UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
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ATLANTA, GEORGIA 30303-8931

May 5, 2003

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

SUBJECT: NORTH ANNA POWER STATION - NRC INTEGRATED INSPECTION REPORT NOS. 50-338/03-02 AND 50-339/03-02

Dear Mr. Christian:

On April 5, 2003, the US Nuclear Regulatory Commission (NRC) completed an inspection at your North Anna Power Station, Units 1 and 2. The enclosed integrated inspection report documents the inspection findings which were discussed on April 15, 2003, with Mr. J. Davis and other members of your staff.

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

This report documents an NRC-identified finding of very low safety significance (Green) which was determined to involve a violation of NRC requirements. However, because of the very low safety significance and because it is entered into your corrective action program, the NRC is treating this finding as a non-cited violation (NCV) in accordance with Section VI.A.1 of the NRC Enforcement Policy. Additionally, two licensee-identified violations which were determined to be of very low safety significance are listed in this report. If you contest any NCV in this report, 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.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of

NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at *http://www.nrc.gov/reading-rm/adams.html* (the Public Electronic Reading Room).

Sincerely,

/RA/

Kerry D. Landis, Chief
Reactor Projects Branch 5
Division of Reactor Projects

Docket Nos.: 50-338, 50-339
License Nos.: NPF-4, NPF-7

Enclosures: NRC Integrated Inspection Reports Nos. 50-338/03-02, 50-339/03-02
w/Attachment: Supplemental Information

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.: 50-338, 50-339
License Nos.: NPF-4, NPF-7

Report Nos.: 50-338/03-02, 50-339/03-02

Licensee: Virginia Electric and Power Company (VEPCO)

Facilities North Anna Power Station, Units 1 & 2

Location: 1022 Haley Drive
Mineral, Virginia 23117

Dates: January 5, 2003 - April 5, 2003

Inspectors: M. Morgan, Senior Resident Inspector
J. Canady, Resident Inspector
R. Chou, Reactor Inspector, RII (Sections 1R08 and 4OA3.5)
B. Crowley, Senior Reactor Inspector, RII (Sections 4OA3.3 and 4OA3.4)
K. Naidu, Senior Reactor Engineer, NRR (4OA3.4)
R. Hamilton, Health Physicist, RII (Section 4OA5)

Approved by: K. Landis, Chief, Reactor Projects Branch 5
Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000338/2003-002, IR 05000339/2003-002; Virginia Electric and Power Co.; 01/05/2003-04/05/2003; North Anna Power Station Units 1 & 2; Inservice Inspection Activities.

The report covered a three month period of inspection by resident inspectors and announced inspections by a regional health physicist, senior reactor inspector, reactor inspector and an NRR senior reactor engineer. One Green non-cited violation (NCV) was identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609 “Significance Determination Process” (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC’s program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process”, Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

Green. An incorrect pipe support installation, which contains a hydraulic snubber, did not meet drawing requirements and resulted in a capacity reduction. This pipe support protects the safety-related Low Head Safety Injection System from failures during seismic and other shock loadings.

An NRC-identified non-cited violation of 10 CFR 50, Appendix B, Criterion V, Instructions, Procedures, and Drawings was identified. This finding is more than minor because the support was incorrectly constructed and affected the objective of the Mitigating Systems cornerstone. Failure to correctly install the hydraulic snubber pipe support reduced the snubber design load capacity and challenged its ability to ensure the Low Head Safety Injection System remains functional following a seismic event. The issue was determined to be of very low safety significance because the as-found condition resulted in no loss of function. (Section 1R08).

B. Licensee-Identified Violations

Two violations of very low safety significance, which were identified by the licensee, have been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee’s corrective action program. The violations are listed in Section 4OA7 of this report.

CONTENTS

Report Details

Summary of Plant Status

Unit 1 began the inspection period at 100% power and entered an end-of-cycle coast down on January 9, 2003. The unit was shut down on February 23 for reactor vessel head replacement and a refueling outage.

Unit 2 began the inspection period defueled with the reactor coolant system loops full and the vessel head removed. The unit was synchronized to the grid on February 2 and 100% power was reached on February 5. The unit operated at this power level until an automatic reactor trip on March 31 due to a “C” steam generator low level coincident with steam flow/feed flow mismatch. The unit was restarted and synchronized to the grid the following day. Full power was reached on April 2. The unit operated at this power level for the remainder of the report period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R04 Equipment Alignment

a. Inspection Scope

The inspectors performed partial walkdowns of systems, structures, and components (SSC) to determine if they were correctly aligned in accordance with appropriate procedures and drawings. The partial walkdowns were performed on the Unit 2A and 2B Motor Driven Auxilliary Feedwater Pumps, the Unit 2J Emergency Diesel Generator (EDG), and the Unit 1J Emergency Diesel Generator. The inspectors also held discussions with operations personnel, verified that the appropriate technical specifications action statement were entered, and reviewed the following documents:

- Unit 2 A and B Motor Drive Auxilliary Feedwater Pumps, (2-PT-71.12, “AFW System Valve Position Verification”) and Station Drawing 12050-FM-074A, “Feedwater System”, Sheets 1 and 3;
- Unit 2 J Emergency Diesel Generator, (2-OP-6.6A, “Emergency Generator Pre-operational Check for 2H and 2J Diesel” and 0-PT-80, “AC Sources Operability Verification”); and,
- Unit1 J Emergency Diesel Generator (1-OP-6.6A, “Emergency Generator Pre-operational Check for 1H and 1J Diesel” and 0-PT-80, “AC Sources Operability Verification”).

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors assessed the implementation of the fire protection program using “NAPS Appendix R Report” and Virginia Power Administrative Procedure (VPAP)-2401, “Fire Protection Program.” The inspectors checked the control of transient combustibles and the material condition of the fire detection and fire suppression systems in the following eight areas:

- Unit 1 Mechanical Equipment Room (elevation 294);
- Unit 1 Cable Spreading Room (elevation 294);
- Unit 1 Main Transformer Areas (transformers A, B, and C);
- Technical Support Center (TSC) Ventilation Room and Support/Assembly Area (elevation 244);
- Unit 2 Cable Spreading Area (elevation 294);
- Unit 2 Mechanical Equipment Room (elevation 294);
- Unit 1 Auxiliary Building/Boric Acid Tank Area (elevation 244); and,
- Unit 1 Cable Vault/Tunnel (elevation 259).

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI) Activities

a. Inspection Scope

The inspectors observed in-process ISI work activities and reviewed selected ISI records. The observations and records were compared to the Technical Specifications (TS) and the applicable Code (American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, 1989 Edition with no addenda) to verify compliance.

Portions of the following Unit 1 ISI were observed:

Ultrasonic (UT)	-	Weld 11, Drawing 11715-WMKS-0104A-1,Safety Injection Line 6"-SI-16-1502-Q1
Liquid Penetrant (PT)	-	Weld 11, Drawing 11715-WMKS-0104A-1,Safety Injection Line 6"-SI-16-1502-Q1
		Weld 92H, Drawing 11715-WMKS-0104A-2,Safety Injection Line 10"-SI-239-153A-Q2
Visual (VT)	-	Tank Support Legs 1 to 4, Drawing 11715-WMKS-SI-TK-2 Line1-SI-TK-2

Qualification and certification records for examiners and nondestructive examination (NDE) procedures for the above ISI examination activities were reviewed. Work Orders and examination documents were reviewed.

The inspectors also performed a general walkdown in nearby areas to assess the condition of the plant.

The inspectors reviewed the licensee's responses to NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity." The inspectors selected samples from the leaks identified by the licensee during the Unit 1 outage and independently observed the components to assess the significance. The inspectors also independently performed a general walkdown inside the containment to look for leaks.

The inspectors reviewed the licensee's implementation of NRC Regulatory Issue Summary (RIS) 2003-01, "Examination of Dissimilar Metal Welds," Supplement 10 to Appendix VIII of Section XI of the ASME Code. The inspectors discussed the issue with the licensee's engineers.

b. Findings

Introduction. A Green NRC-identified Non-Cited Violation (NCV) was identified for failure to correctly install a hydraulic snubber pipe support 1-SI-HSS-107 for safety injection line 10"-SI-238-1502-Q2 as required by the support drawing. The drawing addresses activities required by 10 CFR 50, Appendix B, Criteria V, Instructions, Procedures, and Drawings, which in part requires that activities affecting quality shall be prescribed by documented instructions, procedures, and drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings.

Description. On March 13, 2003, while observing a PT on a safety injection line, the inspectors identified hydraulic snubber (4" bore) pipe support 1-SI-HSS-107 was not correctly installed according to drawing No. 11715-PSSK-104A.10, Sheet 1 of 1. The snubber supports Low Head Safety Injection Pump discharge piping and is not train independent.

The support was installed with three defects: the eye rod of the snubber at the pipe clamp contacted the pipe clamp second pin, which was not designed to take any loads; the pipe clamp size was smaller than required; and the end attachment, snubber, and pipe clamp were not installed in a straight line but with skewed angles more than allowable in both vertical and horizontal directions. The contact between the eye rod and the second pin of the pipe clamp might not provide freedom of movement for the thermal expansion expected during normal operation and might break the pin when temperatures increase and the snubber expands. The installed smaller pipe clamp did not meet the capacity as stated in the manufacturer's design for a 4" bore hydraulic snubber. The installation with skewed angles in the vertical and horizontal directions would reduce the snubber capacity. All three defects would contribute to a snubber capacity reduction and challenge its ability to perform its intended function to resist the shock loads during seismic or transient conditions.

The licensee issued Plant Issue N-2003-1160 for the problems identified by the inspectors. Plant Issue Resolution N-2003-1160-R1 and Category 3 Root Cause Evaluation N-2003-1160-E1 stated that the undersized pipe clamp (3/4" X 4" for 2 and ½" bore snubber) has a capacity of 10.35 Kips (10,350 pounds) which is well below the pipe support design load of 18.13 Kips and documented the snubber as inoperable. The system was also declared inoperable due to affecting both trains. The Resolution did not address capacity reduction due to the skewed angles. The licensee performed further research and found that the latest piping stress analysis and support calculation indicated the design load to be 6.06 Kips instead of 18.13 Kips. The licensee issued Work Order 486858 Task 02 to realign and adjust the snubber. The licensee also sent the as-found condition (skewed angles of 10 degrees in both directions) to the vendor for past operability evaluation. The vendor concluded that the snubber with the skewed angles would not reduce the capacity below 6.06 Kips. Therefore, the licensee concluded that the snubber had been operable.

Analysis. The inspectors determined that this finding was associated with protection against external factors attributes (seismic) and affected the Mitigating Systems Cornerstone. Snubbers are designed to protect safety-related systems and components from failures due to seismic and other transient loadings. This improperly installed snubber challenges its ability to ensure the Low Head Safety Injection System remains capable following a seismic event. The issue was evaluated using the significance determination process based on Plant Issue Resolution N-2003-1160-R1 (Revision 1). The Resolution concluded that the snubber was operable with the reduced capacity for the design load. Since there was no loss of function, this finding was of very low safety significance (Green).

Enforcement. 10 CFR 50, Appendix B, Criteria V, Instructions, Procedures, and Drawings in part requires that activities affecting quality shall be prescribed by documented instructions, procedures, and drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Contrary to the above, on March 13, 2003, it was determined that the licensee failed to correctly install hydraulic snubber pipe support 1-SI-HSS-107 in accordance with design drawing 11715-PSSK-104A.10 Sheet 1 of 1, which resulted in a snubber with reduced capacity. Because the failure to correctly install the pipe support was of very low safety significance and the licensee documented this condition in Plant Issue N-2003-1160, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 50-338/2003-02-01, Failure to Correctly Install Hydraulic Snubber Pipe Support 1-SI-HSS-107.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspectors reviewed training documentation and observed licensed operator requalification training involving designated plant restart operators and supervisors. The inspectors observed simulator training for a mixed crew of supervisors/operators, and training that involved a simulated start-up of the plant to criticality, continued raising of power to the point of adding heat, testing of main turbine, closure of output breakers, and a table top discussion of dilution to criticality. During the observed simulator session, the inspectors evaluated the crew performance in: 1) knowledge of plant technical issues, 2) use of the phonetic alphabet and 3-way communications, and 3) use of command and control techniques by supervisors.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

For the equipment issues described in the plants issues listed below, the inspectors evaluated the licensee's effectiveness of the corresponding preventive and corrective maintenance. The inspectors performed walk-downs of the accessible portions of the system, performed in-office reviews of procedures and evaluations, and held discussions with system engineers. Inspectors compared the licensee's actions with the requirements of the Maintenance Rule (10 CFR 50.65) using VPAP 0815, "Maintenance Rule Program," and Engineering Transmittal CEP-97-0018, "North Anna Maintenance Rule Scoping and Performance Criteria Matrix." Additionally, the inspectors attended some of the licensee's scheduled Maintenance Rule Working Group Meetings.

- N-2002-0807 - reported a(1) status of Unit 1B component cooling pump in the licensee's maintenance rule (MR) database. The component has been placed in the a(1) status earlier but the MR working group had accepted the recommendation for its removal due to the inclusion of cascading hours for the maintenance on the 1-lll Battery;

- N-2003-1466 - failed attempt to open a safety injection to hot leg valve 1-SI-MOV-1890B (remotely) during an operating procedure for flushing down the line. The licensee initiated Work Order 487732-04 to investigate the problem; and,

- N-2003-0753 - the inability of the licensee to fully close Unit 1 reactor coolant system (RCS) Loop Isolation Valve, 1-RC-MOV-1593 (B Reactor Cool Loop Cold Leg Isolation Valve). Work Order 262082-01 was initiated by the licensee to repair the problem.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors reviewed the licensee's scheduled or emergent work activities to assess the management of plant risk. The inspectors evaluated if the assessments of risk were performed in accordance with the requirements of 10CFR50.65 (a)(4) and plant procedures. Additionally, the inspectors reviewed the licensee's actions to minimize the probability of initiating events, maintain the functional capability of mitigating systems, and maintain barrier integrity. The risk impact of performing the following work activities was assessed:

- Work Request 00482854, and Plant Issue N-2003-0561 - Emergent work associated with the Unit 2 B charging pump gear box cooler SW supply inlet check valve (2-SW-609-CKVALV). The pump was tagged out for troubleshooting, preventive maintenance, and repair activities due to cooler problems;

- Work Request 00143254 - the removal from service of the Unit 1 and Unit 2 AMSAC, the emergency switchgear flood wall, performance of the Instrument Air Compressor Accumulator test 1-PT-230-4, and the performance of the Unit 2 RCS power operated relief valve periodic test;

- Work Orders 00485027, 00485109, 00485182 and Plant Issue N-2003-0442 - Emergent work performed on the Unit 1 Instrument Air Compressor (1-1A-C-1). The Unit 1 instrument air compressor was out of service in conjunction with the Unit 2B Steam Dump Valve, the 2C Charging Pump, the Auxiliary Feedwater Pump, the Reserve Station Transformer, and the performance of switchyard work;

- Work Order 00486070-01 and Plant Issue N-2003-0917 - emergent work associated with the Unit 1 Control Rod Drive Mechanism (CRDM) Shroud Cooler (1-HV-E-6). The cooler was tagged out for inspection and repair due to identified leakage (boron coating) problems. The CRDM cooler was out of service in conjunction with the performance of reserve station transformer and the station switchyard work;

- Plant Issue N-2003-1489 - emergent work (review activities) performed on Unit 1 Steam Generator support coverings. The structural steel coatings were inspected by the licensee (as per NRC Generic Safety Issue 191) and they were noted to be peeling; and,

- Work Request WR00151572 and Plant Issue N-2003-1537 - emergent work and engineering assessment performed on the Unit 1 1H EDG #7 cylinder leak. The 1H EDG was taken out of service for inspection and potential repair due to the identified leakage around the cylinder (between 1 and 2 drops per second) problem. The 1H

EDG was inoperable for testing in conjunction with the performance of station switchyard work.

b. Findings

No findings of significance were identified.

1R14 Nonroutine Plant Evolutions and Events

.1 Unit 2 Feedwater Regulating Valve Oscillations

a. Inspection Scope

The inspectors observed operator response and followup activities associated with oscillations of the Unit 2 Feedwater Regulating Valves (FRV). Reactor Power was at approximately 70% on a power ramp to full power following the startup from an extended outage. The operators noted an alarm that indicated that the “C” feedwater recirculation valve to the main condenser had reopened about 15 minutes after it had been closed. The inspectors reviewed operator logs, plant computer data and plant operating procedures to determined what occurred and how the operators responded, and to verify that the response was in accordance with the plant procedures. The licensee placed the event in the corrective action program as Plant Issue N-2003-0465.

b. Findings

No findings of significance were identified.

.2 Unit 2 Trip Due to “C” FRV Failing Closed

a. Inspection Scope

On March 31, 2003, the inspectors observed operator response and followup activities associated with the trip of Unit 2 due to the C feedwater regulating valve failing to the closed position. The inspectors observed that the control room operators used the appropriate procedures and did not encounter any major complications in bringing the plant to a stable condition. The licensee provided appropriate detail about the event in Licensee Reactor Trip Report N-2-03-31-03 and Licensee Event Report (LER) 50-339/2003-001-00.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors evaluated the technical adequacy of operability evaluations to ensure that operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The reviewed operability evaluations were described in the following plant issues:

- N-2003-0041 - leakage that occurred from the 1H EDG jacket cooling pump following the operation of the EDG;
- N-2003-0084 - out of specification glycol concentration in the 1H EDG;
- N2003-0407 - packing leakage coming from the packing stuffing box on Unit 2 A Loop Hot Leg Drain Header Isolation Valve (2-RC-11-VALVE);
- N-2003-0985 - inappropriate fuses appeared to be installed in the “trip and close” circuitry for the Unit 1 Inside Recirculation Spray (RS) Pumps A and B;
- N2003-1034 - the material condition of the Unit 1 Containment Sump grading presented excessive amounts of rust;
- N2003-1239 - cracked fillet weldments found in the thermal sleeve welds in the B accumulator piping to the hot leg; and,
- N-2003-1352 - oscillations of the 1J EDG following hot restart testing.

b. Findings

No findings of significance were identified.

1R16 Operator Work-Arounds (OWAs)

a. Inspection Scope

The inspectors reviewed operator work-around (OWA) 98-OWA-B25, “Secondary side feed water heater reliefs.” This OWA involved large pressure transients following a unit trip from power that causes various secondary side relief valves to lift. Since relief valves do not always reseat, thus requiring isolation of the affected heater, an extra burden on the operations and maintenance departments was the result. This review included potential adverse impacts on the operator's ability to perform abnormal or emergency operating procedures.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

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

- Unit 2 Instrument Air Compressor (2-PT-228, “Atlas Copco ZR3-65 Instrument Air Compressor 2-IA-C-1 Performance Verification”); Work Order 00475550-01, “Grease compressor motor bearings”; and Work Order 00481011-01, “Replace time relays”;

- Unit 1J Emergency Diesel (1-PT-83.12J, “1J Diesel Generator Test (Start by ESF actuation) Followed by 24-hour run and hot restart test” and 1-OP-6.9, “Slow Start and Operation of 1J Emergency Diesel Generator”); Work Order 00486673, “Re-Tighten jacket coolant water coupling bolting”; and Plant Issue N-2003-1004, “Jacket Coolant Water Leak”;

- Unit 1 CC Return from the C RCP Lube Oil, Stator and Shroud Coolers-Outside Containment Isolation Valve (1-PT-212.1, “Valve Inservice Inspection (Component Cooling System”); and Work Order 00487434-01, “Limit switch problem”;

- Unit 1 CC Supply Check Valves to the Reactor Coolant Pump Thermal Barriers, (1-PT-210.7, “Valve inservice inspection (CC supply check valves to RCP thermal barriers”); and Work Order 00459898, “replace Unit 1A RCP thermal barrier CC inlet header check valve”; and,

- Unit 1B Header Control Valve to the B Steam Generator, [1-PT-214.5, “Valve inservice inspection (auxiliary feedwater valve position indication)”]; and Work Order 00487636, “limit switch problem.”

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

a. Inspection Scope

Prior to shutdown of Unit 1 for a planned outage (and vessel head replacement), the inspectors reviewed the licensee outage risk control plan. This review focused on management risk considerations and licensee response strategies for the potential loss of key safety functions during the outage.

Unit 1 was shutdown on February 23, for a planned Spring 2003 refueling outage and vessel head inspection/replacement. Following the unit shutdown, the inspectors toured the Unit 1 control room and containment. During this tour, the inspectors held various discussions with control room personnel, reviewed operation logs and observed control room readiness; specifically, the alignment/configurations of the emergency core cooling and decay heat removal systems for hot standby operation. The inspectors focused much of their review on the shutdown/cooldown parameters maintained during the Unit 1 shutdown to hot standby operations.

During the refueling outage, the inspectors observed, reviewed and evaluated, (as applicable), various activities such as; 1) preparations for use of Unit 1 refueling equipment, 2) results of walkdowns that identified a small number of minor reactor coolant system and charging/ safety injection system valve packing leaks, 3) performance of an 18 month 1H EDG 24 hour surveillance, 4) emergent work repair of the B loop stop isolation valve (1-RC-MOV-1593), and 5) disposition of coating degradations identified on the steam generator support structures. The inspectors also reviewed and evaluated routine outage logs/reports, and various maintenance rule-related activities.

A visual examination (VT-2) of penetration 50 on the vessel head was performed after shutdown and cooldown of the plant and there a "popcorn" boric acid residue was observed. The "popcorn" boric acid residue at the penetration was indicative of a through wall leakage at the penetration. An 8-hour report was made to the NRC operations center for a significant condition affecting a principle safety barrier. The licensee documented this issue in plant issue N-2003-0910.

The inspectors made several other Unit 1 containment entries during the outage. Some of the observations made during these entries included; 1) installation of the old control rod drive mechanisms on the new reactor vessel head, 2) emergent work involving the B safety injection accumulator thermal sleeve repair activities, and 3) control of transient combustible materials and maintenance of fire watch duties during welding/grinding activities. The unit remained in a shutdown condition at the end of the reporting period.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

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

- 1-PT-82H, “1H Emergency Diesel Generator Slow Start Test;”

- 1-PT-52.2A, “Reactor Coolant System Leak Rate (Computer Calculation);”
- 1-PT-52.3, “Unit 1 P-250 Computer and Unit 1 PCS Sump Inleakage Verification;”
- 2-PT-62.2.1, “Recirculation Spray Heat Exchanger Service Water (RSHX SW) Inleakage;”
- 2-PT-63.1A, “Quench Spray System - “A” Subsystem;”
- 2-PT-213.5H, “Valve Inservice Inspection;”
- 2-PT-57.1A, “Emergency Core Cooling Subsystem-Low Head Safety Injection Pump (2-SI-P-A);” and,
- 2-PT-213.8A, “Valve Inservice Inspection (“A” Train of Safety Injection System)”.
- North Anna Power Station Standing Order Number 235, “Monitoring Plan for Increased RCS Leakage.”

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed the details of temporary modification (TM)-1717 to determine whether system operability/availability was affected; configuration control was maintained; and the associated safety evaluation was justifiably implemented. The purpose of the TM was to provide additional backup to Unit 1 station Battery 1-II. An evaluation determined that the battery remained operable with one cell out of service but would become inoperable if another failed. A review of the licensee’s corrective action program by the inspectors revealed two plant issues associated with the TM. Plant issue N-2003-0165 involved the tipping of the cart designed to transport the spare battery jar in the 1-II battery room and the stagging of the battery jar in the 1-II battery room without the availability of the TM paperwork in the control room (N-2003-0172). The TM was determined to be implemented in accordance with the requirements of the administrative procedure VPAP 1403, “Temporary Modifications.”

The inspectors also reviewed the details of TM-1719. The purpose of the TM was to provide a source of water from the emergency condensate storage tank (ECST) for the hydraulic cutters that were used to demolish a portion of the Unit 1 containment structure for the reactor head replacement. The inspectors reviewed applicable work orders, drawings, procedures, and held discussions with licensee personnel. A walkdown of the area where the spool piece had been reinstalled after completion of the hydraulic cutting was performed by the inspector.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation

a. Inspection Scope

On February 18, 2003, the inspectors reviewed the scenerio and observed the performance of the licensee's training drill in the Technical Support Center. The inspectors also attended the post drill critique, held discussions with the emergency preparedness coordinators, and reviewed EPIP-1.01, "Emergency Action Level Table (TAB-J) Security Event." The inspectors observed the use of effective three part communications between the station emergency director and a security representative.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

a. Inspection Scope

The inspectors performed a periodic review of the Unit 1 and 2 PI data reported to the NRC for the following performance indicators:

- High Pressure Safety Injection System Unavailability (Mitigating Systems), and
- Auxiliary Feedwater System Unavailability (Mitigating Systems).

The inspectors reviewed data for the second quarter 2002 to fourth quarter 2002 from the system engineer's unavailability database and from operating logs. The data was compared with that displayed on the NRC's public web site. The PI method of calculation was compared with the guidelines contained in NEI 99-02, "Regulatory Assessment Performance Indicator Guideline."

During plant tours the inspectors also periodically assessed the Occupational Exposure Control Effectiveness and the RETS/ODCM Radiological Effluent Occurrence PIs by determining if high radiation areas (>1R/hr) were properly secured and looking for unmonitored radiation release pathways.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

a. Inspection Scope

The inspectors performed an in-depth review of Unit 2 Charging Flow Control Valve 2-CH-FCV-2122 deficiencies which were associated with valve packing leakage (an emergency core cooling system leakage concern), low flow instability, and valve binding problems. This inspection was to ensure that appropriate root cause evaluations and corrective actions were specified. The inspectors assessed the licensee's actions against the requirements of 10 CFR 50, Appendix B. The following documents were reviewed by the inspectors:

- Root Cause Evaluation Response N-2003-0482-E1;
- Plant Issue Communication N-2003-0482-C1;
- Plant Issues N-2003-0482, 0488, 0547, 0578, 0591 and 0614;
- Work Order 00485082; and,
- Plant Drawing 12050-FM-095C, Chemical and Volume Control System, sheet 1 of 2.

b. Findings and Observations

The licensee determined that the root cause of the valve packing leakage problem was that the design of the stuffing box does not allow packing adjustments to effectively maintain a zero leakage acceptance criteria. The valve binding problem was due to foreign materials caught in the cage assemblies as a result of the manufacturing process. The low flow instability of the valves was corrected through the implementation of a design change. Corrective action for the valve packing leakage problem involved a design change to allow packing adjustments and the valve binding problem required a procedural change to require the inspection of the cage assemblies for foreign material as well as maintaining vendor recommended clearances. The inspectors verified that the root causes and corrective actions taken or planned to correct the conditions were appropriate and timely. No violations or regulatory requirements or findings were identified.

4OA3 Event Followup

.1 (Closed) LER 50-338, 339/2002-001-00: Manual Reactor Trip Due To Control Bank "B" Group 2 Step Counter Inoperable. The licensee noted that the cause of the problem was a defect in the down counter gear during shutdown of the plant for a planned refueling outage. The overall problem was attributed to an equipment error/defect. When the condition was discovered, the licensee immediately declared the step counter inoperable and the action requirements of Technical Specification (TS) 3.1.3.3 were entered. As per TS requirements, the Unit 2 reactor trip breakers were opened within 15 minutes of the discovery of the problem. The long term corrective action involved replacement of the step counter and relay driver during the outage. This item is in the licensee's corrective action program as Plant Issue N-2001-0569.

.2 <u>(Closed) LER 50-338, 339/2001-001-00</u>: Ventilation Flow Outside Technical Specification Limits Due To Inadequate Test. The inspectors reviewed the licensee's corrective actions associated with the failure of the ventilation flow to meet Technical Specifications (TS) 4.7.7.1.b.3 requirements. The licensee noted that the cause of the problem was inadequate testing techniques and the use of flawed testing methodology and test sample locations. They attributed the overall problem to testing ventilation flows at locations of the ventilation system that were not truly representative of the full ventilation system designed flow rates. It was also noted that the ventilation system fans were not adjusted properly and in accordance with the system design requirements. A subsequent walkdown of the ventilation system was performed to ensure that the proper sample points were appropriately marked. The inspectors confirmed that the corrective actions were taken on both Unit 1 and Unit 2. This item is in the licensee's corrective action program as Plant Issue N-2001-0170.

.3 <u>(Closed) LER 50-339/2002-001-00</u>: Reactor Vessel Head Leakage due to Hot Short Cracking and Primary Water Stress Corrosion Cracking: On September 14, 2002, during Unit 2 refueling outage, the licensee identified through-wall leakage on two reactor vessel head penetrations based on the presence of boric acid deposits on the head at these locations. Additionally, four more penetrations were later determined to be leaking. Technical Specification 3.4.6.2 prohibits Reactor Coolant System pressure boundary leakage in Modes 1 through 4. The licensee reasonably assumed that the leakage occurred while at power. The licensee determined that the apparent cause of the event was hot-short cracking, which originated during the fabrication of the reactor vessel head, that was accelerated by primary water stress corrosion cracking. The corrective action was to replace reactor vessel head. The finding was dispositioned in Inspection Report No. 50-339/2002-04 as a non-cited violation.

.4 <u>Replacement of North Anna Unit 1 Reactor Pressure Vessel Head (RPVH) (71007)</u>

.4.1 <u>Containment Restoration Activities</u>

a. <u>Inspection Scope</u>

The inspectors reviewed containment (CNTMT) restoration activities associated with the temporary construction opening, which was approximately 10 feet by 20 feet in the containment liner and 18 feet by 28 feet at the face of the concrete wall, as detailed in the licensee's Design Change Package (DCP) 02-025, "Restoration of Temporary Construction Opening in the Containment Structure for Reactor Pressure Vessel Head Replacement / North Anna Power Station / Unit 1."

Activities associated with CNTMT liner plate welding were reviewed and compared with the ASME Boiler and Pressure Vessel Code (B&PV), Sections III and VIII, 1968 Edition with Addenda through Summer 1969, and welding controls detailed in Bechtel Power Special Processes Manual (SPM). The inspectors reviewed controls for the full penetration liner plate welds and the associated leak chase welds. For the liner plate weld, the inspectors visually inspected the final weld surfaces, observed in-process welding and inspection activities (visual (VT) and magnetic particle (MT)) for weld repair of defects identified by radiographic (RT) examination, and reviewed original and repair radiographic (RT) film. For the leak chase welds, the inspectors observed in-process welding activities for welds A-FW-9 through A-FW-12, L-FW-14 through L-FW-20, N-FW-2, N-FW-3, A-FW-3, and A-FW-4. Also, VT and MT inspection of leak chase welds A-FW-1, A-FW-2 and A-FW-13 through A-FW-16 was witnessed. In addition to observation of in-process work, the inspections included: review of welding procedure (including supporting procedure qualification records), review of welder qualification records, review of welding material testing and certification records, observation of welding material issue and use control, review of in-process weld records, review of Quality Control (QC) involvement in the welding process, review of MT examination records for the completed liner plate weld, observation of preliminary pressure test of 'leak chase" channels after re-welding containment liner, and review of QC and nondestructive examination (NDE) personnel qualification and certification records.

The inspectors reviewed activities associated with installation of CNTMT reinforced concrete and compared activities with the applicable Code, ACI 318-63, Part IV-B, Building Code Requirements for Reinforced Concrete Institute, 1963. Cadwelding splicing activities were reviewed and compared with the following applicable requirements: Bechtel specifications for procurement and installation, equivalent to NAPS (North Anna Power Station) specifications used during original construction; the ASME B&PV Code, Section III, Division 2, 1995 Edition with 1996 Addenda, the applicable Code for splice system qualification tests; Cadweld operator qualification consistent with ASME Section III, Subsection CC-4333.4; and AWS D1.4-98, the applicable Code for welded splices.

The inspectors observed in-process Cadwelding for splices 1V4T, 3H8L, 3H8R, and 'sister splice' T1V18B; observed QC inspections, including in-process and final acceptance, of Cadwelding activities, including inspection of 'sister splice' T1V4B; reviewed in-process records for splices 1V6T, 16T, 17T, 5B, 4B, 18B, 5T, and 19B; and visually inspected completed cadwelds 1V17B, 18B, and 19B.

b. Findings

No findings of significance were identified.

.4.2 Reactor Pressure Vessel Closure Head Activities

a. Inspection Scope

The inspectors observed the installation activities performed by Framatome ANP, Lynchburg, USA, and Juenot, France, related to the RPVH such as: installation of the control rod drive mechanisms (CRDMs), plugs, and core exit thermocouple nozzle assemblies (CETNAs); qualification of the automatic welding machine that is used to weld the CRDMs to the flange; and automatic welding operations. In addition, the inspectors observed NDE on competed welds and repaired welds, and review relevant records on material certifications and personnel qualifications. The inspectors observed the following specific RPVH activities:

- transportation and installation of CRDMs,
- fit up measurements,
- qualification of automatic welding machine (ESAB 6) for canopy seal welding,
- liquid penetrant (PT) inspection of completed canopy seal welds, repair weld, and qualification canopy seal weld coupon used to qualify the ESAB 6,
- seal welding of plug to the CRDM nozzle identified as H4 (penetration 20),
- seal welding of CRDMs E13 (penetration 53), C9 (penetration 40), M6 (penetration 28, and G7 (penetration 4), and
- manual welding to repair weld flaws.

In addition, the inspectors reviewed quality records as detailed in List of Documents Reviewed, Attachment 1, to verify that work was being accomplished and documented in accordance with requirements.

b. Findings

No findings of significance were identified.

.4.3 Quality Assurance (QA) Oversight

a. Inspection Scope

The inspectors reviewed licensee procedures relative to Quality Assurance oversight of contractor activities for the RPVH replacement as detailed in Dominion Procedure NOD-GL-4, "Dominion Reactor Head Replacement VHR Projects Nuclear Oversight Quality Plan. In addition, the inspectors observed in-process QA oversight activities for containment restoration and CRDM installation. The inspectors also reviewed a sample of Activity Reports, Management Summary Reports, Dominion Vendor/Subcontractor Surveillance Reports, Bechtel Quality Surveillance Reports, and Framatome Quality Control Surveillance Reports, all documenting QA observations and findings, to ensure that adequate oversight was being applied.

b. Findings

No findings of significance were identified.

.4.4 Review of Dominion's 10 CFR 50.59 Evaluations for the Replacement RPVH

a. Inspection Scope

The inspectors reviewed the following replacement RPVH DCPs and associated 10 CFR 50.59 evaluation:

- DCP 02-014, Reactor Vessel Head Replacement/NAPS/Unit 1;
- DCP 02-026, Reactor Vessel Head Replacement/Structure Services;
- Modifications/NAPS/Unit 1; and,

- DCP 02-025, Restoration of Temporary Construction Opening in the Containment Structure for Reactor Pressure Vessel Head Replacement/North Anna Power Station/Unit 1.

The DCPs were reviewed to verify that changes between the original RPVH and the replacement RPVH, and modifications resulting from the installation of the replacement RPVH, were properly evaluated in accordance with 10 CFR 50.59.

b. Findings

No findings of significance were identified.

.4.5 Review of RPVH Replacement Lifting and Transportation Program Activities

a. Inspection Scope (71007)

The inspectors reviewed the licensee's heavy load lifting and transportation program to ensure that it met the Updated Final Safety Analysis Report (UFSAR) and regulatory requirements for application to the Unit 1 reactor head replacement. The inspectors walked down the containment to review the polar crane condition and the head runway installation as part of preparation to remove the old head out and lift the new head into containment. The inspectors also walked down the outside at the containment wall opening area to review the outside runway and its supporting structure and the 300 Ton Manitowoc-M250 Mobile Crane condition. The inspectors observed the old RPVH lifted above the refueling floor from storage area, set on the cart, pushed out to the outside platform from the containment using the runway installed, and lifted on the top of a transporter.

The inspectors reviewed the procedures, polar crane maintenance and inspection records, crane operator qualification records, Work Plan and Inspection Record (WPIR), drawings for head lifting steps, runways, and supporting structures, crane lifting capacity, mobile crane M-250 inspection and certificate, mobile crane stability evaluation, mobile crane load drop analysis, the structural calculation for the platform and supporting structure, and design change packages.

b. Findings

No findings of significance were identified.

4OA5 Other

Reactor Vessel Head Replacement Radiation Protection Inspection

a. Inspection Scope

Various aspects of the licencee's radiation protection program controls, planning, preparation, and implementation for Unit 1 and Unit 2 reactor pressure vessel head replacement activities were reviewed and evaluated. Specifically, the inspectors reviewed and evaluated as low as is reasonably achievable (ALARA) planning; dose estimates and dose tracking; exposure controls including temporary shielding; contamination and airborne radioactivity controls; radioactive material management; radiological work plans and controls; emergency contingencies; and project staffing and training plans.

ALARA planning packages for both reactor head replacements were reviewed. The radiation, contamination, and airborne radioactivity surveys in the packages were reviewed for radiological work conditions and the adequacy of prescribed postings and surveys. The inspectors reviewed the radiation work permits in the packages to determine projected exposure, expected conditions, electronic dosimeter dose and dose rate alarm settings, dosimetry requirements, protective clothing/equipment, worker instructions and radiation protection (RP) technician instructions. Revisions to ALARA exposure estimates were reviewed and evaluated against changing work scope/radiological conditions. The ALARA packages from the Unit 2 head replacement were contrasted with those from the Unit 1 head replacement to determine if lessons learned had been implemented, and the lessons learned write-ups were evaluated for content. In addition, the inspectors reviewed internal dosimetry assessments for adequacy of respiratory protection and engineering controls. Corrective action documentation was reviewed for significant trends or recurring problems with work practices and controls. The source terms and resulting doses from the two head replacements were compared by the inspectors and used as a basis for assessing of the ALARA planning. The inspectors reviewed the temporary shielding program and its implementation during the outage.

The inspectors interviewed the RP project leads for both day and night shifts to identify contingencies, problems, and changes in work scope that were incurred during the reactor head replacements. These interviews included reviewing work scope documentation, transportation and burial site documentation, and contingency plans for each step in the relocation of the heads from the reactor vessel to the burial site.

Project staffing and training issues were discussed with the Radiation Protection Manager (RPM) and his staff health physicists.

RP program activities and their implementation were evaluated against 10 CFR 19.12; 10 CFR 20, Subparts B, C, F, G, H, and J; 10 CFR 61; 10 CFR 71; 49 CFR 172; 49 CFR 173 and approved licensee procedures. Licensee guidance documents, records, and data reviewed within this inspection area are listed in List of Document Reviewed.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

.1 Exit Meeting Summary

On April 15, 2003, the resident inspectors presented the inspection results to Mr. J. Davis and other members of his staff who acknowledged the findings. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

.2 Annual Assessment Meeting Summary

On March 20, 2003, the NRC's Chief of Reactor Project's Branch 5 and the Senior Resident Inspector assigned to the North Anna Power Station (NAPS) met with Virginia Electric and Power Company to discuss the NRC's Reactor Oversight Process (ROP) and the NAPS annual assessment of safety performance for the period of January 1, 2002 - December 31, 2002. The major topics addressed were: the NRC's assessment program, the results of the NAPS assessment, and NRC security activities. Attendees included NAPS site management, members of site staff, corporate management and staff, and members of the local news media.

This meeting was open to the public. The presentation material used for the discussion is available from the NRC's document system (ADAMS) as accession number ML030980555. ADAMS is accessible from the NRC Web site at *http://www.nrc.gov/reading-rm/adams.html* (the Public Electronic Reading Room).

4OA7 Licensee-Identified Violations

The following violations are of very low safety significance (Green) and were identified by the licensee. These issues are violations of NRC requirements which met the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for disposition as non-cited violations:

.1 Technical Specification 4.7.7.1.b.3 requires a flow rate of 1000 +/- 10% cubic feet per minute (cfm) from each control room emergency ventilation system supply fan. Contrary to the above, on January 24, 2001, the measured flow from the 2-HV-F-42 fan was 834 cfm. The inspectors verified that the licensee implemented the corrective actions specified in the LER and supporting documents (see Section 4OA3.2). This item is in the licensee's corrective action program as Plant Issue N-2001-0170.

.2 The Code of Federal Regulations 10 CFR 50.59 requires, in part, that an evaluation be performed to determine whether a change to the facility requires a license amendment. On April 1, 2003, the licensee identified a failure to perform a 10 CFR 50.59 evaluation for an Updated Final Safety Analysis Report (UFSAR) change related to the description of preparation for refueling. Section 9.1.4.5 of the UFSAR was changed via UFSAR change FN 92-164 without the performance of an evaluation. The UFSAR originally stated that the reactor is shutdown and cooled to cold shutdown conditions with a final Keff less than or equal to 0.95 (all rods in) and a boron concentration of between 2300 and 2400 ppm. The UFSAR change replaced the original description with the statement that the reactor is shutdown and cooled to cold shutdown conditions in accordance with Technical Specifications (i.e., Keff less than 0.99 and temperature less than or equal to 200 degrees F). Because the Technical Specifications definition of cold shutdown is less conservative than the one replaced in the UFSAR, the change needs to be technically justified. A Safety Review/Regulatory Screen is required to technically justify the changes made to Section 9.1.4.5 via FN 92-164. This issue is documented in Plant Issue N-2003-1460.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

D. Christian, Senior Vice President and Chief Nuclear Officer
K. Barnette, Supervisor, Site Industrial Safety/Fire Protection
J. Breeden, Supervisor Transportation
L. Carter, Project Nuclear Oversight Lead
J. Crossman, Supervisor, Nuclear Engineering
L. Lane, Director, Station Nuclear Safety and Licensing
E. Dryier, Supervisor Dosimetry
M. Dunston, Manager, Site Services
T. Fredette, Electrical Engineering (Emergency Diesel Generator) Systems Engineer
J. Davis, Director, Station Operations and Maintenance
D. Heacock, Site Vice President
E. Hendrixson, Manager, Station Engineering
P. Kemp, Manager, Operations
M. Lane, ALARA Supervisor
J. Leberstien, Supervisor Licensing
T. Maddy, Manager, Station Security
G. Modzelewski, Project Engineer - VHP Inspection
F. Mladen, Manager, Maintenance
N. Nichols, Staff Health Physicist
Q. Parker, Maintenance Rule Coordinator
D. Pleman, Alternate RP Project Lead North Anna Head Replacement
D. Price, Head Replacment Project Manager
P. Quarles, Nuclear Oversight
W. Renz, Director, Security and Emergency Preparedness
H. Royal, Manager, Nuclear Training
C Simms, RP Project Lead North Anna and Surrey Nuclear Plant Head Replacements
M. Sartain, Engineering Project Lead
A. Stafford, Manager, Radiological Protection
M. Whalen, Supervisor Licensing
L. Yates, Nuclear Oversight Team

Contractors

W. Bryant, Framatome Site Manager
S. Bateman, Framatome Quality Assurance
B. Hawkins, Framatome Task Lead
M. Pierce, Quality Assurance
D. Miller, Bechtel Project Manager

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

50-339/2001-002-00	LER	Manual Reactor Trip Due To Control Bank "B" Group 2 Step Counter Inoperable (Section 4OA3.1)
50-338, 339/2001-001-00	LER	Ventilation Flow Outside Technical Specification Limits Due To Inadequate Test (Section 4OA3.2)
50-339/2002-001-00	LER	Reactor Vessel Head Leakage due to Hot Short Cracking and Primary Water Stress Corrosion Cracking (Section 4OA3.3)
50-338/2003-02-01	NCV	Failure to Correctly Install Hydraulic Snubber Pipe Support 1-SI-HSS-107 (Section 1R08)

LIST OF DOCUMENTS REVIEWED

1R07 Heat Sink Performance

Procedures and Standards

- Administrative Procedure VPAP-0811, Rev. 3, Service Water System Inspection and Maintenance Program
- Administrative Procedure VPAP-2201, Rev 12, Nuclear Plant Chemistry Program Standard No. STD-CEN-0021, Rev. 5, Civil Engineering Nuclear Standard - Welded Pipe Attachments

Documents Reviewed

- Work Orders 00327116, 00382431, and 00436294 Perform Boroscope Inspections on Heat Exchanger 1-RS-S-1C
- Letter Serial No. 89-572E, Virginia Electric Power Company to NRC for Supplemental Response to Generic Letter 89-13 Service Water System Problems affecting
- Work Orders 00454097, 00436878, 00448956, 00424906, and 00408457 for Maintenance on Valve 1-SW-MOV-104D

1R08 Inservice Inspection (IS)

Procedures

- Brook Associates, Inc. Field Services Procedures 830994, Revision 1, Reactor Vessel Head Penetration Remote Visual Inspection

- Westinghouse Field Services Procedures MRS-SSP-1353, Revision 1, North Anna Unit 2 Reactor Vessel Head Inspection Tool Operation
- Westinghouse Field Services Procedure WDI-UT-010, Revision 3, Intraspect Ultrasonic procedure for Inspection of Reactor Vessel Head Penetrations, Time of Flight Ultrasonic Longitudinal Wave and Shear Wave
- Administrative Procedure VPAP-0801, Rev. 11, Maintenance Program
- Administrative Procedure VPAP-1103, Rev.7, ASME Section XI Visual Examination Program (VT-1, 2, 3 & General)
- Nondestructive Examination Procedure NDE-UT-803, Rev. 0, Single or Two Side Access Ultrasonic Examination of Austenitic Piping Welds
- Nondestructive Examination Procedure NDE-PT-703, Rev. 3, Liquid Penetrant Examination for Temperatures Less Than 60 Degrees F
- Procedure 0-MPM-0901-01, Rev. 006, Removal, Installation, and As Found Functional Testing of ITT Grinnell Snubbers Using Wyle Test Machine
- Periodic Test Procedure 1-PT-79.7, Rev. 14, Visual inspection of Unit 1 Hydraulic Snubbers
- Periodic Test Procedure 1-PT-52.2A, Rev. 25, Reactor Coolant System Leak Rate (Computer Calculation)
- Periodic Test Procedure 1-PT-48.2, Rev. 002, Visual Inspection of ASME XI Class 2 and 3 Pressure Boundary Components Outside Reactor Containment
- Periodic Test Procedure 1-PT-48.1, Rev. 002,Visual Inspection of ASME XI Class 2 Pressure Boundary Components Inside Reactor Containment
- Administrative Procedure DNAP-1004, Rev.0, Boric Acid Corrosion Control (BACC) Program (Effective Date June 5, 2003)
- Periodic Test Procedure 1-PT-48, Rev. 011, Visual Inspection of Reactor Coolant Pressure Boundary Components
- General Operating Procedure 1-GOP-4.22, Rev. 2, Monitoring Containment Heat Exchanger Performance to Identify RCS Leakage

Plant Issues

- N-2002-2287, CRDM Penetration 21 appears to have exhibited some amount of boric acid leakage during last cycle
- N-2002-2288, CRDM Penetration 31 appears to have exhibited some amount of boric acid leakage during last cycle
- N-2002-2319, CRDM Penetration 57 appears to have exhibited some amount of boric acid leakage during last cycle

Vendor Exam Evaluation Reports (VEs)

- North Anna Unit 2 Reactor Vessel Head Penetration September 2002 J-Groove Weld and Tube OD Surface Eddy Current Inspection Results
- North Anna Unit 2 Reactor Vessel Head Penetration September 2002 Open Housing Scanner and Gap Scanner Inspection.

Other Documents

- North Anna Liquid Penetrant (PT) Examination Report 6622, CRDM 51, 62, 63, Nov 2001
- North Anna PT Examination Report 6676, CRDM 51 Weld Repair (final) Nov. 2001
- North Anna PT Examination Report 6688, CRDM 63 Weld Repair (intermediate) Nov. 2001
- Work Orders 00485894 Remove Snubber and Clamp From Support 1-SI-HSS-701 For NDE
- Ultrasonic Examination Data Record on March 12, 2003 for Weld Mark 11, Drawing 11715-WMKS-0104A-1, Line 6"-SI-16-1502-Q1
- Liquid Penetrant Report on March 12, 2003 for Weld Mark 11, Drawing 11715-WMKS-0104A-1, Line 6"-SI-16-1502-Q1
- Liquid Penetrant Report on March 14, 2003 for Weld Mark 92H, Drawing 11715-WMKS-0104A-2, Line 10"-SI-239-153A-Q2
- VT-3 Visual Examination Report IWF Component on March 12, 2003 For Legs 1 -4, Drawing 11715-WMKS-SI-TK-2, Line 1-SI-TK-2
- Drawing 11715-PSSK-104A.10, Rev. 2 for Pipe Support 1-SI-HSS-107
- Drawing 11715-PSSK-104A.08, Sheets 1 -3, for Pipe Support 1-SI-HSS-701
- Drawing 11715-FM-096A, Rev. 39, Flow/Valve Operating Diagram Safety Injection System
- Periodic Test Procedure 1-PT-79.7,Rev. 14, Visual inspection of Unit 1 Hydraulic Snubbers Performed dated Feb. 23 - March 10, 2003
- North Anna Power Station Standing Order No. 235, Rev. 1
- Ongoing Activities in Response to Bulletin 2002-01
- Periodic Test Procedure 1-PT-46.21, Rev. 012, RCS Pressure Boundary Components Affected by Boric Acid Accumulation Performed on Feb. 23, 2003
- Work Order 00486228 to Repack Valve
- Work Request Tag # 141028
- Work Request Tag # 141025
- Work Request #00085112
- Work Order 00486023 to Repack Valve
- Self Assessment Plan # CEN-02-01, Generic Letter 88-05 Commitment Effectiveness, Dated June 19, 2002
- Response Serial Nos.02-689, 02-168, 02-168A to NRC Bulletin 2002-01, Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity

2OS1 Access Control to Radiologically Significant Areas (71121.01)

Procedures, Instructions, Lesson Plans and Manuals

- Virginia Power Administrative Procedure (VPAP)-2101, Radiation Protection Program, Revision (Rev.) 21
- VPAP-1601, Corrective Action, Rev. 15
- VPAP-1501, Deviations, Rev. 14

- North Anna Power Station Independent Spent Fuel Storage Installation - Safety Analysis Report, 6/30/98

RWPs

- 02-2-2008, Health Physics surveys in Very High Radiation Areas, Rev. 0
- 02-2-2013, Minor Work by Mechanical, Electrical, and Instrumentation Personnel, Rev. 0
- 02-2-2023, Disassemble, Inspect, Repair, Repack, Cut Out and Replace Valves and Flanges in support of the Unit-2 Outage, Rev. 0

Records and Data

- C-HP-1041.023, Attachment 3, Committed Effective Does Equivalent (CEDE)-Ingestion Intake Data, Revision 0, for evaluation dated 12/10/01
- Radiological Survey Map for Unit 2 Containment, 262' Elevation, B Cubical, 9/10/02
- Radiological Survey Map for Unit 2 Containment, 241' Elevation, A Loop Room, 9/9/02

Audits and Self-Assessments

- Audit 01-07: Radiological Protection/Chemistry
- C-HP-1091.232, Attachment 1, Radiological Survey Program Evaluation, Rev 1, Covering period 5/99 to 2/02

Plant Issues

- N-2001-3301, Individual entered a posted High Radiation Area and subsequently a Locked High Radiation Area in U-2 Rector Containment 216' elevation without a digital alarming dosimeter, 11/9/001
- N-2002-2640, Individual entered the protected area with the wrong TLD, 9/30/02

2OS2 As Low As Reasonably Achievable (71121.02)

Procedures, Instruction, Lesson Plans and Manuals

- VPAP-2101, Radiation Protection Program, Rev. 21
- VPAP-2101, Station ALARA Program, Rev. 8
- Source Term Reduction Initiatives at North Anna Power Station, dated 7/31/02

RWPs

- 02-2-2001, Disassemble and reassemble Reactor Vessel Head, Rev. 0
- 02-2-2050, Rector Head penetration eddy current inspection, Rev. 2
- 02-2-2065, Repair 20RC-MOV-2593, Rev2

Records and Worksheets

- ALARA Evaluation & RWP Man-Rem Data as of 9/30/02
- Pre-job ALARA Worksheets and ALARA Action Plans for ALARA Evaluations Numbers:
- 02-008 Install & Remove Scaffolding
- 02-016 Disassemble/Reassemble Reactor Head
- 2-034 Rector Head Inspections
- Quarterly Hot Spot Program Report - Third Quarter 2002
- List of Source Term Removed from 1/01/97 to 9/30/02

Audits and Self-Assessments

- Audit 01-07: Radiological Protection/Chemistry
- C-HP-1091.231, Attachment 1, Exposure Control Program Evaluation, Rev. 0, Covering Period 1/1/99 to 12/31/00

Plant Issues

- N-2002-2624, An individual alarmed the radiation monitor at the main security exit point after having cleared the radiation monitors at the RCS exit point, 9/29/02
- N-2002-2297, Unnecessary radiation exposure to scaffold crew, approximately 120 mrem total for the crew. Scaffolding was erected and disassembled for a job which had been completed and did not require the sue of scaffolding, 9/15/02

2PS2 Radiation Material Processing and Transportation (71122.02)

Procedures, Instruction, Lesson Plans and Manuals

- North Anna Power Station - Updated Final Safety Analysis Report (UFSAR), Sections 11.2 and 11.3, Rev. 36
- VPAP-1601, Corrective Action, Rev. 15
- HPTCTP-5-LP-9, Radioactive Material/Waste Shipping- Waste Classification, Rev. 3 7/27/1998

Records

- Memo: C.A. Tarantino to A.H. Stafford, Third Quarter 2001, Confirmatory Measurements, 10/31/2001
- Results of Radiochemistry Cross Check Program Dominion Generation Framtome ANP DE&S, 1st Quarter 2002

Audits and Self-Assessments

- NUPIC Audit of Teledyne Brown Engineering Environmental Services, 3/12/2001
- Vendor Corrective Action Request to Teledyne Brown Engineering-ES, 2/2/2001

- C-HP-1091.271, Attachment 1, Radioactive Material Control Program Evaluation Revision 0, covering period 4th Quarter 1997 to 4th Quarter 2000

Plant Issues

- N-2002-1963, Trailer delivering cask for resin shipment (02-CNS-04) was identified as having 2 superficial cracks in a rear cover plate, 8/14/2002
- N-2002-0113, Differences among Millstone, North Anna and Surry purchase orders for shipping casks, 1/16/2002
- N-2002-2072, ASP-1 (neutron radiation detector) #3695 failed the return performance check 8/28/2002

4OA1 Performance Indicator Verification (71151)

Procedures

- HPAP-2802, NRC Performance Indicator Program, Rev. 1
- 0-PT-452.01, Radioactive Liquid Effluents, Dose Calculations, Rev 5 (reviewed results of monthly procedural performance for July 2001 to July 2002)
- 0-PT-454.01, Radioactive Gaseous Effluents, Dose Calculations, Rev. 5 (reviewed results of monthly procedural performance for July 2001 to July 2002)

Plant Issues

- N-2001-2989, A Dominion Employee’s CEDE of 31 mrem on his whole body count exceeds the TEDE-ALARA evaluation’s internal dose estimate of 5.85 mrem, 10/1/2001
- N-2001-3301, Individual entered a posted High Radiation Area and subsequently a Lock High Radiation Area in U-2 Reactor Containment 216' elevation without a digital alarming dosimeter, 11/9/2001
- N-2002-2072, ASP-1(neutron radiation detector) #3695 failed the return performance check 8/28/2002

4OA3 Event Followup

Replacement of North Anna Unit 1 RPVH (71007)

Procedures

- 71007 Reactor Vessel Head Replacement Inspection
- Mechanical Maintenance Procedure 0-MCM-1303-02, Rev. 0, Moving Miscellaneous Heavy Load and Concrete Floor Plugs in Containment During Unit Outage with No Fuel in the Reactor Vessel

Documents Reviewed

- P1-REBAR,Welding Procedure Specification, Revision 2

- P1AT-Lh(CVN-20°F), Welding Procedure Specification, Revision 1
- Procedure CP-C-2, CADWeld Rebar Splices, Revision 0
- Repair/Replacement Plan, Program Number 2003-009, Remove and Replace Containment Liner
- Bechtel Nondestructive Examination Standard RT-ASME IIICL B S69, Radiographic Examination, Revision 1
- Dominion Nuclear Oversight Department Guideline NOD-GL-4, Dominion Reactor Vessel Head Replacement Projects Nuclear Oversight Quality Plan, Revision 1
- Bechtel Welding Specification WFMC-1, Welding Filler Material Control, Revision 1
- Vendor Procedure 'Form - 84', Welding, Heat Treatment and Nondestructive Examination Requirements, Revision 3
- Bechtel Special Processes Manual For North Anna 1&2 Nuclear Power Station RPV Head Replacement Project, Revision 4
- Bechtel Procedure WQ-1, Welder Performance Qualification Specification (ASME Section IX), Revision 18
- Bechtel Procedure B-GWS-1, General Welding Standard, Revision 10
- Bechtel Procedure GWS-Structural, General Welding Standard, Revision 6
- Bechtel Procedure GWS-REBAR, General Welding Standard Arc Welding of Reinforcement Steel, Revision 5
- DCP 02-014, Reactor Vessel Head Replacement/ NAPS/ Unit 1
- DCP 02-026, Reactor Vessel Head Replacement/ Structure Services Modifications/ NAPS/ Unit 1
- Design Change No. 02-025, Restoration of Temporary Construction opening in the Containment Structure for Reactor Pressure Vessel Head Replacement / North Anna Power Station / Unit 1
- RT Film and Reader Sheets for Segments 17-17.5-18 (original and repair), 46.5-47, 36-36.5, 31-31.5, 6.5-7, 13-13.5 (Expansion), 16-16.5 (Expansion)
- Cadwelder Qualification Records for Cadwelders 062, 109, 124, 125, 135, 138 and 145
- Certification of Calibration for Machine Used for Testing Cadweld Qualification Specimens
- In-process North Anna Periodic Test Procedure 0-PT-61.1M, Leak Testing Individual Containment Liner Locations, Revision 1-OT01for preliminary leak chase pressure testing
- VT-2 Certification for North Anna Power Station Level II Examiner for leak chase pressure testing
- Welder Qualification Test Records for Welder Symbols BM-10, BM-12, BM-13, BM-14, BM-15, and BM-16
- In-process Records (Field Weld Checklists) for Leak Chase Channel Weld Records for Welds L-FW-14 through L-FW-20, N-FW-2, N-FW-3, A-FW-3, A-FW-4, and A-FW-9 through A-FW-12 .
- Personnel Qualification and Certification Records for a Sample of Cooperheat MQS NDE Examiners
- Personnel Qualification and Certification Records a Sample of Bechtel QC Inspectors and Examiners
- Sample of Dominion NAPS Unit 1 RVHR Project Oversight Team Activity Reports dated 2/10/03 through 3/19/03

- Sample of Dominion NAPS Unit 1 RVHR Project Nuclear Oversight Team (PNOT) Management Summary Reports dated 02/10/2003 through 03/15/2003
- Sample of Bechtel Quality Surveillance Reports 24841-NI-QSSS-03-001 through 24841-NI-QSS-03-0014
- Sample of Framatome Quality Control Surveillance Reports dated 2/28/03 - 3/11/03
- Dominion RPVHR Project Vendor/Subcontractor Surveillance Reports N1-003, N1-007, N1-008, and N1-014
- Welding Receiving Inspection and Material Certification Records: E7018(Lot 2S210C02), E7018(Lot 4D215A04), and E7018(Lot 2J027A01
- Framatome Material Certifications Records For: "Y" Shaped Consumable Insert (Lot PP460); 1/16" Diameter, ER308L Type Weld Rod; Magnaflux Liquid Penetrant Materials; Tungsten Weld Electrodes; and Stainless Steel Flanges
- Framatome Weld Rod Control Records
- Framatome Qualifications Records for Welders and NDE Examiners
- Quality Control Inspection Records (QCIRs) for 4 CETNAs and 12 adapter plugs received by Framatome, Lyncyburg from Framatome, France
- A Sample of Framatome initiated Nonconformance Reports
- A Sample of Framatome Condition Reports
- Process Travelers for the Installation of CRDMs, Plugs, and CETNAs
- Manufacturing Specification For Thin Edge Welding of Components Onto the Reactor Closure Head
- Adaptors - Automatic TIG Orbital Process
- Traveler - Engineering Verification for Items of Reactor Vessel Closure Head Before and After Seal Welds
- Manufacturing Specification For Repairs to Thin Edge Welding of Components onto the Adaptors of Reactor Closure Head Using the Manual TIG Process, Revision 2
- Procedure For Screwing and Unscrewing the CRDMs Onto/From the Reactor Vessel Closure Head
- Adaptors CRDM Type L106A
- Procedure For CRDM Torquing and Welding - Reactor Vessel Head Replacement
- Procedure 54-ISI-240, Visible Solvent Removable Liquid Penetrant Exam Procedure (NA-03-001 & NA-03-002 incorporated), Revision 41
- Framatome Weld Procedure Specification 55-WP8/8/F6SCA1-05, Manual Gas Tungsten Arc Welding (GTAW), Revision 05
- Framatome Weld Procedure Specification 55-WP8/8/F6SCB3-05, Automatic GTAW, Revision 05
- Design Change No. 02-023, Rigging and transport of reactor Pressure Vessel Heads/ NAPS/Unit 1
- Load Test Certification for Head Suspend Ring, Dated Feb. 21, 2003
- Calculation No. 24841-110-C-010, Rev. 0, Evaluation of Crane Stability During Rigging Activities
- Calculation No. 24841-110-C-002, Rev. 0, Load Drop Evaluation During Assembly of the Access Work
- Structural Calculation No. C-2655-36, Rev. 0, Analysis of Work Platforms & Support Structures for Runway by Rigging International
- Calculation No. C-2655-50, Rev. 0, RPV Head Suspension Frame General Analysis and Interface Loads

- Work Plan and Inspection Record (WPIR) No. R-RIG-22, Rev. 0, Reactor Head Rigging
- Work Plan and Inspection Record (WPIR) No. C-MPT-33, Rev. 0, Preassembly of Platform and Runway
- Work Order 0473319, Perform Refueling Inspection for Polar crane
- 300 Ton Manitowoc - M250 Series 1 Lifting Capacities - Heavy Lift Boom + Fixed Jib
- Manitowoc M-250 Crane Inspection Certificate Performed on Feb. 28, 2003
- Crane Operator Qualifications
- Rigging International Drawing No. 2655-310, Sheets 1 - 5, Rev. 0, Runway & Work Platforms General Arrangement
- Rigging International Drawing No. 2655-315, Sheets 1 - 3, Rev. 0, Runway Assembly
- Rigging International Drawing No. 2655-317, Sheets 1 - 6, Rev. 0, Unit 1 RPVH Head, Suspension Ring, & Cart Assembly
- Rigging International Drawing No. 2655-340, Sheets 1 - 2, Rev. 0, Roll-In and Roll-Out of Rector Heads

4OA5 Reactor Vessel Head Replacement Radiation Protection Inspection

<u>Procedures</u>

- Corporate Health Physics Procedure C-HP-1032.030 Radiation Surveys, Revision (Rev.) 2
- C-HP-1032.040, Contamination Surveys, Rev. 4
- C-HP-1032.050, Airborne Radioactivity Surveys, Rev. 3
- C-HP-1032.060, Radiological Posting and Access Control, Rev. 1
- C-HP-1041.020, DAC Hour Determination Based on Bioassay Results, Rev. 1
- C-HP-1041.021, Radioactive Intake Determination Based On Bioassay Results,
- C-HP-1041.023, Internal Dose Calculation Based on Radionuclide Intake, Rev.1
- North Anna Health Physics Procedure HP-1071.040, Packaging and Shipment of Radioactive Material, Rev. 9
- Temporary Health Physics Procedure T-HP-1071.050 Preparing the Reactor Pressure Vessel
- Head (RVPH) For Shipment, Effective 12/7/2002- 9/1/2003

<u>Records</u>

- ALARA Package 2-041, Disassemble/ Reassemble the Unit 2 (U2) Reactor Vessel Head
- ALARA Package 2-040, Unit 2 Photogrammetry for Support of U2 Head Replacement
- ALARA Package 2-042, Cut Hole in U2 Containment Dome and Build Runway for the Reactor Head Replacement
- ALARA Package 2-043, Install Bechtel Top Hat, Move in Bottom Shield Plate to Head Stand, Load Thermal Sleeves into Baskets on Lower Shield Plate, Move Old Reactor Head to Head Stand 216' and Encapsulate. Includes All Associated Support

- Radiation Work Permit (RWP) 02-02-2074, Unit 2 Containment, Yard Area at Hatch and Fuel Building Crane, Rev. 0
- ALARA Package 03-033, (2 RWPs)
- RWP 03-02-3070, Perform Photogrammetry of Reactor Head and Reactor Vessel Flange, 10 Stud Holes, Guide Studs and Drive Shafts, Rev. 0
- ALARA Package 03-034, Cut Hole in Containment Dome and Build Runway System for the Reactor Head Replacement. Includes Installation of Temporary Jib Crane and All Associated Support, Rev. 0
- ALARA Package 03-043, Move in Bottom Shield Plate to Head Stand On 262', Load Thermal Sleeve Into Basket, Install Bottom Shield Plate Onto Old Reactor Head, Install Bechtel Top hat and Encapsulate.
- RWP 03-2-3074, Move Unit 1 Old Reactor Head From Containment 262' Head Stand and Place on SPMT. Move Reactor Head to Fuel Building Crane Enclosure to Offload and Complete Packaging of WMG Bottom Plate and WMG Top hat Shipping Cover. Load Packaging onto 190' Transporter and Ship Offsite. Includes Moving In New Replacement Reactor Head Into Unit 1 Containment., Rev. 1
- Shipping Papers for Unit 2 Reactor Head
- Correspondence: WMG Project 2118E, 11/20/2002, Characterization and Classification of North Anna Unit 2 Reactor Pressure Vessel Head
- Special Nuclear Material (SNM) Exemption Certification, EC-0230-SNM, Rev. 1,
- Manifest 03-ENV-01, 1/7/03
- Envirocare of Utah, Shipping Checklist, 1/6/2003
- Transportation Plan for North Anna Reactor Pressure Vessel Head, 2118-RE-0004, Rev. 0:
  Appendix A, Site Layout Map
  Appendix B, Transportation Route
  Appendix C, Supplemental Emergency Response Plan
- Correspondence: WMG Project 2118E, 12/12/2002, Activation Basis for Characterization and Classification of the North Anna Unit 2 Reactor Pressure Vessel Head

<u>PIP Documents</u>

- Plant Issue N-2003-0217, A Station Mechanic Experienced Symptoms of Heat Stress While Working on Unit-2 Containment.
- Plant Issue N-2003-1017, A HP Technician Identified Two Workers Inside a High Radiation Area (HRA) with Non-telemetric Dosimetry.
- Plant Issue N-2003-1159, While Performing a Walkdown of U1 Containment HP Technician Discovered a Boundary Rope and Placard Lying on the Floor.
- Plant Issue N-2003-1208, HP Technician Discovered a HRA Boundary Placard Improperly Positioned on the U1 Containment 216' Elevator Ramp Swing Gate.
- Plant Issue N-2003-0942, A Security Officer Entered a Posted HRA and Contaminated Area Without Protective Clothing or Alternate Means of DAD Alarm Recognition.

LIST OF ACRONYMS

ASME	-	American Society of Mechanical Engineers
ALARA	-	As Low As Is Reasonably Achievable
B&PV	-	Boiler and Pressure Vessel
CETNA	-	Core Exit Thermocouple Nozzle Assembly
cfm	-	cubic feet per minute
CFR	-	Code of Federal Regulations
CNTMT	-	Containment
CRDM	-	Control Rod Drive Mechanism
DCP	-	Design Change Package
EDG	-	Emergency Diesel Generator
FRV	-	Feedwater Regulating Valve
HP	-	Health Physics
HRA	-	High Radiation Area
ISI	-	Inservice Inspection
LER	-	Licensee Event Report
MT	-	Magnetic Particle Examination
MR	-	Maintenance Rule
NAPS	-	North Anna Power Station
NCV	-	Non-Cited Violation
NDE	-	Nondestructive Examination
OS	-	Occupational Radiation Safety
OWA	-	Operator Work-around
PARS	-	Publicly Available Records
PI	-	Performance Indicator
PMT	-	Post-maintenance Test
ppm	-	parts per million
PT	-	Liquid Penetrant Examination
QA	-	Quality Assurance
QC	-	Quality Control
RCS	-	Reactor Coolant System
RP	-	Radiation Protection
RPV	-	Reactor Pressure Vessel
RPVH	-	Reactor Pressure Vessel Head
RWP	-	Radiation Work Permit
SDP	-	Significance Determination Process
SSC	-	Systems, Structures, and Components
SW	-	Service Water
TM	-	Temporary Modification
TS	-	Technical Specification
TSC	-	Technical Support Center
UFSAR	-	Updated Final Safety Analysis Report
UT	-	Ultrasonic Examination
VT	-	Visual Examination
VHP	-	Vessel Head Penetration

VPAP	-	Virginia Power Administrative Procedure
WPIR	-	Work Plan and Inspection Record