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

August 1, 2013

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

SUBJECT: SURRY POWER STATION, UNIT NOS 1 AND 2 - RELIEF FROM THE REQUIREMENTS OF THE ASME CODE (TAC NOS. MF0329 AND MF0330)

Dear Mr. Heacock:

By letter dated November 9, 2012, as supplemented by letter dated January 14, 2013, the Virginia Electric and Power Company, Dominion (the licensee), submitted Relief Requests CMP-008 and CMP-010. Relief Requests CMP-008 and CMP-010 propose the use of alternate root mean square (RMS) error criteria for sizing flaws that are greater than the requirements of American Society of Mechanical Engineers Boiler and Pressure Vessel Code Case N-695, "Qualification Requirements for Dissimilar Metal Piping Welds," (N-695) at Surry Power Station, Units 1 and 2.

Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55a(g)(5)(iii), the licensee requested approval to use alternative requirements for the reactor vessel nozzle to safe end dissimilar metal butt end welds examination. The proposed alternative uses an alternate depth-sizing qualification for volumetric examinations of these welds from the inside surface and uses a RMS error criterion for sizing flaws that is greater than that allowed by ASME Code. Pursuant to 10 CFR 50.55a(g)(6)(i), the staff has evaluated the licensee's determination that the Code Case requirements are impractical.

The NRC staff determines that granting relief pursuant to 10 CFR 50.55a(g)(6)(i) is authorized by law and will not endanger life or property or the common defense and security, and is otherwise in the public interest giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(g)(5)(iii). Therefore, the NRC grants relief as described in relief requests CMP-008 and CMP-010, as supplemented by the letter dated January 14, 2013, for Surry Power Station, Units 1 and 2, until the end of their fourth 10-year ISI intervals which are scheduled to end on December 13, 2013 and May 9, 2014, respectively.

If you have any questions, please contact the Project Manager, Karen Cotton at 301-415-1438 or via e-mail at karen.cotton@nrc.gov.

Sincerely,

for

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

Docket Nos: 50-280 and 50-281

Enclosure:
Safety Evaluation

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**NUCLEAR REGULATORY COMMISSION**
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST NUMBERS CMP-008 AND CMP-010

REGARDING THE FOURTH TEN YEAR INSPECTION INTERVAL

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION, UNITS 1 AND 2

DOCKET NUMBER 50-280 AND 50-281

## 1.0 INTRODUCTION

By letter dated November 9, 2012 (Agency Wide Documents Access and Management System (ADAMS) Accession Number ML12341A213) as supplemented January 14, 2013 (ADAMS Accession No. ML13025A084), the Virginia Electric and Power Company, Dominion (the licensee), submitted Relief Requests CMP-008 and CMP-010. Relief Requests CMP-008 and CMP-010 propose the use of alternate root mean square (RMS) error criteria for sizing flaws that are greater than the requirements of American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Case N-695, "Qualification Requirements for Dissimilar Metal Piping Welds," (N-695) at Surry Power Station Units 1 and 2. Relief Request CMP-008 covers the examination of welds in Unit 1 and Relief Request CMP-010 covers the examination of welds in Unit 2.

## 2.0 REGULATORY EVALUATION

In its letter dated November 9, 2012, the licensee requested relief from the 0.125 inch RMS error depth sizing acceptance criteria contained in ASME Code Case N-695 pursuant to 10 CFR 50.55a(g)(5)(iii).

ASME Code Case N-695 is accepted for use in Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.147, Revision 16, "Inservice Inspection Code Case Acceptability ASME Section XI, Division 1," and incorporated by reference in 10 CFR 50.55a(b).

10 CFR 50.55a(g)(4)(ii) states, in part, that inservice examination of components conducted during 120-month intervals must comply with the latest edition and addenda of the Code incorporated by reference in 10 CFR 50.55a(b) 12 months before the start of the 120-month inspection interval or the optional ASME Code cases listed in NRC RG 1.147.

10 CFR 50.55a(g)(5)(iii) states, in part, that if the licensee has determined that conformance with a code requirement is impractical, the licensee shall notify the NRC and submit information to support the determinations.

Enclosure

10 CFR 50.55a(g)(6)(i) states, in part that the Commission will evaluate determinations under paragraph (g)(5) of this section that code requirements are impractical and that the Commission may grant such relief and may impose such alternative requirements as it determines is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest giving due consideration to the burden upon the licensee if the requirements were imposed upon the licensee.

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 Commission to grant the relief requested by the licensee.

## 3.0 TECHNICAL EVALUATION

### 3.1 The Licensee's Relief Request

The components covered by Relief Requests CMP-008 and CMP-010 are large-bore reactor inlet and outlet dissimilar metal welds. The welds are all ASME Code Class 1 and are Inspection Category B-F, Item B5.10 "Reactor Vessel Nozzle to Safe-End Butt Welds". The welds are covered by the licensee's risk informed ISI program as Category R-A, Item R1.11, "Elements Subject to Thermal Fatigue".

The vessel material in both cases is SA508 Class 2 ferritic steel. The weld metal is described as austenitic stainless steel and not a nickel alloy weld metal such as alloy 182 or 82.

Table 1: Weld Covered By RR CMP-008 for Surry Power Station Unit 1

| Component Designation | Location | ID | Thickness | Safe End Material |
|---|---|---|---|---|
| 29"-RC-1-2501 R-1-01DM | Loop A Hot Leg | 29" | 2.70" | ASTM A-376 TP 316 |
| 27.5"-RC-3-2501 R-1-17DM | Loop A Cold Leg | 27.5" | 2.56" | SA351 CF8M |
| 29"-RC-4-2501 R-1-01 DM | Loop B Hot Leg | 29" | 2.70" | ASTM A-376 TP 316 |
| 27.5"-RC-6-2501 R-1-17DM | Loop B Cold Leg | 27.5" | 2.56" | SA351 CF8M |
| 29"-RC-7-2501 R-1-01 DM | Loop C Hot Leg | 29" | 2.70" | ASTM A-376 TP 316 |
| 27.5"-RC-9-2501 R-1-17DM | Loop C Cold Leg | 27.5" | 2.56" | SA351 CF8M |

Table 2: Weld Covered By RR CMP-010 for Surry Power Station Unit 2

| Component Designation | Location | ID | Thickness | Safe End Material |
|---|---|---|---|---|
| 29"-RC-301-2S01 R-1-01 DM | Loop A Hot Leg | 29" | 2.70" | ASTM A-376 TP 316 |
| 27.5"-RC-303-2501 R-1-17DM | Loop A Cold Leg | 27.5" | 2.56" | SA351 CF8M |
| 29"-RC-304-2501 R-1-01 DM | Loop B Hot Leg | 29" | 2.70" | ASTM A-376 TP 316 |
| 27.5"-RC-306-2501 R-1-17DM | Loop B Cold Leg | 27.5" | 2.56" | SA351 CF8M |
| 29"-RC-307-2501 R-1-01 DM | Loop C Hot Leg | 29" | 2.70" | ASTM A-376 TP 316 |
| 27.5"-RC-309-2501 R-1-17DM | Loop C Cold Leg | 27.5" | 2.56" | SA351 CF8M |

Surry, Unit 1 and Surry, Unit 2, are both currently in their fourth 10-year Inservice Inspection (ISI) intervals. The ASME Boiler and Pressure Vessel Code (ASME Code) of record for both units is the 1998 Edition of Section XI through the 2000 addenda. Section XI Code Case N-695

(Qualification Requirements for Dissimilar Metal Piping Welds) is referenced in the ISI program. This Code Case is listed in Regulatory Guide 1.147, Rev. 16, Table 1 - "Acceptable Section XI Code Cases". In addition, as required by 10 CFR 50.55a, the ASME Section XI, 2001 Edition is used for Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems."

The volumetric examination specified by Examination Category B-F, Item B5.10, "RPV nozzle to safe-end dissimilar metal (DSM) butt welds" will be performed using the ultrasonic (UT) examination method as described in IWA-2232 and Appendix I. Appendix I, I-2220 requires that ultrasonic examination procedures, equipment, and personnel be qualified by performance demonstration in accordance with Appendix VIII. Dominion will be using NRC-approved Code Case N-695, "Qualification Requirements for Dissimilar Metal Piping Welds" which provides an alternative to Appendix VIII. Code Case N-695 provides an alternative to the Appendix VIII, Supplement 10 requirements for the qualification requirements of DM welds. Paragraph 3.3(c) indicates that examination procedures, equipment, and personnel are qualified for depth-sizing when the RMS error of the flaw depth measurements, as compared to the true depths, do not exceed 0.125 inch.

To date, no inspection vendor has been able to achieve the Appendix VIII-required 0.125 inch RMS error for depth sizing of large bore DM welds from the ID. The Performance Demonstration Initiative (PDI), who administers Appendix VIII testing, has given out partial certifications to procedures and personnel capable of detecting and length sizing cracks. The partial certifications include the achieved RMS error for the inspections. While these partial certifications show that the inspections can be very effective at detecting cracks, these partial certifications do not meet Appendix VIII requirements.

To demonstrate the impracticality of achieving an RMS error of 0.125 inches, the licensee stated:

> The most recent attempt at achieving 0.125 inch RMS error was in early 2008. This attempt, as well as previous attempts, did not achieve the required RMS error value. The qualification attempts have been substantial. The attempts have involved multiple vendors, ultrasonic instruments, personnel, and flaw depth-sizing methodologies, all of which have been incapable of achieving the 0.125 inch RMS error value.

The licensee proposes using an alternative depth-sizing RMS error value greater than the 0.125 inch RMS error value stated in ASME Code Case N-695 for the examination of the welds listed above. The licensee proposes to use an RMS error of 0.224 inches (based on the results achieved by the examination vendor) instead of the 0.125 inches required for Code Case N-695. In the event an indication is detected that requires depth-sizing, the difference between the required RMS error and the demonstrated RMS error will be added to the measured through-wall extent for comparison with the applicable ASME Section XI acceptance criteria. As indicated in the supplemental letter dated January 14, 2013, the licensee provided the following:

1. The welds under consideration will be examined using an ultrasonic technique (UT) which is qualified for flaw detection and length sizing.

2. Indications connected to the inside surface with measured depths of less than 50 percent of the wall thickness will be dispositioned in accordance with an indication the

depth of which is the measured depth plus the root mean square error minus 0.125 inches.

3. For indications connected to the inside surface with measured depths of 50 percent of the wall thickness or greater, the indication will be repaired or a flaw evaluation of the observed indication will be performed. In addition to the information normally contained in flaw evaluations, the evaluation will include:

   a. Information concerning the mechanism which caused the crack,

   b. Information concerning the surface roughness/profile in the area of the pipe/weld required to perform the inspection (i.e., for the area in which the examination is required to be performed, the inner surface profile will be provided), and

   c. Information concerning areas in which the UT probe may "lift off" from the surface of the pipe/weld,

   This flaw evaluation shall be submitted to the NRC for review and approval prior to reactor startup.

The NRC staff considers the three items listed above, including the sentence on submitting the subject flaw evaluation to the NRC for review and approval prior to reactor startup to be part of the alternative as proposed by the licensee. Accordingly, this granting of relief is subject to those items and any departure from them would require a further relief request for NRC review and approval.

Relief Requests CMP-008 and CMP-010 would be applicable to the Surry Power Station, Units 1 and 2, until the end of their fourth ten-year ISI interval. The fourth 10-year ISI interval at Surry, Unit 1, is scheduled to end on December 13, 2013 and the fourth 10-year ISI interval at Surry, Unit 2, is scheduled to end on May 9, 2014.

3.2 <u>NRC Staff Evaluation</u>

As described above, the licensee has requested relief from the requirements of ASME Code Case N-695, "Qualification Requirements for Dissimilar Metal Piping Welds". This code case requires that procedures used to inspect welds from the inside diameter (ID) be qualified by performance demonstration. The acceptance criterion established by the code case is an RMS error of not greater than 0.125 inches.

The staff has confirmed that attempts have been made to qualify ID UT inspection procedures since 2002 and that, to date, no inspection vendor has been able to meet the acceptance criteria established by the code case despite the fact that numerous individuals from several companies have attempted to do so. The staff finds that this repeated inability to qualify ID UT inspection techniques in accordance with ASME Code Case N-695 constitutes an impracticality as described in 10 CFR 50.55a(g)(5)(iii).

While no procedures or personnel are qualified to depth size flaws from the ID of the welds, the probability of detection and length sizing capabilities of the partially-certified procedures and personnel from the ID has been demonstrated to be acceptable.

In July 2012, the NRC staff reviewed the proprietary PDI program (administered by Electric Power Research Institute (EPRI)) data used in blind tests. This review of the PDI data was conducted to verify the information and analysis contained in MRP-2012-11 (provided as attachment 1 of ADAMS Accession No. ML120730196). Based on this review, the NRC staff was able to determine that the addition of the industry-proposed correction factor prior to flaw evaluation for flaws less than 50 percent through-wall satisfactorily reduces the effect of the increased sizing error associated with not meeting the 0.125 RMS error required by Supplement 10 of ASME Section XI and ASME Code Case N-695. The licensee has agreed that if any cracks are detected and measured as 50 percent through-wall depth or greater, and to remain in-service without mitigation or repair, the licensee shall submit flaw evaluations to the NRC for review and approval prior to reactor startup. This evaluation will include the inner profile of the weld, pipe, and nozzle in the region at and surrounding the flaw, and an estimate of the percentage of potential surface areas with UT probe lift-off. Requiring NRC approval for restart when a flaw greater than 50 percent through-wall is discovered and is to be left in service without mitigation or repair addresses the staff concerns with the possibilities of large undersizing errors in deep flaws.

The NRC staff finds that for flaws measured at 50 percent or less through-wall depth adding the industry-proposed correction factor (procedure RMS error - 0.125 inch) to the depths of any flaw found by the inspections and obtaining NRC review and approval prior to startup for any flaws measured as greater than 50 percent through-wall depth provides adequate assurance of structural integrity and leak tightness.

Based on the concerted efforts by the industry to meet the acceptance criteria contained in ASME Code, Section XI, Supplement 10 and ASME Code Case N-695 and the difficulties associated with other inspection methods, the NRC staff finds that meeting the 0.125 inch acceptance criterion is impractical and represents a burden to the licensee.

Based on the above evaluation, the NRC staff finds that this alternative provides reasonable assurance of structural integrity and leak tightness of the subject components.

## 4.0 CONCLUSION

As set forth above, the NRC staff determines that granting relief pursuant to 10 CFR 50.55a(g)(6)(i) is authorized by law and will not endanger life or property or the common defense and security, and is otherwise in the public interest giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(g)(5)(iii). Therefore, the NRC staff grants relief as described in relief requests CMP-008 and CMP-010, as supplemented by the letter dated January 14, 2013, for Surry Power Station, Units 1 and 2, until the end of their fourth 10-year ISI intervals which are scheduled to end on December 13, 2013 and May 9, 2014 respectively.

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

Principal Contributor: Stephen Comblidge

Date: August 1, 2013

If you have any questions, please contact the Project Manager, Karen Cotton at 301-415-1438 or via e-mail at karen.cotton@nrc.gov.

Sincerely,

***/RA by JPaige for/***

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

Docket Nos: 50-280 and 50-281

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