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

January 29, 2008

Carolina Power and Light Company
ATTN: Mr. Robert J. Duncan, II
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 05000400/2007005

Dear Mr. Duncan:

On December 31, 2007, 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 January 16, 2008, 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 four issues of very low safety significance (Green). These issues involved violations of NRC requirements. However, because of their very low safety significance and because they have been entered into your corrective action program, the NRC is treating these issues as non-cited violations, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. Additionaly, one licensee identified violation, which was determined to be of very low safety significance, is listed in section 4OA7 of this report. If you deny 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/

Randall A. Musser, Chief
Reactor Projects Branch 4
Division of Reactor Projects

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

Enclosure: NRC Inspection Report 05000400/2007005
w/Attachment: Supplemental Information

cc w/encl: (See next page)

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Letter to Robert J. Duncan, II from Randall A. Musser dated January 29, 2008

SUBJECT: SHEARON HARRIS NUCLEAR POWER PLANT - NRC INTEGRATED INSPECTION REPORT 05000400/2007005

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket No:	50-400
License No:	NPF-63
Report No:	05000400/2007005
Licensee:	Carolina Power and Light Company
Facility:	Shearon Harris Nuclear Power Plant, Unit 1
Location:	5413 Shearon Harris Road New Hill, NC 27562
Dates:	October 1, 2007 through December 31, 2007
Inspectors:	P. O'Bryan, Senior Resident Inspector M. King, Resident Inspector L. Lake, Senior Reactor Inspector, Section 1R08 N. Staples, Reactor Inspector, Section 4OA2 R. Moore, Senior Reactor Inspector, Section 4OA5
Approved by:	R. Musser, Chief Reactor Projects Branch 4 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000400/2007-005; October 1, 2007 - December 31, 2007; Shearon Harris Nuclear Power Plant, Unit 1; Maintenance Effectiveness, Refueling and Outage activities, and Surveillance Testing.

The report covered a three-month period of inspection by resident inspectors and announced inspections by regional reactor inspectors. Four 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 4, dated December 2006.

A. Inspector-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. A Green self-revealing non-cited violation (NCV) of Technical Specification (TS) 3.7.13 was identified when the B essential services chilled water (ESCW) chiller tripped 2 minutes after it was started on November 5, 2007. The chiller tripped on low refrigerant pressure. The low refrigerant pressure was the result of inadequate seating of the transfer tank isolation valve after maintenance on October 13, 2007. Also contributing to the inoperability of the chiller was the fact that the post maintenance test (PMT) for the maintenance failed to verify that no leakage was occurring through the valve that was operated during maintenance. Therefore, refrigerant slowly leaked from the chiller to the transfer tank, and eventually the amount of refrigerant in the chiller was insufficient for the chiller to operate.

  This finding is greater than minor because it affected the availability and reliability objectives of the Equipment Performance attribute under the Mitigating System cornerstone. Since this finding represents an actual loss of safety function of a single train of technical specification equipment for greater than it’s allowed outage time, the finding was potentially greater than very low safety significance, and phase 2 and 3 analyses were required. A regional Senior Reactor Analyst performed the phase 3 evaluation under the Significance Determination Process for this performance deficiency. The results of this evaluation characterized the performance deficiency as of very low safety significance or Green. The NRC’s SPAR model was used for the analysis with the test and maintenance basic event for the Division B Chilled Water Pump used as the surrogate for the performance deficiency. The basic event was set to TRUE or always failing. The dominant accident sequence was a Small Break Loss of Coolant Accident followed by a failure of the other division’s Emergency Core Cooling System via various support system failures and a failure to provide alternate cooling to Division B’s High Head/ Charging Pump. External event initiators were considered, but were eventually excluded from the final quantification due to the very low core damage frequency contribution from internal initiating events. The cause of this issue is associated with the Resources component of the cross-cutting area of Human Performance, in that the procedures for performing the chiller maintenance did not include adequate operator instructions regarding the proper operation of the isolation valve and adequate post maintenance testing necessary to ensure that the ESCW system would remain available following maintenance. Specifically, the incomplete procedures are related to the cross-cutting aspect of providing complete, accurate and up-to-date design documentation, procedures, and work packages, and correct labeling of components. H.2(c) (Section 1R12)

- Green. A Green self-revealing NCV of TS 6.8.1., Programs and Procedures, was identified for an inadequate clearance order associated with engineering change 62848 on circuit 33 of power panel DP 1B-SB. As a result of the inadequate clearance, the discharge valve for the B emergency service water (ESW) pump, SW-271, would not automatically open when the B emergency service water pump was started. The clearance was inadequate because licensee operators failed to establish the proper plant equipment configuration to support hanging the clearance per procedure OPS-NGGC-1301, Equipment Clearance.

  The failure to establish the proper plant equipment configuration to support the equipment clearance for engineering change 62848 is greater than minor because it is associated with mitigating systems cornerstone attribute of configuration control and affects the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e. core damage). Per NRC Manual Chapter 0609, Appendix G, Shutdown Operations Significance Determination Process, this finding is of very low safety significance (Green) because operators were able to manually open the B ESW pump discharge and valve and maintain it's functionality. This finding was related to the cross-cutting area of human performance and the associated aspect of work planning because the licensee failed to properly configure plant equipment to support the clearance for engineering change 62848. H.3(a) (Section 1R20)

- Green. A Green self-revealing non-cited violation (NCV) was identified for the failure to properly implement operating procedures in accordance with TS 6.8.1. Operator error in procedure implementation of procedure OST-1858, "Remote Shutdown System Operability - Bus Drops Train A" led to the unexpected loss of power of the DP-1A-NNS bus and its associated loads, including the main control room annunciators.

  The finding is greater than minor because it is associated with the Equipment Performance attribute of the Mitigating Systems cornerstones. The finding also 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). The finding was evaluated using MC-0609, appendix G, significance determination for shutdown situations. The finding is considered to have very low safety significance (Green) because the finding did not require a quantitative assessment and therefore screened as green. A quantitative assessment was not required because the finding did not cause a loss of thermal margin, a loss of inventory, or degrade the ability to add inventory to the reactor coolant system. The finding was also related to the work coordination aspect of the cross-cutting area of human performance because the licensee failed to coordinate work associated with the DP-1A-NNS bus. H.3(b) (Section 1R20)

Cornerstone: Barrier Integrity

- Green. The inspectors identified a Green NCV of 10 CFR50, Appendix B, Criterion XVI, "Corrective Actions" when licensee personnel failed to promptly correct a condition adverse to quality. Specifically, six local leak rate test (LLRT) failures occurred between 1989 and 2003 on a service water containment isolation check valve. After the sixth LLRT failure during refueling outage (RFO) 11 in 2003, the licensee initiated a corrective action to disassemble and clean the valve each refueling outage (RFO) as a preventative maintenance activity. This corrective action was not sufficient to correct the cause of the LLRT failures, however, and the valve failed LLRT's during RFO 13 in 2006 and during RFO 14 in 2007.

  The failure to promptly correct the cause of the SW-233 LLRT failures is more than minor because it affected the Barrier Integrity cornerstone of assuring that physical design barriers (e.g. containment) protect the public from radio nuclide releases caused by accidents or events. It is also associated with the cornerstone attribute of system, structure, component and barrier performance. Manual Chapter 0609 Appendix A, Determining the Significance of Reactor Inspection Findings for At-Power Situations, was used to evaluate the significance of this finding. Since the service water supply piping to the non-safety containment air coolers had a second, redundant and functional containment isolation valve, and since this piping is a closed system within containment, the LLRT failure of SW-233 does not represent an actual open pathway in the physical integrity of reactor containment. This finding, therefore, was determined to be of very low safety significance (Green) using the phase 1 screening worksheet for barrier cornerstones. The finding was related to the timely corrective action aspect of the cross-cutting area of problem identification and resolution due to the delays in implementing effective corrective actions. P.1(d) (Section 1R22)

B. Licensee-Identified Violations

A violation of very low safety significance, which was identified by the licensee has been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. This violation and corrective action tracking numbers are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

The unit began the inspection period shutdown for refueling outage 14 (RFO14). The unit remained shutdown until it was restarted on October 22, 2007. On October 27, 2007 the unit reached full rated thermal power, and operated at full power until December 12, 2007 when power was reduced to approximately 50% of rated thermal power for planned maintenance. The unit was returned to 100% of rated thermal power on December 13, 2007 and operated at or near full power for the remainder of the quarter.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection

a. Inspection Scope

On October 24, 2007, a severe thunderstorm warning was issued for the site. The inspectors reviewed actions taken by the licensee in accordance with Procedure AP-300, Severe Weather Response, prior to the onset of that weather, to ensure that the adverse weather conditions would neither initiate a plant event nor prevent any system, structure, or component from performing its design function.

Also, after the licensee completed preparations for seasonal low temperatures, the inspectors walked down the emergency diesel generator and emergency service water systems. These systems were selected because their safety related functions could be affected by adverse weather. The inspectors reviewed documents listed in the attachment, observed plant conditions, and evaluated those conditions using criteria documented in Procedure AP-301, Seasonal Weather Preparations and Monitoring.

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

- AR #148743, Failure of ZDM Cards Due to Low Temperatures
- AR #253133, Fire Protection Valve 3FP-2 Freeze Concern

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 (OOS) for maintenance and testing:

- B train of decay heat removal with the A train out-of-service on October 4, 2007
- A train of spent fuel cooling with the B train out-of-service on October 9, 2007
- A emergency diesel generator with the B emergency diesel generator out-of-service on October 10, 2007
- B train of emergency service water with the A train out-of-service on December 13, 2007.

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 component cooling water 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 alignment of the system 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.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

For the seventeen 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 applicable FSAR sections. Documents reviewed are listed in the Attachment.

- 221', 236', 261', and 286' elevations of the reactor containment building including the following areas: 1-C-1-CHFA, 1-C-1-CHFB, 1-C-1-RCP-1A, 1-C-1-RCP-1B, 1-C-1-RCP-1C, 1-C-1-BAL, 1-C-3-EPA, and 1-C-3-EPB (8 areas).
- Emergency service water pump building and auxiliary intake structure including areas 12-I-ESWPA, 12-I-ESWPB, 12-I-ESWPA-BAL, 12-I-ESWPB-BAL, 5-S-BAL (5 areas).
- The 236' elevation of the reactor auxiliary building including areas 1-A-3-PB and 1-A-3-TA (2 areas).
- The 286' elevation of the reactor auxiliary building including areas 1-A-BATA and 1-A-BATB (2 areas).

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope

Internal Flooding

The inspectors walked down the 236' elevation of the reactor auxiliary building (RAB) containing risk-significant systems, structures and components which are below flood levels or otherwise susceptible to flooding from postulated pipe breaks, to verify that the area configuration, features, and equipment functions were consistent with the descriptions and assumptions used in FSAR section 3.6A.6, Flooding Analysis, and in the supporting basis documents listed in the Attachment. The inspectors reviewed the operator actions credited in the analysis, to verify that the desired results could be achieved using the plant procedures listed in the Attachment.

The inspectors also reviewed the licensee's corrective action documents to identify any flood-related items identified in AR's written from January 1 through December 21, 2007, to verify the adequacy of the corrective actions.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance

a. Inspection Scope

The inspectors observed Engineering Performance Test 163, Generic Letter 89-13 Inspection of the B Component Cooling Water Heat Exchanger, to verify that results were appropriately categorized against the pre-established acceptance criteria. 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. Documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI) Activities (71111.08P, Unit 1)

.1 Piping Systems ISI

a. Inspection Scope

The inspectors reviewed the implementation of the licensee's ISI program for monitoring degradation of the reactor coolant system (RCS) boundary and risk significant piping system boundaries. The inspectors reviewed a sample from the following activities performed during the Unit 1-Fall 2007/RFO14 refueling outage: a) nondestructive examinations (NDE) required by the 1989 Edition, no addenda, of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, b) examinations of reactor pressure vessel (RPV) head and head penetrations in accordance with NRC Order EA-03-009, c) commitments included in the NRC confirmatory action letter (CAL) dated March 29, 2007 covering corrective actions taken to address safety concerns associated with intergranular stress corrosion cracking (IGSCC) of dissimilar metal nozzle welds on pressurizer heads, d) disposition of NDE recordable indications, and e) welding activities as part of repair and replacement activities. Also, as a follow-up to the license renewal inspection conducted in

August 2007, the resolutions to corrective actions associated with missed augmented inspections in the break exclusion area of high energy lines in the main steam and feedwater systems were reviewed.

Specifically, the inspectors reviewed NDE procedures, NDE reports, NDE electronic data (as applicable), equipment calibration and certification records, personnel qualification records, and observed the following NDE activities.

- Baseline ultrasonic (UT) examination of weld overlays for safety valve 1RC-123.
- Bare metal visual examination of the RPV head and head penetrations.
- Bare metal visual examination of the RPV bottom head and head penetrations.

The inspectors reviewed the recordable indication associated with the visual (VT-3) examination of pipe support CC-H-1346 to verify that the evaluation and disposition of an as-found gap of 3/16″instead of a gap of 1/16″ required by design drawings was in accordance with the applicable edition of ASME Section XI, IWB-3000 and IWF-3000.

The inspectors reviewed the following Repair/Replacement Activities for compliance with ASME Code:

- Weld overlays performed in accordance with commitments in the CAL issued March 29, 2007, covering corrective actions taken to address safety concerns associated with IGSCC of dissimilar metal nozzle welds on pressurizer heads. This included Engineering Change 64915-R0, Welding Services Job # 103340, weld overlays completed on pressurizer safety valve nozzles, relief valve nozzle, pressurizer spray nozzle and the surge nozzle. Specifically, the inspectors reviewed weld process control sheets, welding procedure specifications, welding procedure qualification records, welder qualification records, Certified Material Test Reports for weld material, ASME Code reconciliation documents, and NDE reports.

b. Findings

No findings of significance were identified.

.2 Pressurized Water Reactor Vessel Upper Head

a. The inspectors reviewed the implementation of the licensee's ISI program for monitoring degradation of the RPV head and head penetrations in accordance with NRC Order EA-03-009. In addition, the inspectors verified that activities performed were in accordance with the requirements of the order and that indications and/or defects detected were dispositioned in accordance with the ASME Code or an NRC approved alternative.

The inspectors reviewed procedures, NDE reports, equipment calibration records, and personnel qualification records for the following VT-2 activities (examination for leakage) and bare metal visual examinations performed to meet the examination requirements of the NRC Order EA-03-009, and observed the following examinations:

- VT-2 examinations (examination for evidence of leakage), bare metal visual examinations of RPV head and head penetrations
- VT-2 examinations (examinations for evidence of leakage), bare metal visual examinations of RPV bottom head and head penetrations.

b. Findings

No findings of significance were identified.

.3 Boric Acid Corrosion Control (BACC) Program

a. Inspection Scope

The inspectors reviewed the licensee's BACC program activities to ensure implementation with commitments made in response to NRC Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary," and applicable industry guidance documents. Specifically, the inspectors performed an on-site record review of procedures and condition reports documenting the results of containment walkdown inspections. The inspectors also conducted an independent walk-down of the reactor building to evaluate compliance with licensee's BACC program requirements and verify that degraded or non-conforming conditions, such as boric acid leaks, were properly identified and corrected in accordance with the licensee's corrective action program.

The inspectors reviewed a sample of engineering evaluations completed for evidence of boric acid found on systems containing borated water to verify that the minimum design code required section thickness had been maintained for the affected components. Specifically, the inspectors reviewed the following evaluations:

- NCR #194879, Boric Acid Residue on CS-310
- NCR #204873, Boric Acid Residue on CT-33
- NCR #204386, Boric Acid Residue on 3-BR-E0006

b. Findings

No findings of significance were identified.

.4 Steam Generators

The licensee did not perform eddy current examinations of the unit 1 the Steam generators this outage.

.5 Identification and Resolution of Problems

a. The inspectors performed a review of ISI related problems, including welding, and BACC program that were identified by the licensee and entered into the corrective action program as Condition Report (CR) documents. Also reviewed, as a follow-up to Harris

License Renewal Inspection conducted in August 2007, where corrective actions associated with missed augmented inspections in the break exclusion area of high energy lines in the main steam and feedwater systems. The inspectors reviewed the CRs to confirm that the licensee had appropriately described the scope of the problem and had initiated corrective actions. The review also included the licensee's consideration and assessment of operating experience events applicable to the plant. The inspectors performed this review to ensure compliance with 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requirements. The corrective action documents reviewed by the inspectors are listed in the report attachment.

b Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

On November 19, 2007, the inspectors observed licensed-operator performance during requalification simulator training for crew C, to verify that operator performance was consistent with expected operator performance, as described in Simulator Examination Scenario DSS-008. This training tested the operators' ability to respond to a main steamline break inside of the reactor containment building. 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 two degraded SSC/function performance problems or conditions listed below to verify the licensee's handling of these performance problems or conditions in accordance with 10 CFR 50, Appendix B, Criterion XVI, Corrective Action, and 10 CFR 50.65, Maintenance Rule. Documents reviewed are listed in the Attachment.

- B essential services chilled water (ESCW) system chiller tripped 2 minutes after it was started on November 5, 2007.
- B CCW heat exchanger leak on December 12, 2007.

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 #253347, B ESCW Chiller Trip Shortly After Start For Train Swap
- AR #259102, B CCW Heat Exchanger Leaking From Southern Most End Bell

b. Findings

Introduction. A Green self-revealing NCV of Technical Specification (TS) 3.7.13 was identified when the B essential services chilled water (ESCW) chiller tripped 2 minutes after it was started on November 5, 2007. The chiller tripped on low refrigerant pressure. The low refrigerant pressure was the result of inadequate seating of the transfer tank isolation valve after maintenance on October 13, 2007. Also contributing to the inoperability of the chiller was the fact that the post maintenance test (PMT) for the maintenance failed to verify that no leakage was occurring through the valve that was operated during maintenance. Therefore, refrigerant slowly leaked from the chiller to the transfer tank, and eventually the amount of refrigerant in the chiller was insufficient for the chiller to operate and rendered the "B" ESCW chiller inoperable for a period of time greater than that allowed by TS.

Description. The ESCW chiller requires a sufficient amount of refrigerant to operate continuously without tripping on low refrigerant pressure. A refrigerant transfer system is permanently installed to allow refrigerant to be removed from the chiller unit to support maintenance. The refrigerant is temporarily stored in the refrigerant receiver tank which is equipped with a level gauge to verify proper refrigerant level.

During refueling outage 14 (RFO-14), refrigerant was removed from the B chiller to support scheduled maintenance. Following the scheduled maintenance work on the B chiller on October 13, 2007, refrigerant was transferred back to the chiller unit by conducting a pressurized liquid transfer from the receiver tank to the chiller in accordance with procedure OP-148, Essential Services Chilled Water System. OP-148, section 8.9 directs the operator to transfer refrigerant until the level gage on the receiver tank indicates that the tank is empty. Following completion of the refrigerant transfer, the chiller refrigerant transfer valves were realigned to their operational position. The operational position of the transfer system valves ensures that the chiller is isolated from the receiver tank so that refrigerant does not leak back to the receiver tank.

After all maintenance was completed, the chiller was operated several times. However, on November 5, 2007, the B ESCW chiller was started and it tripped on low evaporator pressure after running for two minutes. During troubleshooting it was discovered that the receiver tank, which had been left empty on October 13, 2007 (per OP-148), indicated ¼ full. Plant operators performed a valve line-up on the refrigerant transfer system and found a refrigerant receiver isolation valve, 1CY-7, partially open. After finding this condition, the refrigerant was transferred from the receiver tank back to the chiller, operating parameters were monitored and verified to be normal, and the chiller was declared Operable. Level in the receiver tank was monitored to confirm that 1CY-7 was no longer leaking past its seat.

Since valve 1CY-7 had not been operated since October 13, 2007, and since the majority of the refrigerant leakage past 1CY-7 most likely occurred during chiller operation (the differential pressure across the leaking valve is higher during chiller operation) before being shutdown on October 23, 2007, the chiller was inoperable for greater than the 72 hour allowed outage time in TS 3.7.13.

Analysis. This finding is greater than minor because it affected the availability and reliability objectives of the Equipment Performance attribute under the Mitigating System cornerstone. Since this finding represents an actual loss of safety function of a single train of technical specification equipment for greater than it's allowed outage time, the finding was potentially greater than very low safety significance, and phase 2 and 3 analyses were required. A regional Senior Reactor Analyst performed the phase 3 evaluation under the Significance Determination Process for this performance deficiency. The results of this evaluation characterized the performance deficiency as of very low safety significance or Green. The NRC's SPAR model was used for the analysis with the test and maintenance basic event for the Division B Chilled Water Pump used as the surrogate for the performance deficiency. The basic event was set to TRUE or always failing. The dominant accident sequence was a Small Break Loss of Coolant Accident followed by a failure of the other division's Emergency Core Cooling System via various support system failures and a failure to provide alternate cooling to Division B's High Head/ Charging Pump. External event initiators were considered, but were eventually excluded from the final quantification due to the very low core damage frequency contribution from internal initiating events. The cause of this issue is associated with the Resources component of the cross-cutting area of Human Performance, in that the procedures for performing the chiller maintenance did not include adequate operator instructions regarding the proper operation of the isolation valve and adequate post maintenance testing necessary to ensure that the ESCW system would remain available following maintenance. Specifically, the incomplete procedures are related to the cross-cutting aspect of providing complete, accurate and up-to-date design documentation, procedures, and work packages, and correct labeling of components. H.2(c)

Enforcement. TS 3.7.13 states, in part, that at least two independent Essential Service Water System loops shall be operable while in Modes 1, 2, 3, and 4. Contrary to the above, the B ESCW loop was determined to be inoperable for greater than the TS allowed outage time while in Mode 1. Because this finding is of very low safety significance and has been entered into the corrective action program (AR 253347), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000400/2007005-01, B ESCW chiller inoperable due to refrigerant leakage.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors reviewed the licensee's risk assessments and the risk management actions for the plant configurations associated with the five activities listed below. The inspectors verified that the licensee performed adequate risk assessments, and implemented appropriate risk management actions when required by 10 CFR 50.65(a)(4). For emergenent work, the inspectors also verified that any increase in risk was promptly assessed, and that the appropriate risk management actions were promptly implemented.

- Infrequent surveillance OST-1813, Remote Shutdown System Operability with reduced reactor coolant inventory while shutdown on October 2, 2007.
- Core reload with B train electrical busses out of service on October 19, 2007.
- B train emergency service water train out of service for repairs on October 20, 2007.
- A motor driven auxiliary feedwater pump and air handler AH-85B out of service for repairs on December 5, 2007.
- A emergency service water system and the A main feed pump out of service on December 13, 2007.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed six 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 OPS-NGGC-1305, 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 #248930, C-SA CSIP Breaker Did Not Operate from ACP During OST-1813
- AR #250810, EDG B Oscillations During OST-1824
- AR #250851, Inadvertent Drainage at 1SI-422 and 1SI-433
- AR #251018, Question on 1RH-16 and 1RH-54 Operability
- AR #251437, LT-487 Connector Assembly Bare Conductor
- AR #255869, 1SC-30, B ESW Screen Wash Isolation Valve Indicates Mid-Position

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 function(s) described in the FSAR and TS. The tests included the following:

- OST-1124, Train B 6.9 kV Emergency Bus Undervoltage Trip Actuating Device Operational Test and Contact Check Modes 1-6 after preventative maintenance on the B train electrical busses on October 11, 2007.
- OST-1804, RHR Remote Position Indicating and Timing Test 18 Month Interval Modes 5 and 6 after corrective maintenance on valves 1RH-16 and 1RH-54 on October 12, 2007.
- OST-1824, 1B-SB Emergency Diesel Generator Operability Test 18 Month Interval Modes 1 through 6 and Defueled after preventative maintenance on the B emergency diesel generator on October 16, 2007.
- OST-1812, Auxiliary Feedwater Isolation: ESF Response Time 18 Month Interval Modes 4 through 6 after corrective maintenance on valve 1AF-50 on October 18, 2007.
- OST-1808, Main Steam Isolation: ESF Response Time 18 Month Interval Modes 3 through 5 after corrective maintenance on 1MS-85 on October 19, 2007.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

The inspectors evaluated licensee activities as described below, to verify that licensees considered risk in developing outage schedules, adhered to administrative risk reduction methodologies developed to control plant configuration, developed mitigation strategies for losses of the key safety functions identified below, and adhered to operating license and technical specification requirements that maintained defense-in-depth. Documents reviewed are listed in the Attachment.

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

.1 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 technical specifications 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 are kept cognizant of plant configuration.

- clearance activities
- RCS instrumentation
- electrical power
- decay heat removal
- spent fuel pool cooling
- inventory control
- reactivity control
- containment closure

In addition, the inspectors completed their review of Operating Experience Smart Sample (OpESS) FY2007-03, Crane and Heavy Lift Inspection, Supplemental Guidance for IP-71111.20. Documents reviewed are listed in the Attachment.

b. Findings

Improper Maintenance Clearance Prevented the B ESW Pump from Opening Automatically

Introduction. A Green self-revealing NCV of TS 6.8.1., Programs and Procedures, was identified for an inadequate clearance order associated with engineering change 62848 on circuit 33 of power panel DP 1B-SB. As a result of the inadequate clearance, the discharge valve for the B emergency service water (ESW) pump, SW-271, would not automatically open when the B emergency service water pump was started. The clearance was inadequate because licensee operators failed to establish the proper plant equipment configuration to support hanging the clearance per procedure OPS-NGGC-1301, Equipment Clearance.

Description. On October 19, 2007, while in mode 5, licensee operators started the B ESW pump and noticed that the B ESW pump discharge valve, SW-271, did not automatically open as expected. Operators immediately stopped the B ESW pump and manually opened SW-271 and aligned the system for proper operation. Upon investigating the malfunction, operators discovered that an equipment clearance hung earlier that day on circuit 33 of power panel DP 1B-SB disabled the automatic opening function of SW-271. It was later discovered that clearance writers did not fully consider the impact of the clearance for engineering change 62848, which required the removal of power from the circuit for a design modification of the panel. Although the B ESW train is not required to be operational in mode 5, the B emergency diesel generator and the B train of decay heat removal were being relied upon as protected train equipment to provide core cooling. Therefore, the B train of ESW was necessary to ensure core decay heat removal in the event that off-site power was not available.

Analysis. The failure to establish the proper plant equipment configuration to support the equipment clearance for engineering change 62848 is greater than minor because it is associated with mitigating systems cornerstone attribute of configuration control and affects the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e. core damage). Per NRC Manual Chapter 0609, Appendix G, Shutdown Operations Significance Determination Process, this finding is of very low safety significance (Green) because operators were able to manually open the B ESW pump discharge and valve and maintain it's functionality. This finding was related to the cross-cutting area of human performance and the cross-cutting aspect of work planning H.3(a).

Enforcement. Technical Specification 6.8.1 requires that written procedures be established, implemented, and maintained covering applicable procedures recommended in Appendix A of Regulatory Guide 1.33. Regulatory Guide 1.33 includes procedures for equipment control (e.g. locking and tagging). Contrary to TS 6.8.1, the licensee inadequately controlled plant equipment configuration on October 19, 2007 when circuit 33 of panel DP 1B-SB was placed under clearance, unintentionally disabling the automatic opening function of the discharge valve for the B ESW pump. Because this finding is of very low safety significance and has been entered into the corrective action program (AR 251296), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000400/2007005-02, inadequate clearance for SW-271, the discharge valve for the B emergency service water (ESW) pump.

Loss of Annunciation System Due to Operator Error During Performance Surveillance Testing

Introduction. A Green self-revealing non-cited violation (NCV) was identified for the failure to properly implement operating procedures in accordance with TS 6.8.1. Operator error in procedure implementation led to the unexpected loss of power of the

DP-1A-NNS bus and its associated loads, including the Main Control Room Annunciators.

Description. On October 5, 2007 the licensee was performing procedure OST-1858, "Remote Shutdown System Operability - Bus Drops Train A, 18 Month Interval Modes 5, 6 or Defueled". Power for the DP-1A-NNS bus is normally supplied by one of two 125 VDC battery chargers, but power can also be supplied by the 125 VDC battery if the bus loses its normal power supply from the battery chargers. Section 4.0, Precautions and Limitations Number 14 of OST-1858 states that "this test will deenergize A Train 480 VAC Buses. If the battery is disconnected to DP-1A-SA or DP-1A-NNS when the AC supply to the battery charger is deenergized, the DC bus will also be deenergized." Test team personnel initialed that "the 125 VDC NNS Battery is connected to the bus" in Section 3.0, Prerequisite 8, but the battery was not actually connected to the bus due to ongoing maintenance. Test team personnel were not aware of the actual status of the maintenance on the 125 VDC NNS battery.

Since the 125 VDC NNS battery was not connected to the DP-1A-NNS bus, when power was removed to the battery chargers (Step 25 of Section 7.1), the DP-1A-NNS bus lost power. The DP-1A-NNS bus loads, including main control room annunciators, were unexpectedly lost. Main control room operators did not know why the annunciation system lost power and entered procedure AOP-037, "Loss of Main Control Room Annunciators." Test team personnel continued with OST-1858 and the bus was energized again approximately nine minutes later, restoring power to the Main Control Room Annunciators. Later the same day while performing the same test procedure, test team personnel again removed power to the battery chargers (Step 12 of Section 7.2). The DP-1A-NNS bus was again deenergized and its associated loads were lost for a second time. It was energized again after performing Step 14 of Section 7.2, approximately one minute later.

Analysis. The finding is greater than minor because it is associated with the Equipment Performance attribute of the Mitigating Systems cornerstones. The finding also 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). The finding was evaluated using MC-0609, appendix G, significance determination for shutdown situations. The finding is considered to have very low safety significance (Green) because the finding did not require a quantitative assessment and therefore screened as green. A quantitative assessment was not required because the finding did not cause a loss of thermal margin, a loss of inventory, or degrade the ability to add inventory to the reactor coolant system. The finding was also related to the cross-cutting area of human performance, aspect H.3(b), coordination of work activities.

Enforcement. Technical specification 6.8.1 requires that procedures be implemented for the activities listed in Appendix A of Regulatory Guide 1.33. Contrary to this requirement, licensee personnel did not properly implement procedural requirements during the performance of OST-1858. Specifically, on October 5, 2007, the licensee initialed that the 125 VDC NNS Battery was connected to the DP-1A-NNS bus in Section

3.0, Prerequisite 8 when the battery was not actually connected to the bus. Because of this, while performing Step 25 of Section 7.1, the DP-1A-NNS bus lost power and its associated loads were lost unexpectedly. This also subsequently led to the loss of Main Control Room Annunciators. Because the finding is of very low safety significance and has been entered into the licensee's corrective action program (AR 249554), this violation is being treated as a NCV, consistent with section VI.A of the NRC Enforcement Policy: NCV 05000400/2007005-03, Loss of Annunciation System Due to Operator Error During Performance Surveillance Testing.

.2 Reduced Inventory Conditions

a. Inspection Scope

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

b. Findings

No findings of significance were identified.

.3 Refueling Activities

a. Inspection Scope

The inspectors observed fuel handling operations (removal, inspection, sipping, reconstitution, and insertion) 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 to verify that the location of the fuel assemblies was tracked, including new fuel, from core offload through core reload.

b. Findings

No findings of significance were identified.

.4 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 TS, 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.

.5 Identification and Resolution of Problems

a. Inspection Scope

Periodically, the inspectors reviewed the 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 six surveillance tests identified below, the inspectors witnessed testing and/or reviewed test data, to verify that the systems, structures, and components 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.

- OST-1813, Remote Shutdown System Operability 18 Month Interval Mode 5, 6, or Defueled on October 2, 2007.
- OST-1801, ECCS Throttle Valve, CSIP, and Check Valve Verification 18 Month Interval Mode 5, 6, or Defueled on October 3, 2007.
- OST-1859, Remote Shutdown System Operability - Bus Drops Train B 18 Month Interval Mode 5, 6, or Defueled on October 11, 2007.
- * OST-1104, Containment Isolation Inservice Inspection Valve Test Quarterly Interval Modes 1 through 6 on November 16, 2007.
- OST-1026, Reactor Coolant System Leakage Evaluation, Computer Calculation, Daily Interval Modes 1 through 4 on November 15, 2007.
- ** EST-212, Type C Local Leak Rate Tests on October 4, 2007.

*This procedure included inservice testing requirements.
** This procedure included testing of a large containment isolation valve.

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

- AR #93928, 1SW-233 LLRT Failure
- AR #197683, 1SW-233 Stuck Open

b. Findings

Introduction: The inspectors identified a Green NCV of 10 CFR50, Appendix B, Criterion XVI, "Corrective Actions" when licensee personnel failed to promptly correct a condition adverse to quality. Specifically, six local leak rate test (LLRT) failures occurred between 1989 and 2003 on a service water containment isolation check valve. After the sixth LLRT failure during refueling outage (RFO) 11 in 2003, the licensee initiated a corrective action to disassemble and clean the valve each refueling outage (RFO) as a preventative maintenance activity. This corrective action was not sufficient to correct the cause of the LLRT failures, however, and the valve failed LLRT's during RFO 13 in 2006 and during RFO 14 in 2007.

Description: Valve SW-233 is a containment isolation check valve that is in the normal service water supply piping to the non-safety containment air coolers. The valve internals are made from carbon steel. The valve experienced six LLRT failures during outages up through RFO11 in 2003. After each of these six failures, work orders were written to disassemble, clean, and repair the valve. Work records indicate that valve internals were fouled with rust and debris, and that the disc would not move freely. In each instance, the as-left LLRT was performed with satisfactory results. Corrective actions after the RFO11 LLRT failure included initiating an Engineering Change (EC) request to replace the carbon steel valve internals with stainless steel to correct the corrosive buildup which prevented the free movement of the disc on the valve hinge. Additionally, a preventive maintenance (PM) work order to disassemble and clean the valve was initiated for SW-233 during RFO12 as an interim measure to prevent another LLRT failure until the design EC was implemented. The valve passed the LLRT in the RFO12 outage, however, the valve was disassembled and cleaned prior to the LLRT, so it is unknown if the LLRT would have passed in it's as-found condition.

During the RFO13 outage, the valve was again disassembled and cleaned prior to the LLRT test. However, the valve was initially found partially stuck open and the licensee noted that the valve could not have performed it's safety function of containment isolation in this degraded condition. Corrective actions for the valve failure in RFO13 again included implementing the EC to replace the valve internals material with stainless steel. However, the EC was not implemented in RFO13, and the valve again failed it's LLRT during RFO14 in October, 2007. In the RFO14 LLRT, the valve allowed leakage greater than the allowable containment leakage (L_a). After this failure in RFO14, the licensee implemented the EC to replace the check valve internals with stainless steel.

Analysis: The failure to promptly correct the cause of the SW-233 LLRT failures is more than minor because it affected the Barrier Integrity cornerstone of assuring that physical design barriers (e.g. containment) protect the public from radio nuclide releases caused by accidents or events. It is also associated with the cornerstone attribute of system, structure, component (SSC) and barrier performance. Manual Chapter 0609 Appendix A, Determining the Significance of Reactor Inspection Findings for At-Power Situations, was used to evaluate the significance of this finding. Since the service water supply piping to the non-safety containment air coolers had a second, redundant and functional containment isolation valve, and since this piping is a closed system within containment, the LLRT failure of SW-233 does not represent an actual open pathway in the physical integrity of reactor containment. This finding, therefore, was determined to be of very low safety significance (Green) using the phase 1 screening worksheet for barrier cornerstones. Inspectors also determined that the cause of the finding is related to the 'Timely Corrective Actions' aspect of the Problem Identification and Resolution cross cutting area due to the delays in implementing effective corrective actions (P.1(d)).

Enforcement: 10CFR 50, Appendix B, Criterion XVI, "Corrective Action," requires, in part, that conditions adverse to quality are promptly identified and corrected. Contrary to this requirement, the licensee failed to adequately implement timely corrective actions for a degraded service water containment isolation valve SW-233, resulting in an additional failure during RFO14 in October, 2007. However, because this violation is of very low safety significance, the issue was entered into the correction action program (AR 249347), and the deficient condition has been corrected, this finding is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. NCV 05000400/2007005-04, Failure to correct cause of SW-233 local leak rate failures.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed the temporary modifications described in the Engineering Change and work order listed below, 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.

- WO #970723-01, Temporary power to radiation monitor RM 3561D-SB
- EC #68259, SUT-1A Tank Pressure Data Collection - Temporary Modification

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

- AR #206791, A Start-Up Transformer Fast Pressure Rise LED
- AR #248378, Reactor Trip Entering RFO-014

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 December 4, 2007 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

4OA1 Performance Indicator Verification (PI) Verification

a. Cornerstone: Mitigating Systems

The inspectors sampled licensee submittals for the five PIs listed below. The inspectors looked at the period from fourth quarter 2006 through the third quarter 2007. To verify the accuracy of the PI data reported during that period, PI definitions and guidance contained in NEI 99-02, “Regulatory Assessment Indicator Guideline,” Revision 4, were used to verify the basis in reporting for each data element.

- Mitigating System Performance Index, Emergency AC Power
- Mitigating System Performance Index, High Pressure Safety Injection
- Mitigating System Performance Index, Residual Heat Removal
- Mitigating System Performance Index, Auxiliary Feedwater
- Mitigating System Performance Index, Cooling Water Systems

On a sample basis, the inspectors reviewed operating logs, work history information, maintenance rule information, corrective action program documents, and surveillance procedures to determine the actual time periods the MSPI systems were not available due to planned and unplanned activities. The results were then compared to the baseline planned unavailability and actual planned and unplanned unavailability determined by the Licensee to ensure the data’s accuracy and completeness. Likewise, these documents were reviewed to ensure MSPI component unreliability data determined by the licensee identified and properly characterized all failures of monitored components. The unavailability and unreliability data were then compared with performance indicator data submitted to the NRC to ensure it accurately reflected the performance history of these systems.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems

.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 CAP. The review was accomplished by reviewing daily AR reports.

.2 Annual Sample Review

a. Inspection Scope

The inspectors assessed the effectiveness of the licensee's corrective action program as it relates to the events surrounding the fault pressure trip of the Unit 1A startup transformers (SUT) on September 28, 2007. The inspectors reviewed the licensee's root cause analysis to assess the adequacy of the problem statement, extent of condition, identified root causes, and recommended corrective actions.

In order to assess the licensee's corrective action program as it relates to the engineering change process, the inspectors reviewed the permanent plant modification related to the sudden pressure relay (SPR) installed on the 1A-SUT. The inspectors reviewed this modification to ensure that risk significant functions of the transformers and off-site power were not affected by replacing the SPRs. The inspectors reviewed maintenance and testing documentation, modifications, performance trending, industry and vendor operating experience, and equipment history as identified by work orders, NCRs, and system health reports to assess the licensee's actions to verify and maintain the safety function, reliability and availability of 1A SUT. The review was performed in order to verify that specified acceptance criteria were met and appropriate and that the equipment operation was consistent with the plant's licensing and design bases.

Additionally, the inspectors reviewed documentation of individual WO's and NCRs which were originated since installation of the new SPR's which were initiated to address any identified issues with the SPR's. This effort included a review of the licensee's response to any abnormal indications of the SPR's.

b. Observations and Findings

On September 28, 2007, while reducing power for a refueling outage, a fault pressure trip on 1A-SUT resulted in the loss of the 1A, 1C and 1D electrical buses and caused a reactor/turbine trip with a loss of all reactor coolant pumps and a subsequent auxiliary feedwater (AFW) actuation. The 1A safety bus was reenergized automatically from the 'A' emergency diesel generator (EDG).

The fault pressure trip function for the SUT is provided by a Sudden Pressure Relay (SPR) which senses internal transformer fluid pressure. The SPR is designed to de-energize the SUT in the event of an internal transformer fault as indicated by a characteristic rise in internal fluid pressure within the transformer. Inspection of the transformer following the SPR actuation revealed no damage to the transformer and that an unnecessary actuation had occurred.

During the previous refueling outage in April 2006, an engineering change was implemented which replaced the old SPR design (Qualitrol Series 900) with a new design (Qualitrol Series 930). The series 900 was a mechanical design which detected changes in the transformer internal pressures with the use of a bellows and a linkage that actuated a switch to trip the circuit breaker that de-energized the transformer. The series 930 uses electronic pressure sensors that translate the pressure into an electrical signal. This signal is transmitted to the 930 main control panel, which monitors the transformer pressure and responds to sudden pressure increases occurring in a short period of time. The rapid pressure rise signal from the device is hard wired to require two out of three signals to actuate in order to trip the circuit breaker that de-energizes the transformer. The series 930 has user selectable sensitivity settings. The time it takes for the SPR to respond and trip the transformer is dependent upon how fast the pressure is increasing and, in the case of the series 930, its sensitivity setting. In general, the higher the rate of increase in pressure, the quicker the SPR relay responds. A response curve is defined for the SPR which indicates the acceptable range of response times for a vendor calibrated relay.

The licensee's root cause investigation team concluded that the sensitivity of the new series 930 SPR did not provide enough operating margin for low rates of pressure rise to prevent unnecessary lock out of the SUT. Normal transformer internal pressure changes near the low end of the relay sensitivity range are not modeled by the OEM transformer vendor and are not instrumented and monitored on in service transformers. The licensee followed the vendor's recommendation to use a sensitivity setting of "4" when installing the new relays. This decision was primarily based on the fact that the response curves for sensitivity settings (0 through 4) for the series 930 fall within the vendors generic response curve of the series 900 and successful operating experience with series 930 relays at setting "4" existed at the time of the engineering change. The setting of "4" is more sensitive than the "0" setting and effectively reduced the operating margin which contributed to the lockout of the SUT.

As an interim corrective action, the licensee generated AR 248378-04 to use temporary modification EC 68259, for gathering data on the 1A-SUT to determine the actual operation margin of the transformer. This temporary modification replaced the 1A SUT Qualitrol 930 sudden pressure relay with a Qualitrol 900 original design relay for protection (tripping) purposes and added a new factory calibrated Qualitrol 930 for data monitoring purposes. This configuration returns the protection back to a known acceptable condition with additional monitoring capability.

Although the inspectors did identify weaknesses in the rigor of the engineering analysis used to support the change to the new SPR design, ultimately the inspectors concluded that the licensee's actions were reasonable given the operating experience and vendor calibration curve data available at the time.

Additionally, the inspectors determined that the licensee missed a potential opportunity to identify the low operating margin one year prior to the SUT trip on September 28, 2007. On September 19, 2006, a turbine trip / reactor trip occurred due to an unrelated failed generator relay. Following the trip, a system engineer noted a locked in warning light for one channel of the series 930 relay on the 1A-SUT. An NCR was initiated and an adverse condition investigation was conducted. The conclusion of the investigation was that one of the sensors actuated as a result of the abrupt or sudden loading of the transformer and that the relay performed as designed preventing a spurious signal from unnecessarily tripping the SUT. The corrective action was to reset the warning light.

Following the September 28, 2007 scram, HNP requested that Qualitrol retest the installed series 930 relay. The test results showed that the installed SPR actuated too close to the pressure relay threshold for sensitivity setting "4". The team reviewed the test results of the "tripped" SPR and concluded that the relay was not within calibration. The sensitivity setting was for the "4" setting was slightly more sensitive. It failed to meet test requirements for the setting on all three channels. For a rate of .35PSIG/sec, each channel was out of calibration: Ch.1 (10.00s), Ch.2 (9.78s), and Ch. 3 (9.97s) and the acceptable range is 10.12 - 10.28 seconds. In this case, the relay tripped prematurely.

The inspectors determined that Qualitrol did not fully calibrate the full range of sensitivity settings "0" through "7" for use and that the licensee did not have the ability to perform independent calibrations in the field. The licensee generated ARs 234378-26 and 234378-05 to revise the purchasing specification for the 930 relays to require a calibration certification for the sensitivity range of application. The inspectors concluded that the calibration error contributed to the event, but was not the dominant causal factor

No violations of regulatory requirements occurred.

.3 Annual Sample Review - Operator Work Arounds

a. Inspection Scope

The inspectors reviewed the cumulative effects of deficiencies that constituted operator work arounds to determine whether or not they could affect the reliability, availability, and potential for misoperation of a mitigating system; affect multiple mitigating systems; or affect the ability of operators to respond in a correct and timely manner to plant transients and accidents. The inspectors also assessed whether operator work arounds were being identified and entered into the licensee's corrective action program at an appropriate threshold. Documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

.4 Semi-Annual Trend Review

a. Inspection Scope

The inspectors performed a review of the licensee's corrective action program (CAP) and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors' review was focused on repetitive equipment issues, but also considered the results of inspector CAP item screenings, licensee trending efforts, and licensee human performance results. The inspector's review nominally considered the six-month period of July through December, 2007, although some examples expanded beyond those dates when the scope of the trend warranted. The review also included issues documented outside the normal CAP in system health reports, self assessment reports, and Maintenance Rule assessments. The inspectors compared and contrasted their results with the results contained in the licensee's latest semi-annual trend reports.

The inspectors also evaluated the licensee's trend reports against the requirements of the CAP as specified in CAP-NGGC-0200, Corrective Action Program.

4OA3 Event Follow-up

.1 Turbine Trip/Reactor Trip - September 28, 2007

a. Inspection Scope

The inspectors responded to a reactor trip and turbine trip that occurred on September 28, 2007. The inspectors discussed the trip with operations, engineering, maintenance, and licensee management personnel to gain an understanding of the event and assess followup actions. The inspectors reviewed operator actions taken in accordance with licensee procedures and reviewed unit and system indications to verify that actions and system responses were as expected. The inspectors assessed the licensee's actions to gather, review, and assess information leading up to and following the trip. The inspectors later reviewed the initial investigation report and cause determination to assess the detail of review and adequacy of the cause determination and proposed corrective actions prior to restart.

The licensee's investigation identified the cause of the trip to be a fault pressure trip of the 1A Startup Transformer (SUT). The inspectors reviewed the root cause of the fault pressure trip as an annual inspection sample as documented in Section 4OA2.2 of this report. The inspectors also reviewed the initial notification to verify that it met the requirements specified in NUREG-1022, Event Reporting Guidelines.

b. Findings

No findings of significance were identified.

.2 (Closed) Licensee Event Report (LER) 05000400/2007-03-00. Automatic Reactor Trip following Actuation of a Sudden Pressure Relay and Lockout of the 1A Startup Transformer (SUT).

On September 28, 2007, while reducing power for Refueling Outage (RFO) 14, a fault pressure trip of the 1A-SUT caused a reactor trip/turbine trip with a loss of all reactor coolant pumps (RCP) and a subsequent auxiliary feedwater (AFW) actuation. The plant was at approximately 27 percent power at the time of the trip and all plant loads had been transferred to the SUTs. The 1A SUT tripped due to the actuation of its Qualitrol 930 series sudden pressure fault relay. This deenergized the AC busses powering the A and C RCPs causing a reactor trip and a trip of the B RCP, as designed, resulting in a loss of all forced flow in the reactor coolant system. The AFW system actuated following the reactor trip and operated as designed to stabilize steam generator levels. All systems functioned as required and no other safety systems were actuated. All control rods fully inserted on the reactor trip. The operations staff responded to the event in accordance with applicable plant procedures. The plant stabilized at normal operating no-load reactor coolant system temperature and pressure following the reactor trip. The LER was reviewed by inspectors and no findings of significance were identified and no violation of NRC requirements occurred. The licensee documented the 1A SUT trip in AR #248378. This LER is closed.

4OA5 Other Activities

.1 (Closed) Temporary Instruction (TI) 2515/166, Pressurized Water Reactor Containment Sump Blockage (NRC Generic Letter 2004-02)

a. Inspection Scope

The inspectors verified the implementation of the licensee's commitments documented in their September 1, 2005, response (HNP-05-101) to Generic Letter 2004-02, Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors. The commitments, referenced as corrective actions in the response, included plant modifications and procedure changes. The corrective actions included analysis and testing to determine the appropriate sizing and design for a permanent containment sump screen assembly modification, modifications to reduce min-K insulation contribution to post LOCA debris, and program controls to assure the assumptions regarding post LOCA debris generation and transport remain valid. Additionally, a license basis change to include in the FSAR a description of the deterministic evaluation methodology was listed as an action.

This inspection included review of the sump screen assembly installation procedure, screen assembly modification 10 CFR 50.59 evaluation, structural (debris) loading calculation, and validation testing of the modified sump screen design. The inspectors also reviewed the foreign materials exclusion controls and the completed Quality Assurance / Quality Control records for the screen assembly installation. The inspectors conducted a visual walkdown to verify the screen assembly configuration being installed was consistent with drawings and the tested configuration and verified the design criteria for screen gap were included in the installation instructions and drawing. The sump screen modification was not completed during the inspection. The resident inspectors verified the completion of the containment sump screen installation at the end of the outage.

b. Findings and Observations

No findings of significance were identified.

The licensee's actions stated in their September 1, 2005 response to GL 2004-02 were not complete during this inspection. The following is a listing of the corrective action commitments listed in the licensee's GL 2004-02 response and the status:

1. Modification - Install containment sump screen with perforation size to be determined by downstream effects study. Perforation size determined to be 3/32 inch. Size to be approximately 3000 square feet per sump (2 sumps).

2. Modification - Reduce min-K insulation on pressurizer safety relief valves' and power operated relief valves' piping to reduce contribution to post LOCA debris. Note: alternate action to removal taken to reduce min-K contribution by re-enforcing (banding) min-K cassettes to prevent or limit contribution to one instead of 6 lines per break.

3. Modification - Install trash rack over refueling canal drain.

4. Unspecified corrective actions to address concerns from evaluation of fuel and vessel internals, to be completed by March 31, 2006. Any required design changes to be completed by November 16, 2007. Analysis completed no corrective actions required.

5. Program controls referenced in 2(f) of response:

    1. Change design and licensing bases - prevent replacement of metallic reflective insulation with fiber glass blanket.

    2. Revise plant documents to incorporate limit for allowable quantity of unqualified coatings.

3. Revise deficiency tag procedure to provide additional guidance regarding tags in containment.

4. Procedure change to ensure refuel canal drain trash rack is installed after RFO.

5. Not stated in 2(f): revise procedure(s) to ensure min-K cassette bands are reinstalled after maintenance.

6. Head loss testing on HNP micro-porous insulation materials to validate current (response) assumptions.

7. Assure minimum NPSH available for RHR and CS pumps.

8. No gaps or breaches in excess of minimum sump screen opening (3/32 inch).

9. Trash racks and sump screens will be capable of withstanding applicable design basis loads.

10. Licensing basis change required - description of deterministic evaluation methodology in FSAR; deletion of 50 % assumption.

12. Corrective action not referenced in GL response: take pictures of installed min-K band installation to assist in maintenance re-installation.

13. Not referenced in GL response: revise procedures to assure min-K bands are installed to assure operability of containment sump.

4OA6 Meetings, Including Exit

On January 16, 2007, the resident inspectors presented the inspection results to Mr. Duncan and other members of his staff. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

4OA7 Licensee-Identified Violations

The following finding of very low significance was identified by the licensee and is a violation of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600 for being dispositioned as non-cited violations NCV.

- 10CFR20.1902(a) requires that radiation areas be conspicuously posted with signs bearing the radiation symbol and the words “CAUTION, RADIATION AREA.” Contrary to this requirement, on October 9, 2007, the licensee discovered that unattended equipment (one vacuum and two HEPA filters) were stored outside posted radiation areas and the dose rate was 10 mrem/hr at 30 cm from the equipment, which exceeds the 10CFR20.1902(a) limit of 5 mrem/hr at 30 cm. This was identified in the licensee’s corrective action program as AR 249992. This finding is of very low safety significance because it did not result in significant unintended radiation exposure.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

D. Alexander, Superintendent, Environmental and Chemistry
C. Burton, Director, Site Operations
D. Corlett, Supervisor, Licensing/Regulatory Programs
J. Dills, Manager, Outage and Scheduling
J. Dufner, Manager, Maintenance
R. Duncan, Vice President Harris Plant
M. Findlay, Superintendent, Security
W. Gurganious, Training Manager
K. Henderson, Plant General Manager
C. Kamiliaris, Manager, Nuclear Assessment Section
T. Natale, Manager, Site Support Services
S. O'Connor, Manager, Engineering
J. Pierce, Supervisor, Nuclear Assessment
G. Simmons, Superintendent, Radiation Control
J. Warner, Manager, Operations

NRC personnel

R. Musser, Chief, Reactor Projects Branch 4

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000400/2007005-01	NCV	B Essential Services Chiller inoperable greater than TS allowed outage time (Section 1R12).
05000400/2007005-02	NCV	Inadequate clearance for SW-271, discharge valve for the B emergency service water (ESW) pump (Section 1R20.1).
05000400/2007005-03	NCV	Loss of Annunciation System Due to Operator Error During Performance Surveillance Testing (Section 1R20.1).
05000400/2007005-04	NCV	Failure to correct cause of SW-233 local leak rate failures (Section 1R22).

Closed

05000400/2007-03-00	LER	Automatic Reactor Trip following Actuation of a Sudden Pressure Relay and Lockout of the 1A Startup Transformer (SUT). (Section 4OA3).
05000400, 2515/166	TI	Pressurized Water Reactor Containment Sump Blockage (NRC Generic Letter 2004-02) (Section 4OA5).

LIST OF DOCUMENTS REVIEWED

<u>Section 1R01: Adverse Weather Protection</u>

- AP-300, Severe Weather
- AP-301, Seasonal Weather Preparations and Monitoring
- Work orders 309223 and 694138

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

<u>Partial System Walkdown</u>

Residual Heat Removal system:

- Procedure OP-111, Residual Heat Removal System,
- Drawing 2165-S-1324, Simplified Flow Diagram Residual Heat Removal Systems

Spent Fuel Pool Cooling system:

- Procedure OP-116, Fuel Pool Cooling System,
- Drawing 2165-S-0305 , Simplified Flow Diagram Fuel Pool Cooling Systems

Emergency Diesel Generator system:

- Procedure OP-155, Diesel Generator Emergency Power System,
- Drawing 2165-S-0633, Diesel Generator Emergency Power System

Emergency Service Water system:

- Procedure OP-139, Service Water System.
- Drawing 2165-S-0547 and 2165-S-0548, Circulating and Service Water Systems, sheets 1 and 2.

<u>Complete System Walkdown</u>

- Procedure OP-145, Component Cooling Water
- System Description 145, Component Cooling Water System
- Drawing 2165-S-1319, 1320, 1321, and 1322, Simplified Flow Diagrams Component Cooling Water
- FSAR section 9.2.2, Component Cooling Water
- AR #225449, Ground Fault Relay Actuation on Breaker 1A-SA-8 for A CCW pump
- AR #246215, SW Vent Valve on A CCW Heat Exchanger Thread Leakage
- AR #250530, B CCW Heat Exchanger Channel Head Gasket Leakage
- AR #249825, Manual Valves Required in IST Program
- AR #243132, Q Class Components in CCW System Replaced with non-Q Parts
- Work Orders 1120016, 1135979, 1141164, 1155139, 1128134, 1015599, 904109, 685060, 194337, 1098195, 1097603, 1108612, 1108613, 1108610, 1108607, 1108611, 1108609, 1016468
- Engineering Change 2229, Redesign of A CCW Pump Ground Fault Relay

- Engineering Change 66805, Material Upgrade for CCW Class Three Flow Elements
- Engineering Change 50459, 6.9 kV Motor Causing Spurious Trouble Alarm

Section 1R05: Fire Protection

- FPP-012-02-RAB261,Reactor Auxiliary Building Elevation 261 Fire Pre-Plan
- FPP-012-04-DBG, Diesel Generator Building Fire Pre-Plan
- FPP-012-01-CNMT, Containment Building Fire Pre-Plan
- FPP-012-03-FHB, Fuel Handling Building Fire Pre-Plan
- FPP-012-07-TB, Turbine Building Fire Pre-Plan
- FPP-012-06-WPB, Waste Processing Building Fire Pre-Plan
- FPP-012-08-SEC, Out Building Fire Pre-Plan
- FPP-012-09-LAF, Large Area Fire Pre-Plan
- FPP-012-02-RAB 236, Reactor Auxiliary Building Elevation 236 Fire Pre-Plan
- FPP-012-02-190-216, Reactor Auxiliary Building Elevations 190 and 216 Fire Pre-Plan
- FPP-012-02-RAB286, Reactor Auxiliary Building Elevation 286 Fire Pre-Plan
- FPP-012-02-RAB305-324, Reactor Auxiliary Building Elevations 305 and 324 Fire Pre-Plan

Section 1R06: Flood Protection Measures

FSAR Sections

- 2.4.10, Flooding Protection Requirements
- 3.6A.6, Flooding Analysis

Calculations

- Appendix I to the HNP Probabilistic Safety Assessment, Internal Flooding Analysis
- Calculation #PRA-F/E-5, RAB Unit 1 Elevation 236 Compartment Flood Analysis

Procedures

- AOP-022, Loss of Service Water
- OP-139, Service Water System

Section 1R07: Heat Sink Performance

- PLP-620, Service Water Program (Generic Letter 89-13)
- EPT-163, Generic Letter 89-13 Inspections

Section 1R08: Inservice Inspection Activities

Procedures

- OPT-1519, Rev. 8, Containment Visual Inspection for Boron and Evaluation of Containment Sump leakage Every Refueling Outage Shutdown mode 3
- EPT-859, Rev. 2, 100% Bare Metal Visual Examination for Reactor Vessel Head
- SI-UT-126, Rev. 3 dated April 2007, Procedure for Phased Array Ultrasonic Examination of Weld Overlaid Similar and Dissimilar Metal Welds
- SI-NDE-08, Rev. 1 dated November, 2005, Qualification and Certification of NDE Personnel for Nuclear Applications
- SI-NDE-09, Rev. 0 dated September 2004, Personnel Visual Acuity Examination Procedure
- Document #QP0902, Revision 8 dated September 22, 2004, Qualification and Certification of NDE Personnel
- QAP-2.7, Rev. 13 dated January 19, 2007, Selection Training and Qualification of Non-Destructive Examination Personnel
- QAP-9.16, Rev. 3 dated January 17, 2007, High Temperature Liquid Penetrant Inspection Procedure, Using Color Visible/Solvent Removable Penetrant
- QAP-9.21, Rev. 1 dated April 17, 2007, Liquid Penetrant Inspection Procedure Solvent Removable Visible Dye
- QAP-9.3, Rev. 17, Workmanship and Visual inspection criteria for ASME Welding

Corrective Action and Evaluation Documents

- AR 00250005/NCR 20071010, ISI Pressure Test not performed on buried piping
- AR 00249573/NCR 20071006, Snubbers deleted from program remain in field
- AR236989, Containment IWE/IWL Inspection Program has not been maintained for effective implementation
- AR00239921, Welds beyond the branch connections off the MS lines were not included in the augmented inspection program for break exclusion requirements
- Summary Report # H-04702, Report # VT-04-004, Evaluation of Thread Damage on 3 Studs on Component II-SI-027SI-137 VBB
- NCR #070145, First five beads of weld overlay completed with no water in system on the PZR surge line
- AR00249976/NCR20071009, Snubber SG-H-0305B Failed Functional Test

Other Records

- Assessment #178624, October 23 - 26, 2006 - Harris Inservice Inspection Program Self-Assessment Report
- Installation of Weld Overlays, Engineering Change 64915, Rev 0,/Welding Services (WSI) Job #103340
  1. WSI Traveler No. 103340-01, PZR Overlays Safety Valve 1RC123
  2. WSI Traveler No. 103340-04, PZR Overlays Surge Line
- H-ISI-07-01, dated July 19, 2007, Harris Inservice Inspection and Testing Assessment
- PLP-106, Rev. 42, Technical Specification Equipment List Program

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

Section 1R13: Maintenance Risk Assessments and Emergent Work Evaluation

- OMP-003, Outage Shutdown Risk Management.
- WCM-001, On-line Maintenance.

Section 1R15: Operability Evaluations

- OPS-NGGC-1305, Operability Determinations

Section 1R20: Refueling and Outage Activities

- FHP-020, Refueling Operations
- FHP-014, Fuel and Insert Shuffle Sequence
- AOP-020, Loss of Reactor Coolant System Inventory or Residual Heat Removal While Shutdown
- GP-004, Reactor Startup
- GP-005, Power Operation
- GP-008, Draining the Reactor Coolant System
- OMP-003, Outage Shutdown Risk Management
- OMP-004, Control of Plant Activities During Reduced Inventory Conditions
- OST-1034, Containment Penetrations Test Weekly Interval During Core Alterations and Movement of Irradiated Fuel Inside Containment, and
- OST-1091, Containment Closure Test Weekly Interval During Core Alterations and Movement of Irradiated Fuel Inside Containment
- AR 251296, SW-271 Failed to Open on Start of B ESW Pump
- CWD 2166-B-401
- Engineering Change 62848
- OPS-NGGC-1301, Equipment Clearance
- APP-ALB-002, Annunciator Response Procedure, ESW Header B Return to Auxiliary Reservoir Blocked or Misaligned
- AOP-022, Loss of Service Water
- Engineering Change 62848

Section 4OA1: Performance Indicator Verification

- NEI 99-02, Regulatory Assessment Performance Indicator Guideline
- Calculation HNP-F/PSA-0068, NRC Mitigating System Performance Index Basis Document for Harris Nuclear Plant

Section 4OA2: Identification and Resolution of Problems

- CAP-NGGC-0200, Corrective Action Program.
- Operator Work Around Cumulative Effect Review, October, 2007
- AR #213826, Operator Work Around Aggregate Assessment in Alert Region
- AR #240236, Operator Work Around Cumulative Effects Review
- AR #235815, Untimely Engineering CAP Trend Report Response
- AR #236527, Adverse Trend in Transient Combustibles
- AR #237329, Resource Related Issues - Engineering CAP Trends
- AR #259207, Human Performance Review Board 2007 Trend Results
- TRM-K67-SUT-Tranfromer-MISC-002, SHNPP SUT Transformer External Maintenance, Rev. 0
- TRM-K67-P4755-GENBANK-001.0, Installation and Inspection of the new Qualitrol Electronic Rapid Pressure Relay on Startup 1A Power Transformer
- RD-43881, Startup Power Transformer Connection Diagram, Rev. 2
- RD-43880, Startup Power Transformer Connection Diagram, Rev. 2
- RD-43883, Startup Power Transformer Connection and Wiring Diagram, Rev. 4
- RD-43884, Startup Power Transformer Elementary Diagram, Rev. 4
- RD-43885, Startup Power Transformer Elementary Diagram, Rev. 5
- RD-43886, Startup Power Transformer Elementary Diagram, Rev. 4
- RE-49168, Startup Power Transformer Connection Diagram, Sh. 1, Rev. 2
- RE-49180, Qualitrol VD Electronic Rapid Pressure Relay, Rev. 1
- NCR 206710, Generator Lockout Cause Reactor trip Caused by Degraded Ground Fault Relays
- NGGM-IA-0003, NGG Interface Agreement, Rev. 5
- EC 54349, Transformer Fault Pressure Relay Protection Improvement, Rev. 7
- GEK-63643, GE Instruction Booklet for Start-up Power Transformer

Section 4OA5: Other

- General Procedure (GP) 009, Refueling Cavity Fill, Refueling and Drain of the Refueling Cavity, Modes5-6-5, Rev. 46
- HNP-C/STRU-1102, Top Hat Assembly Qualification Calculation, Rev. 4
- HNP-C/STRU-1103, Analysis of Vortex Suppressor and Trash Racks Structure, Rev. 0
- HNP-C/CONT-1012, Containment Unqualified Coatings Log, Rev. 0
- SD-0023, HNP Containment Bldg. GSI-191 Debris Generation Calculation, Rev. 1
- SD-0026, HNP GSI-191 Recirculation Sump Screen Debris Bed Head Loss, Rev. 2
- Engineering Change (EC) 064377, Replacement Recirculation Sump Screens, Rev. 10
- EC 064378, Specification for Replacement Recirculation Sump Screens, Rev. 1
- HNP-C/STRU-1104, Analysis of Sump Strainer Structure, Rev. 0
- SD-0022, Minimum Post-LOCA Containment Water Level, Rev. 0
- EC 067479, Re-enforcement of Min-K Insulations, Rev. 0
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- EC 067430, Containment Recirculation Sump Level and Temperature Instrumentation Relocation, Rev. 0

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- Administrative Procedure (AP) 038, Deficiency Tag Procedure, Rev. 13
- EGR-NGGC-0005, Engineering Change, Rev. 26
- HNP-C/EO-1224, Qualification of Wire Mesh Doors in Containment, Rev. 3
- AREVA Calculation 51-9029298-000, GSI-191 Downstream Effects Evaluation for HNP, dated 11/21/06
- Nuclear Condition Report (NCR) 00251172, B ECCS Sump Rework, dated 10/18/07
- GP-002, Normal Plant Heatup from Cold Solid to Hot Subcritical, Mode 5 to Mode 3, Rev. 40
- Drawing PGENH013 (1366-098433), Support Bonding for Min-K Insulation, Pressurizer Loop Seals, Rev. 0
- ENERCON Services Report PEHoo6-RPT-003, SOI Evaluation for Min-K Insulation on the Pressurizer Loop Seals At Harris Nuclear Plant, 8/30/07
- Work Order 892954-03, EC 64377, “A” Containment Spray Sump, dated 6/27/07
- Work Order 892954-05, EC 64377, “B” Containment Spray Sump, dated 6/27/07
- Corrective Maintenance (CM) M0206, Pressurizer Power Operated Relief Valve, Rev. 13