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

July 31, 2012

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

SUBJECT: NORTH ANNA POWER STATION - NRC INTEGRATED INSPECTION REPORT 05000338/2012003, and 05000339/2012003

Dear Mr. Heacock:

On June 30, 2012, the U. S. 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 July 31, 2012, with Mr. G. Bischof and other members of your staff.

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

This report documents one NRC-identified finding of very low safety significance (Green) which was determined to involve a violation of NRC requirements. The NRC is treating this finding as a non-cited violation (NCV) consistent with Section 2.3.2 of the NRC Enforcement Policy. If you contest this NCV, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the United States Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001, with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the North Anna Power Station.

Additionally, if you disagree with a cross-cutting aspect assignment in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region II, and the NRC Resident Inspector at the North Anna Power Station.

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

Sincerely,

/RA/

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

Enclosure: Inspection Report 05000338/2012003 and 05000339/2012003
w/ Attachment: Supplemental Information

cc w/Encl: (See page 3)

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ADAMS: ☐ Yes ACCESSION NUMBER: ML12213A637 ☐ SUNSI REVIEW COMPLETE ☐ FORM 665 ATTACHED

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DATE	7/31/2012	7/31/2012	7/31/2012	7/31/2012	7/31/2012		
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Letter to David A. Heacock from Gerald J. McCoy July 31, 2012

SUBJECT: NORTH ANNA POWER STATION – NRC INTEGRATED INSPECTION REPORT 05000338/2012003, and 05000339/2012003

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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos:	50-338, 50-339
License Nos:	NPF-4, NPF-7
Report No:	05000338/2012003 and 05000339/2012003
Licensee:	Virginia Electric and Power Company (VEPCO)
Facility:	North Anna Power Station, Units 1 & 2
Location:	1022 Haley Drive Mineral, Virginia 23117
Dates:	April 1, 2012, through June 30, 2012
Inspectors:	G.Kolcum, Senior Resident Inspector R. Clagg, Resident Inspector L. Lake, Senior Reactor Inspector (Section 1R08)
Accompanied by:	M. Levine, Nuclear Safety Professional Development Program (Training)
Approved by:	Gerald J. McCoy, Chief Reactor Projects Branch 5 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000338/2012-003, 05000339/2012-003; 04/01/2012 – 06/30/2012; North Anna Power Station, Units 1 and 2; Inservice Inspection Activities

The report covered a three month period of inspection by resident inspectors and one reactor inspector from the region. One finding was identified and was determined to be a non-cited violation (NCV). The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). The cross-cutting aspect was determined using IMC 0310, "Components within the Cross Cutting Areas". Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC Identified and Self-Revealing Findings

- Green. A self revealing non-cited violation (NCV) of the required augmented ISI examinations identified in 10 CFR 50.55a(g)(6)(ii)(F), Examination requirements for Class 1 piping and nozzle dissimilar metal butt welds, which implements ASME Code Case N-770-01, that covers alternative examination requirements and acceptance standards for Class 1 PWR Piping and Vessel Butt Welds Fabricated with Alloy 82 and 182 Filler Material was identified for the licensee's failure to identify unacceptable PWSCC indications in the Unit 1 B SG hot leg nozzle safe-end weld. These requirements require in-service examinations to be performed using qualified techniques and with qualified personnel capable to identify primary water stress corrosion cracking (PWSCC) indications. The licensee entered this issue into its corrective action program as condition report CR467649.

  The inspectors determined that the failure to identify the PWSCC indications in the Unit 1 B steam generator (SG) hot leg safe-end weld was a self-revealing performance deficiency that was within the licensee's ability to foresee and correct. Using IMC 0612, the inspectors determined that this finding was of more than minor significance because the failure to identify the PWSCC could have resulted in the potential to allow degradation of the safe-end to proceed undetected. Unchecked PWSCC degradation could have resulted in more significant degradation of the safe-end weld with subsequent degradation of the primary system pressure boundary. The finding is associated with the design control attribute of the Barrier Integrity Cornerstone and affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, examinations of the SG safe-end welds provide assurance that the structural boundary of the reactor coolant system remains capable of performing its intended safety function. The inspectors used IMC 0609, "Significance Determination Process," Attachment 0609.04, "Initial Characterization of Findings," and determined that the finding was of low safety significance (Green) because it did not represent an actual failure of the safe-end pressure retaining boundary. The inspectors identified a cross-cutting aspect in the Human Performance Work Practices cross cutting area, H.4 (c). Specifically, the licensee failed to conduct an adequate briefing with NDE technicians prior to the examination to ensure its successful execution. (Section 1R08)

B. Licensee Identified Violations

None.

REPORT DETAILS

Summary of Plant Status

Unit 1 began the inspection period in a refueling outage that began on March 11, 2012. On May 9, 2012, Unit 1 returned to Rated Thermal Power (RTP) operation and operated at full power for the remainder of the report period.

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

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection

.1 Seasonal Susceptibilities

a. Inspection Scope

The inspectors reviewed the licensee's adverse weather preparations for hot weather operations, specified in 0-GOP-4.1, "Hot Weather Operations," Revision 29, 0-AP-5.5, "EDG Hot Weather Operations," Revision 12, and the licensee's corrective action program (CAP) database for hot weather related issues. The inspectors walked down the risk-significant systems/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.

- Unit 1 & 2 Auxiliary Feedwater (AFW) pump rooms
- Service Water Pump House
- Service Water Valve House
- 1H, 1J, 2H, and 2J Emergency Diesel Generators (EDGs)

b. Findings

No findings were identified.

.2 Review of Offsite Power and Alternate AC Power Readiness

a. Inspection Scope

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

b. Findings

No findings were identified.

.3 Site Specific Events

a. Inspection Scope

The inspectors performed three site specific weather related inspections, listed below, due to anticipated adverse weather conditions. Specifically, the inspectors reviewed licensee adverse weather response procedures and site preparations including work activities that could impact the overall maintenance risk assessments. The documents utilized and reviewed as part of the inspections are listed in the Attachment.

- Forecasted severe thunderstorms and heavy winds on May 9, 2012
- Forecasted severe thunderstorms and heavy winds with hail on June 1, 2012
- Tornado watch on June 26, 2012

b. Findings

No findings were identified.

.4 External Flooding

a. Inspection Scope

The inspectors assessed the external flood vulnerability of the North Anna site. The inspectors verified the condition of the service water reservoir emergency flood protection dike between the service water reservoir, and the plant and related drainage ditches and culverts, in addition to the west side flood protection dike. The inspectors also reviewed applicable station procedures and design documents to assess proper surveillance and maintenance for external flood protection features.

b. Findings

No findings were identified.

1R04 Equipment Alignment

.1 Partial Walkdowns

a. Inspection Scope

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

- Unit 1 direct current distribution battery system after battery testing
- Unit 1 EDG starting air system for 1H EDG during maintenance on 1J EDG
- 1J EDG fuel oil system after leak on fuel oil duplex strainer

b. Findings

No findings were identified.

.2 Complete Walkdown

a. Inspection Scope

The inspectors performed a detailed walkdown and inspection of the Unit 1 Auxiliary Feedwater System and related valves/piping/supports/instrumentation to assess proper alignment and to identify discrepancies that could impact its availability and functional capacity. The inspectors assessed the physical condition and position of related risk significant components based on guidance from internal NRC risk documentation, related TS, UFSAR, and design bases documents. The inspection also included a review of the alignment and the condition of support systems including fire protection, room ventilation, and emergency lighting. Equipment deficiency tags were reviewed as well as the work history and CAP documentation. The condition of the system was discussed with the engineering personnel. The operating procedures, drawings, and other documents utilized and reviewed as part of the inspection are listed in the Attachment.

b. Findings

No findings were identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors conducted focused tours of the six areas listed below that are important to reactor safety to verify the licensee's implementation of fire protection requirements as described in fleet procedures CM-AA-FPA-100, Revision 5, "Fire Protection/Appendix R (Fire Safe Shutdown) Program," CM-AA-FPA-101, "Control of Combustible and Flammable Materials," Revision 3, and CM-AA-FPA-102, "Fire Protection and Fire Safe Shutdown Review and Preparation Process and Design Change Process," Revision 3. 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. Other documents utilized and reviewed as part of the inspections are listed in the Attachment.

- Emergency Diesel Generator 1H Unit 1 (fire zone 9A-1a / EDG-1H) and Emergency Diesel Generator 2H Unit 2 (fire zone 9A-2a / EDG-2H)
- Emergency Diesel Generator 1J Unit 1 (fire zone 9B-1a / EDG-1J) and Emergency Diesel Generator 2J Unit 2 (fire zone 9B-2a / EDG-2J)
- Containment Unit 1 (fire zone 1-1a / RC-1)
- Cable Vault and Tunnel Unit 1 (includes Control Rod Drive Room and Battery Room and Z-27-1) (fire zone 3-1a / CV & T-1)
- Cable Vault and Tunnel Unit 2 (includes Control Rod Drive Room and Battery Room and Z-27-2) (fire zone 3-2a / CV & T-2)
- Service Water Pump House (fire zone 12a / /SWPH), Auxiliary Service Water Pump House (fire zone 13a / ASWPH), Motor-Driven Fire Pump Building (fire zone 26 / FPB), and Service Water valve House (fire area 48a / SWVH)

b. Findings

No findings were identified.

1R06 Flood Protection Measures

a. Inspection Scope

The inspectors performed an annual review of cables located in underground bunkers/manholes. The inspectors evaluated, as appropriate, the four manholes listed below for the following: (1) verified by direct observation that the cables were not submerged in water; (2) verified by direct observation that cables and/or splices appeared intact; (3) verified that drainage or an appropriate dewatering device (sump pump) was in operation; and (4) verified that level alarm circuits were set appropriately to ensure that the cables would not be submerged.

- Manhole 01-BLD-MBAR-50MH-3
- Manhole 01-BLD-MBAR-50MH-4

- Manhole 01-BLD-MBAR-5MH-03
- Manhole 01-BLD-MBAR-5MH-04

b. Findings

No findings were identified.

1R08 Inservice Inspection Activities

a. Inspection Scope

Non-Destructive Examination Activities and Welding Activities: From March 19 - 30, 2012, the inspectors conducted an on-site review of the implementation of the licensee's inservice inspection (ISI) program for monitoring degradation of the reactor coolant system, emergency feed water systems, risk-significant piping and components, and containment systems in Unit 1. The inspector's activities included a review of non-destructive examinations (NDEs) to evaluate compliance with the applicable edition of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Section XI (Code of record: 2004 Edition, with no Addenda), and to verify that indications and defects were appropriately evaluated and dispositioned in accordance with the requirements of the ASME Code, Section XI, acceptance standards or an NRC-approved alternative requirement.

The inspectors directly observed, or reviewed records of, the following NDEs mandated by the ASME Code to evaluate compliance with the ASME Code Section XI and Section V requirements and, if any indications and defects were detected, evaluated whether they were dispositioned in accordance with the ASME Code or an NRC-approved alternative requirement.

- Inspectors observed the ultrasonic examination of reactor pressure vessel head penetration # 47.
- The inspectors reviewed the records of the volumetric ultrasonic (UT) examinations performed on the B steam generator (SG) hot leg safe-end weld.

During the review of the volumetric ultrasonic (UT) examinations performed on the B steam generator (SG) hot leg safe-end weld the inspectors identified that there were unresolved questions on the procedures used to conduct the examinations. This issue is currently being addressed as an Unresolved Issue (URI) presented in "Findings" below.

The inspectors reviewed documentation for the repair/replacement of the following pressure boundary welds. The inspectors evaluated if the licensee applied the pre-service non-destructive examinations and acceptance criteria required by the Construction Code. In addition, the inspectors reviewed the welding procedure specifications, welder qualifications, welding material certifications, and supporting weld procedure qualification records to evaluate if the weld procedures were qualified in accordance with the requirements of the Construction Code and the ASME Code Section IX.

- Work Order 59102301193, Installation of full structural weld overlay on SG B hot leg safe-end.

PWR Vessel Upper Head Penetration (VUHP) Inspection Activities: The licensee performed ultrasonic examinations of the reactor pressure vessel (RPV) head penetrations in accordance with the requirements of 10 CFR 50.55a(g)(4)(ii)(D) and performed the examinations in accordance with ASME Code Case 729-1.

The inspectors observed the examinations performed on RPV head penetration number 47, and reviewed procedures, personnel qualifications and certifications. The licensee did not perform any bare metal visual inspections or repairs on the RPVUHP this outage.

Boric Acid Corrosion Control (BACC) Inspection Activities: The inspectors reviewed the licensee's BACC program activities to ensure implementation with commitments made in response to NRC Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary," and applicable industry guidance documents. Specifically, the inspectors performed an on-site record review of procedures and the results of the licensee's containment walkdown inspections performed during the current spring refueling outage. The inspectors also interviewed the BACC program owner, conducted an independent walkdown of containment to evaluate compliance with licensee's BACC program requirements, and verified that degraded or non-conforming conditions, such as boric acid leaks, were properly identified and corrected in accordance with the licensee's BACC and corrective action programs.

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

- CR465640 – Excessive inactive boric acid packing leak on 1 SI-MOV-1869B

Steam Generator Inspection Activities: The inspectors reviewed the licensee's 2012 Steam Generator Condition Monitoring Assessment and the 2011 Steam Generator Condition and Operating Assessment for North Anna Unit 1. No steam generator tube inspection activities occurred during this outage.

Identification and Resolution of Problems: The inspectors reviewed a sample of ISI-related issues that were identified by the licensee and entered into the corrective action program as condition reports (CRs). The inspectors reviewed the CRs to confirm the licensee had appropriately described the scope of the problem and had initiated corrective actions. The review also included the licensee's consideration and assessment of operating experience events applicable to the plant. The inspectors performed this review to ensure compliance with 10CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requirements. The corrective action documents reviewed by the inspectors are listed in the Attachment.

b. Findings

.1 Missed PWSCC indications on SG safe-end weld

Introduction: A self revealing non-cited violation (NCV) of the required augmented ISI examinations identified in 10 CFR 50.55a(g)(6)(ii)(F), Examination Requirements for Class 1 Piping and Nozzle Dissimilar Metal Butt Welds, was identified for the licensee's failure to identify unacceptable PWSCC indications in the Unit 1 B SG hot leg nozzle safe-end weld. This regulation requires examinations be conducted in accordance with ASME Code Case N-770-01, which covers alternative examination requirements and acceptance standards for Class 1 PWR piping and vessel butt welds fabricated with Alloy 82 and 182 filler material. This regulation requires in-service examinations to be performed using qualified techniques and with qualified personnel capable of identifying PWSCC indications.

Description: During the Spring Unit 1 refueling outage, after machining was performed in preparation for a full structural weld overlay that was being installed as mitigation to PWSCC, two through-wall indications were identified in the Unit 1 B SG hot leg safe-end-to-nozzle weld. The machining had reduced the wall thickness down to approximately 80% of its original thickness and exposed the indications to the surface. These through-wall indications were identified by workers who observed water seeping in localized areas on the surface of the safe-end. An ultrasonic examination qualified to identify PWSCC was performed on this safe-end weld just prior to the machining. This safe-end weld had also been examined for PWSCC with the same UT procedure in 2009.

The licensee entered this issue into their corrective action program with CR 467649 and took corrective actions which included performing a root cause evaluation (RCE), an additional UT examination to identify the extent of the indications, and a review of the safe-end fabrication records. These corrective actions determined that the extent of the indications included two 100% through-wall and three partial through-wall indications, that the indications exceeded the acceptance criteria of ASME Section XI, that the indications were PWSCC, that the indications were in the vicinity of repairs that had been performed during fabrication of the safe-end, and that the indications should have been identified by previous UT examinations performed. The licensee also performed a qualified examination using phased array UT on the other hot leg safe-end weld. The licensee also removed portions of the cracks to facilitate the installation of the full structural weld overlay.

The results of the RCE were presented in report RCE 001078. The evaluation determined that the root causes included that the site NDE organization did not adequately implement their responsibility to ensure effective application of the examination procedure by supplemental examination personnel and that the on-site briefing conducted with NDE technicians was not adequate to ensure successful execution of the examination.

The NRC staff conducted an evaluation to determine the growth of the indications, to determine if the indications could have existed in 2009, and determine if the indications could have grown to an unacceptable size during machining conducted during the preparation for the installation of the full structural weld overlay. The staff issued technical position paper, NRC Staff Technical Position for North Anna Hot Leg to Steam Generator Nozzle Flaws (ML 12202B154) that determined that it was likely that the indications were present in 2009 and should have been detected by UT examination, that for a period of as much as the last 4 months between 2009 and 2012 the indications exceeded Section XI acceptance criteria, and that the machining process did not cause the unacceptable indications to propagate to the safe-end surface.

Based on the above information, the inspectors determined that the indications should have been identified during UT examinations conducted in 2009, resulting in the plant being operated in a nonconforming condition for a period of up to the last 4 months between 2009 and 2012, that the examinations performed on SG safe-end weld did not provide assurance that the structural boundary of the reactor coolant system remains capable of performing its intended safety function.

Analysis: The inspectors determined that the failure to identify the PWSCC indications in the Unit 1 B SG hot leg safe-end weld was a self-revealing performance deficiency that was within the licensee's ability to foresee and correct. The inspectors determined that this finding was of more than minor significance because the failure to identify the PWSCC could have resulted in the potential to allow degradation of the safe-end to proceed undetected. Unchecked PWSCC degradation could have resulted in more significant degradation of the safe-end weld with subsequent degradation of the primary system pressure boundary. The finding was associated with the design control attribute of the Barrier Integrity Cornerstone and affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, examinations of the SG safe-end welds provide assurance that the structural boundary of the reactor coolant system remains capable of performing its intended safety function. The inspectors used IMC 0609, "Significance Determination Process," Attachment 0609.04, "Initial Characterization of Findings," and determined that the finding was of low safety significance (Green) because it did not represent an actual failure of the safe-end pressure retaining boundary.

The inspectors reviewed this performance deficiency for cross-cutting aspects as required by IMC 0310, "Components With Cross-Cutting Aspects." The inspectors identified a cross-cutting aspect in the Human Performance Work Practices cross cutting area, H.4 (c). Specifically, the licensee failed to provide adequate supervision and conduct an adequate briefing with NDE technicians prior to the examination to ensure its successful execution.

Enforcement: 10 CFR 50.55a(g)(6)(ii)(F), Examination Requirements for Class 1 Piping and Nozzle Dissimilar Metal Butt Welds, requires examinations of vessel butt welds fabricated with Alloy 82 and 182 filler material be conducted in accordance with ASME Code Case 770-01, which covers alternative examination requirements and acceptance standards for Class 1 PWR piping, vessel butt welds fabricated with Alloy 82, and 182 filler material. This regulation requires in-service examinations to be performed using qualified techniques and with qualified personnel capable of identifying PWSCC indications. Contrary to the above, the licensee failed to identify PWSCC in the B SG hot leg safe-end weld. Because this finding is of very low safety significance and has been entered into the licensee's corrective action program as condition report CR467649, this violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000338/2012003-01, Failure to identify PWSCC in the unit 1 B SG hot leg safe-end weld.

.2 Examination of SG safe-end weld with possible unqualified ultrasonic examination procedures

Introduction: The inspectors identified an unresolved item (URI) concerning the qualification of the licensee's UT procedure, UT probes meeting procedural requirements and the adequacy of the site specific mock-ups.

Description: The licensee qualified the manual UT procedure in accordance with the EPRI Performance Demonstration Initiative (PDI) process utilizing the PDI procedure IR-2009-358, for site specific qualification. This process allows for qualification that utilizes a site specific mock-up in an open demonstration process. Although this qualification is used by the industry in meeting the requirements of Appendix VIII of Section XI, there are concerns with the inconsistency with respect to the application of robust, blind demonstration approaches versus less rigorous, and open qualifications. This issue was highlighted as a result of the missed indications at North Anna. The licensee requested that EPRI review (TJ) IR-2009-358 to reassess the current validity of the information provided within this document. With respect to the open demonstration process, EPRI has determined the stated position (of using an open demonstration process) in (TJ) IR-2009-358 to continue to be acceptable, which is inconsistent with their approach to the use of Code Case N-770, where EPRI requires that a blind demonstration test be passed. This inconsistency needs to be further discussed and a path forward defined in order to develop guidance for application during either type of performance demonstration.

In addition, the inspectors and members of the NRR staff conducted on-site evaluations of the site specific UT procedure and inspection technique. This evaluation was conducted on the site specific calibration blocks and with the same UT probes that the licensee used to qualify the UT procedure and the qualification of the NDE technicians. Subsequently, the NRC staff requested Pacific Northwest National Laboratory (PNNL) to evaluate the qualification of the manual UT procedure that was used to examine the safe-end weld. The results were presented in PNNL Report PNNL-21546, Evaluation of Manual Ultrasonic Examinations Applied to Detect Flaws in Primary System Dissimilar Metal Welds at North Anna Power Station, (ML12200A216). This report determined that the site specific approach for the manual UT technique does not meet the intent of the requirements of Appendix VIII of ASME Section XI. Also identified was that the probes used to conduct the site examinations did not meet the procedure requirements of licensee procedure ER-AA-NDE-180 for UT probe angles.

This issue remains unresolved until questions associated with the qualification of the UT procedure, including that the probes that did not meet the procedural requirements for UT probe angles and the adequacy of the site specific mock-ups are resolved. This issue is identified as URI 05000338/2012003-02, Examination of SG safe-end weld with possible unqualified ultrasonic examination procedures.

1R11 Licensed Operator Requalification Program

.1 Resident Inspector Quarterly Review

a. Inspection Scope

The inspectors reviewed a licensed operator training session May 9, 2012, which involved a reactor shutdown with a failure of safety related nuclear instruments and spent fuel pool (SFP) low level alarm due to SFP cooling pump seal leak. The scenario required classifications 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 instructor and reviewed with the operators.

b. Findings

No findings were identified.

.2 Operator Observations

a. Inspection Scope

During the inspection period, the inspectors conducted observations of licensed reactor operators actions and activities during the ramp of Unit 1 from 100% power to 90% and return to 100% to conduct turbine valve freedom testing on May 23, 2012, to ensure that the activities were consistent with the licensee procedures and regulatory requirements. These observations took place during both normal and off-normal plant working hours. As part of this assessment, the inspectors observed the following elements of operator performance: (1) operator compliance and use of plant procedures including technical specifications; (2) control board/in-plant component manipulations; (3) use and interpretation of plant instruments, indicators and alarms; (4) documentation of activities; (5) management and supervision of activities; and, (6) communication between crew members.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope

For the three equipment issues listed below, the inspectors evaluated the effectiveness of the respective licensee's preventive and corrective maintenance. The inspectors performed walkdowns of the accessible portions of the systems, performed in-office reviews of procedures and evaluations, and held discussions with licensee staff. The inspectors compared the licensee's actions with the requirements of the Maintenance Rule (10 CFR 50.65), and licensee procedure ER-AA-MRL-10, "Maintenance Rule Program," Revision 5.

- Work Order 59102463766, "Replace Signal Cable with Shielded Cable IAW DC NA-11-01183," on 1J EDG Governor
- MRE015062, "1-SI-MOV-19865C did not open as expected"
- 1-SI-LT-1928 Level Oscillation

b. Findings

No findings were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

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

- Emergent work on 1-RC-MOV-1595
- FME and recovery from 1-RC-MOV-1595
- Leakage on 1-CH-MOV-115B
- Online risk changes due to failure of Unit 1 'A' condensate pump

b. Findings

No findings were identified.

1R15 Operability Determinations and Functionality Assessments

a. Inspection Scope

The inspectors reviewed four operability determinations and functionality assessments, listed below, affecting risk-significant mitigating systems, to assess, as appropriate: (1) the technical adequacy of the evaluations; (2) whether continued system operability was warranted; (3) whether other existing degraded conditions were considered as compensating measures; (4) whether the compensatory measures, if involved, were in place, would work as intended, and were appropriately controlled; and (5) where continued operability was considered unjustified, the impact on TS Limiting Conditions for Operation and the risk significance in accordance with the Significant Determination Process (SDP). The inspectors' review included a verification that operability determinations (OD) were made as specified by procedure OP-AA-102, "Operability Determination," Revision 7.

- OD 000477, "Explain cause of 1J EDG load swings"
- CR471315, "Valve internals broken and missing" on 1-RC-MOV-1595
- OD 000483, "OD to Engineering to 1J EDG broken guard"
- CR4688654, "1J EDG duplex filter fuel oil leak"

b. Findings

No findings were identified.

1R18 Plant Modifications

.1 Temporary Modifications

a. Inspection Scope

The inspectors reviewed the temporary modification for recovery equipment for a stuck fuel assembly rod to verify that the modification did not affect systems operability or availability as described by the TS and UFSAR. In addition, the inspectors verified that the temporary modification was in accordance with VPAP-1403, "Temporary Modifications," Revision 13, and the related work package, that adequate controls were in place, procedures and drawings were updated, and post-installation tests verified the operability of the affected systems. Documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2 Permanent Modifications

a. Inspection Scope

The inspectors reviewed the three completed permanent plant modification design change packages listed below. The inspectors conducted a walkdown of the installations, discussed the desired improvements with system engineers, and reviewed the 10 CFR 50.59 Safety Review/Regulatory Screening, technical drawings, test plans and the modification packages to assess the TS implications.

- Design Change NA-11-01031, “EDG Day Tank Missile Protection Modification”
- Design Change NA-11-01183, “Emergency Diesel Generator Governor Control Replacement (1H, 1J, 2H, and 2J),” Revision 4
- Design Change NA-11-01213, “Seismic Monitoring Instrumentation Upgrade"

b. Findings

No findings were identified.

1R19 Post Maintenance Testing

a. Inspection Scope

The inspectors reviewed five post maintenance test procedures and/or test activities for selected risk-significant mitigating systems listed below, to assess whether: (1) the effect of testing on the plant had been adequately addressed by control room and/or engineering personnel; (2) testing was adequate for the maintenance performed; (3) acceptance criteria were clear and adequately demonstrated operational readiness consistent with design and licensing basis documents; (4) test instrumentation had current calibrations, range, and accuracy consistent with the application; (5) tests were performed as written with applicable prerequisites satisfied; (6) jumpers installed or leads lifted were properly controlled; (7) test equipment was removed following testing; and (8) equipment was returned to the status required to perform in accordance with VPAP-2003, “Post Maintenance Testing Program,” Revision 14. Other documents reviewed are listed in the Attachment.

- 1-PT-82.4A, “1H Diesel Generator Test (Start by ESF Actuation),” Revision 45
- 1-PT-82.12J, “1J Diesel Generator Isochronous Mode (Start by ESF Actuation),” Revision 33
- 1-PT-83.12J, “1J Diesel Generator Test (Start by ESF Actuation) Followed by 24-hour Run and Hot Restart Test,” Revision 21
- WO 59102448226, “01-MS-TV-111A Valve to be disassembled to support valve replacement, valve leaking by
- WO 59102287003, “Uncouple/Recouple Motor to Speed Increaser and Align”

b. Findings

No findings were identified.

1R20 Refueling and Other Outage Activities

a. Inspection Scope

The inspectors reviewed the Outage Safety Review (OSR) and contingency plans for the Unit 1 refueling outage, which began March 11, 2012, to confirm that the licensee had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense-in-depth. The inspectors used NRC inspection procedure 71111.20, "Refueling and Outage Activities," to observe portions of the shutdown, cooldown, refueling, maintenance activities, and startup activities to verify that the licensee maintained defense-in-depth commensurate with the outage risk plan and applicable TS. The inspectors monitored licensee controls over the outage activities listed below.

- Licensee configuration management, including daily outage reports, to evaluate maintenance of defense-in-depth commensurate with the OSR for key safety functions and compliance with the applicable TS when taking equipment out of service.
- Implementation of clearance activities and confirmation that tags were properly hung and equipment appropriately configured to safely support the work or testing.
- Installation and configuration of reactor coolant pressure, level, and temperature instruments to provide accurate indication and an accounting for instrument error.
- Controls over the status and configuration of electrical systems to ensure that TS and outage safety plan requirements were met, and controls over switchyard activities.
- Monitoring of decay heat removal.
- Controls to ensure that outage work was not impacting the ability of the operators to operate the spent fuel pool cooling system.
- Reactor water inventory controls including flow paths, configurations, and alternative means for inventory addition, and controls to prevent inventory loss.
- Controls over activities that could affect reactivity.
- Refueling activities, including fuel handling and sipping to detect fuel assembly.
- Startup and ascension to full power operation, tracking of startup prerequisities, walkdown of the drywell (primary containment) to verify that debris had not been left which could block emergency core cooling system strainers, and reactor physics testing.
- Licensee identification and resolution of problems related to refueling outage activities.

b. Findings

No findings were identified.

1R22 Surveillance Testing

a. Inspection Scope

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

RCS Leakage:

- 1-PT-52.2, “Reactor Coolant System Leak Rate (Hand Calculation),” Revision 38, and 1-PT-52.2A, “Reactor Coolant System Leak rate (Computer Calculation),” Revision 31

Other Surveillance Tests:

- 0-PT-82.13, “Quarterly Test of 0-AAC-DG-OM Alternate AC Diesel Generator (SBO Diesel) on F Transfer Bus,” Revision 23
- 2-PT-34.3, “Turbine Valve Freedom Test,” Revision 43

Containment Isolation Valve:

- 1-PT-61.3, “Containment Type C Testing,” Revision 33, and 1-PT-61.3, “Containment Type C Test Attachment 66, Valve Penetration Diagram – Penetration 105B”

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

a. Inspection Scope

The inspectors performed a periodic review of the following two PIs for Unit 1 and Unit 2 to assess the accuracy and completeness of the submitted data and whether the performance indicators were calculated in accordance with the guidance contained in NEI 99-02, “Regulatory Assessment Performance Indicator Guideline,” Revision 6. The inspection was conducted in accordance with NRC inspection procedure 71151, “Performance Indicator Verification.” Specifically, the inspectors reviewed the Unit 1 and Unit 2 data reported to the NRC for the period April 1, 2011, through March 31, 2012.

Documents reviewed included applicable NRC inspection reports, licensee event reports, operator logs, station performance indicators, and related CRs.

- Reactor Coolant System (RCS) Leakage
- RCS Specific Activity

b. Findings

No findings were identified.

4OA2 Problem Identification and Resolution

.1 Review of Items Entered into the Corrective Action Program

As required by NRC inspection procedure 71152, “Identification and Resolution of Problems,” and in order to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee’s CAP. This review was accomplished by reviewing daily condition record (CR) report summaries and periodically attending daily CR Review Team meetings.

.2 Annual Sample: Review of CR335031, Emergency Diesel Generator Fuel Oil Day Tank Vent Missile Protection

a. Inspection Scope

The inspectors performed a review regarding the licensee’s assessments and corrective actions for CR335031, “Emergency Diesel Generator Fuel Oil Day Tank Vent Missile Protection,” to ensure that the full extent of the issue was identified, an appropriate evaluation was performed, and appropriate corrective actions were specified and prioritized. The inspectors also evaluated the CR against the requirements of the licensee’s CAP as specified in procedure, PI-AA-200, “Corrective Action Program,” Revision 19 and 10 CFR 50, Appendix B.

b. Findings and Observations

No findings of significance were identified. In general, the inspectors verified that the licensee had identified problems at an appropriate threshold and entered them into the CAP database, and had proposed or implemented appropriate corrective actions.

4OA3 Event Followup

.1 (Closed) Licensee Event Report (LER) 05000339/2011-001-00: Inoperable Emergency Diesel Generator Due to Coolant Leak

On August 23, 2011, with both Units 1 and 2 offline as a result of the Central Virginia earthquake, the 2H EDG was manually tripped due to a coolant system leak forty nine minutes after its demand start due to a loss of offsite power. The 1H and 1J EDGs and the 2J EDG continued to operate after the 2H EDG was manually tripped. The alternate AC diesel generator was aligned to re-energize the 2H emergency bus. The direct cause of the manual tripping of the 2H EDG was a failed gasket that caused a coolant leak. Maintenance procedures did not provide an adequate level of detailed instructions on proper installation of the gasket between the exhaust belt and the coolant inlet bypass fitting. Specifically the procedures lacked guidance on adhesive cure times and details regarding how to tighten the adjusting fastener without impacting the gasket joint compression.

The licensee entered this problem in their CAP as CR439091. All exhaust belt triangles on all four diesels were inspected to identify any mis-positioned or extruded gaskets. The enforcement action of this issue is as discussed in NRC Inspection Report 05000338/2012010. This LER is closed.

.2 (Closed) Licensee Event Report (LER) 05000338, 339/2011-003-00: Dual Unit Reactor Trip and ESF Actuations During Seismic Event with a Loss of Offsite Power

On August 23, 2011, at 1351 hours, with both units in Mode 1 at 100% power, a magnitude 5.8 earthquake occurred approximately 11 miles WSW of the North Anna Power Station. The earthquake caused an automatic reactor trip of both units and a loss of offsite power. The station response to the event focused on stabilizing the units and restoring power to affected buses. Once immediate correction actions were performed, plant walkdowns and inspections were conducted. These inspections did not identify any significant physical or functional damage to safety-related plant systems, structures and components (SSCs), and only limited damage to non-safety related, non-seismically designed SSCs.

The licensee entered this problem in their CAP as CR439052. Additional inspection aspects and safety analyses of this event were discussed in NRC Inspection Reports 05000338, 339/2011011, 05000338, 339/2011012 and 05000338, 339/2011013. This LER is closed.

4OA5 Other Activities

.1 Quarterly Resident Inspector Observations of Security Personnel and Activities

a. Inspection Scope

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

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

b. Findings

No findings were identified.

.2 (Closed) Unresolved Item (URI) 05000338/2011012-01: Unsealed Penetration on Unit 1 Motor Driven Auxiliary Feedwater Pump Room Equipment

a. Inspection Scope

As described in URI 05000338/2011012-01, discussions were held between NRC inspectors and Dominion engineers regarding the high energy line break (HELB) analysis for the AFW pipe tunnels that connected the main steam valve house and the AFW pump rooms as well as the potential impact the seismic event may have had on the condition of the tunnel structure. A physical inspection of the tunnel was conducted by NAPS personnel on October 18, 2011. While no structural issues related to the seismic event were noted in the tunnel, Dominion personnel did identify issues in the Unit 1 pipe tunnel. The inspectors interviewed station personnel and performed an extensive review of the licensee's procedures, documents associated with the unsealed penetration and corrective action program documents regarding this issue. This URI is closed.

b. Findings

Following the identification of the unsealed penetration into the Motor Driven Auxiliary Feedwater (MDAFW) pump room, additional engineering analysis was performed by the licensee to determine what the impact of steam entering the room would have had on the installed equipment through the use of a more detailed analysis of the conditions that would exist following a crack in the steam line. The licensee developed and implemented a design change to seal the opening and prevent steam from entering the MDAFW pump room in the event of a main steam line break within the pipe tunnel. The material used to seal the penetration provides both environmental protection from the steam as well as being unaffected by the effects of a seismic event. The licensee's analysis and updated engineering technical evaluation was reviewed by the NRC once completed.

The inspectors reviewed CR 448893, "Missing Seal in AFW Tunnel Opening to U1 MDAFWP Room," and CR 449171, "Unit 1 AFW pipe tunnel inspection required to answer NRC HELB question." This involved a design control issue regarding space configuration where the space should have been sealed completely. This issue was associated with the design control attribute of the Mitigating Systems Cornerstone, but was not more than minor because it did not affect the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences.

4OA6 Meetings, Including Exit

.1 Exit Meeting Summary

On July 31, 2012, the resident inspectors presented the inspection results to Mr. G. Bischof and other members of the staff, who acknowledged the findings. The inspectors asked the licensee whether any of the material examined during the inspection should be considered proprietary. No proprietary information was identified.

.2 Annual Assessment Meeting Summary

On May 1, 2012, the NRC's Chief of Reactor Projects Branch 5 and the Resident Inspectors assigned to the North Anna Power Station met with Virginia Electric and Power Company to discuss the NRC's Reactor Oversight Process and the NRC's annual assessment of North Anna's safety performance for the period of January 1 through December 31, 2011. The major topics addressed were the NRC's assessment program, and the results of the North Anna Power Station assessment. Attendees included North Anna site management, members of the site staff, journalists from several local TV stations, a representative from the office of one of Virginia's State Senators, two members from the local county's Board of Supervisors, and seven members of the public.

This meeting was open to the public. The presentation material used for the discussion and the list of attendees is available from the NRC's document system (ADAMS) as accession number ML081200874. ADAMS is accessible from the NRC Web site at http://www/nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

ATTACHMENT: SUPPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel:
M. Becker, Manager, Nuclear Outage and Planning
G. Bischof, Site Vice President
M. Crist, Plant Manager
J. Daugherty, Manager, Nuclear Maintenance
R. Evans, Manager, Radiological Protection
B. Gaspar, Manager, Nuclear Site Services
C. Gum, Manager, Nuclear Protection Services
E. Hendrixson, Director, Nuclear Engineering
S. Hughes, Manager, Nuclear Operations
P. Kemp, Project Manager, Station Improvement Initiatives
J. Leberstien, Technical Advisor Licensing
F. Mladen, Director, Nuclear Safety and Licensing
J. Plossl, Supervisor, Nuclear Station Procedures
J. Schleser, Manager, Nuclear Organizational Effectiveness
D. Taylor, Supervisor, Station Licensing
R. Wesley, Manager, Nuclear Training
M. Whalen, Technical Advisor Licensing

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened and Closed

05000338/2012003-01	NCV	Failure to identify PWSCC in the Unit 1 B SG hot leg safe-end weld (Section 1R08)

Opened

05000338/2012003-02	URI	Examination of SG safe-end weld with possible unqualified ultrasonic examination procedures (Section 1R08)

Closed

05000339/2011-001-00	LER	Inoperable Emergency Diesel Generator Due to Coolant Leak (Section 4OA3.1)
05000338, 339/2011-003-00	LER	Dual Unit Reactor Trip and ESF Actuations During Seismic Event with a Loss of Offsite Power (Section 4OA3.2)
05000338/2011012-01	URI	Unsealed Penetration on Unit 1 Motor Driven Auxiliary Feedwater Pump Room Equipment (Section 4OA5.2)

Discussed
None

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

0-AP-4.1, "Severe Weather Conditions," Revision 54
DC 06-139, "Missile Shield Extension for Auxiliary Feedwater Pump House Ventilation Intake North Anna Power Station/Units 1 and 2," 07-14-2006
NAPS UFSAR, Table 3.2-1, "Structures, Systems, and Components that are Designed to Seismic and Tornado Criteria," Revision 47
NAPS UFSAR, Chapter 3.3, "Wind and Tornado Design Criteria," Revision 47
TN-32 Dry Storage cask Topical Safety Analysis Report
NUHOMS-HD Horizontal Modular Storage System for Irradiated Nuclear Fuel UFSAR, Rev 3

Section 1R04: Equipment Alignment

NAPS UFSAR, Chapter 10.4.3, "Condensate and Feedwater Systems," Revision 47
Technical Specifications for North Anna Power Station Units 1 and 2, Section 3.7.5 "Auxiliary Feedwater (AFW) System"
NUREG/CR-6837 (ML 082540184), "Auxiliary Feedwater System Risk-Based Inspection Guide for the North Anna Nuclear Power Plants"
DWG 11715-FM-074A, "Flow/Valve Operating Numbers Diagram Feedwater System North Anna Power Station Unit 1 Virginia Power," Sheet 3, Revision 43
DWG 11715-FC-12J, "Pipe Tunnel From Quench Spray Pump House to Auxiliary Feedwater Pump House," Sheet 1, Revision 5
DWG 11715-FC-12K, "Pipe Tunnel From Quench Spray Pump House to Auxiliary Feedwater Pump House," Sheet 1, Revision 5
DWG 11715-MSK-107G1, "Pipe Stress Summary Sheet Yard Piping – North Reactor Containment – Sheet 1 North Anna Power Station - Unit 1," Revision 7
DWG 11715-MSK-107G2, "Pipe Stress Summary Sheet Yard Piping – North Reactor Containment – Sheet 2 North Anna Power Station - Unit 1," Revision 6
1-OP-31.2A, "Valve Checkoff-Auxiliary Feedwater," Revision 23
ER-NA-PFM-314, "Inspection of Auxiliary Feedwater Tunnel Piping Supports," Revision 0

Section 1R05: Fire Protection

1-FS-S-2, "Fire Fighting Plan for Cable Vault and Tunnel and 280' Rod Drive Unit 1 Safe Shutdown Equipment," Revision11
2-FS-S-2, "Fire Fighting Plan for Cable Vault and Tunnel and 280' Rod Drive Unit 2 Safe Shutdown Equipment," Revision10
1-FS-SW-1, "Service Water Pumphouse Unit 1 and 2," Revision1
NAPS Appendix R Report, Revision 30

Section 1R08: Inservice Inspection Activities

Procedures
EP-AP-BAC-10, Rev. 8, Boric Acid Corrosion Control Program
ER-AP-BAC-102, Rev. 9, Boric Acid Corrosion Control Program Evaluation
ER-AP-BAC-101, Rev. 7, Boric Acid Corrosion Control Program Inspections

ER-AA-NDE-UT-810, Rev. 2, Ultrasonic Examination of Dissimilar Metal Welds in Accordance with ASME Section XI, Appendix VIII.

Corrective Action Documents
CR465638 – 1-SI-MOV-1890 body to flange leak
CR465640 – Excessive inactive boric acid packing leak on 1-SI-MOV-1869
CR466576 – Dry boric acid on bolting of 1-SI-86
CR465931 – Boric Acid on main flange of 1-P-1A, B, C

Other
First quarter Boric Acid Program Health Report
Drawing 11715-WMKS-RC-E-1A.2P – Inservice Inspection detail drawing Steam Generator Detail and Sections Unit 1
Design Change NA-11-01047/DCU-3041 – Full Structural Weld Overlay
Ultrasonic data sheets for B SG hot leg safe-end 11715-WMKS-RC-E-1C.2/29-RC-7/N-SE2
Welding procedure specification WPS-08-43-S-001-204321
Welding procedure specification WPS-03-03-S-602-204321

Section 1R18: Plant Modifications

Westinghouse DWG10004358, “Stud Hole Mount,” Revision 0
Westinghouse DWG10004359, “Stud Hole Mount Delivery Tool,” Sheets 1-2, Revision 0
Westinghouse DWG10004360, “Buoyancy Compensator,” Sheets 1-4, Revision 0
Westinghouse DWG10004361, “Fuel Assembly Support Tool,” Sheets 1-9, Revision 0
Westinghouse DWG10004362, “Fuel Assembly Dislodging Tool,” Sheets 1-2, Revision 0
Westinghouse DWG10004362, “North Anna Fuel Assembly Support Tool Template,” Sheets 1-9, Rev 0
Westinghouse letter dated March 30, 2012, “North Anna Unit 1 Hanging Fuel Assembly Recovery Upper Core Plate and Hold-down Spring Assessment”
Westinghouse letter dated March 30, 2012, “Assessment of the Impact to the Reactor Vessel Due to the Use of a Fuel Assembly Catching Device During Removal of a Stuck Fuel Assembly”
Westinghouse Calculation CN-RIDA-12-26, “Impact Loading of Fuel Assembly on Support Tool for North Anna”
Westinghouse Procedure STD-QP-1998-8189, “Anchors and Special Fuel Assembly Cruciform Handling Tool Qualification Procedure,” Rev 0
Westinghouse Procedure NSD-SVS-TOP-845, “Installation and Operation of the Stud Hole Mount and Template for Stuck Fuel Assembly at North Anna 1,” Rev 0
Westinghouse Procedure NSD-SVS-TOP-847, “Installation and Operation of the Fuel Assembly Support Tool and Dislodging Tool at North Anna 1,” Rev 0
Westinghouse Procedure NSD-ENG-QP-846, “Qualification Procedure for North Anna Unit 1 Stuck Fuel Assembly Support Tool and Delivery Tool,” Rev 0
Westinghouse Specification NSD-ENG-FS-842, “Functional Specification for North Anna Unit 1 Stuck Fuel Assembly Recovery Tooling,” Rev 0
Westinghouse Data Package NSD-ENG-QP-843-DP, “Stud Hole Mount and Stud Hole Mount Delivery Tool Data Package,” Rev 0
Westinghouse Data Package NSD-ENG-QP-844-DP, “Restraint Tool Template Data Package,” Rev 0
Westinghouse Data Package NSD-ENG-QP-848-DP, “Dislodge Tool Data Package,” Rev 0

ETE-NA-2012-0024, “Fuel Assembly Capture Device, Dislodge Tool, and Cruciform Handling Tool Evaluation”
ODM000249, “Fuel Assembly 0X5 Stuck to Upper Internals During Lift,” Rev 1
0-OP-4.59, “Operation of the Special Fuel Assembly Cruciform Handling Tool (SFACHT),” Rev 0

<u>Section 1R19: Post Maintenance Testing</u>

ACE019087, “Lower Seat Ring Area and Gasket Found Damaged on 1-MS-TV-111A,” Rev 0
OD000461, “Complete OD for 1-MS-TV-111B leakby through valve seat”
WO 59102287003, “02-CH-P-1B Gear Uncouple/Recouple Motor to Speed Increaser & Align

<u>1R22: Surveillance Testing</u>
1-PT-61.3, “Containment Type C Test,” Revision 33
1-PT-61.3 (2012-04-29) (SD-5KTBO3Y-CGD-CP), “Containment Type C Test,” Revision 33, Attachment 66, “Valve Penetration Diagram – Penetration 105B”
1-PT-52.2A (LFF5GO-CGD-CP), “Reactor Coolant System Leak Rate (Computer Calculation),” Rev 31
1-PT-52.2 (SD-57FNPSC-CGD-CP), “Reactor Coolant System Leak Rate (Hand Calculation,), Rev 38

LIST OF ACRONYMS

ADAMS	Agencywide Document Access and Management System
CA	Corrective Action
CAP	Corrective Action Program
CFR	Code of Federal Regulations
CR	Condition Report
EDG	Emergency Diesel Generator
IMC	Inspection Manual Chapter
JPM	Job Performance Measures
LHSI	Low Head Safety Injection
NCV	Non-cited Violation
NRC	Nuclear Regulatory Commission
OD	Operability Determination
PARS	Publicly Available Records
PI	Performance Indicator
QS	Quench Spray
RCE	Root Cause Evaluation
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RTP	Rated Thermal Power
SDP	Significance Determination Process
SR	Surveillance Requirements
TDAFWP	Turbine Driven Auxiliary Feedwater Pump
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
VEPCO	Virginia Electric and Power Company
VPAP	Virginia Power Administrative Procedure
WO	Work Order