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

March 16, 2015

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 NO. 2 – ALTERNATIVES TO THE REQUIREMENTS OF THE ASME CODE AND USE OF A SUBSEQUENT EDITION AND ADDENDA OF THE ASME B&PV CODE, SECTION XI (TAC NO. MF4129)

Dear Mr. Heacock:

By letter dated May 7, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14133A006), as supplemented by letter dated September 18, 2014 (ADAMS Accession No. ML14266A349),Virginia Electric and Power Company (Dominion, the licensee), requested alternatives to the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section XI for examination of the pressurizer surge nozzle inner radius section and system pressure testing of the bottom of the reactor vessel. Additionally, the licensee requested use of a subsequent edition and addenda of the ASME B&PV Code, Section XI for inservice inspection (ISI) examinations of certain components.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a, the licensee submitted two Relief Requests. Relief Request S2-I5-ISI-02 proposes an alternative to the weld examination requirements specified in the 1998 Edition of the ASME Code, Section XI, Table IWB-2500-1 explicitly required for use by the condition in 10 CFR 50.55a(b)(2)(xxi) for Category B-D, Item B3.120 components. Relief Request S2-I5-SPT-01 proposes an alternative to the VT-2/pressure testing requirement of Table IWB-2500-1 of the Section XI, 2004 Edition for Category B-P, Item B15.10 components. Pursuant to 10 CFR 50.55a(z)(2), the licensee requested to use the proposed alternatives on the basis that it would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(g)(4)(iv), the licensee also requested to use the 2007 Edition through the 2008 Addenda of the ASME Code, Section XI, subject to the conditions listed in 10 CFR 50.55a(b), for examinations of Examination Category B-L-1 components at Surry Power Station, Unit 2 (SPS2).

The NRC has determined that authorizing the use of the proposed alternatives in relief request S2-I5-ISI-02 and S2-15-SPT-01 provide reasonable assurance of structural integrity and leak tightness, and that complying with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the 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)(2). Therefore, the NRC authorizes relief pursuant to 10 CFR 50.55a(z) for the fifth 10-year inservice inspection interval at Surry Power Station, Unit 2, which is currently scheduled to end on May 9, 2024.

Also, the NRC has determined that the use of a subsequent Edition of ASME Code, Section XI requirements is acceptable. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(g)(4)(iv). Therefore, the NRC staff authorizes the use of the 2007 Edition through the 2008 Addenda of the ASME Code, Section XI for Examination Category B-L-1 for the fifth 10-year ISI interval at Surry Power Station, Unit 2.

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,

Shaun Williams for

Robert J. Pascarelli, Chief
Plant Licensing Branch 2-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No.: 50-281

Enclosure:
Safety Evaluation

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FIFTH 10-YEAR INSERVICE INSPECTION INTERVAL

RELIEF REQUESTS S2-I5-ISI-02, S2-I5-SPT-01 AND

REQUEST FOR SUBSEQUENT EDITION/ADDENDA

VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)

SURRY POWER STATION, UNIT 2

DOCKET NUMBER 50-281

1.0 INTRODUCTION

By letter dated May 7, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14133A006), as supplemented by letter dated September 18, 2014 (ADAMS Accession No. ML14266A349),Virginia Electric and Power Company (Dominion, the licensee), requested alternatives to the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section XI for examination of the pressurizer surge nozzle inner radius section and system pressure testing of the bottom of the reactor vessel. Additionally, the licensee requested use of a subsequent edition and addenda of the ASME B&PV Code, Section XI for inservice inspection (ISI) examinations of certain components.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a, the licensee submitted two Relief Requests. Relief Request S2-I5-ISI-02 proposes an alternative to the weld examination requirements specified in the 1998 Edition of the ASME Code, Section XI, Table IWB-2500-1 explicitly required for use by the condition in 10 CFR 50.55a(b)(2)(xxi) for Category B-D, Item B3.120 components. Relief Request S2-I5-SPT-01 proposes an alternative to the VT-2/pressure testing requirement of Table IWB-2500-1 of the Section XI, 2004 Edition for Category B-P, Item B15.10 components. The basis for the alternatives is that complying with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(g)(4)(iv), the licensee also requested to use the 2007 Edition through the 2008 Addenda of the ASME Code, Section XI, subject to the conditions listed in 10 CFR 50.55a(b), for examinations of Examination Category B-L-1 components at Surry Power Station, Unit 2 (SPS2).

Enclosure

These requests are associated with the ISI requirements of the ASME B&PV Code, Section XI, for the fifth 10-year ISI interval at SPS2.

2.0 REGULATORY EVALUATION

The 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 Regulatory Guide 1.147.

The 10 CFR 50.55a(z) states, in part, that alternatives to the requirements of paragraphs (b) through (h) may be used, when authorized by the NRC, if the licensee demonstrates that (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 analysis of the regulatory requirements, the NRC staff concludes that regulatory authority exists to authorize the proposed alternatives S2-I5-ISI-02 and S2-I5-SPT-01 pursuant to 10 CFR 50.55a(z)(2).

The licensee also proposes the use of a later edition of the Code in accordance with 10 CFR 50.55a(g)(4)(iv). The NRC staff notes that 10 CFR 50.55a(g)(4)(iv) states:

> Inservice examination of components and system pressure tests may meet the requirements set forth in subsequent editions and addenda of the ASME Code provided that they are incorporated by reference in 10 CFR 50.55a(b), subject to the conditions listed in 10 CFR 50.55a(b), and subject to Commission approval. Portions of editions or addenda may be used provided that all related requirements of the respective editions or addenda are met.

Given that the licensee has proposed the use of a later edition of the ASME Code pursuant to 10 CFR 50.55a(g)(4)(iv) and that 10 CFR 50.55a(g)(4)(iv) specifically permits the use of later editions of the ASME Code subject to technical criteria which will be considered below, the NRC staff finds that regulatory authority exists to authorize the use of a subsequent edition of the ASME Code, as requested by the licensee.

3.0 TECHNICAL EVALUATION

3.1 Relief Request S2-I5-ISI-02

Component Affected

ASME Code Class:	1
Examination Category:	B-D
Item No.:	ASME Code, Section XI, Item B3.110 (2004 Edition) ASME Code, Section XI, Item B3.120 (1998 Edition)
ISI Component ID:	PZR Surge Line Nozzle 2-RC-E-2, Weld 15NIR [nozzle-inside-radius]
Description:	Nozzle Inner Radius Section (Pressurizer Surge Nozzle)

ASME Code Requirements

The current Code of Record for Surry, Unit 2, is the 2004 Edition of the ASME Code with no addenda. The Surry fifth 10-year ISI interval started on May 10, 2014, and is currently scheduled to end on May 9, 2024.

The 2004 Edition of ASME Code, Section XI, Table IWB-2500-1, Category B-D, Item B3.110, requires a volumetric examination of Pressurizer Surge Line Nozzle-to-Vessel Weld. The 2004 Edition of Section XI does not require an examination of the pressurizer surge NIR. However, 10 CFR 50.55a(b)(2)(xxi)(A) mandates use of the 1998 Edition of Section XI for the examination requirements of full penetration welded nozzles in vessels.

Category B-D, Item B3.120 of Table IWB-2500-1 in the 1998 Edition of the ASME Code, Section XI requires a volumetric examination of the NIR section of the pressurizer surge nozzle; however, 10 CFR 50.55a(b)(2)(xxi)(A) allows an enhanced visual (VT-1) on the inside surface in lieu of a volumetric requirement that is performed from the outside surface.

The Licensee's Alternative and Basis for Relief Request

The licensee's proposal is for the pressurizer surge line nozzle-to-vessel inner radius section to be examined as part of the normally scheduled Class 1 system leakage tests during each refueling outage. The surveillance requirements of Technical Specifications (TSs) that determine the reactor coolant system leak rate and the containment atmosphere radioactivity will be met as part of normal reactor operation.

The pressurizer surge line nozzle is located under the pressurizer skirt and is surrounded by 78 heater penetrations. The insulation and cables for the pressurizer heaters, the heater penetrations and cables, and the pressurizer skirt restrict access to the nozzle. The integrally cast nozzles have an irregular profile, a rough surface which can interfere with ultrasonic inspections. Any ultrasonic examination on this nozzle was described as "best effort" by the licensee.

A remote visual inspection would only achieve partial coverage. This examination would be partially obscured by the thermal sleeves, which extends beyond the inside radius area into the volume of the pressurizer.

The licensee estimated the dose to perform the nozzle inner radius inspections in the letter dated May 7, 2014. The dose estimate is 56 man-rem if all 78 heater cables have to be disconnected and pulled back. Temporary shielding is considered impractical as placement of the shielding material would obstruct accessibility to the examination surface.

As described in a letter dated May 20, 2014 (ADAMS Accession No. ML14148A166) Westinghouse performed an evaluation to address the impact of operational transients for Surry to account for insurge/outsurge transients in addition to design transients in the pressurizer lower head. The evaluation is documented in Dominion Technical Report: LR-1020/LR-2020, "License Renewal Project Time-Limited Aging Analyses Review" and in Westinghouse WCAP-15607, "Evaluation of Pressurizer Insurge/Outsurge for Surry and North Anna" (Proprietary Class 2).

The results of the Westinghouse evaluation show that the Cumulative Usage Factor (CUF) for the surge line NIR is 0.29 (inside surface) and 0.11 (outside surface). The CUF estimates are based upon the number of design basis cycles for the pressurizer, which is 200 cycles. The number of cycles is tracked over the lifetime of the plant. As noted in the Surry Updated Final Safety Analysis Report, Section 4.1.4, the 200 cycle estimate has been retained for the 60-year renewed operating license period. These CUFs are considerably less than the design limit of 1.0, showing a low potential for failure in this area.

NRC Staff Evaluation

The licensee is proposing to perform VT-2 examinations of Pressurizer Surge Line Nozzle-to-Vessel Weld as part of the normally scheduled ASME Code, Class 1 system leakage test each refueling outage in lieu of the ASME Code and 10 CFR 50.55a(b)(2)(xxi)(A) requirements. The licensee has stated that in order for the licensee to volumetrically examine Pressurizer Surge Line Nozzle-to-Vessel Weld and pressurizer NIR section, it would have to remove the insulation and heater cables exposing the licensee's personnel to an estimated dose of 56 man-REM and the potential for personnel contamination from newly exposed surfaces.

The requirements for examinations of inner nozzle radii were developed in the ASME Code in reaction to the discovery of thermal fatigue cracks in the inner-radius section of boiling water reactor feedwater nozzles. These thermal fatigue cracks were the result of internal water temperature fluctuations in the feedwater system. The NRC staff is unaware of any operating experience involving degradation (i.e., indications) in pressurizer NIR sections or for any reactor or steam generator NIR sections at pressurized water reactor plants.

The calculated CUF of the surge line NIR for the design basis 200 cycles during the life of the plant is 0.29 (inside surface) and 0.11 (outside surface), which are considerably less than the design limit of 1.0.

The NRC staff finds that the ultrasonic examination limitations described by the licensee (obstructions to search unit manipulation and coverage, irregular O.D. profile, rough surface coupling condition and attenuating grain structure) would likely limit the volumetric examination coverage to below the ASME Code requirement of "essentially 100 percent" and require a relief request from 10 CFR 50.55a(g)(5)(iii) for the missed coverage. Additionally, based on the description of the pressurizer access provided in the licensee's submittal, the alternative VT-1 examination with a remote visual technology would have limited coverage.

The NRC staff also finds that in order for the licensee to volumetrically examine Pressurizer Surge Line Nozzle-to-Vessel Weld and pressurizer NIR section, it would have to remove the insulation and heater cables exposing the licensee's personnel to an estimated dose of 56 man-REM and the potential for personnel contamination from newly exposed surfaces. The NRC staff also finds that use of temporary shielding to mitigate exposure would be impractical because the shielding material would further obstruct the examination surface. These issues pose a hardship on the licensee.

Therefore pursuant to 10 CFR 50.55a(z)(2), the NRC staff has determined that based on the above, the ASME Code-required volumetric examination and/or the optional visual examination discussed in 10 CFR 50.55a(b)(2)(xxi)(A) would impose a hardship on the licensee without a compensating increase in quality and safety. The NRC staff has determined that the required system leakage test, the Boric Acid Corrosion Control Program, the reactor coolant leak detection systems, and the low component CUF, and no industry operational experience of pressurizer NIR material degradation provide reasonable assurance of structural integrity and leak tightness.

3.2 Relief Request S2-I5-SPT-01

Component Affected

ASME Code Class:	1
Examination Category:	B-P
Item No.:	ASME Code, Section XI, Item B15.10 (2004 Edition)
ISI Component ID:	Reactor Vessel
Description:	Bottom of the Reactor Vessel

ASME Code Requirements

The current Code of Record for Surry, Unit 2, is the 2004 Edition of the ASME Code with no addenda. The Surry fifth 10-year ISI interval started on May 10, 2014, and is currently scheduled to end on May 9, 2024.

The 2004 Edition of ASME Code, Section XI, Table IWB-2500-1, Category B-P, Item B15.10, requires a visual examination for leakage (VT-2) during a system leakage test (IWB-5220) every refueling outage prior to plant startup. The system leakage test shall be conducted at a pressure not less than the pressure corresponding to 100% rated reactor power.

The Licensee's Alternative and Basis for Relief Request

Dominion will continue to monitor for potential leakage on the bottom of the reactor vessel with other surveillance requirements and alarms. The applicable TSs requirements will be met throughout the fifth 10-year ISI interval. Surry TS 3.1.C permits 1 gallon per minute (gpm) unidentified leakage and 10 gpm identified Reactor Coolant System (RCS) leakage. RCS leakage rates are calculated daily and trended. Leakage from the bottom of the reactor would be collected in the in-core sump room and discharged to the containment sump. The in-core sump room and containment sump level switches provide alarms and pump control signals that would alert control room operators of a high level condition. Leakage rates or changes which exceeds action levels prompts operators to enter procedures requiring identification of the source(s) of leakage.

The containment atmosphere is monitored by gaseous and particulate radiation monitors that provide indication in the Main Control Room. In addition, samples of the containment atmosphere are analyzed on a monthly basis for particulate, iodine, and noble gases with the results trended to identify any adverse indications.

The license also plans to perform a visual examination (VT-2) with the insulation in place for the bottom of the reactor vessel as soon as conditions allow entry into the in-core area following reactor shutdown.

In the May 7, 2014 submittal Dominion says;

To meet the Section XI pressure and temperature requirements for the system leakage test of the reactor vessel, the reactor containment at SPS2 is required to be at sub-atmospheric pressure. Station administrative procedures require that self-contained breathing apparatus be worn for containment entries under these conditions. This requirement significantly complicates the visual examination (VT-2) of the bottom of the reactor vessel during testing. Access to the bottom of the reactor vessel requires the examiner to descend several levels by ladder and navigate a small entrance leading to the reactor vessel. In addition to these physical constraints, the examiner must contend with extreme environmental conditions, such as, elevated air temperatures due to reactor coolant at temperatures above 500 degrees F and limited air circulation in the vessel cubicle.

The hardship of performing this test/examination arises less from the time constraint created by the use of bottled air or the involved radiation levels, but rather more from conditions that exist during the test. During the test the reactor coolant system is at the operational temperature of greater than 500 degrees F and the containment is sub-atmospheric. Performing the examination in these conditions is complicated by the following factors:

- The need to use a self-contained breathing apparatus (SCBA) with a fullface respirator.
- The need to access the bottom of the vessel under sub-atmospheric conditions which requires the examiner to descend several levels by ladders and to navigate a small hatch wearing the SCBA.
- The physical environment caused by the heat generated by a vessel elevated to a temperature of > 500 degrees F coupled with a lack of ventilation.

These factors increase the safety hazard associated with the examination. To place the examiner under this increased risk and burden is not justified. This combination of conditions does not exist during the refueling outage when the proposed alternative examination would take place. The proposed alternative examination would be performed under conditions that are safer and allow for a more thorough examination.

The licensee also noted that there has been no operational experience at SPS2 with evidence of leakage on the bottom of the reactor vessel. Also, SPS2 performed volumetric examination of the reactor vessel bottom mounted nozzles with no issues being identified in May 2005. SPS2 has implemented the ASME Code Case N-722-1, "Additional Examinations for PWR Pressure Retaining Welds in Class 1 Components Fabricated With Alloy 600/82/182 Materials Section XI, Division 1," bare metal visual examinations (VE) that are conducted every other refueling outage and no leakage issues have been identified.

NRC Staff Evaluation

The NRC staff has evaluated the proposed alternative S2-I5-SPT-01 pursuant to 10 CFR 50.55a(z)(2). The NRC staff focused on whether compliance with the specified requirements of 10 CFR 50.55a(g), or portions thereof, would result in hardship or unusual difficulty, and if there is a compensating increase in the level of quality and safety despite the hardship.

In ASME Code, Section XI, system leakage tests are performed each refueling outage in accordance with ASME Code, Section XI, Table IWB-2500-1, Examination Category B-P, Item B15.10. System leakage tests are conducted at a test pressure not less than the nominal operating pressure associated with 100% rated reactor pressure. The ASME Code requires that a VT-2 examination be conducted by examining the accessible external exposed surfaces of pressure retaining components for evidence of leakage. The VT-2 examinations may be performed on insulated or non-insulated surfaces of components.

The licensee proposed, as an alternative, performance of a VT-2 visual examination for evidence of leakage of the bottom of the reactor vessel every refueling outage as soon as conditions allow entry into the in-core area during reactor shutdown. In addition, the licensee noted that the Surry TS require monitoring of the reactor coolant leak rate, and the leak rates are evaluated and trended on a daily basis. Also containment atmosphere radiation monitors and in-core area and containment sumps provide indication and alarms to control room operators.

The proposed VT-2 examination will look for evidence of boric acid leakage/corrosion, the examination will be performed in accordance with the Code requirements, except that it will be performed while the RCS is pressurized to between 300 and 340 psig while the plant is in a refueling outage and the containment is at atmospheric conditions. Furthermore, in addition to the proposed VT-2 examination performed every outage as well as bare-metal VE performed every other outage, the licensee states that it will continue to ensure that the surveillance requirements that monitor leakage will be satisfied. Lastly, the containment atmospheric radiation monitors and the in-core room and containment sumps have indications and alarms in the control room that trigger operator action for adverse conditions. In addition, the licensee states that it will perform a bare-metal VE every other refueling outage as required by ASME Code-Case N-722-1. The NRC staff has determined that in the event of a leak, the proposed alternative examination provides reasonable assurance of structural integrity and leak tightness.

The NRC staff agrees that the environment in containment during system leakage test makes inspecting the lower RPV head penetrations very difficult and creates unnecessary personnel safety challenges. The licensee's proposed alternative would allow examiners to perform a VT-2 examination in an environment where they are not encumbered by high temperatures and sub-atmospheric conditions that would be present while the plant is at a pressure corresponding to 100% rated reactor power thus allowing time for a more thorough inspection and reducing personnel safety issues. The NRC staff has, therefore, determined that performing the ASME Code-required examinations at the bottom of the reactor vessel during the system leakage test would result in hardship without a compensating increase in the level of quality and safety.

3.3 Request To Use Subsequent Edition/Addenda

<u>The License's Request</u>

The ASME Code components affected by the licensee's request are as follows:
ASME Section XI, Examination Categories:

- B-L-1, Pressure Retaining Welds in Pump Casings

The Code of Record for the current Surry, Unit 2, ISI intervals for Code Class 1, 2 and 3 components is the ASME Section XI, 2004 Edition. The examinations would not be required to be performed under the Code of Record until the last period of the current inspection interval.

In lieu of the current Code of Record, ASME Section XI, 2004 Edition, the licensee proposes using the ASME Section XI, 2007 Edition through the 2008 Addenda for Category B-L-1 components. The licensee noted that examination categories B-L-1 are not included in the 2007 Edition through the 2008 Addenda of the Code.

<u>NRC Staff Evaluation</u>

10 CFR 50.55a(g)(4)(iv) contains 4 criteria which must be met prior to use of a subsequent edition of the ASME Code these criteria are:

1. The proposed edition/addendum of the ASME Code is incorporated by reference in 10 CFR 50.55a(b)

2. The proposed edition/addendum of the ASME Code is subject to the conditions listed in 10 CFR 50.55a(b)

3. The licensee shall request Commission approval to use the proposed edition/addendum of the ASME Code

4. If only portions of editions or addenda are to be used all related requirements of the respective editions or addenda must be met

In evaluating the first criterion, i.e., that the proposed edition/addendum of the Code has been incorporated by reference in 10 CFR 50.55a(b), the NRC staff notes that 10 CFR 50.55a(b)(2) incorporates by reference the ASME Code Section XI from the 1970 Edition through the 1976 Winter Addenda, and the 1977 Edition through the 2007 Edition with the 2008 Addenda, which was proposed by the licensee. Therefore, the NRC finds that the first criterion has been satisfied.

In evaluating the second criterion, i.e., that the conditions listed in 10 CFR 50.55a(b) are satisfied for the specific proposed subsequent edition and addenda of the ASME Code, Section XI, the NRC staff notes that 10 CFR 50.55a(b) sets no conditions on Examination Category B-L-1 of the 2007 Edition, 2008 Addenda of the ASME Code, Section XI. Therefore, the NRC staff finds that the second criterion has been satisfied.

In evaluating the third criterion, i.e., that the licensee shall request Commission approval to use the proposed edition/addendum of the ASME Code, the NRC staff notes that the licensee's relief request constitutes a request to the Commission for approval to use a subsequent edition/addendum of the ASME Code. Therefore, the NRC staff finds that the third criterion has been satisfied.

In evaluating the fourth criterion, i.e., that if portions of subsequent editions or addenda of the ASME Code, Section XI are used, all related requirements of the respective editions or addenda must be met, the NRC staff is satisfied that the licensee has listed all related requirements in

Examination Category B-L-1 of the 2007 Edition, 2008 Addenda of the ASME Code, Section XI. Therefore, the NRC staff finds that the fourth criterion has been satisfied.

Based on the above, the NRC staff finds that the criteria contained in 10 CFR 50.55a(g)(4)(iv) are satisfied and that the licensee's request to use the 2007 Edition, 2008 Addenda of the ASME Code Section XI for Examination Category B-L-1 is acceptable.

4.0 CONCLUSION

As set forth above, the NRC staff determines that authorizing the use of the alternatives presented in relief request S2-I5-ISI-02 and S2-15-SPT-01 provide reasonable assurance of structural integrity and leak tightness, and that complying with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the 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)(2). Therefore, the NRC authorizes relief pursuant to 10 CFR 50.55a(z) for the fifth 10-year inservice inspection interval at Surry Power Station, Unit 2, which is currently scheduled to end on May 9, 2024.

Also as set forth above, the NRC staff determines that the use of a subsequent Edition of ASME Code, Section XI requirements is acceptable. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(g)(4)(iv). Therefore, the NRC staff authorizes the use of the 2007 Edition through the 2008 Addenda of the ASME Code, Section XI for Examination Category B-L-1 for the fifth 10-year ISI interval at Surry Power Station, Unit 2.

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

Principal Contributor(s):

Stephen E. Cumblidge
Keith M. Hoffman

Date: March 16, 2015

Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(2). Therefore, the NRC authorizes relief pursuant to 10 CFR 50.55a(z) for the fifth 10-year inservice inspection interval at Surry Power Station, Unit 2, which is currently scheduled to end on May 9, 2024.

Also, the NRC has determined that the use of a subsequent Edition of ASME Code, Section XI requirements is acceptable. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(g)(4)(iv). Therefore, the NRC staff authorizes the use of the 2007 Edition through the 2008 Addenda of the ASME Code, Section XI for Examination Category B-L-1 for the fifth 10-year ISI interval at Surry Power Station, Unit 2.

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 SWilliams for/

Robert J. Pascarelli, Chief
Plant Licensing Branch 2-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No.: 50-281

Enclosure:
Safety Evaluation

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