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

October 5, 2021

Mr. Robert T. Simril
Site Vice President
Catawba Nuclear Station
Duke Energy Carolinas, LLC
4800 Concord Road
York, SC 29745

SUBJECT: CATAWBA NUCLEAR STATION, UNITS 1 AND 2 – STAFF EVALUATION RELATED TO AGING MANAGEMENT PROGRAM AND INSPECTION PLAN OF REACTOR VESSEL INTERNALS PER MRP-227, REVISION 1-A (EPID L-2020-LLQ-0006)

Dear Mr. Simril:

By letter dated October 15, 2020, Duke Energy Carolinas, LLC (Duke Energy, the licensee) submitted the aging management program (AMP) and inspection plan for the Catawba Nuclear Station (Catawba), Units 1 and 2, reactor vessel internals (RVI), to the U.S. Nuclear Regulatory Commission (NRC) for review and approval.

Materials Reliability Program (MRP) 227, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guideline," Revision 1-A, and its supporting reports were used as the technical bases for developing the Catawba RVI AMP and inspection plan. This submittal is to fulfill regulatory commitment No. 14 in its NRC-approved license renewal application as documented in Appendix D of NUREG-1772, "Safety Evaluation Report Related to the License Renewal of McGuire Nuclear Station, Units 1 and 2, and Catawba Nuclear Station, Units 1 and 2."

The NRC staff has reviewed the inspection plan for the Catawba RVI AMP and inspection plan and verified that it is consistent with the inspection and evaluation guidelines of MRP-227, Revision 1-A, and is, therefore, acceptable. The licensee addressed the action item specified in MRP-227, Revision 1-A, appropriately. Therefore, the NRC staff concludes that the licensee has met license renewal commitment No. 14, as documented in Appendix D of NUREG-1772.

If you have any questions, please contact me at (301) 415-0615 or by e-mail at Zackary.Stone@nrc.gov.

Sincerely,

Zackary R. Stone

Digitally signed by Zackary R. Stone
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Zackary R. Stone, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos: 50-413 and 50-414

Enclosure:
Staff Evaluation

cc: Listserv

STAFF EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

AGING MANAGEMENT PROGRAM AND INSPECTION PLAN OF

REACTOR VESSEL INTERNALS PER MRP-227 REVISION 1-A

DUKE ENERGY CAROLINAS, LLC

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-413 AND 50-414

1.0 INTRODUCTION

By letter dated October 15, 2020 (Reference 1), Duke Energy Carolinas, LLC (Duke Energy, the licensee) submitted the aging management program (AMP) and inspection plan for the Catawba Nuclear Station (Catawba), Units 1 and 2, reactor vessel internals (RVI).

The Catawba RVI AMP and inspection plan is based on Materials Reliability Program (MRP) 227, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guideline," Revision 1-A (Reference 2). The licensee submitted the Catawba RVI AMP and inspection plan to fulfill Regulatory Commitment No. 14 in its U.S. Nuclear Regulatory Commission (NRC)-approved license renewal application as documented in Appendix D of NUREG-1772, "Safety Evaluation Report Related to the License Renewal of McGuire Nuclear Station, Units 1 and 2, and Catawba Nuclear Station, Units 1 and 2" (Reference 3). The AMP and inspection plan describes RVI inspections through the period of extended operation.

Attachment 1 of the licensee's letter dated October 15, 2020, contains the Catawba RVI AMP and inspection plan, and Attachment 2 contains the fuel management and fuel design to address the applicability limitations in MRP-227, Revision 1-A.

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR) CFR Part 54, "Requirements for renewal of operating licenses for nuclear power plants," addresses the requirements for plant license renewal process. The regulation at 10 CFR 54.21, "Contents of application - technical information," requires that each application for license renewal contain an integrated plant assessment and an evaluation of time-limited aging analyses. The plant-specific integrated assessment shall identify and list those structures and components subject to an aging management review and demonstrate that the effects of aging (e.g., cracking, loss of material, loss of fracture toughness, dimensional changes, and loss of preload) will be adequately managed so that their intended functions will be maintained consistent with the current licensing basis during the period of extended operation as required by 10 CFR 54.29(a).

Structures and components subject to an AMP shall encompass those structures and components that are referred to as "passive" and "long-lived." Passive structures and components perform an intended function, as described in 10 CFR 54.4, without moving parts or without a change in configuration or properties. Long-lived structures and components are not subject to replacement based on a qualified life or specified time period. The scope of components considered for inspection under MRP-227, Revision 1-A, includes core support structures, typically denoted as Examination Category B-N-3 by the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, and those RVI components that serve an intended license renewal safety function pursuant to criteria in 10 CFR 54.4(a)(1). The scope of the program does not include active RVI components (e.g., vent valve discs, shafts, or hinge pins), or consumable components such as fuel assemblies, reactivity control assemblies, and nuclear instrumentation because these components are not typically within the scope of the components that are required to be subject to an AMP as defined by the criteria set forth in 10 CFR 54.21(a)(1).

The NUREG-1800, “Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants” (Reference 4), includes that NRC staff should confirm that aging degradation effects should be managed in RVI component when entering the period of extended operation. Some owners of pressurized-water reactors (PWR) units were granted renewed licenses contingent on a commitment to conform to the recommendations specified in Section XI.M16 of NUREG-1801, "Generic Aging Lessons Learned (GALL) Report – Final Report," Revision 2 (Reference 5). Section XI.M16 requires that the applicant provide a commitment in the final safety analysis review supplement to: (1) participate in the industry programs for investigating and managing aging effects on RVI components; (2) evaluate and implement the results of the industry programs as applicable to the RVI components; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for RVI components to the NRC for review.

3.0 TECHNICAL EVALUATION

3.1 NRC Staff Assessment Approach

The NRC staff assessment of the Catawba RVI AMP and inspection plan focused on determining whether the licensee adequately incorporated the guidelines specified in MRP-227, Revision 1-A, and addressed the action item in the NRC safety evaluation (SE) of the topical report.

3.2 Catawba RVI AMP and Inspection Plan

By letter dated June 16, 2010 (Reference 6), the licensee notified the NRC of its intent to revise its commitments for RVI inspections from those that currently exist in Section 18.2.22 of Catawba’s updated final safety analysis report (Reference 7) to the NRC-approved inspection guidelines based on MRP-227. As part of its license renewal application, the licensee stated that the Catawba RVI program is subject to future enhancements as the industry understanding of degradation continues to change.

By letter dated March 19, 2014 (Reference 8), the licensee notified the NRC of its intent to submit a Catawba RVI inspection plan to implement MRP-227-A (or the latest NRC approved revision, currently MRP-227, Revision 1-A) no later than two years before the initial inspection. The licensee stated that the expected initial RVI inspection dates are Fall 2024 and Fall 2025 for Catawba, Units 1 and 2, respectively, but may be subject to change. The 40-year operating licenses for Catawba, Units 1 and 2, end on December 6, 2024 and February 24, 2026, respectively. By letter dated December 5, 2003 (Reference 9), the NRC issued the renewed facility operating licenses for Catawba, Units 1 and 2, that end on December 5, 2043 for both Units.

The licensee stated that Catawba RVI program uses a combination of prevention, mitigation, and condition monitoring. Where applicable, credit is taken for existing programs, such as water chemistry program, the ASME Section XI inservice inspection (ISI) program, control rod guide tube support pin replacement project, and the bottom-mounted instrumentation thimble tube program. The licensee stated that the existing programs are augmented with the inspections and evaluations recommended by MRP-227, Revision 1-A. The licensee also uses a companion industry guide, MRP-228, Revision 3, “Materials Reliability Program: Inspection Standard for Pressurized Water Reactor Internals – 2018 Update” (Reference 10), which provides guidance on the qualification and/or demonstration of the required nondestructive evaluation (NDE) techniques and other criteria pertaining to the actual performance of the inspections. The licensee stated that the revision of MRP-228 in effect at the time of the inspection will be used for future inspections.

The Electric Power Research Institute (EPRI) in cooperation with the Pressurized-Water Reactor Owners Group (PWROG) developed WCAP-17096-NP, Revision 2, “Reactor Internals Acceptance Criteria Methodology and Data Requirements” (Reference 11), for inspections performed under MRP-227, Revision 1-A. By letter dated January 3, 2017 (Reference 12), the NRC approved the generic use of WCAP-17096-NP-A, Revision 2.

The licensee stated that WCAP-17096-NP-A, Revision 2, and the guidance within PWROG-17071-NP, Revision 0, “WCAP-17096-NP-A Interim Guidance” (Reference 13), provides an acceptable methodology to evaluate inspection findings from MRP-227 examinations. The interim guidance within PWROG-17071-NP focuses on areas where the currently approved acceptance criteria guidance within WCAP-17096-NP-A, Revision 2, is inconsistent with other industry guidance that has been issued since the NRC approval of WCAP-17096-NP-A, Revision 2.

By letter dated July 31, 2019, PWROG transmitted for NRC review and approval WCAP-17096-NP, Revision 3, which includes the interim guidance from PWROG-17071-NP and updates to the acceptance criteria guidance for MRP-227, Revision 1, and applicability to initial license renewal (60-year operating license) and subsequent license renewal (80-year operating license) based on current industry knowledge. The NRC staff has not approved WCAP-17096-NP, Revision 3, but once approved, the licensee may use the NRC-approved WCAP-17096-NP to perform inspections for the Catawba RVI components.

The NRC staff has approved the licensee’s aging management of Catawba RVI components as discussed in NUREG-1772. Section 5 and Appendix B of Catawba RVI AMP and inspection plan state that the Catawba RVI AMP complies with: (1) GALL report, (2) NRC Final License Renewal Interim Staff Guidance (ISG) LR-ISG-2011-04, “Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized Water Reactors” (Reference 14), and (3) MRP-227, Revision 1-A. The licensee committed to inspect RVI components as shown in Commitment No. 14 in Appendix D to NUREG-1772. Table B-1 of Catawba RVI AMP and inspection plan provides a listing of the Catawba, Units 1 and 2, RVI components as part of license renewal application.

The GALL report and LR-ISG-2011-04 reference MRP-227-A and not MRP-227, Revision 1-A. The licensee stated that its commitment to implement MRP-227, Revision 1-A, necessitates that the aging management strategy in the original license renewal application be updated. The NRC staff finds that as RVI components age, an appropriate aging management is needed to minimize and detect RVI degradation early to allow for the repair or replacement of the degraded component(s). The NRC staff notes that MRP-227, Revision 1-A, does not alter the RVI AMP of the GALL report and can be used in place of MRP-227-A. Therefore, the NRC staff finds it acceptable that the Catawba RVI AMP and inspection plan follows MRP-227, Revision 1-A.

3.3 MRP-227, Revision 1-A

By letter dated January 9, 2012 (Reference 15), EPRI submitted the NRC-approved MRP-227-A, Revision 0, for generic use which contains the NRC's SE. By letter dated December 3, 2019 (Reference 2), EPRI submitted the NRC-approved version, MRP-227, Revision 1-A, for generic use, which also contains the NRC's SE.

By letter dated February 19, 2020 (Reference 16), the NRC identified items in MRP-227, Revision 1-A, that required clarification. By letter dated May 4, 2020 (Reference 17), EPRI clarified items in MRP-227, Revision 1-A. By email dated June 9, 2020 (Reference 18), the NRC accepted and verified that clarification in the EPRI letter represents an addendum or errata of MRP-227, Revision 1-A, and no further changes were needed to be made to the MRP-227, Revision 1-A.

MRP-227, Revision 1-A, identifies RVI components that require inspections with associated schedules. For each component, the topical report requires a specific type of NDE, specifies evaluation criteria to disposition inspection results, and provides implementation procedures. The topical report uses a screening and ranking process to aid in identifying required inspections for RVI components. The topical report identifies and considered the following eight aging mechanisms that affect RVI components: stress corrosion cracking, irradiation-assisted stress corrosion cracking, wear, fatigue, thermal aging embrittlement, irradiation embrittlement, void swelling and irradiation growth, thermal and irradiation-enhanced stress relaxation or irradiation-enhanced creep. The topical report categorizes the RVI components as "primary," "expansion," "existing programs," and "no additional measures."

Primary components are those RVIs that are highly susceptible to the effects of at least one of the eight aging mechanisms. The primary components also include those which have shown a degree of tolerance to a specific aging degradation effect, but for which no highly susceptible component exists or for which no highly susceptible component is accessible.

Expansion components are those RVIs that are highly or moderately susceptible to the effects of at least one of the eight aging mechanisms, but for which engineering evaluations and safety assessments have shown a degree of tolerance to those effects. The schedule for implementation of aging management requirements for expansion components will depend on the findings from the examinations of the primary components at individual plants. The expansion group also consists of those components for which an increased scope of the primary component sample is specified based on degradation detected in the primary sample (i.e., increased sampling of a primary component).

Existing program components are those RVIs that are susceptible to the effects of at least one of the eight aging mechanisms and for which generic and plant-specific existing AMP elements (e.g., ASME Code, Section XI) are sufficient for managing those effects.

No additional measures components are those RVIs for which the effects of all eight aging mechanisms are below the screening criteria. Additional components were placed in the no additional measures group as a result of the failure modes, effects, and criticality analysis (FMECA); the engineering evaluations; and safety assessments. No further action is required to manage the aging of the no additional measures components.

Section 4 of MRP-227, Revision 1-A, provides specific examination methods, examination frequencies/schedules, and examination coverages for the above RVI components. The topical report clarifies that categorization and analysis are not intended to supersede any requirements in the ASME Code, Section XI. The topical report further clarifies that any components that are classified as core support structures, as defined in the ASME Code, Section XI, IWA-9000, and covered by the ASME Code, Section XI, Table IWB-2500-1, Category B-N-3, have requirements that remain in effect and may only be altered as allowed by 10 CFR 50.55a or plant-specific licensing documentation.

Action Item 1 in Section 4.0 of the NRC SE for MRP-227, Revision 1-A, requires licensees to inspect baffle former bolts and submit evaluations to the NRC. Catawba's response to Action Item 1 is discussed further in this SE. The NRC staff requires that applicants or licensees who wish to use this topical report submit to the NRC its RVI inspection program in accordance with its existing license renewal commitment.

3.4 Existing Programs

MRP-227, Revision 1-A, credits existing programs that manage aging effects of RVI components. Below are the existing relevant programs at Catawba, Units 1 and 2.

Chemistry Control Program

The licensee stated that the Catawba, Units 1 and 2, chemistry control program is an existing program that monitors and controls the chemistry of the reactor coolant system to minimize aging effects of RVI components such as cracking and loss of material that are caused by corrosion and stress corrosion cracking. This program includes provisions specified for the verification of proper chemistry control and aging management, such that the intended functions of plant components will be maintained during the period of extended operation. The program includes periodic sampling of primary water for the known detrimental contaminants to maintain their concentrations below levels known to result in loss of material or cracking. The NRC staff has approved the Catawba, Units 1 and 2, chemistry control program as part of the license renewal application review as discussed in NUREG-1772. The licensee has not asked to deviate from the approved program; therefore, the NRC staff finds that the chemistry control program is acceptable.

ASME Code, Section XI, ISI Program

The Catawba, Units 1 and 2, ASME Code, Section XI, ISI program is an existing inspection program that includes examinations of the reactor vessel core support structure components in accordance with the ASME Code, Section XI, Subsection IWB-2500. The licensee examines the core support structures using VT-3 visual examination methods each interval (examination

Category B-N-3). Table 4-9 in MRP-227, Revision 1-A, identifies the RVI components to be examined by the ASME Code, Section XI, ISI program. The NRC staff has approved the ISI plan as part of the license renewal application review as discussed in NUREG-1772. The MRP-227, Revision 1-A, uses the same program, therefore, the NRC staff finds that the ISI program is acceptable.

Bottom-Mounted Instrumentation Thimble Tube Inspection Program

Table 4-9 in MRP-227, Revision 1-A, identifies the bottom mounted instrumentation (BMI) system flux thimble tubes as an existing program component that requires inspection. The licensee stated that the Catawba, Units 1 and 2, BMI thimble tube inspection program manages loss of material due to wear of the flux thimble tube materials. It implements the guidance of NRC Bulletin 88-09, "Thimble Tube Thinning in Westinghouse Reactors" (Reference 19), that requires a thimble tube wear inspection procedure be established and maintained. The Catawba, Units 1 and 2, program uses an inspection methodology, such as eddy current testing, to inspect the flux thimble tubes periodically and to monitor wall thinning and predict when tubes would require repair or replacement based on a wall thickness trending report. The program establishes appropriate acceptance criteria (percent through-wall wear) based on industry guidance and includes margin to allow for factors such as instrument uncertainty, uncertainties in wear scar geometry, and other potential inaccuracies, as applicable, to the inspection methodology WCAP-12866, Revision 0, "Bottom Mounted Instrumentation Flux Thimble Wear," (Reference 20). The NRC staff finds that Catawba, Units 1 and 2, BMI thimble tube inspection program satisfies NRC Bulletin 88-09 and is, therefore, acceptable.

Control Rod Guide Tube Support Pin Replacement Project

The control rod guide tube support pins are used to align the bottom of the control rod guide tube assembly into the top of the upper core plate. Because of support pin cracking experienced in the industry, the licensee replaced the original support pins with support pins that were fabricated from Alloy X-750 in the 1980's. Later, industry operating experience showed cracking in Alloy X-750 support pins. The licensee replaced the Alloy X-750 support pins with Type 316 stainless steel (SS) support pins at Catawba, Units 1 and 2, in Fall 2000 and Fall 2001, respectively.

The licensee stated that support pins fabricated from Type 316 SS are not susceptible to primary water stress corrosion cracking which was the primary failure mechanism for Alloy X-750 support pins. MRP-191, Revision 1, "Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design" (Reference 21), categorizes the guide tube support pins as Category A, which is assigned to components for which the aging effects are below the screening criteria or for which aging degradation significance is minimal.

Section 4.5 of MRP-227, Revision 1-A, defines the Type 316 SS support pins as a no additional measures component. Thus, the Type 316 SS support pins are not included in Table 4-9 of MRP-227, Revision 1-A, and do not require a plant-specific AMP. The licensee stated that no additional inspections are required by the supplier or per MRP-227, Revision 1-A, and the support pins should remain functional for the period of extended operation. The NRC staff finds that so far Type 316 SS support pins have not shown significant degradation; therefore, it is acceptable that the Type 316 SS support pins do not require a plant-specific aging management program.

Power Uprating

By letter dated April 29, 2016 (Reference 22), the NRC approved the Catawba, Units 1 and 2, measurement uncertainty recapture (MUR) power uprate. As part of power uprate approval, the NRC staff determined that the effect of the MUR power uprate on the RVI evaluation was acceptable. Therefore, the NRC staff finds that the Catawba RVI AMP and inspection plan is not significantly affected by the MUR power uprate.

3.5 RVI Degradations

Below is a discussion of how the licensee manages the specific RVI degradations with respect to MRP-227-Revision 1-A at Catawba, Units 1 and 2.

Reactor Internals Guide Tube Wear

The PWROG developed a tool to predict wear of the upper internals control rod guide tube (guide card wear) and lower guide tube. WCAP-17451-P, Revision 2, "Reactor Internals Guide Tube Wear – Westinghouse Domestic Fleet Operational Projections" (Reference 23), documents the guide plates (cards) initial inspection schedule and acceptance criteria for Westinghouse nuclear steam supply system (NSSS) designed plants. Table 4-3 of MRP-227, Revision 1-A, referenced WCAP-17451-P noting that the "latest NRC-reviewed or approved version" is applicable. The licensee stated that it followed WCAP-17451-P, Revision 2, for control rod guide plate (card) inspections.

Catawba, Units 1 and 2, are 4-loop plants with a 17x17 A guide tube design that has used ion nitride rod cluster control assemblies (RCCAs). The licensee stated that recent operating experience at U.S. Westinghouse NSSS plants that have 17x17 A or 17x17 AS style guide tubes and have switched to ion nitride RCCAs indicates that the rate of guide card wear has outpaced the wear predications in WCAP-17451-P, Revision 1 (Reference 24). This issue was determined to have a potential nuclear safety consequence and was reported to the NRC as a defect, pursuant to 10 CFR Part 21.

For Catawba, Unit 1, the licensee inspected all 53 guide tubes at rodded locations during the Fall 2018 outage per WCAP-17451-P, Revision 2. The licensee relocated two of the worst-worn guide tubes to unrodded locations in the upper internals and replaced them with unworn spare guide tubes previously in the unrodded positions. All other relevant indications identified were dispositioned with no operability concerns. Additionally, Catawba, Unit 1, replaced the ion nitride RCCAs with chrome plated RCCAs in Spring 2020. Catawba, Unit 1, is following the guidance in WCAP-17451-P, Revision 2, for future management of guide card and lower guide tube continuous guidance wear.

For Catawba, Unit 2, the licensee inspected all 53 guide tubes at rodded locations during the Fall 2016 outage per WCAP-17451-P, Revision 1, criteria. The licensee relocated eight of the worst-worn guide tubes to unrodded locations in the upper internals and replaced them with unworn spare guide tubes previously in the unrodded locations during the Fall 2016 and Spring 2018 outages. The guide card wear measurements from the Fall 2016 outage were reanalyzed with guidance from WCAP-17451-P, Revision 2. Additionally, Catawba replaced the ion nitride RCCAs with chrome plated RCCAs in Fall 2019. Catawba, Unit 2, is following the guidance in WCAP-17451-P, Revision 2, for future management of guide card and lower guide tube continuous guidance wear.

Although the licensee used WCAP-17451-P, Revision 1, in lieu of Revision 2, to monitor the degraded guide tubes for Unit 2 in 2016, the licensee did use WCAP-17451-P, Revision 2, to reanalyze the wear at the guide card. The licensee stated that it will use WCAP-17451-P, Revision 2, for future guide tubes at Unit 2. The licensee will use the latest version of WCAP-17451-P, Revision 2, to inspect the wear occurring at control rod guide tubes in both units; therefore, the NRC staff finds that the aging management for the guide tubes is acceptable.

Baffle-Former Bolt Degradation

The industry has discovered degraded baffle-former bolts in 4-loop downflow plants through MRP-227 inspections. On August 1, 2016, Westinghouse issued Nuclear Safety Advisory Letter (NSAL) 16-01, Revision. 1, “Baffle-Former Bolts” (Reference 25), which categorizes plants based on their susceptibility to baffle former bolt degradation and recommends re-inspection techniques and intervals. On July 25, 2016 (Reference 26), MRP issued MRP 2016-021 which provides interim guidance regarding baffle former bolt inspections for Tier 1 plants as defined in NSAL-16-01. On March 15, 2017 (Reference 27), MRP issued MRP 2017-009 which provided interim guidance for baffle former bolt inspections for all plants. The interim guidance within MRP 2016-021 and MRP 2017-009 was incorporated into MRP-227, Revision 1-A, Table 4-3.

Catawba, Units 1 and 2, are upflow configuration plants and, therefore, fall into the Tier 4 plant category described in NSAL-16-01. An upflow configuration has been shown to reduce the incidence of baffle jetting damage to fuel and to reduce the bolt loads due to pressure differentials across the baffle under normal operating and expected faulted conditions. Based on MRP-227, Revision 1-A, and MRP 2017-009, the licensee will perform a baseline volumetric (UT) examination no later than 35 effective full-power years (EFPY), which corresponds to the Fall 2024 outage for Unit 1 and the Fall 2025 outage for Unit 2.

By letter dated January 17, 2018 (Reference 28), MRP issued MRP 2018-002 regarding baffle-former bolt expansion inspection requirements. The licensee stated that if it discovers significant baffle-former bolt clustering as defined in MRP 2018-002, it will perform a one-time VT-3 visual examination of barrel-former bolts within three fuel cycles. The VT-3 examination coverage will include the barrel-former bolts adjacent to the large clusters of baffle-former bolts with unacceptable indications, as defined in MRP 2018-002.

The licensee will follow the industry guidance to inspect baffle former bolts in addition to following MRP-227, Revision 1-A. Therefore, NRC finds that the aging management for the baffle former bolts is acceptable.

Core Barrel Operating Experience

During Spring 2018 inspections, a PWR plant identified crack-like surface indications at the core support barrel assembly welds, specifically, one vertically-oriented indication at the middle girth weld and 45 indications adjacent to the middle axial weld. On July 17, 2019 (Reference 29), MRP published MRP 2019-009 regarding core barrel and core support barrel inspection requirements. MRP 2019-009 shows that age-related cracking at axial welds is not likely to be any more prevalent or safety significant than cracking at circumferential welds. The licensee stated that, although the MRP-227, Revision 1-A, and Catawba RVI AMP and inspection plan may allow for an inspection to overlook circumferential cracking in an axial weld, the monitoring of girth welds remains a reasonable surrogate for identification of age-related cracking of the core barrel prior to any significant degradation which could impact shutdown capability and core damage. The licensee stated that EVT-1 enhanced visual examination continues to be an acceptable visual examination method for detecting cracking in the barrel welds. The licensee further stated that MRP 2019-009 is effective at all Westinghouse-designed PWR units as of February 1, 2020. Catawba, Units 1 and 2, fall into Group 2 of MRP 2019-009, and the licensee is evaluating the "Good Practice" guidance.

The NRC staff finds that the core barrel operating experience is still evolving, and that the licensee is cognizant of the latest industry guidance regarding the adequacy of the inspections. Since the licensee follows Nuclear Energy Institute (NEI) 03-08 (Reference 30) and MRP-227, Revision 1-A, the NRC staff finds the aging management for the core barrel is acceptable.

Clevis Bearing Stellite Wear Surface and Clevis Insert Bolts

On August 25, 2014, Westinghouse published Technical Bulletin (TB) 14-5, “Reactor Internals Lower Radial Support Clevis Insert Cap Screw Degradation” (Reference 31), which summarizes the operating experience for the clevis bearing stellite wear surface and clevis insert bolts, as well as the root cause and the applicability of the root cause on Westinghouse PWRs. TB-14-5 also discusses the safety implications of the operating experience and root cause of clevis degradation and recommends inspections as part of the AMP. The licensee committed to perform a VT-1 visual examination of the clevis insert fasteners in its license renewal application. The licensee stated that it will follow the recommendations of MRP-227, Revision 1-A, as supplemented by TB-14-5 for the aging management of the clevis.

The licensee will follow the inspection requirements of MRP-227, Revision 1-A, and industry guidance in Westinghouse TB-14-5. Therefore, the NRC staff finds that the licensee’s aging management for clevis bearing stellite wear surface and clevis insert bolts is acceptable.

Thermal Sleeve Flange Wear

On December 7, 2015 (Reference 32), Westinghouse published TB-07-2, Revision 3, “Reactor Vessel Head Adapter Thermal Sleeve Wear,” describing the wear at thermal sleeve flanges associated with the reactor vessel closure head penetration nozzles. On July 9, 2018 (Reference 33), Westinghouse published NSAL-18-1, “Thermal Sleeve Flange Wear Leads to Stuck Control Rod,” notifying the wear in thermal sleeve flange. While there have been no reported events of control rods failing to insert into the core when required, Westinghouse reported this issue to the NRC under 10 CFR Part 21. On May 7, 2019 (Reference 34), the PWROG published PWROG-16003-P, Revision 2, “Evaluation of Potential Thermal Sleeve Flange Wear,” to provide the technical basis and acceptance criteria for evaluating thermal sleeve flange wear.

On February 14, 2020 (Reference 35), Westinghouse published NSAL-20-1, “Reactor Vessel Head Control Rod Drive Mechanism Penetration Thermal Sleeve Cross-Sectional Failure,” which states that Westinghouse plants that operate in a T-cold configuration, and that have thermal sleeves with a collar below the flange, are potentially susceptible to cracking and separation of the flange from the sleeve. Catawba, Units 1 and 2, fall into the Table 1 design category listed in NSAL-20-1. The licensee stated that it is evaluating the operating experience and intends to comply with the recommendations in NSAL-20-1 and the NEI 03-08 “Needed” guidance for the aging management of the thermal sleeves.

The licensee inspected thermal sleeve flanges for wear at Catawba, Unit 2, during the Spring 2018 refueling outage in accordance with Westinghouse TB-07-2 using the evaluation methodology and acceptance criteria within PWROG-16003-P, Revision 1 (Reference 36). This inspection indicated that no rodded locations exceeded the generic separation acceptance criteria established in PWROG-16003-P, Revision 1.

The licensee inspected the thermal sleeve flange for wear at Catawba, Unit 1, during the Fall 2018 refueling outage in accordance with MRP 2018-027 and NSAL-18-1 using the methodology and acceptance criteria within PWROG-16003-P, Revision 1. The licensee indicated that no rodded locations exceeded the generic separation acceptance criteria established in PWROG-16003-P, Revision 1. The licensee stated that it will use plant-specific wear rates and acceptance criteria based on the methodology in PWROG-16003-P for future management of thermal sleeve flange wear at Catawba, Units 1 and 2.

The licensee has inspected the thermal sleeve flanges, which do not have wear exceeding the acceptance criteria, and the licensee will follow the industry guidance for future management of thermal sleeve flange wear at Catawba, Units 1 and 2. Therefore, the NRC staff finds the licensee's aging management of thermal sleeve flange wear is acceptable.

3.6 MRP-227 Revision 1-A Applicability

Section 6 of Catawba RVI AMP and inspection plan discusses how MRP-227, Revision 1-A, is applicable to Catawba, Units 1 and 2, in terms of the reactor design, RVI material specifications, stresses, and operating conditions. To demonstrate the applicability, Section 6 of Catawba RVI AMP and inspection plan discusses how Catawba complies with general assumptions in Section 2.4 of MRP-227, Revision 1-A. Attachment 2 to the October 15, 2020, letter, discusses how Catawba complies with the fuel management and design assumption as specified in MRP-227, Revision1-A.

One of the general assumptions in Section 2.4 of MRP-227, Revision 1-A, states that if major plant-specific differences from the inputs to the FMECA process described in MRP-189 and 191 are identified, then plant owners must determine and document the impact, if any, on the aging management strategy. The licensee stated that Catawba, Units 1 and 2, complies with all general assumptions except the assumption which states that the components and material class of each functional component are as listed in the latest revision MRP-191. The licensee identified two components that were not specifically included within MRP-191, Revision 1, -- the irradiation specimen holder (spring) and the flux thimble anti-vibration sleeves. The irradiation specimen holder (spring) is a subcomponent of the irradiation specimen plug assembly, which is a subcomponent of the irradiation specimen holder assembly. The irradiation specimen holder (spring) was originally included in MRP-191, Revision 0 and Revision 1, as part of the irradiation specimen plug assembly, but the spring piece-part was not specifically listed. The flux thimble anti-vibration sleeves are protective sleeves for the flux thimbles. They were installed within Catawba, Unit 1, to reduce the clearance within the lower internals bottom mounted instrumentation columns and limit the amount of vibratory motion of the flux thimbles. The flux thimble anti-vibration sleeves were not specifically listed alongside the flux thimble tubes or bottom-mounted instrumentation assemblies within MRP-191, Revision 0 or Revision 1. For MRP-191, Revision 2 (Reference 37), both components were added to the scope of the expert panel review and screening, categorization, and ranking. The irradiation specimen holder (spring) and the flux thimble anti-vibration sleeves were assigned to safety and economic Category A and thus require no additional measures. The licensee stated that considering these results from MRP-191, Revision 2, the irradiation specimen holder at Catawba, Units 1 and 2, and the flux thimble anti-vibration sleeves at Catawba, Unit 1, do not require changes to the MRP-227, Revision 1-A, aging management requirements or any further actions. The licensee contended that with no changes to the susceptibility or degradation mechanisms of concern, the FMECA and functionality analysis are still acceptable.

The licensee analyzed the irradiation specimen plug (spring) and flux thimble anti-vibration sleeves based on FMECA method in MRP-227, Revision 1-A. The NRC staff determines that based on its analysis, the licensee does not need to change MRP-227, Revision 1-A, inspection requirements for Catawba, Units 1 and 2, with respect to the irradiation specimen plug (spring) and the flux thimble anti-vibration sleeves. Therefore, the NRC staff find the Catawba RVI inspection plan acceptable. Additionally, the NRC finds that the Catawba RVI AMP and inspection plan satisfies the general assumptions of MRP-227, Revision 1-A.

The NRC staff reviewed Attachment 2 to the October 15, 2020, letter, and determined that the licensee has appropriately addressed the fuel management and design of the Catawba, Units 1 and 2, reactors comply with the fuel management and design assumption specified in MRP-227, Revision1-A, and, therefore, are acceptable.

3.7 NRC Conditions on MRP-227, Revision 1-A

In its review of MRP-227, Revision 1-A, the NRC requested that owners who use the topical report address the five items described within the NRC letter dated February 19, 2020. The licensee stated that out of the five items, only Item No. 5, "Confirmation of Changes to Tabular Footnote References for TR [topical report] Tables 4-1 through 4-6 and 4-8 and 4-9," pertains to Catawba, Units 1 and 2. The licensee explained that within Item No. 5, only Tables 4-3 and 4-6 of MRP-227, Revision 1-A, are applicable to Catawba, Units 1 and 2, and they are related to the footnote numbering changes for various RVI components. As shown in Section 6.1.3 of Catawba RVI AMP and inspection plan, the licensee discussed how MRP-227, Revision 1-A, has addressed Item 5 per the supplemental information provided by EPRI in letter dated May 4, 2020 (Reference 17). The licensee sufficiently clarified Item 5 associated with MRP-227, Revision 1-A; therefore, the licensee has resolved the issues raised by the NRC in its letter dated February 19, 2020.

Action Item 1 of Section 4.0 of the NRC SE for MRP-227, Revision 1, requires licensees to inspect baffle former bolts and submit evaluations to the NRC. The licensee stated that it has not yet inspected baffle-former bolts at Catawba, Units 1 and 2. The licensee further stated that Catawba, Units 1 and 2, are categorized as Tier 4 within NSAL-16-01, Revision 1. The licensee indicated that as directed in MRP-227, Revision 1-A, Catawba will perform the baseline ultrasonic examination of baffle-former bolts no later than 35 EFPY. Catawba will determine a baffle-former bolt re-examination interval based on inspection findings. The licensee explained that if the inspection findings do not meet the examination acceptance criteria defined in Section 5 of MRP-227, Revision 1-A, it will disposition the findings by plant-specific evaluation per the NEI 03-08, Needed Requirement, as specified in Section 7.5 of MRP-227, Revision 1-A. The licensee stated that it will document the findings in the Catawba plant corrective action program. The licensee stated that if atypical or aggressive baffle-former bolt degradation, as defined in MRP 2017-009, is observed, it will follow the interim guidance within MRP 2017-009 to determine the permitted reinspection interval. The licensee stated that it would submit to the NRC the plant-specific evaluation for information within 1 year following the outage in which degradation was found in accordance with the action item.

The NRC staff finds that Catawba, Units 1 and 2, will inspect baffle-former bolts in accordance with MRP-227, Revision 1-A, and the applicable interim guidance. The NRC staff notes that the proposed action is consistent with the requirements in Action Item 1 of the NRC's SE because:

(1) if aggressive baffle-former bolt degradation is discovered, the licensee will submit its evaluation within one year following the outage in which degradation was discovered, and (2) if the inspection interval will be lengthened or will exceed the maximum inspection interval recommended in MRP-2017-009, the licensee will submit to the NRC for information at least one year prior to the end of the current reinspection interval. The NRC staff finds that the licensee has satisfied Applicant/Licensee Action Item 1 of the NRC SE of MRP-227, Revision 1-A, and, therefore, the Catawba RVI AMP and inspection plan is acceptable with respect to Action Item 1.

3.8 Catawba Implementation of MRP-227, Revision 1-A

Section 7 of MRP-227, Revision 1-A, states that the RVI inspection guidance be implemented in accordance with NEI 03-08 protocol which specifies three implementation categories--Mandatory, Needed, and Good Practice. Any deviation from the implementation procedure requires a written report.

The primary component inspection scope is identified in Tables 7-1 and 7-2 of Catawba RVI AMP and inspection plan for Units 1 and 2, respectively. The NRC staff verified that Tables 7-1 and 7-2 are consistent with Table 4-3 of MRP-227, Revision 1-A, as modified by any applicable interim guidance. The NRC notes that the RVI of Catawba, Units 1 and 2, in Tables 7-1 and 7-2 do not contain baffle edge bolts (item number W5), internals hold down spring (W8) and thermal shield flexures (W9) which are shown in Table 4-3 of MRP-227, Revision 1-A. The NRC staff finds this acceptable because Catawba, Units 1 and 2, reactors do not use these three components.

Tables 7-1 and 7-2 of Catawba RVI AMP and inspection plan show the due dates for the inspections of the Catawba primary components, which represent the latest opportunity for Catawba to perform the initial examinations. The licensee stated that Catawba may inspect primary components earlier than the due dates listed. Many of the initial inspection due dates in MRP-227, Revision 1-A, are based on the start of the period of extended operation. The licensee stated that the period of extended operation begins at midnight on December 6, 2024, for Catawba, Unit 1, and at midnight on February 24, 2026, for Catawba, Unit 2. The licensee explained that the refueling outages corresponding to inspection due dates that are based on EFPY are conservative estimates based on 18-month fuel cycles and are subject to change. The licensee stated that it will perform subsequent examinations after the initial examinations in accordance with MRP-227, Revision 1-A, as modified by any applicable interim guidance. The NRC staff finds acceptable that Catawba, Units 1 and 2, may inspect primary components earlier (initial examination) than the due dates as specified in Tables 7-1 and 7-2 of Catawba RVI AMP and inspection plan because the licensee needs to perform the examination during the specific refueling outage that occurs before the beginning of the period of the extended of operation to satisfy the examination due date.

The licensee stated that the examination schedule for expansion components will depend on the findings from the examinations of the primary components as discussed in Appendix C, Table C-2, of Catawba RVI AMP and inspection plan. Appendix C of Catawba RVI AMP and inspection plan also provides the examination acceptance criteria and expansion criteria for each of the RVI components and are consistent with MRP-227, Revision 1-A.

The licensee stated that should a change occur in plant operational practices or should operating experience result in changes to the projections, it will update on affected plant documentation in accordance with approved procedures.

The NRC staff finds that the Catawba RVI AMP and inspection plan is consistent with the specified inspection component, examination methods, and acceptance criteria of MRP-227, Revision 1-A; and is, therefore, acceptable.

3.9 On-Going Industry Programs and NEI 03-08 Guidelines

The NRC staff notes that Section 7 of MRP-227, Revision 1-A, discusses the implementation guidelines which follow NEI 03-08. Section 5 of Catawba RVI AMP and inspection plan discusses the implementation procedure based on NEI 03-08 guidance. As part of its license renewal application, the licensee stated that it would participate in industry activities associated with RVI-related issues and that the Catawba RVI AMP and inspection plan is subject to future enhancements as the industry's understanding of degradation continues to improve.

Several of the industry topical reports developed for the aging management of PWR internals are issued under the implementation protocol of NEI 03-08 and any interim guidance. Appendix B to NEI 03-08, Implementation Protocol, defines the processes and expectations for implementing industry guidance, and requires that the specific implementation category for requirements be identified by the topical reports. The licensee stated that its RVI AMP and inspection plan is based on the scope defined in MRP-227, Revision 1-A, and it will also implement the guidance in topical reports issued under the implementation protocol of NEI 03-08 in accordance with the licensee's administrative procedures, including later revisions and interim guidance. The licensee further stated that a failure to meet a Needed or a Mandatory requirement is a deviation from the NEI 03-08 guidelines and a written justification for the deviation must be prepared and approved as described in Appendix B to NEI 03-08 and Catawba administrative procedures, including notification to the NRC for information.

The NRC staff finds that the licensee is actively participating in the industry-wide effort in the aging management of the Catawba RVI components and has procedures to adopt the industry interim guidance as specified in the Catawba AMP and inspection plan. Specifically, the NRC staff finds that the licensee has: (1) administrative controls to monitor the aging of RVI components with respect to the industry guidance, and (2) procedures to report to the NRC if its aging management program deviates from industry guidelines. Therefore, the NRC staff finds that the Catawba RVI AMP and inspection plan is acceptable with respect to MRP-227, Revision 1-A.

4.0 CONCLUSION

The NRC staff finds that (1) the licensee has adequately addressed Action Item 1 in the NRC SE for MRP-227, Revision 1-A, and (2) the Catawba RVI AMP and inspection plan is consistent with the inspection and evaluation guidance of MRP-227, Revision 1-A, and is, therefore, acceptable. Based on the above, the NRC staff concludes that the licensee has demonstrated that the Catawba RVI AMP and inspection plan will adequately manage the aging effects and provide reasonable assurance structural integrity of the RVI components through the period of extended operation.

The NRC staff notes that: (1) it's approval of the Catawba RVI AMP and inspection plan does not reduce, alter, or otherwise affect the ASME Code, Section XI, ISI requirements, or any licensing basis requirements related to the ISI of RVI components at Catawba, Units 1 and 2, and (2) If the licensee wishes to use a new version of MRP-227, Revision 1-A, in the future, the new version of the topical report must have prior NRC review and approval.

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Date: October 5, 2021

SUBJECT: CATAWBA NUCLEAR STATION, UNITS 1 AND 2 – STAFF EVALUATION RELATED TO AGING MANAGEMENT PROGRAM AND INSPECTION PLAN OF REACTOR VESSEL INTERNALS PER MRP-227, REVISION 1-A (EPID L-2020-LLQ-0006) DATED OCTOBER 5, 2021

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