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Attn: Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

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License No.: NPF-12

**DOMINION ENERGY SOUTH CAROLINA (DESC)**
**VIRGIL C. SUMMER NUCLEAR STATION (VCSNS) UNIT 1**
**ALTERNATIVE REQUEST TO DEFER ASME CODE SECTION XI INSERVICE INSPECTION EXAMINATIONS FOR PRESSURIZER AND STEAM GENERATOR PRESSURE-RETAINING WELDS AND FULL PENETRATION WELDED NOZZLES**
**RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI)**

By letter dated June 9, 2022 (Agencywide Document Access and Management System Package Accession No. ML22160A477), Dominion Energy South Carolina, Inc. (DESC), submitted an alternative request to the inservice inspection (ISI) requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) for the pressurizer (PZR) and steam generator (SG) welds at Virgil C. Summer Nuclear Station (VCSNS).

Specifically, pursuant to Title 10 of the Code of Federal Regulations, Part 50, Section 55a, Paragraph (z)(1) (10 CFR 50.55a(z)(1)), DESC proposed an increase to the ISI interval for the PZR and SG welds from the current ASME Code, Section XI 10-year requirement, thereby deferring the required ISI examinations for the remainder of current fourth 10-year ISI interval through the fifth 10-year ISI interval ending on December 31, 2033. DESC referred to the results of the probabilistic fracture mechanics (PFM) analyses in the following Electric Power Research Institute (EPRI) non-proprietary reports as the primary basis for the deferral of the ISI examinations:

- EPRI Technical Report 3002015905, "Technical Bases for Inspection Requirements for PWR [Pressurized Water Reactor] Pressurizer Head, Shell-to-Head, and Nozzle-to-Vessel Welds," 2019 (hereinafter referred to as "EPRI report 15905," ADAMS Accession No. ML21021A271).
- EPRI Technical Report 3002015906, "Technical Bases for Inspection Requirements for PWR Steam Generator Class 1 Nozzle-to-Vessel Welds and Class 1 and Class 2 Vessel Head, Shell, Tubesheet-to-Head and Tubesheet-to-Shell Welds," 2019 (hereinafter referred to as "EPRI report 15906," ADAMS Accession No. ML20225A141).
- EPRI Technical Report 3002014590, "Technical Bases for Inspection Requirements for PWR Steam Generator Class 1 Feedwater and Main Steam Nozzle-to-Shell Welds and Nozzle Inside Radius Sections," 2019 (hereinafter referred to as "EPRI report 14590," ADAMS Accession No. ML19347B107).

In an email dated November 3, 2022, from Ed Miller, NRC Senior Project Manager, to Gary Miller and Yan Gao of Dominion Energy, the Nuclear Regulatory Commission (NRC) staff requested additional information to facilitate their review of the subject alternative request. The NRC's request for additional information (RAI) and the DESC responses are provided in Attachment 1 to this letter.

Should you have any questions, please contact Yan Gao at (804) 273-2768.

Respectfully,

James E. Holloway
Vice President – Nuclear Engineering and Fleet Support

Commitments made in this letter:

1. *VCSNS commits to performing the required ISI examinations for these seven welds (PZR welds 1-2100A-1, 1-2100A-5, 1-2100A-8, and SG welds 2-1100-17, 2-1100-15, 2-1100-23, and 2-1100-23IR) in the third Inspection Period of the fifth ISI Interval.*

Attachment:

1. Response to NRC Request for Additional Information

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**Attachment 1**

**Response to NRC Request for Additional Information**

TABLE OF CONTENTS

# RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

# ALTERNATIVE REQUEST TO DEFER ASME CODE SECTION XI INSERVICE INSPECTION EXAMINATIONS FOR PRESSURIZER AND STEAM GENERATOR PRESSURE-RETAINING WELDS AND FULL PENETRATION WELDED NOZZLES

## 1.0 BACKGROUND

By letter dated June 9, 2022 (Agencywide Document Access and Management System Package Accession No. ML22160A477), Dominion Energy South Carolina, Inc. (DESC), submitted an alternative request (AR) to the inservice inspection (ISI) requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) for the pressurizer (PZR) and steam generator (SG) welds at Virgil C. Summer Nuclear Station (VCSNS). The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the AR and generated a request for additional information (RAI) [3.4] in order to complete their review. Section 2 below provides the RAI items and the corresponding responses.

## 2.0 NRC RAIs

### 2.1 RAI Description

*Background*

*By letter dated June 9, 2022 (Agencywide Document Access and Management System Accession Number ML22160A477), Dominion Energy South Carolina, Inc. (the licensee) submitted to the United States Nuclear Regulatory Commission (NRC), a proposed alternative to the inservice inspection (ISI) requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) of the pressurizer (PZR) and steam generator (SG) welds of Virgil C. Summer Nuclear Station Unit 1 (VCSNS).*

*Specifically, pursuant to Title 10 of the Code of Federal Regulations, Part 50, Section 55a, Paragraph (z)(1) (10 CFR 50.55a(z)(1)), the licensee is proposing to increase the ISI interval for the PZR and SG welds of VCSNS from the current ASME Code, Section XI 10-year requirement, thereby deferring the required ISI examinations for the remainder of current fourth 10-year ISI interval through the fifth 10-year ISI interval ending on December 31, 2033. The licensee referred to the results of the probabilistic fracture mechanics (PFM) analyses in the following Electric Power Research Institute (EPRI) non-proprietary reports as the primary basis for the deferral of the ISI examinations:*

- *EPRI Technical Report 3002015905, "Technical Bases for Inspection Requirements for PWR [Pressurized Water Reactor] Pressurizer Head, Shell-to-Head, and Nozzle-*

*RAI texts are italic and underlined

*to-Vessel Welds," 2019 (hereinafter referred to as "EPRI report 15905," ADAMS Accession No. ML21021A271).*

- *EPRI Technical Report 3002015906, "Technical Bases for Inspection Requirements for PWR Steam Generator Class 1 Nozzle-to-Vessel Welds and Class 1 and Class 2 Vessel Head, Shell, Tubesheet-to-Head and Tubesheet-to-Shell Welds," 2019 (hereinafter referred to as "EPRI report 15906," ADAMS Accession No. ML20225A141).*
- *EPRI Technical Report 3002014590, "Technical Bases for Inspection Requirements for PWR Steam Generator Class 1 Feedwater and Main Steam Nozzle-to-Shell Welds and Nozzle Inside Radius Sections," 2019 (hereinafter referred to as "EPRI report 14590," ADAMS Accession No. ML19347B107).*

*Regulatory Basis*

*The NRC has established requirements in 10 CFR Part 50 to protect the structural integrity of structures and components in nuclear power plants. Among these requirements are the ISI requirements of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a to ensure that adequate structural integrity of PZR and SG vessels (including their welds) is maintained through the service life of the vessels. Therefore, the regulatory basis for the following requests for additional information (RAIs) has to do with demonstrating that the proposed alternative ISI requirements would ensure structural integrity of the PZR and SG welds of VCSNS, and thereby would provide an acceptable level of quality and safety per 10 CFR 50.55a(z)(1) for the welds.*

*RAI-1*

*Issue*

*The submittal references probabilistic and deterministic analyses in the above EPRI reports to estimate potential fatigue crack growth in the subject PZR and SG welds and to justify application of these analyses to the proposed ISI interval extension for the welds. The licensee presented plant-specific information to demonstrate that the referenced analyses in the EPRI reports would bound the subject PZR and SG welds, including the ISI history of the welds.*

*Leveraging PFM analyses to define the basis for risk-informing inspection requirements requires knowledge of both the current and future behavior of the material degradation and the associated uncertainties applicable to the subject PZR and SG welds. Confidence in the results of these analyses hinges on the assurance that the PFM model adequately represents, and will continue to represent, the degradation behavior in the subject PZR and SG welds. The NRC staff has determined that, when considering extended examination intervals, adequate performance monitoring through inspections is needed to ensure that the assumptions of the PFM model remain valid, and that novel or unexpected degradation is detected and dispositioned in a timely fashion. In Section 5.0 of the submittal, the licensee stated that some of the subject PZR and SG welds have not yet received the fourth ISI interval examinations and that the required ASME Code, Section XI examinations will be performed during the sixth ISI interval ending on*

*December 31, 2043. Given this inspection scenario, the VCSNS inspection history in Attachment 4 to the submittal, and the licensee's proposed extended examination intervals, the staff noted that some of the subject PZR and SG welds will be in operation for more than 20 years without inspections. Specifically, these welds are: 1-2100A-1, 1-2100A-5, 1-2100A-8 in the PZR, and 2-1100-17, 2-1100-15, 2-1100-23, and 2-1100-23IR in the SG. The licensee, however, did not provide a performance monitoring scheme for these welds.*

*The licensee discusses the system leakage test as "providing further assurance of safety" for the proposed alternative. However, the NRC staff notes that the visual examinations performed during system leakage tests may not provide sufficient information to ensure that the PFM model continues to predict the material behavior and that emergent degradation is discovered and dispositioned in a timely fashion. Specifically, visual examinations may not directly detect the presence or extent of degradation; may not provide direct detection of aging effects prior to potential loss of structure or intended function; and do not provide sufficient validating data necessary to confirm the modeling of degradation behavior in the subject PZR and SG welds.*

*Request*

*For PZR welds 1-2100A-1, 1-2100A-5, 1-2100A-8, and SG welds 2-1100-17, 2-1100-15, 2-1100-23, and 2-1100-23IR:*

a. *Describe the performance monitoring that will be implemented with this proposed alternative to ensure that the PFM model adequately represents, and will continue to represent, the degradation behavior in the subject components commensurate with the duration of the requested alternative. The three VCSNS PZR welds mentioned above are of particular concern because these welds are in the region of the PZR that is subject to additional thermal cycling due to insurge/outsurge transients.*

b. *Explain how this performance monitoring will provide, over the extended examination interval, (1) direct evidence of the presence and extent of degradation, (2) validation and confirmation of the continued adequacy of the PFM model; and (3) timely detection of novel or unexpected degradation.*

c. *If through this performance monitoring indications are detected that exceed the acceptance standards of ASME Code, Section, IWB-3500, confirm that they will be evaluated as required by ASME Code, Section XI (which includes requirements for successive inspections and additional examinations) and describe other actions (if any) specified in the plant's corrective action program to ensure that the integrity of the component is adequately maintained.*

*RAI-2*

*Issue*

*The NRC staff requests the following clarification and explanation on the VCSNS transient information provided in Attachment 2 to the submittal.*

*Request*

a. *Pages 4 and 6 of the attachment states that the VCSNS 60-year projected Heatup and Cooldown cycles is 114, which is different than the number of Heatup/Cooldown cycles of 200 in Tables A5 and A6 of the attachment for VCSNS. Clarify the 60-year Heatup and Cooldown cycle projection for VCSNS.*

b. *Similarly, pages 4 and 6 of the attachment states that the VCSNS 60-year projected reactor trip cycles 147 (5 for Case C), which is different than the number of reactor trip cycles in Tables A5 and A6 of the attachment for VCSNS. Clarify the 60-year reactor trip cycle projection for VCSNS.*

c. *In Table A7 of the attachment, explain why the number of Heatup/Cooldown cycles of 58 for the VCSNS SG feedwater nozzle is different than the VCSNS 60-year projected Heatup/Cooldown cycles (114) stated in pages 4 and 6 of the attachment.*

## 2.2 RAI Responses

### Response to RAI-1

a. Performance monitoring supporting this Alternative Request began with the fourth 10-year ISI Interval which begins on January 1, 2014. Seven of the ten pressurizer welds (70%) and three of the seven steam generator welds/components (43%) included in this request have been examined in the fourth ISI Interval with no degradation detected. The remaining three pressurizer welds and four steam generator welds examinations are scheduled to be performed during the last refueling outage of the fourth interval (fourth interval ends on December 31, 2023). The Alternative Request would allow deferral of these exams. To address the concern of some of the PZR and SG welds being in operation for more than 20 years without inspections, VCSNS commits to performing the required ISI examinations for these seven welds (PZR welds 1-2100A-1, 1-2100A-5, 1-2100A-8, and SG welds 2-1100-17, 2-1100-15, 2-1100-23, and 2-1100-23IR) in the third Inspection Period of the fifth ISI Interval (fifth interval ends on December 31, 2033). Performance monitoring for all required welds will resume with the start of the sixth ISI Interval beginning January 1, 2034, in accordance with the Section XI, 10-year ISI Interval plan.

   Future examinations of these PZR and SG welds throughout the industry will provide additional opportunities to detect known degradation mechanisms, as described in Section 6.0 the EPRI Technical Reports, and will also provide the opportunity to detect any new or unexpected degradation mechanisms that may occur in the future for the subject welds. If a new degradation mechanism is identified during future

industry examinations, VCSNS will follow the industry guidance to address the new degradation mechanism.

Based on the PSI/ISI inspection scenarios for the VCSNS PZR and SG welds and the results of the DFM and PFM analysis of EPRI Technical Reports 3002015905 [3.1], 3002015906 [3.2], and 3002014590 [3.3], DESC believes that the performance monitoring described is adequate.

b. (1) The performance monitoring plan provided in the response above includes inspection sampling that will provide direct evidence of the presence and extent of any degradation over the extended examination interval for these welds.

(2) The components in the proposed alternative have operated for a minimum of 28 years and up to a maximum of 40 years without the identification through inspection of any service-induced degradation. This excellent operating history is validation and confirmation of the conservative nature of the PFM and DFM models used in the EPRI Technical Reports 3002015905 [3.1], 3002015906 [3.2], and 3002014590 [3.3]. This also shows that the models will predict future behavior conservatively.

(3) The performance monitoring schedule described above will provide timely detection of any novel or unexpected degradation in these components.

c. If during the performance monitoring examinations described above, indications are detected that exceed the applicable ASME Code, Section XI acceptance standards, they will be evaluated in accordance with ASME Code, Section XI requirements.

Based on the above discussion, DESC concludes that the performance monitoring examinations for the welds covered by the proposed Alternative Request ensure the safe operation of the components for the duration of the proposed extended examination interval.

**Response to RAI-2**

a. 114 cycles (pages 4 and 6) is the projected number of Heatup/Cooldown cycles over the 60-year operating period. The projected cycles are based on; 1) captured events that fit within design transient parameters, and; 2) evaluated "out of bounds" events that are subsequently classified as design transient events. "Out of bounds" events are captured transients that do not fall within the design transient parameters. VCSNS uses the WESTEMs Thermal Event Monitoring module to track plant transients by monitoring various design transients at multiple locations within the plant. The maximum cycle count for all the locations is then identified for each unique design transient and used to determine the projected cycles for that design transient. The 114 projected Heatup/Cooldown cycle value is based on the cycle count at the steam generator feedwater nozzle.

200 cycles (Tables A5 and A6) is the number of analyzed Heatup/Cooldown cycles that bound what is expected to occur over the life of the plant as defined in the Westinghouse Delta 75 Steam Generator Design Specification. The limit of 200

cycles is used in the Delta 75 steam generator stress analysis to verify structural integrity.

b. Similar to the response above, 147 cycles (pages 4 and 6) is the projected number of Reactor Trip cycles over the 60-year operating period. 400 cycles (Tables A5 and A6) is the number of analyzed Reactor Trip cycles that bound what is expected to occur over the life of the plant as defined in the Westinghouse Delta 75 Steam Generator Design Specification. Table A5 separates the 400 analyzed Reactor Trip cycles into three (3) different Reactor Trip transients (defined by Note 2 in Table A5). Table A6 presents the summation of the three (3) different Reactor Trip transients for a total of 400 analyzed Reactor Trip cycles.

c. 58 cycles (Table A7) is the projected number of Heatup/Cooldown cycles over the 60-year operating period based only on captured events that fit within the design transient parameters at the steam generator feedwater nozzle. This cycle count does not consider "out of bounds" events. An evaluation of "out of bounds" events at the steam generator feedwater nozzle results in 114 projected Heatup/Cooldown cycles over the 60-year operating period. For consistency, Table A7 should be updated to reflect 114 projected Heatup/Cooldown cycles over the 60-year operating period.

## 3.0 REFERENCES

3.1 ADAMS Accession No. ML21021A271, EPRI Technical Report 3002015905, "Technical Bases for Inspection Requirements for PWR Pressurizer Head, Shell-to-Head and Nozzle-to-Vessel Welds," Palo Alto, California, 2019

3.2 ADAMS Accession No. ML20225A141, EPRI Technical Report 3002015906, "Technical Bases for Inspection Requirements for PWR Steam Generator Class 1 Nozzle-to-Vessel Welds and Class 1 and Class 2 Vessel Head, Shell, Tubesheet-to-Head and Tubesheet-to-Shell Welds," Palo Alto, California, 2019

3.3 ADAMS Accession No. ML19347B107, EPRI Technical Report 3002014590, "Technical Bases for Inspection Requirements for PWR Steam Generator Feedwater and Main Steam Nozzle-to-Shell Welds and Nozzle Inside Radius Sections," Palo Alto, California, 2019

3.4 Email from Ed Miller (NRC) to Gary Miller and Yan Gao of Dominion Energy regarding RAI for VC Summer – L–2022–LLR-0052, November 3, 2022