 758, RFO-11 Inspections

Corrective Action Program Documents

- AR 0091348 Contamination Occurrence In Clean Area
- AR 0091671 Lost DRD Inside Containment
- AR 0091652 Personnel Contamination Occurrence Due to Poor Work Practice
- AR 0091642 Incore Sump Dose Rates Higher than Anticipated
- AR 0091624 HRA Posting/Barricading of RCB 221 Ladders
- AR 0091508 Worker in the RCA was Unaware of their RWP Number
- AR 0091692 RAD Material Vacuum Hose Not Secure
- AR 0091694 Dose Rate Set point Consistency

Records and Data

- Health Physics Day/Night Shift Turnover Logs, April 28, 2003, through May 2, 2003

Section 2OS2: ALARA Planning and Controls

Procedures

- NGGC-ADM-0105, ALARA Planning, Rev. 6
- AP-002, Plant Conduct of Operations, Rev. 38
- AP-110, Pre-Job Briefings, Rev. 8
- AP-400, Conduct of Diving Operations, Rev. 1
- AP-512, Use of Respiratory Protection Equipment, Rev. 23
- AP-530, ALARA, Rev. 7
- AP-535, Performing Work in Radiation Control Areas, Rev. 15
- DOS-NGGC-0003, XE-133 Skin Dose Calculation, Rev. 3
- DOS-NGGC-0004, Administrative Dose Limits, Rev. 7
- HPS-NGGC-0003, Radiological Posting, Labeling And Surveys, Rev. 8
- HPS-NGGC-0014, Total Exposure Manage Radiation Work Permits, Rev. 0
- MNT-NGGC-0003, Radiation Shielding Use, Rev. 8
- NGGM-PM-0002, Radiation Control & Protection Manual, Rev. 33

Records Reviewed

- Calendar Year (CY) 2002, Dose Breakdown Graph
- CY 2002 Daily Net Generation Graph
- Annual Doses 1992-2002, Graphical Data
- Steam Generator (SG) Dose-rate Data for Hot leg, Cold leg and Center of Bowl Dose-rates
- Shearon Harris Plant Three-Year Rolling Average vs. Top Quartile Plants, CY 2000

through CY 2002
- 2002 ALARA Report
- Spreadsheet: Pressurizer Water Reactor Collective Radiation Exposure Three-year rolling Averages, CY 2000 through CY 2002.
- Spreadsheet: RFO-11 (Refueling Outage 11) ALARA Plan Projections
- Spreadsheet: Historical Steam Generator Survey Data (1988-2000) for previous SG.

Records

- ALARA Committee Meeting Minutes, December 2002-March 2003
- ALARA 5 year plan 2002-2006
- Temporary Shielding Requests (TSR) 03-005, Shield [Seal Table] Transition Tubing, Cleaning Solution Cart., 11/11/02
- TSR 03-012, Install Shielding Inside Lower [reactor head] Shroud Doors, Around Thermocouple Conduits and Mechanisms, 11/11/02
- TSR 03-017, Provide General Background Shielding on SG Platforms, 11/11/02
- TSR 03-018, Shield Valve [1CS-491] and Piping to Prevent LHRA Occurrence, 11/11/02
- TSR 03-022, Hang Shield Wall to Reduce General Area Dose Rates While Reactor Head Is On Head Stand, Shield Around Reactor Head and Walkway by “B” SG and RCP., 11/11/02
- Harris Nuclear Plant Final Safety Analysis Report Chapter 11, Radioactive Waste Management
- Harris Nuclear Plant Final Safety Analysis Report Chapter 12, Radiation Protection

ALARA Work Packages (AWP)

- AWP 03-001, RFO-11 Reactor Headwork/ Refueling, Rev. 0
- AWP 03-007, Installation and Removal of Insulation, Rev. 0
- AWP 03-008, Seal Table Maintenance Activities, Rev. 0
- AWP 03-011, Reactor Head BMV and ASME Section XI Inspections, Rev. 0
- AWP 03-012, Health Physics Activities During RFO-11, Rev. 0
- AWP 03-024, General Maintenance (includes snubbers), Rev. 0
- AWP 03-028, Shielding Activities, Rev. 0

ALARA Work Plan Reviews

- AWP-03-024, Miscellaneous Maintenance Activities, 5/11/03
- AWP 03-024, General Maintenance (Snubbers), 5/8/03
- AWP 03-028, Shielding Activities During RFO 11, 4/30/03
- AWP 03-034, Refueling Activities, 5/12/03
- AWP-03-035, ISI RFO-11, 5/3/03
- AWP 03-035, ISI RFO-11, 5/8/03
- AWP 03-039, Scaffolding Activities for RFO-11, 5/5/03
- AWP 03-040, Insulation Activities for RFO-11, 5/4/03
- Memo to ALARA File, Inprogress Review (Overall), 5/4/03
- Memo to ALARA File, Inprogress Review (Overall), 5/8/03

Corrective Action Program Documents

- AR-54852 Wrong RWP Identified to I&C for Entry into HRA, 1/28/02
- AR-56257 Substandard Protective Clothing Problems, 2/22/02

- AR-59099 Dose Estimate Exceeded because PT-01BD Location Wrong, 4/11/02
- AR-64165 Dose Projection Underestimated for PRT Work, 6/27/02
- AR-83940 Hotspots from Resin Sluice Requires Flush Plan, 2/6/03
- AR-87051 Adverse Trend: Radiological Postings (81426, 82408, 84990,86100, 86892), 4/10/03

Section 4OA1: Performance Indicator Verification

Procedures

- CAP-NGGC-0200, Corrective Action Program, Rev. 7
- CAP-NGGC-0201, Self Assessment Program, Rev. 6
- REG-NGGC-0009, NRC Performance Indicators, Rev. 1
- OST-1026, Reactor Coolant System Leakage Evaluation, Computer Calculation, Daily Interval, Modes 1-2-3-4, Rev. 19

Corrective Action Program Documents

- AR-84078, Abnormal Release from PVS-1 due to Cask Vent (84003), 2/7/03
- AR-84003, PVS-1 Increase During Fill/ Vent of IF-304 BWR Cask, 2/7/03
- AR-86892, LHRA Posted Without Completing Requirements of AP-504
- AR-87919, Permanent Lead Shielding Found Removed in RAB 236 Letdown Heat Exchanger Area (from valve 1CS-95), 03/18/03
- Significant Adverse Condition Investigation Form (Root Cause Analysis) for AR-84003
- Annual Radioactive Effluent Release Report, January 1, 2002, to December 31, 2002, April 2003

Section 4OA5: Other Activities

- Westinghouse Letter PGN-03-40, dated May 28, 2003, entitled Upper Head Region Bulk Fluid Temperature Design Basis
- Procedure EPT-859, “100 % Bare Metal Visual Examination of the Reactor Pressure Vessel Head”
- Calculation, HNP-M/MECH-1091, “Effective Degradation Years for the Reactor Vessel Head”
- Procedure CM-M0070, “Reactor Vessel Mirror Insulation Disassembly/Reassembly Procedure”