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

April 29, 2009

Mr. David A. Christian
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 05000338/2009002 AND 05000339/2009002

Dear Mr. Christian:

On March 31, 2009, the U. S. Nuclear Regulatory Commission (NRC) completed an inspection at your North Anna Power Station Units 1 and 2. The enclosed inspection report documents the inspection findings which were discussed on April 21, 2009, with Mr. Daniel Stoddard 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 NRC-identified finding of very low safety significance (Green) which was determined to be a violation of NRC requirements. However, because of the very low safety significance of the issue and because it was entered into your corrective action program, the NRC is treating this violation as a non-cited violation (NCV) consistent with Section VI.A.1 of the NRC's Enforcement Policy. If you contest this NCV, 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 North Anna Power Station.

Additionally, if you disagree with the characterization of 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, Region II, and the NRC Resident Inspector at the North Anna Power Station. The information you provide will be considered in accordance with Inspection Manual Chapter 0305.

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
License Nos.: NPF-4, NPF-7

Enclosure: Inspection Report 05000338/2009002 and 05000339/2009002

w/Attachment: Supplemental Information

cc w/encl. (See next page)

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Letter to David A. Christian from Gerald J. McCoy dated April 29, 2009

SUBJECT: NORTH ANNA POWER STATION – NRC INTEGRATED INSPECTION REPORT 05000338/2009002 AND 05000339/2009002

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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos:	50-338, 50-339
License Nos:	NPF-4, NPF-7
Report No:	05000338/2009002 and 05000339/2009002
Licensee:	Virginia Electric and Power Company (VEPCO)
Facility:	North Anna Power Station, Units 1 & 2
Location:	1022 Haley Drive Mineral, Virginia 23117
Dates:	January 1, 2009 through March 31, 2009
Inspectors:	J. Reece, Senior Resident Inspector R. Clagg, Resident Inspector R. Hamilton, Senior Health Physicist (Sections 2OS1, 4OA5.2, and 4OA5.3) R. Williams, Reactor Inspector (Section 1R08) E. Michel, Senior Reactor Inspector (Section 1R08, 4OA5.4) R. Chou, Reactor Inspector (Section 1R08)
Approved by:	Gerald J. McCoy, Chief Reactor Projects Branch 5 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000338/2009-002, 05000339/2009-002; 01/01/2009 – 03/31/2009; North Anna Power Station, Units 1 and 2; Routine Integrated Inspection Report

The report covered a 3-month period of inspection by resident inspectors, one senior health physicist, two reactor inspectors and one senior reactor inspector. One finding was identified and was determined to be a non-cited violation (NCVs). The significance of most findings is identified 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 0305, "Operating Reactor Assessment Program." 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 Non-cited Violation (NCV) of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified by the NRC for multiple examples of a failure to accomplish a procedure for activities affecting quality which rendered both trains of high head safety injection (HHSI) inoperable. The licensee entered this issue into their corrective action program (CAP) as CR114725.

This finding had a credible impact on safety because both trains of the HHSI were rendered inoperable, and manual operator action was required to place at least one train in service. The inspectors determined the finding was more than minor because it impacted the mitigating systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences, and the related attribute of human performance which involved the failure to adequately accomplish procedures. The inspectors evaluated the finding using the significance determination process (SDP) and determined that a Phase III evaluation was required. A regional Senior Reactor Analyst performed a Phase 3 evaluation under the SDP. The performance deficiency was determined to be of very low safety significance (Green). The evaluation was accomplished using the NRC's probabilistic risk assessment computer model of the plant with Emergency Diesel Generator 1J and the Boron Injection Tank's inlet motor operated valve 1867A set to always fail. The model was quantified, assuming the configuration lasted for nine hours. The dominant accident sequences were Losses of Offsite Power as the initiating event followed by the failure through various mechanisms of the 1H emergency diesel generator and the Alternate Alternating Current Diesel Generator. Also, neither the failed Emergency Diesel Generators nor offsite power was recovered prior to core damage. The key assumptions were that Unit 2 was constructed similar enough that the Unit 1 probabilistic risk assessment model could be used and the duration of the configuration was nine hours.

This finding involved the cross-cutting area of human performance, the component of decision-making and the aspect of safety-significant decisions using a systematic process, especially when faced with uncertain or unexpected plant conditions, to ensure safety is maintained, because the personnel performing quality related activities involving 2-SI-MOV-2867A failed to make adequate decisions affecting nuclear safety while performing procedures (H.1.a). (Section 4OA3)

B. Licensee Identified Violations

None

REPORT DETAILS

Summary of Plant Status

Unit 1 began the period at full Rated Thermal Power (RTP) and operated at full power until March 8, 2009, when the unit entered a planned refueling outage. The unit remained in a refueling outage for the rest of the report period.

Unit 2 began the period at full RTP and operated at or near full RTP for the entire report period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection

Site Specific Event

a. Inspection Scope

The inspectors performed two site specific weather related inspections due to anticipated adverse weather conditions. During January 2009, and again in March 2009, the licensee prepared for an adverse weather forecast of chilling temperatures. Specifically, the inspectors reviewed the licensee responses to extreme cold weather at the plant site for the following risk significant areas:

- Unit 1 and Unit 2 Refueling Water Storage Tanks on January 14, 2009
- Unit 1 and Unit 2 Auxiliary Feedwater (AFW) Pump rooms on March 2, 2009

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

a. Inspection Scope

The inspectors conducted three equipment partial alignment 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.

- ‘B’ train Service Water (SW) System during planned major maintenance on ‘A’ train SW components

- Unit 1 'B' Quench Spray (QS) system during planned maintenance on ‘A’ train components
- Unit 1 'A' train Motor Driven Auxiliary Feedwater (MDAFW) and turbine driven AFW systems during planned maintenance on the 'B' train AFW system components

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors conducted tours of the four samples listed below that are important to reactor safety to verify the licensee’s implementation of fire protection requirements as described in Virginia Power Administrative Procedure (VPAP)-2401, Revision 29, “Fire Protection Program.” 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.

- Battery Room 1 – I Unit 1 (fire zone 7A-1 / BR1-I), Battery Room 2 – I Unit 2 (fire zone 7A-2 / BR2-I), Battery Room 1 – II Unit 1 (fire zone 7B-1 / BR1-II), Battery Room 2 – II Unit 2 (fire zone 7B-2-II), Battery Room 1 – III Unit 1 (fire zone 7C-1 / BR1-III), Battery Room 2 – III Unit 2 (fire zone 7C-2 / BR2- III), Battery Room 1 – IV Unit 1 (fire zone 7D-1 / BR1-IV)., Battery Room 2 – IV Unit 2 (fire zone 7D-2 / BR2-IV)
- Emergency Diesel Generator 1H Unit 1 (fire zone 9A-1a / EDG-1H) and Emergency Diesel Generator 2H Unit 2 (fire zone 9A-2a / EDG-2H)
- Turbine-Driven Auxiliary Feedwater Pump Room Unit 1 (fire zone 14A-1a / TDAFW-1), Turbine-Driven Auxiliary Feedwater Pump Room Unit 2 (fire zone 14A-2a / TDAFW-2), Motor-Driven Auxiliary Feedwater Pump Room Unit 1 (fire zone 14B-1a / MDAFW-1), Motor-Driven Auxiliary Feedwater Pump Room Unit 2 (fire zone 14B-2a / MDAFW-2)
- Containment Unit 1 (fire zone 1-1a / RC-1)

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI) Activities

.1 Non-Destructive Examination (NDE) Activities and Welding Activities

a. Inspection Scope

From March 16 to March 20, 2009, the inspectors reviewed the implementation of the licensee’s In-service Inspection (ISI) program for monitoring degradation of the reactor coolant system (RCS) boundary and risk significant piping boundaries of

Unit 1. The inspectors' activities consisted of an on-site review of NDE and welding activities to evaluate compliance with the applicable edition of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Section XI (Code of record: 1989 Edition with no Addenda), and to verify that indications and defects (if present) were appropriately evaluated and dispositioned in accordance with the requirements of the ASME Code, Section XI acceptance standards.

The inspectors' review of NDE activities specifically covered examination procedures, NDE reports, equipment and consumables certification records, personnel qualification records, and calibration reports (as applicable) for the following examinations:

- UT of circumferential pressurizer vessel weld, N1-B2-11-001
- Automated UT of the reactor pressure vessel and nozzle welds, N1.R1.11.026, N1.R1.11.027, and N1.R1.11.030

The inspectors' review of welding activities specifically covered the welding activity listed below in order to evaluate compliance with procedures and the ASME Code. The inspector reviewed the work order, repair and replacement plan, weld data sheets, welding procedures, procedure qualification records, welder qualification records, and NDE reports.

- WO 59049419702 – Perform 12" Bore Overlay on 1C SI Accumulator, ASME Class I

b. Findings

Introduction: An unresolved item (URI) was identified for an issue regarding ASME required examinations of the Unit 1 containment liner in several sump areas.

Description: On September 30, 2007, while performing maintenance and cleanup in the Unit 1 sump area, the licensee discovered that the top of a containment pressure test connection plug had corroded away and was missing. This discovery was documented in condition report (CR) 021245. The original CR was assigned to a minor work request which was later canceled prior to work commencement. Later, the licensee initiated CR325382, apparent cause evaluation (ACE) 017479 and operability determination (OD) 000272 to address this issue during the spring 2009 Unit 1 outage.

The typical pressure test connection consists of a 3 1/2" diameter standard pipe cap with a 1/8" diameter screwed half coupling and 1/8" screwed socket head pipe plug. This setup was originally used to locally pressure test a plugged grout injection hole in an embedded 3/4" plate that is part of the containment liner. The grout injection hole below the protective cover was originally used during plant construction to backfill grout under the sump liner. It is plugged with a threaded 2" diameter hex head pipe plug and sealed with a 1/4" fillet weld. All of the above mentioned materials are carbon steel. The exposed area of the test connection was originally coated with a zinc-based paint for corrosion inhibition. The test connection was directly exposed to the boric acid containing wetted environment of the containment sump.

This led to general corrosion of the test connection and socket head pipe plug and allowed borated water from the sump to directly contact the carbon steel containment liner and 2” diameter hex head pipe plug.

The licensee identified several other sump areas with test connections exhibiting corrosion including four locations where the 1/8” screwed socket head pipe plugs were missing and instituted a work order to clean and redrill the test connections to accept a 1/4" stainless steel pipe plug. Subsequently, the licensee initiated corrective actions and performed successful pressure tests on the connections at 45 psig for 15 minutes. Similar issues exist for the Unit 2 sump area and will be addressed under request for engineering assistance (REA) 2008-094 during the next Unit 2 outage.

This issue is unresolved pending completion of NRC review of the licensee’s ACE, and a related corrosion analysis. It is identified as URI 05000338/2009002-01, Degradation of the Containment Sump.

.2 PWR Vessel Upper Head Penetration (VUHP) Inspection Activities

a. Inspection Scope

Inspections during this outage consisted of visual examinations conducted above the reactor pressure vessel upper head to identify potential boric acid leaks from pressure-retaining components. The inspectors specifically reviewed examination procedures, personnel training and qualification records, reports for the visual inspection of pressure retaining components above the head performed every outage, and reviewed the licensee’s calculations for effective degradation years (EDYs). No reactor vessel augmented examination required by 10CFR50.55a(g)(6)(ii)(d) were required to be performed during this outage.

b. Findings

No findings of significance were identified.

.3 Boric Acid Corrosion Control (BACC) Inspection Activities

a. Inspection Scope

The inspectors reviewed the licensee’s BACC program activities to ensure implementation with commitments made in response to NRC Generic Letter 88-05, “Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary,” and applicable industry guidance documents. Specifically, the inspectors performed an on-site record review of procedures and the results of the licensee’s containment walk-down inspections performed during the Unit 1 Spring 2009 outage. The inspectors also interviewed the BACC program owner and conducted a walk-down of the reactor building to evaluate compliance with licensee’s BACC program requirements and verify that degraded or non-conforming conditions, such as boric acid leaks identified during the containment walk-down, were properly identified and corrected in accordance with the licensee’s BACC and corrective action programs.

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

- CR019200 – Boric acid found on components while performing 1-PT-48.2B
- CR020882 – Inactive boric acid leak on 1-RS-P-3A
- CR020995 – Inactive boric acid leak on 1-RH-030

b. Findings

No findings of significance were identified.

.4 Steam Generator (SG) Tube Inspection Activities

a. Inspection Scope

The inspectors reviewed activities, plans, condition monitoring and operational assessments, the pre-outage degradation assessment, and procedures for the inspection and evaluation of the steam generator Inconel Alloy 690TT tubing for Unit 1 SGs 'B' and 'C' to determine if the activities were being conducted in accordance with TS and applicable industry standards. Data gathering, analysis, and evaluation activities were reviewed.

The inspectors reviewed data results to verify the adequacy of the licensee's primary, secondary, and resolution analyses. The inspectors also reviewed video tapes for the secondary inspection of the steam drum and the top of the tube sheet for corrosion and foreign objects.

The inspectors reviewed equipment, data operators, and analyst certifications and qualifications, including medical exams.

The inspectors reviewed data for the following tubes:

SG B: R33C30, R17C12, R26C90, R4C51, R10C50, R1C16, R19C35, R27C68, R23C56, R22C56, R21C56, R1C10, R3C58, R20C30, and R21C71

SG C: R32C53, R31C59, R46C57, R47C55, R33C22, R23C45, R20C40, and R13C11

b. Findings

No findings of significance were identified.

.5 Identification and Resolution of Problems

a. Inspection Scope

The inspectors performed a review of ISI-related problems, including welding, BACC, and SG inspections that were identified by the licensee and entered into the corrective action program as CRs. The inspector reviewed the CRs to confirm that the licensee had appropriately described the scope of the problem and had initiated corrective actions.

The review also included the licensee's consideration and assessment of operating experience events applicable to the plant. The inspectors performed this review to ensure compliance with 10CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requirements. The corrective action documents reviewed by the inspectors are listed in the report attachment.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program

a. Inspection Scope

The inspectors reviewed two crew scenarios associated with the annual requalification process. The first scenario involved a failure of a pressurizer pressure transmitter, a component cooling pump trip with failure of the standby pump to start, a reactor coolant system leak and subsequent large break loss of coolant accident, a failure of the containment depressurization actuation automatic signal and respective switch for the operator at the controls, a failure of the 'A' train low head safety injection pump, and a failure of the 'A' train low head safety injection pump auto-start.

The second scenario involved a dropped control rod, a trip of a main feedwater pump with a subsequent feedwater line break, a failure of the main turbine to auto-trip on a reactor trip, a trip of the turbine driven auxiliary feedwater pump with a failure to reset, and a degradation of the 'B' train motor driven auxiliary feedwater pump.

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 of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

For the two equipment issues listed below, the inspectors evaluated the effectiveness of the corresponding 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 system engineers. The inspectors compared the licensee's actions with the requirements of the Maintenance Rule (10 CFR 50.65) using ER-AA-MRL-10, Revision 2, "Maintenance Rule Program."

- MRE006838: “MRule Evaluation to Engineering for failure of 1-EG-BC-02,” (1H Emergency Diesel Generator (EDG) Battery Charger)
- MRE010155: “2H EDG air start system has engine lube oil in it”

b. Findings

No findings of significance were identified.

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 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 complying 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.

- Entry into 0-AP-41.1, “Service Water Array Icing,” while ‘A’ train SW header was removed from service for multiple maintenance activities
- Entry into 0-AP-41.1, “Service Water Array Icing”, for possible service water spray array icing (emergent) due to ambient temperature and snow
- Emergent work on Unit 1 ‘1H’ EDG due to ground on respective battery charger
- 2009 Unit 1 refueling outage plan change safety review for emergent work on 1-SI-MOV-1869A and 1-SI-MOV-1869B
- Emergent work on Unit 2 Quench Spray 2-QS-MOV-202B
- Emergent entry into 0-AP-8, Revision 6, "Response to Grid Instability," for Units 1 and 2 due to switchyard voltage greater than 530 kV

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed four operability evaluations affecting risk-significant mitigating systems, listed below, 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 SDP.

The inspectors' review included a verification that determinations of operability were made as specified by procedure OP-AA-102, Revision 3, "Operability Determination."

- CR 322188: "Drains are not flowing on the Unit 1 Service Water Spray Arrays"
- CR 321868: "Internal circumferential scratches visible at rolled joints on evaporator tube bundle for 2-HV-E-4A"
- CR 325458: "Outboard Seal/Gland Leakage of 2-FW-P-2 Steam Turbine"
- CR 326122: "High voltage on 500KV Grid"

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing

a. Inspection Scope

The inspectors reviewed five post maintenance test procedures and/or test activities, as appropriate, for selected risk-significant mitigating systems 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 licensee procedure VPAP-2003, Revision 13, "Post Maintenance Testing Program."

- WO 00797190-01: Replace expansion joint 1-SW-REJ-8A
- WO 59101835964: Replace square root card for 1-FW-FI-100C associated with 1-FW-P-3A
- WO 59080501001: Lube flexible couplings and clean lube oil coolers for 1-CH-P-1A
- WO 59078284801: Install oil sample port in accordance with design change package (DCP) 07-131 service water pump
- WO 59101866064: Repair of Unit 1 1-CC-P-1B

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities

a. Inspection Scope

The inspectors reviewed the Outage Safety Plan (OSP) and contingency plans for the Unit 1 refueling outage, which began March 8, 2009, to confirm that the licensee had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense-in-depth. The inspectors used Inspection Procedure 71111.20, "Refueling and Outage Activities," to observe portions of the shutdown, cooldown, and maintenance activities to verify that the licensee maintained defense-in-depth commensurate with the outage risk plan and applicable TS. The inspectors monitored licensee controls over the outage activities listed below.

- Licensee configuration management, including daily outage reports, to evaluate defense-in-depth commensurate with the outage safety plan and compliance with the applicable TS when taking equipment out of service.
- Adequate implementation of the boric acid corrosion control program by performing a containment walkdown after shutdown to identify boric acid leaks.
- Controls over the status and configuration of electrical systems and switchyard to ensure that TS and outage safety plan requirements were met.
- Licensee implementation of clearance activities to ensure equipment was appropriately configured to safely support the work or testing.
- Decay heat removal processes to verify proper operation and that steam generators, when relied upon, were a viable means of backup cooling.
- Installation and configuration of reactor coolant instruments to provide accurate indication and an accounting for instrument error.
- 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.
- Heat-up 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. Reactor Coolant System (RCS) integrity was verified by reviewing RCS leakage calculations and containment integrity was verified by reviewing the status of containment penetrations and containment isolation valves.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

For the six surveillance tests listed below, the inspectors examined the test procedures, witnessed testing, and 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 Tests:

- 1-PT-71.3Q, Revision 42, “1-FW-P-3B, ‘B’ Motor-Driven AFW Pump and Valve Test”
- 1-PT-14.1, Revision 48, “Charging Pump 1-CH-P-1A”
- 1-PT-63.1A, Revision 34, “Quench Spray System – ‘A’ Subsystem”
- 1-PT-57.1A, Revision 53, “Low Head Safety Injection Pump (1-SI-P-1A)”
- 1-PT-75.2A, Revision 49, “Unit 1 ‘A’ Service Water Pump (1-SW-P-1A)”

Containment Isolation Valve:

- 1-PT-61.3, Revision 31, “Containment Type C Test (1-SI-106)”

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Controls To Radiologically Significant Areas

Access Controls. The inspectors reviewed and evaluated licensee guidance and its implementation for controlling and monitoring worker access to radiologically significant areas and tasks associated with the 2009 Unit 1 (U1) Refueling Outage (RFO). The inspectors evaluated changes to, and adequacy of, procedural guidance; directly observed implementation of established administrative and physical radiological controls; appraised radiation worker and technician knowledge of, and proficiency in implementing, Radiation Protection (RP) activities; and assessed radiation worker exposures to radiation and radioactive material.

The inspectors directly observed the administrative and engineered controls for radiologically significant areas. The inspectors discussed and reviewed the controls and their implementation for Locked High Radiation Area (LHRA), LHRA greater than 15 rem/ per hour, and Very High Radiation Area (VHRA) keys and for storage of irradiated material within the spent fuel pool in detail. Additionally the inspectors discussed and reviewed the licensee’s controls for areas where dose rates could change significantly because of plant shutdown and refueling operations. The inspectors discussed the expected response to unexpected radiological conditions on backshift with two responsible RP Supervisors.

For selected tasks, the inspectors reviewed radiation work permit (RWP) details and attended pre-job briefings to assess communication of radiological control requirements to workers. Occupational worker adherence to selected RWPs and Health Physics Technician (HPT) proficiency in providing job coverage were evaluated through direct observations, remote observations, and interviews with licensee staff. Electronic dosimeter alarm set points and worker stay times were evaluated against applicable radiation survey results. For High Radiation Area (HRA) tasks involving significant dose gradients, the inspectors evaluated the use and placement of whole body and extremity dosimetry to monitor worker exposure.

Postings and physical controls established within the radiologically controlled area (RCA) for access to the U1 reactor containment building (RCB), the U1 and Unit 2 (U2) reactor auxiliary building (RAB) locations, radioactive material storage locations, decontamination building, and Independent Spent Fuel Storage Installation (ISFSI) were evaluated during facility tours. Results were compared to current licensee surveys and assessed against established postings and radiation controls. Licensee controls were observed for selected U1 and U2 RAB LHRA and VHRA locations.

The inspectors evaluated implementation and effectiveness of licensee controls for both airborne and external radiation exposure. The inspectors directly observed processes used for externally contaminated individuals, including those with potential uptakes of radioactive material. The inspectors reviewed administrative and physical controls including air sampling, barrier integrity, engineering controls, and postings for tasks having the potential for individual worker internal exposures to exceed 30 millirem committed effective dose equivalent.

Radiation protection activities were evaluated against UFSAR, TS, and 10 CFR Parts 19 and 20 requirements. Detailed procedural guidance and records reviewed for this inspection area are listed in Sections 2OS1 and 4OA5 of the attachment to this report.

Problem Identification and Resolution. Licensee corrective action program (CAP) documents associated with access controls to radiologically significant areas were reviewed and assessed. The inspectors evaluated the licensee’s ability to identify, characterize, prioritize, and resolve the identified issues in accordance with procedure PI-AA-200, Corrective Actions, Revision 2. Licensee CAP documents associated with access control issues, personnel radiation monitoring, and personnel exposure events which were reviewed and evaluated in detail during inspection of this program area are identified in Sections 2OS1, and 4OA5 of the attachment to this report.

The inspectors completed the 21 specified line-item samples detailed in inspection procedure (IP) 71121.01.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

a. Inspection Scope

Cornerstone: Initiating Events

The inspectors performed a periodic review of the three following Unit 1 and 2 PI's 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." 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 January 2008 through December 2008. Documents reviewed included applicable NRC inspection reports, licensee event reports, operator logs, and station performance indicators.

- Unplanned Scrams per 7000 Critical Hours
- Unplanned Power Changes per 7000 Critical Hours
- Unplanned Scrams With Complications

b. Findings

No findings of significance 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 Annual Sample: Closure of a Substantive Cross-cutting Aspect

a. Inspection Scope

The inspectors performed a review regarding the licensee's assessments and corrective actions for a substantive cross-cutting aspect (SCCA) in the area of human performance with a cross-cutting theme in the aspect of resources, complete, accurate and up-to-date design documentation, procedures, work packages, and correct labeling of components (H.2.c), as defined in Manual Chapter 0305, "Operating Reactor Assessment Program." This SCCA was the result of four findings in a twelve month assessment period with the same cross-cutting aspect of H.2.c, and was communicated to the licensee in the 2008 Mid-Cycle Performance Review and Inspection Plan letter, dated September 2, 2008, to the licensee. The inspectors also evaluated the actions against the requirements of the licensee's CAP as specified in procedure, PI-AA-200, Revision 5, "Corrective Action Program," and 10 CFR 50, Appendix B.

b. Findings and Observations

No findings of significance were identified. The inspectors reviewed the licensee's corrective actions contained in CR096021, "3 significant events in the last yr tied to detail in plant procedures," and specifically the related evaluations documented in common cause analysis (CCA) 000050 initiated in April, 2008. This CCA included a proprietary independent review and analysis contracted by the licensee. From both the independent and licensee's analysis, multiple corrective actions (CAs) and self assessment reports (SARs) were initiated. The inspectors determined that the following CAs were completed:

- CA081135: "Communicate results of PUA (procedure use and adherence)/Documentation Common Cause Analysis"
- CA081136: "Determine how to incorporate technical procedure reviews into SAs (self assessments)"
- CA086492: "Track Self-Assessment procedure change"
- CA081137: "Perform SA of select procedure/documentation completeness and accuracy"
- CA124730: "Perform extent of condition review of approximately 40 Mechanical Maintenance procedures"
- CA081138: "Document the strategy and the progress to review/upgrade station procedures"
- SAR000562: "Risk-based Technical Procedure Quality Assessment - I&C"
- SAR000563: "Risk-based Technical Procedure Quality Assessment - Electrical"
- SAR000564: "Risk-based Technical Procedure Quality Assessment - Mechanical"
- SAR000565: "Risk-based Technical Procedure Quality Assessment - Operations"
- SAR000566: "Risk-based Technical Procedure Quality Assessment - Control Operations"

The inspectors also determined that the following CAs were still in progress:

- CA089594: "Identify procedures that do not have ranges for acceptance criteria." This CA has a due date of February 27, 2009, and is under management review
- CA129542: "Control Operations to track completion of COEL procedure concern 09-0001." The procedure concern evaluation (COEL) has a due date of August 8, 2009
- CA089597: "Evaluate progress on I&C procedure upgrade progress." This CA has a due date of December 31, 2009
- SAR000567: " Risk-based Technical Procedure Quality Assessment - Engineering"
- Additional actions are pending relating to engineering procedures.

The inspectors continue to monitor the licensee's progress for the above CAs/SAR not yet completed. The licensee also completed the following additional CAs:

- CA083155/CA083156: The licensee initiated a requirement to perform a procedure use and adherence (PUA)/procedure quality (PQ) check during one of four required human performance (HU) evaluations each month. The licensee will track and trend the results of the HU evaluations on a monthly basis.
- CA081717/CA081718: The licensee will ensure the outage BOOST team has a focus on PUA/PQ and document the results.

- CA081719: The licensee provided additional training to supplemental employees arriving at the site to support outage work.
- CA081720/CA090786: The licensee developed two key performance indicators to measure performance in PUA/PQ and a track/trend process with routine leadership review.
- CA081721: The licensee added a HU tool task preview causal code to guidance and reference document, PI-AA-200-2001, Revision 1, "Trending."
- Procedure, PI-AA-200, Revision 5, "Corrective Action," was revised to require a CR for cross-cutting aspects that are identified as a result of a self-revealing or NRC identified finding. Additionally, a Level 2 cause evaluation is required for all cross-cutting aspects to ensure a appropriate review and CAs to resolve the issue.

The 2008 Mid-Cycle Performance Review and Inspection Plan letter stated that the SCCA could be cleared if licensee's CAs are effective as demonstrated by no new findings with the same cross-cutting aspect for the remainder of 2008. There were no new findings relating to the SCCA and, consequently, the SCCA was closed out in the annual assessment letter documented in NRC inspection report number 05000338/2009001 and 05000339/2009001.

4OA3 Event Followup

.1 (Closed) Licensee Event Report (LER) 05000339/2008-002-00: Both Trains of High Head Safety Injection Inoperable Due To Human Performance Error

Introduction. A Green NCV of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified by the NRC for multiple examples of a failure to accomplish a procedure for activities affecting quality which rendered both trains of high head safety injection (HHSI) inoperable. The licensee entered this issue into their CAP as CR114725.

Description. During a refueling outage with Unit 2 in Mode 3, the licensee removed the 2J EDG from service for maintenance at 03:03 on October 17, 2008. At approximately 22:00 on October 17, 2008, the licensee started work order (WO) 59101792367, “manually adjust/monitor thrust on 2-SI-MOV-2867A,” to obtain thrust/leakage data in accordance with a troubleshooting package. At 23:36 on October 17, 2008, the 'A' train HHSI was declared inoperable due to work performed by WO 59101792367. Per TS 3.8.1, Condition B, the 'B' train valve, 2-SI-MOV-2867B, became inoperable at 03:26 hours, October 18, 2008, since its respective emergency power source, 2J EDG, was inoperable for maintenance; however, the licensee failed to recognize this as evidenced by a review of their logs. At 11:25 on October 18, 2008, the licensee realized that both trains of HHSI were inoperable and entered TS 3.0.3. At 12:05 the licensee manually opened 2-SI-MOV-2867A to return the 'A' train of HHSI to service which allowed exit of TS 3.0.3. Subsequently, the licensee entered this issue in their CAP as CR114725, and performed an apparent cause evaluation, ACE014023.

As part of the LER review and closeout, the inspectors performed a detailed review of the related ACE and identified the following three examples of failure to adequately accomplish a procedure for activities affecting quality:

- The troubleshooting package for WO 59101792367 was prepared as directed by administrative procedure, MA-AA-103, Revision 1, “Conduct of Troubleshooting,” of which Attachment 2, Section 3.8, “Develop Instructions,” required the licensee to describe the equipment configuration during the troubleshooting to bound the effects of the troubleshooting and prevent creating an undesirable or unanalyzed equipment configuration, and identify the impact of the troubleshooting on plant equipment. While this section of the package stated "attached," the inspector determined that the attachment to the troubleshooting package containing the actual troubleshooting steps did not have any statements as specifically required. Therefore, the licensee failed to meet the requirements specified in procedure MA-AA-103 which was a failure to adequately accomplish a procedure.

- The troubleshooting package for WO 59101792367, stipulated the performance of procedure, 2-OP-8.11, Revision 3, “Troubleshooting ‘C’ Boric Acid Storage Tank Dilution,” section 5.3, “Testing 2-SI-MOV-2867A, BIT Inlet Isolation Valve, and 2-SI-MOV2867B, BIT Inlet Isolation Valve, for leak by,” up to step 5.10. Step 5.3.1 of this procedure states, “Verify Initial Conditions are satisfied.” Step 3.1 of the Initial Conditions states, “Review the equipment status to verify station configuration supports the performance of the procedure.” The inspectors determined that if step 3.1 had been properly accomplished, the licensee would have recognized that taking 2-SI-MOV-2867A out of service with the 2J EDG inoperable would render both trains of HHSI inoperable.

- A tag-out, 2-08-SI-0502, was associated with WO 59101792367 and was verified by two senior reactor operators. Licensee procedure OP-AA-200, Revision 2, “Equipment Clearance,” step 3.2.3, “Approval of a Tag-Out,” has multiple bulleted requirements of which one states, “Verify that the tag-out is in compliance with Technical Specification and regulatory requirements including maintaining redundant equipment operable.” The inspectors determined that if the two SROs had properly accomplished the procedure to verify the tagout, they would have recognized that the removal of 2-SI-MOV-2867A from service with the 2J EDG inoperable would have caused the inoperability of both trains of HHSI.

The inspectors noted that the ACE stated, "The apparent cause is inadequate emergent issues management. Other contributing causes include inadequate verbal instructions/communication, inadequate written instructions/communication, simultaneous multiple tasks, inadequate teamwork, judgment and inadequate planning." However, the inspectors determined that the ACE failed to specifically identify or discuss the examples above which were significant barriers that could have prevented the inoperability of both trains of HHSI or a loss of a safety function.

Analysis. The inspectors determined that the failure to accomplish procedures as noted in the examples above for an activity affecting quality, i.e., the troubleshooting of 2-SI-MOV-2867A, was a performance deficiency. This finding had a credible impact on safety because both trains of the HHSI were rendered inoperable, and manual operator action was required to place at least one train in service. The inspectors determined the finding was more than minor because it impacted the mitigating systems cornerstone objective to ensure the availability of the HHSI system, and the related attribute of human performance which involved the failure to adequately accomplish procedures.

The inspectors evaluated the finding using the SDP and determined that a Phase 2 evaluation was required. A regional Senior Reactor Analyst performed a Phase 3 evaluation under the SDP and determined the performance deficiency was of very low safety significance (Green). The evaluation was accomplished using the NRC's Probabilistic Risk Assessment computer model of the plant with Emergency Diesel Generator 1J and the Boron Injection Tank's inlet Motor Operated Valve 1867A set to always fail. The model was quantified, assuming the configuration lasted for nine hours. The dominant accident sequences were Losses of Offsite Power as the initiating event followed by the failure through various mechanisms of the 1H Emergency Diesel Generator and the Alternate Alternating Current Diesel Generator. Also, neither the failed Emergency Diesel Generators nor offsite power was recovered prior to core damage. The key assumptions were that Unit 2 was constructed similar enough that the Unit 1 Probabilistic Risk Assessment model could be used and the duration of the configuration was nine hours.

During the troubleshooting activities for 2-SI-MOV-2867A, the licensee failed to use the systematic process which already existed in their procedures to properly identify the impact of the existing inoperability of the 2J EDG on the operability of the HHSI system. Therefore, the inspectors determined that this finding involved the cross-cutting area of human performance, the component of decision making, and the aspect of safety-significant decisions using a systematic process, especially when faced with uncertain or unexpected plant conditions, to ensure safety is maintained. (H.1.a)

Enforcement. 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," states, in part, that activities affecting quality shall be accomplished in accordance with procedures. Contrary to the above, on October 17 and 18, 2008, the licensee failed to adequately accomplish procedure steps involving MA-AA-103, 2-OP-8.11 and OP-AA-200, which resulted in the inoperability of both trains of HHSI. Because the finding is of very low safety significance and because it has been entered into the licensee's CAP as CR114725, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000339/2009002-02, Failure to Accomplish Procedures Renders Both Trains of High Head Safety Injection System Inoperable.

.2 (Closed) Supplemental Licensee Event Report (LER) 05000339/2008-001-01 Manual Reactor Trip Due to Shutdown Bank 'A' Group Step Counter Deviation Greater Than Allowed

On February 8, 2008, with Unit 2 in Mode 3, zero percent power and preparing for a unit restart following a planned maintenance outage, the operators received a rod control urgent failure alarm during withdrawal of the 'A' shutdown bank control rods. The operators noted that the respective group step counters had deviated by three steps and consequently, in accordance with Technical Requirements Manual 3.1.3, initiated a manual reactor trip. In this supplemental LER the licensee documented their determination that the direct cause of the rod control urgent failure was two spread pins, 1 and 3, located on the slave cycler moveable decoder card, A503, in the logic cabinet which caused an intermittent open circuit of the two redundant ground connections on the A503 card. The licensee revised their procedures to incorporate a vendor practice to periodically reform the pins. The licensee documented the corrective actions associated with this event in CR090778.

The inspectors' review of this supplemental LER concluded that there was no additional information that changed the original characterization of no findings of significance documented in NRC Integrated Inspection Report 05000338/2008005 and 05000339/2008005. This LER is closed.

4OA5 Other

.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 of significance were identified.

.2 (Closed) Temporary Instruction (TI) 2515/173 Ground Water Protection Initiative (GPI)

a. Inspection Scope

On February 2 through 5, 2009, the inspectors reviewed North Anna Power Station groundwater protection actions for compliance with the industry initiative described in the NEI Document "Industry Ground Water Protection Initiative- Final Guidance Document, August 2007" (NRC document reference ML072600292 and ML072610036).

The inspectors review covered the licensee site characterization with regard to site hydrology and geology, site risk assessment with respect to systems structures and components that posed a credible risk for licensed material to reach ground water, on-site ground water monitoring programs, established programs for remediation in the event of inadvertent releases, and record keeping which allows for proper planning and resource allocation for the eventual decommissioning of the site.

The inspectors reviewed the communication aspects of the sites program. These aspects included the requirements for initial and periodic briefing of sites GPI program with designated State and Local officials, voluntary notifications of State/Local officials and follow-up notifications of the NRC regarding any significant

on-site spills or environmental water samples exceeding the criteria in the Radiological Environmental Monitoring Program.
The programmatic inclusion of follow-on NRC documentation requirements including 30 day reporting and annual reporting was also reviewed. The inspectors reviewed program to determine if it included the oversight commitments of having a self assessment of the program not later than 12/31/2008 and periodically at an interval not to exceed 5 years there after and a separate review under the auspices of NEI within one year of the initial self assessment and periodically thereafter at an interval not to exceed 5 years.

The inspectors reviewed various program documents including procedures, condition reports, self assessment and corrective actions and determined that the licensee's program was implemented consistent with the NEI document with one action scheduled but not yet completed. The independent NEI review is scheduled for June 2009, approximately 6 months after the self-assessment. The requirements specified in TI 2515/173 have been met.

b. Findings

No findings of significance were identified.

.3 IP 60855.1 Radiological Controls

a. Inspection Scope

The inspectors observed and evaluated implementation of radiological controls, including RWPS and postings, and discussed the controls with an HPT and HP supervisory staff.

Radiological control activities for ISFSI areas were evaluated against 10 CFR part 20, 10 CFR Part 72, and applicable licensee procedures. Documents reviewed are listed in Section 4OA5 of the attachment to this report.

b. Findings

No findings of significance were identified.

.4 (Open) Temporary Instruction (TI) 2515/172, Reactor Coolant System Dissimilar Metal Butt Welds (DMBWs)

a. Inspection Scope

From March 16-20, 2009, the inspectors reviewed the licensee's activities related to the inspection and mitigation of DMBWs in the Reactor Coolant System (RCS) to ensure that the licensee activities were consistent with the industry requirements established in the Materials Reliability Program (MRP) document MRP-139, Primary System Piping Butt Weld Inspection and Evaluation Guidelines, July 2005. This inspection was limited to the review of licensee activities with regard to the scoping, classification, inspection, and mitigation of dissimilar metal butt welds in accordance with the industry requirements of MRP-139.

TI 2515/172 was performed in 2008 and documented in Inspection Report 2008005. During that time a program review (per TI 2515/172 paragraph 03.05) was performed.

b. Findings and Observations

No findings of significance were identified.

MRP-139 Baseline Inspections

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

Yes. The licensee has performed all required baseline inspections at the time of this review.

Pressurizer – The licensee installed Full Structural Weld Overlays on all PRZ DMBWs within the scope of the MRP-139 program during the fall 2007 refueling outage for unit 1, and the spring 2007 refueling outage for unit 2. Unit 1 weld overlays were ultrasonically inspected per the associated relief request requirements this outage (spring 2009) and Relief Request NDE-005. The inspectors reviewed documentation for the UT exams of the Safety Nozzle 'A', Safety Nozzle 'C', and Relief Nozzle 'D'; there were no recordable indications. The inspectors also verified future exams of the PZR weld overlays will be scheduled in accordance with the requirements of the associated relief requests and Section XI of the ASME BPVC Code, Non-mandatory Appendix Q.

Reactor Vessel Nozzles – The hot and cold leg nozzles from the reactor vessel are stainless steel and not susceptible to PWSCC. Therefore, no MRP-139 inspections are required for those welds.

Steam Generator (SG) – When the SGs for Unit 2 were replaced in February 1996, an Alloy 52 inlay was installed over the DMBWs for the hot and cold Leg nozzles. Therefore, the SG hot and cold leg DMBWs for Unit 2 are not in contact with the RCS and no subsequent MRP-139 inspections are required for those welds.

The Steam Generators for Unit 1 were replaced in 1994, but there were no inlays applied to the DMBWs of Unit 1. The licensee has scheduled the remaining baseline volumetric examinations for Unit 1 hot legs greater than 14 inches and the cold legs during the spring 2009 outage. This outage (spring 2009) the licensee inspected both hot and cold leg nozzles of all three Unit 1 SGs with a PDI qualified phased array UT procedure, with no service related indications. This meets the deadline for welds at hot leg temperatures and >14" nominal pipe size of December 31, 2009; and welds at cold leg temperatures of December 31, 2010.

Therefore, the licensee has met the MRP-139 deadlines for baseline examinations of all welds scoped into the MRP-139 program.

2) Is the licensee planning to take any deviations from MRP-139 requirements?

No, the licensee has not submitted any requests for deviation from MRP-139 requirements.

Volumetric Examinations

The inspectors selected the following DMBW for the volumetric examination review: Unit 1, SG “A” Hot Leg primary loop nozzle.

1) For each examination inspected, was the activity performed in accordance with the examination guidelines in MRP-139, Section 5.1, for unmitigated welds or mechanical stress improved welds and consistent with NRC staff relief request authorization for overlaid welds?

Yes. The licensee conducted the examination per ER-NA-NDE-UT-815, Rev 0, which utilizes Zetec_OmniScanPA_03, which is certified by PDI; and met the requirements for unmitigated weld overlays of MRP-139, Section 5.1. The configuration of the SG nozzles welds were such that they required the development of a mockup. The inspectors verified the licensee’s evaluation was in accordance with PDI’s “Dissimilar Metal Weld Mock Up Criteria,” Rev A.

2) For each examination inspected, was the activity performed by qualified personnel?

Yes. The examination was conducted by PDI certified personnel for procedure Zetec_OmniScanPA_03.

3) For each examination inspected, was the activity performed such that deficiencies were identified, dispositioned, and resolved?

Yes, the activity was performed such that deficiencies were identified, dispositioned, and resolved using the licensee’s corrective action program.

Weld Overlays

Sample not available.

Mechanical Stress Improvement (Not Applicable)

Sample not available.

In-service Inspection Program

This reporting requirement was addressed previously in inspection report 05000338,338/2008005; no new information was noted during this inspection.

.5 (Closed) Temporary Instruction (TI) 2515/176, EDG TS Surveillance Requirements Regarding Endurance and Margin Testing

Inspection activities for TI 2515/176 were previously completed and documented in inspection report 05000338, 339/2008005, and this TI is considered closed at North Anna Power Station; however, TI 2515/176 will not expire until August 31, 2009. The information gathered while completing this temporary instruction was forwarded to the Office of Nuclear Reactor Regulation for review and evaluation.

4OA6 Meetings, Including Exit

.1 Exit Meeting Summary

On April 21, 2009, the resident inspectors presented the inspection results to Mr. Daniel Stoddard 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.

.2 Ground Water Protection Inspection Exit

On February 5, 2009, the inspectors discussed the results of the Ground Water Protection Inspection (TI 2515/173) with Mr. Dan Stoddard and members of his staff.

.3 Occupational RP Inspection Exit

On March 12, 2009, the inspectors discussed results of the Occupational RP Inspection (IP 71121.01) with Mr. Dan Stoddard and members of his staff.

.4 ISI and SGISI Inspection Exit

An exit meeting for the ISI and SGISI portions was conducted on March 20, 2009 with licensee management.

ATTACHMENT: SUPPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel:

W. Anthes, Manager, Nuclear Maintenance
M. Bradley, Health Physics Supervisor
J. Breeden, Supervisor, Radioactive Analysis and Material Control
B. Britt, BACC Program Manager
E. Dreyer, Health Physics Supervisor
R. Evans, Manager, Radiological Protection and Chemistry
E. Hendrixson, Director, Nuclear Safety and Licensing
T. Huber, Director Nuclear Engineering
S. Hughes, Manager, Nuclear Operations
P. Kemp, Supervisor, Station Licensing
L. Lane, Plant Manager
M. Lane, Health Physics Supervisor
G. Lear, Manager, Organizational Effectiveness
T. Maddy, Manager, Nuclear Protection Services
G. Marshall, Manager, Nuclear Outage and Planning
C. McClain, Manager, Nuclear Training
F. Mladen, Manager, Nuclear Site Services
S. Morris, Supervisor Material ISI/NDE
B. Morrison, Supervisor Nuclear Engineering
D. Plemen, Health Physics Supervisor
J. Scott, Supervisor, Nuclear Training (operations)
R. Simmons, Health Physics Supervisor
D. Stoddard, Site Vice President
M. Young, Health Physics Supervisor

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000338/2009002-01	URI	Degradation of the Containment Sump (Section 1R08)

Closed

05000339/2008-002-00	LER	Both Trains of High Head Safety Injection Inoperable due to Human Performance Error (Section 4OA3.1)
05000339/2008-001-01	LER	Manual Reactor Trip Due to Shutdown Bank ‘A’ Group Step Deviation Greater Than Allowed (Section 4OA3.2)
05000338, 339/2515/173	TI	Ground Water Protection Initiative (GPI) (Section 4OA5.2)
05000338, 338/2515/176	TI	EDG TS Surveillance Requirements Regarding Endurance and Margin Testing (Section 4OA5.5)

Opened and Closed

05000339/2009002-02	NCV	Failure to Accomplish Procedures Renders Both Trains of High Head Safety Injection System Inoperable (Section 4OA3)

Discussed

05000338/2515/172	TI	Reactor Coolant System Dissimilar Metal Butt Welds (DMBWs) (Section 4OA5.4)

LIST OF DOCUMENTS REVIEWED

Section 1R08 Inservice Inspection (ISI) Activities

Procedures

0-MCM-1801-01 Rev 19, Welding Safety-Related and Seismic-Related Equipment – 10/10/07
ER-AA-NDE-UT-815 Rev 0, "Procedure for Encoded Phased Array Ultrasonic Examination of Dissimilar Metal Piping Welds"
ER-AA-NDE-130 Rev 1, "Storage and Control of Calibrated NDE Equipment, Calibration Standards and Consumable NDE Materials"
ER-AA-NDE-UT-702 Rev 2, Ultrasonic Examination of Ferritic Vessel Welds Greater Than 2.0 in Thickness – 09/08/08
ER-AA-NDE-VT-604 Rev 0, "Visual Examination for Leakage of PWR Reactor Head Penetrations"
ER-AA-NDE-VT-607 Rev 0, "VE Examination of Pressure Retaining Welds in Class 1 Components Fabricated with Alloy 600/82/182 Materials"
ER-AA-NDE-VT-605 Rev 0, "IWE Visual Examination Procedure"
ER-AP-BAC-10 Rev 3, Boric Acid Corrosion Control Program
ER-AP-BAC-101 Rev 1, Boric Acid Corrosion Control Program (BACCP) Inspections
ER-AP-BAC-102 Rev 2, Boric Acid Corrosion Control Program (BACCP) Evaluations
ER-NA-BAC-101-1001 Rev 0 , NAPS Site Specific Boric Acid Corrosion Control Program (BACCP) Inspection and Evaluation Requirements
1-PT-46.21, RCS Pressure Boundary Components Affected By Boric Acid Accumulation
1-PT-48.1A Rev 3, Visual Inspection of Class 2 and 3 Bolted Components Inside Reactor Containment – A Group
1-PT-48.2A Rev 2, Visual Inspection of Class 2 and 3 Bolted Components Outside Reactor Containment – A Group
1-PT-48.4, Completed Bare Metal Visual Inspection of Vessel BMI Nozzles, Unit 1, September 11, 2007
1-PT-48.5 Rev 4, Leakage Inspection Above Reactor Vessel Head
1-PT-48 Rev 17, Visual Inspection of Reactor Coolant Pressure Boundary Components
1-PT-48.6, Completed Vessel Head Bare Metal Visual Inspection, Unit 1, September 24, 2007
2-PT-48.7, Completed Vessel Head Volumetric Inspection, Unit 2, April 6, 2007
54-ISI-801 Rev 2, Automated UT of PWR Vessel Shell Welds
54-ISI-821 Rev 2, ID Automated Ultrasonic Examination of Austenitic and Dissimilar Metal Piping Welds for Detection and Length Sizing
54-ISI-855 Rev 4, Automated Ultrasonic Examination of Reactor Vessel Nozzle to Shell Welds and Inner Radius Regions from the Nozzle Bore
Boric General Training, Boric Acid Corrosion Control Program (BACCP) Inspections, ER-AP-BAC-101
NAP-SGPMS-001, Rev. 12, North Anna Site Specific Eddy Current Analysis Guidelines North Anna Unit 1 1R20 (U1R20)
EPRI, Rev. 7, Steam Generator Management Program: Pressurized Water Reactor Steam Generator Examination Guidelines
ER-AA-ISI-10, Rev. 0, ASME Section XI Inservice Inspection Program
Areva 54-ISI-400, Rev. 17, Multi-Frequency Eddy Current Examination of Tubing
ER-AP-SGP-103, Rev. 0, Steam Generator Condition Monitoring and Operational Assessments
ER-AP-SGP-102, Rev. 0, Steam Generator Degradation Assessment
ER-AP-SGP-101, Rev. 2, Steam Generator Program

Calculations

Corrective Action Documents

CR326310	“Programmatic CR to generation WO”
CR325382	“U1 Outage Emergent DCP”
CR326618	“Work request – repair RS sump area leak chase pressure connections”
OD000272	“Address containment integrity and boric acid on carbon steel”
CR019200	“Boric acid found on components while performing 1-PT-48.2B”
CR021733	“Boric acid was found on 1-CH-ICV-3266”
CR020995	“Inactive boric acid leak on 1-RH-030”
CR020882	“Inactive boric acid leak on 1-RS-P-3A”
CR021245	“Leak chase test tap in Unit 1 containment sump rusted off”
CR326926	“Foreign Objects Found in “A” Steam Generator During the Foreign Object Search Examination”
CR327363	“Camera Probe Wedged or Stuck in “C” Steam Generator During the Final Pass of the Post Lance Inner Tube Bundle Inspection”
*CR327436	“Calibration of 1-CH-FT-1154A is required based on a review of shutdown data”
*CR327503	“Boric acid noted on pipe insulation in RC 241 penetration area”
*CR327575	“Minor boric acid discovered at various instrument valves”
*CR327582	“Boric acid found on fitting of 1-CH-PT-1154”
*CR327579	“Boric acid found on ICV’s”

*Documents created as a direct result of this inspection.

Other

CA016608	BACCP Evaluation performed for CR019200
CA018141	BACCP Evaluation performed for CR020995
CA017883	BACCP Evaluation performed for CR020882

07-0792, North Anna Power Station, N1R19 Outage Form OAR-1 Owner’s Activity Report, 12/06/07

08-0595, Virginia Electric and Power Company (Dominion) North Anna Power Station Unit 1 Fourth Interval Inservice Inspection Plan and Associated Proposed Alternatives and Relief Requests, 10/07/08

08-0595A, Virginia Electric and Power Company (Dominion) North Anna Power Station Unit 1 Review and Approval Schedule – Fourth Interval Inservice Inspection Plan and Associated Proposed Alternatives and Relief Requests, 01/30/2009

08-0595B, Virginia Electric and Power Company (Dominion) North Anna Power Station Unit 1 Revised Relief Request CS-001 For Snubber Examination and Testing For the Fourth Interval ISI Program, 02/18/2009

09-097, Virginia Electric and Power Company (Dominion) North Anna Power Station Units 1 and 2 Eliminate Commitment Associated With Bottom-Mounted Instrumentation Penetration Nozzles on the Reactor Pressure Vessel Lower Head Examination Frequency, 02/18/2009

09-114, Virginia Electric and Power Company (Dominion) North Anna Power Station Units 1 and 2 Relief Request CMP-24 and CMP-25, 02/18/2009

Tapes on Visual Inspections for SG 1B Steam Drum & 1A Cold Leg Top of Tube Sheet Post Lance Foreign Objects

Areva 51-9101656-00, North Anna Unit 1 1R20 – EPRI Appendix H Eddy Current Technique Review

Steam Generator Inspection Plan and Schedule

Steam Generator Data Acquisition and Data Analysis Daily Reports

Steam Generator Condition Monitoring and Operational assessment, North Anna Unit 1, SG C, September 2004 Refueling Outage

Steam Generator Condition Monitoring and Operational assessment, North Anna Unit 1, SG B, March 2006 Refueling Outage EOC 18
Eddy Current Inspection Equipment Certification and Calibration Records
Data Acquisition and Data Examiner Training, Qualification, Certification, and Vision Examination Records
Areva Eddy Current Examination Technique Specification Sheets – ETSS-RPC002-MIZ80-R0, ETSS-RES001-MIZ80-R0, ETSS-RPC001-MIZ80-R0, and ETSS-BOB001-MIZ80-R0
ET-N-07-0091, Rev. 1, Steam Generator Condition Monitoring and Operational Assessment EOC 19/ REOC 10
ET-N-09-0016, Rev. 0, North Anna Unit 1 – Spring 2009 Steam Generator Degradation Assessment
Westinghouse letter LTR-SGDA-06-79, SUBJ: North Anna Unit 2 Replacement Steam Generator Safe End to Inlet and Outlet Nozzle Welds, June 28, 2006
ER-AA-MAT-11, "Alloy 600 Management Program," Rev 4
Relief Request NDE-005
59- W893-00044, Vendor Technical Manual, "Model W-11009-A1 Controlled Leakage Seal Reactor Coolant Pump," Rev 9
Dominion Memorandum, Subject: Previous NAPS BMI Bare Metal Visual Examinations, Rev 01, March 17, 2009

Section 2OS1: Access Controls to Radiologically Significant Areas

Procedures, Manuals and Guidance Documents

- C-HP-1031.023, RWP Dosimetry: Exposure control Support, Revision 5
- C-HP-1031.024, Administrative Dose Control, Revision 5
- C-HP-1031.025, Dosimetry Requirements For Site Restricted Areas, Revision 3
- C-HP-1032.090, Providing Job Coverage Using Remote Monitoring Technology, Revision 0
- C-HP-1041.011, Evaluating And Tracking Intakes Of Radioactive Material, Revision 4
- C-HP-1041.020, DAC-Hour Determination Based On Bioassay Results, Revision 3
- C-HP-1041.021, Radionuclide Intake Determination Based On Bioassay Results, Revision 6
- C-HP-1041.022, Internal Dose Calculation Based On DAC-Hour Exposure. Revision 5
- C-HP-1041.023, Internal Dose Calculation Based On Radionuclide Intake, Revision 7
- RP-AA-201, Access Controls for High and Very High Radiation Areas, Revision 2
- RP-AA-202, Radiological Posting, Revision 0
- RP-AA-221, Radiological Survey Records, Revision 0
- RP-AA-222, Radiation Surveys, Revision 0
- RP-AA-223, Contamination Surveys, Revision 0
- RP-AA-224, Airborne Radioactivity Surveys, Revision 0
- RP-AA-230, Personnel Contamination Monitoring and Decontamination, Revision 1
- RP-AA-231, Radiological Control Areas, Revision 0
- RP-AA-240, Discrete Radioactive Particle Control, Revision 0
- RP-AA-261, Control Of Radiological Diving Activities, Revision 0
- RP-AA-260, Control Of Radiography, Revision 1
- RP-AA-262, Steam Generator Primary Side Work Controls, Revision 0
- RP-AA-265, Failed Fuel Action Plan, Revision 0

Surveys, Data, Records
- RWP 08-1224, Load, Transport, and Store NUHOMS Spent Fuel Dry Storage Cask. To Include Video, Fuel Shuffle, Prepping Cask, Support Activities (RP, Decon, Welding, QC, Crane), Cask Disassembly/Reassembly, Leak Tests and Associated Activities at the ISFSI Pad.
- RWP 08-1502, General Entry During Subatmospheric Conditions For The Purpose Of Walkdowns, Inspections, Radiological Surveys, Minor Maintenance and Adjustments.
- RWP 08-2215, Disassemble, inspect, repair, repack, cutout and replace valves and flanges in support of the U-2 Outage. To include calibrations and repairs on 2-RC-LT-RVLIS sensors. To include all associated support.
- RWP 08-2247, Lift Rx Head Shield and Replace Interim Lift Ring Block. Includes All Associated Support.
- RWP 08- 2252, Lift and Set Reactor Vessel Head. Includes Inspection in Cavity and all Support Functions.
- RWP 09-3215 Disassemble, Inspect, Repair, Repack, Cut Out And Replace Valves And Flanges In Support Of U1 Outage. To Include Calibrations And Repairs On 1-RC-LT-RVLIS.
- RWP 09-3220 Perform ultrasonic, Magnetic Particle, and Liquid Penetrant Examinations for the ASME IX In-service Inspection Program. Prep Welds, Remove/ Replace Support Clamps and Permanent Marking of Welds with Low Stress Stamps. Inspection of Upper and Lower Taps on Loop Stop Valve. Perform Secondary Erosion and Corrosion Inspection. Include NDE/UT Thickness Inspection on Feedwater, Main Steam, and Blowdown Lines. Volume in drive C has no label.
- RWP 09-3260 Perform B and C Steam Generator Inspection, Manway and Diaphragm Removal/ Reinstallation During the Unit 1 2009 Refueling Outage.

CAP Documents (Condition Reports)
- CR092500
- CR093845
- CR096047
- CR096830
- CR100852
- CR101550
- CR102219
- CR102619
- CR103616
- CR109698
- CR109427
- CR110550
- CR317468
- CR318085
- CR319593
- CR324692

Section 4OA5: Other Activities

TI 2515/173 Ground Water Protection Initiative

Procedures
- CH-71.T01, Tritium In Water: By Liquid Scintillation Counting, Revision 3
- CM-AA-BPM-10, Buried Piping Monitoring Program, Revision 0
- HP-3051.020, Ground Water Protection Program, Revision 0

- RP-AA-502, Ground Water Protection Program, Revision 0
- RP-AA-503, Radiological Decommissioning Records – 10 CFR 50.75(g) Program, Revision 0
- RP-AA-504, Remediation Process for the Ground Water Protection Program, Revision 0
- VPAP-2103N, North Anna ODCM, Pages 43 and 44, Revision 15
- VPAP-2802, Groundwater Protection Voluntary Communication Notification and Reports, Pages 186 and 187, Revision 30
- T-HSP-ENV-001, Enhanced Groundwater Initiative, Revision 2 (Expired Interim Procedure)

Documents
- SA 000586, Ground Water Protection Initiative, 12/22/2008
- MACTEC Engineering and Consulting, Inc. Report of Monitoring Well Installation and Sampling Activities North Anna Nuclear Power Station, 07/2/2008
- ANI Guideline 07-01, Potential for Unmonitored and Unplanned Off-Site Releases of Radioactive Material, March 2007

Central Reporting System Documents
- CA 004551
- CA 019202
- CA 019203
- CA 090435
- CR 009765
- CR 011000
- CR 117932
- CR 017997
- SAA 003220
- SAA 003222
- SAR 000586

ISFSI Radiological Controls
- Procedure 0-HSP-ISFSI-001, Independent Spent Fuel Storage Installation (ISFSI), Health Physics TLD Survey Surveillance, Revisions 3, 4 and 5
- Procedure 0-HSP-ISFSI-002, Nuhoms Dry Spent Fuel Storage System; Preparation, Loading, Transport, and T. S. Surveillance Surveys, Revision 2
- Spreadsheet: Controlled/ Restricted Area Dose Trending, 3/7/09

LIST OF ACRONYMS

ADAMS	Agencywide Document Access and Management System
ALARA	As Low As Reasonably Achievable
CA	Corrective Action
CAP	Corrective Action Program
CFR	Code of Federal Regulations
CR	Condition Report
EDG	Emergency Diesel Generator
HP	Health Physics
HPT	Health Physics Technician
HRA	High Radiation Area
IMC	Inspection Manual Chapter
IP	Inspection Procedure
ISFSI	Independent Spent Fuel Storage Installation
JPM	Job Performance Measures
LHRA	Locked High Radiation Area
LHSI	Low Head Safety Injection
NAPS	North Anna Power Station
NCV	Non-cited Violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
OD	Operability Determination
PARS	Publicly Available Records
PCP	Process Control Program
PI	Performance Indicator
QS	Quench Spray
RAB	Reactor Auxiliary Building
RCB	Reactor Containment Building
RCE	Root Cause Evaluation
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RFO	Refueling Outage
RP	Radiation Protection
RWP	Radiation Work Permit
RTP	Rated Thermal Power
SDP	Significance Determination Process
SR	Surveillance Requirements
TDAFWP	Turbine Driven Auxiliary Feedwater Pump
TS	Technical Specifications
U1	Unit 1
U2	Unit 2
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
VEPCO	Virginia Electric and Power Company
VHRA	Very High Radiation Area
VPAP	Virginia Power Administrative Procedure
WO	Work Order