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

December 13, 2023

Mr. Steven M. Snider
Vice President, Oconee Nuclear Station
Duke Energy Carolinas, LLC
7800 Rochester Highway
Seneca, SC 29672-0752

SUBJECT: OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3 – RE: AUTHORIZATION OF ALTERNATIVE TO USE RR-22-0174, “RISK-INFORMED CATEGORIZATION AND TREATMENT FOR REPAIR/REPLACEMENT ACTIVITIES IN CLASS 2 AND 3 SYSTEMS SECTION XI, DIVISION 1” (EPID L-2022-LLR-0060)

Dear Mr. Snider:

By letter dated July 27, 2022, as supplemented by letters dated March 9 and October 20, 2023, Duke Energy Carolinas, LLC (Duke Energy, the licensee) requested authorization of proposed alternative RR-22-0174 to the requirements of American Society of Mechanical Engineers (ASME), Section XI, “*Rules for Inservice Inspection of Nuclear Power Plant Components*” for Oconee Nuclear Station (ONS), Units 1, 2, and 3, and Keowee Hydro Station, Units 1 and 2. Specifically, Duke Energy requested to use Code Case N-752, “*Risk-Informed Categorization and Treatment for Repair/Replacement Activities in Class 2 and 3 Systems Section XI, Division 1*,” for determining the risk-informed categorization and for implementing alternative treatment for repair/replacement activities on moderate and high energy Class 2 and 3 items in lieu of certain ASME Code Section XI, paragraph IWA-1000, IWA-4000, and IWA-6000 requirements.

Duke Energy submitted the request pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a “*Codes and Standards*,” on the basis that the proposed alternative would provide an acceptable level of quality and safety in accordance with 10 CFR 50.55a(z)(1).

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed RR-22-0174, and concludes, as set forth in the enclosed safety evaluation, that the licensee has adequately addressed the regulatory requirements set forth in 10 CFR 50.55a(z)(1). Therefore, the NRC staff authorizes the proposed alternative in relief request RA-22-0174 for the duration of the current renewed operating license for Oconee Units 1, 2, and 3. Applicability for Keowee Hydro Station Units 1 and 2 is associated with the renewed operating license of Oconee Units 1, 2, and 3.

Code Case N-752 has not been approved by the NRC or incorporated by reference for generic use. Therefore, the NRC reviewed the Duke Energy submittal as a plant-specific request for Oconee.

All other ASME OM Code requirements for which relief or an alternative was not specifically requested and approved remain applicable.

If you have any questions, please email Shawn.Williams@nrc.gov.

Sincerely,

Bo M. Pham Digitally signed by Bo M. Pham Date: 2023.12.13 21:45:10 -05'00'

Bo M. Pham, Director
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation Division of

Docket Nos. 50-269, 50-270, and 50-287

Enclosure:
Safety Evaluation

cc: Listserv

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

PROPOSED ALTERNATIVE REQUEST RA-22-0174

“RISK-INFORMED CATEGORIZATION AND TREATMENT FOR REPAIR/REPLACEMENT

ACTIVITIES IN CLASS 2 AND 3 SYSTEMS SECTION XI, DIVISION 1”

DUKE ENERGY CAROLINAS, LLC

OCONEE NUCLEAR STATION, UNIT NOS. 1, 2, AND 3

DOCKET NOS. 50-269, 50-270, AND 50-287

1.0 INTRODUCTION

By letter dated July 27, 2022, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML22208A031), as supplemented by letters dated March 9, 2023, (ML23068A015), and October 20, 2023 (ML23293A267), Duke Energy Carolinas, LLC (Duke Energy, the licensee) requested authorization of proposed alternative RR-22-0174 to the requirements of American Society of Mechanical Engineers (ASME), Section XI, “*Rules for Inservice Inspection of Nuclear Power Plant Components*” for Oconee Nuclear Station (ONS), Units 1, 2, and 3, and Keowee Hydro Station, Units 1 and 2. Specifically, Duke Energy requested to use Code Case N-752, “*Risk-Informed Categorization and Treatment for Repair/Replacement Activities in Class 2 and 3 Systems Section XI, Division 1*,” for determining the risk-informed categorization and for implementing alternative treatment for repair/replacement activities on moderate and high energy Class 2 and 3 items in lieu of certain ASME Code Section XI, paragraph IWA-1000, IWA-4000, and IWA-6000 requirements.

Duke Energy submitted the request pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a “*Codes and Standards*,” on the basis that the proposed alternative would provide an acceptable level of quality and safety in accordance with 10 CFR 50.55a(z)(1). Code Case N-752 has not been approved by the NRC or incorporated by reference for generic use. Therefore, the NRC reviewed the Duke Energy submittal as a plant-specific request for Oconee.

From August 1 to October 10, 2023, the NRC staff participated in a virtual regulatory audit. The NRC staff performed the audit to ascertain the information needed to support its review of the application and develop requests for additional information (RAIs), as needed. On October 10, 2023 (ML23219A140), the NRC staff issued an audit summary report.

2.0 REGULATORY EVALUATION

2.1 Regulations

The regulations in 10 CFR 50.55a(g)(4), "*Inservice inspection standards requirement for operating plants*," state, in part, that ASME Code Class 1, 2, and 3 components must meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in the Section XI of additions and addenda of the ASME BPV Code and that are incorporated by reference.

The regulations in Section 50.55a(z), "*Alternatives to codes and standards requirements*," of 10 CFR state, in part, that alternatives to the requirements of 10 CFR 50.55a(b) through (h) of this section or portions thereof may be used, when authorized by the Director, Office of Nuclear Reactor Regulation. An alternative must be submitted and authorized prior to implementation. The licensee must demonstrate that: (1) the proposed alternative would provide an acceptable level of quality and safety, or (2) compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

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

The regulation in 10 CFR 50.69, "*Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors*."

The regulations in 10 CFR 50.54(a)(3), "*Conditions of licenses*," states, in part, that each licensee described in paragraph (a)(1) of this section may make a change to a previously accepted quality assurance program description included or referenced in the Safety Analysis Report without prior NRC approval, provided the change does not reduce the commitments in the program description as accepted by the NRC. Changes to the quality assurance program description that do not reduce the commitments must be submitted to the NRC in accordance with the requirements of Sec. 50.71(e). In addition to quality assurance program changes involving administrative improvements and clarifications, spelling corrections, punctuation, or editorial items, the following changes are not considered to be reductions in commitment:

… (ii) The use of a quality assurance alternative or exception approved by an NRC safety evaluation, provided that the bases of the NRC approval are applicable to the licensee's facility; …

2.2 Regulatory Guidance

Regulatory Guide 1.178, Revision 2, "*Plant-Specific, Risk-Informed Decisionmaking for Inservice Inspections of Piping,*" April 2021 (ML21036A105).

Regulatory Guide 1.174, Revision 3, "*An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis*," January 2018 (ML17317A256).

Regulatory Guide 1.177, Revision 2, "*Plant-Specific Risk-Informed Decisionmaking: Technical Specifications*," January 2021 (ML20164A034).

3.0 TECHNICAL EVALUATION

3.1 Applicable Code Edition and Addenda

The current edition for the Inservice Inspection (ISI) interval for Oconee Nuclear Station (ONS) Units 1, 2, and 3 and Keowee Hydro Station, Units 1 and 2, is the American Society of Mechanical Engineers (ASME) Code, Section XI, 2007 Edition with the 2008 Addenda. All units are in the fifth inspection interval, which started on July 15, 2014, and is scheduled to end on July 15, 2024. In its letter dated July 27, 2022, the licensee states that use of subparagraphs IWA-4540(b) and IWA-4340 of the 2017 Edition of the Section XI of the ASME BPV Code is acceptable for Oconee.

3.2 ASME Code Components Affected

As stated in the application dated July 27, 2022:

> This request applies to ASME Class 2 and 3 items or components except the following:
>
> 1. Piping within the break exclusion region [> Nominal Pipe Size (NPS) 4 (DN 100)] for high energy piping systems[1] as defined by the Owner.
> 2. That portion of the Class 2 feedwater system [> NPS 4 (DN 100)] of pressurized water reactors (PWRs) from the steam generator (SG), including the SG, to the outer containment isolation valve.

This request does not apply to Class CC[1] and MC[2] items.

3.3 Applicable Code Requirements

ASME Code, Section XI, Subsection IWA provides the requirements for repair/replacement activities including the following:

- IWA-1320 specifies group classification criteria for applying the rules of ASME Section XI to various Code Classes of components. For example, the rules in IWC apply to items classified as ASME Class 2 and the rules in IWD apply to items classified as ASME Class 3.
- IWA-1400(f) requires Owners to possess or obtain an arrangement with an Authorized Inspection Agency (AIA).
- IWA-1400(j) requires Owners to perform repair/replacement activities in accordance with written programs and plans.
- IWA-1400(n) requires Owners to maintain documentation of a Quality Assurance Program in accordance with 10 CFR 50 or ASME NQA-1, Parts II and III.
- IWA-4000 specifies requirements for performing ASME Section XI repair/replacement activities on pressure-retaining items or their supports.
- IWA-6210(d) and (e), specify Owner reporting responsibilities such as preparing Form NIS-2, Owner's Report for Repair/Replacement Activity.

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[1] Class CC items are concrete containment items for which the requirements are in ASME Code, Subsection IWL of Section XI defined by Section III, Division 2, Article CC-1000.
[2] Class MC items are metal containment or liners of concrete containments for which the requirements are in ASME Code, Subsection IWE of Section XI described in Section III, Subsection NE, Article NE-1110.

- IWA-6350 specifies that the following ASME Section XI repair/replacement activity records must be retained by the Owner: evaluations required by IWA- 4160 and IWA-4311, Repair/Replacement Programs and Plans, reconciliation documentation, and NIS-2 Forms.

3.4 Proposed Alternative

Duke Energy proposes to use Code Case N-752 as an alternative for the ASME Code requirements specified in Section 3 of its submittal. The licensee states, in part, that Code Case N-752 provides a process for determining the risk-informed categorization and treatment requirements for Class 2 and 3 pressure-retaining items or the associated supports and that the process may be applied on a system basis or on individual items within selected systems. In its letter dated July 27, 2022, the licensee states, in part, that,

> Code Case N-752 categorization methodology relies on the conditional core damage and large early release probabilities associated with postulated ruptures. Safety significance is generally measured by the frequency and the consequence of the event. However, the risk-informed process categorizes components solely based on consequence, which measures the safety significance of the component given that it ruptures (component failure is assumed with a probability of 1.0). This approach is conservative compared to including the rupture frequency in the categorization as this approach will not allow the categorization of SSCs [structures, systems, and components] to be affected by any changes in frequency due to changes in treatment. It additionally applies deterministic considerations (e.g., defense in depth, safety margins) in determining safety significance. Additional detail is provided Section 5.2.
>
> The risk-informed process categorizes components as either high safety-significant (HSS) or LSS (low safety significance). HSS components must continue to meet ASME Section XI rules for repair/replacement activities. LSS components are exempt from ASME Section XI repair/replacement requirements and can be repaired/replaced in accordance with treatment requirements established by the Owner. The treatment requirements must provide reasonable confidence that each LSS item remains capable of performing its safety-related functions under design basis conditions. Component supports, if categorized, are assigned the same safety significance, HSS or LSS, as the highest passively ranked segment within the bounds of the associated analytical pipe stress model. The categorization and treatment requirements of Code Case N-752 are consistent with those in 10 CFR 50.69.
>
> It should be noted that Code Case N-752 is based on ANO-2 relief request ANO2 R&R-004, Revision 1, dated April 17, 2007 (Reference 8.8 – ML071150108), as supplemented by Entergy. The NRC approved relief request ANO2-R&R-004, Revision 1, in a safety evaluation dated April 22, 2009 (Reference 8.9 – ML090930246). The ANO-2 relief request was developed to serve as an industry pilot for implementing a risk-informed repair/replacement process that included a risk-informed categorization process and treatment requirements.

Duke Energy is not requesting NRC approval to implement 10 CFR 50.69 in this relief request. This process would not apply to Class 1 items and systems. The process requires the Owner to define alternative treatment requirements and confirm with reasonable confidence that each

LSS item remains capable of performing its safety-related function. These treatment requirements must cover items such as design control, procurement, installation, configuration control, and corrective actions.

The NRC staff authorized the ANO licensee to utilize Request for Alternative ANO2-R&R-004, Revision 1 (ML071150108) for determining the risk-informed categorization and for implementing alternative treatment for repair/replacement activities on moderate and high energy Class 2 and 3 items at ANO, Unit 2. By letter dated April 22, 2009 (ML090930246), the NRC staff authorized the alternative.

Duke states the proposed alternative is based on similar request from ANO. By letter dated May 27, 2020, Entergy submitted alternative request EN-20-RR-001 to NRC for ANO to use Code Case N-752 (ML20148M343). By letter dated May 19, 2021, the NRC authorized the plant-specific methodology in EN-20-RR-001 for ANO (ML21118B039).

3.5 NRC Staff Evaluation

The NRC evaluated the licensee's submittal, as supplemented, to determine if the proposed alternative met an acceptable level of quality and safety, as required by the regulations, and described in Section 2.0 of this safety evaluation (SE).

3.5.1 PRA Technical Acceptability

The proposed plant-specific approach for Oconee takes advantage of the ANO precedents and utilizes the risk-informed categorization process in Appendix I of Code Case N-752 for ASME Class 2 and 3 systems. The process requires confirmation of the technical adequacy of the probabilistic risk assessment (PRA) for its risk-informed inservice inspection (RI-ISI) to confirm the applicability to categorization, including verification of assumptions on equipment reliability. The alternative authorized for ANO2-R&R-004, Revision 1 for ANO, Unit 2 (ML071150108), demonstrated adequate PRA technical requirements, as outlined in the NRC staff's safety evaluation dated April 22, 2009 (ML090930246) and has been used by numerous nuclear power plants for the risk-informed categorization and treatment of Class 2 and 3 systems.

The NRC staff's review of the ONS PRA was based on staff's previous determinations that the PRA model was found acceptable to support issuance of Oconee Nuclear Station, Units 1, 2, and 3, Issuance of Amendments Regarding Adoption of Technical Specification Task Force (TSTF) -425, Revision 3, "*Relocate Surveillance Frequencies to Licensee Control Risk Informed Technical Specification Task Force (RITSTF) Initiative 5b*," dated March 21, 2011 (ML110470446), and License Amendment "*Oconee Nuclear Station, Units 1, 2, and 3, Issuance of Amendments Regarding Transition to a Risk-Informed, Performance-Based Fire Protection Program in Accordance with 10 CFR 50.48(c)*," dated December 29, 2010 (ML103630612). The model has had routine PRA maintenance updates applied. In addition, all findings for the internal events model were reviewed and closed in February 2019 using the process documented in Appendix X to Nuclear Energy Institute (NEI) 05-04, "*Close-out of Facts and Observations*," as accepted by the NRC.

In its letter dated July 27, 2022, the licensee stated, in part, that,

> The ONS Code Case N-752 categorization process for the internal events and flooding hazard uses the plant-specific PRA model.

> The Duke Energy risk management process ensures that the PRA model used in this application reflects the as-built and as-operated plant for each of the ONS units.
>
> The PRA models described above have been assessed against RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2.
>
> Duke Energy shall review changes to the plant, operational practices, applicable plant, and industry operational experience, and, as appropriate, update the PRA and categorization and treatment processes. Duke Energy shall perform this review in a timely manner but no longer than once every two refueling outages. This approach is consistent with the feedback and adjustment process of 10 CFR 50.69(e).

Although the passive methodology proposed in RR-22-0174 is similar to that used in the risk informed inservice inspection (RI-ISI) program, the licensee confirmed that it will continue to review and assess the existing PRAs to verify that they support the evaluations required by the proposed alternative as part of its program to maintain a feedback and process adjustment process consistent with that of 10 CFR 50.69(e) to update the PRA, categorization, and treatment processes based on review of changes to the plant, operational practices and applicable plant and industry operational experiences. Although Oconee does not have an approved 10 CFR 50.69 program, the NRC finds this approach for PRA technical adequacy, feedback and process adjustment to be acceptable.

<u>Active Function Evaluation</u>

In its response to RAI No. 4b. in letter dated March 9, 2023, the licensee states that for pressure retaining components that have a passive function as well as an active function, the proposed alternative categorization process only applies to the pressure boundary function of these components and no treatment changes will be applied to the active function as a result of implementing the proposed alternative. In its Audit response to RAI No. 1 in letter dated October 20, 2023, the licensee notes that the consequence evaluation methodology of the proposed alternative must address not only the postulated failure of the subject pressure boundary component (e.g., loss of a flow path) but also other direct and indirect effects (e.g., due to spray or flooding) associated with the pressure boundary failure. The conditional core damage probability (CCDP) analysis incorporates the impact of both direct and indirect failures including any effects on the active function. Therefore, while treatment requirements for the active portion of the pressure retaining components are not within the scope of the proposed alternative, the assessment of the impact to the active function is required by the proposed plant-specific methodology.

In its Audit response to RAI 1, Item 2 in letter dated October 20, 2023, the licensee notes that the proposed categorization methodology is the consequence evaluation portion of Electric Power Research Institute (EPRI) TR-112657 Revision B-A, "*Revised Risk-Informed Inservice Inspection Procedure*" (ML013470102), which is the foundational methodology for several risk-informed applications related to structures, systems, and components (SSCs) that perform pressure boundary functions. These applications include ASME Code Case 660, "*Risk-Informed Safety Classification for Use in Risk-Informed Repair/Replacement Activities Section XI, Division I*," RI-ISI programs, and ANO-R&R-004, Revision 1. Relative risk measures such as Fussell-Vesely (F-V) and Risk Reduction Worth (RRW) are not applied for these applications, in

part, because passive components and pressure retaining portion of active components typically have very low failure rates/probabilities; and common cause failures probabilities are also very low and would reach orders of magnitude below the truncation levels of the PRA. As such, using relative importance measures such as F-V and RRW identifies the vast majority of pressure boundary components and pressure retaining functions of active components as low safety significant. The F-V and RRW importance measures are often used for the selection of candidates for improvement and enhanced maintenance, whereas the Conditional Core Damage Probability (CCDP) criteria, applied in Code Case N-752, and thus, Oconee plant-specific request RR-22-0174, is useful for identifying components that should be prevented from failing using repair/replacement, planned maintenance, and other treatment requirements.
In its response to RAI No. 2 in letter dated October 20, 2023, the licensee notes that,

> Section 1420, item c: Changes in configuration, design, materials, fabrication, examination, and pressure-testing requirements used in the repair/replacement activity shall be evaluated, as applicable, to ensure the structural integrity and leak tightness of the system are sufficient to support the design bases functional requirements of the system.

These requirements, in addition to those outlined in the relief request as explained in this Safety Evaluation, provide reasonable confidence that passive components and pressure retaining functions of active components will continue to perform their design-basis function, and, therefore, would not impact the basis for not using F-V.

<u>Risk Tables</u>

The proposed alternative references Code Case N-752 Section I-3.3.2 which allows for the use of risk tables as identified in Table I-1, I-2, I-3 and I-4 in lieu of CCDP or Conditional Large Early Release Probability (CLERP). In its letter dated October 20, 2023, the licensee notes that the same methodology is allowed in Code Case N-660 as endorsed in Regulatory Guide 1.147 (ML21181A222). This methodology was also approved for ANO-R&R-004, Revision 1. The licensee also notes that improvements in PRA technical adequacy supports the use of CCDP and CLERP directly; however, the risk tables will continue to be used as a comparison to the results obtained from the PRA. As explained in both ANO-R&R-004, Revision 1, and Oconee RR-22-0174 proposed plant-specific use of Code Case N-752, differences in consequence rank between the use of risk tables and quantitative indices shall be reviewed, justified, and documented or the higher consequence rank assigned.

<u>Review of Key Principles</u>

NRC staff evaluated the application to the RG 1.174 Key Principles. These key principles are:

Principle 1: The proposed licensing basis change meets the current regulations unless it is explicitly related to a requested exemption.

Principle 2: The proposed licensing basis change is consistent with the defense-in-depth [DID] philosophy.

Principle 3: The proposed licensing basis change maintains sufficient safety margins.

Principle 4: When the proposed licensing basis change results in an increase in risk, the increase should be small and consistent with the intent of the Commission's policy statement on safety goals for the operations of nuclear power plants.

Principle 5: The impact of the proposed licensing basis change should be monitored by using performance measures strategies.

Key Principle 1:

NRC staff finds that key principle 1 is not applicable to an alternative requested submitted in accordance with 10 CFR 50.55a(z)(1).

Key Principle 2:

In supplement dated October 20, 2023, in response to RAI No. 1, the licensee addressed how the proposed alternative is consistent with the defense-in-depth philosophy.

The licensee described the use of the "Consequence Evaluation" methodology contained in Code Case N-752 and the use of the PRA model crediting only the unaffected equipment to obtain Conditional Core Damage Probability (CCDP) and Conditional Large Early Release Probability (CLERP) values.

The licensee stated:

> In the ASME Code Case N-752 categorization methodology, the failure of the component's pressure boundary function is assumed with a probability of 1.0 and only the consequence evaluation is performed. The consequence evaluation is coupled with additional deterministic considerations (e.g., DID, safety margins) required by ASME Code Case N-752 Section I-3.4.2 in determining safety significance.

The licensee discussed the defense-in-depth applied to the 10 CFR 50.69 methodology. The licensee reasoned that because the Code Case N-752 categorization process is identical to the 10 CFR 50.69 methodology for pressure boundary components, it captures identical system impacts and results in the same conclusion for passive/pressure boundary components, thereby, the same defense-in-depth philosophy is inherent to Code Case N-752.

Based on the information provided in the application and above, the NRC staff concludes that the proposed change is consistent with defense-in-depth philosophy.

Key Principle 3:

In supplement dated October 20, 2023, in response to RAI No. 3, the licensee discussed how the proposed alternative maintains sufficient safety margins.

The licensee stated:

> Safety margins for LSS items, in accordance with ASME Code Case N-752, are maintained by the alternative treatment process ensuring, with reasonable

> confidence, the items remain capable of performing its safety-related function under design-basis conditions along with the other programs in place at Oconee to monitor, inspection, operate, and maintain installed equipment.

The licensee also stated:

> The 10 CFR 50.69 and ASME Code Case N-752 categorization processes use a 1.0 failure probability which provides a conservative risk-informed categorization result and also minimizes uncertainty. The categorization process assumes failure in all cases which shows, regardless of what treatment code or standard is used, the categorization process bounds the change in failure frequency arising from changes in treatment. Nevertheless, as stated above, 10 CFR 50.69 and ASME Code Case N-752 (e.g. -1420) requires that the Owner ensure with reasonable confidence that each LSS item remains capable of performing its safety-related functions under design-basis conditions.

The licensee further stated:

> Other programs and process remain in place such as design control, 10 CFR 50.59 change control process, supply chain / procurement processes, corrective action / problem identification and resolution, testing and monitoring programs (e.g. RI-ISI, IST, License Renewal Aging Management, Flow Accelerated Corrosion, Erosion, Raw Water Program, Buried Pipe Program, etc.), and Technical Specifications (including surveillances). These programs allow the licensee to monitor the condition of components, identify degradation, and correct the degradation in a timely manner.

Based on the information provided in the application and above, the NRC staff concludes that the proposed change maintains sufficient safety margins.

Key Principle 4:

In the licensee's supplement dated October 20, 2023, in response to RAI No. 1, the licensee discussed how the proposed alternative is consistent with the intent of the Commission's Policy Goal statement that result in small changes to core damage frequency or risk.

The licensee stated:

> The passive categorization process is driven by the consequence of failure in that the process conservatively assumes that a failure occurs with a probability of 1.0. As such, some postulated passive failures will be categorized as HSS while, from a pure risk perspective, they may be low safety significant. As an example, postulated failures with conditional core damage probability (CCDP) values of 5 E-04 are HSS per the passive categorization process. However, many passive components have failure frequencies of 1E-08 and lower. Thus, if failure frequency were to be considered, they may be shown quantitatively to be low safety significant.

The staff notes that the proposed changes in treatments is not expected to result in significant changes to existing low failure frequencies and there is reasonable confidence that the affected SSCs would retain the capability and reliability of the design basis function, as discussed in

Section 3.5.2. Therefore, the staff concludes that the proposed change would result in at most small changes to core damage frequency or risk in accordance with the Commission's Policy Goal statement.

Key Principle 5:

In the licensee's submittal dated July 27, 2022, the licensee described how the impact of the proposed changes would be monitored using performance management strategies.

The licensee stated:

> Duke Energy shall review changes to the plant, operational practices, applicable plant, and industry operational experience, and, as appropriate, update the PRA and categorization and treatment processes. Duke Energy shall perform this review in a timely manner but no longer than once every two refueling outages.

The licensee also stated:

> Baseline examination (e.g., preservice examination) of the items affected by the repair/replacement activity, if required, shall be performed in accordance with requirements of the applicable program(s) specifying periodic inspection of items.

The licensee further stated:

> Conditions that would prevent an LSS item from performing its safety related function(s) under design basis conditions will be corrected in a timely manner. For significant conditions adverse to quality, measures will be taken to provide reasonable confidence that the cause of the condition is determined, and corrective action taken to preclude repetition. Corrective action of adverse conditions associated with LSS items will be identified and addressed in accordance with Duke Energy's existing corrective action program.

To gain additional risk insights, the NRC staff performed an independent site-specific assessment using the Oconee Standardized Plant Analysis Risk (SPAR) model. The results conclude the risk associated with utilizing the Code Case N-752 categorization methodology is consistent with RG 1.174 and RG 1.177 risk acceptance guidelines.

Based on the information provided in the application and above, the NRC staff concludes that the proposed changes provide reasonable confidence that LSS items would be monitored appropriately using performance management strategies.

<u>Risk Conclusion</u>

Based on the above, the NRC staff finds, with reasonable assurance, that the ONS PRAs reflects the as-built, as-operated plants to support the safety significance categorization of RR-22-0174, and that the feedback and process adjustments will provide reasonable confidence that the PRA will be maintained in a manner to support the categorization and treatment for the repair/replacement of Class 2 and 3 items. In addition, the NRC staff finds the application to be consistent with RG 1.174 Key Principles.

3.5.2 Alternative Code and Standards Acceptability

Alternative Treatment

In evaluating the licensee's alternative treatment requirements of RR-22-0174, the NRC staff considered the past precedent of previous NRC approved methods relating to risk-informed treatment of structures, systems, and components (SSCs) for nuclear power plants. As noted in the licensee's submittal, these include previous NRC approval of the use of Arkansas Nuclear One precedents and 10 CFR 50.69, "*Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors*." While the licensee has not requested to implement 10 CFR 50.69 at ONS Units 1, 2, and 3, the licensee specified that the treatment requirements in its proposed alternative, which relies on the ANO precedents and plant-specific applicability of Code Case N-752, are consistent with scope of the requirements for similar Low Safety Significance (LSS) SSCs listed in 10 CFR 50.69(b)(1). While Code Case N-752 has not been approved by the NRC or incorporated by reference for generic use, the NRC staff finds that it has some applicable treatment for plant-specific evaluation and use.

Codes and Standards Alternative

The NRC staff's review of the specific alternative codes and standards identified potential areas of uncertainty in assessing the quality of the proposed alternate treatment. In Section 5.2.E of Basis for Use in the letter dated July 27, 2022, the licensee listed the alternative treatments related to Paragraph 1420 of Code Case N-752. The NRC staff found Sections 5.2.E.1 through 5.2.E.10 are equivalent to Subparagraphs 1420(a) through (j) of Code Case N-752. The NRC staff again notes that the Code Case N-752 has not been approved by the NRC or incorporated by reference for generic use, therefore, the NRC staff's review focused on the plant-specific regulatory and technical evaluation.

In reviewing the licensee's plant-specific alternative treatment wording of Section 5.2.E of its letter dated July 27, 2022, the NRC staff evaluated the alternative requirements in lieu of current regulatory requirements for codes and standards. The NRC staff recognizes that the general basis for the proposed alternative's approach was to replace the requirements of Section XI of the ASME BPV Code with requirements from the original Construction Code, Owner's Requirements, and nationally recognized codes, standards, or specifications applicable to the LSS categorized item as permitted by the licensing basis. However, the NRC's independent review determined that the specific language of the following Sections (discussed below) may allow for the use of Owner's options in lieu of specific codes and standards, which the staff identified as a potential concern to the quality of the licensee's proposed alternative treatment.

Section 5.2.E.6 of the licensee's submittal states the following:

> The repair methods of nationally recognized post-construction codes and standards (e.g., PCC-2, API-653) applicable to the item *may* [emphasis added] be used.

The NRC staff notes that the language would allow the Owner to choose whether to apply a nationally recognized post-construction code or standard applicable to the item, raising concerns about the quality of the alternative treatment.

In its letter dated October 20, 2023, the licensee provided additional basis for the use of the "may" in its Basis for Use and stated:

> This paragraph [5.2.E.6] allows the use of nationally recognized post-construction codes and standards applicable to the item. The word ’shall’ is not used because ’shall’ would establish that these nationally recognized post-construction codes and standards must be used. The use of the word ‘may’ retains the flexibility to use Section XI code, if desired or if there is not a nationally recognized post-construction codes or standard applicable to the item.

The NRC staff considers Section XI of the ASME BPV Code to be a nationally recognized post-construction code, therefore, it’s use would still be allowed whether the term “may” or “shall” was used. If there is not a nationally recognized post-construction code or standard applicable to the item, then the default would be expected to utilize the original Construction Code, as the original Construction Code is what the licensee’s license allows for operation with this repaired/replaced safety item. Additionally, post-construction codes have similar citations, such as Section 101-3.11, “*Examination*” on page 3 of ASME PCC-2.

The versatility allowed by alternative post-construction codes to the current regulatory requirement of Section XI provides significant flexibility, but there remains a requirement for use of an established code. In Section 5.2.E.7 of the letter dated July 27, 2022, the licensee states, in part, that,

> Performance of repair/replacement activities, and associated NDE, shall be in accordance with the Owner's Requirements and, *as applicable* [emphasis added], the Construction Code, or post-construction code or standard, selected for the repair/replacement activity. Alternative examination methods may be used *as approved by the Owner*. [emphasis added].

The NRC staff finds that this language could be read to only require the licensee to perform repair/replacement activities by the Owner’s Requirements with the option to follow the Construction Code or post-construction code or standard as deemed applicable by the Owner for the selected repair/replacement activity. Further, even if a code or standard is deemed applicable and chosen by the Owner for the selected repair/replacement activity, the allowance of alternative examination methods as approved by the Owner can significantly change the level of quality and safety relative to following a nationally approved code or standard. For example, a licensee may choose to substitute a volumetric examination with a visual examination that could be performed by a plant walkdown with insulation in place. If a nationally approved code or standard for construction or post-construction clearly identifies an examination method for a repair/replacement activity, the option for the Owner to change that method raises potentially significant uncertainty to the level of quality for the performance of a repair/replacement activity, if not exercised judiciously. Owner’s Requirements, as defined in the 2007 Edition with 2008 Addenda of Section XI of the ASME BPV Code, are defined as those requirements when a Construction Code is not specified, address plant-specific requirements of the Construction Code, or invoke plant-specific requirements that are in excess of Construction Code requirements. The allowances of Sections 5.2.E.6 and 5.2.E.7 could potentially allow the Owner to determine what codes and standards could be applied and then change specific provisions without adequate justification.

In its letter dated October 20, 2023, the licensee provided additional rationale to support the basis and intent of the language in the various parts of Section 5.2.E, to provide additional assurance that its program does “*allow for alternatives to the code or standard, but not wholesale use of Owner’s Requirements to substitute for code requirements*.” While the NRC staff believes that a clearly defined code or standard is preferable for the predictability and clarity of the alternate treatment to be implemented, the NRC concludes that the licensee's intent in RR-22-0174 is to leverage flexibility in treatment alternatives, specifically for LSS components, through a methodology based on the NRC approved ANO2 R&R-004 precedent and plant-specific evaluation for ONS. Because the proposed alternate treatment is limited to LSS component, with controls in place as described by the licensee's supplemental audit responses, the NRC staff finds that the code and standard, as described, provides an acceptable level of quality and safety.

3.5.3 Quality Assurance

The proposed alternative would allow LSS items to be exempt from ASME Code, Section XI, IWA-1400(n), which requires the licensee to document repair and replacement activities via a Quality Assurance Program in accordance with 10 CFR Part 50 Appendix B or ASME NQA-1. The licensee's submittal cites footnote (1) in Code Case N-752, which states, "*If compliance with 10 CFR 50 Appendix B or NQA-1 is required at the Owner's facility, IWA-1400(o) is not exempt*" (NRC staff notes that the reference of IWA-1400(o) vs. IWA-1400(n) is due to different edition and addenda of the ASME Code, but that the content is the same). For clarity, while the term "exempt" is used in the cited footnote, the proposed alternative does not exempt the LSS components from Appendix B requirements, as any exemption from an NRC regulatory requirement in 10 CFR Part 50 would need to be requested and considered under 10 CFR 50.12 or other more specific provisions, as appropriate. However, the proposed alternative allows for altering the treatment of those LSS components under the provisions of Appendix B.

Duke Energy stated its intent to update the fleet's Quality Assurance Program Description (QAPD) for safety-related Class 2 and 3 SSCs identified as LSS in accordance with Code Case N-752 to not be required to meet the requirements of Duke Energy's QAPD. The treatments are specified in the July 27, 2022, submittal and clarified in the supplements. The March 9, 2023, supplement includes a draft QAPD update which states that Duke Energy plans to use current QAPD processes and procedures with additional controls for the treatment of Class 2 and 3 LSS SCCs to ensure continued capability and reliability of the design-basis function. In accordance with 10 CFR 50.54(a), when the use of a quality assurance exception is approved by an NRC safety evaluation, licensees may make changes to a previously accepted QAPD without prior NRC approval provided the bases of the approval are applicable to the licensee's facility. In this case, the NRC already previously approved similar quality assurance program manual (QAPM) change for Entergy Operations, Inc. (Entergy), submitted under auspices of 10 CFR 50.54(a)(4), in a safety evaluation (SE) dated May 19, 2021 (ML21132A279). In its submittal dated October 26, 2020 (ML20300A324), Entergy proposed changes to its QAPM which would allow sites that have been authorized to utilize Code Case N-752 to use the alternative repair/replacement categorization and treatment requirements of Code Case N-752 in lieu of the corresponding sections of ASME Section XI. Further, treatment of safety-related SSCs (identified as LSS) Class 2 and 3 SSCs in accordance with Code Case N-752 are not required to meet the requirements of the QAPM. Instead, Entergy would develop program elements describing treatment of these LSS SSCs to ensure continued capability and reliability of the design basis function. The procedures governing these treatment activities are classified as safety-related and therefore, under the jurisdiction of 10 CFR 50, Appendix B. The staff reviewed the proposed change to the Entergy QAPM and concluded that the proposed alternative, as described above, still met the requirements of Appendix B, which includes the treatment requirements of Code Case N-752 and the additional safety-related procedures developed to address program elements of the treatment requirements of the LSS.

The NRC staff confirmed that the changes to the QAPD proposed by Duke Energy are consistent with the changes approved by the NRC staff to Entergy's QAPM as documented in the SE dated May 19, 2021, therefore, it is not considered a reduction in commitment in accordance with 10 CFR 50.54(a)(3)(ii).

3.6 NRC Staff Conclusion

Based on information provided, the NRC staff finds that: (1) the proposed risk categorization methodology will satisfactorily classify the affected Class 2 and 3 components as HSS or LSS, (2) the alternate treatment requirements in the proposed alternative will provide reasonable assurance that each LSS item remains capable of performing its safety-related function, (3) the current risk informed ISI program will continue, (4) the licensee's corrective action program will continue to provide actions to correct conditions that could prevent an LSS item from performing its safety function, (5) the feedback and process adjustment will allow timely update of the elements of this program, (6) the licensee's PRA has sufficient technical quality to support this application, and (7) the repair/replacement program quality elements will provide reasonable assurance that the LSS items remain capable of performing their design safety function. Therefore, the NRC staff finds that the proposed alternative will provide an acceptable level of quality and safety.

4.0 CONCLUSION

The NRC staff has determined that the proposed alternative in the licensee's request referenced above would provide an acceptable level of quality and safety.

The NRC staff concludes that the licensee has adequately addressed the regulatory requirements set forth in 10 CFR 50.55a(z)(1).

The NRC staff authorizes the use of proposed alternative RA-22-0174 at Oconee Nuclear Station, Unit 1, 2, and 3 for the remainder of the current renewed operating licenses for Oconee Units 1, 2, and 3, as shown below. Applicability for Keowee Hydro Station, Units 1 and 2, is associated with the renewed operating license of Oconee Units 1, 2, and 3.

	Docket Number	License Expires
Unit 1	05000269	02/06/2033
Unit 2	05000270	10/06/2033
Unit 3	05000287	07/19/2034

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

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Date: December 13, 2023

SUBJECT: OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3 – RE: AUTHORIZATION OF ALTERNATIVE TO USE RR-22-0174, “RISK-INFORMED CATEGORIZATION AND TREATMENT FOR REPAIR/REPLACEMENT ACTIVITIES IN CLASS 2 AND 3 SYSTEMS SECTION XI, DIVISION 1” (EPID L-2022-LLR-0060) DATED DECEMBER 13, 2023

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