UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 6, 2017

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

SUBJECT: NORTH ANNA POWER STATION, UNITS 1 AND 2 – PROPOSED INSERVICE INSPECTION ALTERNATIVES N1-I4-NDE-009 AND N2-I4-NDE-004 (CAC NOS. MF9298 AND MF9299; EPIDS L-2016-LLR-0018)

Dear Mr. Stoddard:

By letter dated November 30, 2016, as supplemented by letter dated June 14, 2017 (Agencywide Documents Access and Management System Accession Nos. ML16340B092 and ML17171A230, respectively), Virginia Electric and Power Company (Dominion, the licensee), submitted requests to the U.S. Nuclear Regulatory Commission (NRC) for the use of proposed alternatives to certain inservice inspection (ISI) interval requirements for the North Anna Power Station, Units 1 and 2 (NAPS Units 1 and 2). Specifically, the licensee's proposed alternatives N1-I4-NDE-009 and N2-I4-NDE-004 would eliminate the volumetric examinations required by the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, Examination Category B-G-1, Item B6.40, Pressure retaining bolting greater than 2-inches, Reactor Vessel – Threads in Flange. Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50.55a(z)(1), the licensee requested approval for the use of the proposed alternatives, in accordance with an industry initiative analyzed in Electric Power Research Institute Report No. 3002007626, "Nondestructive Evaluation: Reactor Pressure Vessel Threads in Flange Examination Requirements."

The NRC staff has reviewed the subject request and concludes, as set forth in the enclosed safety evaluation, that the proposed alternatives will provide an acceptable level of quality and safety. Pursuant to 10 CFR 50.55a(z)(1), the staff authorizes the use of the proposed alternatives for the duration of the fourth 10-year ISI intervals at NAPS Unit 1 (until April 30, 2019) and NAPS Unit 2 (until December 13, 2020).

All other ASME Code requirements for which alternatives or relief were not specifically requested and approved remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

If you have any questions, please contact the Project Manager, Randy Hall, at 301-415-4032 or via e-mail at Randy.Hall@nrc.gov.

Sincerely,

Michael T. Markley

Michael T. Markley, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-338 and 50-339

Enclosure:
Safety Evaluation

cc w/encl: Distribution via Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST FOR ALTERNATIVE NOS. N1-I4-NDE-009 AND N2-I4-NDE-004

REACTOR VESSEL FLANGE THREAD EXAMINATIONS

FOURTH TEN-YEAR INSERVICE INSPECTION INTERVALS

VIRGINIA ELECTRIC AND POWER COMPANY

NORTH ANNA POWER STATION, UNITS 1 AND 2

DOCKET NOS. 50-338 AND 50-339

1.0 INTRODUCTION

By application dated November 30, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16340B092), as supplemented by letter dated June 14, 2017 (ADAMS Accession No. ML17171A230), Virginia Electric and Power Company (Dominion, the licensee) submitted request numbers N1-I4-NDE-009 and N2-I4-NDE-004 in accordance with Paragraph 50.55a(z)(1) of Title 10 of the *Code of Federal Regulations* (10 CFR) for a proposed alternative to the requirements of 10 CFR 50.55a, "Codes and standards," for North Anna Power Station, Units 1 and 2 (NAPS Units 1 and 2). The U.S. Nuclear Regulatory Commission (NRC) staff issued a request for additional information (RAI) regarding the licensee's application, as documented in a memorandum dated May 17, 2017 (ADAMS Accession No. ML17137A093).

The proposed alternative would allow the licensee to eliminate the examination of threads in the reactor pressure vessel (RPV) flange required by Examination Category B-G-1, Item No. B6.40, in Section XI, "Rules for Inservice Inspection (ISI) of Nuclear Power Plant Components," of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code) for the fourth 10-year ISI intervals of NAPS Units 1 and 2. Specifically, pursuant to 10 CFR 50.55a(z)(1), the licensee requested to use the alternative on the basis that it will provide an acceptable level of quality and safety. The licensee confirmed in the June 14, 2017, letter in its response to RAI 1 that the alternative is requested only for the fourth 10-year ISI intervals of NAPS Units 1 and 2.

2.0 REGULATORY EVALUATION

The regulations in 10 CFR 50.55a(g)(4) state, in part, that ASME Code Class 1, 2, and 3 components (including supports) must meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in Section XI of the applicable editions and addenda of the ASME Code to the extent practical within the limitations of design, geometry, and materials of construction of the components. The threads in the RPV flange are categorized as ASME Code Class 1 components. Therefore, per 10 CFR 50.55a(g)(4), ISI of these threads must be performed in accordance with Section XI of the applicable edition and addenda of the ASME Code.

The regulations in 10 CFR 50.55a(z) state, in part, that alternatives to the requirements in paragraphs (b) through (h) of 10 CFR 50.55a may be authorized by the NRC if the licensee demonstrates that: (1) the proposed alternative provides an acceptable level of quality and safety, or (2) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

For the fourth 10-year ISI intervals at NAPS, Units 1 and 2, the Code of record for the inspection of ASME Code Class 1, 2, and 3 components is the 2004 Edition of the ASME Code, Section XI, with no addenda. The fourth 10-year ISI intervals end on April 30, 2019, for NAPS Unit 1 and December 13, 2020, for NAPS Unit 2.

3.0 TECHNICAL EVALUATION

3.1 Licensee's Request

3.1.1 ASME Code Components Affected

The proposed alternative applies to threads in the RPV flange subject to ASME Code, Section XI, Examination Category B-G-1, Item No. B6.40.

3.1.2 Applicable ASME Code Edition and Addenda

The applicable ASME Code edition for the NAPS Units 1 and 2 fourth 10-year ISI intervals is the 2004 Edition of Section XI with no Addenda.

3.1.3 Applicable ASME Code Requirement

The licensee has requested an alternative to the examination requirements in Examination Category B-G-1, Item No. B6.40, which is listed in Table IWB-2500-1, "Examination Categories" of the ASME Code, Section XI. This item requires volumetric examination, every ISI interval, of all the threads in RPV flange stud holes, as indicated in Figure IWB-2500-12 "Closure Stud and Threads in Flange Stud Hole" of the ASME Code, Section XI.

3.1.4 Licensee's Proposed Alternative and Basis for Use

The licensee is proposing to eliminate the examination of the threads in the RPV flange, as required by Examination Category B-G-1, Item No. B6.40, of the ASME Code, Section XI, for the fourth 10-year ISI intervals of NAPS Units 1 and 2. The licensee's request is based on an evaluation by the Electric Power Research Institute (EPRI) documented in EPRI Technical Report No. 3002007626 (EPRI report), "Nondestructive Evaluation: Reactor Pressure Vessel Threads in Flange Examination Requirements," dated March 2016 (ADAMS Accession No. ML16221A068). The licensee's submittal included information from the EPRI report regarding the generic stress analysis and flaw tolerance evaluation, with additional plant-specific information to demonstrate applicability of the EPRI results. The submittal also included information from the EPRI report regarding operating experience and potential degradation mechanisms for the threads in the RPV flange.

Additionally, the licensee stated that, prior to returning the reactor to service during each refueling outage, it inspects, cleans, and lubricates the threads in the RPV flange prior to reinstalling the RPV studs, to detect and mitigate general degradation.

3.2 NRC Staff's Evaluation

The licensee relied on the EPRI report for the technical basis for the proposed alternative to eliminate examination of threads in the RPV flange. The NRC staff focused its evaluation of the proposed alternative on the deterministic stress analyses and flaw tolerance evaluation in the EPRI report, but also considered operating experience and potential degradation mechanisms. Each of these topics was discussed in the EPRI report and in the licensee's submittal.

By letter dated January 26, 2017 (ADAMS Accession No. ML17006A109), the NRC staff authorized Southern Nuclear Operating Company, Inc. (SNC) to use a similar alternative at Vogtle Electric Generating Plant, Units 1 and 2, and Joseph M. Farley Nuclear Plant, Unit 1 that was also based on the generic stress analysis and flaw tolerance evaluation in the EPRI report. Section 3.2.1 of the safety evaluation (SE) for the SNC authorization (SNC SE) documents the NRC staff's evaluation of the EPRI report, and concludes that EPRI's generic stress analysis and flaw tolerance evaluation are acceptable and the results can be used to support the elimination of the threads in RPV flange examination. For Dominion's proposed alternative, the NRC staff relied on this previous evaluation and focused on the plant-specific threads in RPV flange information to determine if EPRI's generic stress analysis and flaw tolerance evaluation are applicable to NAPS Units 1 and 2.

3.2.1 Operating Experience

The EPRI report included the results of a survey of U.S. nuclear reactors taken in 2015 and early 2016 of the volumetric examination results for threads in the RPV flange (Table 3 of the licensee's submittal). The survey included 33 boiling water reactor (BWR) units and 61 pressurized water reactor (PWR) units. The total number of examinations for all 94 units was 10,662, with no reportable indications. The NRC staff finds that these survey results offer ample supporting evidence that the threads in the RPV flange are performing their function without a credible threat to the structural integrity of the RPV flange.

3.2.2 Potential Degradation Mechanisms

Section 5, "Evaluation of Potential Degradation Mechanisms," of the EPRI report provides an evaluation of the susceptibility of the threads in the RPV flange to the following degradation mechanisms: pitting, intergranular attack, corrosion, fatigue, stress corrosion cracking, crevice corrosion, velocity phenomena, de-alloying corrosion and general corrosion, stress relaxation, creep, mechanical wear and mechanical/thermal fatigue. The EPRI report concluded that the only potential degradation mechanisms applicable to the threads in the RPV flange are mechanical and thermal fatigue. To address the potential for mechanical or thermal fatigue, the licensee referred to the generic stress analysis and flaw tolerance analysis in the EPRI report.

The NRC agrees that mechanical and thermal fatigue are the only potential degradation mechanisms for the threads in RPV flanges of NAPS Units 1 and 2. The other degradation mechanisms listed in the EPRI report (e.g., stress corrosion cracking and creep) are not credible degradation mechanisms for the threads in the RPV flange, because they are not in contact with the reactor coolant and they are not in the operating temperature range where metal creep can occur.

3.2.3 Stress Analysis

Section 6.1, "Stress Analysis," of the EPRI report describes the determination of stresses at the critical location in the threads in the RPV flange. These stresses were used as input into the flaw tolerance evaluation, which is discussed in Section 3.2.4 of this safety evaluation. The stress analysis was performed using a three-dimensional, symmetric finite element model (FEM) of a portion of the threads in the RPV flange, RPV shell immediately below the flange, and a symmetric half of an RPV stud. Geometric parameters, such as number of RPV studs, stud diameter, RPV inside diameter, and flange thickness at the threads, were used to create the FEM. The loads applied in the FEM were the preload on the RPV studs, internal pressure, and thermal loads due to heatup and cooldown.

In the SNC SE, the NRC staff concluded that the generic EPRI stress analysis is acceptable and that the resulting stresses can be used in the subsequent flaw tolerance evaluation. For Dominion's proposed alternative, the NRC staff relied on its previous evaluation and conclusion regarding the generic EPRI stress analysis, and focused on the plant-specific threads in RPV flange information to determine the applicability of the generic stress analysis to NAPS Units 1 and 2.

Finite Element Model

As discussed in the EPRI report, bounding geometric parameters were used to create an FEM. The EPRI report states that the PWR design was used as a representative geometry for the FEM because of its higher design pressure and temperature. In Table 1 of Attachment 1 of the submittal, the licensee showed the NAPS Units 1 and 2 geometric parameters along with those used in the bounding analysis in the EPRI report. The NRC finds the selection of a PWR design acceptable for NAPS Units 1 and 2 because both units are PWRs. Additionally, the NRC staff finds the PWR geometric parameters in the EPRI report acceptable because they bound the geometric parameters of NAPS Units 1 and 2.

Applied Loads

With respect to preload stress, in Table 1 of the Attachment 1 of the submittal, the licensee showed geometric parameters of NAPS Units 1 and 2 and compared them to the bounding values used in the EPRI calculation of preload stress on the RPV studs. The NRC staff verified the geometric parameters the licensee provided using the NAPS Units 1 and 2 updated final safety analysis report (UFSAR) and the corresponding value of the calculated preload stress for each unit. The NRC staff determined that the 42,338 pounds per square inch (psi) preload stress used in the EPRI analysis bounds the calculated preload stress for NAPS Units 1 and 2. The NRC staff was concerned that the calculated preload stress for NAPS Units 1 and 2 may be different than the actual preload stress. Therefore, as documented by memorandum dated May 17, 2017, the NRC staff requested the licensee to confirm that the 42,338 psi preload stress used in the EPRI analysis bounds the actual preload stress for NAPS Units 1 and 2. By letter dated June 14, 2017, in its response to RAI 2, the licensee confirmed it.

The stress analysis in the EPRI report evaluated reactor heatup, but not a reactor cooldown. In the SNC SE, the NRC staff found that the use of heatup or cooldown has no effect on the fatigue crack growth calculation (evaluated in Section 3.2.4 of this SE for NAPS Units 1 and 2) because it would produce the same stress range in the calculation. The EPRI thermal transient analysis assumed a 100 degrees Fahrenheit per hour heatup rate for the reactor coolant until the operating temperature was reached. The heatup rate is acceptable because it is greater than or equal to the maximum allowed reactor coolant heatup rate specified in the NAPS Units 1 and 2 UFSAR.

Based on the above, the NRC staff concludes that the applied loads used in the EPRI stress analysis are acceptable for NAPS Units 1 and 2.

3.2.4 Flaw Tolerance Evaluation

Section 6.2, "Flaw Tolerance Evaluation," of the EPRI report describes how the crack driving force, or stress intensity factor, K_I, due to the applied loads was determined. The flaw tolerance analysis, including the crack growth analysis, was based on the principles of linear elastic fracture mechanics. The stresses in the region of the root of the threads in the FEM were used to determine the critical location based on the largest tensile axial stress. A flaw was simulated by inserting crack tip elements in the FEM originating from this critical location, which enabled K_I to be determined. The flaw was modeled around the critical thread and orientated such that the axial stresses act normal to the face of the flaw. Four flaw depths were modeled to determine the variation of K_I with flaw depth, and the maximum applied K_I was compared to the maximum value allowed by subarticle IWB-3600 of the ASME Code, Section XI. A flaw growth evaluation was then performed with a postulated initial flaw size at the root of the critical thread to show that the structural integrity of the threads in the RPV flange was not compromised for 80 years of plant life.

In the SNC SE, the NRC staff documented its conclusion that the generic EPRI flaw tolerance evaluation is acceptable. For Dominion's proposed alternative, the NRC staff relied on its previous evaluation and conclusion, and focused on the plant-specific threads in RPV flange information to determine the applicability of the generic flaw tolerance analysis to NAPS Units 1 and 2.

The generic EPRI flaw tolerance evaluation included simulations of a postulated flaw of four sizes inserted into the FEM to determine K_I due to preload, internal pressure, and heat-up transient. The maximum applied K_I around the postulated flaw was determined for each flaw depth for two load cases: (1) preload only and (2) preload with heat-up and pressure. The first case occurs during tensioning of the RPV bolts, and the second case occurs during reactor heatup to operating temperature and pressure. The EPRI report identified a maximum applied K_I of 17.4 kilopounds per square inch - square root inch (ksi$\sqrt{in}$) for the first case and 19.8 ksi$\sqrt{in}$ for the second case. The maximum applied K_I of 19.8 ksi$\sqrt{in}$ is less than the allowable value of 69.6 ksi$\sqrt{in}$, which is based on the RPV flange fracture toughness (K_{IC}) value at the upper shelf temperatures of the lower bound K_{IC} curve applicable to ferritic steels in Appendix A to the ASME Code, Section XI. Since the maximum applied K_I is less than the allowable value, the NRC staff concludes that the threads in RPV flange are reasonably flaw tolerant at operating temperatures.

The EPRI report did not include a comparison of the maximum applied K_I value of 17.4 ksi$\sqrt{in}$ for the preload case to the allowable value of K_I at the temperature appropriate for the preload case. Therefore, the NRC staff requested the licensee to provide this comparison. By letter dated June 14, 2017, in its response to RAI 3, the licensee stated the allowable K_I value is 44.27 ksi$\sqrt{in}$ for the preload case for the NAPS Units 1 and 2 RPV flanges and that this allowable K_I value provides sufficient margin from the applied K_I value of 17.4 ksi$\sqrt{in}$. The licensee calculated the allowable K_I value for the NAPS Units 1 and 2 RPV flanges based on the nil-ductility transition reference temperature (RT_{NDT}) of -22°F and the K_{IC} curve in Appendix A to ASME Code, Section XI. The NRC staff verified the RT_{NDT} value of -22°F from the NAPS Units 1 and 2 UFSAR, and the licensee's calculation of the allowable K_I value using the evaluation methods in the ASME Code, Section XI. Therefore, the NRC staff finds the licensee's comparison of applied K_I to the allowable K_I acceptable for the preload case. Accordingly, the NRC staff concludes that the NAPS Units 1 and 2 threads in RPV flanges are reasonably flaw tolerant at preload temperatures.

The SNC SE stated that for a postulated flaw of 0.2 inches from the root of thread, the crack would grow by 0.005 inch over 80 years of reactor operation. The 0.005 inch crack growth value was provided in a letter dated October 24, 2016 (ADAMS Accession No. ML16298A049) during the review of the SNC request. The NRC staff concluded in the SNC SE that this amount of crack growth was acceptable. Additionally, the NRC staff determined this crack growth length is bounding using the fatigue crack growth curves in Figure A-4300-1 in ASME Code, Section XI, Appendix A. The crack growth evaluation in the EPRI report also assumed 50 reactor heatup/cooldown cycles per year and 5 bolt preloads per year. The NRC staff confirmed that these assumptions are conservative for NAPS Units 1 and 2.

3.2.5 Technical Conclusion

The NRC staff determined that the licensee has demonstrated that the deterministic stress analysis and flaw tolerance evaluation in the EPRI report are bounding for the threads in RPV flanges of NAPS Units 1 and 2. Therefore, the NRC staff determined that elimination of the ASME Code-required examination of threads in the RPV flanges of NAPS Units 1 and 2 is acceptable, because the licensee has provided reasonable assurance of structural integrity of the threads in RPV flanges without this examination for the duration of the fourth 10-year ISI intervals of NAPS Units 1 and 2.

4.0 CONCLUSION

As set forth above, the NRC staff determined that the licensee's proposed alternative to eliminate the ASME Code-required examination of threads in the RPV flange during the NAPS Units 1 and 2 fourth 10-year ISI intervals provides an acceptable level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1). Therefore, the NRC staff authorizes the use of the proposed alternative in request numbers N1-I4-NDE-009 and N2-I4-NDE-004, as supplemented, for the duration of the fourth 10-year ISI intervals for NAPS Units 1 and 2.

All other ASME Code requirements for which relief was not specifically requested and approved remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: David Dijamco

Date: December 6, 2017

SUBJECT: NORTH ANNA POWER STATION, UNITS 1 AND 2 - PROPOSED INSERVICE INSPECTION ALTERNATIVES N1-I4-NDE-009 AND N2-I4-NDE-004 (CAC NOS. MF9298 AND MF9299; EPIDS L-2016-LLR-0018) DATED DECEMBER 6, 2017

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