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

December 15, 2022

Mr. Daniel G. Stoddard
Senior Vice President and Chief Nuclear Officer
Innsbrook Technical Center
5000 Dominion Blvd.
Glen Allen, VA 23060-6711

SUBJECT: VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1 - RE: EXTENSION OF STEAM GENERATOR PRIMARY INLET NOZZLE DISSIMILAR METAL WELD INSPECTION INTERVAL (EPID L-2022-LLR-0033)

Dear Mr. Stoddard:

By letter dated March 10, 2022 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML22069B117 (Request) and ML22069B118 (Proprietary Attachment), as supplemented by letter dated October 27, 2022 (ML22300A220), Dominion Energy South Carolina (Dominion, the licensee), submitted a request for the Virgil C. Summer Nuclear Station, Unit No. 1 (Summer), to the U.S. Nuclear Regulatory Commission (NRC or Commission) for a proposed alternative to certain American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPV Code), Section XI.

Specifically, Dominion requested a one-time extension of the Summer steam generator primary inlet nozzle dissimilar metal weld inspection interval from five years to nominally nine years for the volumetric examination. In its supplement dated October 27, 2022, the licensee modified the request for two possible durations for the alternative. The first was through the fall refueling outage (RFO) in 2024 and the second was through the spring RFO in 2026. The NRC staff's review considered the shorter duration through the fall RFO in 2024.

The NRC staff has reviewed the alternative request and concludes, as set forth in the enclosed safety evaluation, that Dominion has adequately addressed all the regulatory requirements set forth in 10 CRR 50.55a(z)(1). All other ASME Code, Section XI requirements for which relief was not specifically requested and approved remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

If you have any questions, please contact the Project Manager, Ed Miller at 301-415-2481 or via e-mail at Ed.Miller@nrc.gov.

Sincerely,

Michael T. Markley

Digitally signed by Michael T. Markley
Date: 2022.12.15 07:57:06 -05'00'

Michael T. Markley, Chief
Plant Licensing Branch 2-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-395

Enclosure:
Safety Evaluation

cc: Listserv

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

EXTENSION OF STEAM GENERATOR PRIMARY INLET NOZZLE

DISSIMILAR METAL WELD INSPECTION INTERVAL

DOMINION ENERGY SOUTH CAROLINA

VIRGIL C. SUMMER NUCLEAR STATION, UNIT 1

DOCKET NO. 50-395

1.0 INTRODUCTION

By letter dated March 10, 2022 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML22069B117 (Request) and ML22069B118 (Proprietary Attachment), as supplemented by letter dated October 27, 2022 (ML22300A220), Dominion Energy South Carolina (Dominion, the licensee), submitted a request for the Virgil C. Summer Nuclear Station, Unit No. 1 (Summer), to the U.S. Nuclear Regulatory Commission (NRC or Commission) for a proposed alternative to certain American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPV Code), Section XI.

2.0 REGULATORY EVALUATION

Adherence to Section XI of the ASME Code is mandated by 10 CFR 50.55a(g)(4), which states, in part, that ASME Code Class 1, 2, and 3 components will meet the requirements, except the design and access provisions and the pre-service examination requirements, set forth in the ASME Code, Section XI.

Paragraph 10 CFR 50.55a(z) states, in part, that alternatives to the requirements of 10 CFR 50.55a(b)-(h) may be used, when authorized by the Director, Office of Nuclear Reactor Regulation, if (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.

The licensee has submitted this request on the basis that a proposed alternative would provide an acceptable level of quality and safety in accordance with 10 CFR 50.55a(z)(1).

Enclosure

3.0 TECHNICAL EVALUATION

3.1 Licensee's Proposed Alternative

Applicable Code Edition and Addenda

The applicable Code Edition and Addenda is the ASME Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, 2007 Edition through 2008 Addenda.

American Society of Mechanical Engineers (ASME) Code Components Affected

The request is applicable to the following components and associated dissimilar metal welds (DMW):

- 1-4100A-31 (DM) – 'A' SG Inlet Nozzle to Safe End DMW
- 1-4200A-28 (DM) – 'B' SG Inlet Nozzle to Safe End DMW
- 1-4300A-29 (DM) – 'C' SG Inlet Nozzle to Safe End DMW

ASME Code Requirement for Which Alternative Is Requested

Paragraph 10 CFR 50.55a(g)(6)(ii)(F)(1) requires licensees to implement the requirements of ASME BPV Code Case N770-5, subject to the conditions specified in paragraphs (g)(6)(ii)(F)(2) through (16) of the section.

The proposed alternative covers examinations required by American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Case N-770-5, "Alternative Examination Requirements and Acceptance Standards for Class 1 PWR Piping and Vessel Nozzle Butt Welds Fabricated with UNS N06082 or UNS W86182 Weld Filler Material With or Without Application of Listed Mitigation Activities, Section XI, Division 1" (ASME Code Case N-770-5).

Specifically, the examinations are for welds that fall under ASME Code Case N-770-5, Inspection Item A-2: "Unmitigated butt weld at Hot Leg operating temperature < 625 °F (329 C)." Components inspected under Item A-2 require a visual examination each refueling outage and volumetric examination every five years.

Proposed Alternative and Basis for Use

The licensee is proposing a one-time extension of the steam generator (SG) primary inlet nozzle dissimilar metal (DM) weld inspection interval at Virgil C. Summer Nuclear Station (Summer), Unit 1 from five years to nominally six years for the volumetric examination only. No changes are requested to the visual examination requirements of the SG inlet nozzle DM welds. The SG inlet nozzle DM welds were volumetrically examined in the fall of 2018. The next scheduled volumetric examination for these welds is in the spring of 2023. The licensee requested to defer the volumetric examination.

Duration of Proposed Alternative

Dominion requested a one-time extension of the Summer steam generator primary inlet nozzle DMW inspection interval from five years to nominally nine years for the volumetric examination. In its supplement dated October 27, 2022, the licensee modified the request for two possible durations for the alternative. The first was through the fall refueling outage (RFO) in 2024 and the second was through the spring RFO in 2026. The NRC's review considered the shorter duration through the fall RFO in 2024.

3.2 NRC Staff Evaluation

The licensee requested an extension of the inspection interval for the subject N-770-5, Inspection Item A-2 welds from five years to six years. The basis used to demonstrate the acceptability of extending the examination interval for Code Case N-770-5, Inspection Item A-2 components is contained in the site-specific weld crack growth analysis described in Enclosure 3 of the letter dated March 10, 2022, and the crack growth information provided in its supplement letter dated October 27, 2022.

The weld crack growth analyses is intended to demonstrate that the welds possess adequate thickness to protect against failure due to primary water stress corrosion cracking (PWSCC). In the weld crack growth analyses, an inside surface flaw that is 1.7 mm (0.065 in), or fifty percent of the depth of the inlay is postulated in the PWSCC-resistant Alloy 52 inlay. Then, the analysis calculated the amount of time for the flaw to reach the maximum allowable end-of-evaluation period flaw size. This maximum allowable end-of-evaluation period flaw size would be the largest flaw size that could exist in the welds and still be acceptable according to ASME Code, Section XI. Crack growth was calculated based on the PWSCC growth mechanism through both the Alloy 52 inlay and the Alloy 82 DM welds.

Using the same inputs and methodology discussed in the original request, the licensee's supplement letter dated October 27, 2022, provided analyses at two lower Factors of Improvement (15 and 23), and proposed two possible durations for the alternative. The first was through the fall refueling outage (RFO) in 2024 and second was through the spring RFO in 2026. The Nuclear Regulatory Commission (NRC) is currently working with the PWSCC Crack Growth Rate Expert Panel, which is evaluating the crack growth PWSCC data for Alloys 52 and 152, including several data sets not included in the letter dated March 10, 2022. The NRC staff has determined that until the work of the Expert Panel is completed that longer-term alternatives are challenging to support and that the Fall of 2024 is the more appropriate duration for proposed alternative RR-4-26. If this work is completed in the near future, the NRC would consider authorizing longer duration alternative requests.

The results of the analysis justify a longer examination interval for the welds than the five years currently allowed for Code Case N-770-2, Inspection Item A-2 welds. Based on the results for the SG inlet nozzle DM welds hot leg, an examination interval of up to six years is acceptable. The plant-specific PWSCC growth results, in the attached Westinghouse letter and technical report, supports the request for the Summer Unit 1 SG inlet nozzle DM welds to be examined after a duration of six years from the previous refueling outage inspections in fall of 2018. This would allow Summer Unit 1 to perform the inspection during the fall 2024 refueling outage.

In the event that a crack is present and grows faster than predicted by the crack growth analyses, the licensee has programs in place to detect leakage though the welds during outages and during operation. The licensee performs bare metal visual examinations of the subject welds every refueling outage. The licensee has a boric acid corrosion control program that monitors for signs of boric acid deposits. Additionally, the licensee uses containment radiation monitors and monitors containment sump levels plant operation to detect substantial reactor coolant system leakage and/or increases in radiation levels within containment that would occur in the unlikely event of leakage through the welds.

Based on the above, the NRC staff finds that the licensee's analysis demonstrates that it is unlikely that an unacceptable crack would be present during the extended interval. Additionally, if such a crack were to develop, the licensee has monitoring programs and administrative controls in place that would likely detect the crack and allow remedial action to be taken. Therefore, the NRC concludes that re-examination interval provides an acceptable level of quality and safety through the RFO in fall 2024.

4.0 CONCLUSION

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

Principal Contributor: S. Cumblidge, NRR

Date: December 15, 2022

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