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

July 29, 2005

EA-05-145

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. 05000338/2005003, 05000339/2005003, 07200016/2005002 AND EXERCISE OF ENFORCEMENT DISCRETION

Dear Mr. Christian:

On June 30, 2005, the United States Nuclear Regulatory Commission (NRC) completed an inspection at your North Anna Power Station, Units 1 and 2, and the North Anna Independent Spent Fuel Storage Installation. The enclosed integrated inspection report documents the inspection findings, which were discussed on June 30, 2005, with Mr. Larry Lane and other members of your staff.

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

Based upon the results of this inspection, no finding of significance was identified. However, a licensee-identified violation which was determined to be of very low safety significance is listed in Section 4OA7 of this report. If you contest any non-cited violation 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, D.C. 20555-0001; with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the North Anna Power Station.

Additionally, based on review of Licensee Event Reports and a followup inspection, the inspectors determined that small amounts of pressure boundary leakage from reactor vessel head penetrations occurred prior to reactor vessel head replacements. Because Technical Specification (TS) 3.4.13 requires that with any reactor coolant pressure boundary leakage, the plant be placed in hot standby within 6 hours, the NRC concluded that violations of the TSs occurred. The violations involved reactor coolant system pressure boundary leakage not avoidable by reasonable quality assurance measures and management controls that were employed by you. Although these issues constitute violations of NRC requirements, we have concluded that your actions did not contribute to the degraded conditions and, thus, no performance deficiencies were identified. Based on these facts, I have been authorized, after consultation with the Director, Office of Enforcement to exercise enforcement discretion in accordance with Section VII.B.6 of the Enforcement Policy and refrain from issuing enforcement action for the violations. An evaluation was performed and we have determined that violations associated with three leaking nozzles on Unit 2, identified in October 2001, involved a risk of low to moderate safety significance. A violation associated with a leaking nozzle on Unit 1, identified in March 2003, involved a risk of very low safety significance. The reactor heads at both North Anna units have been replaced with new heads constructed of material that is less susceptible to cracking and leaking of penetrations. This generic problem is the subject of NRC Bulletins 2001-01 and 2002-02, as well as, NRC Order EA 03-009 and its first revision. NRC actions to generically address this problem, have resulted in new requirements for licensees to effectively examine the reactor vessel head penetrations for flaws on a periodic basis.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response, if any, will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at *http://www.nrc.gov/reading-rm/adams.html* (the Public Electronic Reading Room).

Sincerely,

/RA by Joseph Shea Acting For/

Charles A. Casto, Director
Division of Reactor Projects

Docket Nos.: 50-338, 50-339, 72-016
License Nos.: NPF-4, NPF-7, SNM-2507

Enclosures: Inspection Reports 05000338/2005003, 05000339/2005003 and 07200016/2005002 w/Attachment: Supplemental Information

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-338, 50-339, 72-016
License Nos.:	NPF-4, NPF-7, SNM-2507
Report Nos.:	05000338/2005003, 05000339/2005003, 07200016/2005002
Licensee:	Virginia Electric and Power Company (VEPCO)
Facilities:	North Anna Power Station, Units 1 & 2 North Anna Independent Spent Fuel Storage Installation
Location:	1022 Haley Drive Mineral, Virginia 23117
Dates:	April 1, 2005 - June 30, 2005
Inspectors:	J. Reece, Senior Resident Inspector G. Smith, Acting Senior Resident Inspector G. Wilson, Resident Inspector L. Garner, Project Engineer B. Crowley, RII (Sections 4OA3.1 and 4OA3.2)
Approved by:	K. Landis, Chief, Reactor Projects Branch 5 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000338/2005-003, IR 05000339/2005-003, IR 07200016/2005-002; 04/01/2005 - 06/30/2005; North Anna Power Station Units 1 & 2, and North Anna Independent Spent Fuel Storage Installation. Maintenance Effectiveness.

The report covered a three-month period of inspection by the resident inspectors and a project engineer from the region. One licensee identified Non-cited Violation (NCV) was documented. 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 Finding

Cornerstone: Mitigating Systems

TBD. A self-revealing finding was identified for failure to comply with Technical Specification 5.4.1 requirements to establish, implement, and maintain procedures. Failure to provide sufficient instructions in a maintenance procedure resulted in the faulty retermination of a 480 volt molded-case circuit breaker in December 2004. This subsequently led to an electrical fault and the loss of two safety-related 480 volt motor control centers, 1J1-2N and 1J1-2S, for approximately 7.5 hours on May 1, 2005.

The finding was more than minor because it affected the reactor safety mitigating system cornerstone objective to ensure availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences and the barrier integrity cornerstone objective to provide reasonable assurance that physical design barriers such as containment protects the public from radio nuclide releases caused by accidents or events. This finding is unresolved pending completion of the significance determination assessment and contains a resource issue of the cross-cutting aspects of Human Performance. (Section 1R12)

B. Licensee-Identified Violation

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

REPORT DETAILS

Summary of Plant Status

Unit 1 and Unit 2 began the inspection period at 100 percent power and remained at or near 100 percent power for the entire reporting period except for minor power reductions to perform required periodic testing and a Unit 2 load reduction on June 1, 2005, to ~87% power for tube leak repairs in the ‘B’ condenser waterbox.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection

a. Inspection Scope

The inspectors reviewed the licensee’s adverse weather preparations for hot weather operations specified in 0-GOP-4.1, “Hot Weather Operations,” and the licensee’s correction action data base for hot weather related issues. The inspectors walked down the two risk-significant areas listed below to verify compliance with the procedural requirements and to verify that the specified actions provided the necessary protection for the structures, systems, or components. Documentation reviewed is listed in the attachment.

- Unit 1 & 2 Service Water (SW) pump structure & SW valve house; and,
- Unit 1 & 2 Emergency Diesel Generator Rooms.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

a. Inspection Scope

The inspectors conducted three equipment alignment partial walkdowns to evaluate the operability of selected redundant trains or backup systems, listed below, with the other train or system inoperable or out of service. The inspectors reviewed the functional system 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. Documents reviewed are listed in attachment.

- Unit 2 2J Emergency Diesel Generator (EDG) during the 2H surveillance test;
- Unit 1 and 2 “A” Service Water Header, while “B” Service Water Header out for planned maintenance; and,

- Unit 1 1J EDG while two year overhaul maintenance was being performed for 1H EDG.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

Fire Area Tours

a. Inspection Scope

The inspectors conducted tours of the nine areas listed below and important to reactor safety to verify the licensee's implementation of fire protection requirements as described in Virginia Power Administrative Procedure (VPAP)-2401, "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.

- Emergency Diesel Generator 1H Unit 1 (fire zone 9A-1a/ EDG-1H);
- Emergency Diesel Generator 1J Unit 1 (fire zone 9B-1a/ EDG-1J);
- Fuel Oil Pump Room - Motor Control Center Room (fire zone 10C/ MCC);
- Turbine-Driven Auxiliary Feedwater Pump (TDAFW) Room Unit 2 and Motor-Driven Auxiliary Feedwater Pump (MDAFW) Room Unit 2 (fire zones 14A-2a / TDAFW-2 and 14B-2a / MDAFW-2);
- Charging Pump Cubicle 1-1A and 1-1B (fire zones 11Aa / CPC-1A and 11Ba / CPC-1B);
- Battery Room 1 - I Unit 1 and Battery Room 2 - I Unit 2 (fire zones 7A-1 / BR1-I and 7A-2 / BR2- II);
- Battery Room 1 - III Unit 1 and Battery Room 2 - III Unit 2 (fire zones 7C-1 / BR1-III and 7C-2 / BR2-III);
- Charging Pump Cubicle 1-1C (fire zone 11Ca / CPC-1C); and,
- Charging Pump Cubicle 2-1C (fire zone 11Fa / CPC-2C).

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope

The inspectors reviewed internal flood protection measures at the turbine building/Emergency Switchgear interface area. Flooding in the turbine building could impact risk-significant components in the Emergency Switchgear room if the turbine building flood mitigation features were degraded. Turbine building flood protection features were observed to verify that they were installed and maintained consistently with the plant design basis.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program

a. Inspection Scope

The inspectors observed an annual licensed operator requalification simulator examination on June 14, 2005. The scenario, Simulator Examination Guide SXG-54, involved high Reactor Coolant System (RCS) activity due to a fuel failure, followed by failures of the pressurizer Power Operated Relief Valve (PORV), Pressurizer Level Master and the RCS, resulting in a large break loss of coolant accident.

The scenario required classification and notifications that were counted for NRC performance indicator input. 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 instructors 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 licensee’s effectiveness of the corresponding 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 (MR) (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.” Documents reviewed are listed in attachment.

- Plant Issues N-2004-0232, N-2004-1318, N-2005-1171, and N-2004-4415 associated with 1J EDG MR performance criteria and unavailability performance criteria while observing maintenance activities during 1J EDG 18 months overhaul; and,
- Plant Issue N-2005-1615 associated with an inadequate corrective maintenance procedure which resulted in the loss of 1J1 2N and 2S buses.

b. Findings

Inadequate Maintenance Procedure Resulted in an Electrical Fault Which Deenergized Two Safety-Related Motor Control Centers

Introduction. A self-revealing finding was identified for failure to comply with TS 5.4.1 in that an inadequate procedure resulted in the loss of two safety-related 480V buses.

Description. On May 1, 2005, a control room operator started 1-SW-P-7, 'C' recirculation spray radiation monitor sample pump, in accordance with 0-GOP-4.4, "Equipment Rotation," Rev. 18 for a monthly operational check. Immediately after placing the switch to start, a feeder breaker, 1-EE-BKR-14J-5 opened to deenergize two safety-related 480 volt MCCs, 1J1-2N and 1J1-2S. The operators entered TS Limiting Condition for Operation (LCO) 3.8.9 with an allowed outage time of 8 hours for loss of one train of an engineered safety feature 480 volt electrical bus. The licensee's initial investigation showed that the supply breaker, 1-EE-BKR-1J1-2N-B5, for 1-SW-P-7 had tripped open along with the upstream feeder breaker. Additional visual inspections of all molded-case circuit breakers on 1J1-2N and 1J1-2S were performed with minor arcing damage discovered on the breaker for 1-SW-P-7 between the 'B' and 'C' phase field connection terminations. The internals of this breaker bucket showed discoloration indicative of a possible flashover event. No other problems were detected in either MCC; both MCCs were restored to operable status; and the LCO was exited after 7.5 hours.

The licensee's root cause team noted that one of the termination screws for the field cable connection to the thermal overload relay exhibited damage indicative of an arcing fault. Further review determined that the screw associated with the 'B' phase mounting hardware had penetrated the insulation on the 'C' phase. The resulting fault caused a flashover event within the cubicle and resulted in the upstream feeder breaker tripping on overcurrent with the subsequent loss of the 480V buses. The inspectors reviewed Work Order (WO) 00489851-01 which documented the most recent preventive maintenance (PM) on 1-EE-BKR-1J1-2N-B5. This WO was performed on December 28, 2004, and included the 5 year over current trip testing. Both the PM and the testing were controlled by procedure 0-EP-0304-01, "Testing/Replacing 480 Volt Breaker Assemblies," Rev. 41. The inspectors reviewed the procedure and determined that the event was caused by inadequate instructions. The procedure was subsequently revised based on the conclusions of the root cause team to include sufficiently detailed instructions to ensure proper termination of the affected wiring.

Analysis. The inspectors determined that the performance deficiency consisted of an inadequate procedure that resulted in faulty retermination of breaker wiring. The loss of the 480V buses resulted in the partial loss of 'B' train safety-related components involved with safety injection flow and containment recirculation spray heat exchangers. The inspectors referenced IMC 0612 and determined that the finding is more than minor because it affected the reactor safety mitigating system cornerstone objective to ensure availability, reliability and capability of systems that respond to initiating events to prevent core damage and the barrier integrity cornerstone objective to provide reasonable assurance that physical design barriers such as containment protect the public from radio nuclide releases caused by accidents or events. The attribute of procedure quality was affected for each aforementioned cornerstone. The inspectors referenced IMC 0609, for the significance determination process and determined that a Phase II analysis was required because the finding affected two cornerstones. This finding is unresolved pending completion of the significance determination assessment. The procedure problem is a resource issue involving the cross-cutting aspect of Human Performance.

Enforcement. TS 5.4.1 requires that written procedures shall be established, implemented, and maintained covering the activities in the applicable procedures recommended by Regulatory Guide (RG) 1.33, Revision 2, Appendix A, February 1978, of which part 9.e. specifies general procedures for the control of maintenance work. Contrary to the above, on December 28, 2004, maintenance procedure 0-EPM-0304-01 was not adequate, in that, it failed to provide sufficient instructions to preclude faulty retermination of wiring in breaker 1-EE-BKR-1J1-2N-B5. This led to an electrical fault and the loss of 1J1-2N and 1J1-2S MCCs on May 1, 2005. This violation is being treated as a URI pending significance determination, and is identified as URI 05000338/2005003-01, Inadequate Maintenance Procedure Results in Loss of Safety Related 480V Buses. This finding is in the licensee's corrective action program as Plant Issue N-2005-1615.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

The inspectors evaluated, as appropriate, for the seven work activities listed below: (1) the effectiveness of the risk assessments performed before maintenance activities were conducted; (2) the management of risk; (3) that, upon identification of an unforseen situation, necessary steps were taken to plan and control the resulting emergent work activities; and (4) that 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.

- Review of maintenance for week of April 5, 2005, including 2-FW-P-1C, 2-SA-C-1, rack work, switchyard, load shed, 1-PT-36.1A, 2-PT-75.2A, and 75.2B;

- Review of maintenance for week of April 17, 2005, including 1-EE-EG-1J, rack work, 1-PT-33.7, and 1-BC-P-1A;
- Review of maintenance during loss of offsite power testing (1/2-PT-71.14/15), including 1-HV-AC-7, RSWT level channel cals, switchyard, 0-ACC-06-OM, 0-PT-82.12, 2-FW-P-1C, 2-CW-P-10, and rack work;
- Review of the emergent work control associated with the loss of two Unit 1 480V MCCs 1J1-2N and 1J1-2S caused by a fault in 1-SW-P-7, “Unit 1 ‘C’ RS Radiation Monitor Sample Pump,” molded case circuit breaker;
- Review of emergent work associated with 2-FW-P-2 trip valve (2-MS-TV-215) loose trunnion bolts as described in Plant Issue N-2005-2264;
- Review of emergent work control associated with the loss of service water loop due to the inability of 2-SW-MOV-214B to automatically close on a containment depressurization actuation (CDA); and,
- Review of maintenance for June 9, 2005, including 1-EE-EG-1H, rack work, 1-HV-AC-6, and switchyard work.

b. Findings

No findings of significance were identified.

1R14 Operator Performance During Non-Routine Evolutions and Events

a. Inspection Scope

The inspectors reviewed operator logs and plant computer data for the events listed below to determine if plant and operator responses were in accordance with plant design, procedures, and training. The inspectors also evaluated performance and equipment problems to ensure that they were entered the corrective action program.

- The inspectors evaluated the response of the Unit 1 control room operators on April 1, 2005, for entry into abnormal procedure 0-AP-10, “Loss of Power” during an unexpected trip of the feeder breaker for 1-MCC-1B1-3 which caused a loss of power to the motor-operated valves 1-FW-154B and 1-FW-150A; and,

- The inspectors evaluated the response of the Unit 1 control room operators on May 1 and 2, 2005, for entry into abnormal procedure 0-AP-10, “Loss of Power” during an unexpected trip of the feeder breaker for 1-EE-MCC-1J1-2N and 2S, causing the loss of Unit 1 480V “J” emergency bus, and loss of power to several safety-related Motor Operated Valves (MOVs).

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed seven 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; (5) where continued operability was considered unjustified, the impact on TS Limiting Conditions for Operation and the risk significance in accordance with the Significance Determination Process. The inspectors' review included a verification that the operability determinations were made as specified by Procedure VPAP-1408, "System Operability."

- Plant Issue N-2005-1050, Dominion Substation Engineering group identified problem of the process used in the manufacture of Exide type 3CC-7 batteries which may result in premature degradation of the battery;
- Plant Issue N-2005-1462, following removal of control side radiator banks on 1J EDG, calcium deposits were identified on the tube sheets and around the tube openings;
- Plant Issue N-2005-1712, internal inspection of 1-SW-REJ-ISA1 has revealed a delamination in the inner tube of the expansion joint;
- Plant Issue N-2005-1747, 10 CFR 21 reporting of defects and non-compliance: Woodard Governor "Compensating" EG Series Actuations;
- Plant Issue N-2005-1774, unacceptable flow rates noted during 0-PT-76.4.2, "Control Room Pressure Envelope Flow Balance Verification," Revision 10;
- Plant Issue N-2005-2320, oil leak on Unit 1 TDAFW pump outboard bearing which resulted in an operable but degraded determination; and,
- Plant Issue N-2005-2264, NRC identified problem with loose trunnion bolts on Unit 2 TDAFW pump trip valve.

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds

a. Inspection Scope

The inspectors reviewed the cumulative effects of operator workarounds (OWAs) to assess: (1) the effect on the reliability, availability, and potential for mis-operation of a system; (2) the potential for increasing an initiating event frequency or affecting multiple mitigating systems; and (3) the cumulative effects on the ability of the operators to respond in a correct and timely manner to plant transients and accidents. The inspectors reviewed the current OWAs to determine if there were other conditions which would require actions to compensate for equipment problems or deficiencies.

The inspectors also reviewed OWA-110, Unit 1 ‘2A’ Recirculation Spray seal head tank high alarms to determine if the mitigating system function was affected or if the operator’s ability to implement abnormal and emergency operating procedures was impacted. The inspectors reviewed the addition of OWA-110 to the licensee OWA list and discussed the added OWA with the licensee in the context of the licensee operator being able to perform the OWA during and following transients.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing

a. Inspection Scope

The inspectors reviewed six 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 its safety function. The inspectors verified that these activities were performed in accordance with licensee procedure VPAP-2003, “Post Maintenance Testing Program.”

- Procedure 2-PT-71.2Q, “2-FW-P-3A, A Motor Driven Auxiliary Feedwater Pump and Valve Test,” per WO 52631401 for PMT-LKT-MM-001 for Post Maintenance External Leakage Test;
- Procedure 0-MCM-0100-01, “Inspection and Repair of Pumps in General,” per WO 52680201, to install new seal and gland follower on 1-CH-P-1A;
- Procedure 0-MCM-0701-37, “Radiator Inspection and Testing for EDG” per WO 052881;
- Procedure 0-EPM-0302-02, “BBC-ITE 480 Volt K-Line Breaker and Associated Switchgear Cubicle Maintenance” per WO 00600011-01;
- Procedure 2-PT-57.1A, “Emergency Core Cooling Subsystem - Low Head Safety Injection Pump (2-SI-P-1A)” performed following planned preventive maintenance as an emergent item relating to Plant Issues N-2005-1679 and N-2005-1690; and,
- Procedure 0-MOP-6.94, “0-AAC-DG-OM Alternate AC Diesel Generator (Station Blackout Diesel), per WO 493740.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

For the seven surveillance tests listed below, the inspectors examined the test procedure, 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 the technical specifications were met:

- 1-PT-36.21, "Verifying Interlocks for Breaker 15H7, Charging Pump 1-CH-P-1C;"
- 1-PT-36.28, "Verifying Interlocks for Breaker 15J7, Charging Pump 1-CH-P-1C;"
- 1-PT-71.2Q, "1-FW-P-3A, A Motor-Driven AFW Pump and Valve Test;" (IST)
- 1-PT-82.12J, "Isochronums Mode with ESF Start of 1J EDG;"
- 2-PT-82.4A, "2H Diesel Generator Test (Start by ESF Actuation);"
- 0-PT-76.4.2, "Control Room Pressure Envelope Flow Balance Verification;" and,
- 1-PT-71.15, "Loss of Offsite Power - Train A Operational Test for Auxiliary Feedwater Pumps."

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed Temporary Modification 2005-1749, "Unit 1 A RWST Refrigeration Unit," to verify that the modification did not affect system operability or availability as described by the TS and UFSAR. In addition, the inspectors verified that the installation of the temporary modification was in accordance with the work package, that adequate controls were in place, procedures and drawings were updated, and post-installation tests verified the operability of the affected systems.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems

.1 Daily Review

As required by Inspection Procedure 71152, "Identification and Resolution of Problems," and in order to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee's corrective action program. This review was accomplished by reviewing daily Plant Issues summary reports and periodically attending daily Plant Issue Review Team meetings.

.2 Semi-Annual Trend Review

a. Inspection Scope

As required by Inspection Procedure 71152, "Identification and Resolution of Problems," the inspectors performed a review of the licensee's corrective action program plant issues reports to identify trends that could indicate the existence of a more significant safety issue. The inspectors' review considered the results of daily inspector plant issue item screening discussed in section 4OA2.1 above, licensee trending efforts, and licensee human performance results. The inspector's review nominally considered the six month period of January through June 2005, although some examples expanded beyond those dates when the scope of the trend warranted. The review also covered areas not documented in plant issues reports such as: departmental problem/challenges lists, system health reports, quality assurance audit/surveillance reports, self assessment reports, and maintenance rule assessments. The inspectors compared and contrasted their results with the results contained in the licensee's latest quarterly trend reports. Corrective actions associated with a sample of the issues identified in the licensee's trend report were reviewed for adequacy.

b. Assessment and Observations

No findings of significance were identified. The inspectors evaluated the licensee trending methodology and observed that the licensee had performed reviews. The inspectors noted that the licensee has recently reorganized their approach to trending and has assigned specific resources to exclusively identify negative trends and report those findings through the corrective action system. In addition to the new effort, the licensee routinely reviewed cause codes, key words, and system links to identify potential trends in their plant issue report data. The inspectors compared licensee process results with the results of the inspectors' daily screening and did not identify any discrepancies or potential trends that the licensee had failed to identify except as noted below.

During the review period, the inspectors noted and discussed with the licensee several issues which could indicate an adverse trend in human performance. These issues pertained to human performance errors in electrical maintenance on safety-related equipment. Subsequently, the licensee performed a collective review of observations and Plant Issues generated during this time frame. The licensee concluded in their April 15, 2005, trend report (Plant Issue N-2005-0478) that a significant adverse trend in electrical maintenance work practices existed. Specifically, the report identified several human performance errors made by the electrical department involving inattention to detail, lack of situational awareness, and inadequate procedures. These errors were identified based on the number of plant issue reports throughout the review period that documented the need to rework equipment after maintenance had previously been performed. Noteworthy examples of rework items involved incorrectly configured contacts on the control power relay and speed switch for the 1H EDG which increased its outage time (Plant Issue N-2005-2141/2148); incorrectly configured contacts on the auxiliary switches for the pressurizer back-up heaters (Plant Issue N-2005-0080); quench spray pump breakers (Plant Issue N-2005-0080) which caused entry into technical specifications action statements; and the reactor trip breaker incorrectly configured contacts which caused a reactor trip (Plant Issue N-2004-2301).

The inspectors reviewed the licensee's corrective actions associated with these issues using the guidance contained in Inspection Procedure 71152 as well as other baseline inspection procedures. The inspectors determined that the licensee's planned and implemented corrective actions appear to be reasonable and appropriate. To correct these issues, the licensee instituted a procedure upgrade to enhance existing procedures and develop new ones to prevent incorrectly configuring contacts in electrical components. The licensee also completed informal shop training in reading and understanding circuit diagrams, and instituted the use of task preparation forms to help prompt their thinking before work is performed.

4OA3 Event Followup

.1 (Closed) Licensee Event Reports (LERs) 05000339/2001-003-00 and 05000339/2001-003-01: Reactor Vessel Head Nozzle Through-Wall Leakage Due to Lack of Weld Fusion During Original Fabrication

In October 2001, a qualified bare-head visual (VT) examination for leakage of the Unit 2 reactor pressure vessel head (RPVH) control rod drive mechanism (CRDM) penetrations was performed in accordance with NRC Bulletin 2001-01. The bare head VT inspection identified boric acid deposits indicative of leakage at penetrations 51, 62, and 63. Followup liquid penetrant (PT) inspection of the J-Groove weld surfaces identified indications at the toe of the J-Groove weld in all three penetrations. The indications were similar to those identified at four penetrations in Unit 1 in September 2001, and were initially dispositioned as being in the clad and not requiring further characterization. After discussions with the NRC, the licensee decided to investigate the indications by excavation of an indication area in penetration 63. After excavation to a depth of approximately 1 inch and a length of 2 and 3/4 inches, PT inspection revealed an indication extending almost the full length of the excavation. In addition, a "boat" sample was taken from an indication area in penetration 62 weld. Metallurgical analysis showed the indication to be caused by predominantly hot-short cracking. Based on the investigation (excavation and metallurgical analysis) the licensee concluded that the flaws penetrated deep into the welds at the butter to J-Groove weld interface and that penetration 63 apparently had a through-wall leak. All three penetration J-Groove welds were repaired using a weld embedded flaw overlay technique. The three penetrations were also inspected using inside diameter (ID) eddy current (ET) and ultrasonic (UT) and outside diameter UT examinations. Although some shallow indications were identified, it was concluded that the ET and UT examinations demonstrated that there was no evidence of a flaw propagating from the OD of the penetration or the penetration to weld fusion zone toward the ID and around the penetration circumferentially.

The inspectors reviewed docketed correspondence and licensee documentation associated with the inspection, identified leakage, and disposition of penetration 63 to determine if the leakage was a result of a licensee performance deficiency at the time of the 2001 inspection. This included a sample of the inspection reports/results (VT, UT, ET and PT) from the 2001 and 2002 inspections, Engineering Transmittals documenting inspection results and disposition of results, and Plant Issues Reports. In addition, the inspectors held discussions with inspection and engineering personnel involved in disposition of the inspection results (VT, UT, ET, and PT) from the 2001 and 2002 inspections.

The inspectors concluded that at the time of identification of the leak at penetration 63 in 2001, the licensee's decisions were reasonable. This was the first inspection of the Unit 2 head and therefore the first opportunity to discover leaks. The PT indications at the toe of the J-Groove welds were initially considered in the clad and not associated with cracks/leaks through the weld or penetration as identified at other utilities. At the time of the inspection, techniques and characterization of findings were still being developed. After discussions with the NRC, the licensee decided to investigate the indications further and determined that the indications were caused predominately by hot short cracking and penetrated deep into the welds. This information, when combined with the results of the bare head VT inspection lead the licensee to the conclusion that leakage had probably occurred at penetration 63. The violation (pressure boundary leakage) was not avoidable by quality assurance measures and management controls, which were considered reasonable at the time, and no performance deficiency was identified.

The leaking penetrations were assessed for risk by the NRC. The risk assessment process assumes a Weibull distribution for the time that leakage started in each nozzle that has been observed to be leaking. Based on industry experience, it is assumed that any nozzle on which the annulus above the J-groove is weld wetted by leakage, will have a probability of 0.2 for a circumferential crack to initiate. A probabilistic distribution of circumferential crack growth rates is estimated on the basis of stress level and material properties for each leaking nozzle. The distributions for leakage duration and crack growth rates are combined to obtain a distribution for the probability that a nozzle that has been leaking could develop a circumferential crack greater than 330E, as a function of plant age. The growth of a circumferential crack to 330E is assumed to cause nozzle ejection, resulting in a medium break Loss of Coolant Accident (MBLOCA). The probability that a nozzle ejection would occur in the year prior to the inspection of interest is the increase in MBLOCA frequency for the unit due to the leakage. The resulting change in the core damage frequency is calculated by multiplying this change in the MBLOCA frequency by the conditional probability that a MBLOCA will result in core damage at the affected unit. The assessment considered the fact that the penetrations had not previously been examined by UT technique to eliminate the possibility of circumferential cracking, which would reduce the risk. The associated risk of operating with vulnerable penetrations can be reduced by performing the necessary non-destructive examinations and eliminating the possibility that circumferential cracking has initiated. The probability of CRDM nozzle ejection, which causes a MBLOCA, is applied to the licensee's SPAR model. The assessment determined this issue to have resulted in a low to moderate safety significance.

Pressure boundary leakage was a violation of requirements, in that, Technical Specification 3.4.13 limits operational leakage to “No pressure boundary leakage” while in Modes 1-4. However, as discussed in the NRC’s Enforcement Policy, the NRC may refrain from issuing enforcement action for violations resulting from matters not within the licensee’s control, such as equipment failures that were not avoidable by reasonable Licensee quality assurance measures or management controls. Based on the circumstances of this violation, the NRC considers it appropriate to exercise enforcement discretion in accordance with Section VII.B.6 of the Enforcement Policy and refrain from issuing enforcement action for this violation.

.2 (Closed) LER 05000338/2003-001-00, Reactor Vessel Head Penetration Nozzle Leakage

On March 4, 2003, during a scheduled refueling outage, VT inspection identified an apparent through-wall leak of the Unit 1 RPVH at penetration 50. The inspection was to followup on previous inspection results from the outage in 2001. The conclusion that leakage had occurred was based on a boric acid deposit approximately one half-inch in diameter at the lower side of the penetration-to-head transition. Only visual inspection was performed with no verification with other NDE techniques. At the time of the inspection, the licensee had already made the decision to replace the Unit 1 head during the outage in process. The 2001 inspection of Penetration 50 identified flaw indications (PT indications at the toe of the J-Groove weld) that were not repaired. The inspectors performed an inspection to determine if the leakage at Penetration 50 was a result of a licensee performance deficiency at the time of the 2001 inspection.

The inspectors reviewed docked correspondence and licensee documentation associated with the inspection and leakage of penetration 50. This included a sample of the inspection reports/results (VT, UT, ET, and PT) from the 2001 inspections, Engineering Transmittals documenting inspection results and disposition of results, Plant Issues Reports, and the Unit 1 Justification for Continued Operations after the Unit 2 inspection identified through-wall leakage from penetrations with similar indications. In addition, the inspectors held discussions with inspection and engineering personnel involved in disposition of the inspection results from the 2001 inspection.

Based on industry experience (specifically identification of circumferential RPVH penetration cracking above the J-Groove weld), NRC Information Notice 2001-05, NRC Bulletin 2001-01, and EPRI Materials Reliability Program (MRP) committee request, the licensee committed to perform a bare metal VT inspection of the Unit 1 RPVH during the fall of 2001 refueling outage. The MRP committee requested that all plants ranked high in leakage susceptibility perform the bare metal VT inspection for leakage. North Anna Unit 1 was included in the highly ranked plants. The commitment to perform the bare metal VT inspection included performing ET inspection under the head to detect small surface connected flaws, contingent upon availability and acceptable performance of necessary equipment and personnel.

The bare metal visual of the penetration to bare metal head surface, using a combination of robotic cameras and a boroscope, initially reported 34 of the 65 penetrations with relevant indications of boric acid. The VT inspection was difficult because of boric acid streaks and spatter residue on the head surface due to an active conoseal leak. In addition, there had been a history of conoseal leakage. Further review of the tapes of the inspection resulted in acceptance of 7 of the 34 penetrations. Based on a combination of the available inspection techniques at the time (OD and ID ET examinations, ID UT examinations, and PT examinations), and flaw growth evaluations, the licensee concluded that there were no through-wall leaks on the other 27 penetrations and the head was acceptable for another cycle of operation.

Since the VT inspection of penetration 50 initially characterized the penetration as having evidence of significant leakage, all available NDE techniques were used to try and determine if cracks were present in the J-Groove weld or if through-wall flaws existed. The surface of the J-Groove weld and the penetration OD surface below the J-Groove weld were ET examined. No indications were identified on the penetration OD and one non-recordable indication was identified on the surface of the J-Groove weld. ET of the ID tube surface covering 98% the weld area did not detect any indications. PT inspection of the J-Groove weld surface identified one non-service induced flaw on the surface of the weld and several other indications at the toe of the weld on the clad side of the weld. The non-service induced flaw was near the penetration tube side of the weld and was removed by grinding. The indications at the toe of the weld were considered to be in the clad and not relevant to identification of flaws or primary water stress corrosion cracking (PWSCC) in the weld surface or the penetration material. (Similar PT indications were identified in J-Groove welds of three other penetrations.) A partial UT of the penetration tube ID at the weld was performed from 196 degrees to 293 degrees. A portion of the thermal sleeve was removed to gain additional access of the penetration above the centering ring for UT and ET. The penetration ID was UT and ET inspected and no indications were identified. Based on the NDE performed on Penetration 50, the licensee concluded that no evidence of through-wall flaws existed and therefore, the boric acid identified at penetration 50 was from an external source. The inspectors considered the UT to be adequate to identify circumferential cracks above the J-Groove weld. As noted above, there was an active conoseal leak and there had been a history of conoseal leaks.

After Unit 1 was returned to service, an inspection of the Unit 2 RPVH identified apparent leakage at three penetrations. PT inspection of the J-Groove welds for these three penetrations revealed PT indications at the toe of the welds similar to those identified in Unit 1 penetrations, including penetration 50. Removal of a “boat” sample and metallurgical analysis of the indications in one of the Unit 2 welds (penetration 62) determined that the PT indications were caused by hot-short cracking in the J-Groove weld butter layers, which resulted in through-wall leaks. Based on the Unit 1 PT indications being similar to the Unit 2 indications, it was concluded that the Unit 1 indications were also caused by hot-short cracking and could result in through-wall leakage. Therefore, Justification for Continued Operation (JCO) C02-02 was issued for Unit 1.

During the 2003 refueling outage, as followup to the 2001 Unit 1 inspection findings, the licensee performed a VT inspection on Unit 1 penetration 50 and found boric acid deposits at the penetration to head intersection indicating through-wall leakage.

Based on review of documentation and interviews with personnel associated with disposition of the inspection findings in 2001, the inspectors concluded that at the time, the licensee's decisions were reasonable. At the time, inspection techniques and characterization and resolution of findings were still being developed. The licensee did not consider the PT indications on the head side of the J-Groove weld, which they thought were in the clad, to be associated with the cracking and/or leakage in the penetration or weld material previously identified at other plants. The violation (pressure boundary leakage) was not avoidable by quality assurance measures and management controls, which were considered reasonable at the time, and no performance deficiency was identified.

The leaking penetration was assessed for risk by the NRC. The assessment considered the fact that the nozzle had previously been examined by UT technique to eliminate the possibility of circumferential cracking, which would reduce the risk. The assessment determined this issue to have resulted is a risk of very low safety significance.

Pressure boundary leakage was a violation of requirements, in that, Technical Specification 3.4.13 limits operational leakage to "No pressure boundary leakage" while in Modes 1-4. However, as discussed in the NRC's Enforcement Policy, the NRC may refrain from issuing enforcement action for violations resulting from matters not within the Licensee's control, such as equipment failures that were not avoidable by reasonable Licensee quality assurance measures or management controls. Based on the circumstances of this violation, the NRC considers it appropriate to exercise enforcement discretion in accordance with Section VII.B.6 of the Enforcement Policy and refrain from issuing enforcement action for this violation.

4OA4 Cross Cutting Aspects of Findings

Section 1R12 of this report describes an inadequate procedure which led to a faulted condition in a molded case circuit breaker and the subsequent loss of two safety-related MCCs. The procedure problem is a resource issue involving a cross-cutting aspect of Human Performance.

4OA5 Other Activities

.1 Review of the Operation of an Independent Spent Fuel Storage Installation (60855)

a. Inspection Scope

Inspectors reviewed the normal operations of the Independent Spent Fuel Storage Installation (ISFSI). The inspectors walked down the ISFSI pad to assess the material condition of the casks, the installation of security equipment, and the performance of the monitoring systems. In preparation for an upcoming cask loading, the inspectors reviewed licensee cask loading and handling procedures and reviewed previous cask loading and ISFSI related plant issues and corrective actions status. The inspectors also reviewed bridge crane lubrication / inspection work orders completion data and calibration data sheets for equipment that would be used during cask loading. Specific documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

.2 (Closed) Temporary Instruction (TI) 2515/163 "Operational Readiness of Offsite Power"

a. Inspection Scope

The inspectors performed Temporary Instruction 2515/163, "Operational Readiness of Offsite Power." The inspectors collected and reviewed licensee procedures and supporting information pertaining to the offsite power system specifically relating to the areas of offsite power operability, the maintenance rule (10 CFR 50.65), and the station blackout rule (10 CFR 50.63). The inspectors reviewed this data against 10 CFR 50 Appendix A General Design Criterion (GDC) 17, "Electric Power Systems," and the requirements of 10 CFR 50.63, 10 CFR 50.65, and Plant Technical Specifications. This information was forwarded to the Office of Nuclear Reactor Regulation for further review.

b. Findings

No findings of significance were identified.

4OA6 Meetings, including Exit

On June 30 and July 14, 2005, the senior resident inspector presented the inspection results to Mr. Larry Lane and other members of the staff. The licensee acknowledged the findings. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

4OA7 Licensee-Identified Violation

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

North Anna Unit 2 Renewed Facility Operating License, No. NPF-7, Section 2.D, "Fire Protection," states that VEPCO shall implement and maintain in effect all provisions of the approved fire protection program as described in the licensee's UFSAR Section 9.5.1.1, Design Basis, includes the station's Technical Requirement's (TR) Manual. TR 7.10 states that passive fire protection features listed in Table 7.10-1, which includes fire retardant coating/wrap, are required to be OPERABLE. Contrary to the above, on

June 21, 2005, the licensee discovered a small section (less than 2 square inches) of missing fire wrap on a unistrut conduit clamp for 2-CC-P-1A, during the implementation of 0-PT-108.4, "Visual Inspection–Auxiliary Building Fire Retardant Coatings, Cable Tray Fire Stops, Required by Appendix R." The inspectors reviewed IMCs 0612 and 0609 and consulted with regional fire protection inspectors and determined that the finding was of very low safety significance given the small area of missing fire wrap. The licensee has this finding documented in their corrective action program as Plant Issue N-2005-2307.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

G. Bischof, Director, Nuclear Safety and Licensing
W. Corbin, Director, Nuclear Engineering
J. Crossman, Assistant Manager, Nuclear Operations
J. Davis, Site Vice President
R. Evans, Manager, Radiological Protection
R. Foster, Supply Chain Manager
S. Hughes, Manager, Nuclear Operations
P. Kemp, Supervisor, Nuclear Safety & Licensing
J. Kirkpatrick, Manager, Maintenance
L. Lane, Director, Operations and Maintenance
J. Leberstien, Licensing Technical Advisor
T. Maddy, Manager, Nuclear Protection Services
M. Main, Component Engineer
F. Mladen, Manager, Nuclear Site Services
B. Morrison, Assistant Engineering Manager
H. Royal, Manager, Nuclear Training
M. Sartain, Manager, Nuclear Engineering
R. Williams, Component Engineer

LIST OF ITEMS OPENED AND DISCUSSED

Opened

05000338/2005003-01	URI	Inadequate Maintenance Procedure Results in Loss of Safety Related 480V Buses (Section 1R12)

Closed

05000339/2001-003-00	LER	Reactor Vessel Head Nozzle Through-Wall Leakage Due to Lack of Weld Fusion During Original Fabrication (Section 4OA3.1)
05000339/2001-003-01	LER	Reactor Vessel Head Nozzle Through-Wall Leakage Due to Lack of Weld Fusion During Original Fabrication (Section 4OA3.1)
05000338/2003-001-00	LER	Reactor Vessel Head Penetration Nozzle Leakage (Section 4OA3.2)
2515/163	TI	Operational Readiness of Offsite Power (Section 4OA5.2)

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

- Plant Issue N-2005-1096, “Unit 1 Rod Drive Room area ambient temperature exceeded 95 degrees F for approximately 12 hours”
- Plant Issue N-2005-2081, “2-SA-C-1 tripped on LP air hi temp due to running unloaded with hi ambient temperature”
- Plant Issue N-2005-2093, “1J, 2H, and 2J emergency diesel room ambient temperatures were found to be all in excess of 104 degrees but less than the analyzed limit”

Section 1R04: Equipment Alignment

Documents

- Procedure 2-OP-6.2, Operator of 2J Emergency Diesel Generator from Control Room, revision 36
- Procedure 2-OP-6.2A, Valve Checkoff - 2J Diesel Engine Cooling Water, revision 8
- Procedure 2-OP-6.4A, Valve Checkoff - 2J Diesel Engine Lube Oil System, revision 6
- Procedure 2-OP-6.8A, Valve Checkoff - Emergency Generator Fuel Oil System, Revision 8
- UFSAR, Section 9.5, Other Auxiliary Systems, revision 40
- Emergency Diesel Generator System Drawing 12050-FM-107A, B, C
- Procedure 0-OP-49.1A, Valve Checkoff - Service Water System, revision 38
- Procedure 0-MOP-49.09, Service Water Header Removal

Drawings

- 11715-FM-078A. Service Water System - Sheet 1 of 5, Rev. 62
- 11715-FM-078A. Service Water System - Sheet 2 of 5, Rev. 34
- 11715-FM-078A. Service Water System - Sheet 3 of 5, Rev. 58
- 11715-FM-078A. Service Water System - Sheet 4 of 5, Rev. 91
- 11715-FM-078A. Service Water System - Sheet 5 of 5, Rev. 4
- 11715-FM-078B Service Water System - Sheet 1 of 4, Rev. 33
- 11715-FM-078B Service Water System - Sheet 3 of 4, Rev. 34
- 11715-FM-078C Service Water System - Sheet 1 of 2, Rev. 54
- 11715-FM-078C Service Water System - Sheet 2 of 2, Rev. 42

Section 1R12: Maintenance Effectiveness

Procedures

- 0-GOP-4.4, Equipment Rotation, Rev. 18
- 0-AP-10, Loss of Electrical Power, Rev.44

Documents

- WO 00489851-01
- Risk Assessment associated with the loss of two MCCs

Plant Issues

- N-2005-0478, possible adverse trend detected in electrical maintenance work practices
- N-2005-1615, supply breaker 14J-5 tripped open when starting 1-SW-P-7 for equipment rotation
- N-2005-1240, feeder breaker tripped open causing loss of 1-EP-MCC-1B1-3
- N-2005-1246, feeder breaker removed and inspected after tripping for fault on MCC bus

Drawings

- 11715-FE-1R, Load List for 1EE-MCC-1J1-2S
- 11715-FE-1R, Load List for 1EE-MCC-1J1-2N

Section 1R15: Operability Evaluations

- N-PMTE-2003-0053, PM Task Evaluation for inspection frequency of 2-MS-TV-215
- 0-MCM-0412-01, "Repair of Gimpel Valves (Terry Turbine Trip Valves)"
- 59-G005-00001, "Easy Flow body Combined Trip Throttle Valve," vendor technical manual for the terry turbine trip valve (2-MS-TV-215)
- WO 00523173-01, PM for lubrication of 2-MS-TV-215
- WO 00513885-01, PM for lubrication of 2-MS-TV-215

Section 4OA2: Identification and Resolution of Problems

Plant Issues

- N-2005-2301, during performance of 2-PT-36.1A (train A Reactor Protection and ESF Logic Actuation Logic Test), a reactor trip occurred
- N-2004-4274, 1-QS-MOV-100A failed to stroked closed electrically during 1-PT-63.1A, (U-1 "A" QS)
- N-2004-4299, while working WO 520580-01 on breaker 14h1-1, found 52 cell switch set up opposite of esk-6k
- N-2004-4730, shortly after unit 2J EDG shutdown for maintenance run annunciator JC7, EDG 2J interlocks not reset was received in the MCR
- N-2004-4735, about two minutes after the EDG shutdown during maintenance run the 2J interlock alarm activated
- N-2005-0080, while performing PMT on 2-EE-BKR-24H1-6 (WO 525829-01) after 9 ½ year inspection performed by Waukesha found contacts 26 and 28 reversed
- N-2005-0478, using information in the Corrective Action System data base, a possible adverse trend has been detected in electrical maintenance work practices, dealing with plant critical equipment and status control
- N-2005-2141, after reinstalling fuses in the 1H EDG as per 1-MOP-6.90, several alarms were locked in on the 1H EDG Room annunciator panel which were unexpected
- N-2005-2148, while clearing tags on 1H diesel HSR picked up causing alarms on the 1H diesel annunciator panel

Section 4OA3: Event Followup

Engineering Documents

- Engineering Transmittal (ET) N 02-102, Reactor Vessel Head Penetration Inspection North Power Station, Unit 2, Revision 0
- ET N 01-142, Reactor Vessel Head Inspection Results North Anna Power Station, Unit 1, Revision 0, 1, and 2
- ET N 01-180, Reactor Vessel Head Inspection Results North Anna Power Station, Unit 2, Revisions 0, 1, and 2
- ET N 01-190, Reactor Vessel Head Extent of Condition, North Anna Power Station, Units 1 and 2, Revision 0
- Justification for Continued Operation (JCO) C-02-02, Unit 1 Reactor Vessel Head Penetrations
- Category 1 Root Cause Evaluation Response - N-2002-2287-E1, Unit 2 Reactor Head Through-Wall Leakage
- Category 1 Root Cause Evaluation Response - N-2002-2287-E2, Unit 2 Reactor Head Through-Wall Leakage (Penetration 21)
- Category 1 Root Cause Evaluation Response - N-2003-0910-E2, Unit 1 Reactor Vessel Head Through-Wall Leakage
- Category 2 Root Cause Evaluation Response N-2001-3310-E1, Unit 2, Determine Root Cause of Liquid Penetrant Indications in Reactor Vessel Nozzle 63
- Westinghouse WCAP-15777, Metallurgical Investigation of Cracking of the Alloy 82/182 J-Groove Weld of the Reactor Vessel Head Penetration at North Anna Unit 2 Station

Plant Issues

- N–2001-2994, Unit 1, Failure of VT-2 Examination (Bare Metal Visual) of Reactor Pressure Vessel Head Penetrations, including Resolution R1
- N–2002-3383, Unit 1, Operability Evaluation of Reactor Vessel Head (Penetration 50) Inadequate because of not Evaluate Root Cause of Boric Acid Around Penetration 50
- N–2002-2287, Unit 2, Apparent Leakage of Reactor Vessel Head Penetration 21, including Resolution Nos. R1, R2, R3, R4, and R5
- N–2001-3310, Unit 2, Liquid Penetrant Indications in Reactor Vessel Head Penetration 63, including Communication C1 and Resolutions R1, R2, R3, R4, R5, and R6
- PI –2001-3248, Unit 2, VT-2 (Bare Metal Visual) Examination of Penetrations 51, 62, and 63 Rejectable due to Boric Acid Deposits on the Bare Head

Records

- Sample of VT, UT, ET, and PT Examination Reports and Examination Summaries for 2001 Unit 1, 2001 Unit 2 and 2002 Unit 2 RPVH Examinations

Section 4OA5: Other Activities - Review of the Operation of an Independent Spent Fuel Storage Installation

Procedures

- 0-OP-4.33. "Pre-Cask Loading Verification"

- 0-OP-4.35, "TN-32 Cask Loading and Handling"

Plant Issues

- N-2005-0304, ISFSI cask lifting rig has been missing it 's brass inset for approximately 1 year
- N-2005-1285, when securing from configuration Bravo in the fuel building, the discharge damper of 1-HV-F-7A (fuel building exhaust fan) was determined to be failed closed by local observation and no flow charge to "B" vent stack
- N-2005-1357, fuel building exhaust from 1-HV-MOD-102A (1-HV-F-7A) discharge damper only open approximately 50%

Section 4OA5: Other Activities - Temporary Instruction 2515/163, Operational Readiness of Offsite Power

Procedures

- 1-OP-26.8 "500 KV Switchyard Voltage," Rev. 12
- 0-AP-8 "Response to Grid Instability," Rev. 1
- 0-AP-10, "Loss of Electrical Power," Rev.44
- System Operations Procedures Manual Chapter 2 "Transmission Operation," Rev. 3
- DNAP-2000 "The Dominion Work Management Process," Rev. 3
- SEAP-0002 "Shift Technical Advisor," Rev. 7
- ECA-0.0 "Loss of All AC Power," Rev. 19