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

June 2, 2017

Mr Daniel G. Stoddard
President and Chief Nuclear Officer
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060

SUBJECT: NORTH ANNA POWER STATION - NRC EVALUATION OF CHANGES, TESTS, AND EXPERIMENTS INSPECTION REPORT 05000338/2017007 AND 05000339/2017007

Dear Mr. Stoddard:

On April 21, 2017, the U.S. Nuclear Regulatory Commission completed an inspection at your North Anna Power Station Units 1 and 2, and discussed the results of this inspection with Mr. Lane and other members of your staff. Inspectors documented the results of this inspection in the enclosed inspection report.

NRC inspectors documented two findings of very low safety significance (Green) that involved violations of NRC requirements and one Severity Level IV traditional enforcement violation with no associated finding in this report. The NRC is treating these violations as non-cited violations consistent with Section 2.3.2.a of the Enforcement Policy.

If you contest the violations or significance of these non-cited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region II; the Director, Office of Enforcement; and the NRC resident inspector at the North Anna Power Station.

This letter, its enclosure, and your response (if any) will be made available for public inspection and copying at http://www.nrc.gov/reading-rm/adams.html and at the NRC Public Document Room in accordance with 10 CFR 2.390, "Public Inspections, Exemptions, Requests for Withholding."

Sincerely,

/RA/

Jonathan H. Bartley, Chief
Engineering Branch 1
Division of Reactor Safety

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

Enclosure:
Inspection Report 05000338/2017007
and 05000339/2017007 w/Attachment:
Supplementary Information

cc: Distribution via Listserv

SUBJECT: NORTH ANNA POWER STATION - NRC EVALUATION OF CHANGES, TESTS, AND EXPERIMENTS INSPECTION REPORT 05000338/2017007 AND 05000339/2017007 dated June 2, 2017

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

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-338, 50-339
License Nos.:	NPF-4, NPF-7
Report Nos.:	05000338/2017007, 05000339/2017007
Licensee:	Virginia Electric and Power Company (VEPCO)
Facility:	North Anna Power Station, Units 1 and 2
Location:	Mineral, Virginia 23117
Dates:	April 17, 2017 - April 21, 2017
Inspectors:	T. Fanelli, Senior Reactor Inspector (Team Leader) M. Greenleaf, Reactor Inspector C. Stott, Reactor Inspector (RIV) W. Satterfield, Reactor Inspector (Trainee)
Approved by:	Jonathan Bartley, Chief Engineering Branch 1 Division of Reactor Safety

Enclosure

SUMMARY

Inspection Report (IR) 05000338/2017007, 05000339/2017007; 04/17/2017-04/21/2017; North Anna Power Station, Units 1 and 2; Evaluations of Changes, Tests, and Experiments.

This report covers a one-week onsite inspection by an NRC team of three inspectors and one inspector trainee. The team identified two Green non-cited violations (NCVs) and one Severity Level IV (SL IV) traditional enforcement violation. The significance of inspection findings are indicated by their color (i.e., greater than Green, or Green, White, Yellow, Red) and determined using IMC 0609, “Significance Determination Process” dated April 29, 2015. Cross-cutting aspects are determined using IMC 0310, “Components Within the Cross Cutting Areas” dated December 4, 2014. All violations of NRC requirements are dispositioned in accordance with the NRC’s Enforcement Policy dated November 1, 2016. The NRC’s program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 6.

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

Cornerstone: Barrier Integrity

- <u>Green</u>: The NRC identified a Green non-cited violation of 10 CFR 50.49(f) for failing to qualify structures, systems, and components (SSCs) (eight motor control centers) located in a radiation harsh environment in accordance with IEEE Std. 323-1974 Section 5, “Principles of Qualification.” In response to this issue, the licensee performed an operability determination and determined that the motor control centers (MCCs) were operable based on the material similarity of the original SSCs and the new SSCs. This issue has been entered into the corrective action program as CR 1065894

  The performance deficiency was determined to be more than minor because it was associated with the Equipment Performance attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of the safety related AC Power System. Specifically, the failure to perform environmental qualification for SSCs subject to a harsh environment, during which they must perform a safety function adversely affected the reliability of that equipment when called upon. This finding was not assigned a cross-cutting aspect because the issue did not reflect current licensee performance. (Section 1R17.b.1)

- <u>Green</u>: The NRC identified a Green non-cited violation of 10 CFR 50.49(e)(5) for failing to base the qualified life of structures, systems, and components (SSCs) (i.e. Nordel O-rings) on the known limits of extrapolation in accordance with IEEE 323 Sections 6.5.3, “Extrapolation,” and 6.5.4 “Determination of Qualification.” In response, the licensee determined that the affected components remained operable because the age of the O-rings in question was within the original qualification. The licensee entered this into their corrective action program as CR 1065957.

  The performance deficiency was determined to be more than minor because if left uncorrected, the performance deficiency would have the potential to lead to a more significant safety concern. Specifically, the failure to properly determine the qualified life and replace the O-rings at the required time interval would adversely affect the reliability of that equipment when called upon to respond to initiating events and prevent undesirable consequences. This finding was not assigned a cross-cutting aspect because the issue did not reflect current licensee performance. (Section 1R17.b.2)

Other Findings: Traditional Enforcement

- SL IV: The team identified a Severity Level IV non-cited violation of 10 CFR 50.59(c)(2), "Changes, Tests, and Experiments," for the licensee's failure to obtain a license amendment, as specified by Nuclear Energy Institute (NEI) 96-07 Section, 4.3.2, prior to implementing a change that increased the likelihood of a malfunction of a safety-related dike. This has been entered into the licensee corrective action program as condition report 1065945.

  The violation was dispositioned using the traditional enforcement process in accordance with the NRC Enforcement Policy, Subsection 2.2.2 Revised August 1, 2016, because the issue affected the NRC's ability to perform its regulatory oversight function. The NRC Enforcement Policy, Section 6.1, "Violation Examples for Reactor Operations," Subsection 6.1.d.2 specified that violations of 10 CFR 50.59 which resulted in conditions that were evaluated by the Significance Determination Process (SDP) as being of very low safety significance represented a severity level IV violation. The regional senior reactor analyst performed a screening analysis to determine the significance of the violation. Using very conservative failure frequencies for ductile iron pipe used in water systems, and a conservative initiating event frequency for an independent simultaneous rainfall capable of filling the dike, the finding was determined to be of very low safety significance. The inspector determined that the detailed risk evaluation confirmed that a severity level IV violation was appropriate. Crosscutting aspects are not assigned to traditional enforcement violations. (Section 1R17.b.3)

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R17 Evaluations of Changes, Tests, and Experiments (71111.17T)

.a Inspection Scope

Evaluations of Changes, Tests, and Experiments: The inspectors reviewed six safety evaluations performed pursuant to Title 10, *Code of Federal Regulations* (CFR) 50.59, “Changes, tests, and experiments,” to determine if the evaluations were adequate and that prior NRC approval was obtained as appropriate. The inspectors also reviewed fifteen screenings where licensee personnel had determined that 10 CFR 50.59 evaluations were not necessary. The inspectors reviewed these documents to determine if:

- The changes, tests, or experiments performed were evaluated in accordance with 10 CFR 50.59 and that sufficient documentation existed to confirm that a license amendment was not required;
- The safety issues requiring the changes, tests or experiments were resolved;
- The licensee conclusions for evaluations of changes, tests, or experiments were correct and consistent with 10 CFR 50.59; and
- The design and licensing basis documentation used to support the change was updated to reflect the change.

The inspectors used, in part, Nuclear Energy Institute (NEI) 96-07, “Guidelines for 10 CFR 50.59 Implementation,” Revision 1, to determine acceptability of the completed evaluations and screenings. The NEI document was endorsed by the NRC in Regulatory Guide 1.187, “Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments,” dated November 2000.

This inspection constituted 21 evaluations, screenings, and/or applicability determination samples as defined in Inspection Procedure (IP) 71111.17-05. Documents reviewed are listed in the Attachment.

.b Findings

.1 Failure to Qualify MCCs in Cable Penetration areas in accordance with 10 CFR 50.49

Introduction: The NRC identified a Green NCV of 10 CFR 50.49(f) for failing to qualify structures, systems, and components (SSCs) (eight motor control centers) located in a radiation harsh environment per 10 CFR 50.49.

Description: Design change NA-11-00002 installed an alternate power supply in a 480V motor control center (MCC) for spent fuel pit cooling pump 1-FC-P-1B. The MCC was located in environmental qualification (EQ) zones AB-259B. The zone ID (AB-259B) is the same for Unit 1 and Unit 2. Inspectors noted that the EQ zone manual designated these zones as a harsh environment but the design change lacked an EQ review.

During the inspectors review, it became apparent that the licensee did not treat SSCs in these zones as EQ components. The licensee used technical report (TR) EQ-0034, Rev. 0, dated 3/7/1988, to reduce the original dose for these zones, which justified the removal of the SSCs in these zones from the EQ program. This was because the replacement parts received for the original MCCs (Klockner-Moeller) did not meet the original EQ test report for the zones (STR132778-1 dated 2/5/1979) or an approved equivalent test report. Since 1988, the multiple SSC modifications and MCC replacements in these zones did not consider environmental qualification. The North Anna Power Station's EQ program nevertheless identified these zones as subject to a radiation harsh environment at 3.72E+04rads total integrated dose. The inspectors determined that the TR was a non-conservative analysis of the zone dose rate. In addition, there are SSCs in these zones that meet the definition of electric equipment important to safety covered by 10 CFR 50.49.

The inspectors noted that Section 3.11.2.9 of North Anna Power Station's Updated Final Safety Analysis Report (UFSAR) stated that replacement electrical SSCs under the scope of 10 CFR 50.49 are to be qualified in accordance with 10 CFR 50.49. To accomplish this, procedure CM-AA-EQ-10, "Fleet EQ Program Descriptions" required electrical SSCs to be qualified in accordance with IEEE Std. 323-1974. The standard IEEE 323-1974, Section 5, "Principles of Qualification," specified, that the capability of all Class 1E equipment for performing its required functions shall be demonstrated for assurance that the severity of the qualification methods equal or exceed the maximum anticipated service requirements and conditions.

The inspectors determined that none of the modified or replaced SSCs in the zone were environmentally qualified according to IEEE 323-1974 to meet 10 CFR 50.49.

<u>Analysis:</u> The inspectors determined that the licensee's failure to qualify the SSCs in EQ zone AB-259B in accordance with IEEE Std. 323-1974 Section 5, "Principles of Qualification," was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the Equipment Performance attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of the safety related AC Power System. Specifically, the failure to perform environmental qualification for SSCs subject to a harsh environment, during which time they must perform a safety function adversely affected the reliability of that equipment when called upon. The finding was evaluated using IMC 0609, Appendix A, "The Significance Determination Process (SDP) for Findings at-Power, and determined the finding was of very low safety significance (Green) because the finding was a deficiency affecting the qualification of a mitigating SSC but did not affect the operability or functionality. This finding was not assigned a cross-cutting aspect because the issue did not reflect current licensee performance.

<u>Enforcement:</u> 10 CFR 50.49(f) required, that, each item of electric equipment important to safety under the scope of the rule to be qualified by any one of the following methods:

(1) Testing an identical item of equipment under identical conditions or under similar conditions with a supporting analysis to show that the equipment to be qualified is acceptable

(2) Testing a similar item of equipment with a supporting analysis to show that the equipment to be qualified is acceptable
(3) Experience with identical or similar equipment under similar conditions with a supporting analysis to show that the equipment to be qualified is acceptable
(4) Analysis in combination with partial type test data that supports the analytical assumptions and conclusions

Contrary to the above, since at least 1988, the licensee failed to qualify electric equipment important to safety in EQ zone AB-259B by any one of the following methods:

(1) Testing an identical item of equipment under identical conditions or under similar conditions with a supporting analysis to show that the equipment to be qualified is acceptable
(2) Testing a similar item of equipment with a supporting analysis to show that the equipment to be qualified is acceptable
(3) Experience with identical or similar equipment under similar conditions with a supporting analysis to show that the equipment to be qualified is acceptable
(4) Analysis in combination with partial type test data that supports the analytical assumptions and conclusions

In response to this issue, the licensee performed an operability determination and determined that the MCCs were operable based on the material similarity of the original SSCs and the new SSCs. Because this issue is of very low safety significance (Green) and has been entered into the corrective action program as CR 1065894, this violation is being treated as a NCV consistent with Section 2.3.2 of the NRC Enforcement Policy. This violation is identified as NCV 05000338/2017007-01 and 05000339/2017007-01, Failure to Qualify MCCs in Cable Penetration areas in accordance with 10 CFR 50.49.

.2 <u>Failure to Qualify EGS brand Quick Disconnects in Accordance With IEEE Std. 323-1974</u>

<u>Introduction:</u> The NRC identified a Green non-cited violation of 10 CFR 50.49(e)(5) for failing to base the qualified life of structures, systems, and components (SSCs) (i.e. Nordel O-rings) on the known limits of extrapolation.

<u>Description:</u> Design change NA-15-00101 described the replacement of Conex seal assemblies on Unit 1 pressurizer pressure transmitters with EGS quick disconnect cables. These cables are required to operate in a harsh environment. Original testing for the O-rings used accelerated aging to determine the mathematical model (Arrhenius) that established a 10 year qualified life at approximately 150°F. Using this previously established mathematical model, the licensee subsequently extrapolated a 74.2-year qualified life, based on 125°F as calculated in QDR-N-34.4, “EGS Quick Disconnect Connectors Qualified Life Calculation 3." The calculation did not consider the known limits of extrapolation. The inspectors noted that the licensee EQ program did not base qualified life extrapolations on the known limits of extrapolation as required by IEEE 323-1974.

For new and replacement SSCs, procedure CM-AA-EQ-10, “Fleet EQ Program Descriptions,” required that the EGS quick disconnects be qualified in accordance with IEEE Std. 323-1974. Section 6.5.4 of IEEE Std. 323-1974 specified, in part, that the qualified life of equipment shall be “based upon the known limits of extrapolation of the time dependent environmental effects if an accelerated aging test was used to determine the mathematical model.” The inspectors determined that without considering the limits of extrapolation there is no expectation that adequate margins exist to form a reasonable expectation that these SSCs could perform their safety function when called upon. Many quality standards discuss the processes and methods for considering the known limits of extrapolation related to the Arrhenius model, and IEEE Std. 323-1974 Section 6.3.3, specifically identifies IEEE Std. 101-1972, “IEEE Guide for Statistical Analysis of Thermal Life Test Data” as one such quality standard. This standard describes statistical analyses for data from thermally accelerated aging tests, and discusses the suitability of extrapolation and the confidence limits on establishing a qualified life curves.

Analysis: The inspectors determined that the licensee’s failure to base the qualified life of SSCs (i.e. Nordel O-rings) upon the known limits of extrapolation of the time dependent environmental effects in accordance with IEEE 323 Sections 6.5.3, “Extrapolation,” and 6.5.4 “Determination of Qualification,” was a performance deficiency. The performance deficiency was determined to be more than minor because if left uncorrected, the performance deficiency would have the potential to lead to a more significant safety concern. Specifically, the failure to properly determine the qualified life and replace the O-rings at the required time interval would adversely affect the reliability of that equipment when called upon to respond to initiating events and prevent undesirable consequences. The finding was evaluated using IMC 0609, Appendix A, “The Significance Determination Process (SDP) for Findings at-Power.” The finding is of very low safety significance (Green) because the finding was a deficiency affecting the qualification of a mitigating SSC but did not affect the operability or functionality. This finding was not assigned a cross-cutting aspect because the issue did not reflect current licensee performance.

Enforcement: Title 10 CFR Part 50.49(e)(5), Aging, stated, in part, “Consideration must be given to all significant types of degradation which can have an effect on the functional capability of the equipment. The equipment must be replaced or refurbished at the end of this designated life unless ongoing qualification demonstrates that the item has additional life.” Contrary to the above, since September 15, 1992, the licensee failed to consider all significant types of degradation which can have an effect on the functional capability of the equipment and replace or refurbish equipment at the end of its designated life unless ongoing qualification demonstrated that the item had additional life. Specifically, the licensee did not adequately consider the impact of the known limits of extrapolating the qualified life of Nordel O-rings via the Arrhenius model. In response, the licensee determined that no immediate safety concern existed, as the O-rings in question had not been installed for longer than their original qualification. Because this issue is of very low safety significance (Green) and has been entered into the licensee corrective action program as CR 1065957, this violation is being treated as a NCV consistent with Section 2.3.2 of the NRC Enforcement Policy. This violation is identified as NCV 05000338/2017007-02 and 05000339/2017007-02 Failure to Qualify EGS Quick Disconnects in Accordance With IEEE Std. 323-1974

.3 Failure to Obtain NRC Review and Approval for Changes to Safety-Related Dike West of Unit 2 Turbine Building

Introduction: The team identified a Severity Level IV non-cited violation of 10 CFR 50.59(c)(2), "Changes, Tests, and Experiments," for the licensee's failure to obtain a license amendment, as specified by NEI 96-07 Section, 4.3.2, prior to implementing a change that increased the likelihood of a malfunction of a safety-related dike.

Description: The licensee's UFSAR and Individual Plant Examination of External Events (IPEEE) identified that the a flood protection dike west of the Unit 2 turbine building was safety-related because it protected both trains of Class 1E switchgear located in the lower levels of the Units 1 & 2 Turbine buildings. The flood path was identified as the excavation pit for the abandoned Units 3 & 4 construction west of the Unit 2 turbine building. The IPEEE specified three potential water sources. One was water from Lake Anna through dike VI (not safety-related) and the unit 3 & 4 intake tunnels, second was water from the service water pond overtopping, and third was water from storm runoff.

Design change DCP07-016 dated August 15, 2013, installed two non-safety-related water headers within the safety-related flood protection dike west of the Unit 2 turbine building. One header was a 120-psi, 12-inch inner-diameter ductile iron (DI) pipe used for the fire protection system (FPS), and the second header was a 60-psi, 2-inch diameter high-density polyethylene (HDPE) pipe used for the domestic water system (DWS). The design change buried these headers to a depth of about 7 to 8 feet down from the top and inside from the dike's slope face to protect the pipes from tornado damage. Prior to this, the dike did not have any pressurized water sources that could cause degradation or add to the potential dike failure frequency.

The inspectors identified a new dominant failure mode from the DI pipe failure frequency, which is present all year regardless of whether the dike is retaining water. Degradation from internal erosion (voiding) from pipe leakage or from pipe rupture could challenge the structural integrity of the dike whenever it was required to retain water. The inspectors identified failure data for earthen dikes from the U.S. Army Core of Engineers of approximately 1E-4/year. The inspectors obtained failure data for DI pipe from two studies. One by the University of Texas at Arlington dated 2015 for the Center for Underground Infrastructure Research and Education, and a second from Utah State University dated 2012. The inspectors noted that these studies only considered significant pipe failures excluding leakages. The two studies used for DI pipe failure data specified that failure frequency to be from 4.9 to 14 failures per 100 mile/year. For the dike, this translated to a DI pipe failure frequency from 4.64E-03/year to 1.33E-02/year. The inspectors determined that the increase in likelihood from significant pipe failures alone conservatively met the twofold increase criteria established in NEI 96-07 section 4.3.2, example 8. This was because just 4% to 1.5% of the dike DI pipe failure frequencies would double the dike failure frequency from 1E-4 to 2E-4. Licensee procedure CM-AA-400, "10 CFR 50.59 and 10 CFR 72.48 - Changes, Tests and Experiments," revision 7, used NEI 96-07 as detailed guidance and definitions for answering the questions in 10 CFR 50.59. The team determined that the licensee should have answered 10 CFR 50.59(c)(2) criterion (ii) as yes, thus requiring NRC review and approval.

The licensee's 50.59 evaluation for criterion (ii) stated, "the addition of the FPS and DWS piping in the dike does not introduce the possibility of a change in the likelihood of a malfunction evaluated in the UFSAR because the UFSAR does not evaluate a malfunction of the dike: That is, the UFSAR does not evaluate flooding events during which partial or complete flood protection as provided by the dike has been lost." However, licensee calculation 25161-G-060, "Slope Stability Analysis for Flood Protection Dike," revision 1, did analyze pipe failures. The analysis stated, in part, "In the case of larger leaks where visible seepage could occur on the slope face, some erosion of the face would happen if this situation was allowed to continue for extended periods." The inspectors noted that the licensee established an undetectable leak rate of approximately 30 to 60 gallons per minute. In addition, the analysis stated, in part, "…although considerable damage would be caused to the dike if the high pressure flow from this 12-in. pipe were allowed to go unchecked for an extended period of time. However, this will not happen since the operator will become aware of a fire protection system pipe break or leak in various ways. An initiation of action to isolate the break in the piping would occur in a timely fashion." The inspectors noted that none of the analysis evaluated the increase in likelihood of the dike malfunctioning it only allowed for timely repair of the dike after failure.

In addition to the increase in likelihood trigger for criterion (ii), NEI 96-07 Section 4.3.2, stated, in part, "departures from … the General Design Criteria [GDCs] (Appendix A to Part 50) are not compatible with a 'no more than minimal increase.'" The inspectors determined that the change departed from GDC 3 and GDC 4. Which specified, in part, for GDC 3, "firefighting systems (the FPS header) be designed to assure that their rupture does not significantly impair the safety capability of SSCs," and specified, in part, for GDC 4, "SSCs shall be appropriately protected against discharging fluids that may result from equipment failures and from events and conditions outside the nuclear power unit." The inspectors determined that this modification did not meet GDC 3, because the licensee analysis specified that a break or leak from the FPS header would cause considerable damage to the dike. The modification did not meet GDC 4 because the modification did not appropriately protect the dike from discharging fluids from the FPS or DWS headers.

Analysis. The violation was dispositioned using the traditional enforcement process in accordance with the NRC Enforcement Policy, Subsection 2.2.2 Revised August 1, 2016, because the issue affected the NRC's ability to perform its regulatory oversight function. The NRC Enforcement Policy, Section 6.1, "Violation Examples for Reactor Operations," Subsection 6.1.d.2 specified that violations of 10 CFR 50.59 which resulted in conditions that were evaluated by the Significance Determination Process (SDP) as being of very low safety significance represented a severity level IV violation. The regional senior reactor analyst performed a screening analysis to determine the significance of the violation. Using very conservative failure frequencies for ductile iron pipe used in water systems, and a conservative initiating event frequency for an independent simultaneous rainfall capable of filling the dike, the finding was determined to be of very low safety significance. The inspector determined that the detailed risk evaluation confirmed that a severity level IV violation was appropriate. Crosscutting aspects are not assigned to traditional enforcement violations.

Enforcement: Title 10 CFR 50.59(c)(2)(ii), stated, in part, "a licensee shall obtain a license amendment prior to implementing a proposed change, test, or experiment if the change, test, or experiment would … result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component, important to safety previously evaluated in the final safety analysis report." Contrary to the above, from August 15, 2013, to April 20, 2017, the licensee failed to obtain a license amendment prior to making a change that resulted in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the final safety analysis report. Specifically, the licensee failed to obtain a license amendment prior to implementing a change that increased the likelihood of a malfunction of a safety-related dike. The licensee entered the issue into the corrective action program as condition report 1065945. Because the significance was determined to be of very low safety significance and because the licensee entered it into its corrective action program, this violation is being treated as a non-cited violation, consistent with Section 2.3.2.a of the NRC Enforcement Policy: NCV 05000338/2017007-03 and 05000339/2017007-03, Failure to Obtain NRC Approval for Changes to Safety-Related Dike West of Unit 2 Turbine Building for Flood Mitigation Strategy

4OA6 <u>Meetings, Including Exit</u>

On April 21, 2017, the team presented inspection results to Mr. L. Lane and other members of the licensee's staff. The team verified that no proprietary information was retained by the inspectors or documented in this report.

ATTACHMENT: SUPPLEMENTARY INFORMATION

SUPPLEMENTARY INFORMATION

KEY POINTS OF CONTACT

Licensee personnel
E. Hendrixson, Director, Nuclear Site Engineering
L. Hilbert, Plant Manager
L. Lane, Site Vice President
J. Leberstien, Technical Consultant, Licensing
A. Lerch, Consulting Engineer, Electrical Design
J. McGinnis, Technical Specialist, Licensing
J. Mettale, Supervisor, Electrical Design
S. Morris, Manager, Engineering Design
B. Stanley, Director, Nuclear Station Safety & Licensing
D. Taylor, Manager, Licensing

NRC personnel
G. Croon, Senior Resident Inspector, North Anna
S. Cuadrado, Resident Inspector, North Anna

LIST OF ITEMS OPENED, CLOSED, DISCUSSED, AND UPDATED

Opened and Closed

05000413, 414/2017007-01	NCV	Failure to Qualify MCCs in Cable Penetration areas in accordance with 10 CFR 50.49
05000413, 414/2017007-02	NCV	Failure to Qualify EGS Quick Disconnects in Accordance With IEEE Std. 323-1974
05000413, 414/2017007-03	NCV	Failure to Obtain NRC Review and Approval for Changes to Safety-Related Dike West of Unit 2 Turbine Building

LIST OF DOCUMENTS REVIEWED

10 CFR 50.59 Evaluations
DCP07-016, Site Preparations for Unit 3 – Fire Protection and Domestic Water System Modifications, dated 1/20/11
NA-12-00037, Unit 2 Zinc Injection Installation, dated 4/10/13
NA-12-00052, Unit 1 Station Service Bus Relays, dated 8/82013
NA-14-00076, Abandonment of Unit 2 Core Exit Thermocouples, dated 10/42014
NA-15-00005, Solid State Protection System Digital Card Replacement, dated 6/10/2015
NA-14-00006, Unit 1 S/G Blowdown Sodium Analyzer Upgrade, 12/16/2014

10 CFR 50.59 Screenings
DCP07-132, Lube Oil Sample Test Port Installation / Naps / U2, dated 10/30/07
NA-09-00166, Modified Service Water Pump Spare – Equivalency Evaluation, dated 10/21/09
NA-11-00002, Phase 2 to Install Alternate Power Supply to Spent Fuel Pit Cooling Pump 1-FC-P-1B, dated 5/22/ 2012
NA-11-01097, CMIS – Steam Flow / Feed Flow Coincident With Low Steam Generator Level Reactor Trip Function Elimination / Naps / Unit 1, dated 9/3/13
NA-12-00012, AC Voltage Transducer Replacement / Unit 2, dated 9/27/2012
NA-12-00041, Repair Reactor Coolant Loop Stop Valve 2-RC-MOV-2593, dated 8/21/12
NA-13-00027, RCP Floating Ring Seal Removal 1-Rc-P-1a-Pump / Naps / Unit 1, dated 7/18/13
NA-13-00064, Replacement and Elimination of 2J Under-Voltage Relays – Equivalency, dated 1/28/2014
NA-13-00068, 1J Emergency Diesel Generator K4 Relay Replacement, dated 8/11/2014
NA-13-00105, Casing Cooling Tank Low-Low Level Setpoint Change, dated 1/30/2014
NA-14-00046, Control Room Recorder Equivalency Replacement, dated 8/5/2014
NA-14-00085 Modification to Unit 2 Pressurizer Vent Valves Tail Pipe Mech, dated 06/30/15
NA-15-00020 Unit 2 Steam Generator Insulation Replacement Mech, dated 05/13/15
NA-15-00085, EDG Fuel Oil Transfer Pump Equivalent Replacement / NAPS / Unit 1, dated 08/25/15
NA-15-00101, EGS Quick Disconnect Electrical Connectors for 1-RC-LT-1459, 1-RC-LT-1460, and 1-RC-LT-1461, dated 7/12/2016

Calculations
SE-0008, Head Loss Calculation with Unit 3 and 4 Piping Removed from the Flow Paths, Rev. 0
25161-G-060, Slope Stability Analysis for Flood Protection Dike, Rev. 1
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<u>Condition Reports generated as a result of the inspection</u>
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