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

June 25, 2015

Mr. David A. Heacock
President and Chief Nuclear Officer
Virginia Electric and Power Company
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: NORTH ANNA POWER STATION, UNIT NO. 2, PROPOSED INSERVICE INSPECTION (ISI) ALTERNATIVE N2-I4-NDE-002 (TAC NO. MF4981)

Dear Mr. Heacock:

By letter dated October 6, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14283A044), as supplemented by letter dated March 2, 2015 (ML15069A227), Virginia Electric and Power Company (the licensee) proposed an alternative to certain inservice inspection (ISI) program requirements for its North Anna Power Station, Unit 2.

The regulation in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(g)(6)(ii)(D), states, in part, that all licensees of pressurized water reactors must augment their ISI program with American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (ASME) Code Case N-729-1, "Alternative Examination Requirements for PWR [Pressurized Water Reactor] Reactor Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds, Section XI, Division 1," subject to conditions specified in 10 CFR 50.55a paragraphs (g)(6)(ii)(D)(2) through (6).

Specifically, the proposed alternative is to extend the CC N-729-1 Item B4.40 volumetric/surface examination schedule for the reactor vessel closure head nozzles and partial penetration welds from a ten year to fifteen year period. This would extend the schedule by five years for the next examination for North Anna Power Station, Unit 2 from Spring of 2016 to the Spring of 2022.

The NRC staff has reviewed the subject request and concludes, as set forth in the enclosed safety evaluation, that the proposed alternative provides an acceptable level of quality and safety in accordance with 10 CFR 50.55a(z); and therefore authorizes the proposed alternative.

All other requirements of the ASME Code, Section XI, and 10 CFR 50.55a(g)(6)(ii)(D) for which relief was not specifically requested and authorized herein by the NRC staff remain applicable, including the third party review by the Authorized Nuclear Inservice Inspector.

If you have any questions, please contact the Project Manager, V Sreenivas, at 301-415-2597 or via e-mail at V.Sreenivas@nrc.gov.

Sincerely,

Robert J. Pascarelli, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-339

Enclosure:
Safety Evaluation

cc w/encl: Distribution via Listserv

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

PROPOSED ALTERNATIVE TO ASME CODE CASE N-729-1

AS CONDITIONED BY THE REQUIREMENTS OF 10 CFR 50.55a(g)(6)(ii)(D) FOR

INSPECTION OF REACTOR VESSEL CLOSURE HEAD NOZZLES

VIRGINIA ELECTRIC AND POWER COMPANY

NORTH ANNA POWER STATION, UNIT 2

DOCKET NO. 50-339

1.0 INTRODUCTION

By letter dated October 6, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14283A044), as supplemented by letter dated March 2, 2015 (ADAMS Accession No ML15069A227), Virginia Electric and Power Company (the licensee) proposed an alternative to certain inservice inspection (ISI) program requirements for its North Anna Power Station, Unit 2.

Specifically, the alternative proposes to extend the CC N-729-1 Item B4.40 volumetric/surface examination schedule for the reactor vessel closure head nozzles and partial penetration welds from a ten year to fifteen year period. The proposed alternative would extend the schedule by five years for the next examination for North Anna Power Station, Unit 2 from the Spring of 2016 to the Spring of 2022. The provision for alternatives 10 CFR 50.55a(z)(1) states that alternatives to the requirements of paragraph (g) of 10 CFR 50.55a may be used, when authorized by the NRC, if the licensee demonstrates that the proposed alternatives would provide an acceptable level of quality and safety.

2.0 REGULATORY EVALUATION

The regulation in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(g)(6)(ii)(D), states, in part, that all licensees of pressurized water reactors must augment their ISI program with American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (ASME) Code Case N 729-1, "Alternative Examination Requirements for PWR [Pressurized Water Reactor] Reactor Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds, Section XI, Division 1," subject to conditions specified in 10 CFR 50.55a paragraphs (g)(6)(ii)(D)(2) through (6) of this section.

Enclosure

The staff notes that the licensee's proposed alternative was submitted pursuant to 10 CFR 50.55a(a)(3). 10 CFR Part 50, section 50.55a was revised, as noticed in the Federal Register on November 5, 2014 (79 FR 65776-65814). A part of that revision was the resequencing of the parts of 50.55a such that the provisions for proposing alternatives is moved from 10 CFR 50.55a(a)(3) to 50.55a(z). Accordingly, the staff is authorizing this proposed alternative pursuant to 10 CFR 50.55a(z)(1).

As discussed in this safety evaluation, the NRC staff finds that regulatory authority exists for the licensee to request and the Commission to authorize the proposed alternative.

10 CFR 50.55a(g)(6)(ii), states, in part, that the Commission may require the licensee to follow an augmented inservice inspection (ISI) program for systems and components for which the Commission deems that added assurance of structural reliability is necessary.

3.0 TECHNICAL EVALUATION

3.1 Components Affected

The affected components are ASME Class 1, Reactor Vessel Closure Head (RVCH) Penetration Nozzles and partial penetration welds, which are fabricated from Inconel SB-167 (Alloy 690) UNS N06690. The nozzle J-groove welds are fabricated from ERNiCrFe-7 (UNS N06052) and ENiCrFe-7 (UNS W86152), 52/152 weld materials. The original NAPS RVCH, which contained penetration nozzles which were manufactured with Alloys 600/82/182 materials, was replaced with a new RVCH using Alloys 690/52/152 material for the penetration nozzles during the Fall 2007 refueling outage.

3.2 Inservice Inspection Interval

The proposed duration of the alternative is for the fourth and fifth ten-year ISI intervals until the next exam is required to be performed, which is in the Fall 2022 outage.

3.3 ASME Code of Record

The ASME Section XI Code of Record for the fourth 10-year ISI interval is the 2004 Edition with no Addenda. NAPS adopts a later code edition for the 5th ISI Interval when that interval is entered on December 14, 2020.

3.4 ASME Code and/or Regulatory Requirements

Section 50.55a(g)(6)(ii)(D) of 10 CFR requires, in part, licensees shall augment their inservice inspection program in accordance with ASME Code Case N-729-1 subject to the conditions specified in paragraphs (2) through (6) of 10 CFR 50.55a(g)(6)(ii)(D). ASME Code Case N-729-1, Table 1, Inspection Item B4.40 requires volumetric/surface examination be performed within one inspection interval (nominally 10 calendar years) of its inservice date for a replaced RVCH. The required volumetric/surface examinations would thus have to be completed by fall 2017 in order to fulfill the requirements of N-729-1.

3.5 Proposed Alternative

The licensee proposes to delay the next required inspection for a period of approximately 5 years. As indicated in its March 2, 2015 submittal, the licensee proposes to accomplish the inspection in accordance with ASME Code Case N-729-1 and 10 CFR 50.55a(g)(6)(ii)(D) during refueling outage 28 which is scheduled for spring 2022.

3.6 Licensee's Basis for Use of the Proposed Alternative

The licensee's basis for use of the proposed alternative is based primarily on three topics of consideration. The first topic addresses the concept that the inspection interval in Code Case N-729-1 is based on primary water stress corrosion cracking (PWSCC) crack growth rates for Alloy 600/82/182. The second topic addresses a bare metal visual examination conducted on the licensee's replacement RVCH in 2012. The third topic addresses a plant-specific factor of improvement analysis conducted by the licensee.

In addressing its first basis for use of the proposed alternative, the licensee asserts that the inspection intervals contained in ASME Code Case N-729-1 for alloy 600/82/182 are based on Re-inspection years (RIY) equal to 2.25 and that this value is based on PWSCC crack growth rates as defined in the 75th percentile curve contained in MRP 55, "Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Thick-Wall Alloy 600 Material", and MRP 115, "Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Alloy 82, 182, and 132 Welds." The licensee further asserts that the PWSCC crack growth rates of alloy 690/52/152 are significantly lower than those of alloy 600/82/182 and, therefore, merit a longer inspection interval. The licensee bases that assertion on: a) the lack of cracking in other 690 components such as steam generators and pressurizers in the approximately 20 years that alloy 690 has been in service in these components; b) the failure to observe cracking in inspections already performed in replacement heads (9 of 40 replacement heads have been examined which includes heads which operate at higher temperatures than the head under consideration); c) the similarity of the inspected heads to the head under consideration regarding configuration, manufactures, design and operating conditions; and d) laboratory test data for alloys 690/52/152 as contained in MRP-375, "Technical Basis for Reexamination Interval Extension for Alloy 690 PWR Reactor Vessel Top Head Penetration Nozzles."

In addressing its second basis for use of the proposed alternative, the licensee stated that a bare metal visual examination was performed in 2012 on the NAPS replacement RVCH in accordance with ASME Code Case N-729-1, Table 1, item B4.30. This visual examination was performed by VT-2 qualified examiners on the outer surface of the RVCH including the annulus area of the penetration nozzles. This examination did not reveal any indications of nozzle leakage (e.g. boric acid deposits) on the surface or near a nozzle penetration. Also, the licensee stated that no alternative examination processes are proposed to those required by ASME Code Case N-729-1, as conditioned by 10 CFR 50.555a(g)(6)(ii)(D). The Visual (VT-2) examinations and acceptance criteria as required by item B4.30 of Table 1 of ASME Code Case N-729-1 are not affected by this request and will continue to be performed on a frequency not to exceed every 5 calendar years.

In addressing its third basis for use of the proposed alternative, the licensee made a plant specific calculation of the required factor of improvement in the crack growth rate of Alloy

690/52/152 as compared to the crack growth rate of alloy 600/82/182. In making this calculation the licensee used the actual temperature of the head and conservatively assumed that calendar years were equal to effective full power years. Based on this calculation, the licensee determined that an improvement factor of 9 was required to meet the proposed and desired inspection interval of 15 calendar years. The licensee then proposed that because the required factor of improvement (9) was smaller than the factor of improvement of 20 which bounded most of the MRP 375 data for alloy 690/52/152, the use of a factor of improvement of 9 would not result in a reduction in safety and was, therefore, justified.

The licensee stated that their analysis showed significant margin to ensure that Alloy 690 nozzle base and Alloy 52/152 weld materials used in the NAPS replacement RVCH provide for a reactor coolant system pressure boundary where the potential for PWSCC has been shown by analysis and by years of positive industry experience to be remote. As such, the licensee found the technical basis sufficient to ensure public health and safety by extending the inspection frequency of the RVCH nozzle at NAPS from a maximum of 10 years to a new maximum of 15 years.

3.7 NRC STAFF EVALUATION

In evaluating the technical sufficiency of the licensee's proposed alternative, i.e., a one-time extension of the volumetric/surface examination interval contained in ASME Code Case N-729-1 from 10 years to not longer than 15 years, the NRC staff considered each of the three aspects of the licensee's basis for use of the proposed alternative.

Due to concerns about PWSCC, many pressurized water reactor (PWR) plants in the United Stated and overseas have replaced reactor vessel closure heads containing Alloy 600/182/82 nozzles with heads containing Alloy 690/152/52 nozzles. The inspection frequencies developed in Code Case N-729-1 for RVCH penetration nozzles using Alloy 600/182/82 were developed based, in part, on those material's crack growth rate equations documented in MRP-55 and MRP-115. The licensee's primary technical basis is to present crack growth rate data for the new more crack resistant materials, Alloy 690/152/52, and demonstrate an improvement factor (IF) of these materials versus the older Alloy 600/82/182 materials. This improvement factor would then provide the basis for the extension of the inservice inspection frequency requested by the licensee in their proposed alternative.

In evaluating the licensee's first technical basis for use of the proposed alternative, the NRC staff notes that the licensee uses MRP-375. This document, in part, summarizes numerous Alloy 690/152/52 crack growth rate data from various sources to develop IFs for the crack growth rate equations provided in MRP-55 and MRP-115. While the NRC staff finds the licensee's assertions and/or interpretations to be reasonable, this safety evaluation does not constitute approval of MRP-375. The NRC staff has not validated all of the data used by the MRP-375 document, and the NRC staff does not consider it appropriate to use all of the data from that document to review the licensee's relief request. In conjunction with information provided by the licensee, the NRC staff review will rely upon Alloy 690/152/52 crack growth rate data from two NRC contractors: Pacific Northwest National Laboratory (PNNL) and Argonne National Laboratory (ANL). This data is documented in a data summary report and can be found under ADAMS Accession No. ML14322A587. The NRC confirmatory research generally supports the contention that the crack growth rate of alloy 690/52/152 is more crack resistant but differs from the MRP-375 data in some respects.

The PNNL and ANL data summary report includes crack growth rate data up to approximately 20% cold work based on the observation of local strains in welds and weld dilution zone data. However, the NRC staff did not consider the weld dilution zone data in its assessment. This is because the limited weld dilution zone data that is currently available has shown higher crack growth rates than are commonly observed for alloy 690/152/52 material. The high crack growth rates in weld dilution zones may be due to the reduced chromium present in these areas. The NRC staff chose to exclude the weld dilution zone data from this analysis due to the limited number of data points available, the variability in results, and due to the limited area of continuous weld dilution for flaws to grow through. For example, in the case of the highest measured crack growth rates, a flaw would have to travel in the heat affected zone of a j-groove weld along the low alloy steel head interface. It is not fully apparent to the NRC staff how accelerated crack growth in very small areas of weld dilution zone would result in a significantly increased probability of leakage or component failure during a relatively short extension of the required inspection interval. Exclusion of these data may be reevaluated as additional data become available; a better understanding of the existing data is obtained; or if a longer extension of the inspection interval is requested. Upon evalution of the licensee's first basis for use of this proposed alternative, the NRC staff finds that the impact of these weld dilution zone crack growth rates on the change in volumetric inspection frequency is not relevant.

In evaluating the licensee's second basis for use of the proposed alternative, the NRC staff finds that the past bare metal visual examination on the head under consideration is a reasonable means to demonstrate the absence of leakage through the nozzle/J-groove weld prior to the time the examination was conducted. The NRC staff also finds that performance of future bare metal visual examinations in accordance with CC N-729-1, is adequate to demonstrate the absence of leakage at or prior to the time the examinations are conducted. Finally, the NRC staff finds that the CC N-729-1 frequency for bare metal visual examinations in conjunction with the revised frequency of volumetric examinations is sufficient to provide reasonable assurance of the structural integrity of the RVCH.

In evaluating the licensee's third basis for use of the proposed alternative, the NRC staff found that the licensee's calculated improvement factor of 9, to support an extension of the ASME Code Case N-729-1 inspection frequency of 2.25 RIY to 15 calendar years, was acceptable by NRC staff calculation. The NRC staff also found that the application of an IF of 9 to the 75th percentile curves in MRP 55 and 115 bounded essentially all of the NRC data included in the PNNL and ANL data summary report. Therefore, the NRC staff found that this analysis supports the concept that a volumetric inspection interval for the RVCH of not more than 15 calendar years does not pose a higher risk than that associated with an alloy 600/182/82 RVCH inspected at intervals of 2.25 RIY. Hence, the NRC staff found the licensee's technical basis to be acceptable.

Therefore, the NRC finds that the proposed alternative provides an acceptable level of quality and safety as required by 10 CFR 50.55a(a)(z)(1).

4.0 CONCLUSION

As set forth above, the NRC staff has determined that the alternative method proposed by the licensee in RR N2-I4-NDE-002 will provide an acceptable level of quality and safety for the examination frequency requirements of the reactor vessel closure head. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements

set forth in 10 CFR 50.55a(a)(z)(1). Therefore, the NRC staff authorizes the one time use of Relief Request RR N2-I4-NDE-002 at NAPS, Unit 2 for the duration up to and including the 28th refueling outage that is scheduled to commence in Spring 2022 and which will occur in the fifth ten-year ISI inspection interval.

All other requirements of the ASME Code, Section XI, and 10 CFR 50.55a(g)(6)(ii)(D) for which relief was not specifically requested and authorized herein by the NRC staff remain applicable, including the third party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: Margaret T. Audrain

Date: June 25, 2015

If you have any questions, please contact the Project Manager, V Sreenivas, at 301-415-2597 or via e-mail at V.Sreenivas@nrc.gov.

Sincerely,

/RA/

Robert J. Pascarelli, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-339

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Safety Evaluation

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