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

March 1, 2016

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. 1, RELIEF REQUEST NO. N1-I4-SPT-006, INSERVICE INSPECTION PROGRAM (CAC NO. MF6250)

Dear Mr. Heacock:

By letter dated May 19, 2015, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15147A016), as supplemented by letter dated September 9, 2015 (ADAMS No. ML15258A183), Virginia Electric and Power Company (the licensee) submitted an alternative to the inservice inspection (ISI) requirements of the American Society of Mechanical Engineers *Boiler and Pressure Vessel Code* (ASME Code), Section XI, for North Anna Power Station (NAPS), Unit 1.

Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55(a)(3)(ii), relief request N1-I4-SPT-006 proposed an alternative on the basis that compliance with ASME Code Section XI would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Specifically, the licensee requests relief from the ASME Code, Section XI, IWB-5222(b) requirement to extend the reactor coolant pressure boundary (RCPB) for the system leakage test to be conducted at or near the end of the ISI interval on a specific Class 1 piping segment.

The paragraph headings in 10 CFR 50.55a were change by *Federal Register* Notice dated November 5, 2014 (79 FR 65776), which became effective on December 5, 2014 (e.g., 10 CFR 50.55a(a)(3)(i) is now 50.55a(z)(1), and 50.55a(a)(3)(ii) is now 50.55a(z)(2)). See the cross reference tables, which are cited in the notice, in the ADAMS Accession No. ML14015A191 and ADAMS package Accession No. ML14211A050.

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the subject request and concludes, as set forth in the enclosed safety evaluation, that the proposed alternative in accordance with 10 CFR 50.55a(z)(2) is authorized by law and provides a reasonable assurance of structural integrity and that complying with specified requirements would result in a hardship without a compensating increase in the level of quality and safety. Therefore, the NRC staff grants Relief Request N1-I4-SPT-006 at NAPS Unit 1 for the remainder of the fourth 10-year ISI interval, which ends on April 30, 2019.

All other ASME Code, Section XI, requirements for which alternatives or relief was not specifically requested and approved in the subject request remain applicable, including a third party review by the Authorized Nuclear Inservice Inspector.

The detailed results of the NRC staff review are provided in the enclosed safety evaluation. If you have any questions, please contact Mr. V Sreenivas of my staff at 301-415-2597 or by electronic mail at V.Sreenivas@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 No. 50-338

Enclosure: Safety Evaluation

cc w/encl: Distribution via Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FOURTH 10-YEAR INSERVICE INSPECTION INTERVAL

REQUEST FOR RELIEF NO. N1-I4-SPT-006

NORTH ANNA POWER STATION, UNIT 1

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-338

1.0 INTRODUCTION

By letter dated May 19, 2015, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15147A016), as supplemented by letter dated September 9, 2015 (ADAMS Accession No. ML15258A183), Virginia Electric and Power Company (the licensee) submitted an alternative to the inservice inspection (ISI) requirements of the American Society of Mechanical Engineers *Boiler and Pressure Vessel Code* (ASME Code), Section XI, for North Anna Power Station (NAPS), Unit 1.

Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55(a)(3)(ii), relief request N1-I4-SPT-006 proposed an alternative on the basis that compliance with ASME Code Section XI would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Specifically, the licensee requests relief from the ASME Code, Section XI, IWB-5222(b) requirement to extend the reactor coolant pressure boundary (RCPB) for the system leakage test to be conducted at or near the end of the ISI interval on a specific Class 1 piping segment.

In this relief request, the licensee proposes an alternative to the requirements of article IWB- 5222(b) of Section XI of the ASME Code pursuant to 10 CFR 50.55a(a)(3)(ii). The NRC notes that on November 5, 2014, the NRC reorganized 10 CFR 50.55a (79 FR 65776), and relief requests that had been previously covered by 10 CFR 50.55a(a)(3)(i) are now covered under the equivalent 10 CFR 50.55a(z)(1) and relief requests previously covered by 10 CFR 50.55a(a)(3)(ii) are now covered under the equivalent 10 CFR 50.55a(z)(2).

2.0 REGULATORY EVALUATION

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in the ASME Code, Section XI, "Rules for ISI of

Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components.

Pursuant to 10 CFR 50.55a(z), alternatives to the requirements of paragraph (g) of 10 CFR 50.55a may be used when authorized by the Director, Office of Nuclear Reactor Regulation. A proposed alternative must be submitted and authorized prior to implementation. The licensee must demonstrates (1) the proposed alternatives would provide 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.

Based on the above, and subject to the following technical evaluation, the NRC staff finds that regulatory authority exists for the licensee to request and the NRC to authorize the alternative requested by the licensee.

3.0 TECHNICAL EVALUATION

3.1 System/Component(s) Affected:

The components affected are ASME Code Class 1 Normal Pipe Size (NPS) 2" (two-inch) NPS 2" Auxiliary Spray Piping, which is part of the extended Class 1 RCPB that requires pressurization each 10-year inspection interval. The components are part of the chemical volume control system (CVCS), and in accordance with IWB-2500 (Table IWB-2500-1), they are classified as Examination Category B-P, Item B 15.10. The segments include:

- Segment Boundary (valve-to-valve) from 1-CH-HCV-1311 to 1-CH-328
- Approximate Length – 100 ft.
- Drawing: 11715-CBM-095C-4, Sheet 1 of 2

3.2 Applicable Code Edition and Addenda

The ASME Code of Record for the fourth 10-year ISI interval at North Anna Power Station, Unit 1, is the 2004 Edition with no Addenda.

3.3 ASME Code Requirements (As stated by the licensee)

IWB-5222(b) states, "The pressure retaining boundary during the system leakage test conducted at or near the end of each inspection interval shall extend to all Class 1 pressure retaining components within the system boundary." IWB-5221(a) states, "The system leakage test shall be conducted at a pressure not less than the pressure corresponding to 100% rated reactor power." The paragraphs require pressurization at reactor coolant system nominal operating pressure for this portion of the extended Class 1 boundary at or near the end of the interval.

3.4 Duration of Relief Request

The licensee submitted this relief request for the forth 10-year ISI interval which began on May 1, 2009, and ends on April 30, 2019.

3.5 Licensee's Basis for Requesting Relief (As sated by the licensee)

> Pressurizer pressure is maintained via normal pressurizer spray, which uses the Reactor Coolant Pumps (RCP). Normal pressurizer spray is controlled by the pressurizer Pressure Control System which automatically controls the pressurizer environment. The primary purpose of the auxiliary spray line is for pressure control when the RCPs are not running (i.e., during a post accident condition when it is desired to decrease Reactor Coolant System (RCS) pressure). Operation of the auxiliary spray line at hot standby or power would lead to an unnecessary plant transient. To meet Code requirements, the normally closed upstream isolation valve 1-CH-HCV-1311 must be opened to pressurize the subject pipe segment. Water in this line is supplied from the Charging System which operates at a pressure slightly greater than the RCS normal operating pressure. Therefore, opening of valve 1-CHHCV-1 311 at hot standby or power would increase pressurizer spray flow which will cause an adverse reduction in RCS pressure. In addition, this piping segment is at containment ambient temperature and with the RCS at normal operating temperature, this test would create a thermal shock transient in the spray piping and spray nozzle.

3.6 Proposed Alternative (As stated by the licensee)

> The RCPB pipe segment described in this request is a portion of the auxiliary pressurizer spray line, which is not normally pressurized. This request proposes to perform an ASME Code Section XI, Table IWB-2500-1 and IWB-5221 system leakage test with the isolation valve 1-CH-HCV-1311 in the normally closed position as an alternative to the system leakage test requirements of IWB-5222(b) for this piping segment. This examination will be performed at nominal operating pressure associated with 100% reactor power after satisfying the ASME Code required hold time.

In its letter dated May 19, 2015, the licensee stated that,

> Testing this pipe segment at full RCS pressure does not provide a compensating increase in the level of quality or safety for the following reasons:
>
> 1. The design pressure rating of this pipe segment is the same as the RCPB; however the operating pressure of the pipe segment is well below the normal RCS operating pressure.
>
> 2. This segment is isolated from the RCS pressure during normal operating conditions.
>
> 3. This segment is subject to ASME Code required VT-2 visual examination. This examination is performed with the segment isolated from the RCS and the RCS at its normal operating pressure and temperature. This examination is performed each refueling outage and is sufficient to identify any structural defects that could potentially challenge the integrity of the segment during normal operation.

In its letter dated September 9, 2015, the licensee provided further information in response to NRC staff request for additional information (RAI):

- In response to RAI-2, the licensee provided a list of materials for the auxiliary spray piping and discussed the pipe construction. The auxiliary spray piping is constructed of the materials A376-TP316, S/160, A403-WP316, S/160, BW;A182-F316, S/160;A351 GR CF8 that meets the design pressure and temperature ratings of 2735 psig and 650°F, respectively.

Component	Material
Piping 2" NPS	A376-TP316, S/160
Pipe Fittings (elbows/reducers)	A403-WP316, S/160, BW
1-CH-328	A182-F316,S/160
1-CH-HCV-1311	A351 GR CF8

- In its response to RAI-3, the licensee stated, that there are 23 butt welded connections in the auxiliary spray piping, none of which have shown signs of degradation due to fatigue or stress corrosion cracking. No operating experience related to degraded welded connections on this line was identified, partially due to the fact that because the potential for thermal shock during the performance of system pressure testing has been recognized.

- In response to RAI-4, the licensee stated, in part, that the butt-welded locations along the subject piping segment, which are considered high safety significant (HSS) and classified with no degradation mechanism in the Risk-Informed ISI Program, have not been selected for the Fourth Inspection Interval at this time. Prior to implementation of the Risk-Informed ISI Program, at various times throughout the plants operation, several of the welds have had surface examinations, liquid penetrant examinations performed with no recordable indications.

- In response to RAI-5, the licensee stated that there is no history of leakage identified in the subject piping segment between valves 1-CH-HCV-1311 and 1-CH-328.

- In response to RAI-6, the licensee stated that all of the auxiliary spray piping is insulated. Although no examination hold time is required for Category B-P components, a 4 hour hold time is used to meet the required IWA-5213 insulated component hold time requirements for the other Examination Categories.

- In response to RAI-7, the licensee stated that the subject piping segment between 1-CH-HCV-131 1 and 1-CH-328 is assumed to be pressurized to RCS pressure due to minor valve leakage. However; since no connection is available for mounting temporary instrumentation this assumption has not been validated.

- In response to RAI-8, the licensee stated that a plant modification would be required to install the appropriate vents/drains to facilitate use of an external pump to pressure test the auxiliary spray piping segment.

- In response to RAI-9, the licensee stated that approval of the proposed Inservice Inspection Alternative N1-I4-SPT-006 would minimize the total dose received by craft personnel required to install test connections and examination personnel performing the inspections. Historically, the contact dose rates have been 400 - 1500 mrem/hr (1-CH-328) and 50 - 250 mrem/hr (1-CH-HCV-1311).

4.0 NRC STAFF EVALUATION

The NRC staff has evaluated N1-I4-SPT-006 pursuant to 10 CFR 50.55a(z)(2). The NRC staff focuses on whether compliance with the specified requirements of 10 CFR 50.55a(g), or portions thereof, would result in hardship or unusual difficulty, without a compensating increase in the level of quality and safety.

IWB-5221(a) requires that the system leakage test be performed at a pressure not less than the pressure corresponding to 100% rated reactor power. Additionally, IWB-5222(b) requires that the pressure retaining boundary during the system leakage test conducted at or near the end of each inspection interval extend to all Class 1 pressure-retaining components within the system boundary. The licensee has proposed an alternative to the system leakage test requirements of the ASME Code for the line segments based on hardship.

The licensee stated that this pipe is normally separated from the RCS through double isolation, and is, therefore, unpressurized. The licensee also stated that due to minor valve leakage, this pipe is believed to be at normal RCS pressure (this cannot be verified because there are no test taps on the pipe segment to allow for actual measurement). The NRC staff finds that the design pressure for the pipe is in excess of the RCS pressures, so that it is acceptable for this segment to be either unpressurized or at normal RCS pressure. The NRC staff also finds the four-hour hold time for the VT-2 examinations to allow leakage to become visible as acceptable.

The licensee states that the auxiliary pressurizer line includes at least one manual valve which provides an isolation point to obtain double isolation of the RCPB, and these valves are generally maintained closed during normal operation. In order to perform the ASME Code-required test on the subject pipe segment, it would be necessary to manually open the inboard valves to pressurize the line segment. Pressurization by this method would defeat the RCS double isolation, resulting in potential safety concerns. The NRC staff finds this to be an unnecessary hardship without a compensating increase in quality and safety. In addition, approval of this proposed alternative would minimize the total dose received by the examination personnel performing the inspections.

The licensee also states that there is no way to safely pressurize this segment of pipe with an external pump. Pressurizing the segment with an external pump with the reactor at hot standby or operating conditions would cause coolant to be directed to the pressurizer spray head. This would cause a transient condition within the reactor by unnecessarily lowering the core pressure. The NRC staff finds this to be an unnecessary hardship without a compensating increase in quality and safety.

Based on the material of construction, low usage service conditions, possible incurred radiation dose, and the ASME Code-compliant VT-2 examination of the segments performed each outage, the NRC staff concludes that there is reasonable assurance of structural integrity. The NRC staff concludes that imposition of the ASME Code requirement to extend pressure retaining boundary to all Class 1 components within the system boundary for the system leakage test at the end of the ISI interval would result in hardship without a compensating increase in the level of quality and safety.

5.0 CONCLUSION

As set forth above, the NRC staff concludes that complying with the specified ASME Code requirement to extend the pressure retaining boundary to this Class 1 component within the system boundary for system leakage tests at or near the end of the interval for the Class 1 piping segment described would result in hardship to the licensee without a compensating increase in the level of quality and safety. The NRC staff also concludes the proposed alternatives provide a reasonable assurance of structural integrity. Therefore, pursuant to 10 CFR 50.55a(z)(2), the NRC authorizes the use of Relief Request N1-I4-SPT-006 for the remainder of the fourth 10-year ISI interval for NAPS Unit 1. The fourth 10-year ISI interval for NAPS Unit 1 began on May 1, 2009, and is scheduled to be completed on April 30, 2019.

All other requirements of the ASME Code, Section XI for which relief has not been specifically requested and approved remain applicable, including a third party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: Donald Becker

Date: March 1, 2016

The detailed results of the NRC staff review are provided in the enclosed safety evaluation. If you have any questions, please contact Mr. V Sreenivas of my staff at 301-415-2597 or by electronic mail at V.Sreenivas@nrc.gov.

Sincerely,

/RA/

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

Docket No. 50-338
Enclosure: Safety Evaluation

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