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

April 30, 2007

Virginia Electric and Power Company
ATTN: Mr. David A. Christian
Sr. Vice President and
Chief Nuclear Officer
Innsbrook Technical Center - 2SW
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: SURRY POWER STATION - NRC INTEGRATED INSPECTION REPORT NOS. 05000280/2007-02 AND 05000281/2007-02 AND ANNUAL ASSESSMENT MEETING SUMMARY

Dear Mr. Christian:

On March 31, 2007, the United States Nuclear Regulatory Commission (NRC) completed an inspection at your Surry Power Station, Units 1 and 2. The enclosed report documents the inspection findings which were discussed on April 11 and 23, 2007, with Mr. Jernigan and other members of your staff.

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

Based on the results of this inspection, four self-revealing findings and one NRC-identified finding of very low safety significance (Green) were identified. Four of these findings were determined to involve violations of NRC requirements. However, because of their very low safety significance and because they have been entered into your corrective action program, the NRC is treating these issues as non-cited violations (NCVs) in accordance with Section VI.A.1 of the NRC's Enforcement Policy. In addition, one licensee-identified violation, which was determined to be of very low safety significance (Green), is listed in Section 4OA7 of this report. If you deny any of these NCVs you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the United States 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 Surry Power Station.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component 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/

Eugene F. Guthrie, Chief
Reactor Projects Branch 5
Division of Reactor Projects

Docket Nos.: 50-280, 50-281
License Nos.: DPR-32, DPR-37

Enclosure: Integrated Inspection Report
5000280,281/2007002
w/Attachment: Supplemental Information

cc w/encl.: (See page 3)

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Letter to David A. Christian from Eugene F. Guthrie dated April 30, 2007
SUBJECT: SURRY POWER STATION - NRC INTEGRATED INSPECTION REPORT NOS. 05000280/2007-02, 05000281/2007-02

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.: 50-280, 50-281

License Nos.: DPR-32, DPR-37

Report Nos.: 05000280/2007002, 05000281/2007002

Licensee: Virginia Electric and Power Company (VEPCO)

Facility: Surry Power Station, Units 1 & 2

Location: 5850 Hog Island Road
Surry, VA 23883

Dates: January 1 - March 31, 2007

Inspectors: N. Garrett, Senior Resident Inspector
D. Arnett, Resident Inspector
M. Bates, Senior Operations Engineer (Section 1R11.1)
B. Caballero, Operations Engineer (Section 1R11.1)
J. Reece, Senior Resident Inspector, North Anna Power Station (Section 4OA3.3 and 4OA3.4)
R. Rodriguez, Reactor Inspector (Section 4OA3.4)

Approved by: E. Guthrie, Chief, Reactor Projects Branch 5
Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000280/2007-02, IR 05000281/2007-02, on 01/01/2007 - 03/31/2007; Surry Power Station Units 1 & 2; Maintenance Risk Assessment and Emergent Work, Identification and Resolution of Problems, and Followup of Events.

The report covered a three month period of inspection by Resident Inspectors, a Senior Operations Engineer, an Operations Engineer, a Senior Resident Inspector from another site and a Reactor Inspector. An in-office review was conducted of an unresolved item. Five Green findings, four of which were 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. NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. A Green self-revealing finding was identified for not performing an adequate extent of condition review in accordance with the licensee's established procedures. The potential consequences of having siding torn from the Turbine Building if Unit 2 experienced steam relief valve actuations, as had occurred on Unit 1 on June 29, 2006, was not recognized. Consequently on October 7, 2006, the actuation of Unit 2 steam relief valves resulted in the loss of normal power to two emergency buses on Unit 1 and one emergency bus on Unit 2 due to flying debris impacting electrical conductors. The finding was entered into the licensee's corrective action program as Condition Report (CR) 003598. Sections of the Turbine Building near the discharge of the steam relief valves were temporarily strengthened while additional long term corrective actions were being evaluated.

  The finding is more than minor due to its impact on the Initiating Events objective to limit the likelihood of those events that upset plant stability and the related attribute of human performance. This finding was of very low safety significance (Green) because the increase in risk was limited by the duration of the condition. The cause of the finding was directly related to the appropriate and timely corrective actions aspect of the problem identification and resolution cross-cutting area because sufficient information was available for the licensee to have identified potential damage to plant equipment and taken actions to limit it when the Unit 2 steam relief valves actuated. (Section 4OA3.3)

Cornerstone: Mitigating Systems

- Green. The inspectors identified a Green, non-cited violation (NCV) of 10 CFR 50.65 (a)(4), “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” for failure to assess the increase in risk that results from proposed maintenance activities. Specifically, in assessing the increase in risk of planned maintenance activities, the licensee failed to adjust their on-line risk evaluation for operation with a pressurizer power operated relief valve (PORV) block valve closed. The licensee entered this violation in their corrective action program as CR007055 for resolution.

  This finding is more than minor because it relates to risk management, in that, the maintenance risk assessments failed to consider the unavailability of a PORV, a risk significant component. The inspectors determined that the finding is of very low safety significance (Green) since the inclusion of the closed PORV block valve into the risk assessments showed no appreciable increase in plant risk during the performed maintenance activities. The cause of the finding was directly related to the proper work planning aspect of the human performance cross-cutting area because the licensee had not included the PORV block valve in their on-line risk monitoring program. (Section 1R13)

- Green. A self-revealing, non-cited violation of Technical Specification 6.4.D, “Unit Operating Procedures and Programs”, was identified for failure to follow procedure. Specifically, foreign material was left inside a Unit 1 control rod guide tube which prevented the full insertion of control rod K-14 during a manual reactor trip. The procedure in use during maintenance specifically required an inspection for and removal of all foreign material from the control rod guide tube. The licensee entered this violation in their corrective action program as CR002285 for resolution, performed a root cause analysis, and determined corrective actions.

  The finding is more than minor because it affects the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences to the reactor core and resulted in a degraded rod control system. The inspectors determined that the finding is of very low safety significance (Green) since only one control rod was affected. The cause of the finding was directly related to the procedural compliance aspect of the human performance cross-cutting area because personnel failed to adequately perform a search for foreign material as required by procedures. (Section 4OA2)

- Green. A self-revealing non-cited violation of Technical Specification 6.4.A, “Unit Operating Procedures and Programs,” was identified for failure to have adequate written procedures for normal startup of systems and components involving nuclear safety. Specifically, the licensee failed to have adequate procedural instructions to preclude the Unit 2 charging pump component cooling water system from becoming air bound when returning it to service. As a result all three Unit 2 charging pumps, also emergency core cooling system (ECCS) pumps, were declared inoperable due to lack of seal cooling. This violation was entered in the licensee corrective action program as CR000031 for resolution, which included performing an apparent cause analysis, and determining corrective actions.

The finding is more than minor because a procedural error that results in a consequence, in this case the inoperability of three ECCS pumps, is more than of minor safety significance. The significance of the finding was determined to be of very low safety significance (Green) due to the short period of time the cooling water system was unavailable. The cause of the finding was directly related the complete documentation and component labeling aspect of the human performance cross-cutting area because procedural instructions for venting the component cooling water system were not adequate. (Section 4OA3.1)

- Green. A self-revealing, non-cited violation of 10 CFR 50.63 was identified regarding a breaker control circuit design deficiency which prevented the licensee from supplying the 1J emergency bus on Unit 1 from the alternate AC diesel generator. The problem occurred during a transient on October 7, 2006, in which the actuation of Unit 2 steam relief valves resulted in the loss of normal power to two emergency buses on Unit 1 and one emergency bus on Unit 2 due to flying debris impacting electrical conductors. The licensee installed a modification to correct the breaker circuit design deficiency.

  The finding is more than minor because it impacted the Mitigating Systems objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage) and the related attribute of design control. This finding was of very low safety significance (Green) because the recovery of the alternate AC diesel generator's ability to energize a safety bus after four hours was credible. (Section 4OA3.4)

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

Unit 1 began the report period operating at or near 80 percent rated thermal power (RTP) due to a dropped rod. On January 6, the unit was shut down to replace the control rod drive mechanism cable for control rod K-8. The unit was placed on-line January 7, and reached full RTP on January 8. The unit operated at or near full RTP for the remainder of the report period.
Unit 2 operated at or near full RTP for the entire report period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection

a. Inspection Scope

On January 25 and February 5, the licensee entered operations checklist OC-21, "Severe Weather Checklist" when temperatures were predicted to drop to well below freezing. The inspectors reviewed the checklist, reviewed operator logs, and performed focused walkdowns in the turbine building, auxiliary building, safeguards, fire pump house, high level and low level intakes to verify proper operator actions were performed to prevent freezing of equipment.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

.1 Partial System Walkdowns

a. Inspection Scope

The inspectors performed the following four partial system walkdowns to verify correct system alignment. The inspectors checked for correct valve and electrical power alignments by comparing positions of valves, switches, and breakers to the procedures and drawings listed in the Attachment. Additionally, the inspectors reviewed the corrective action system to verify that equipment alignment problems were being identified and properly resolved.

- Unit 1 'B' Safety Injection pump, 1-S1-P-1B while 1-SI-P-1A was out for a maintenance package
- Unit 1 'A' & 'C' Charging pumps, 1-CH-P-1A/C while 1-CH-P-1B was out for a maintenance package

- Unit 2 ‘B’ Containment Spray pump, 2-CS-P-1B while 2-CS-P-1A was undergoing maintenance
- Unit 1 'B’ Emergency Service Water (ESW) Pump, 1-SW-P-1B and 1 'C’ ESW Pump, 1-SW-P-1C while 1 'A’ ESW Pump, 1-SW-P-1A was tagged out for a maintenance package

b. Findings

No findings of significance were identified.

1R05 Fire Protection

.1 Fire Area Walkdowns

a. Inspection Scope

The inspectors conducted tours of the following ten areas to assess the adequacy of the fire protection program implementation. The inspectors checked for the control of transient combustibles and the condition of the fire detection and fire suppression systems (using “SPS Appendix R Report,”) in the following areas:

- Unit 1 Emergency Switchgear Room
- Unit 2 Emergency Switchgear Room
- Unit 1 Normal Switchgear Room
- Unit 2 Normal Switchgear Room
- Number 1 Emergency Diesel Generator
- Unit 2 Safeguards
- Auxiliary Building 2' level
- Auxiliary Building 13' level
- Auxiliary Building 27' level
- Auxiliary Building 35' level

b. Findings

No findings of significance were identified.

.2 Annual Fire Brigade Drill

a. Inspection Scope

The inspectors observed a fire brigade drill to evaluate the readiness of the licensee’s personnel to fight fires. Specific aspects evaluated were: use of protective clothing and self contained breathing apparatus; fire hose deployment and reach; approach into the fire area; effectiveness of communications among the fire brigade members and the control room; sufficiency of fire fighting equipment brought to the fire scene; and the drill objectives and acceptance criteria.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope

The inspectors reviewed the Updated Final Safety Analysis Report (UFSAR) and the Individual Plant Examination (IPE) of Non-Seismic External Events and Fires for analyzed external and internal floods. Walkdowns were performed in the turbine building, emergency switchgear room, manhole number 3, and auxiliary building to review compliance with procedures for internal and external flooding. In addition, the inspectors walked down floor drain back water stop valves, various expansion joint shields, and flood and spill control dams. The inspectors reviewed completed preventive maintenance and surveillance records for the turbine building sump pumps and floor drain back water stop valve replacement. The documents reviewed are listed in the Attachment of the report.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance

a. Inspection Scope

The inspectors evaluated the condition of the Unit 1 component cooling (CC) heat exchanger, 1-CC-E-1B during annual cleaning. The inspectors discussed the heat exchanger monitoring program with engineering personnel and reviewed the current heat exchanger program document ER-AA-HTX-10, “Heat Exchanger Program”. The inspectors observed the condition of the heat exchanger prior to the performance of tube scraping performed under Maintenance Work Order (MWO) 749015-01. The inspectors reviewed the performance results of surveillance procedure 1-OSP-SW-003, Rev 18, “Macrofouling of CC HX 1-CC-E-1B”, prior to and following the performance of the cleaning.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program

.1 Biennial Licensed Operator Requalification Review

a. Inspection Scope

The inspectors reviewed facility operating history and associated documents in preparation for this inspection. While onsite the inspectors reviewed documentation, interviewed licensee personnel and observed the administration of operating tests associated with the licensee's operator requalification program. Each of the activities performed by the inspectors was done to assess the effectiveness of the licensee in implementing requalification requirements identified in 10 CFR Part 55, "Operators' Licenses." The evaluations were also performed to determine if the licensee effectively implemented operator requalification guidelines established in NUREG 1021, "Operator Licensing Examination Standards for Power Reactors," and Inspection Procedure 71111.11, "Licensed Operator Requalification Program." The inspectors also evaluated the licensee's simulation facility for adequacy for use in operator licensing examinations using ANSI/ANS-3.5-1998, "American National Standard for Nuclear Power Plant Simulators for use in Operator Training and Examination." The inspectors observed two crews during the performance of the operating tests. Documentation reviewed included written examinations, job performance measures, simulator scenarios, licensee procedures, on-shift records, licensed operator qualification records, watchstanding and medical records, simulator modification requests and performance test records, operator feedback, and remediation plans. Documents reviewed during the inspection are listed in the Attachment to this report.

b. Findings

Introduction: The inspectors identified an Unresolved Item (URI) associated with a licensed reactor operator (RO) who incorrectly re-activated his license in November 2006. As part of the reactivation, the RO performed portions of his plant tour without being escorted by an active licensed operator.

Description: In preparing for the biennial licensed operator requalification inspection, the inspectors requested that the licensee have available for review, the reactivation documentation for all operators who had reactivated their license since the previous licensed operator requalification inspection. As part of this documentation, the inspectors requested records for the escorts who accompanied the reactivated operators during their required plant tour. The licensee performed a review of these records prior to the inspectors arriving at the facility and in the process discovered that one RO had not correctly performed the required plant tour. The licensee entered the issue into their corrective action program as CR006467. The RO subsequently re-performed tours of the plant, with a licensed operator escort, for all areas that were in question.

After arriving onsite, the inspectors reviewed operator license reactivation records for four licensed operators. These four individuals were the only licensed operators who had reactivated their license since the last biennial licensed operator requalification inspection and were currently maintaining an operating license. Documentation for reactivation was made on “Return to Active License Status Certification” which is Attachment 1 to Licensed Operator Requalification Program Guide, Appendix C. The inspectors reviewed the information contained in Attachment 1 and corroborated the licensee’s discovery that one RO had not correctly completed all portions of the required plant tour. The Auxiliary Building and Fuel Building were two areas of the plant that the RO did not correctly completed the tour.

The inspectors reviewed the guidance in the Licensed Operator Requalification Program Guide to determine the level of detail that is provided for reactivation plant tours. The guidance is specific in stating that an active RO or SRO, as appropriate, shall accompany the individual on the tour and sign the “Return to Active License Status” form.

<u>Analysis</u>: Licensed Operator Requalification Program Guide, Appendix C, Attachment 1, specifically requires an active licensed RO or SRO, as appropriate, to escort the reactivating operator in all areas that watchstanders normally tour. The requirements of 10 CFR 55.53(f), Conditions of a License, also require the complete plant tour to be performed under the direction of an RO or SRO, as appropriate.

Based on the information the licensee was able to provide, the inspectors determined that one reactivating RO was not escorted by an active licensed operator when performing a tour of the Auxiliary Building and Fuel Building. The inspectors determined that the licensee identified issue of failing to correctly perform a complete plant tour with another active licensed operator is a performance deficiency because the licensee must satisfy the requirements of 10 CFR 55.53 for license reactivation, as well as the requirements of Licensed Operator Requalification Program Guide, Appendix C, Attachment 1.

<u>Enforcement</u>: 10 CFR 55.53(f), Conditions of a License, states in part, that the licensee has completed a minimum of 40 hours of shift functions under the direction of an operator or senior operator, as appropriate, and in the position to which the individual will be assigned. The 40 hours must have included a complete tour of the plant and all required shift turnover procedures.

The reactivating RO did not perform a complete plant tour under the direction of an active RO or SRO, as required by 10 CFR 55.53(f), Conditions of a License. The licensee entered the issue into their corrective action program as CR006467. Additional NRC review of all the circumstances involving this issue is required before a final determination can be made, therefore, this item is identified as an Unresolved Item (URI) 05000280,281/2007002-01, Failure to properly reactivate an RO license. The item is unresolved pending NRC review of additional information.

.2 Quarterly Licensed Operator Requalification Review

a. Inspection Scope

The inspectors observed licensed operator performance during simulator training session RQ-07.2-SP-1 to determine whether the operators:

- were familiar with and could successfully implement the procedures associated with recognizing and recovering from loss of pressurizer master controller, failure of a vital bus, failure of a turbine impulse pressure channel, a reactor coolant system leak, and a large break loss of coolant accident
- recognized the high-risk actions in those procedures
- were familiar with related industry operating experiences

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

For the two equipment issues described below, the inspectors evaluated the licensee's effectiveness of the corresponding preventive and corrective maintenance. For each selected item below, the inspectors performed a detailed review of the problem history and surrounding circumstances, evaluated the extent of condition reviews as required, and reviewed the generic implications of the equipment and/or work practice problem. Inspectors performed walkdowns of the accessible portions of the system, performed in-office reviews of procedures and evaluations, and held discussions with system engineers. Inspectors compared the licensee's actions with the requirements of the Maintenance Rule (10 CFR 50.65), VPAP 0815, "Maintenance Rule Program," and the Surry Maintenance Rule Scoping and Performance Criteria Matrix.

- Unit 2 Safety Injection Check Valves, and
- Unit 1 and Unit 2 Turbine Driven Auxiliary Feedwater Pump Valves

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors evaluated the adequacy, accuracy, and completeness of the following six plant risk assessments performed prior to changes in plant configuration for maintenance activities or in response to emergent conditions. When applicable, inspectors assessed if the licensee entered the appropriate risk category in accordance with plant procedures. Specifically, the inspectors reviewed:

- Plan of the Day (POD) for the week of January 8 - 13, for schedule changes incorporated while bringing Unit 1 critical, extension of 2-OPT-FW-002, and adding of 2-IA-TV-201B due to it not re-opening
- POD for the week of January 13 - 19, 2007, for schedule changes and risk impact including rescheduling of risk significant surveillances
- POD for the week of January 22 - 26, 2007 for schedule changes and the addition of risk significant surveillances, adding of 2-RC-PCV-2455C, ‘C’ PORV due to its block valve being closed and the extension of 1-CH-P-1B, Unit 1 ‘B’ Charging Pump
- POD for the week of February 3 - 9, 2007 for schedule changes and risk impact including failure of the steam dump temperature control valve, rescheduling risk significant maintenance, extension of work on the 1 'A’ component cooling heat exchanger
- POD for the week of February 24 - March 2 for schedule changes and risk impact including rescheduling of risk significant surveillances, failure of the Unit 2 emergency switchgear room air handling unit, 2-VS-AC-6, extension of maintenance on the Unit 2 ‘D’ main condenser waterbox and extension of maintenance Unit 1 ‘D’ component cooling heat exchanger
- POD for the week of March 5 - 9, 2007 for the extension of 1-CC-E-1D and 2-IA-C/D-4B and addition of 2-OPT-FW-006 along with an inside containment trip valve being declared inoperable

b. Findings

Introduction: The inspectors identified a Green non-cited violation (NCV) for a failure to comply with 10 CFR 50.65 (a)(4), “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants”, by not including the Unit 2 pressurizer power operated relief valves (PORV) in the on-line risk evaluation when the PORV block valves were closed.

Description: On January 21, 2007, the licensee identified an adverse trend on the Unit 2 pressurizer PORV tailpipe temperature. Several times during the period January 21 to February 1, the licensee closed and re-opened PORV block valves in an attempt to determine which PORV was contributing to the tailpipe temperature rise. One PORV block valve was maintained open at all times during this period. The licensee did not enter the closed PORV block valve in the plant on-line risk monitor until prompted by the inspectors on February 1. During the period January 21 to February 1, the licensee removed and returned to service various risk significant components for maintenance. During this period, the risk assessments performed to manage the risk, while these components were out of service for maintenance, were not correct since the evaluations had not included the closed PORV block valve.

Analysis: The inspectors determined that the failure to include the closed PORV block valve in risk assessments was a performance deficiency. The inspectors referenced

Appendix B of IMC 0612 and determined that the performance deficiency was a finding. The finding is more than minor when screened through Section 3, Minor Question (5)(a), "The licensee on-line risk assessment failed to consider risk significant SSC and support systems that were unavailable during maintenance." The inspectors referenced IMC 0609, Appendix K, to evaluate the significance relative to risk. The baseline risk for Unit 2 is 8.80E-6/yr and risk for Unit 2 with the pressurizer PORV block valve closed is 8.90E-6/yr. Given the baseline CDF for Unit 2 of 8.80E-6/yr, the incremental increase in risk of 1.0E-7/yr and the short exposure times for 2-RC-MOV-2536 of approximately 16.1 hours and 2-RC-MOV-2535 of approximately 214.7 hours the incremental core damage probability deficit was 1.84E-10 and 2.44E-9 respectively. When the deficit is below the 1.0E-6, the finding is considered to be of very low safety significance. Thus, the inspectors determined that the finding is of very low safety significance (Green) since the inclusion of the closed PORV block valve into the risk assessments showed no appreciable increase in plant risk during the performed maintenance activities.

The cause of the finding was directly related to the proper work planning aspect of the human performance cross-cutting area because the licensee had not included the PORV block valve in their on-line risk monitoring program.

Enforcement: 10 CFR 50.65(a)(4) states, in part, that before performing maintenance activities the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities. Contrary to this, during the time period from January 21, 2007, to February 1, 2007, the risk assessment performed prior to maintenance activities did not include all the applicable risk significant SSC. Because this finding is of very low safety significance and because it was entered into the licensee's corrective action program as Condition Report 007055, this violation is being treated as an NCV, consistent with Section VI.A of the NRC enforcement Policy: NCV 05000280,281/2007002-02, Failure to Perform a Risk Assessment Related to Shutting the Unit 2 Pressurizer Power Operated Relief Valve.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors evaluated the technical adequacy of the six operability evaluations to ensure that operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The operability evaluations were described in the condition reports listed below:

- Condition Report (CR) 006265, The valve disk for 2-CC-806 is not seated properly in the closed direction
- CR 006540 Incorrect vendor weld procedure used during fabrication of Unit 2 Containment Sump
- CR 006621 Evaluation needed for cooling of auxiliary feedwater (AFW) lube oil coolers at minimum AFW recirculation flow
- CR 005930 During NRC walkdown found an Orange Card reading breaker not installed properly in cubical; questioned breaker operability

- CR 009110 Evaluate Charging Pump Discharge check valve criteria
- CR 009543 1-CC-P-2A rotating backwards during pump swap

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing

a. Inspection Scope

The inspectors reviewed the six post maintenance test procedures and activities associated with the repair or replacement of the following components to determine whether the procedures and test activities were adequate to verify operability and functional capability following maintenance of the following equipment:

- Maintenance Work Order (MWO) 745718-02, Replacement of 2-SW-445, service water check valve
- MWO 00746163-01, Mechanical PM for 1-CH-P-1B, ‘B’ Charging Pump
- MWO 00747390-01, Clean and Inspect - Electrical test of 2-SI-P-1B, unit ‘2' B Safety Injection pump
- MWO 00755592-01, Repair of, ‘A’ Emergency Service Water Pump 1-SW-P-1A
- MWO 00728517-01, Mechanical PM of 1-SI-MOV-1862A, LHSI pump A suction from RWST
- MWO 00773428-01, Trouble shoot and repair rod motion

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities (Unit 1 Forced Outage)

a. Inspection Scope

The inspectors observed portions of the reactor plant shutdown and startup for the Unit 1 outage to affect repairs on the K-8 control rod drive mechanism control cable inside containment. The outage began on January 6, 2007, and ended January 7, 2007. The inspectors reviewed the status and configuration of electrical systems to verify they met technical specification requirements, selected control room activities to verify the licensee was controlling reactivity in accordance with the technical specifications, and plant configuration changes on a sampling basis to verify technical specifications and administrative procedure prerequisites were met prior to changing plant configurations.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

For the seven surveillance tests listed below, the inspectors examined the test procedure and either witnessed the testing and/or reviewed test records to determine whether the scope of testing adequately demonstrated that the affected equipment was functional and operable:

Surveillance Tests

- 1-OPT-EG-001, “Number 1 Emergency Diesel Generator Monthly Start Exercise”
- 2-OSP-SI-004, “SI Check Valve Troubleshooting”
- 2-ST-FW-002, “Evaluation of Auxiliary Feedwater System Vibration”
- 2-PT-8.4, “Consequence Limiting Safeguards (Hi-Train)”
- 1-PT-8.1, “Reactor Protection System Logic (For Normal Operations)”

Inservice Test

- 2-OPT-FW-003, “Turbine Drive Auxiliary Feedwater Pump 2-FW-P-2"
- 2-OPT-CS-002, “Containment Spray System Test”

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation

a. Inspection Scope

The inspectors observed the announced emergency response training drill conducted on March 26, 2007, to assess the licensee’s performance in emergency classification, off-site notification and protective action recommendations. The drill included emergency response actions taken by the management team in the Technical Support Center (TSC). This drill evaluation is included in the Emergency Response Performance Indicator statistics.

b. Findings

No findings of significance were identified.

4 OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems

.1 Daily Review of Plant Issues

a. Inspection Scope

As required by Inspection Procedure 71152, “Identification and Resolution of Problems”, and in order to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee’s corrective action program. This review was accomplished by reviewing hard copies of each condition report, attending daily screening meetings, and accessing the licensee’s computerized database.

b. Findings

No findings of significance were identified.

.2 Focused Review of Plant Issues

a. Inspection Scope

The inspectors performed an in-depth review of the root cause evaluation and corrective actions for the failure of Unit 2 control rod K-14 to drop into the core following a reactor trip. This issue was documented in the corrective action program as Condition Report (CR) 002285. The review was performed to ensure the full extent of the issue was identified, an appropriate evaluation was performed, and appropriate corrective actions were specified and prioritized. The inspectors evaluated the plant issue against the requirements of the licensee’s corrective action program as delineated in Station Administrative Procedure VPAP-1601, “Corrective Action,” and 10 CFR 50 Appendix B, Criterion XVI, “Corrective Action.”

b. Findings

Introduction: A Green self-revealing non-cited violation (NCV) of Technical Specification 6.4.D, “Unit Operating Procedures and Programs,” was identified for failure to follow procedures, in that, foreign material was not identified and removed from the core during a refueling outage. Subsequently, control rod K-14 failed to insert into the core during a manual reactor trip.

Description: On October 7, 2006, a Green self-revealing finding was identified when Unit 2 control rod K-14 failed to fully insert into the core following a manual reactor trip. During the Unit 2 refueling outage which was completed in May 2005, the licensee replaced the split pins and inserts for the control rod guide tubes under procedure MRS-SSP-1750.0, “Guide Tube Support Pin Replacement at Surry Unit 2 (VIR)”. The procedure replaced all of the guide tube inserts with a flexureless insert and removed all of the flexures used to hold the old style inserts in place. The licensee procedure required that a final video inspection be performed to ensure no debris that could affect control rod insertion is in the upper or lower guide tube. During the manual trip in October 2006, control rod K-14 did not fully insert into the reactor core. The licensee exercised the rod to verify free movement and the rod stuck at 66 steps out of the core. Prior to core disassembly, the control rod was fully inserted into the core. Following reactor plant disassembly during Refueling Outage 2006, subsequent inspections by the licensee determined the K-14 rod cluster control assembly was damaged and free control rod movement was prevented by foreign material inside the control rod guide tube. The licensee determined the foreign material had been video-taped following completion of maintenance during the Unit 2 refueling outage in 2005 but was not identified by the technicians reviewing the tape at that time.

Analysis: The inspectors determined that the failure to identify and remove foreign material from the reactor is a performance deficiency. The inspectors referenced Appendix B of IMC 0612 and determined that the performance deficiency was a finding. The finding was screened through the questions in Appendix E and determined to be more than minor. The specific example is b. in Section 4. This example states that a procedural error that results in a consequence is a more than minor violation. The licensee did not follow the procedural requirement to ensure there was no debris left in the control rod guide tube which could affect control rod motion. The foreign material left in the control rod guide tube resulted in damage to and prevented free motion of the rod cluster control assembly (RCCA). This performance deficiency affected the Reactor Safety/Mitigating Systems Cornerstone by reducing the availability and reliability of systems that respond to initiating events and resulted in a degraded reactivity control system. The performance deficiency screens as Green through all of the questions in the mitigating systems cornerstone worksheet.

The cause of the finding was directly related to the procedural compliance aspect of the human performance cross-cutting area because personnel failed to adequately perform a search for foreign material as required by procedures.

Enforcement: Technical Specification 6.4.D requires, in part, that all procedures described in section 6.4.A and 6.4.B shall be followed. Contrary to the above, during the performance of procedure MRS-SSP-1750.0, “Guide Tube Support Pin Replacement at Surry Unit 2 (VIR)”, the licensee failed to identify and remove debris in the upper or lower guide tubes which could affect control rod insertion. As a result, foreign material in the guide tube prevented insertion of control rod K-14. Because this finding is of very low safety significance and because it was entered into the licensee’s corrective action program as Condition Report 002285, this violation is being treated as an NCV, consistent with Section VI.A of the NRC enforcement Policy: NCV 05000281/2007002-03, Foreign Material Prevents Insertion of Unit 2 Control Rod K-14.

4OA3 Followup of Events

.1 (Closed) Licensee Event Report (LER) 05000281/2006-001-00, Charging Pump Component Cooling Water System Inoperable Due to inadequate Venting

a. Inspection Scope

The inspectors reviewed LER 05000281/2006-001-00 to review the circumstances surrounding the event and the appropriateness of the licensee's corrective actions.

b. Findings

Introduction: A Green self-revealing non-cited violation (NCV) of Technical Specification 6.4.A, "Unit Operating Procedures and Programs," was identified for an inadequate procedure which resulted in an inoperable Unit 2 charging pump component cooling system and the three charging pumps, which also serve as emergency core cooling pumps, being declared inoperable.

Description: A Green self-revealing finding was identified when the Unit 2 charging pump component cooling (CPCC) system became air bound which resulted in declaring the three Unit 2 charging pumps (high head safety injection pumps) inoperable. On July 27, 2006, the licensee removed the Unit 2 'A' charging pump component cooling water intermediate seal cooler from service to repair a leaking union on the discharge of the seal cooler. During the return to service, main control annunciator 2D-G, SW or CC PPS DISCH TO CHG PPG LO PRESS lit, the standby pump started, but no system flow was observed. The licensee secured the CPCC pumps and declared the system inoperable. The licensee declared the three Unit 2 charging pumps (high head safety injection pumps) inoperable and entered Technical Specification 3.0.1 which requires the unit to be in hot shutdown within six hours. The licensee vented the system and restarted a CPCC pump. The system was returned to service; which returned the charging pumps to service. The licensee determined the system became air bound due to inadequate procedural direction for venting the system after maintenance. The CPCC system and the charging pumps were inoperable for 1 hour 38 minutes.

Analysis: The inspectors determined that the failure to have an adequate procedure to properly vent the CPCC system was a performance deficiency. The performance deficiency is a finding which is more than minor since the finding is similar to example (b) of Section 4 to IMC 0612, Appendix E, in that, a procedural error that results in a consequence is more than of minor safety significance. In this case, air in the CPCC system prevented cooling of the seals for all three charging pumps. The finding is in the Mitigating Systems Cornerstone.

A Significance Determination Process (SDP) Phase 3 was performed using the NRC's risk software. The finding was modeled by setting the basic events for the charging pump closed cooling water system to a failed state. The loss of the charging pump closed cooling water system was potentially very significant, because of the importance of the charging pump to high pressure sequences in the risk model. The significance was highest for the Small Loss of Coolant Accident (LOCA), Medium LOCA and Steam GeneratorTube Rupture accident initiators. The significance of the finding was determined to be of very low safety significance, Green, due to the very short period of time the cooling water system was unavailable.

The cause of the finding was directly related the complete documentation and component labeling aspect of the human performance cross-cutting area because procedural instructions for venting the component cooling water system were not adequate.

Enforcement: Technical Specification 6.4.A requires, in part, that detailed written procedures with appropriate check-off lists and instructions shall be provided for normal startup and shutdown of all systems and components involving nuclear safety of the station. Contrary to the above, the licensee procedure to return the CPCC system to service contained inadequate venting instructions. As a result, the charging pump component cooling system became air bound which resulted in the three charging pumps (high head safety injection pumps) being declared inoperable. Because this finding is of very low safety significance and because it was entered into the licensee's corrective action program (Condition Report 000031), this violation is being treated as an NCV, consistent with Section VI.A of the NRC enforcement Policy: NCV 05000281/2007002-04, Inadequate Procedural Instructions Results in Inoperable Charging Pump Component Cooling Water System. This LER is considered closed.

.2 (Closed) LER 05000280/2006-001-00, Voluntary Report - Motor Driven AFW Pump High flow condition with Low Steam Generator Pressure

On May 4, 2006, with Unit 1 defueled and Unit 2 operating at 100 percent power, an AFW system design issue, related to the original plant design, was discovered. The design concern is a motor driven AFW pump unanalyzed high flow condition that could occur during a limited time frame with low steam generator pressure and reactor coolant system (RCS) temperature between 350°F and 500°F. Specific postulated scenarios with certain initiating events, along with an assumed single failure, could result in no operating AFW pumps on a unit. However, the Surry plant design includes an AFW cross-connect and the licensee has established temporary controls to preclude the high flow condition. The inspectors reviewed the licensee operability determination, Plant Issue S-2006-1810, CR 001000 and the planned corrective actions. Temporary modifications S1-06-088 for Unit 1 and S2-06-040 for Unit 2 were prepared to disable the auto-open function on two Motor Operated Valves (MOVs) to one steam generator and to close these two MOVs when the RCS temperature is between 350°F and 500°F. After performing special tests and analyzing the resultant pump curves, the licensee has placed operational limits on the AFW system that would adequately prevent the postulated high flow condition. This LER is considered closed.

.3 <u>(Closed) URI 05000280, 281/2006011-03</u>, Review of Licensee Corrective Actions to Preclude Recurrence of Damage From Cross-under Relief Valve Actuations

a. <u>Inspection Scope</u>

The inspectors completed a review and characterization of URI 05000280, 281/2006011-03. The inspectors reviewed the licensee's corrective action database for related documented issues and problems to determine the adequacy of any corrective actions implemented. The inspectors also interviewed licensing personnel to obtain similar documentation from legacy corrective action databases.

b. <u>Findings</u>

<u>Introduction</u>: A Green self-revealing finding was identified for not performing an adequate extent of condition review, as specified in their own standards established in VPAP-1601, for a Unit 1, June 26, 2006, event. This resulted in a loss of normal offsite power to three emergency buses.

<u>Description</u>: On October 7, 2006, Unit 2 was manually tripped during a transient based on indications associated with main steam flow, main steam pressure, and steam generator feedwater flow and level perturbations. During the transient, exhaust steam discharging from the Unit 2 cross-under piping relief valves (CURVs) impacted the adjacent turbine building siding and created flying debris. The flying debris impacted the 'A' and 'C' Reserve Service Station Transformers' (RSST) electrical conductors resulting in loss of normal offsite power to both Unit 1 and one of the Unit 2 emergency buses. Additional details associated with the event are contained in NRC Special Inspection Report 05000280, 281/2006011. The finding was entered into the licensee's corrective action program as Condition Report (CR) 003598. Observations of the Special Inspection inspectors resulted in the licensee temporarily strengthening sections of the Turbine Building near the discharge of the CURVs while additional long term corrective actions were being evaluated.

The inspectors identified several previous events involving cross-under relief valve (CURV) actuations. Two of these, one in 1996 and the other on June 29, 2006, documented damage to the turbine building. Only the June 29, 2006, event, documented in Plant Issue S-2006-2851, was considered as relevant to current licensee performance. From the information contained in the plant issue and associated documentation, the inspectors concluded that the licensee's extent of condition review was inadequate. The plant issue stated that, "Several whole sheets and several partial sheets of metal were caught in the path of the steam flow and hurled across the unit 1 alleyway towards the "D" building approximately 100 feet away. In their extent of condition review, the licensee recognized the need to identify actions to prevent Turbine Building wall degradation due to CURV actuations on both Units. However, the extent of condition review was not adequate, in that, the potential consequences from Turbine Building wall damage due to a Unit 2 CURV actuations were not identified. Consequently, compensatory corrective actions were not taken to limit those consequences, commensurate with their potential safety significance. Specifically, prior to October 7, 2006, the potential damage to the RSST transformer electrical conductors was not recognized and addressed by the extent of condition review, even though the distance from these conductors to the Unit 2 CURVs exhaust lines was less than the distance the debris was scattered during the Unit 1, June 29, 2006, event.

Licensee procedure, VPAP-1601, “Corrective Action Program,” establishes the measures to be taken to assure that failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly corrected. Procedure VPAP-1501, “Deviations,” included conditions that affect plant reliability as deviations. With respect to extent of condition, VPAP-1601 defined ‘extent’ as “Identification of impact or potential impact on the opposite unit, related or similar equipment, and related documents.” The inspectors further noted that compensatory action is defined as “Action taken to temporarily address a deficient condition until permanent corrective actions can be implemented. Normally identified as short term corrective action.”

Analysis: The inspectors identified a performance deficiency in which the licensee failed to perform an adequate extent of condition review for Plant Issue S-2006-2851. Sufficient information was available for the licensee to have identified the potential for more significant damage to plant equipment due to flying debris if a similar transient involving CURV actuations occurred on Unit 2. Not performing an adequate extent of condition review in accordance with their own standards established in VPAP-1601 was a finding. The inspectors reviewed Inspection Manual Chapter (IMC) 0612 and determined that the finding is more than minor due to its impact on the objectives of two different reactor safety cornerstones: (1) the Initiating Events objective to limit the likelihood of those events that upset plant stability and the related attribute of human performance, and (2) the Mitigating Systems objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage) and the related attribute of human performance.

A Phase 3 SDP analysis was performed for the condition that led to a loss of offsite power to the three emergency buses while Unit 1 remained in service. The condition, as it existed, could lead to a Loss of Offsite Power (LOOP) to the site if Unit 1 had tripped while the emergency buses were not energized from offsite sources. The increase in the likelihood of a loss of offsite power was determined by estimating the likelihood of a plant trip during the time the condition existed. Risk was then calculated by multiplying this probability by the Conditional Core Damage Probability for a LOOP as determined by the NRC's risk model. Because the increase in risk was limited by the duration of the condition, this resulted in a Green finding.

The cause of the finding was directly related to the appropriate and timely corrective actions aspect of the problem identification and resolution cross-cutting area because sufficient information was available for the licensee to have identified potential damage to plant equipment and taken actions to limit it when the Unit 2 CURVs actuated.

Enforcement: No violation of regulatory requirements was identified. The inspectors determined that the finding did not represent a noncompliance because it occurred on nonsafety-related plant equipment. This finding is of very low safety significance (Green) and was entered into the licensee's corrective action program as Condition Report (CR) 003598. The finding is identified as 05000280, 281/2007002-05: Failure to Perform an Adequate Extent of Condition Review for Unit 1 June 29, 2006, Turbine Building Damage Event.

.4 (Closed) URI 05000280, 281/2006011-02, Breaker Design Deficiency Prevents the AAC DG from being Loaded as Designed

a. Inspection Scope

The inspectors completed a review and characterization of URI 05000280, 281/2006011-02. The inspectors reviewed the licensee's design basis for compliance with the station blackout requirements of 10 CFR 50.63, "Loss of all alternating current power," and consulted with the appropriate NRC technical staff.

b. Findings

Introduction: The inspectors identified a Green, self-revealing, NCV of 10 CFR 50.63 regarding a breaker control circuit design deficiency.

Description: On October 7, 2006, Unit 2 experienced a secondary transient which resulted in operation of the cross-under relief valves (CURVs) causing portions of the turbine building siding to become detached. Pieces of siding contacted conductors of two transformers causing faults and lockouts of the 'A' and 'C' reserve station service transformers (RSST) which resulted in an undervoltage (UV) condition on the 1H, 1J, and 2J 4160V emergency buses and subsequent automatic starting and loading of the #1 and #3 emergency diesel generators (EDG) to restore power to the 1H and 2J buses. The #3 EDG can supply the 1J or 2J emergency buses and preferentially loads the 2J bus on a UV involving both buses. Additionally, the same UV conditions automatically started the alternate AC diesel generator (AAC DG). However during the event, the licensee could not load the AAC DG onto the 1J emergency bus as designed by closing the 1J normal supply breaker, 15J8, from the main control room. The trip coil to breaker 15J8 was found energized due to the lockout relays associated with the faults on the RSSTs. With the trip coil energized, the breaker could not be closed. Additional details associated with the cause of the event are contained in NRC Special Inspection Report 05000280, 281/2006011.

The inspectors independently verified that the cause of the energized breaker trip coil was a breaker circuit design deficiency. Furthermore, the design deficiency would result in a similar breaker condition for station blackouts which involve a fault on one or more of the RSSTs. The inspectors also reviewed the licensee's design basis of the AAC DG which was installed to meet the licensing basis function of 10 CFR 50.63, which states in part, "If the AAC source can be demonstrated by test to be available to power the shutdown buses within 10 minutes of the onset of a station blackout, then no coping analysis is required." The inspectors verified the licensee had no coping analysis and consulted with additional NRC technical staff regarding the regulatory requirements. Since the operators would have to manually reset the lockout relay(s) at the switchgear room, which would take more than 10 minutes, the inspectors concluded that the design deficiency involving the breaker prevented the licensee from complying with the ACC DG's licensing basis function. As described in the Special Inspection Report, the licensee installed a modification to correct the breaker circuit design deficiency.

Analysis: The inspectors determined that the licensee's failure to properly design the breaker control circuitry to allow loading of an emergency bus by the AAC DG within 10 minutes was contrary to the requirements of 10 CFR 50.63 and therefore, a performance deficiency or finding. The inspectors consulted inspection manual chapter (IMC) 0612 and determined that the finding is more than minor because it impacted the Mitigating Systems objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage) and the related attribute of design control. An NRC Region II Senior Reactor Analyst conducted a Phase 3 evaluation. The evaluation concluded that the performance deficiency was of very low safety significance (Green). The key assumptions associated with the evaluation were:

- The performance deficiency would only cause an inability for the AAC DG to energize a safety bus for plant-centered and weather initiated Losses of Offsite Power.

- Recovery from the performance deficiency after four hours was credible. Such a recovery would allow the AAC DG to energize a safety bus.

The dominant core damage sequences involved the Loss of Offsite Power followed by failure of the other Emergency Diesel Generators via various mechanisms without recovery of offsite power within four hours or recovering a diesel generator within that same time period.

Since the design deficiency occurred at the time of the installation of the AAC DG, the causes of the design deficiency were not considered as reflective of current performance deficiencies, and a cross-cutting aspect is not applicable.

Enforcement: 10 CFR 50.63, states, in part, "If the AAC source can be demonstrated by test to be available to power the shutdown buses within 10 minutes of the onset of a station blackout, then no coping analysis is required." Contrary to the above, no coping analysis was performed and the test used by the licensee to demonstrate the ability to meet the required 10 minutes was determined to be inadequate, in that, a breaker control circuit design deficiency was revealed by the October 7, 2006, event. For station blackouts with RSST fault(s), the AAC DG would not be available to power shutdown buses until relays in a remote location from the control room were reset. Because this finding is of very low safety significance and is in the licensee's corrective action program as CR002805, this violation is being treated as an NCV, consistent with Section

VI.A of the NRC Enforcement Policy: NCV 05000280, 281/2007002-06, Breaker Control Circuit Design Deficiency Results in Failure to Supply Emergency Bus 1J.

4OA6 Meetings, Including Exit

.1 Exit Meeting Summary

On April 11, 2007, the resident inspectors presented the inspection results to Mr. Sloane and other members of his staff who acknowledge the findings. On April 23, 2007, the inspection results associated with two additional unresolved items were discussed with Mr. Jernigan and his staff.

The inspectors confirmed that proprietary information was not provided or examined during the inspection.

.2 Annual Assessment Meeting Summary

Subsequent to the end of this inspection period, on April 23, 2007, the NRC's Chief of Reactor Projects Branch 5 and the Resident Inspector assigned to the Surry Power Station met with Virginia Electric and Power Company to discuss the NRC's Reactor Oversight Process (ROP) and the NRC's annual assessment of Surry's safety performance for the period of January 1 through December 31, 2006. The major topics addressed were the NRC's assessment program and the results of the Surry assessment. Attendees included Surry site management, members of site staff, and corporate management.

This meeting was open to the public. The presentation material used for the discussion and the list of attendees is available from the NRC's document system (ADAMS) as accession numbers ML071160013 and ML071160286, respectively. ADAMS is accessible from the NRC Web site at http://www/nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

4OA7 Licensee-Identified Violations

The following finding of very low significance was identified by the licensee and is a violation of NRC requirements which meets the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as a Non-Cited Violation (NCV).

- 10 CFR 50.65(a)(4), requires, in part, that before performing maintenance activities, the licensee shall access and manage the increase in risk that may result from the proposed maintenance activity. Contrary to the above, on February 4, 2007, the licensee failed to access the risk of removing the Alternate AC (AAC) diesel generator (DG) from service. The licensee failed to recognize that tagging out the breaker, 0-ACC-BKR-04M1-2, open for maintenance rendered the AAC DG inoperable. The licensee discovered the problem on February 5 and returned the AAC DG to service. In accordance with example (e) of Section 7 of Inspection Manual Chapter (IMC) 0612, Appendix E, “Examples of Minor Issues,” the issue is more than minor. This finding was identified in the licensee’s corrective action program as CR007157, 007241, and 007561. This finding is of very low safety significance because the actual risk change occurred for a short period of time and the change in risk was not large.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

M. Adams, Director, Nuclear Station Safety and Licensing
M. Crist, Manager, Operations
B. Garber, Supervisor, Licensing
J. Grau, Manager, Nuclear Oversight
E. Hendrixson, Director, Site Engineering
D. Jernigan, Site Vice President
L. Jones, Manager, Radiation Protection and Chemistry
C. Luffman, Manager, Protection Services
R. Simmons, Manager, Outage and Planning
K. Sloane, Director, Nuclear Station Operations and Maintenance
B. Stanley, Manager, Maintenance
M. Wilson, Manager, Training

NRC

E. Guthrie, Chief, Branch 5, Division of Reactor Projects, Region II

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000280,281/2007002-01	URI	Potential Failure to Properly Reactivate an RO License (Section 1R11.1)

Opened and Closed

05000281/2007002-02	NCV	Failure to Include the Pressurizer Power Operated Relief Valve in the On-line Risk Assessment Prior to Other Maintenance Activities (Section 1R13)
05000281/2007002-03	NCV	Failure to Follow Procedures During Maintenance Resulting in a Stuck Control Rod (Section 4OA2)
05000281/2007002-04	NCV	Inadequate Procedural Instructions Results in Inoperable Charging Pump Component Cooling Water System (Section 4OA3.1)

05000280,281/2007002-05	NCV	Failure to Perform an Adequate Extent of Condition Review for Unit 1 June 29, 2006, Turbine Building Damage Event (Section 4OA3.3)
05000280, 281/2007002-06	NCV	Breaker Control Circuit Design Deficiency Results in Failure to Supply Emergency Bus 1J (Section 4OA3.4)

Closed

05000281/2006-001-00	LER	Charging Pump Component Cooling Water System Inoperable Due to Inadequate Venting (Section 4OA3.1)
05000280/2006-001-00	LER	Voluntary Report - Motor Driven AFW Pump High flow condition with Low Steam Generator Pressure (Section 4OA3.2)
05000280, 281/2006011-03	URI	Review of Licensee Corrective Actions to Preclude Recurrence of Damage From Cross-under Relief Valve Actuations (Section 4OA3.3)
05000280, 281/2006011-02	URI	Breaker Design Deficiency Prevents the AAC DG from being Loaded as Designed (Section 4OA3.4)

LIST OF DOCUMENTS REVIEWED

Section 1R04: Equipment Alignment

Plant Procedures
0-OP-SW-002A, Emergency Service Water System Alignment
1-OP-SI-001, Filling the LHSI and OSRS Pump Seal Head Tank
1-OP-CH-001/2/3, CVCS Operations
0-OP-CS-001/2/3, Containment Spray System Operations

Plant Drawings
11448-FM-071A
11448-FMC-89C
11448-FMC-88A
11548-FM-84A

Section 1R06: Flood Protection Measures

Plant Procedures
0-ECM-0901-02, Opening and Sealing of Fire Barriers
1-OSP-PL-001, Performance Test of Turbine Building Sump Pumps 1-PL-P-2A, 1-PL-P-2B, 1-PL-P-2C (Turbine Building Sump No. 1) Completed 12/4/06
1-OSP-PL-002, Performance Test of Turbine Building Sump Pumps 1-PL-P-2D, 1-PL-P-2E, 1-PL-P-2F (Turbine Building Sump No. 2) Completed 12/6/06 and 1/31/07
2-OSP-PL-001, Performance Test of Turbine Building Sump Pumps 2-PL-P-2A, 2-PL-P-2B, 2-PL-P-2C (Turbine Building Sump No. 3) Completed 12/12/06

Maintenance Work Orders (MWO)
MWO 731353-01, Dike, Dams, Shields - 0-MPM-1900-01, Periodic inspection of Flood and Spill Protection Dikes, Dams, and Expansion Joint Shields
MWO 733145-01, Flood Det Tst - E45 - 0-EPM-0805-01, Station Flood Detection Testing
MWO 750433-01, Back Flow Preventers - 0-MPM-1900-02, Flood Protection Floor Drain Back Water Stop Valve Replacement (Inside Radiological Controlled Area)
MWO 750434-01, Back Flow Preventers - 0-MPM-1900-02, Flood Protection Floor Drain Back Water Stop Valve Replacement
MWO 759977-04, Seal Manhole #3 penetration 01-EP-MH-3

Section 1R11: Licensed Operator Requalification

Procedures
Licensed Operator Requalification Program, Program Guide, Dominion Nuclear Training, April 2001.
FIG-01, LOR Program, Rev 17
FIG-02, LORP Sample Plan Basis Document, Revision 7.
FIG-03, Program Change Control, Revision 12
FIG-04, Licensed Operator/STA Trending Program, Revision 2
FIG-05, Conduct of Annual LORP Examinations, Revision 22.
FIG-06, Simulator Performance Mode Scenarios, Revision 3)
FIG-07, Active License Status Maintenance, Revision 4
FIG-10, Simulator Instructor Initial Continuing Training Programs, Rev 5
FIG-17, Critical Task Development, Rev 0
FIG-18, Examination Security, Rev 1
FIG-20, Simulator Training Needs Assessment, Rev 0
FIG-21, Simulator Scenario Based Testing, Rev 1
FIG-22, Clarification of Operations Standards, Rev 0
FIG-23, Instructor Guidelines, Rev 23
FIG-24, Continuing Training Operations Program Exam Failure, Rev 1
FIG-25, Just-In-Time Training, Rev 1
FIG-27, Periodic Review of Key Operator Response Times, Rev 2
TRCP-006, Nuclear Training Program Implement, Revision 12.
TRCP-3007, Simulator Performance Testing, Revision 0.
TRCP-3006, Simulator Configuration Management, Revision 5.

TRCP-3002, Simulator Modification Record (SMR), Revision 9.
TRCP-0007, Nuclear Training Program Evaluation, Revision 17.

Simulator Tests
0-SPS-ANSI-09, Operability Test – 100% Steady State One Hour Run, 2006.
0-SPS-ANSI-08, Operability Test – 75% Steady State One Hour Run, 2006.
0-SPS-ANSI-04, Operability Test – 25% Steady State One Hour Run, 2006.
0-SPS-ANSI-10, Operability Test – Manual Reactor Trip, 2006.
0-SPS-ANSI-11, Operability Test – Simultaneous Trip of All Feedwater Pumps, 2006.
0-SPS-ANSI-12, Operability Test – Simultaneous Trip of All MS Isol. Valves, 2006.
0-SPS-ANSI-13, Operability Test – Simultaneous Trip of All Reactor Coolant Pumps, 2006.
0-SPS-ANSI-14, Operability Test – Trip of a Single Reactor Coolant Pump, 2006.
RQ-07.1-SE-8, Surry Simulator Scenario Based Testing Checklist, 01/19/2007.
RQ-07.1-SE-7, Surry Simulator Scenario Based Testing Checklist, 01/18/2007.

Simulator Scenarios
RQ-07.1-SE-7, Containment Pressure Detector Failure, Loss of Condenser Vacuum, ATWS, LBLOCA, HI HI CLS Failure, CS Pump Failure, Automatic / Manual RMT Failure, 12/22/2006, Rev. 0.
RQ-07.1-SE-8, Dropped Rod Followed By SGTR, 12/22/2006, Rev. 0.

Job Performance Measures
JPM 26.09, Rev 2, Place a Main Feedwater Regulating Valve on the Jacking Device
JPM 41.01, Rev 7, Locally Emergency Borate Using 1-CH-MOV-1350
JPM 18.02B, Rev 11, Transfer Semi-Vital Bus Power Suppy
JPM 52.10 Rev 13, Isolate/Vent Safety Injection Accumulators
JPM 65.01, Rev 8, Respond to a Continuous Rod Insertion

Condition Reports Resulting From Inspection
CR006467, Operator License Reactivation Procedure Non-compliance, 01/17/2007

LOR Training Feedback
RQ-5.3 (4/12/05-7/8/05)
RQ-5.4 (7/26/05-9/9/05)
RQ-5.5 (9/13/05-10/14/05)
RQ-5.6 (10/18/05-11/18/05)
RQ-5.7 (11/29/05-12/23/05)
RQ-6.1 (1/10/06-2/1/06)
RQ-6.3 (4/4/06-6/9/06)
RQ-6.4 (6/20/06-7/21/06)
RQ-6.5 (8/1/06-9/1/06)
RQ-6.6 (9/12/06-10/13/06)

Miscellaneous
Licensed Operator Medical Records (21)
Licensed Operator Training Feedback (2005 and 2006)
Reactivation Records (4)

Licensed Personnel Quarterly On-Shift Time Records (8)
Written Exam Remediation Packages (9)
Simulator Remediation Package (1)

Section 1R12: Maintenance Effectiveness

System Health Report for Safety Injection System
System Health Report for Check Valves
Surry Check Valve Condition Monitoring Program Condition Monitoring Plan, Groups 2-071a and 2-071e
ET S 03-004, Rev.0, Backleakage Acceptance Criteria for Check Valves 1/2-SI-241, 242, & 243 in 1/2-OPT-SI-014, Surry Power Station, Unit 1 & 2
Engineering Transmittals 95-0455, Rev. 0, Mechanical Agitation of Check Valves, Surry Power Station, Unit 1 & 2
CME 00-0039, Rev. 0, Bases for Acceptance Criteria in 1/2-OPT-SI-14, Surry Power Station, Unit 1 & 2
System Health Report for Auxiliary Feedwater System

Condition Reports (CR)
CR003298 - Seat Damage on 2-SI-82 Check Valve
CR004107 - 2-SI-241 was mechanically agitated during the performance of 2-OPT-SI-014
CR006843 - Increased U2 RCS (Reactor Coolant System) leakage, suspect U2 RWST (Refueling Water Storage Tank) inleakage
CR008237 - Unit 2 Steam Driven Aux Feed Pump
CR001972 - Suspect leakby of 1-MS-PCV-102A/ and or B

Plant Issues
S-2000-2623, S-2000-2579, S-2001-2623, S-2005-2087, S-2204-2018, S-2005-2672, S-2006-2164, S-2002-3725

Maintenance Work Orders (MWO)
301153-01, 3015335-01, 409274-02, 439369-01, 600156-01, 758334-01, 769000-01, 768999-01

Plant Procedures
2-OPT-SI-002, Refueling Test of the Low Head Safety Injection Check Valves to the Cold Legs
2-OPT-SI-014, Cold Shutdown Test of SI Check Valves to RCS Hot and Cold Legs
2-OSP-SI-004, SI Check Valve Troubleshooting
0-MCM-0417-01, Velan Swing Check Valves Inspection and Overhaul
1(2)-OP-FW-001/2/3, Motor Driven AFW Pumps Startup and Shutdown

Vendor Technical Manual
38-V637-00005, Maintenance Manual VEL-FBBM for 2 1/2" - 24" Forged Bolted Bonnet Gate and Globe Valves and Bolted Cover Check Valves

Plant Drawings
11548-FM-089A
11548-FM-089B
11448-FM-068A
11548-FM-086A

Section 1R20: Refueling and Other Outage Activities (Unit 1)

Plant Procedures
1-GOP-2.2, Unit shutdown, Less Than 30% to HSD
1-GOP-2.3, Unit Shutdown, Stabilized at HSD
1-GOP-1.4, Unit Startup, HSD to 2% Reactor Power
1-OP-RX-004, The Calculation of Estimated Critical Condition
1-OP-RX-006, Withdrawal of the Control Banks to Critical Condition