UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
245 PEACHTREE CENTER AVENUE NE, SUITE 1200
ATLANTA, GEORGIA 30303-1257

July 29, 2010

Mr. David A. Heacock
President and Chief Nuclear Officer
Virginia Electric and Power Company
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060

SUBJECT: NORTH ANNA POWER STATION – NRC INTEGRATED INSPECTION REPORT 05000338/2010003, 05000339/2010003, AND 07200056/2010001

Dear Mr. Heacock:

On June 30, 2010, the U. S. Nuclear Regulatory Commission (NRC) completed an inspection at your North Anna Power Station Units 1 and 2 and the North Anna Independent Spent Fuel Storage Installation. The enclosed integrated inspection report documents the inspection findings which were discussed on July 26, 2010, with Mr. Michael Crist and other members of your staff.

The inspection examined activities conducted under your licenses as they related 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.

This report documents one self-revealing finding and one NRC-identified finding of very low safety significance (Green) which were determined to be violations of NRC requirements. The report also documents one apparent violation for which significance is yet to be determined. However, because of the very low safety significance of these issues and because they were entered into your corrective action program, the NRC is treating these as non-cited violations (NCV) consistent with Section VI.A.1 of the NRC Enforcement Policy. If you wish to contest these non-cited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington DC 20555-001; 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 North Anna Power Station. In addition, if you disagree with the cross-cutting aspect assigned to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, RII, and the NRC Senior Resident Inspector at the North Anna 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 the 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/

Gerald J. McCoy, Chief
Reactor Projects Branch 5
Division of Reactor Projects

Docket Nos. 50-338, 50-339, 72-056
License Nos. NPF-4, NPF-7, SNM-2507

Enclosure: Inspection Report 05000338/2010003, 05000339/2010003, and 07200056/2010001
w/ Attachment: Supplemental Information

cc w/ encl. (See next page)

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Letter to David A. Heacock from Gerald J. McCoy dated July 29, 2010

SUBJECT: NORTH ANNA POWER STATION – NRC INTEGRATED INSPECTION REPORT 05000338/2010003, 05000339/2010003, AND 07200056/2010001

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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-338, 50-339, 72-056
License Nos.:	NPF-4, NPF-7, SNM-2507
Report No:	05000338/2010003, 05000339/2010003, and 07200056/2010001
Licensee:	Virginia Electric and Power Company (VEPCO)
Facility:	North Anna Power Station, Units 1 & 2, and North Anna Independent Spent Fuel Storage Installation
Location:	1022 Haley Drive Mineral, Virginia 23117
Dates:	April 1, 2010 through June 30, 2010
Inspectors:	J. Reece, Senior Resident Inspector R. Clagg, Resident Inspector J. Rivera-Ortiz, Senior Reactor Inspector, Section 1R08 (SGISI only) R. Williams, Reactor Inspector, Sections 1R08, 4OA5.3 R. Carrion, Senior Reactor Inspector, Sections 4OA5.4
Approved by:	Gerald J. McCoy, Chief Reactor Projects Branch 5 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000338/2010003, 05000339/2010003, 07200056/2010001; 04/01/2010 – 06/30/2010; North Anna Power Station, Units 1 and 2 and the North Anna Independent Spent Fuel Storage Installation: Maintenance Effectiveness, Identification and Resolution of Problems, and Event Follow-up.

The report covered a 3 month period of inspection by resident inspectors and reactor inspectors from the region. Two findings were identified and were determined to be non-cited violations (NCVs). One finding was determined to be an apparent violation for which significance is yet to be determined. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). The cross-cutting aspect was determined using IMC 0310, "Components within the Cross Cutting Areas." 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.

<u>NRC Identified and Self-Revealing Findings</u>

Cornerstone: Initiating Events

- <u>TBD</u>. An apparent violation of Technical Specifications 5.4.1a was identified by the inspectors for the failure to adequately implement procedural requirements resulting in operation of the 'A' reactor coolant system (RCS) pump (RCP) beyond the motor high bearing temperature limit of 195 degF for approximately 10 minutes. The licensee entered this problem into their corrective action program as corrective action 170278 associated with condition report 382725.

  The inspectors determined that the failure to implement an alarm response procedure to trip the 'A' RCP in a timely manner was a performance deficiency. The finding was more than minor because it could be reasonably viewed as a precursor to a significant event due to RCP motor operation in an unknown condition of bearing performance in which the melting of Babbitt material can lead to excessive shaft vibrations and consequent adverse impact on RCP seal performance leading to a seal loss of coolant accident. In accordance with NRC Inspection Manual Chapter 0609, "Significant Determination Process," the inspectors performed a Phase 1 analysis and determined the finding required a phase 2 evaluation because assuming worst case degradation the TS limit for RCS leakage would have been exceeded. The significance of this finding is to be determined (TDB) pending completion of a phase 3 evaluation. This finding involved the cross-cutting area of human performance, the component of decision making and the aspect of decision communications, H.1(c), because a reactor operator failed to communicate the loss of component cooling to the RCP motors to the senior reactor operator which led to a failure to trip the 'A' RCP on exceeding the motor bearing high temperature limit. (Section 4OA3.2)

Cornerstone: Mitigating Systems

- Green: A self-revealing non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," was identified for the failure to promptly identify and correct a condition adverse to quality for the breaker associated with 2-RH-MOV-2700, Loop 'A' Hot Leg to RH Pump Isolation Valve. The licensee entered this problem into their corrective action program as condition report 372940.

  The inspectors determined that the licensee's failure to promptly correct a known condition adverse to quality, as required by 10 CFR 50, Appendix B, Criterion XVI, was a performance deficiency. The inspectors reviewed IMC 0612, Appendix E and determined the finding was more than minor because it was similar to examples 4d and 4f. The phase 1 screening resulted in a need to perform phase 2 and phase 3 evaluations due to the finding resulting in the loss of mitigating function, specifically the ability to perform decay heat removal. A phase 3 analysis was performed by a regional senior risk analyst in accordance with the guidance of NRC Inspection Manual Chapter 0609 Appendix A. The significance determination process phase 3 risk evaluation resulted in a risk increase for the finding <1E-6 for core damage frequency (CDF) and <1E-7 for large early release frequency (LERF). The dominant sequence involved a steam generator tube rupture, followed by failure of the RHR system, and failure of the operators to refill the emergency condensate storage tank to continue secondary side cooling. The analysis assumed the operators, given the additional time while cooling the core using the secondary side, would be able to manually open 2-RH-MOV-2700. The finding was characterized as of very low safety significance (Green). The cause of this finding involved the cross-cutting area of problem identification and resolution, the component of corrective action program, and the aspect of implementation of corrective action (P.1(d)) because the licensee failed to correct the safety issue that existed with 2-RH-MOV-2700 in a timely manner, commensurate with its safety significance and complexity. (Section 4OA2.3)

Cornerstone: Barrier Integrity

- Green: A non-cited violation of 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Plants," was identified by the inspectors for the failure to demonstrate that the performance of a Unit 1 pressurizer power operated relief valve (PORV), 1-RC-PCV-1455C, was being effectively controlled through preventative maintenance in accordance with 10 CFR 50.65 (a)(2) and establish performance goals and monitor against those goals in accordance with 10 CFR 50.65 (a)(1). The licensee entered this into their corrective action program as condition report 374734.

  The inspectors determined that the failure to demonstrate that the performance of 1-RC-PCV-1455C, was being effectively controlled through preventative maintenance in accordance with 10 CFR 50.65 (a)(2) was a performance deficiency. The inspectors reviewed Inspection Manual Chapter 0612, Appendix B and determined the finding was more than minor because it affected the Barrier Integrity cornerstone objective of providing reasonable assurance that physical design barriers (e.g. reactor coolant system (RCS) protect the public from radionuclide releases caused by accidents or events. Specifically, RCS equipment and barrier performance, in that the PORVs provide protection to the RCS by preventing brittle fracture at low temperature conditions and protect RCS integrity at high temperature conditions.

The inspectors determined that this performance deficiency was a separate consequence of the degraded performance associated with 1-RC-PCV-1455C. Therefore, in accordance with the guidance provided in NRC Inspection Procedure 71111.12, Appendix D, this issue was determined to be a maintenance rule Category II finding and was of very low safety significance (Green). The cause of this finding involved the cross-cutting area of human performance, the component of work practices, and the aspect of human error prevention (H.4(a)), because the licensee failed to verify that the correct maintenance rule function performance criteria was being used to conduct maintenance rule evaluations. (Section 1R12)

Licensee Identified Violations

None

REPORT DETAILS

Summary of Plant Status

Unit 1 began the period at Rated Thermal Power (RTP) and operated at or near RTP for the entire report period.

Unit 2 began the inspection period in a scheduled refueling outage which started March 21, 2010. On April 26, 2010, Unit 2 returned to service but experienced a main generator /reactor trip on April 27, 2010. The unit returned to RTP on May 2, 2010, and continued until May 28, 2010, when the unit experienced a reactor trip during a lightning storm. Unit 2 was back at RTP by June 4, 2010, but experienced another reactor trip during a lightning storm on June 16, 2010. The unit returned to RTP on June 20, 2010, and remained at RTP through the end of the report period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection

.1 Review of Offsite Power and Alternate AC Power Readiness

a. Inspection Scope

The inspectors verified that plant features, and procedures for operation and continued availability of offsite and alternate alternating current (AC) power systems were appropriate.

The inspectors reviewed the licensee’s procedures affecting those areas, and the communications protocols between the transmission system operator and the nuclear power plant to verify that the appropriate information was exchanged when issues arose that could impact the offsite power system. The inspectors evaluated the readiness of the offsite and alternate AC power systems by reviewing the licensee’s procedures that address measures to monitor and maintain the availability and reliability of the offsite and alternative AC power systems.

b. Findings

No findings were identified.

.2 Seasonal Susceptibilities

a. Inspection Scope

The inspectors reviewed the licensee’s adverse weather preparations for hot weather operations, specified in 0-GOP-4.1, “Hot Weather Operations,” Revision 25, and the licensee’s corrective action program (CAP) database for hot weather related issues. The inspectors walked down three risk-significant systems/areas on each unit listed below to verify compliance with the procedural requirements and to verify that the specified actions provided the necessary protection for the structures, systems, or components.

- Unit 1 & 2 motor-driven and turbine-driven auxiliary feedwater (AFW) pump rooms
- Alternate AC (AAC) diesel generator room
- Unit 1 & 2 ‘H’ and ‘J’ emergency diesel generator (EDG) rooms

b. Findings

No findings were identified.

1R04 Equipment Alignment

.1 Partial Walkdowns

a. Inspection Scope

The inspectors conducted three equipment alignment partial walkdowns to evaluate the operability of selected redundant trains or backup systems, listed below, with the other train or system inoperable or out of service. The inspectors reviewed the functional systems descriptions, Updated Final Safety Analysis Report (UFSAR), system operating procedures, and Technical Specifications (TS) to determine correct system lineups for the current plant conditions. The inspectors performed walkdowns of the systems to verify that critical components were properly aligned and to identify any discrepancies which could affect operability of the redundant train or backup system

- Unit 2 'A' service water (SW) header during ‘B’ SW header outage for 2-SW-MOV-208B maintenance
- Unit 2 ‘2J’ EDG during a maintenance outage for ‘2H’ EDG
- Pressurizer heaters Group 1 due to breaker inoperability for Group 4 as required by TS

b. Findings

No findings were identified.

.2 Complete Walkdown

a. Inspection Scope

The inspectors performed a detailed walkdown and inspection of the SW system to the Unit 1 and Unit 2 control room chillers to assess proper alignment and to identify discrepancies that could impact its availability and functional capacity. The inspection included a review of the associated piping supports, valve alignments, and other related components such as expansion joints. Equipment deficiency tags were reviewed and the condition of the system was discussed with the engineering personnel. Documents reviewed are listed in the attachment.

b. Findings

No findings were identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors conducted tours of the six areas listed below that are important to reactor safety to verify the licensee's implementation of fire protection requirements as described in fleet procedures CM-AA-FPA-100, "Fire Protection/Appendix R (Fire Safe Shutdown) Program," Revision 0, CM-AA-FPA-101, "Control of Combustible and Flammable Materials," Revision 1, and CM-AA-FPA-102, "Fire Protection and Fire Safe Shutdown Review and Preparation Process and Design Change Process," Revision 0. The inspectors evaluated, as appropriate, conditions related to: (1) licensee control of transient combustibles and ignition sources; (2) the material condition, operational status, and operational lineup of fire protection systems, equipment, and features; and (3) the fire barriers used to prevent fire damage or fire propagation.

- Containment Unit 2 (fire zone 1-2a / RC-2)
- Main Steam Valve House Unit 2 (includes MG Set Room) (fire zone 17-1a / MSVH-1)
- Cable Vault and Tunnel Unit 1 (includes Control Rod Drive Room and Z-27-1)(fire zone 3-1a / CV & T-1)
- Cable Vault and Tunnel Unit 2 (includes Control Rod Drive Room and Z-27-2)(fire zone 3-2a / CV & T-2)
- Cable Tray Spreading Room Unit 1 (fire zone 4-1b / CSR-1) and Cable Tray Spreading Room Unit 2 (fire zone 4-2b / CSR-2)
- Main and Station Service Transformers (fire zone Z-8C / XFMRS), Security Auxiliary Power Supply Building (fire zone Z-39 / APSB), and AAC Building (fire zone Z-52 / AAC)

b. Findings

No findings were identified.

1R08 Inservice Inspection Activities

From March 28, 2010, to April 2, 2010, the inspectors conducted a review of the implementation of the licensee's Inservice Inspection (ISI) Program for monitoring degradation of the reactor coolant system, steam generator tubes, emergency feedwater systems, risk-significant piping and components and containment systems.

The inspections described in Sections 1R08.1, 1R08.2, 1R08.3, 1R08.4 and 1R08.5 below constituted one inservice inspection sample as defined in Inspection Procedure 71111.08-05.

.1 Piping Systems ISI

a. Inspection Scope

The inspectors observed the following non-destructive examinations (NDEs) mandated by the American Society of Mechanical Engineers (ASME) Code Section XI to evaluate compliance with the ASME Code Section XI and Section V requirements and, if any indications and defects were detected, to evaluate if they were dispositioned in accordance with the ASME Code or an NRC-approved alternative requirement.

- Liquid Penetrant (PT) examination of recirculation spray 12” pipe to valve weld, Class 2
- PT examination of control rod drive mechanism welds No. 43, 48 and 60, Class 1
- Ultrasonic (UT) examination of ten (10) 6” reactor vessel closure studs (removed), Class 1
- UT examination of ten (10) reactor vessel closure head nuts, Class 1
- UT examination of ten (10) reactor vessel closure washers, Class 1
- UT examination of ten (10) reactor vessel concave closure washers, Class 1

During the non-destructive surface and volumetric examinations performed since the previous refuelling outage, the licensee did not identify any recordable indications that were analytically evaluated for continued service. Therefore, no NRC review was completed for this inspection procedure attribute.

The inspectors reviewed the following pressure boundary welds completed for risk-significant systems during the last Unit 2 refuelling outage to evaluate if the licensee applied the preservice non-destructive examinations and acceptance criteria required by the construction Code and the ASME Code Section XI. In addition, the inspectors reviewed the welding procedure specification and supporting weld procedure qualification records to evaluate if the weld procedure(s) were qualified in accordance with the requirements of the Construction Code and ASME Code Section IX.

- WO774483-01 “Replace 1B RCP Seal Injection Inlet Header Drain Valve, 2-CH-288”, Class 1

The inspectors reviewed the results of the visual examination (VE) for the bottom-mounted instrument penetrations to ensure examinations were being performed in accordance with the requirements of ASME Code Case N-722-1 and 10 CFR 50.55a(g)(6)(ii)(E).

b. Findings

No findings were identified.

.2 Reactor Pressure Vessel Upper Head Penetration Inspection Activities

a. Inspection Scope

For the Unit 2 vessel upper head, no examination was required pursuant to 10 CFR 50.55a(g)(6)(ii)(D) for the current refueling outage. The previous bare metal visual (BMV) examination for the vessel upper head was performed during the March of 2007 refueling outage and next scheduled BMV examination is during the September 2011 refueling outage. The previous volumetric examination for the vessel upper head was performed during the March of 2007 refueling outage and next scheduled volumetric examination is during the March 2013 refueling outage. The inspectors also reviewed the calculation for the number of effective degradation years (EDY). The EDY for the vessel upper head was 6.724 years.

b. Findings

No findings were identified.

.3 Boric Acid Corrosion Control (BACC)

a. Inspection Scope

The inspectors performed an independent walkdown of portions of borated systems which recently received a licensee boric acid walkdown and evaluated if the licensee's BACC visual examinations emphasized locations where boric acid leaks could cause degradation of safety-significant components.

The inspectors reviewed the following licensee evaluations of reactor coolant system components with boric acid deposits to evaluate if degraded components were documented in the corrective action system. The inspectors also evaluated the corrective actions for any degraded reactor coolant system components against ASME Code Section XI and other licensee committed documents:

- CA089637, "Other to Eng to evaluate 2-CH-FCV-2160 in accordance with the BACC Program"
- CA158683, "Perform an evaluation to determine the extent of any boric acid degradation"

The inspectors reviewed the following corrective actions related to evidence of boric acid leakage to evaluate if the corrective actions completed were consistent with the requirements of the ASME Code Section XI and 10 CFR Part 50, Appendix B, Criterion XVI.

- CR117130, "Possible boric acid deposit on pipe in overhead of 24' containment"
- CR117205, "3CHS*V645, boric acid deposit observed on packing gland area"
- CR117211, "3CHS*V987, boric acid deposit observed on packing gland area"
- CR118048, "2-QS-ICV-3022 has dry boric acid buildup on the downstream vent gooseneck"

b. Findings

No findings were identified.

.4 Steam Generator (SG) Tube Inspection Activities

a. Inspection Scope

The inspectors reviewed the Unit 2 eddy current (EC) examination activities performed in SG B and evaluated them against the licensee's TS, commitments made to the NRC, ASME Section XI, and Nuclear Energy Institute (NEI) 97-06 (Steam Generator Program Guidelines). The inspectors conducted the following inspection activities:

- Reviewed EC examination status reports and discussed them with the lead Level III analyst to ensure that all tubes with relevant indications were appropriately screened for in-situ pressure testing. In particular, the inspectors assessed whether assumed NDE flaw sizing accuracy was consistent with data from the Electric Power Research Institute examination technique specification sheets (ETSS) or other applicable performance demonstrations. None of the steam generator (SG) tubes examined met the criteria for in-situ pressure testing.

- Reviewed the last Condition Monitoring and Operational Assessment report in conjunction with the EC status reports on indications to assess the licensee prediction capability for maximum tube degradation.
- Reviewed the latest Degradation Assessment report and vendor's examination plan to assess the scope of the inspection and verify it included potential areas of tube degradation identified in prior outage SG tube inspections, industry operating experience, and NRC generic communications. The inspectors also verified that appropriate inspection scope expansion criteria were applied based on inspection results of active and new degradation mechanisms. Based on the EC examination results, no new degradation mechanisms were identified and no EC scope expansion was required.
- Reviewed the licensee's repair criteria to ensure it was consistent with plant TS.
- Reviewed the primary-to-secondary leakage (e.g., SG tube leakage) history for the last operating cycle. The inspectors found that primary-to-secondary leakage was below three gallons per day, or the detection threshold, during the previous operating cycle.
- Reviewed documentation to ensure that data analysts, EC probes, and equipment configurations were qualified to detect the active and potential SG tube degradation mechanisms in accordance with the applicable industry standards.
- Reviewed a sample of site-specific ETSSs to ensure that their qualification was consistent with Appendix H, Performance Demonstration for Eddy Current Examination, of the Electric Power Research Institute Pressurized Water Reactor Steam Generator Examination Guidelines, Revision 7.
- Directly observed EC data acquisition for 13 tubes examined in the top-of-tubesheet expansion transition with Rotating Pancake Coil probe and 30 tubes examined full length with Bobbin probe.
- Reviewed EC data with a qualified analyst for tubes R28C84, R28C82, R37C28, R9C69, R10C66, and R33C44.
- Directly observed portions of the Foreign Object Search and Removal activities on the secondary side of SG B, in particular for the top-of-tubesheet region.

b. Findings

No findings were identified.

.5 Identification and Resolution of Problems

a. Inspection Scope

The inspectors performed a review of ISI/SG related problems entered into the licensee's corrective action program and conducted interviews with licensee staff to determine if;

- the licensee had established an appropriate threshold for identifying ISI/SG related problems;
- the licensee had performed a root cause (if applicable) and taken appropriate corrective actions; and
- the licensee had evaluated operating experience and industry generic issues related to ISI and pressure boundary integrity.

The inspectors performed these reviews to evaluate 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 Attachment to this report.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program

a. Inspection Scope

The inspectors observed an operator requalification simulator scenario which involved a reactor coolant system (RCS) leak, a partial loss of control room annunciators, a loss of instrument air outside of containment, a pressurizer pressure instrument failure resulting in a failed open pressurizer power operated relief valve, and a control rod ejection causing a small break loss of coolant accident.

The inspectors observed crew performance in terms of communications; ability to take timely and proper actions; prioritizing, interpreting, and verifying alarms; correct use and implementation of procedures, including the alarm response procedures; timely control board operation and manipulation, including high-risk operator actions; and oversight and direction provided by the shift supervisor, including the ability to identify and implement appropriate TS actions. The inspectors observed the post training critique to determine that weaknesses or improvement areas revealed by the training were captured by the instructor and reviewed with the operators.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

For the two equipment issues listed below, the inspectors evaluated the effectiveness of the respective licensee's preventive and corrective maintenance. The inspectors performed walkdowns of the accessible portions of the systems, performed in-office reviews of procedures and evaluations, and held discussions with licensee staff. The inspectors compared the licensee’s actions with the requirements of the Maintenance Rule (10 CFR 50.65), and licensee procedure ER-AA-MRL-10, “Maintenance Rule Program,” Revision 4.

- CR328709, “MRE010495, MRule evaluation to engineering for 1-RC-PCV-1455C went closed with control switch in ‘open’”
- CR355000, “Failed diaphragm found torn at bolt holes”

b. Findings

Failure to Demonstrate Effective Control of Pressurizer PORV 1-RC-PCV-1455C Performance in Accordance with the Maintenance Rule

Introduction: A Green, non-cited violation (NCV) of 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Plants," was identified by the inspectors for failure to demonstrate that the performance of Unit 1 pressurizer power operated relief valve (PORV), 1-RC-PCV-1455C, was being effectively controlled through preventative maintenance in accordance with 10 CFR 50.65 (a)(2).

Description: Between February 18, 2009, and October 24, 2009, the licensee incurred three failures of 1-RC-PCV-1455C. The inspectors reviewed licensee procedure ER-AA-MRL-100, "Implementing Maintenance Rule", Revision 1, and other licensee documents regarding these failures.

The inspectors reviewed maintenance rule evaluation (MRE) 010337 for the February 18, 2009, failure of a comparator card for input control logic to 1-RC-PCV-1455C and noted that the licensee classified this failure as a maintenance rule functional failure (MRFF) and a maintenance preventable functional failure (MPFF), and determined that a 10 CFR 50.65 (a)(1) evaluation was not warranted.

The inspectors reviewed MRE010495, Revision 0, for the March 26, 2009, failure of the valve actuator diaphragm for 1-RC-PCV-1455C and noted that the licensee classified this failure as a MRFF, but not a MPFF, and determined that a 10 CFR 50.65 (a)(1) evaluation was not warranted. The inspectors identified that the licensee improperly utilized a draft version of apparent cause evaluation (ACE) 017534, "ACE to Eng to investigate the failed 1-RC-PCV-1455C diaphragm." This issue was documented in CR368436, "MRE010495 needs revision," and the MRE was subsequently revised to classify this failure as a MPFF. However, the inspectors determined that the licensee incorrectly concluded a 10 CFR 50.65 (a)(1) evaluation was not required because they assumed the performance criteria for the affected maintenance rule function, RC006, was 2 MPFFs per monitoring period.

The inspectors reviewed MRE011280 for the October 24, 2009, failure of a solenoid operated valve (SOV) which prevented 1-RC-PCV-1455C from opening on demand and noted that the licensee classified this failure as a MRFF, a MPFF, and determined that a 10 CFR 50.65 (a)(1) evaluation was not warranted in part because "MR Performance Monitoring Criteria has not been exceeded."

The inspectors reviewed maintenance rule function RC006, which was applicable to all three MREs discussed above, and identified that it listed the performance criteria for the pressurizer PORVs to be "1 MPFF per PORV" in a 2 year period of time. The inspectors determined that the licensee failed to recognize that the two MPFFs described above exceeded the performance criteria of RC006, and did not demonstrate that the performance of 1-RC-PCV-1455C was being effectively controlled through preventive maintenance as required by the Maintenance Rule. The licensee documented this issue in CR374734, "Maintenance rule evaluations used incorrect performance criteria," and placed 1-RC-PCV-1455C in 10 CFR 50.65 (a)(1) monitoring status.

Analysis: The inspectors determined that the failure to demonstrate that the performance of 1-RC-PCV-1455C, was being effectively controlled through preventative maintenance in accordance with 10 CFR 50.65 (a)(2) was a performance deficiency. The inspectors reviewed Inspection Manual Chapter (IMC) 0612, Appendix B and determined the finding was more than minor because it affected the Barrier Integrity cornerstone objective of providing reasonable assurance that physical design barriers (e.g. RCS) protect the public from radionuclide releases caused by accidents or events. Specifically, RCS equipment and barrier performance, in that the pressurizer PORVs provide protection to the RCS by preventing brittle fracture at low temperature conditions and protect RCS integrity at high temperature conditions. The inspectors determined that this performance deficiency was a separate consequence of the degraded performance associated with 1-RC-PCV-1455C. Therefore, in accordance with the guidance provided in NRC Inspection Procedure 71111.12, Appendix D, this issue was determined to be a maintenance rule Category II finding and was of very low safety significance (Green). The cause of this finding involved the cross-cutting area of human performance, the component of work practices, and the aspect of human error prevention (H.4(a)) because the licensee failed to verify that the correct maintenance rule function performance criteria was being used to conduct maintenance rule evaluations.

Enforcement: 10 CFR 50.65 (a)(1) requires, in part, that holders of an operating license shall monitor the performance or condition of structures, systems, and components (SSCs) within the scope of the monitoring program as defined in 10 CFR 50.65 (b) against established goals, in a manner sufficient to provide reasonable assurance that such SSCs are capable of fulfilling their intended functions. 10 CFR 50.65 (a)(2) states, in part, that monitoring as specified in 10 CFR 50.65 (a)(1) is not required where it has been demonstrated that the performance or condition of an SSC is being effectively controlled through the performance of appropriate preventive maintenance, such that the SSC remains capable of performing its intended function. Contrary to this, from March 26, 2009, to October 24, 2009, the licensee failed to identify and properly account for two MPFFs of 1-RC-PCV-1455C, which demonstrated that the performance or condition of this component was not being effectively controlled through the performance of appropriate preventive maintenance and, as a result, that goal setting and monitoring was required. Because the finding is of very low safety significance (Green) and because the problem has been entered into the licensee's CAP as CR374734, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000338/2010003-01, Failure to Demonstrate Effective Control of Pressurizer PORV 1-RC-PCV-1455C Performance in Accordance with the Maintenance Rule.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors evaluated, as appropriate, the six activities listed below for the following: (1) effectiveness of the risk assessments performed before maintenance activities were conducted; (2) management of risk; (3) upon identification of an unforeseen situation, necessary steps were taken to plan and control the resulting emergent work activities; and (4) maintenance risk assessments and emergent work problems were adequately identified and resolved. The inspectors verified that the licensee was in compliance with the requirements of 10 CFR 50.65 (a)(4) and the data output from the licensee's safety monitor associated with the risk profile of Units 1 and 2.

- Emergent entry into 0-AP-8, "Response to Grid Instability," Revision 6, due to loss of real time contingency analysis on April 4, 2010
- Entry into 0-AP-41, "Severe Weather Conditions," Revision 49, with service water pump house (SWPH) missile shield door unable to be closed
- Loss of 'B' RSST and respective 2H emergency bus during a lightning storm.
- Extension of 2H EDG unavailability time due to maintenance issues occurring during planned major overhaul
- Extension of AAC diesel generator unavailability due to emergent issue during scheduled maintenance
- Emergent entry into 0-AP-8, "Response to Grid Instability," Revision 6, due to heavy load action issued by dispatcher on June 24, 2010

b. Findings

No findings were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed six operability evaluations, listed below, affecting risk-significant mitigating systems, to assess, as appropriate: (1) the technical adequacy of the evaluations; (2) whether continued system operability was warranted; (3) whether other existing degraded conditions were considered as compensating measures; (4) whether the compensatory measures, if involved, were in place, would work as intended, and were appropriately controlled; and (5) where continued operability was considered unjustified, the impact on TS Limiting Conditions for Operation and the risk significance in accordance with the Significant Determination Process (SDP). The inspectors' review included a verification that determinations of operability were made as specified by Procedure OP-AA-102, "Operability Determination," Revision 5.

- CR376673, review of Operability Determination (OD) 000371, "Loose Fastener Evaluation for U-1 Removable Sections"
- OD000370, "Vertical lateral box frame on 4-WS-468-151-Q3, 4B MCR chiller not supported"
- OD000363, "Potential for Fuel Rod Channel Closure in AREVA Advanced Mark-BW Fuel"
- CR378311, Unit 1 electrical penetration room fire protection CO_2 barrier (approximately 1 ft^2) does not conform to drawing requirements
- CR381198, review of OD000373, "Please evaluate the operability of the 1A2 Service Water Spray Array"
- OD000377, "Provide an operability determination for 2-HV-F-40B which has a hole in the suction boot"

b. Findings

No findings were identified.

1R19 Post Maintenance Testing

a. Inspection Scope

The inspectors reviewed five post maintenance test procedures and/or test activities for selected risk-significant mitigating systems listed below, to assess whether: (1) the effect of testing on the plant had been adequately addressed by control room and/or engineering personnel; (2) testing was adequate for the maintenance performed; (3) acceptance criteria were clear and adequately demonstrated operational readiness consistent with design and licensing basis documents; (4) test instrumentation had current calibrations, range, and accuracy consistent with the application; (5) tests were performed as written with applicable prerequisites satisfied; (6) jumpers installed or leads lifted were properly controlled; (7) test equipment was removed following testing; and (8) equipment was returned to the status required to perform in accordance with Virginia Power Administrative Procedure (VPAP) -2003, “Post Maintenance Testing Program,” Revision 13.

- WO 59101753651, Replace AVR Card
- WO 59101999711, Assemble valve and install new operator on 2-SI-MOV-2867B, BIT inlet isolation valve
- WO 59101776417, Seal inspection and replacement on 2-RC-P-1B, ‘B’ reactor coolant pump
- WO 59101688396, Actuator diaphragm replacement on 1-SW-TCV-102B, ‘B’ charging pump lube oil heat exchanger service water control valve
- WO 59102140850, Replace ‘2H’ EDG governor

b. Findings

No findings were identified.

1R20 Refueling and Other Outage Activities

a. Inspection Scope

The inspectors used NRC Inspection Procedure 71111.20, “Refueling and Outage Activities,” to continue their observation portions of the refueling, maintenance activities, and startup activities to verify that the licensee maintained defense-in-depth commensurate with the outage risk plan and applicable TS for the Unit 2 refueling outage (RFO), which began March 21, 2010. The inspectors monitored licensee controls over the outage activities listed below.

- Licensee configuration management, including daily outage reports, to evaluate maintenance of defense-in-depth commensurate with the outage safety plan requirements for key safety functions and compliance with the applicable TS when taking equipment out of service
- Installation and configuration of reactor coolant pressure, level, and temperature instruments to provide accurate indication and an accounting for instrument error
- Monitoring of decay heat removal processes to verify proper operation and that steam generators, when relied upon, were a viable means of backup cooling
- Controls to ensure that outage work was not impacting the ability to operate the spent fuel pool cooling system during and after core offload

- Reactor water inventory controls including flow paths, configurations, and alternative means for inventory addition, and controls to prevent inventory loss
- Reactivity controls to verify compliance with TS and activities which could affect reactivity were reviewed for proper control within the outage risk plan
- Refueling activities for compliance with TS, to verify proper tracking of fuel assemblies from the spent fuel pool to the core, and to verify foreign material exclusion was maintained
- Containment closure activities, including a detailed containment walkdown prior to startup, to verify that evidence of leakage did not exist and that debris had not been left which could affect the performance of the containment sump
- Heatup and startup activities to verify TS, license conditions, and other requirements, commitments, and administrative procedure prerequisites for mode changes were met prior to changing modes or plant conditions. RCS integrity was verified by reviewing RCS leakage calculations and containment integrity was verified by reviewing that status of containment penetrations and containment isolation valves.
- Licensee identification and resolution of problems related to refueling outage activities
- Licensee management of worker fatigue including waiver requests, self declarations and fatigue assessments as available
- Licensee control of refuelling activities

b. Findings

No findings were identified.

1R22 Surveillance Testing

a. Inspection Scope

For the five surveillance tests listed below, the inspectors examined the test procedures, witnessed testing, or reviewed test records and data packages, to determine whether the scope of testing adequately demonstrated that the affected equipment was functional and operable, and that the surveillance requirements of TS were met. The inspectors also determined whether the testing effectively demonstrated that the systems or components were operationally ready and capable of performing their intended safety functions.

In-Service Test:

- 2-PT-64.8, “Flow Test of the Inside Recirculation Spray Pumps,” Revision 28
- 2-PT-64.1.2, “Outside Recirculation Spray Pump 2-RS-P-2B,” Revision 28
- 1-PT-64.4B, “Casing Cooling Pump (1-RS-P-3B) Test,” Revision 21

Other Surveillance Tests:

- 2-PT-83.125, “2J Diesel Generator Test (Start by ESF Actuation) Followed by 24 hour run and hot restart test,” Revision 21

Containment Isolation Valve:

- 2-PT-61.3.3, “Containment Purge Valves as Left Pressure Test,” Revision 21

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

a. Inspection Scope

The inspectors performed a periodic review of the two following Unit 1 and 2 PIs to assess the accuracy and completeness of the submitted data and whether the performance indicators were calculated in accordance with the guidance contained in NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6. The inspection was conducted in accordance with NRC Inspection Procedure 71151, "Performance Indicator Verification." Specifically, the inspectors reviewed the Unit 1 and Unit 2 data reported to the NRC for the period April 1, 2009, through March 31, 2010. Documents reviewed included applicable NRC inspection reports, licensee event reports, operator logs, station performance indicators, and related CRs.

- Reactor Coolant System Specific Activity
- Reactor Coolant System Leakage

b. Findings

No findings were identified.

4OA2 Identification and Resolution of Problems

.1 Review of Items Entered into the Corrective Action Program

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 CAP. This review was accomplished by reviewing daily CR report summaries and periodically attending daily CR Review Team meetings.

.2 Semi-Annual Trend Review – Foreign Material Exclusion (FME)

a. Inspection Scope

The inspectors performed a review of the licensee's corrective action program documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors' review was focused on repetitive equipment and corrective maintenance issues but also considered the results of daily inspector corrective action program item screening discussed in Section 4OA2.1. The review included issues documented outside the normal corrective action program in system health reports, corrective maintenance work orders, component status reports, site monthly meeting reports, and maintenance rule assessments. The inspectors' review nominally considered the six month period of January through June, 2010, although some examples expanded beyond those dates when the scope of the trend warranted.

The inspectors compared and contrasted their results with the results contained in the licensee's latest integrated quarterly assessment report. Corrective actions associated with a sample of the issues identified in the licensee's trend report were reviewed for adequacy.

b. Assessment and Observations

No findings were identified. In general, the licensee has identified trends and addressed them in their corrective action program. The inspectors noted an increasing trend for CAP documents regarding issues with FME during the Spring, 2010, Unit 2 refueling outage. Subsequent review of the licensee's second quarter trend report verified that the licensee had also identified an adverse trend relative to FME problems. The inspectors identified multiple CRs associated with FME during the first and second quarters, 2010. Of interest were those CRs which presented a challenge to the barriers against the release of radioactive isotopes, the fuel cladding and the RCS pressure boundary regarding the SG tubes. This involves foreign material left within the reactor coolant system which could become lodged within a fuel assembly or within SG tubes and penetrate the cladding or the tube, respectively, due to flow induced fretting. The CRs identified as FME issues which could challenge the fuel cladding were:

- CR375152, "Debris located on Lower Core Plate found during visual inspection"
- CR375156, "Metal debris was found during as-found inspection after lower internals lift U2"
- CR375339, "During Rx vessel 10 yr ISI noted 2 wire strands on vessel core barrel ledge"
- CR376168, "Cavity FMEA," (foreign material not logged found within the foreign material exclusion boundary surrounding the reactor cavity)
- CR319099, "North Anna 2 confirmed single fuel rod failure" (subsequently determined to result from unidentified debris)

The CRs identified as FME issues which could challenge SG tube integrity were:

- CR373940, "Foreign Material found and retrieved from 2-RC-E-1C"
- CR374212, "Small Piece of Foreign Material in 2-RC-E-1A"
- CR374737, "Foreign Material found in secondary side of 2-RC-E-1B during FOSAR"

The inspectors verified that the licensee has entered the trend problem with foreign material control into their CAP as CR371668 and the associated common cause analysis, CCA000134, "CAAR directed: CCA to Maintenance to perform common cause associated with FME." The inspectors will continue to monitor the licensee's corrective actions regarding FME.

.3 Annual Sample:

Review of CR372940, "Open contactor failed for 2-EE-BKR-2H1-2S-D1"

a. Inspection Scope

The inspectors reviewed the licensee's assessments and corrective actions for the following inspection sample: CR372940, "Open contactor failed for 2-EE-BKR-2H1-2S-D1." The condition report was reviewed to ensure that the full extent of the issue was identified, an appropriate evaluation was performed, and appropriate corrective actions were specified and prioritized. The inspectors also evaluated the condition report against the requirements of the licensee's corrective action program as specified in PI-AA-200, "Corrective Action", and 10 CFR 50, Appendix B.

b. Findings and Observations

Failure to Promptly Correct a Condition Adverse to Quality for 2-RH-MOV-2700 Breaker

Introduction: A self-revealing NCV of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," was identified for the failure to promptly identify and correct a condition adverse to quality for the 2-RH-MOV-2700, Loop 'A' Hot Leg to RH Pump Isolation Valve, breaker.

Description: On March 21, 2010, 2-RH-MOV-2700, Loop 'A' Hot Leg to RH Pump Isolation Valve, failed to open when the licensee attempted to place residual heat removal in service during the most recent RFO. The licensee determined that 3 of 4 plastic pins used to secure the coil of the open contactor for the motor operated valve (MOV) breaker had failed, allowing the coil to move and subsequently block the free movement of the core, which in turn prevented the contacts from changing state and supplying power to the MOV. The licensee initiated CR372940, "Open contactor failed for 2-EE-BKR-2H1-2S-D1," and ACE018065, "Open contactor failed for 2-EE-BKR-2H1-2S-D1," to investigate the failure and institute corrective actions. The inspectors reviewed ACE018065 and noted that the licensee had previously completed an operating experience evaluation OEE000051, "OPEX000118:OE23372: Failure of a Klockner-Moeller contactor utilizing plastic mounting pins for securing coil assembly (Grand Gulf)" in January, 2007. The inspectors reviewed OEE000051 and noted that this type of contactor was susceptible to plastic coil mounting pin failure which may prevent the contactor from functioning on demand and that it was applicable to North Anna. The inspectors also noted that the licensee instituted corrective action in January 2007 under CR005777, "Review of breaker program and consolidation of program enhancements completed." The inspectors reviewed CR005777 and noted the corrective actions included procedure changes to inspect all applicable contactors for damaged coil mounting pins. However, the inspectors verified the conclusions in ACE018065 in which no short term corrective actions were initiated under CR005777 and that the normal preventative maintenance cycle was adequate for inspecting the coil mounting pins. The inspectors found that for safety related breakers, such as the one in question, this could be as long as five years from when the corrective actions were instituted. The inspectors also found that the breaker for 2-RH-MOV-2700 had not yet been inspected at the time of the failure.

The inspectors concluded that the presence of plastic pins susceptible to failure in the 2-RH-MOV-2700 breaker was a known condition adverse to quality. The inspectors also concluded that the licensee failed to promptly correct this condition adverse to quality as required by 10 CFR 50, Appendix B, Criterion XVI and that this resulted in the failure of 2-RH-MOV-2700 to open on demand.

Analysis: The inspectors determined that the licensee's failure to promptly correct this known condition adverse to quality as required by 10 CFR 50, Appendix B, Criterion XVI was a performance deficiency. The inspectors reviewed IMC 0612, Appendix E and determined the finding was more than minor because it was similar to examples 4d and

4f. The phase 1 screening resulted in a need to perform phase 2 and phase 3 evaluations due to the finding resulting in the loss of mitigating function, specifically the ability to perform decay heat removal. A phase 3 analysis was performed by a regional senior risk analyst in accordance with the guidance of NRC Inspection Manual Chapter 0609 Appendix A. The significance determination process phase 3 risk evaluation resulted in a risk increase for the finding <1E-6 for core damage frequency (CDF) and <1E-7 for large early release frequency (LERF). The dominant sequence involved a steam generator tube rupture, followed by failure of the RHR system, and failure of the operators to refill the emergency condensate storage tank to continue secondary side cooling. The analysis assumed the operators, given the additional time while cooling the core using the secondary side, would be able to manually open 2-RH-MOV-2700. The finding was characterized as of very low safety significance (Green). The cause of this finding involved the cross-cutting area of problem identification and resolution, the component of corrective action program, and the aspect of implementation of corrective action (P.1(d)) because the licensee failed to correct the safety issue that existed with the 2-RH-MOV-2700 breaker in a timely manner, commensurate with its safety significance and complexity.

Enforcement: 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," states in part that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. Contrary to the above, on March 21, 2010, the licensee failed to promptly correct a known condition adverse to quality involving the 2-RH-MOV-2700 breaker which resulted in the failure of the valve to open on demand. Because the finding is of very low safety significance (Green) and it was entered into the licensee's CAP as CR372940, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000339/2010003-02, Failure to Promptly Correct a Condition Adverse to Quality for 2-RH-MOV-2700 Breaker.

4OA3 Event Followup

.1 Unit 2 Automatic Reactor Trip

The inspectors responded to an automatic reactor trip of Unit 2 on April 27, 2010, due to a main generator/turbine trip during testing of the new digital automatic voltage regulator for the main generator. The inspectors discussed the trip with operations, engineering, and licensee management personnel to gain an understanding of the event and assess follow up actions. The inspector 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 will perform a detailed review of the cause of the event during a subsequent review of the licensee's respective licensee event report. The inspectors also reviewed the initial licensee notifications to verify that the requirements specified in NUREG-1022, "Event Reporting Guidelines," Revision 2, were met.

.2 Unit 2 Automatic Reactor Trip During a Lightning Storm

a. Inspection Scope

The inspectors responded to an automatic reactor trip of Unit 2 on May 28, 2010, due to low RCS flow in the 'B' loop. The inspectors discussed the trip with operations, engineering, and licensee management personnel to gain an understanding of the event and assess follow up 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 will perform a detailed review of the cause of the event during a subsequent review of the licensee's respective licensee event report. The inspectors also reviewed the initial licensee notifications to verify that the requirements specified in NUREG-1022, "Event Reporting Guidelines," Revision 2, were met.

b. Findings

Failure to Follow Procedure to Trip 'A' Reactor Coolant Pump on High Bearing Temperature.

Introduction: An apparent violation (AV) of TS 5.4.1a was identified by the inspectors for the failure to adequately implement procedural requirements resulting in operation of the 'A' RCS pump (RCP) beyond the motor high bearing temperature limit of 195 degF for approximately 10 minutes.

Description: On May 28, 2010, at 0004 hours a lightning strike resulted in a loss of power to the 'B' reserve station service transformer (RSST) and consequent loss of power to the downstream '2H' emergency bus and semi-vital bus. This resulted in a start of the standby main feedwater pump due to loss of feedwater pressure instrumentation and consequent low voltage on '2B' station service bus (SSB) which also feeds the 'B' RCP. Undervoltage relays on the '2B' SSB resulted in a fast transfer attempt to the 'B' RSST which was already de-energized. This caused loss of power to the 'B' RCP and subsequent reactor trip. The loss of power to the 2H emergency bus also resulted in a loss of power to and closure of the component cooling (CC) trip valves for the RCP motors and resultant alarms, "RCP 1A [B,C] CC Return Lo Flow."
The inspectors reviewed the related alarm response procedure for RCP 1A CC Return Lo Flow, 2-AR-C-C1, Revision 1, and noted the following steps:

- Step 1.5 identified a probable cause for the alarm as a loss of power to respective CC trip valves
- Step 2.4 states, "IF CC is lost to pump and motor bearings exceed 195°F or pump bearings exceed 225°F, THEN GO TO 2-E-0, Reactor Trip or Safety Injection AND trip the RCP."
- Step 2.5 also refers the operator to 2-AP-15, "Loss of Component Cooling," Revision 18, of which step 10 requires in part for the operator to monitor RCP temperatures, motor bearing temperature less than 195 degF. The response not obtained column requires the operator to go to 2-E-0, "Reactor Trip or Safety Injection," and when the reactor is tripped then stop affected RCPs

The inspectors noted that CC was lost to the RCP motors on loss of power to the 2H emergency bus at ~0004 hours. The inspectors reviewed the respective times from the plant computer system (PCS) at which the first 'A' RCP motor bearing exceeded 195 degF and noted the upper thrust bearing temperature exceeded 195 degF at ~0021 hours. The inspectors also noted the 'A' RCP motor was not tripped until ~0031 hours, and the upper thrust bearing temperature peaked at ~221.7 degF approximately one minute later but did not return to less than 195 degF until ~0135 hours.

The inspectors reviewed the vendor's RCP manual for information regarding the potential impact of RCP motor bearing degradation on the pump seals and identified documentation in Addenda 19 that indicated that one of the major causes of seal failures is high vibration. The inspectors interviewed engineering personnel who noted that the Babbitt is ~.375 inches thick on the thrust and radial bearings. The inspectors determined that the lack of cooling to the bearings would result in overheating and eventual melting of the Babbitt on bearing surfaces to yield increased bearing to shaft clearances and further result in an increase in shaft vibration. The licensee entered this problem into their corrective action program as corrective action 170278 associated with condition report 382725.

Analysis: A performance deficiency was identified by the inspectors for the failure to adequately implement procedural requirements of 2-AR-C-C1 to trip the 'A' RCP in response to motor bearing temperatures exceeding 195 degF. This finding had a credible impact on safety due to the operation of the 'A' RCP beyond the vendor's analysis for adequate, long term component safety. The finding was more than minor because it could be reasonably viewed as a precursor to a significant event due to RCP motor operation in an unknown condition of bearing performance in which the melting of Babbitt material can lead to excessive shaft vibrations and consequent adverse impact on RCP seal performance leading to a seal loss of coolant accident. In accordance with NRC IMC 0609, "Significant Determination Process," the inspectors performed a Phase 1 analysis and determined the finding required a phase 2 evaluation because assuming worst case degradation the TS limit for RCS leakage would have been exceeded. The significance of this finding is to be determined pending completion of a phase 3 evaluation. This finding involved the cross-cutting area of human performance, the component of decision making and the aspect of decision communications, H.1(c), because a reactor operator failed to communicate the loss of component cooling to the RCP motors to the senior reactor operator which led to the failure to trip the 'A' RCP on exceeding the motor bearing high temperature limit.

Enforcement: TS 5.4.1a requires, in part, that written procedures shall be implemented per Regulatory Guide 1.33, Appendix A, of which part 5 specifies procedures for abnormal, off-normal, or alarm conditions. Contrary to this, on May 28, 2010, the licensee failed to adequately implement procedural requirements in 2-AR-C-C1 resulting in operation of the 'A' RCP beyond the motor high bearing temperature limit of 195 degF for approximately 10 minutes. Pending determination of safety significance, this finding is identified as an AV, 05000339/2010003-03, Failure to Follow Procedure to Trip 'A' Reactor Coolant Pump on High Bearing Temperature.

4OA5 Other Activities

.1 Quarterly Resident Inspector Observations of Security Personnel and Activities

a. Inspection Scope

During the inspection period, the inspectors conducted observations of security force personnel and activities to ensure that the activities were consistent with the licensee security procedures and regulatory requirements relating to nuclear plant security. These observations took place during both normal and off-normal plant working hours.

These quarterly resident inspector observations of security force personnel and activities did not constitute any additional inspection samples. Rather, they were considered an integral part of the inspectors' normal plant status review and inspection activities.

b. Findings

No findings were identified.

.2 Institute of Nuclear Power Operations (INPO) Plant Assessment Report Review

a. Inspection Scope

The Senior Resident Inspector and the DRP Branch Chief reviewed the final report for the INPO plant assessment of North Anna Power Station issued in December 2009. The report was reviewed to ensure that issues identified were consistent with the NRC perspectives of licensee performance and to verify if any significant safety issues were identified that required further NRC follow-up.

b. Findings

No findings were identified.

.3 Reactor Coolant System Dissimilar Metal Butt Welds (Temporary Instruction (TI) 2515/172, Revision 1)

a. Inspection Scope

Based on the schedule of dissimilar metal butt weld (DMBW) examinations under MRP-139, no examinations were required for the current Unit 2 refueling outage (N2R20) and hence none were performed. Additionally, the licensee had not made any changes to the MRP-139 inspection program since the NRC had previously reviewed this program.

b. Observations

In accordance with requirements of TI 2515/172, Revision 1, the inspectors evaluated and answered the following questions:

(1) Implementation of the MRP-139 Baseline Inspections

1. Have the baseline inspections been performed or are they scheduled to be performed in accordance with MRP-139 guidance?

   This portion of the TI was not inspected during the period of this inspection report, but was previously covered in NRC Inspection Report 05000339/2008005.

2. Is the licensee planning to take any deviations from the MRP-139 baseline inspection requirements of MRP-139? If so, what deviations are planned, what is the general basis for the deviation, and was the NEI- 03-08 process for filing a deviation followed?

This portion of the TI was not inspected during the period of this inspection report, but was previously covered in NRC Inspection Report 05000339/2008005.

(2) Volumetric Examinations

This portion of the TI was not inspected during the period of this inspection report, but was previously covered in NRC Inspection Report 05000339/2008004.

(3) Weld Overlays

This portion of the TI was not inspected during the period of this inspection report, but was previously covered in NRC Inspection Report 05000339/2008004.

(4) Mechanical Stress Improvements (SI)

There were no mechanical stress improvement activities performed or planned by this licensee to comply with their MRP-139 commitments.

(5) Application of Weld Cladding and Inlays

There were no weld cladding or inlay activities performed or planned by this licensee to comply with their MRP-139 commitments.

(6) Inservice Inspection Program

1. Has the licensee prepared an MRP-139 inservice inspection program? If not, briefly summarize the licensee's basis for not having a documented program and when the licensee plans to complete preparation of the program.

   No. The licensee did not have a standalone MRP-139 inservice inspection program document. However, the licensee's MRP-139 inservice inspection program was included in their ASME Section XI Inservice Inspection Program (ISI Program) and also attached as augmented inspections to the inservice inspection program. The inspectors reviewed the North Anna Unit 2 Third Interval ISI Plan. The licensee had revised the Third Interval ISI Plan to reflect the examination methods and frequencies for the MRP-139 ISI requirements.

2. In the MRP-139 inservice inspection program, are the welds appropriately categorized in accordance with MRP-139? If any welds are not appropriately categorized, briefly explain the discrepancies.

   Yes. The welds were appropriately categorized by the licensee responsible engineer.

3. In the MRP-139 inservice inspection program, are the inservice inspection frequencies, which may differ between the first and second intervals after the MRP-139 baseline inspection, consistent with the inservice inspections frequencies called for by MRP-139?

   Yes. The licensee plans inspection frequencies for welds in the MRP-139 ISI program to be consistent with the requirements of MRP-139.

4. If any welds are categorized as H or I, briefly explain the licensee's basis of the categorization and the licensee's plans for addressing potential PWSCC.

   The six DMBWs on the pressurizer were classified as category C after the full structural weld overlays were applied. Therefore, no DMBWs are categorized as H or I.

5. If the licensee is planning to take deviations from the MRP - 139 inservice inspection guidelines, what are the deviations and what are the general bases for the deviations? Was the NEI 03-08 process for filing deviations followed?

   The licensee had not planned to take any deviations from MRP-139 requirements.

b. Findings

No findings were identified.

.4 On-Site Fabrication of Components and Construction of an Independent Spent Fuel Storage Installation (ISFSI) at Operating Plants (IP 60853, revised 01/16/2008)

a. Inspection Scope

The inspectors conducted a review of licensee and contractor activities, including management and quality control/quality assurance (QC/QA) oversight and radiation protection, of the installation of the nuclear horizontal modular system (NUHOMS) horizontal storage modules (HSM) on ISFSI Pad #2 to verify that the individuals performing quality-related activities were trained, qualified, and familiar with the specified design, designated installation techniques, and quality controls. The inspectors also reviewed approved procedures, drawings, and purchase orders of the HSMs to determine if they were consistent with design commitments and requirements contained in the Safety Analysis Report. The inspectors also reviewed supplier nonconformance reports, licensee condition reports, and audits to determine if the findings were appropriately dispositioned and corrective actions implemented in a time frame commensurate with their safety significance, per the Quality Assurance Program.

b. Findings

No findings were identified.

.5 (Closed) URI 05000338, 339/2009004-02: Fire Brigade Performance

a. Inspection Scope

As described in Unresolved Item (URI) 05000338, 339/2009004-02, the inspectors identified issues related to controller performance and impact on the overall drill performance, characterization of critique objectives and how the objectives were met relative to the requirements of the licensee's fire protection program. The inspectors performed an extensive review of the licensee's fire protection program procedures, records of completed drills, and corrective actions relating to the issues identified by the inspectors.

b. Findings

On September 22, 2009, the inspectors observed an unannounced fire brigade drill, conducted at the Unit 1 cable tray spreading room, and the post drill critique immediately following completion of the drill. The inspectors reviewed 0-FPMP-10.0, "Conduct of Fire Drills," Rev. 7 and noted that the purpose of the procedure was to "assess the effectiveness of the Fire Brigade." The inspectors also noted that section 6.5 contained multiple steps directing the conduct, observation, and critique of fire drills. Specifically step 6.5 requires "Conduct of the fire drill….." step 6.5.8 requires observation of the drill "per the Fire Drill Evaluation Form…" and step 6.5.13 and 6.5.14 requires a critique of the drill with drill evaluation members and the participating fire brigade members. As discussed in URI 05000338, 339/2009004-02 the inspectors identified issues relating to controller performance and impact on the overall drill performance, characterization of critique objectives and how the objectives were met relative to the requirements of the licensee's fire protection program. Drill issues were captured by the licensee in the corrective action system as Condition Report (CR) 349154. Corrective actions taken by the licensee include:

- Conducting a self-assessment of fire brigade training and fire drill program
- Observation of fire drills by licensee management
- Modification of licensee procedures
- Modification of fire drill critique process

The inspectors concluded that the failure to conduct the drill in accordance with procedure 0-FPMP-10.0 was a performance deficiency (PD). The PD was not more than of minor significance because it did not affect the fire brigade's ability to extinguish the fire within the required time. This URI is closed.

4OA6 Meetings, Including Exit

.1 Inservice Inspection/TI 2515-172 Exit Meeting

On April 2, 2010, the inspectors presented the inspection results to licensee management. The licensee acknowledged the inspection results. The inspectors confirmed that none of the potential report input discussed was considered proprietary. Proprietary material received during the inspection was returned to the licensee.

.2 Quarterly Exit Meeting

On July 26, 2010, the senior resident inspector presented the inspection results to Mr. Michael Crist and other members of the staff, who acknowledged the findings. The inspectors asked the licensee whether any of the material examined during the inspection should be considered proprietary. No proprietary information was identified.

.3 Annual Assessment Meeting

On May 3, 2010, the NRC's Chief of Reactor Projects Branch 5 and the Resident Inspectors assigned to the North Anna Power Station held an open public meeting to discuss the NRC's Reactor Oversight Process and the NRC's annual assessment of North Anna's safety performance for the period of January 1 through December 31, 2009. The major topics addressed were the NRC's assessment program, and the results of the North Anna Power Station assessment. Attendees included Dominion Virginia Power corporate management, members of the North Anna site management team, and a member of the local newspaper media.

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 number ML081200874. ADAMS is accessible from the NRC Web site at http://www/nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

.4 On-Site Fabrication of Components and Construction of an ISFSI

An exit meeting was conducted on June 4, 2010, with licensee management. The inspector presented the results to Larry Lane, Site Vice President, and other members of the licensee's staff.

ATTACHMENT: SUPPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel:

W. Anthes, Manager, Nuclear Maintenance
M. Crist, Plant Manager
R. Evans, Manager, Radiological Protection and Chemistry
C. Gum, Manager, Nuclear Protection Services
T. Huber, Director, Nuclear Engineering
S. Hughes, Manager, Nuclear Operations
P. Kemp, Supervisor, Station Licensing and Manager, Organizational Effectiveness
L. Lane, Site Vice President
M. Becker, Manager, Nuclear Outage and Planning
F. Mladen, Director, Station Safety and Licensing
B. Morrison, Supervisor Nuclear Engineering
B. Scanlan, Manager, Nuclear Site Services
J. Scott, Supervisor, Nuclear Training (operations)

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000339/2010003-03	AV	Failure to Follow Procedure to Trip 'A' Reactor Coolant Pump on High Bearing Temperature (Section 4OA3.2)

Opened and Closed

05000338/2010003-01	NCV	Failure to Demonstrate Effective Control of a Pressurizer PORV 1-RC-PCV-1455C Performance in Accordance with the Maintenance Rule (Section 1R12)
05000339/2010003-02	NCV	Failure to Promptly Correct a Condition Adverse to Quality For 2-RH-MOV-2700 Breaker (Section 4OA2.3)

Closed

URI 05000338, 339/2009004-02	URI	Fire Brigade Performance (Section 4OA5.5)

LIST OF DOCUMENTS REVIEWED

Section 1R04.2: Equipment Alignment – Complete Walkdown

Procedures
- 0-OP-49.4A, Revision 6, "VALVE CHECKOFF - CONTROL ROOM A/C SERVICE WATER"
- 0-MOP-49.1, Revision 4, "SERVICE WATER FLOODING IN AUXILIARY BUILDING"
- 0-PT-75.1, Revision 20, "SERVICE WATER SYSTEM - VALVES (MONTHLY)"
- 0-PT-75.14, Revision15, "SERVICE WATER WALL THICKNESS MONITORING"
- 0-PT-75.15, Revision 6, "GENERIC LETTER 89-13 SERVICE WATER SYSTEM TESTING REQUIREMENTS COORDINATION"

Condition Reports
- CR381777, "2-HV-H-16 leak was identified to be "through wall" leak of class 3 piping"
- CR376197, "B SW return line, 4"-WS-468-151-Q3, not in contact with pipe support"
- CR105653, "Piping Elbow downstream of 2-SW-280 has through wall leak."
- CR016271, "SW supply pipe to 1-HV-E-4B identified as having a pit below Code minimum wall"
- CR013761, "Service Water supply pipe to 1-HV-E-4B approaching code minimum wall thickness"
- Plant Issue N-1999-1604, "MIC leak the 4" A header SW supply line for the Unit 2 control room chillers"
- Plant Issue N-2005-0036, "MIC leak downstream of 2-HV-MOV-213C, 4" SW outlet of 2-HV-E-4C"

Drawings
- SPH-PHLD-1, Sheet 2 of 26 pipe support location isometric
- 11715-ECI-105AVD, Revision 1, Sheet 1 of 2, piping isometric SW 4" chiller pipe
- 11715-ECI-105AVD, Revision 0, Sheet 2 of 2, piping isometric SW 4" chiller pipe
- 11715-ECI-105AVC, Revision 1, Sheet 1 of 2, piping isometric SW 4" chiller pipe
- 11715-ECI-105AVB, Revision 1, Sheet 2 of 2, piping isometric SW 4" chiller pipe
- 11715-ECI-105AS, Revision 0, Sheet 1 of 1, piping isometric SW 4" chiller pipe
- 11715-ECI-105AVE, Revision 0, Sheet 1 of 2, piping isometric SW 4" chiller pipe
- 11715-FB-040D, Revision 48, flow/valve operating numbers diagram, air conditioning condenser water system
- 11715-PSSK-105AVD.06, Revision 0, Sheet 1 of 1, pipe support sketch, 2-WS-PH-H50.01
- 11715-FM-078A, Revision 63, Sheet 1 of 5, flow/valve operating numbers diagram, service water system
- 11715-FM-078A, Revision 100, Sheet 4 of 5, flow/valve operating numbers diagram, service water system
- 11715-FB-040D, Revision 43, Sheet 1 of 3, flow/valve operating numbers diagram, air conditioning condenser water system
- 11715-FB-040D, Revision 48, Sheet 2 of 3, flow/valve operating numbers diagram, air conditioning condenser water system

Other Documents
- UFSAR, Section 9.2.1 Service Water System
- TS 3.7.8, Service Water System
- Design bases document, Service Water System
- Listing of work orders, open and closed for SW piping/supports re control room chillers

Section 1R08: Inservice Inspection Activities

Procedures
- 03-9130128-00, Secondary Side Visual Inspection Plan and Procedure for Dominion North Anna Unit 2, Revision 2
- 2-PT-48.6, "Vessel Head Bare Metal Visual Inspection," Revision 0, 2/6/2006
- 2-PT-48.6, "Vessel Head Bare Metal Visual Inspection," Revision 1, 9/25/2008
- 2-PT-48.7, "Vessel Head Volumetric Inspection," Revision 1, 9/25/2008
- 2-PT-54.1, "Reactor Pressure Vessel Effective Degradation Years Calculation," Revision 1, 4/26/2004
- 54-ISI-24-31, Written Practice for Personnel Qualification in Eddy Current Examination, Revision 07/08/2008
- 54-ISI-400, Multi-Frequency Eddy Current Examination of Tubing, Revision 17
- 54-ISI-79-11, Zetec Eddynet® Suite Software Checkout Procedure
- ER-AA-NDE-PT-300, "ASME Section XI Liquid Penetrant Examination Procedure," Revision 4
- ER-AA-NDE-PT-301, "Balance of Plant (BOP) Liquid Penetrant Examination Procedure," Revision 3
- ER-AA-NDE-UT-805, "Straight Beam Ultrasonic Examination of Studs and Bolts in Accordance with ASME Section XI, Appendix VIII," Revision 0
- ER-AA-NDE-VT-601, "VT-1 Visual Examination Procedure," Revision 2
- ER-AA-NDE-VT-604, "Visual Examination for Leakage of PWR Reactor Head Penetrations," Revision 0
- ER-AA-NDE-VT-607, "VE Examination of Pressure Retaining Welds in Class 1 Components Fabricated with Alloy 600/82/182 Materials," Revision 0
- ER-AP-BAC-10, "Boric Acid Corrosion Control Program," Revision 5
- ER-AP-BAC-101, "Boric Acid Corrosion Control Program (BACCP) Inspections," Revision 4
- ER-AP-BAC-102, "Boric Acid Corrosion Control Program (BACCP) Evaluations," Revision 5
- ER-NA-AUG-101, "," Revision 1
- MA-AA-1002, "Leakage Management," Revision 5
- MCM-0400-35, "Repacking Manual Valves," Revision 11
- MCM-1006-01, "Repair of Safety-related Piping and Component Bolted Flange Joints," Revision 19
- MCM-1801-01, "Welding Safety-related and Seismic-related Equipment," Revision 19
- NAP-SGPMS-001, North Anna Site Specific Eddy Current Analysis Guidelines, Revision 13

Condition Reports
- CA089637, "Other to Eng (Morris) to evaluate 2-CH-FCV-2160 in accordance with the BACC Program"
- CA158683, "Perform an evaluation to determine the extent of any boric acid degradation"
- CR 373940, "Foreign Material found and retrieved from 2-RC-E-1C"

- CR 374176, 2-RC-E-1B 7th Support Plate wrapper plug not in expected alignment
- CR 374212, “Small Piece of Foreign Material in 2-RC-E-1A”
- CR 374737, “Foreign Material Found in Secondary Side of 2-RC-E-1B during FOSAR”
- CR117130, “Possible boric acid deposit on pipe in overhead of 24’ containment”
- CR117205, “3CHS*V645, boric acid deposit observed on packing gland area”
- CR117211, “3CHS*V987, boric acid deposit observed on packing gland area”
- CR118048, “2-QS-ICV-3022 has dry boric acid buildup on the downstream vent gooseneck”
- CR373134, “Relevant Conditions and Blistering on Unit 2 Containment Liner”
- CR373299, “Unit 2 inside recirc spray pump test channel conn plugs found degraded”
- CR373604, “U2 ctmt liner – keyway area test connections (W1 &P6) have surface corrosion”
- CR373906, “DCU 10-567:3 vent connections found in the RS Sump, no id’d on drawings”
- CR373925, “Three rejectable indications were found in Weld 6A on Line 12”-RS-407-153A-Q2”
- CR374532, “Additional work required for 2-RS-PP-12.00-RS-PIPE-407-153AQ2”
- CR374666, “Minor boric acid found during 2-PT-48.3”
- CR374670, “Boric acid found on manifold o f2-RC-PT-2458B”
- CR374673, “Boric acid found on pipe cap of 2-SI-291”
- CR374820, “Examination of reactor vessel studs should be performed from the bottom surface”

Other Documents

- “Virginia Electric and Power Company (Dominion) North Anna Power Station Unit 2 Inservice Inspection Plan for the Third Inspection Interval,” Revision 10
- 1275031, Inconel 600/Inconel 690 Equivalency for Eddy Current Testing, Revision 0
- 51-5001223-00, Appendix H Equivalency Cable Lengths, July 1997
- 51-5002881-00, Appendix H Equivalency, MRPC Exams Probe Extensions, Cable Length, and Motor Unit Length
- 51-9133110-000, North Anna Unit 2, 2R20 – EPRI Appendix H/Appendix I Eddy Current Technique Review
- 549030575-000, MIZ®-80iD Eddy Current Tester Implementation Document
- Certificate of Authenticity for Eddy Current Probes Serial Numbers: S/N 515112, 515118, 515122, 515124, 464933, 464934, 513481, 513484, 513486, 513488, 453924, 464808, 501547, 509330, 517435, 517439, 454086, 453969, 453980, 505302, 505305, 456621, 515483, 515484, 505527, 505546, 505547, 463434, 464937, 515100, 515103, 515104, 515119, 515125, 515131, 514589, 514293, 514294, 514295, 428565, 454143, 505549, 362845, and 463453
- Certificate of Calibration for Eddy Current Tester Model MIZ-80iD, Serial Numbers 015, 021, 032, 048, 011, 043, 011, 032, 021, 015, 032, 039, 043, and 048
- Certificate of Compliance for Calibration Block 94-6692 (Heat No. A58767A)
- Certificate of Contaminant Report for Spotcheck Cleaner/Remover (Batch No. 09L08K), Spotcheck Penetrant (Batch Nos. 05M15K and 06G16K) and Spotcheck Developer (Batch No. 08H01K)
- Certificates of Personnel Qualification for 17 Qualified Data Analysts
- Certified Material Test Reports for Heat Nos. G10882, 37349, 83718, DF7967 and DT7591; R.V. Closure Stud Heat No. X7225
- Certified Test Report for Couplant 072-S
- ET-N-10-007, North Anna Unit 2 – Spring 2010 Steam Generator Degradation Assessment, Revision 0

- ETSS_BOB001_MIZ80_R0, North Anna Unit 2, Bobbin Probe: Standard ASME Code Examination for Parent Tubing
- ETSS_RPC001_MIZ80_R0, North Anna Unit 2, RPC Probe for Examination of Tubesheets, Dented and Non-dented TSPs, Expansion Transitions, Freespan Dents, and Freespan Bobbin Indications.
- ETSS_RPC002_MIZ80_R0, North Anna Unit 2, RPC Probe for Examination of Low Row U-bends and U-bend Special Interest
- Letter from Kevin J. Hacker, Dominion Corporate Level III to Dominion Site Level III's, "Character Resolution Card Certification," 9/30/2009
- Liquid Penetrant Examination Report Nos. BOP-PT-08-092, BOP-PT-08-112, PT-10-008, PT-10-010, PT-10-011 and PT-10-012
- Measuring and Test Equipment Calibration Certificate for digital thermometer, SN 1073BGCY and NDE07
- MIZ®-30 Eddy Current Tester Implementation Document, 05/31/1994
- North Anna Power Station Units 1 & 2 Technical Specifications, Section 5.5.8 Steam Generator Program
- Procedure Qualification Record Nos. 801 Rev 2, 805 Rev 3, 809 Rev 2, 830 and 831
- Purchase Order Nos. 45558354, 45536689, 45502071 and 45143613
- Site Specific Performance Demonstration records for 17 Qualified Data Analysts
- Steam Generator Condition Monitoring and Operational Assessment – North Anna Unit 2 – October 2005 Refueling Outage
- Steam Generator Condition Monitoring and Operational Assessment – North Anna Unit 2 – March 2007 Refueling Outage EOC18/REOC8
- Transducer Pulse Characterization SN 7969
- Ultrasonic Instrument Linearity Report No. L-10-011
- UT Bolting/Stud Examination Report Nos. UT-10-104, UT-10-105, UT-10-106, UT-10-107, UT-10-108, UT-10-109, UT-10-110, UT-10-111, UT-10-112 and UT-10-113
- UT Calibration Report Nos. CAL-10-028 and CAL-10-033
- Visual Exam of Equipment and Components (VT-1) Report Nos. VT-10-52, VT-10-53, VT-10-54, VT-10-55, VT-10-56, VT-10-57, VT-10-58, VT-10-59, VT-10-60, VT-10-61, VT-10-62, VT-10-63, VT-10-64, VT-10-65, VT-10-66, VT-10-67, VT-10-68, VT-10-69, VT-10-71, VT-10-72, VT-10-73, VT-10-74, VT-10-75, VT-10-76, VT-10-77, VT-10-78, VT-10-79, VT-10-86, VT-10-86 and VT-10-87
- Visual Examination for Boric Acid Detection Report for Reactor Vessel BMI, 3/26/2010
- Weld Data Sheet for WO774483
- Weld Technique Sheet No. 803
- Work Order Package WO774483, "Replace ¾" Kerotest with Velan"
- Zetec Eddynet®Suite Examination Summary File /mnt/na2bc2/raw/SG2BCCAL00006/SCR031C076I001
- Zetec Eddynet®Suite Examination Summary File /mnt/na2bc2/raw/SG2BHCAL00001/DHR037C028I060
- Zetec Eddynet®Suite Examination Summary File /mnt/na2bh2/raw/SG2BHCAL00021/DHR999C999I002

<u>Section 4OA5.4: Other Activities</u>

<u>Procedures</u>

- GMP-C-105, General Maintenance Procedure – Grouting, Revision 4
  GMP-C-117, General Maintenance Procedure – Core Drilling of Concrete and Masonry Structures, Revision 2
- GMP-C-128, Horizontal Storage Module (HSM-H) Assembly for Spent Fuel, Revision 2

- MA-AA-101, Fleet Lifting and Material Handling, Revision 4
- MA-AA-MCR-101, Mobile Cranes, Revision 2
- MA-AA-MHL-101, Material Handling, Revision 1
- MA-AA-OCR-101, Overhead Cranes/Hoists, Revision 1
- MA-AA-RHW-101, Rigging Hardware, Revision 1
- MA-AA-VSE-101, Certification of Vendor Supplied Lifting and Handling Equipment, Revision 0
- NF-AA-NSF-103, NUHOMS Horizontal Storage Module Array Installation Final Documentation Package and Dominion Certificate of Compliance Preparation, Revision 2

Audits and Surveillances
- Oversight Audit 07013: North Anna NUHOMS ISFSI (Special Audit) and Audit Report
- Vendor Surveillance as reported in Trip Report 2009-01, dated 10/05/09
- Vendor Surveillance as reported in Trip Report 2009-03, dated 10/28/09
- Vendor Surveillance as reported in Trip Report 2009-08, dated 12/09/09
- Vendor Surveillance as reported in Trip Report 2009-09, dated 12/11/09

Drawings
- Vendor Drawing 10495-7100, ISFSI – Phase II HSM Configuration (Sheets 1,2, and 3), Revision 0
- Vendor Drawing NUH-03-7101, General Arrangement (Sheets 1, 2, 3, 4, 5, and 6), Revision 1
- Vendor Drawing NUH-03-7103, Base (Sheets 1, 2, 3, 4, and 5), Revision 2
- Vendor Drawing NUH-03-7104, Roof (Sheets 1, 2, and 3), Revision 2
- Vendor Drawing NUH-03-7107, DSC Support Structure (Sheets 1, 2, 3, 4 and 5), Revision 3
- Vendor Drawing NUH-03-7102, Main Assembly (Sheets 1, 2, 3, 4, 5, and 6), Revision 1
- Vendor Drawing NUH-03-7110, Heat Shields (Sheets 1 and 2), Revision 2
- Vendor Drawing NUH-03-7111, Erection Hardware (Sheets 1 and 2), Revision 2
- Vendor Drawing NUH-03-7112, Fasteners (Sheets 1 and 2), Revision 2

Condition Reports (CRs)
- CR365105, ISFSI Pad #2 Concrete Flatness Does Not Meet Specification Requirements
- CR379638, Daily Critical Observation – NUHOMS Module Installation
- CR380158, Minor Cosmetic Damage to NUHOMS Components

Supplier Nonconformance Reports (SNCRs)
- SNCR 09-C-003, Purchase Order #70191225 & Master Services Agreement #46017934
- SNCR 10-C-001, Dominion Purchase Order #70191225
- SNCR 10-C-003, Dominion Purchase Order #70191225

Other
- Nuclear Engineering Standard DNES-AA-NAF-NSF-0001, HSM Installation, Revision 0
- Vendor Specification NUH-03-0218, Field Erection of NUHOMS HSM-H Array, Revision 4
- Technical Report NE-1522, North Anna Independent Spent Fuel Storage Installation NUHOMS HD 10 CFR 72.212 Evaluation, Revision 4

LIST OF ACRONYMS

ACE	Apparent Cause Evaluation
ADAMS	Agencywide Document Access and Management System
ASME	American Society of Mechanical Engineers
CA	Corrective Action
CAP	Corrective Action Program
CFR	Code of Federal Regulations
CR	Condition Report
DBD	Design Basis Document
EDG	Emergency Diesel Generator
IMC	Inspection Manual Chapter
ISI	Inservice Inspection
ISFSI	Independent Spent Fuel Storage Installation
JPM	Job Performance Measures
LHSI	Low Head Safety Injection
NCV	Non-cited Violation
NDE	Non-Destructive Examinations
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NUHOMS	Nuclear Horizontal Module Storage
OD	Operability Determination
PARS	Publicly Available Records
PI	Performance Indicator
PM	Preventative Maintenance
QS	Quench Spray
RCE	Root Cause Evaluation
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RFO	Refueling Outage
RTP	Rated Thermal Power
SDP	Significance Determination Process
SR	Surveillance Requirements
TDAFWP	Turbine Driven Auxiliary Feedwater Pump
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
VEPCO	Virginia Electric and Power Company
VPAP	Virginia Power Administrative Procedure
WO	Work Order