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

November 18, 2022

Mr. Shawn Gibby
Vice President of Nuclear Engineering
Duke Energy Corporation
526 S. Church Street, EC-07H
Charlotte, NC 28202

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2; CATAWBA NUCLEAR STATION, UNITS 1 AND 2; MCGUIRE NUCLEAR STATION, UNITS 1 AND 2, AND SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1 – AUTHORIZATION OF RA-19-0352 REGARDING USE OF ALTERNATIVE FOR REACTOR PRESSURE VESSEL HEAD CLOSURE STUD EXAMINATIONS (EPID L-2020-LLR-0156)

Dear Mr. Gibby:

By letter RA-19-0352, dated December 1, 2020, as supplemented by letter dated January 31, 2022, Duke Energy Carolinas, LLC, and Duke Energy Progress, LLC (Duke Energy or the licensee) proposed an alternative to Section XI IWB-2500(a), Table IWB-2500-1, Examination Category B-G-1, Item No. B6.20 of the Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (B&PV Code) for the Brunswick Steam Electric Plant (BSEP), Units 1 and 2, Catawba Nuclear Station (CNS), Units 1 and 2, McGuire Nuclear Station (MNS), Units 1 and 2, and the Shearon Harris Nuclear Power Plant (SHNPP), Unit 1.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(1), the licensee proposed an alternative to the requirement to perform inservice volumetric or surface examinations of Examination Category B-G-1, Item Number B6.20, reactor pressure vessel head (RPVH) closure studs on a 10-year inservice inspection (ISI) interval. The proposed alternative is to extend the frequency of RPVH closure stud volumetric or surface examinations for the remainder of the currently licensed operating periods for BSEP, Units 1 and 2, CNS, Units 1 and 2, MNS, Units 1 and 2, and SHNPP, Unit 1.

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 licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1). Additionally, the NRC has determined that there is reasonable assurance that structural integrity of the reactor vessel head closure studs for the BSEP, Units 1 and 2, CNS, Units 1 and 2, MNS, Units 1 and 2, and SHNPP, Unit 1, will be maintained for the remainder of the current licensed operating periods. Therefore, the NRC staff authorizes the use of the proposed alternative for the facilities requested in the licensee's application for current licensed operating periods for each unit, as shown in Table 1 in Section 3.6 of the attached safety evaluation.

All other ASME B&PV Code, Section XI, requirements which are not modified by the NRC staff's approval of the licensee's request remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

If you have any questions, please contact Michael Mahoney at 301-415-3867 or via e-mail at Michael.Mahoney@nrc.gov.

Sincerely,

David J. Wrona

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Date: 2022.11.18 13:31:21 -05'00'

David J. Wrona, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos.: 50-324, 50-325, 50-369, 50-370, 50-400, 50-413, and 50-414

Enclosure:
Safety Evaluation

cc:

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST RA-19-0352 – REQUEST FOR USE OF ALTERNATIVE FOR REACTOR

PRESSURE VESSEL CLOSURE STUD EXAMINATIONS

DUKE ENERGY CAROLINAS, LLC

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

DUKE ENERGY PROGRESS, LLC

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NOS. 50-413, 50-414, 50-369, 50-370,

50-324, 50-325, and 50-400

1.0 INTRODUCTION

By letter RA-19-0352, dated December 1, 2020, as supplemented by letter dated January 31, 2022 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML20336A033 and ML22032A142, respectively), Duke Energy Carolinas, LLC, and Duke Energy Progress, LLC (Duke Energy or the licensee) requested U.S. Nuclear Regulatory Commission (NRC or Commission) approval to use a proposed alternative to Section XI IWB-2500(a), Table IWB-2500-1, Examination Category B-G-1, Item No. B6.20, of the Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (B&PV Code) for the Brunswick Steam Electric Plant (BSEP), Units 1 and 2, Catawba Nuclear Station (CNS), Units 1 and 2, McGuire Nuclear Station (MNS), Units 1 and 2, and the Shearon Harris Nuclear Power Plant (SHNPP), Unit 1.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(1), the licensee proposed an alternative to the requirement to perform inservice volumetric or surface examinations of Examination Category B-G-1, Item Number B6.20, reactor pressure vessel head (RPVH) closure studs on a 10-year ISI interval. The proposed alternative is to extend the frequency of RPVH closure stud volumetric or surface examinations for the remainder of the currently licensed operating periods for BSEP, Units 1 and 2, CNS, Units 1 and 2, MNS, Units 1 and 2, and SHNPP, Unit 1.

Enclosure

2.0 REGULATORY EVALUATION

Regulatory Requirements

ASME Section XI IWB-2500(a), Table IWB-2500-1, Examination Category B-G-1, Item No. B6.20 requires volumetric examination of the RPVH closure studs once during each Section XI inspection interval, nominally every 10 years.

Additionally, per Note (7) of Table IWB-2500-1 (B-G-1), “when bolts or studs are removed for examination, surface examination meeting the acceptance criteria of IWB-3515 may be substituted for volumetric examination.”

Guidance

NRC Regulatory Guide (RG) 1.65, “Materials and Inspections for Reactor Vessel Closure Studs,” Revision 1 (ML092050716).

3.0 NRC TECHNICAL EVALUATION

3.1 ASME Code Components Affected

Component:	Reactor Pressure Vessel Closure Studs
Code Class:	Class 1
Examination Category:	B-G-1 (Pressure Retaining Bolting, Greater than 2 inch in Diameter)
Code Item Number:	B6.20 – Closure Studs
Description:	Reactor Pressure Vessel Closure Studs volumetric and surface examinations

3.2 Applicable Code Edition and Addenda

2007 Edition of the ASME Code, Section XI, through 2008 Addenda

3.3 Applicable Code Requirements

ASME Code, Section XI, IWB-2500(a), Table IWB-2500-1, Examination Category B-G-1, Item No. B6.20.

3.4 Reason for Request

In letter dated December 1, 2021, the licensee stated the following for the reason for their proposed alternative:

> The Electric Power Research Institute (EPRI) recently developed a technical basis report [8.1] that determined the requirement to perform inservice volumetric or surface exams of RPV closure studs (Examination Category B-G-1, Item No. B6.20) could be eliminated for the current operating periods … without increasing plant risk or posing any safety concerns for the RPV. The EPRI Report provides the supporting technical basis to allow longer inspection intervals for the closure studs while maintaining the appropriate safety margins required by ASME Section XI, Division 1.

3.5 Alternative and Basis for Use

In Section 5.0 of Enclosure 1 to the submittal, the licensee stated that the proposed alternative is to extend the frequency of the RPVH closure stud volumetric or surface examination for the remainder of the currently licensed operating periods for the subject units of the Duke Energy fleet, see Table 1 in Section 3.6 of this safety evaluation, which results in the proposed length of extension shown for each unit. The current required frequency of examination for the RPVH closure studs is once every 10-year ISI interval.

The licensee referred to the results of the deterministic fracture mechanics (DFM) analyses in the following EPRI Report as the primary basis for proposing to increase the ISI intervals for the requested reactor vessel head closure studs: non-proprietary EPRI Report 3002014589, “Technical Basis for Optimization of the Volumetric Examination Frequency for Reactor Vessel Studs,” November 2018 (included as Enclosure 2 of the January 31, 2022, supplement). This report will be referred to hereinafter as “EPRI Report 14589” and contains stress analyses and DFM-based flaw tolerance analyses of RPVH closure studs of pressurized water reactors (PWRs) and boiling water reactors (BWRs). The NRC did not review EPRI Report 14589 for generic use, and this alternative request does not extend beyond the plant-specific authorization for the subject units of the Duke Energy fleet.

3.6 Duration of Proposed Alternative

The licensee is requesting approval of the proposed alternative for the remainder of the current operating license periods, as shown in Table 1 below.

Table 1

Plant/Unit	**Current ISI interval and its Start Date**	**Current Operating License Period End Date**	**Date of Last Category B-G-1 Examination**	**Proposed Length of Extension**
BSEP, Unit 1	5th, 05/11/2018	09/08/2036	03/30/2018	18 years
BSEP, Unit 2	5th, 05/11/2018	12/27/2034	04/11/2017	17 years
CNS, Unit 1	4th, 08/29/2015	12/05/2043	12/05/2015	28 years
CNS, Unit 2	4th, 08/29/2015	12/05/2043	03/19/2015	28 years
MNS, Unit 1*	5th, 12/01/2021	06/12/2041	03/31/2013	28 years
MNS, Unit 2	4th, 07/15/2014	03/03/2043	09/19/2015	27 years
SHNPP, Unit 1	4th, 09/09/2017	10/24/2046	04/27/2009	37 years

*When the licensee submitted their application on December 1, 2020, McGuire, Unit 1 was in the 4th ISI interval, with an end date of November 30, 2021. The McGuire, Unit 1 4th ISI interval ended on November 30, 2021, and the 5th ISI interval commenced on December 1, 2021.

3.7 NRC Staff Evaluation

3.7.1 Degradation Mechanisms

In Attachment 1 to the submittal, the licensee referred to the evaluation of potential degradation mechanisms in Section 2 of EPRI Report 14589, in which EPRI considered the following three aging effects for the RPVH closure studs: (1) cumulative fatigue damage or fatigue cracking,

(2) stress corrosion cracking (SCC), and (3) loss of material due to wear, general corrosion, or pitting/crevice corrosion. EPRI concluded that the most plausible degradation mechanisms for the RPVH closure studs is mechanical fatigue and fatigue cracking. The NRC staff noted that these aging effects are in the NRC staff's guidance for aging management of reactor vessel head closure studs for both initial license renewal (NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," Rev. 2, December 2010, ML103490041) and subsequent license renewal (NUREG-2191, Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report," published July 2017, ML17187A031 (Vol. 1) and ML17187A204 (Vol. 2)).

In Attachments 2 through 8 to the submittal, the licensee stated that the Reactor Vessel Closure Stud program of each subject Duke Energy unit includes preventive measures to use stable lubricants, and specifically, to prohibit the use of molybdenum disulfide. Prohibiting the use of molybdenum disulfide mitigates the effects of SCC in the RPVH closure studs. The licensee also provided in the attachments the material strength values of the RPVH closure studs of the subject Duke Energy fleet units to show that the values are within the values in the NRC staff guidance values for aging management of RPVH closure studs with regard to reducing the potential for SCC. For the BSEP and SHNPP units, the NRC staff verified that a Reactor Vessel Closure Stud program that includes preventive measures to prohibit the use of molybdenum disulfide is included in the updated final safety analysis report (UFSAR) for the units. For the CNS and MNS units, the licensee explained in the supplement that preventive measures within site maintenance procedures of the units are compliant with RG 1.65, Rev. 1, which prohibits the use of lubricants with deliberately added halogens, sulfur, or lead, including molybdenum disulfide. The licensee also stated that the CNS and MNS units are committed to RG 1.65, Rev. 1, within each site's UFSAR, Chapter 1. The NRC staff reviewed the information in Attachments 2 through 8 to the submittal and the supplement and finds that the potential for SCC for the RPVH closure studs of the subject Duke Energy fleet units has been adequately addressed because the licensee has implemented preventive measures that reduce the potential for SCC.

In Attachments 2 through 8 to the submittal, the licensee stated that for each of the subject Duke Energy fleet units, no unacceptable leakage has been observed from the RPVH closure flange during the required ASME Code, Section XI Pressure Test since the most recent volumetric examination of the RPVH closure studs; this was to show that the potential for loss of material due to wear, general corrosion, or pitting/crevice corrosion is adequately being addressed through monitoring of leakage because these degradation mechanisms are predicated on there being active leakage. In Enclosure 1 of the submittal, the licensee stated that detailed procedures are used during each refueling outage for the removal, care, and visual inspections of the RPVH closure studs, and that these activities provide further assurance that degradation is detected and mitigated prior to returning the reactor to service. The NRC staff reviewed the information in Enclosure 1 of the submittal and Attachments 2 through 8 to the submittal, and finds that the potential for loss of material due to wear, general corrosion, or pitting/crevice corrosion of the RPVH closure studs of the subject Duke Energy fleet units has been adequately addressed because the licensee did not observe unacceptable leakage from the reactor vessel closure flange during the required ASME Code, Section XI Pressure Test since the most recent volumetric examination of the studs and the visual inspections performed during each refueling outage as part of the maintenance activities are sufficient to detect wear on the threads of the RPVH closure studs that may not be leakage-related.

Based on the above discussion, the NRC staff finds the licensee's conclusion that mechanical fatigue and fatigue cracking are the most plausible degradation mechanisms for the reactor vessel head closure studs to be acceptable for the plant-specific alternative request for the subject units of the Duke Energy fleet. The licensee stated that both of these mechanisms were addressed in the stress analysis and flaw tolerance assessment in EPRI Report 14589, which are discussed in the following sections.

3.7.2 Stress Analysis

3.7.2.1 Selection of Limiting Geometries

In Section 3.4 of EPRI Report 14589, EPRI selected a limiting (i.e., bounding) geometry for the RPVH closure studs of PWRs and a limiting geometry for the RPVH closure studs of BWRs. EPRI created models of these limiting geometries to perform finite element analyses (FEA, discussed in Section 3.7.2.3 of this SE).

In Attachments 2 through 8 to the submittal, the licensee provided plant-specific geometrical parameters pertaining to the RPVH closure studs, which includes the stud diameter of each of the subject Duke Energy fleet unit that show that each unit's RPVH closure stud geometry is bounded by the selected geometries in EPRI Report 14589. The NRC staff reviewed the information in the attachments to confirm that the RPVH closure stud geometry of each of the subject Duke Energy fleet units is bounded by selected geometry in EPRI Report 14589.

The NRC staff reviewed the geometric parameters from which EPRI based its selection of limiting geometries of the RPVH closure assemblies used for the models for the FEA. EPRI selected the limiting geometries based on parameters that maximize the stresses in the RPVH closure studs. The NRC staff finds the geometrical ratios of HIR/HTK[1] and HIR/SDIA[2] to be acceptable because these ratios maximize the membrane and bending stresses in the RPVH closure studs. The NRC staff also finds that the geometry of the RPVH closure studs of each of the subject Duke Energy fleet unit is acceptable because the plant-specific values of the ratios are bounded by those in EPRI Report 14589.

3.7.2.2 Selection of Loads and Transients

In Section 3.3 of EPRI Report 14589, EPRI stated that three types of inputs were applied to the FEA models to simulate the following load states and transients: stud preload force, coolant temperature along the inner surfaces of the RPVH closure head and RPV, and internal pressure. The NRC staff reviewed the information on the various loads discussed in the subsections under Section 3.3 of EPRI Report 14589, as discussed next.

In Section 3.3.1 of EPRI Report 14589, EPRI stated that the maximum stress during the tensioning load was determined directly, without the use of FEA, as pure membrane stress that is based on the maximum applied tensioner force. The NRC staff finds this approach acceptable since the membrane stress due to tensioning load may be easily determined (without FEA) by dividing the tensioner force by the stud cross-sectional area.

In Section 3.3.2 of EPRI Report 14589, EPRI stated that the stud preload was determined using FEA and that the stud preload stresses are consistent with the preload force for plants with the geometry used to create the limiting RPVH closure assembly geometries. The licensee provided

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[1] Ratio of the reactor vessel head inside radius to the reactor vessel head thickness.

[2] Ratio of the reactor vessel head inside radius to the reactor vessel head closure stud shank outer diameter

confirmation and additional explanation in the supplement related to stud preload based on NRC staff request. The NRC staff reviewed the information on stud preload in EPRI Report 14589 and the supplement. The NRC staff determined that the licensee's explanation in the supplement that the internal pressure does not add stress to the RPVH closure studs is not acceptable. The internal pressure would add stress to the RPVH closure stud in addition to the preload force; but because the flanges that are joined by the RPVH closure studs are much stiffer than the RPVH closure studs, the additional stress would be small compared to the preload force because a majority of the additional force due to internal pressure would be taken up by the flanges (therefore relieving the clamping force acting on the flanges). Therefore, the NRC staff determined that even though the licensee's explanation regarding the additional stress on the studs due to internal pressure is not acceptable, it would have little impact on the stress analysis in EPRI Report 14589 because the additional stress would be small. Based on the NRC staff's review of the information on stud preload in EPRI Report 14589 and the supplement, the NRC staff finds the approach for determining stud preload acceptable because they are based on the limiting RPVH closure assembly geometries that were evaluated in Section 3.7.2.1 of this SE, and the licensee provided adequate additional confirmation and clarification on the stud preload.

In Sections 3.3.3 and 3.3.4 of EPRI Report 14589, EPRI discussed the hydrotest, steady state operation, and transients loading conditions selected for stress analyses. EPRI included detailed descriptions of the transients in Appendix A of EPRI Report 14589 and stated that the transients were selected based on plant survey and are typical of design basis transients. In Section 2.4 of EPRI Report 14589, EPRI stated that the transient information requested in the survey included the Normal, Upset, Emergency, and Faulted transients relevant to the reactor vessel head closure region of PWRs and BWRs. EPRI also stated that the survey included design basis and typical operating conditions for pressure and temperature of PWRs and BWRs. The licensee provided confirmation and additional explanation in the supplement related to the following loading conditions based on the NRC staff's request: leakage test, normal operating, seismic, and loss of coolant accident (LOCA). The NRC staff reviewed the information on loading conditions in EPRI Report 14589 and the supplement. The NRC staff finds the loading conditions acceptable because (1) they are based on actual plant conditions and transients, and therefore represent a reasonable set of loading conditions for PWRs and BWRs; and (2) the licensee provided adequate additional confirmation and explanation about the leakage test, normal operating conditions, and seismic and LOCA conditions.

In Attachments 2 through 8 to the submittal, the licensee provided plant-specific loading and transient information that shows that each of the subject Duke Energy fleet units are bounded by the loads and transients selected and analyzed in EPRI Report 14589 as discussed above. The NRC staff reviewed the information in the attachments to confirm that the loads and transients of the subject Duke Energy fleet units are bounded by those selected and analyzed in EPRI Report 14589.

Based on the discussion above and the plant-specific information in the submittal, the NRC staff finds the loads and transients in EPRI Report 14589 to be acceptable for the RPVH closure studs of the subject units of the Duke Energy fleet.

3.7.2.3 Finite Element Analyses

In Sections 3.1 and 3.2 of EPRI Report 14589, EPRI described the FEA modeling and approach for determining stresses in the reactor vessel head closure studs due to the loads and transients discussed in the previous Section. EPRI used the limiting reactor vessel head closure geometries for a PWR and BWR discussed in Section 3.4 of the report. Because of symmetry, the models consist of a single reactor vessel head closure stud (including the nut and washer), the two flange components joined by the stud, and portions of the closure head and upper vessel. The NRC staff reviewed the details of the FEA modeling and approach (element types used, boundary conditions, symmetry assumptions, use of the thermal solution as input to the structural solution, etc.) and finds that they are consistent with standard FEA practice.

Based on the discussion above and the plant-specific information on geometry and transients in the submittal that was evaluated in Sections 3.7.2.1 and 3.7.2.2 of this SE, the NRC staff finds the stresses shown in Tables 3-3 and 3-4 of EPRI Report 14589 to be acceptable for referencing for the RPVH closure studs of the subject units of the Duke Energy fleet.

3.7.3 Flaw Tolerance Assessment

3.7.3.1 Applied Stress Intensity Factors

In Section 4.1 of EPRI Report 14589, EPRI described the calculation of applied stress intensity factors (SIFs) applicable to the reactor vessel head closure studs under the various loads described in Section 3 of the report. EPRI showed the resulting applied SIFs as a function of flaw depth for various loads in Figures 4-2 and 4-3 of EPRI Report 14589 for a PWR RPHV closure stud and BWR RPVH closure stud, respectively. The NRC staff noted that these SIFs are based on linear elastic fracture mechanics and are for a semi-elliptical surface crack in threaded bolts under membrane and bending stress. The NRC staff finds the applied SIF values in Figures 4-2 and 4-3 of EPRI Report 14589 acceptable based on independent applied SIF calculations using the stress values from Tables 3-3 and 3-4 of the report.

Based on the discussion above and the plant-specific information on geometry and transients in the submittal that was evaluated in Sections 3.7.2.1 and 3.7.2.2 of this SE, the NRC staff finds the applied SIF results from Section 4.1 of EPRI Report 14589 to be acceptable for referencing for the reactor vessel head closure studs of the subject units of the Duke Energy fleet.

3.7.3.2 Allowable Flaw Size

Limiting Flaw Size Evaluation Method

In Section 4.2.1 of EPRI Report 14589, EPRI discussed the evaluation method for determining the limiting flaw size (i.e., allowable flaw size) in the RPVH closure studs. EPRI stated that a safety factor of 2.0 was applied on the primary loads based on the methods in nonmandatory Appendix G, Paragraphs G-2215 and G-2222 of the ASME Code, Section XI. EPRI cited a 2017 NRC SE for the Vogtle Electric Generating Plant and Joseph M. Farley Nuclear Plant (ML17006A109) that authorized a plant-specific alternative examination request for the reactor pressure vessel threads-in-flange. The reactor pressure vessel threads-in-flange are the components into which the RPVH closure studs are threaded. EPRI stated that the use of the methods of Appendix G of ASME Code, Section XI, is consistent with the NRC position in the 2017 NRC safety evaluation regarding the plant-specific reactor vessel threads-in-flange.

EPRI noted in the report that the methods in Paragraphs G-2215 and G-2222 of Appendix G of ASME Code, Section XI, are for vessel components. EPRI also noted that for bolting materials (i.e., the RPVH closure studs), the recommended methods for evaluating fracture prevention are in Article G-4000 of ASME Code, Section XI, which refers to Welding Research Council Bulletin (WRCB) 175, “PVRC Recommendations on Toughness Requirements for Ferritic Materials.”

EPRI stated that the evaluation methods in WRCB 175 are used primarily to define toughness criteria for bolts with a reference flaw size, and not to evaluate flaws with defined structural (i.e., safety) factors. EPRI stated that WRCB 175 is considerably older than other references that are cited for fracture mechanics evaluations in bolted joints and none of the solutions discussed in it regarding bolting are specific for bolted joints. Thus, EPRI used the safety factors in Paragraphs G-2215 and G-2222 of Appendix G of ASME Code, Section XI, for vessels, with the applied SIFs described in Section 4.1 of EPRI Report 14589, to define the allowable flaw size for the postulated flaws in the RPVH closure studs.

The NRC staff confirmed the information in Appendix G of ASME Code, Section XI and WRCB 175 that EPRI cited, as discussed above. Even though EPRI acknowledged that the safety factor of 2.0 in Appendix G of ASME Code, Section XI is for vessels, the NRC staff assessed the basis for the safety factor. Chapter 30 of the Companion Guide to the ASME Code (Volume 2) explains that a safety of factor of 2.0 applied on the SIF due to the pressure loading (i.e., primary load) combined with a conservative postulated flaw size (i.e., a depth of one-quarter of the vessel thickness) ensures a safety factor of 3 on primary load that is consistent with the ASME Code, Section III design stress limits on vessels. The licensee explained in the supplement that the RPVH closure stud cross-sectional area with the limiting flaw size (based on a safety factor of 2.0 on primary load) determined for the limiting case (i.e., BWR case) analyzed in EPRI Report 14589 is greater than the minimum RPVH closure stud cross-sectional area determined from the design requirements of Appendix E of the ASME Code, Section III for the RPVH closure studs of the BSEP units. The licensee therefore concluded that "the [ASME Code,] Section III design margins for the bolting material are maintained even with all [RPVH closure] studs cracked to the limiting flaw size."

The NRC staff reviewed the information in Section 4.2.1 of EPRI Report 14589, the supplement, and Appendix E of the ASME Code, Section III. The NRC staff determined that the licensee's conclusion that the ASME Code, Section III, design margins for the studs of the BSEP units are maintained even with all RPVH closure studs cracked to the limiting flaw size is acceptable because the Appendix E equations are for the total design cross-sectional area of all RPVH closure studs. The NRC staff also determined that the limiting flaw size based on a safety factor of 2.0 on primary load would result in significant structural margin for the RPVH closure studs of the subject units of the Duke Energy fleet and is therefore acceptable for the RPVH closure studs of the units. This determination is based on (1) the total remaining ligament cross-sectional area of the RPVH closure studs of the limiting case with the postulated limiting flaw size is greater than the total design cross-sectional area of the RPVH closure studs of the limiting units (i.e., the BSEP units) requested in the submittal; and (2) the total remaining ligament cross-sectional area assumes all the RPVH closure studs are cracked.

Based on the discussion above and the plant-specific information on geometry and transients in the submittal that was evaluated in Sections 3.7.2.1 and 3.7.2.2 of this SE, the NRC staff finds the methodology in EPRI Report 14589 for determining the limiting flaw size in the RPVH closure studs to be acceptable for the RPVH closure studs of the subject units of the Duke Energy fleet.

Fracture Toughness

In Section 4.2.2 of EPRI Report 14589, EPRI selected a fracture toughness (K_{IC}) value of 190 kilo-pound per square inch (ksi) square root inches (ksi$\sqrt{in}$) for the flaw tolerance assessment of the RPVH closure studs based on Charpy impact testing and fracture toughness data of SA-540 steels used for RPVH closure studs reported in a 1977 paper in the Journal of Pressure Vessel

Technology (Seeley, R.R., et al., "Fracture Toughness Properties of SA-540 Steels for Nuclear Bolting Applications," Journal of Pressure Vessel Technology [JPVT], August 1977). Figures 6 and 7 of the 1977 JPVT paper show the Charpy impact (and lateral expansion) testing and fracture toughness data, respectively. In Attachments 2 through 8 to the submittal, the licensee showed that the RPVH closure studs of each of the subject Duke Energy fleet units are specified as SA-540 steel, and in the supplement confirmed that all the available Charpy impact values of the RPVH closure studs of each of the subject Duke Energy units are above the lower Charpy impact property curve shown in Figure 7 of the 1977 JPVT paper. The licensee cited the 1977 JPVT paper, which states that "the upper shelf fracture maximum toughness values range from 190 ksi√in (209 [megapascal square root meter] MPa√m) to 240 ksi√in (264 MPa√m) at room temperature," for selecting the K_{IC} value of 190 ksi√in and in the supplement provided additional clarification on this K_{IC} value.

The NRC staff reviewed the information on fracture toughness in EPRI Report 14589, the attachments to the submittal, the supplement, and the 1977 JPVT paper. The NRC staff observed from the fracture toughness data of the heats of SA-540 steel presented in Figure 7 of the 1977 JPVT that the upper shelf K_{IC} value could be lower than 190 ksi√in at operating temperatures. However, considering the significant structural margin (evaluated in Section 3.6.3.2, Subsection "Limiting Flaw Size Evaluation Method," of this SE) and the conservative flaw crack growth (FCG) calculations (evaluated in Section 3.7.4 of this SE), the NRC staff determined that even if the K_{IC} value is lower than 190 ksi√in, the time to grow the postulated initial flaw size to the limiting flaw size would be greater than the 17 years and 18 years requested for the BSEP units, which are the limiting units in the submittal.

Based on the discussion above, the NRC staff finds that even if the K_{IC} value is lower than 190 ksi√in, the proposed ISI extension periods for the RPVH closure studs of the subject units of the Duke Energy fleet are not impacted.

Stud Ductility

The NRC staff also noted that the RPVH closure studs of the subject Duke Energy units are at the upper shelf regime of fracture toughness in which the material of the RPVH closure studs has sufficient ductility. This observation is based on (1) the lowest temperature permitted for stud preload by any of the reactor vessel tensioning procedures at the subject Duke Energy units is 60°F (as stated in the supplement), which is at the upper shelf temperature of the fracture toughness data in the 1977 JPVT paper; and (2) the licensee's confirmation in the supplement that all available Charpy impact values of the reactor vessel head closure studs of each of the units are above the lower Charpy impact property curve in the 1977 JPVT paper. The fracture toughness being in the upper shelf regime means that if a postulated flaw in a RPVH closure stud reaches the limiting size, the RPVH closure stud would be in a ductile mode of fracture (as opposed to a brittle mode of fracture that can occur in reactor vessel steels at 60°F when a postulated flaw reaches the limiting flaw size). Also, since the applied service stresses (based on the stress values in Tables 3-3 and 3-4 of EPRI Report 14589) are lower than the nominal yield strength of 150 ksi for SA-540 steels, the crack driving forces would be lower than the elastic-plastic material resistance of the RPVH closure stud, which provides further assurance that there will be no brittle fracture of the stud.

Maximum Allowable Flaw Size

In Section 4.2.3 of EPRI Report 14589, EPRI presented maximum allowable flaws sizes in the RPVH closure studs of PWRs and BWRs. EPRI determined the maximum allowable flaw size by setting the total applied SIF (due to primary load with a safety factor 2.0 plus secondary load) equal to fracture toughness and solving for flaw size. EPRI showed this determination graphically in Figures 4-4 and 4-5 of EPRI Report 14589 for PWR and BWR studs, respectively. The NRC staff reviewed the limiting flaw sizes (which are the crack depths at which the "2 K_{pri} + K_{sec}" curves intersect the K_{IC} curve) shown in these two figures. The NRC staff determined that because the K_{IC} value could be lower than 190 ksi√in (evaluated in Section 3.6.3.2, Subsection "Stud Ductility," of this SE), the limiting flaw sizes could be smaller than those shown in Figures 4-4 and 4-5 of EPRI Report 14589. The NRC staff determined that the smaller limiting flaw size for PWR RPVH closure studs as a result of a lower K_{IC} value has no impact on the FCG calculation shown in Figure 4-6 of EPRI Report 14589 because the amount of growth is minimal as shown in the figure. Therefore, for the PWR units in the submittal, i.e., the CNS, MNS, and SHNPP units, the NRC staff determined that the smaller limiting flaw size has insignificant impact on the proposed ISI extension periods for the RPVH closure studs of these units. In the previous Section of this SE on fracture toughness, the NRC staff evaluated the impact of the lower K_{IC} value for the BSEP units. A lower K_{IC} value leads to a smaller limiting size for the BWR stud, and the NRC staff determined that the time to grow the postulated initial flaw size to the smaller limiting flaw size would be greater than the 17 years and 18 years ISI extension periods requested for the BSEP units.

Based on the discussion above and the evaluations in Sections 3.6.3.1, 3.6.3.2, Subsection "Limiting Flaw Size Evaluation Method," and 3.6.3.2, Subsection "Fracture Toughness," of this SE, the NRC staff finds that even with limiting flaw sizes smaller than those in Figures 4-4 and 4-5 of EPRI Report 14589 due to a lower K_{IC} value, the proposed ISI extension periods for the RPVH closure studs of the subject units of the Duke Energy fleet, are not impacted.

3.7.4 Fatigue Crack Growth Analyses

In Section 4.3 of EPRI Report 14589, EPRI discussed the FCG analyses for the reactor vessel head closure studs of PWRs and BWRs. EPRI stated that the FCG evaluation used the largest fatigue cycle loading to bound the full set of operating ranges that occur during plant operation and that the transient frequency is based on the results of a plant survey. EPRI stated that the FCG rate is based on those in the ASME Code, Section XI, A-4300, which are rates for low alloy ferritic steels exposed to air environments. EPRI postulated an initial flaw depth of 0.3 inch and used the range of applied SIFs in Section 4.1 of EPRI Report 14589 (evaluated in Section 3.6.3.1 of this SE) and the A-4300 FCG rates to calculate the time to grow from the postulated initial flaw depth to the limiting flaw sizes determined in Section 4.2.3 of EPRI Report 14589. EPRI stated that the postulated initial flaw depth is based on a reference flaw depth in WRCB 175 and is larger than the 0.157-inch minimum flaw size detectable by inspection. In the supplement, the licensee confirmed that there were no relevant indications detected during the ASME Code, Section XI, ISI examinations of the RPVH closure studs of the subject Duke Energy units. The licensee also provided additional explanation and sensitivity studies on the FCG analyses in the supplement that showed that the A-4300 FCG rate is appropriate for SA-540 steel and that the FCG analyses in EPRI Report 14589 contained substantial conservatisms.

The NRC staff reviewed the information in Section 4.3 of EPRI Report 14589, the supplement, WRCB 175, and the ASME Code, Section XI, A-4300. The NRC staff determined that the postulated initial depth of 0.3 inch for the FCG analyses is acceptable for the RPVH closure studs of the subject Duke Energy units because there were no relevant indications detected during the ASME Code, Section XI ISI examinations of the RPVH closure studs. The NRC staff also determined that the A-4300 FCG rate is appropriate for SA-540 steel because the licensee's comparison of the A-4300 FCG rate with the FCG rate for martensitic steel that was tested at a broad range of yield strengths showed reasonable agreement of the two rates. Lastly, the NRC staff determined that the FCG analyses in EPRI Report 14589 contained substantial conservatisms because of the licensee's sensitivity study that assumed three times the A-4300 FCG (with bounding cycle counts for low-cycle transients) and showed lower growths than those shown in Figures 4-6 and 4-7 of EPRI Report 14589. The NRC staff noted that the conservative FCG analyses provide reasonable assurance that uncertainties in the FCG rate, as well as in the applied loads that drive the rate, were adequately accounted for in the RPVH closure studs of the subject units of the Duke Energy fleet.

Based on the discussion above and the evaluations in Sections 3.7.2, 3.6.3.1, and 3.6.3.2 of this SE, the NRC staff finds the FCG analysis for the PWR studs to be acceptable for the reactor vessel head closure studs of the CNS, MNS, and SHNPP units. In Section 3.6.3.2, Subsection "Fracture Toughness," of this SE, the NRC determined that the fracture toughness value could be lower than the value assumed in EPRI Report 14589, and thus lowering the limiting flaw size for the BWR RPVH closure studs analyzed in the report. The smaller limiting flaw size would also lower the growth time period for the BWR RPVH closure studs, and the NRC staff determined that the lower growth time period is greater than the 17 years and 18 years ISI extension periods requested for the BSEP units.

3.7.5 Performance monitoring

The licensee provided information in the supplement on performance monitoring of the RPVH closure studs of the units in the submittal whose proposed ISI extension periods are greater than 20 years, i.e., the CNS, MNS, and SHNPP units.

The licensee stated in the submittal that detailed procedures are used during each refueling outage for the removal, care, and visual inspection of the RPVH closure studs and the threads-in-flange. The licensee clarified in the supplement the activities performed during the controlled maintenance activities at the CNS, MNS, and SHNPP units. Specifically, the licensee explained that the procedure performed for tensioning the RPVH closure studs results in a higher stress in the studs than the stud preload stress, and that thus, the tensioning process for the RPVH closure studs of the CNS, MNS, and SHNPP units acts like a proof test for the RPVH closure studs.

The licensee further explained that a RPVH closure stud with the limiting flaw size will likely be revealed during the proof test by not meeting the elongation acceptance criteria. The licensee stated that elongation measurements of the studs are taken before and after tensioning and are verified by Quality Control examiners and documented to ensure accuracy. The licensee stated that the tensioning process acting as a proof test and the verified elongation measurements for each RPVH closure stud ensure confidence in the integrity of the tensioned studs for periods longer than 20 years.

The licensee also stated that the ASME Code, Section XI ISI examinations on the RPVH head closure studs at two other nuclear plants (not part of the subject proposed alternative) within the Duke Energy fleet, the Oconee Nuclear Station, Units 1, 2, and 3 (Oconee), and H.B. Robinson Steam Electric Plant, Unit No. 2 (Robinson), and within the industry, will continue to be performed and that any relevant indications or new degradation mechanisms identified would be entered into the Duke Energy Corrective Action Program. The licensee stated that this operating experience will be evaluated and extent of condition examinations, if required by the evaluation, would be performed for the CNS, MNS, and SHNPP units.

The NRC staff reviewed the description of the controlled maintenance procedures in the submittal and the supplement. The NRC staff calculated the expected elongation value during tensioning of the RPVH closure stud with a postulated limiting flaw size and compared it with the range of acceptable elongation value from available tensioning procedures of PWR RPVH closure studs. The NRC staff determined that the expected elongation value is less than the range of acceptable elongation values, which means that a postulated limiting flaw size in the RPVH closure stud would not be detected during the tensioning process. The NRC staff also determined that because the range of acceptable stud elongation values is larger than the expected elongation value of a flawed RPVH closure stud, the precision and accuracy of the instruments used for stud elongation measurements are not sufficient to measure the elongation of a flawed stud with confidence. Based on its review of the information in the submittal, the supplement, and available tensioning procedures, and its calculations of stud elongation, the NRC staff finds that the stud tensioning procedure would not detect a flaw in a stud, and therefore, is not an acceptable performance monitoring measure for the RPVH closure studs of the CNS, MNS, and SHNPP units.

The NRC staff also finds that while the visual inspections of the RPVH closure studs performed as part of the controlled maintenance activities at the CNS, MNS, and SHNPP units are good practice for checking the general condition of the studs, they are not sufficient to detect a flaw within the studs caused by a new or unexpected degradation, and therefore, they alone are not an acceptable performance monitoring measure for the RPVH closure studs of the units.

Additionally, the NRC staff finds that while the continued ASME Code, Section XI ISI RPVH closure stud examinations within the industry provide operating experience data for studs, they would not constitute an adequate performance monitoring measure for the studs at the CNS, MNS, and SHNPP units because the data would not be Duke Energy fleet stud performance data.

Even though the performance monitoring measures discussed above are not acceptable, the NRC staff reviewed the information in the supplement on the handling of RPVH closure stud operating experience from Oconee and Robinson. The NRC staff noted that the continued ASME Code, Section XI ISI examinations of the RPVH closure studs at Oconee and Robinson will provide sufficient operating experience data (i.e., stud performance data) for the studs at the CNS, MNS, and SHNPP units during the period of ISI extension proposed for each unit because (1) the Oconee and Robinson studs are made of the same material as the studs at the CNS, MNS, and SHNPP units; and (2) the studs at the two units are under a similar environment and similar service loads as those for the studs at the CNS, MNS, and SHNPP units. The NRC staff confirmed from the site UFSARs that the RPVH closure studs of Oconee and Robinson are made of the same material as those of the CNS, MNS, and SHNPP units. The NRC staff noted that the units at Oconee and Robinson, and the CNS, MNS, and SHNPP units are PWR units, which means that the RPVH closure studs of the units would all experience similar service loads. The NRC staff also noted that because the RPVH closure studs of the units are not exposed to the reactor coolant, the studs would all be in a similar environment. Based on the discussion above, the NRC staff finds that the continued ASME Code, Section XI ISI examinations of the RPVH closure studs at Oconee and Robinson combined with the controlled maintenance activities at the CNS, MNS, and SHNPP units will provide adequate performance monitoring of the RPVH closure studs of the CNS, MNS, and SHNPP units during the period of ISI extension proposed for each unit.

3.8 NRC Staff Technical Evaluation Summary

The NRC staff finds that the reasons below collectively provide reasonable assurance that structural integrity of the RPVH closure studs for the Brunswick Steam Electric Plant, Units 1 and 2, Catawba Nuclear Station, Units 1 and 2, McGuire Nuclear Station, Units 1 and 2, and the Shearon Harris Nuclear Power Plant, Unit 1, will be maintained for the remainder of the current licensed operating periods for each unit, as shown in Table 1 in Section 3.6 of this safety evaluation:

1. Plant-specific information that demonstrates acceptability of the stress and flaw tolerance analyses in EPRI Report 14589 for the RPVH closure studs for each unit based on the evaluations in Section 3.0 of this safety evaluation.

2. Additional information that demonstrates the conservatisms in the structural margin and FCG analysis with respect to the RPVH studs of the units.

3. Adequate performance monitoring of the RPVH studs of the units.

4. Plant-specific information that shows sufficient ductility of the RPVH studs of the units.

5.0 CONCLUSION

As set forth above, the NRC staff has determined that the proposed alternative in the licensee's request referenced above would provide an acceptable level of quality and safety and concludes that the licensee has adequately addressed the regulatory requirements set forth in 10 CFR 50.55a(z)(1).

The NRC staff, therefore, authorizes the use of proposed alternative RA-19-0352 at Brunswick Steam Electric Plant, Units 1 and 2, Catawba Nuclear Station, Units 1 and 2, McGuire Nuclear Station, Units 1 and 2, and the Shearon Harris Nuclear Power Plant, Unit 1, for the remainder of the current licensed operating periods for each unit, as shown in Table 1 in Section 3.6 of this safety evaluation.

The NRC's authorization of the proposed alternative does not infer or imply the approval of EPRI Report 14589 for generic use.

All other ASME Code, Section XI, requirements for which an alternative was not specifically requested and authorized remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

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Date: November 18, 2022

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2; CATAWBA NUCLEAR STATION, UNITS 1 AND 2; MCGUIRE NUCLEAR STATION, UNITS 1 AND 2, AND SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1 – AUTHORIZATION OF RA-19-0352 REGARDING USE OF ALTERNATIVE FOR REACTOR PRESSURE VESSEL HEAD CLOSURE STUD EXAMINATIONS (EPID L-2020-LLR-0156) DATED NOVEMBER 18, 2022

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