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

October 19, 2023

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

SUBJECT: CATAWBA NUCLEAR STATION, UNIT NOS. 1 AND 2; SHEARON HARRIS NUCLEAR POWER PLANT, UNIT NO. 1; MCGUIRE NUCLEAR STATION, UNIT NOS. 1 AND 2; OCONEE NUCLEAR STATION, UNIT NOS. 1, 2, AND 3; AND H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 – ISSUANCE OF ALTERNATIVE TO PRESSURIZER WELDS (EPID L-2023-LLR-0020)

Dear Mr. Gibby:

By letter dated February 17, 2023, as supplemented by letter dated July 20, 2023, Duke Energy Carolinas, LLC and Duke Energy Progress, LLC (Duke Energy, the licensee), submitted a request for the Catawba Nuclear Station, Units 1 and 2 (CNS1 and CNS2); Shearon Harris Nuclear Power Plant, Unit 1 (HNP), McGuire Nuclear Station, Units 1 and 2 (MNS1 and MNS2); Oconee Nuclear Station, Units 1, 2, and 3 (ONS1, ONS2, and ONS3); and H. B. Robinson Steam Electric Plant, Unit 2 (RNP); to the U.S. Nuclear Regulatory Commission (NRC or Commission) for a proposed alternative to certain American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI examination requirements.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(1), the licensee proposed to forgo ASME Section XI examination requirements for the requested pressurizer welds. The regulation in 10 CFR 50.55a(z)(1) requires the licensee to demonstrate that the proposed alternative provides an acceptable level of quality and safety.

The NRC staff has reviewed the subject request and concludes, as set forth in the enclosed safety evaluation, that Duke Energy has adequately addressed the requirements in 10 CFR 50.55a(z)(1). Therefore, the NRC staff authorizes the use of the proposed alternative for Duke Energy through the 5th inservice inspection (ISI) interval for CNS1 and 2 and HNP and through the 6th ISI interval for RNP; MNS1 and 2; and ONS1, 2, and 3.

All other ASME BPV 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 Shawn Williams at (301) 415-1009 or by e-mail at Shawn.Williams@nrc.gov.

Sincerely,

David J. Wrona
Digitally signed by David J. Wrona
Date: 2023.10.19 09:22:02 -04'00'

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

Docket Nos.: 50-413, 50-414, 50-400, 50-369, 50-370, 50-269, 50-270, 50-287, and 50-261

Enclosure:
Safety Evaluation

cc: Listserv

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

INSPECTION INTERVAL EXTENSION FOR PRESSURIZER WELDS

DUKE ENERGY CAROLINAS, LLC

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

H.B. ROBINSON STEAM ELECTRIC PLANT, UNIT 2

DOCKET NOS. 50-413, 50-414, 50-400, 50-369, 50-370, 50-269, 50-270, 50-287, AND 50-261

1.0 INTRODUCTION

By letter dated February 17, 2023 (Agencywide Documents Access and Management System Accession No. ML23048A148), as supplemented by letter dated July 20, 2023 (ML23201A141), Duke Energy Carolinas, LLC and Duke Energy Progress, LLC (Duke Energy, the licensee), submitted a request for the Catawba Nuclear Station, Units 1 and 2 (CNS1 and CNS2); Shearon Harris Nuclear Power Plant, Unit 1 (HNP); McGuire Nuclear Station, Units 1 and 2 (MNS1 and MNS2); Oconee Nuclear Station, Units 1, 2, and 3 (ONS1, ONS2, and ONS3); and H. B. Robinson Steam electric Plant, Unit 2 (RNP); to the U.S. Nuclear Regulatory Commission (NRC or Commission) for a proposed alternative to certain American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI examination requirements.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(1), the licensee proposed to forgo ASME Code, Section XI examination requirements for the requested pressurizer (PZR) welds as discussed in Request for Alternative RA-22-0257. The regulation in 10 CFR 50.55a(z)(1) requires the licensee to demonstrate that the proposed alternative provides an acceptable level of quality and safety. The NRC staff reviewed the proposed alternative request for CNS1 and 2; HNP, MNS1 and 2; ONS1, 2, and 3; and RNP as a plant-specific alternative.

Enclosure

2.0 REGULATORY EVALUATION

The pressurizer (PZR) pressure-retaining welds at the subject Duke units are ASME Code Class 1 components, whose inservice inspections (ISIs) are performed in accordance with the applicable edition of Section XI, "*Rules for Inservice Inspection of Nuclear Power Plant Components,*" of the ASME Code, as required by 10 CFR 50.55a(g).

The regulations in 10 CFR 50.55a(g)(4) state, in part, components that are classified as ASME Code Class 1, 2, and 3 must meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in the ASME Code, Section XI, to the extent practical within the limitations of design, geometry, and materials of construction of the components.

The regulations in 10 CFR 50.55a(z) state, in part, that alternatives to the requirements in paragraphs (b) through (h) of 10 CFR 50.55a may be used when authorized by the NRC if the licensee demonstrates that: (1) the proposed alternative 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.

3.0 TECHNICAL EVALUATION

3.1 Licensee's Proposed Alternative

Applicable Code Edition and Addenda

The applicable ISI interval and associated codes of record for the subject plants are summarized in Table 1.

Table 1: Section XI Codes of Record for Subject Plants

Plant	ISI Interval	ASME Section XI Code Edition/Addenda	Current Interval Start Date	Current Interval End Date
CNS1 and CNS2	Fourth	2007 Edition through 2008 Addenda	08/19/2015	12/06/2024 (Unit 1) 02/24/2026 (Unit 2)
HNP	Fourth	2007 Edition through 2008 Addenda	09/09/2017	09/08/2027
MNS1	Fifth	2007 Edition through 2008 Addenda	12/01/2021	11/30/2031
MNS2	Fourth	2007 Edition through 2008 Addenda	07/15/2014	2/29/2024
ONS1, ONS2, and ONS3	Fifth	2007 Edition through 2008 Addenda	07/15/2014	07/15/2024
RNP	Fifth	2007 Edition through 2008 Addenda	07/21/2012	02/19/2023

American Society of Mechanical Engineers (ASME) Code Components Affected

ASME Code Class: Class 1

Examination Category: B-B, "Pressure Retaining Welds in Vessels Other Than Reactor Vessels"

	B-D, “Full Penetration Welded Nozzles in Vessels”
Item Numbers:	B2.11 for PZR shell-to-head welds, circumferential B2.12 for PZR shell-to-head welds, longitudinal B3.110 for PZR nozzle-to-vessel welds
Component IDs:	The nine tables in Section 1 of the Enclosure to the licensee’s submittal lists the component identifications (IDs) affected for each subject plant.

ASME Code Requirement for Which Alternative Is Requested

For ASME Code Class 1 welds in the PZR, the ISI requirements are those specified in Subarticle IWB-2500 of the ASME Code, Section XI, which requires the licensee to perform volumetric examinations as specified in ASME Code, Section XI, Table IWB-2500-1, for each Examination Category and Item No. listed below once every 10-year ISI interval.

- Examination Category B-B, Item No. B2.11, PZR Shell-to-Head Welds, Circumferential
- Examination Category B-B, Item No. B2.12, PZR Shell-to-Head Welds, Longitudinal
- Examination Category B-D, Item No. B3.110, PZR Nozzle-to-Vessel Welds

The NRC staff confirmed that the ASME Code ISI requirements listed above did not change in the latest edition of ASME Code, Section XI incorporated by reference in 10 CFR 50.55a.

Reason for Proposed Alternative

In Section 4.0 of the Enclosure to the submittal, the licensee stated that the Electric Power Research Institute (EPRI) performed assessments in the following non-proprietary report of the basis for the ASME Code, Section XI examination requirements for the subject PZR welds.

- 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 (hereafter referred to as “EPRI report 15905,” ML21021A271).

The assessments include a survey of inspection results from 74 domestic and international nuclear units and flaw tolerance evaluations using probabilistic fracture mechanics (PFM) and deterministic fracture mechanics (DFM). The licensee stated that this report was developed consistent with EPRI’s White Paper on PFM (ML19241A545) and Regulatory Guide 1.245, “*Preparing Probabilistic Fracture Mechanics Submittals*” (ML21334A158). The licensee stated that based on the conclusions in the EPRI report supplemented by plant-specific evaluations, the licensee requested an alternative to the ASME Code, Section XI examination requirements for the subject PZR welds.

The NRC staff noted that EPRI report 15905 was not submitted or reviewed as a topical report. The NRC staff reviewed the proposed alternative request as a plant-specific alternative for each of the subject plants. The NRC did not review the EPRI report for generic use, and this review does not extend beyond the plant-specific authorization.

<u>Proposed Alternative and Basis for Use</u>

In Section 5.0 of the Enclosure to the submittal, the licensee described the proposed alternative for each plant, as follows.

For CNS1 and 2, the licensee requested to defer the examinations for Item Numbers B2.11, B2.12, and B3.110 through the end of the 5^{th} inspection interval. According to the licensee, the subject PZR welds of CNS1 and 2 received the required preservice inspection (PSI) examinations followed by ISI examinations through the 2^{nd} period of the current 4^{th} ISI interval. The licensee further explained that the proposed alternative leads to an extension for CNS1 of 19.83 years and 20 years for CNS2 from the end of the 3^{rd} ISI interval for both units.

For HNP, the licensee requested to defer the examinations for Item Numbers B2.11, B2.12, and B3.110 through the end of the 5^{th} inspection interval. The licensee stated that the subject PZR welds received the required PSI examinations followed by ISI examinations through the 1^{st} period of the current 4^{th} ISI interval. The licensee further explained that the proposed alternative leads to an extension of 19.58 years from the end of the 3^{rd} ISI interval.

For MNS1 and 2, the licensee requested to defer the examinations for Item Numbers B2.11, B2.12, and B3.110 through the end of the 6^{th} ISI interval. The licensee stated that the subject PZR welds of MNS1 and 2 received the required PSI examinations followed by ISI examinations through the 4^{th} ISI interval. The licensee further explained that the proposed alternative leads to an extension of 20 years from the end of the 4^{th} ISI interval of each unit.

For ONS1, 2, and 3, the licensee requested to defer the examinations for Item Numbers B2.11, B2.12, and B3.110 from only the 6^{th} ISI interval requirements. The licensee stated that the subject PZR welds received the required PSI examinations followed by ISI examinations through the 5^{th} ISI interval.

For RNP, the licensee requested to defer the examinations for Item Numbers B2.11 and B2.12 from only the 6^{th} ISI interval. The licensee stated that the subject PZR welds received the required PSI examinations followed by ISI examinations through the 5^{th} ISI interval.

Figure 1 demonstrates the licensee's proposed alternative from Figure 1-1 of the supplement. Additionally, for each subject unit, Table 1-2 of the supplement shows the length of time from the last inspection to end of proposed alternative for that unit.

Figure 1: Licensee’s Proposed Alternative and Performance Monitoring Plan

Plant \ Year	2012	2013	2014	2015	2016	2017	2018	2019	2020	2021	2022	2023	2024	2025	2026	2027	2028	2029	2030	2031	2032	2033	2034	2035	2036	2037	2038	2039	2040	2041	2042	2043	2044	2045	2046
Catawba 1	3rd Interval			4th Interval								4th Interval		5th Interval				X[1]							6th Interval - ASME Code PZR Requirements Resume							12/5/2043			
Catawba 2	3rd Interval			4th Interval								4th Interval				5th Interval							X[1]		6th Interval - ASME Code PZR Requirements Resume							12/5/2043			
H.B Robinson 2	5th Interval											6th Interval							7/31/2030																
McGuire 1	4th Interval									5th Interval										X[1]	6th Interval									6/12/2041					
McGuire 2	3rd Interval		4th Interval										5th Interval								X[1]			6th Interval								3/3/2043			
Oconee 1, 2, 3	4th Interval		5th Interval										6th Interval									#													
Shearon Harris 1	3rd Interval					4th Interval						4th Interval					5th Interval			X			X				6th Interval - ASME Code PZR Requirements Resume								10/24/2046

Notes:

1. Two separate exam item numbers are scheduled for the same refueling outage to minimize radiation exposure for related or adjacent exams. (i.e., Pressurizer circumferential (B2.11) and longitudinal (B2.12) welds are scheduled for the same outage since these welds intersect or a Pressurizer Safety and Relief nozzle are scheduled in the same outage due to their proximity to each other.)

LEGEND	
(green)	Inspection Interval prior to Alternative RA-22-0257
X	Scheduled Performance Monitoring Exam
(orange)	Deferral Period per RA-22-0257
(blue)	Subsequent Inspection Interval: Reverts Back to ASME Code Requirements
(red)	Current License Period End Date
#	Oconee Current License Period End Date: Unit 1 - 2/6/2033; Unit 2 - 10/6/2033; Unit 3 - 7/19/2034

Duration of Proposed Alternative

The licensee requested to apply the proposed alternative for the remainder of the current fourth 10-year ISI interval through the end of the fifth 10-year ISI interval for CNS1 and 2, and HNP. The licensee requested to apply the proposed alternative for the fifth and sixth 10-year intervals for MNS1 and 2. The licensee requested to apply the proposed alternative for the sixth 10-year ISI interval for ONS1, 2, and 3, and RNP.

Basis for Proposed Alternative

In Section 5.0 of the Enclosure to the submittal, the licensee referred to the results of the PFM analyses in EPRI report 15905 mentioned above, an additional PFM sensitivity study, and plant-specific PFM evaluations for ONS1, 2, and 3, as the bases for the proposed alternative.

3.2 NRC Staff Evaluation

The NRC staff's review focused on evaluating the applicability of the PFM analyses in Section 8.3 of EPRI report 15905, and verifying whether the DFM and PFM analyses in the report support the proposed alternative. The NRC staff previously reviewed a similar request based on EPRI report 15905. That request was in support of a Salem Generating Station, Units 1 and 2, submittal (ML20218A587, hereafter "Salem submittal"). As part of the previous review of the Salem submittal, the NRC staff conducted a thorough review of the applicable aspects of the EPRI report and documented its review in the associated, plant-specific Salem safety evaluation (SE) (ML21145A189). For the Duke Energy review, the NRC staff considered the referenced information and focused on the plant-specific application of the EPRI report for the subject Duke Energy units. The NRC staff also evaluated the plant-specific analyses for ONS1, 2, and 3 in Attachments 7 and 8 of the Duke Energy submittal. Using a risk-informed approach, the NRC staff also confirmed that the proposed alternative provides sufficient performance monitoring.

3.2.1 Degradation Mechanisms

The NRC staff reviewed the submittal for plant-specific circumstances that may indicate presence of a degradation mechanism and activity sufficiently unique to the subject Duke Energy units to merit additional consideration. The NRC staff found no evidence of conditions at the subject Duke Energy units that would require consideration of a unique degradation mechanism beyond application of the information the licensee referenced from EPRI report 15905. Specifically, the NRC staff reviewed the materials, stress states, and consistency of chemical environment (i.e., reactor coolant) of the subject PZR welds and found them to be consistent with the assumptions made in the EPRI report. Therefore, the NRC staff finds that consideration of additional degradation mechanisms beyond those from the EPRI report is not necessary.

3.2.2 PFM Analysis

The NRC staff confirmed that the PFM analysis referenced by the licensee for the Duke Energy submittal is consistent with the approach taken in the technical arguments presented in the Salem submittal and explicitly referenced in the Duke Energy request. The original review of this approach is documented in the Salem SE. The NRC staff reviewed the application of this approach, as proposed in the Duke Energy request, and determined that the PFM analysis is consistent with the previously approved precedent in the Salem submittal. Therefore, the NRC staff finds the proposed PFM analysis to be appropriate for the Duke Energy application. The NRC staff noted that the acceptance criterion of 1×10^{-6} failures per year (also termed Probability of Failure, PoF) is tied to that used by the NRC staff in the development of 10 CFR 50.61a, "*Alternate fracture toughness requirements for protection against pressurized thermal shock events*" and other similar reviews. In that rule, the reactor vessel through-wall crack frequency (TWCF) of 1×10^{-6} per year for a pressurized thermal shock event is an acceptable criterion because reactor vessel TWCF is conservatively assumed to be equivalent to an increase in core damage frequency, and as such meets the guidance in Regulatory Guide (RG) 1.174, "*An Approach to for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,*"(ML17317A256). This assumption is conservative because a through-wall crack in the reactor vessel does not necessarily increase the likelihood of core damage. The discussion of TWCF is explained in detail in the technical basis document for 10 CFR 50.61a, NUREG-1806 "*Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61),"* August 2007 (ML072830074).

The NRC staff also noted that the TWCF criterion of 1×10^{-6} per year was generated using a very conservative model for reactor vessel cracking. In addition, this criterion exists within a context of reactor pressure vessel surveillance programs and inspection programs. The NRC staff finds that the licensee's use of 1×10^{-6} failures per year based on the reactor vessel TWCF criterion is acceptable for the requested PZR welds (a) the impact of a PZR vessel failure would be less than the impact of a reactor vessel failure on overall risk; (b) the subject PZR welds have substantive, relevant, and continuing inspection histories and programs; and (c) the estimated risks associated with the individual welds are mostly much lower than the system risk criterion (i.e., the system risk is dominated by a small sub-population which can be considered the principal system risk for integrity). The NRC staff further noted that comparing the probability of leakage to the same criterion is conservative because leakage is less severe than rupture. The use of a PoF criteria such as 1×10^{-6} per year for individual welds may not be appropriate generically, but based on the discussion above, the NRC staff finds the application of this criterion acceptable for this plant-specific review for the subject PZR welds for Duke Energy.

Lastly, the NRC staff noted that the acceptance criterion of 1×10^{-6} failures per year is lower, and thus more conservative, than the criterion the NRC staff accepted in proprietary report BWRVIP-05, "BWR [Boiling Water Reactor] Vessel and Internals Project: BWR Reactor Pressure Vessel Weld Inspection Recommendation, September 1995"; non-proprietary report BWRVIP-108NP-A, "*BWR Vessel and Internals Project: Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii*, October 2018" (ML19297F806); and non-proprietary report BWRVIP-241NP-A, "*BWR Vessel and Internals Project: Probabilistic Fracture Mechanics Evaluation for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii,* October 2018" (ML19297G738). These EPRI reports were developed prior to or around the time the rules for pressurized thermal shock (PTS) were reevaluated, and as such the acceptance criterion for failure frequency in the reports is based on the guidelines for PTS analysis in RG 1.154, "*Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors*" that were available at the time. The NRC staff also noted that the BWR Vessel and Internals Project topical reports included substantive inspection aspects that were critical to the NRC's findings.

Based on the discussion above, the NRC staff finds the use of the acceptance criterion of 1×10^{-6} failures per year for PoF acceptable for the Duke Energy plant-specific alternative request.

3.2.3 Parameters Most Significant to PFM Results

The NRC staff reviewed the submittal for plant-specific aspects of the Duke Energy alternative request that may diverge from the Salem submittal, as explicitly referenced in the Duke Energy request, concerning parameters most significant to PFM results in EPRI report 15905. For the Westinghouse-designed units in the submittal (CNS1/2, HNP, MNS1/2, and RNP) the NRC staff confirmed that the review conclusions in the Salem SE applied to the Duke Energy submittal and found that the parameters most significant to PFM results would be the same and consistent with the NRC staff's reviews documented in the Salem SE, and consequently the approach taken in those reviews appropriately applies to the current review for Duke Energy. For the plant-specific PFM analyses for the ONS units (which are Babcock & Wilcox (B&W)-designed) in Attachment 8 to the submittal, the NRC staff determined that the parameters most significant to PFM would be the same as those in the EPRI analyses because the inputs and methodology in the ONS-specific analyses are similar or the same as those used in EPRI report 15905.

As discussed in the Salem SE, the sensitivity analysis (SA), sensitivity study (SS), and the NRC staff's observations on the PROMISE software identified the following significant parameters or aspects of the PFM analyses that warrant a close evaluation: stress analysis, fracture toughness, flaw crack growth (FCG) rate coefficient (or simply FCG rate), and effect of ISI schedule and examination coverage. The NRC staff also evaluated the flaw density of 1.0 per weld used by the licensee for the subject PZR welds. The NRC staff discussed and closely evaluated each in the next five sections of this SE. The NRC staff also evaluated other parameters or aspects of the analyses in Section 3.2.9 of this SE.

3.2.4 Stress Analysis

3.2.4.1 Selection of Components and Materials

In Attachments 1 through 5 of the submittal, the licensee evaluated the plant-specific applicability of the components and materials selected and analyzed in EPRI report 15905 to the subject PZR welds of Duke Energy. The acceptability of meeting the criteria, however, depends on the acceptability of the component and material selection described in the EPRI report, which the NRC staff evaluated below.

In Section 4 of EPRI report 15905, EPRI discussed the variation among PZR designs. EPRI used this information for finite element analyses (FEA, see Section 3.2.4.4 of this SE) to determine stresses in the analyzed components, which the licensee referenced for the corresponding Westinghouse-designed PZRs requested for Duke Energy. In selecting the components, EPRI considered geometry, operating characteristics, materials, field experience with respect to service-induced cracking, and the availability and quality of component-specific information.

The NRC staff reviewed Section 4 of EPRI report 15905 and finds that the PZR configurations selected in the report for stress analysis are acceptable representatives for the corresponding PZR components requested for the Westinghouse-designed Duke Energy units in the plant-specific alternative request. Specifically, the radius-to-thickness (R/t) ratios of these Duke Energy PZR components, provided in Tables 1 and 2 of the Enclosure to the submittal, are either bounded by or equivalent to the R/t ratios analyzed in the EPRI report. To verify the dominance of the R/t ratio, the NRC staff reviewed the through-wall stress distributions in Section 7 of the EPRI report to confirm that the pressure stress is dominant, which would confirm the dominance of the R/t ratio. Accordingly, the NRC staff finds that EPRI's conclusion about the R/t ratio being the dominant parameter in evaluating the various PZR configurations to be acceptable for the Westinghouse-designed Duke Energy units in the plant-specific alternative request.

In Attachment 7 to the submittal, the licensee developed finite element models based on the unique plant-specific geometry, namely the heater bundle shell region, of the ONS PZRs. The licensee used this model for the plant-specific FEA for the ONS PZRs in Attachment 7 to the submittal.

Additionally, Tables 1-1, 2-1, 3-1, 4-1, and 5-1 of the submittal state that the Duke Energy PZR shell/head and nozzles meet the applicability criteria in EPRI report 15905 regarding weld and nozzle configuration. The NRC staff confirmed that the EPRI report criteria regarding weld and nozzle configuration are met.

Section 9.4 of EPRI report 15905 addresses criteria for plant-specific applicability of the analysis and indicates that materials are acceptable if they conform to ASME Code, Section XI, Nonmandatory Appendix G, paragraph G-2110. The licensee addressed these criteria in Tables 1-1, 2-1, 3-1, 4-1, and 5-1 of the submittal. The licensee stated that the materials of construction for the PZR components are as reported in Table 2. The licensee addressed the fracture toughness of ONS PZR vessel head and shell materials in Section 2.2.5 of Attachment 8 to the submittal.

Table 2: Materials of Construction

Unit	**PZR Component**	**Material**
CNS1/2	vessel heads	SA-533, Grade A, Class 2
	vessel shell	SA-533, Grade A, Class 2
	nozzles	SA-508, Class 2
MNS1/2	vessel heads	SA-533, Grade A, Class 2
	vessel shell	SA-533, Grade A, Class 2
	nozzles	SA-508, Class 2A
HNP	vessel heads	SA-533, Grade A, Class 2
	vessel shell	SA-533, Grade A, Class 2
	nozzles	SA-508, Class 2A
RNP	vessel heads	SA-302, Grade B
	vessel shell	SA-302, Grade B
	nozzles	N/A (not seeking relief for nozzle-to-vessel welds)
ONS1/2/3	vessel heads	SA-212, Grade B or SA-516, Grade 70
	vessel shell	SA-212, Grade B or SA-516, Grade 70
	nozzles	SA-508, Grade 1, Class 1

The NRC staff evaluated the ONS PZR vessel head and shell materials in Section 3.2.5 of this SE and verified that all other materials in Table 2 above conform with ASME Code Section XI, Nonmandatory Appendix G. Therefore, the NRC staff finds that the materials for Duke Energy meet the material applicability criterion.

Based on the discussion above, the NRC finds that Duke Energy PZRs are acceptable with regard to component configuration and materials.

3.2.4.2 Selection of Transients

In Section 5.2 of EPRI report 15905, EPRI discussed the thermal and pressure transients under normal and upset conditions considered relevant to the PZR shell and associated welds. EPRI developed a list of transients for analysis applicable to the PZRs analyzed in the report, based on transients that have the largest temperature and pressure variations.

The NRC staff evaluated the transient selection in EPRI report 15905 in detail, as discussed in the Salem SE. The NRC staff confirmed that the applicable aspects of the transients discussed in the Salem SE apply equally to this review for Duke Energy. The NRC staff reviewed the discussion of transients in Section 5.2 of EPRI report 15905 and determined that the transient selection defined in the report is reasonable for the Duke Energy plant-specific alternative request because the selection was based on large temperature and pressure variations that are conducive to FCG and are expected to occur in PWRs. The NRC staff then compared the analysis in the EPRI report to plant-specific information provided in the licensee's submittal.

In Tables 1-3 and 1-4 (CNS1/2), Tables 2-3 and 2-4 (MNS1/2), Tables 3-3 and 3-4 (HNP), Tables 4-3 and 4-4 (RNP), and Tables 5-3 and 5-4 of the submittal (ONS1/2/3) , the licensee evaluated the plant-specific applicability of the transients selected in EPRI report 15905 to the PZRs of Duke Energy. The NRC staff reviewed these tables and confirmed that the Duke Energy PZRs are bounded by the criteria in the EPRI report.

In the analyses in EPRI report 15905 there were no separate test conditions included in the transient selection. The licensee stated on page 9 of 23 of the Enclosure to the submittal that pressure tests (i.e., system leakage tests) for Duke Energy are performed at normal operating conditions and no hydrostatic testing has been performed since the plant began operation. The NRC staff noted that since the pressure tests are performed at normal operating conditions, they are part of Heatup/Cooldown, and therefore test conditions need not be analyzed as a separate transient.

Based on the discussion above, the NRC staff finds that Duke Energy meets the transient applicability criteria in EPRI report 15905. Therefore, the analyzed transient loads for the requested PZR welds at Duke Energy are acceptable.

3.2.4.3 Other Operating Loads

Weld residual stress and clad residual stress are addressed in EPRI report 15905. In Section 5.0 of the Enclosure to the submittal, the licensee described the use of the same weld residual stress and clad residual stress in the plant-specific PFM evaluations for the ONS units. The NRC staff documented the review of weld residual stress and clad residual stress analyzed in the EPRI report in the Salem SE. The NRC staff confirmed that no Duke Energy plant-specific aspects of this submittal warranted additional consideration, noting in particular (1) the relatively low sensitivity of the EPRI results on residual stress (Table 8-14 of EPRI report 15905) and the SS conducted on stress; and (2) the small impact of clad residual stress on the PFM results. Based on this, the NRC staff finds that there is a very low probability that plant-specific aspects of other operating loads (e.g., weld residual stress and clad residual stress) would have a significant effect on the probability of leakage or rupture beyond the studies documented in the EPRI report.

Based on the discussion above, the NRC staff finds the treatment of other loads described in this section of the SE acceptable for the requested PZR welds of Duke Energy.

3.2.4.4 Finite Element Analysis

The NRC staff reviewed the fine element analysis (FEA) conducted in EPRI report 15905 and documented its review in detail in the Salem SE. For the Westinghouse-designed units in the submittal, the NRC staff confirmed that no Duke Energy plant-specific aspects of this application warranted further review.

In the plant-specific stress analysis for the ONS units in Attachment 7 to the submittal, the licensee performed FEA based on the PZR geometric configuration of the ONS units. In the supplement, the licensee demonstrated through stress calculations that the impact of piping interface loads on the stresses at the bottom head welds of ONS PZRs was small. The NRC staff reviewed the FEA in Attachment 7 to the submittal and the supplement and determined that the modeling approach used with regards to element types, boundary conditions, and symmetry assumptions, is similar to that used in the FEA in EPRI report 15905.

Based on the discussion referenced above, the NRC staff determined that the pressure and thermal stresses calculated through FEA in EPRI report 15905 and in Attachment 8 of the submittal are acceptable for the requested PZR welds of Duke Energy.

3.2.5 Fracture Toughness

In EPRI report 15905, EPRI assumed for fracture toughness of ferritic materials an upper-shelf K_{IC} value of 200 ksi√in based on the upper-shelf fracture toughness value in the ASME Code, Section XI, A-4200. EPRI treated K_{IC} as a random parameter normal distribution with a mean value of 200 ksi√in and a standard deviation of 5 ksi√in, stating that these assumptions are consistent with the BWRVIP-108 project. Further discussion of this topic as it relates to the EPRI reports, and to plant-specific applications, is contained in the Salem SE. The NRC staff confirmed that the evaluations documented in the Salem SE apply to the Westinghouse-design units in the Duke Energy submittal without further plant-specific considerations. As discussed in Section 3.2.4 of this SE, Duke Energy meets the material criteria in the EPRI report, and thus the NRC staff determined that the assumed fracture toughness parameters above are applicable to the Westinghouse-design units in the submittal.

In the plant-specific PFM evaluations for the ONS units in Attachment 8 to the submittal, the licensee used a K_{IC} of 106 ksi√in, based on ASME Code, Section XI, Appendix C, as a conservative lower bound fracture toughness for the ONS PZR materials. The NRC staff finds this lower bound fracture toughness value acceptable because the licensee compared it with the value based on Charpy V-notched absorbed energy of the material and determined the former to be a lower value.

Based on the discussion referenced above and the discussion in Section 3.2.4 of this SE, which confirmed that the materials are acceptable for the requested PZR welds of Duke Energy, the NRC staff finds the fracture toughness models in the referenced EPRI report and in Attachment 8 to the submittal acceptable for the requested PZR welds of Duke Energy.

3.2.6 Flaw Density

In Section 5.0 of the Enclosure, the licensee stated that a flaw density of 1.0 per weld was used in the evaluations in EPRI report 15905. Additionally, in the plant-specific PFM evaluations for the ONS units in Attachment 8 to the submittal, the licensee used a flaw density of 1.0 per weld. This flaw density is based on the flaw density the NRC staff determined acceptable as documented in the December 19, 2007, SE for BWRVIP-108 (ML073600374). Using this flaw density and estimated volumes of the subject PZR welds, the NRC staff finds that the assumed flaw density for the subject PZR welds is reasonable. Based on this discussion and the discussion in Section 3.2.4 of this SE, which confirmed that the materials and geometric criteria are acceptable for the requested PZR welds of Duke Energy, the NRC staff finds the appropriate flaw density has been considered, and therefore acceptable, for the requested PZR welds of Duke Energy.

3.2.7 Fatigue Crack Growth Rate

The NRC staff reviewed the FCG rate used in EPRI report 15905 and documented its review in detail in the Salem SE. The NRC staff confirmed that no plant-specific aspects of the Duke Energy submittal warranted further review with regards to FCG rate. Based on the discussions referenced above, the NRC staff finds that the ASME Code, Section XI, A-4300 FCG rate used in EPRI report 15905 is acceptable for the requested PZR welds of Duke Energy.

3.2.8 ISI Schedule and Examination Coverage

EPRI analyzed various ISI schedules (or scenarios) in Chapter 8 of EPRI report 15905. For the Westinghouse-designed units, the NRC staff reviewed the applicable aspects of the ISI schedule and examination coverage modeling used in the EPRI report and documented its review in detail in the Salem SE. Because ONS1, 2, and 3 are B&W-designed units, the licensee provided a separate analysis that included the plant-specific ISI scenarios for the units. The various ISI scenarios considered are shown in Table 3 of this SE.

The licensee provided information on the inspection history of the requested PZR welds of Duke Energy in the following tables in the submittal: Tables 1-5 and 1-6 (CNS1/2), 2-5 and 2-6 (MNS1/2), 3-5 (HNP), 4-5 (RNP), and 5-5 through 5-7 (ONS1/2/3). These tables indicate that, except for MNS1, there were no unacceptable indications found during these examinations, i.e., no flaws that exceeded the ASME Code, Section XI acceptance standards. In the supplement, the licensee clarified that these tables include all indications detected and evaluated for continued service. For MNS1, the licensee stated that a flaw evaluation for the 1PZR-1 weld (Circumferential Lower Shell-to-Head weld) of MNS1 determined that the five detected circumferential subsurface indications were acceptable per ASME Code, Section XI, IWB-3600, for continued service for the intended service life of the vessel. The licensee also stated that per ASME Code Case N-526, re-examination was not required and that previously recorded indications have remained the same and no new indications were identified during the most recent 4th ISI interval examination. The NRC staff confirmed that ASME Code Case N-526 is listed in Regulatory Guide 1.147, Revision 20, as an acceptable code case without conditions and that the licensee can use the code case without NRC review and approval.

Finally, the inspection history shows that some of the examination coverages did not meet the ASME Code, Section XI examination coverage requirement of 90 percent or greater. However, licensees are required to submit a relief request under 10 CFR 50.55a(g)(5)(iii) for ASME Code, Section XI examination requirements that are determined by the licensee to be impractical, which typically includes examination coverages that do not meet the requirement. The NRC staff also noted an examination coverage of as low as 25.2 percent for the ONS units. The NRC staff discussed the impact of this low examination coverage on the PFM results in Section 3.2.10 of this SE.

Based on this discussion, the NRC staff finds the Duke Energy inspection history of the subject PZR welds to be acceptable. Thus, given the discussion above on the inspection history of the requested PZR welds of Duke Energy, the NRC staff finds that the PFM analyses of EPRI report 15905 adequately represent the requested components for Duke Energy with respect to ISI schedule and examination coverage.

3.2.9 Other Considerations

The NRC staff reviewed the application and associated references concerning initial flaw depth and length distribution, probability of detection, models, uncertainty, convergence, and DFM analysis. The NRC staff previously reviewed the applicable aspects of these topics as used in EPRI reports 15905 and documented their review in detail in the Salem SE. For the Westinghouse-designed units, the NRC staff confirmed that no plant-specific aspects of the submittal warranted further review. The NRC staff reviewed the DFM analyses the licensee provided in Section 2.0 of Attachment 8 to the submittal for ONS1, 2, and 3. The licensee performed the DFM analyses with an initial flaw depth based on the maximum depth specified in the ASME Code, Section XI, acceptance standards. The results of the licensee's DFM analyses for the PZR welds of the ONS units showed that it takes a minimum of 208 years for the total applied stress intensity factor to reach the fracture toughness. This result provides additional confirmatory information to the PFM analyses that were performed for a period of 80 years. Based on this discussion, the NRC staff finds that the submittal is acceptable with regards to these modeling aspects, and therefore, is acceptable for the requested PZR welds of Duke Energy.

3.2.10 PFM Results Relevant to Proposed Alternative

In EPRI report 15905, EPRI stated that performing only PSI examination without any other post-PSI examinations is acceptable for 80 years of plant operation while maintaining plant safety. The NRC staff does not find this conclusion acceptable since it does not account for the effect of the combination of the most significant parameters or the added uncertainty of low probability events. More significantly, the NRC staff considers this conclusion to be a solely risk-based approach inconsistent with NRC policy that calls for risk insights to be considered together with other factors rather than sole reliance on risk-based approaches. Post fabrication examinations are critical in supporting necessary performance monitoring goals including monitoring and trending; bounding uncertainties; validating/confirming analytical results; and providing timely means to identify novel and/or unexpected degradation.

Duke Energy evaluated the PFM scenarios on pages 11-14 of the Enclosure to the submittal in order to determine which scenarios best represent the inspection history of the subject Duke PZRs. The scenarios identified by the licensee are found in Table 3.

Table 3: Examination Scenarios in the PFM Study

Unit	PFM Scenario
CNS1	PSI+10+20+30+60
CNS2	PSI+10+20+30+60
MNS1	PSI+10+20+30+60
MNS2	PSI+10+20+30+60
ONS1/2/3	PSI+10+20+30+40+70
HNP	PSI+10+20+30+60
RNP	PSI+10+20+30+40+70

Duke Energy stated that the limiting scenario this alternative request is PSI+10+20+30+60 for the Westinghouse-designed units, i.e., CNS1 and 2, HNP, MNS1 and 2, and RNP. The licensee then performed a SS evaluation with this limiting scenario since this scenario was not specifically considered in EPRI report 15905 using the limiting case, PRSHC-BW-2C, in the report. The licensee also performed separate PFM evaluations for the ONS1, 2, and 3, because they are B&W-designed units. The licensee reported that these analysis cases resulted in PoFs less than the acceptance criterion of 1×10^{-6} per year. The licensee performed an additional SS for ONS1, 2, and 3 with a limiting examination coverage of 25.2 percent and reported that the PoF is less than 1×10^{-6} per year.

3.2.11 Performance Monitoring

Performance monitoring, such as ISI programs, is a necessary component described by the NRC five principles of risk-informed decision making. Analyses, such as PFM, work along with performance monitoring to provide a mutually supporting and diverse basis for facility condition and maintenance that is within its licensing basis. An adequate performance monitoring program must provide direct evidence of the presence and extent of degradation, validation of continued appropriateness of associated analyses, and a timely method to detect novel/unexpected degradation. The NRC staff described these characteristics at various public meetings (see ML22060A277, ML23033A667, and ML23114A034).

The initial proposed alternative for Duke Energy would have resulted in a significant amount of time before another examination was performed on the subject welds under the submittal. The NRC staff requested the licensee to provide a performance monitoring plan in order to verify that the assumptions of the PFM analysis remain valid throughout the period of the proposed alternative and can provide a timely method to detect novel/unexpected degradation (see RAI-1 in the July 20, 2023, supplement).

In response to RAI-1, the licensee proposed 10 PZR weld examinations to be performed throughout the period of the proposed alternative across 5 of the 9 units covered by the proposed alternative (i.e., CNS1 and 2, HNP, and MNS1 and 2). This is 10 PZR weld examinations out of the 50 ASME Code required examinations for these five units for the PZR. Each of the ASME Code item numbers covered by the proposed alternative (i.e., B2.11, B2.12, and B3.110) will be examined as part of the proposed performance monitoring plan. The licensee also stated that for the 6th ISI intervals of CNS1 and 2 and HNP, the ASME Code, Section XI required examination will resume. The licensee's proposed performance monitoring plan schedule is shown in Figure 1-1 of Enclosure 1 to the July 20, 2023, supplement, reproduced in this SE as Figure 1.

The NRC staff reviewed the proposed performance monitoring plan in terms of PZRs, rather than in terms of number of required Item No. examinations. ASME Code, Section XI requires that one PZR (i.e., all welds specified in the ASME Code, Section XI, for the PZR) be examined per unit per ISI interval. Accordingly, Table 4 shows the NRC staff's calculation of the number of total PZRs required to be examined by the ASME Code, Section XI for the licensee's proposed alternative.

Table 4: Calculation of Total ASME Code Required PZRs

Site	# of units	# of ISI intervals	ASME Code Required PZRs = units × intervals
CNS	2	2	4
MNS	2	2	4
HNP	1	2	2
ONS	3	1	3
RNP	1	1	1
Total			14

The NRC staff determined, through binomial statistics and Monte Carlo methods, that a 25 percent sample of the total ASME Code required number of PZRs would be an adequate performance monitoring sample over the subject alternative period. This leads to a sample of 0.25 × 14 = 4 PZRs (rounding up). The PZR equivalents proposed in the licensee's performance monitoring plan is shown in Table 5.

Table 5: Number of PZR Equivalent Exams

Unit	# of Section XI exams	# of performance monitoring exams	PZR Equivalents = PM exams / required exams
Catawba 1	10	2	0.2
Catawba 2	10	2	0.2
McGuire 1	10	2	0.2
McGuire 2	10	2	0.2
Harris	10	2	0.2
Total			1.0

In addition to the PZR equivalents calculated in Table 5, CNS1 and 2 and HNP will resume ASME Code, Section XI required exams for the 6th ISI interval for those respective plants. This leads to a total PZR equivalent of 1 + 3 = 4 for the subject time period, which meets the NRC staff position of at least 25 percent sample over the alternative period. Consequently, the NRC staff finds that the quantity of examinations over the subject alternative period is acceptable.

The NRC staff reviewed the timing of examinations to ensure that the proposed examinations in the performance monitoring plan would provide a reasonably continuous source of data supporting the characteristics of acceptable performance monitoring. Specifically, data would continue to become available on a cadence reasonably commensurate with ASME Code requirements, but on a fleet basis rather than an individual unit basis. Based on the proposed examinations during the alternative periods, and periods in which unmodified ASME Code inspections requirements would be in force, the NRC staff finds that the examinations proposed in the performance monitoring plan will provide an appropriately continuous stream of data.

As part of the proposed performance monitoring plan in the supplement, the licensee described actions they would take in the event that degradation was discovered as part of performance monitoring activities. The licensee stated that detected indications would be evaluated and dispositioned according to the rules of ASME Code, Section XI. Furthermore, the licensee stated that detected indications exceeding the acceptable criteria of ASME Code, Section XI, IWB-3500 would result in additional examinations at the CNS1 and 2, HNP, and MNS1 and 2, plants within the first or second refueling outage of discovering the indication. The licensee stated that domestic and international operating experience would be entered into the Duke Energy Corrective Action Program to determine if additional examinations are required.

In addition, the submittal the licensee discussed system leakage tests as "providing further assurance of safety" for the proposed alternative. The NRC staff noted that the visual examinations performed during system leakage tests 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 welds. However, the NRC staff noted that leakage tests provide complementary additional performance monitoring to the ISI examinations. This additional assurance increases confidence that the proposed quantity of examinations, in concert with other on-going activities, will provide an acceptable level of performance monitoring for the subject PZR components.

Based on the above discussion and given the supplemental information in the RAI response, the NRC staff determined that inspections for the subject PZR components could be deferred during the proposed period because an adequate level of performance monitoring is maintained for the components.

4.0 CONCLUSION

As set forth above, the NRC staff determined that the licensee's proposed alternative as discussed above for the requested components provides an acceptable level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all the regulatory requirements set forth in 10 CFR 50.55a(z)(1). Therefore, the NRC staff authorizes the use of the proposed alternative in Request for Alternative RA-22-0257 for Duke Energy through the 5th ISI interval for CNS1 and 2 and HNP and through the 6th ISI interval for RNP; MNS1 and 2; and ONS1, 2, and 3.

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.

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Date: October 19, 2023

SUBJECT: CATAWBA NUCLEAR STATION, UNIT NOS. 1 AND 2; SHEARON HARRIS NUCLEAR POWER PLANT, UNIT NO. 1; MCGUIRE NUCLEAR STATION, UNIT NOS. 1 AND 2; OCONEE NUCLEAR STATION, UNIT NOS. 1, 2, AND 3; AND H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 – ISSUANCE OF ALTERNATIVE TO PRESSURIZER WELDS (EPID L-2023-LLR-0020) DATED OCTOBER 19, 2023

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