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

June 12, 2024

Bob Coffey
Executive Vice President, Nuclear
and Chief Nuclear Officer
Florida Power & Light Company
NextEra Energy Seabrook, LLC
Mail Stop: EX/JB
700 Universe Blvd.
Juno Beach, FL 33408

SUBJECT: SAINT LUCIE NUCLEAR PLANT, UNITS 1 AND 2; TURKEY POINT NUCLEAR PLANT, UNITS 3 AND 4; SEABROOK NUCLEAR PLANT; AND POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2 - PROPOSED ALTERNATIVE FRR 23-01 TO USE ASME CODE CASE N-752-1, "RISK-INFORMED CATEGORIZATION AND TREATMENT FOR REPAIR/REPLACEMENT ACTIVITIES IN CLASS 2 AND 3 SYSTEMS SECTION X1, DIVISION 1" (EPID L-2023-LLR-0009)

Dear Bob Coffey:

By letter dated March 15, 2023 (Agencywide Documents Access and Management System Accession No. ML23074A155), NextEra Energy Point Beach, LLC/NextEra Energy Seabrook, LLC/Florida Power & Light Company (NextEra, the licensee) submitted a request to the U.S. Nuclear Regulatory Commission (NRC) for the use of an alternative to certain American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI requirements at St. Lucie Nuclear Plant, Units 1 and 2 (St. Lucie); Turkey Point Nuclear Plant, Units 3 and 4 (Turkey Point); Seabrook Nuclear Plant (Seabrook); and Point Beach Nuclear Plant, Units 1 and 2 (Point Beach). Specifically, the licensee requested to use Code Case N-752-1, "Risk-Informed Categorization and Treatment for Repair/Replacement Activities in Class 2 and 3 Systems Section X1, Division 1" in lieu of certain requirements in ASME Section XI, Sub-Paragraphs IWA-1320, IWA-1400, IWA-4000, IWA-6210, IWA-6211, IWA-6212, IWA-6220, and IWA-6350.

Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(1), the licensee requested to use the proposed alternative on the basis that the alternative provides an acceptable level of quality and safety. ASME Code Case N-752-1 has not been approved by the NRC staff or incorporated by reference for generic use. Therefore, the NRC staff reviewed the licensee's request as plant-specific requests for Turkey Point, St. Lucie, Point Beach, and Seabrook.

The NRC staff has reviewed the proposed alternative, and, as set forth in the enclosed safety evaluation, the NRC staff determines that the proposed alternative provides an acceptable level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1). Therefore, the

NRC staff authorizes the proposed alternative for the remainder of the fifth and the entirety of the sixth inservice inspection intervals at Turkey Point, the remainder of the fifth inservice inspection interval at St. Lucie, Unit 1, the remainder of the fourth and the entirety of the fifth inservice inspection intervals at St. Lucie, Unit 2, the remainder of the sixth inservice inspection intervals at Point Beach, and the remainder of the fourth inservice inspection interval at Seabrook.

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.

In its request, the licensee mentioned in several places that risk-informed categorization process described in Code Case N-752-1 is "consistent" with the risk-informed categorization process described in 10 CFR 50.69. For the licensee's facilities that have been approved to use a risk-informed categorization process under 10 CFR 50.69, this authorization does not change any of the licensee's obligations with regards to the 10 CFR 50.59 risk-informed categorization process for structures, systems, and components.

If you have any questions, please contact the NextEra Fleet Project Manager Michael Marshall at 301-415-2871 or michael.marshall@nrc.gov.

Sincerely,

JAMIE PELTON

Digitally signed by JAMIE PELTON
Date: 2024.06.12 15:58:54 -04'00'

Jamie Pelton, Deputy Director
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-335, 50-389, 50-250,
50-251, 50-443, 50-266, 50-301

Enclosure:
Safety Evaluation

cc: ListServ

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

PROPOSED ALTERNATIVE FRR 23-01 TO USE ASME CODE CASE N-752-1,

"RISK INFORMED CATEGORIZATION AND TREATMENT FOR REPAIR/REPLACEMENT

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

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

SEABROOK STATION, UNIT NO. 1

ST. LUCIE PLANT, UNIT NOS. 1 AND 2

TURKEY POINT NUCLEAR GENERATING UNIT NOS. 3 AND 4

NEXTERA ENERGY RESOURCES AND FLORIDA POWER & LIGHT COMPANY, ET AL.

DOCKET NOS. 50-250, 50-251, 50-266, 50-301, 50-335, 50-389, AND 50-443

1.0 INTRODUCTION

By letter dated March 15, 2023 (Agencywide Documents Access and Management System Accession No. ML23074A155), NextEra Energy Point Beach, LLC/NextEra Energy Seabrook, LLC/Florida Power & Light Company (NextEra, the licensee) requested the use of an alternative to certain requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code to use Code Case N-752-1, "Risk-Informed Categorization and Treatment for Repair/Replacement Activities in Class 2 and 3 Systems Section X1, Division 1" in lieu of certain requirements in ASME Section XI, Sub-Paragraphs IWA 1320, IWA 1400, IWA-4000, IWA-6210, IWA-6211, IWA-6212, IWA-6220, and IWA-6350.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(1), the licensee requested to use the proposed alternative on the basis that the alternative provides an acceptable level of quality and safety. ASME Code Case N-752-1 has not been approved by the U.S. Nuclear Regulatory Commission (NRC) staff or incorporated by reference for generic use. Therefore, the NRC staff reviewed the licensee's request as plant-specific requests for Turkey Point, St. Lucie, Point Beach, and Seabrook.

Enclosure

2.0 REGULATORY EVALUATION

2.1 Regulations

The following requirements are applicable to this request:

- Section 50.55a(g)(4), "Inservice inspection standards requirement for operating plants," of 10 CFR

- Section 50.55a(z)(1), "Acceptable level of quality and safety," of 10 CFR

2.2 Regulatory Guidance

The NRC staff used the following guidance in the evaluation of this request:

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

- RG 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)

- RG 1.177, Revision 2, "Plant-Specific Risk-Informed Decisionmaking: Technical Specifications," January 2021 (ML20164A034)

2.3 Applicable Precedents

The NRC staff considered and used the following prior NRC approvals in the evaluation of this request:

- U.S. NRC letter to Entergy Operations, Inc., "Arkansas Nuclear One, Units 1 and 2 - Approval of Request for Alternative from Certain Requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (EPID L-2020-LLR-0076)," May 19, 2021 (ML21118B039)

- U.S. NRC letter to Entergy Operations, Inc., "Arkansas Nuclear One, Units 1 and 2 - Request for Approval of Change to the Entergy Quality Assurance Program Manual (EPID L-2020-LLQ-0005)," May 19, 2021 (ML21132A279)

3.0 TECHNICAL EVALUATION

3.1 The Licensee’s Proposed Alternative

The applicable ASME Code editions and addenda for the applicable inservice inspection (ISI) intervals are specified in the table below for each plant.

Plant	ISI Interval	ASME Section XI Code of Record	Interval Start	Interval End
Turkey Point, Unit 3	5th	2007/2008	02/22/2014	02/21/2024
	6th	2019	02/22/2024	02/24/2034
Turkey Point, Unit 4	5th	2007/2008	04/15/2014	04/14/2024
	6th	2019	04/15/2024	04/14/2034
St. Lucie, Unit 1	5th	2007/2008	02/11/2018	02/10/2028
St. Lucie, Unit 2	4th	2007/2008	08/08/2013	08/07/2023
	5th	2019	08/08/2023	08/07/2033
Point Beach, Unit 1	6th	2017	08/01/2022	07/31/2032
Point Beach, Unit 2	6th	2017	08/01/2022	07/31/2032
Seabrook, Unit 1	4th	2013	08/19/2020	08/18/2030

3.2 ASME Code Components Affected

All ASME Class 2 and 3 items or components except the following:

- Class CC and MC items
- Piping within the break exclusion region [> Nominal Pipe Size (NPS) 4 (DN 100)] for high energy piping systems as defined by the Owner
- That portion of the Class 2 feedwater system [>NPS 4 (>DN 100)] of pressurized water reactors from the steam generator, including the steam generator, to the outer containment isolation valve

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 of IWC apply to items classified as ASME Class 2 and the rules in IWD apply to items classified as ASME Class 3.
    - Applies to all applicable NextEra Editions/Addenda.
- IWA-1400(f) or (g) requires Owners to possess or obtain an arrangement with an Authorized Inspection Agency.
    - 1400(f) applies to the 2007 Edition I 2008 Addenda
    - 1400(g) applies to the 2013, 2017 and 2019 Edition

- IWA-1400(j) or (k) requires Owners to perform repair/replacement activities in accordance with written programs and plans.

  - 1400(j) applies to the 2007 Edition I 2008 Addenda
  - 1400(k) applies to the 2013, 2017 and 2019 Edition

- IWA-1400(n) or (o) requires Owners to maintain documentation of a Quality Assurance Program in accordance with 10 CFR 50 or ASME NQA-1, Parts II and III.
  - 1400(n) applies to the 2007 Edition I 2008 Addenda
  - 1400(0) applies to the 2013, 2017 and 2019 Editions

- IWA-4000 specifies requirements for performing ASME Section XI repair/replacement activities on pressure retaining items or their supports.

  - Applies to all applicable NextEra Editions/Addenda.

- IWA-6210(e) - defines the preparation of code data reports.

  - Applies to 2007 Edition/2008 Addenda

- IWA-6211(d) and (e) define the preparation and required timing of code data reports and certification required from Repair/Replacement organizations other than the Owner.

  - Applies to the 2013, 2017 and 2019 Edition

- IWA-6212 repeats the requirement for certification by a Repair/Replacement Organization and refers to Appendix T as an example.

  - Applies to the 2013, 2017 and 2019 Edition

- IWA-6220 repeats the IWA-4150 requirement that a Repair/Replacement Plan be prepared for all repair/replacement activities, requires code data reports be completed, provides the required timing for completion of code data reports, identified certification requirements for code data reports and includes the requirement for maintaining an index of Repair/Replacement plans.

  - Applies to the 2017 and 2019 editions only.

- 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.

  - Applies to all applicable NextEra Editions/Addenda.

3.4 Proposed Alternative

NextEra requested to implement ASME Code Case N-752-1, without exception, at the NextEra nuclear stations as an alternative to certain ASME Code requirements for Turkey Point, St. Lucie, Point Beach, and Seabrook. In the submittal dated March 15, 2023, the licensee stated, in part:

> This requested implementation includes the categorization of passive SSCs [structures, systems, and components] (e.g., piping) and implement alternative special treatment activities limited to the repair/replacement activities for Class 2 and 3 pressure retaining items or their associated supports. For components that have both active and passive functions, only the passive function will be categorized. The alternative treatments associated with ASME Code Case N-752-1 will not be applied to the parts/components associated with the active function. ASME Code Case N-752-1 may be applied on a system basis or on individual items within selected systems. ASME Code Case N-752-1 will not be applied to Class 1 items.
>
> ASME Code Case N-752-1 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 to be affected by any changes in frequency due to changes in treatment. It additionally applies deterministic considerations (e.g., defense in depth [DID], safety margins) in determining safety significance.
>
> The risk-informed process categorizes components as either HSS [high-safety-significant function] or LSS [low-safety-significant function]. 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.

3.5 NRC Staff Evaluation

The NRC evaluated the licensee's request to determine if the proposed alternative met an acceptable level of quality and safety. The NRC staff reviewed the proposed alternative as a risk-informed request because the proposed alternative includes the use of a risk-informed process described in Appendix I of Code Case N-752-1. In evaluating the licensee's proposed alternative, the NRC staff considered the past precedent of previous NRC approved methods relating to risk-informed treatment of SSCs for nuclear power plants. Specifically, The NRC staff authorized the licensee for the Arkansas Nuclear One, Units 1 and 2 (ANO) to utilize alternative

ANO2-R&R 004, Revision 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 at ANO-2. By letter dated April 22, 2009, the NRC staff authorized the alternative (ML21118B039).

Probabilistic Risk Assessment Technical Acceptability

The proposed plant-specific approach for all NextEra sites utilizes the risk-informed categorization process in Appendix I of Code Case N-752-1 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).

The NRC staff's review of the PRA technical adequacy of all NextEra sites was based on the staff's previous determinations that the PRA models were found acceptable to support approval of risk-informed applications. In its submittal dated March 15, 2023, the licensee stated that the PRA has been approved by the NRC for Point Beach Nuclear Plant, Units 1 and 2, in the issuance of amendments regarding adoption of 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors" dated November 26, 2018 (ML18289A378); for St. Lucie Plant, Units 1 and 2, in the issuance of amendments regarding adoption of "Risk-Informed Completion Times in Technical Specifications," dated July 2, 2019 (ML19113A099); for Turkey Point Nuclear Generating Units 3 and 4, in the issuance of amendments regarding adoption of "Risk-Informed Completion Times in Technical Specifications," dated December 3, 2018 (ML18270A429). For Seabrook Station, in its response, dated October 30, 2019 (ML19305A301), to a "One-Time Change Seabrook Technical Specifications Onsite Power Distribution Requirements," the staff noted that all findings for the internal events model were reviewed and closed using the process documented in Appendix X to NEI 05-04, "Close-out of Facts and Observations" as accepted by the NRC.

The licensee states that its procedures require a PRA update that includes, but is not limited to operational practices and procedures, operational experience, and plant design changes. Furthermore, this review is required for all NextEra sites on a frequency not to exceed two refueling cycles. The NRC staff noted that these elements are consistent with the feedback and process adjustment described in Code Case N-752-1 that requires the review changes to the plant operational practices, applicable plant and industry operational experience, and, as appropriate, update the PRA and categorization and treatment processes.

Active Function Evaluation

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 resulting of implementing the proposed alternative. However, 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 failures including any effects of 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.

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 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 failure 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-752-1, and thus NextEra specific requests FRR-23-01, is useful for identifying components that should be prevented from failing using repair/replacement, planned maintenance and other treatment requirements. Code Case N-752-1 notes:

> 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 alternative request provide reasonable confidence that passive components and pressure retaining functions of active components will continue to perform their design basis functions, and, therefore, would not impact the basis for not using F-V.

Risk Tables

The proposed alternative references Code Case N-752-1 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. As explained in both ANO-R&R-004, Revision 1, and Code Case N-752-1, 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.

Review of Key Principles

The NRC staff evaluated the submittal using 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 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:

In the submittal dated March 15, 2023, the licensee stated that its request to use Code Case N-752-1 with no exceptions or deviations, including all definitions. Code Case N-752-1 requires the licensees to define alternative treatment requirements that confirm with reasonable confidence that each LSS item remains capable of performing its safety-related functions under design-basis conditions. Code Case N-752-1 describes requirements that must be addressed by the licensee's "to confirm with reasonable confidence that each LSS item remain capable of performing its safety-related functions under design-basis conditions." The requirements from Code Case N-752-1 include:

a) The Owner shall establish administrative controls for these repair/replacement activities.

b) The fracture toughness requirements of the original Construction Code and Owner's Requirements shall be met.

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.

d) Items used for repair/replacement activities shall meet the Owner's Requirements or revised Owner's Requirements as permitted by the licensing basis.

e) Items used for repair/replacement activities shall meet the Construction Code to which the original item was constructed. Alternatively, items used for repair/replacement activities shall meet the technical requirements of a nationally recognized code, standard, or specification applicable to that item (e.g., ASME, ANSI, AWS, AISC, AWWA, API 650, API 620, MSS SPs, or TEMA), as permitted by the licensing basis.

f) The repair methods of nationally recognized post-construction codes and standards (e.g., ASME PCC-2, API-653) applicable to the item may be used.

g) Performance of repair/replacement activities, and associated NDE [non-destructive examination], shall be in accordance with the Owner's Requirements and, as applicable, 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. NDE personnel may be qualified in accordance with IWA-2300, in lieu of the Construction Code.

h) Pressure-testing of the repair/replacement activity shall be performed in accordance with the requirements of the Construction Code selected for the repair/replacement activity, or shall be established by the Owner.

i) Baseline examination of the items affected by the repair/replacement activity, if required, shall be performed in accordance with the requirements of the Owner's program for periodic inspection of the item selected for examination.

j) These provisions do not negate or affect Owner commitments to regulatory and enforcement authorities having jurisdiction at the plant site.

The NRC staff finds that the licensee's adherence to the above elements covered in Code Case N-752-1 for repair/replacement activities provides reasonable confidence that each LSS item will remain capable of performing its safety-related function. The repair/replacement program quality elements will ensure that the LSS items remain capable of performing their design safety function.

In addition, ASME Code Case N-752-1 specifies that the owner (i.e., the licensee) is responsible for confirming "with reasonable confidence that each LSS items remains capable of performing its safety related functions under design-basis conditions" when defining requirements for design, procurement, installation, etc., for LSS items. As such, owners (i.e., licensee) must select an appropriate code or standard for performing repair/replacement activities on LSS items.

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 staff concludes that the proposed alternative permits acceptable flexibility in treatment alternatives, specifically for Class 2 and 3 LSS components, through a methodology based on the NRC-approved alternative ANO2-R&R-004 precedent and NextEra's plant-specific evaluation. Because the proposed alternative treatment is limited to LSS components, with defined treatment requirements (e.g., design control, corrective action, etc.) described in the enclosure to the licensee's submittal, the NRC staff finds that the codes and standards, as described, provide an acceptable level of quality and safety.

Key Principle 2:

In the submittal dated March 15, 2023, the licensee stated that its request to use Code Case N-752-1 with no exceptions or deviations, including all definitions. The categorization process described in Code Case N-752-1 includes the consideration of DID. According to Appendix I of Code Case N-752-1, the categorization process demonstrates DID philosophy is maintained if the following requirements in Code Case N-752-1 are met:

- Reasonable balance is preserved among prevention of core damage, prevention of containment failure or bypass, and mitigation of an offsite release.

- There is no over-reliance on programmatic activities and operator actions to compensate for weaknesses in the plant design.

- System redundancy, independence, and diversity are preserved commensurate with the expected frequency of challenges, consequences of failure of the system, and associated uncertainties in determining these parameters.

- Potential for common cause failures is taken into account in the risk analysis categorization.

- Independence of fission-product barriers is not degraded.

In the submittal dated March 15, 2023, the licensee stated, in part:

> The risk-informed methodology of ASME Code Case N-752-1 may be applied on a system basis or on individual items within selected systems. Paragraph -1100 of ASME Code Case N-752-1 states: "This Case may be applied on a system basis, including all pressure retaining items and their associated supports, or on individual items categorized LSS within the selected systems." While this is the case, the risk informed methodology is applied to the pressure boundary function of the individual components within the system. The risk informed methodology contained in ASME Code Case N-752-1 requires that the component's pressure boundary function be assumed to fail with a probability of 1.0, and all impacts caused by the loss of the pressure boundary function be identified. This would include identifying impacts of the pressure boundary failure on the component under evaluation, identifying impacts of the pressure boundary failure of the component on the system in which the component resides, as well as identifying impacts of the pressure boundary failure of the component on any other plant SSC. This includes direct effects (e.g., loss of the flow path) of the component failure and indirect effects of the component failure (e.g., flooding, spray, pipe whip, loss of inventory). This comprehensive assessment of total plant impact caused by a postulated individual component failure is then used to determine the final consequence ranking. As such, the final consequence rank of the individual component would be the same regardless of whether the entire system or only the individual component is subject to the risk informed methodology.

The proposed alternative does not alter any SSCs and will have no effect on layers of defense, or system redundancy. Additionally, the proposed alternative requires that DID philosophy be maintained. Therefore, the NRC staff concludes that the proposed change is consistent with the DID philosophy.

Key Principle 3:

In the submittal dated March 15, 2023, the licensee stated that its request to use Code Case N-752-1 with no exceptions or deviations, including all definitions. According to Appendix I of Code Case N-752-1, the categorization process shall verify sufficient margins in engineering analysis and supporting data and margin shall incorporated when determining performance characteristics for Class 2 and 3 SSCs identified as LSS. According to the Code Case N-752-1, sufficient margins are maintained by ensuring that safety analysis acceptance criteria in the plant licensing basis are met, or proposed revisions account for analysis and data uncertainty. If sufficient margins cannot be maintained, the categorization process described in Code Case N-752-1 requires that Class 2 or 3 SSC be identified as HSS, which will continue to meet the requirements of Section XI.

The proposed alternative requires the verification of sufficient margin for Class 2 or 3 SSC prior to applying the alternative requirements for LSS. If sufficient margin cannot be verified, then the requirements of Section XI still apply. Therefore, the NRC staff concludes that the proposed change maintains sufficient safety margin and provide an acceptable level of quality and safety.

Key Principle 4:

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 CCDP values of 5E-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 NRC staff notes that the proposed changes in treatments are 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.

Therefore, the NRC 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 March 15, 2023, the licensee described how the impact of the proposed changes would be monitored using performance management strategies. The licensee stated:

> NextEra 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. NextEra 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 NextEra's existing corrective action program.

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

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 (QA) program in accordance with 10 CFR Part 50 Appendix B or ASME NQA-1. The licensee's submittal mentions footnote (1) in Code Case N-752-1, 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. In its submittal dated March 15, 2023, NextEra/FPL stated it revised, in accordance with 10 CFR 50.54(a)(3), its fleet Quality Assurance Topical Report (QATR) for safety-related Class 2 and 3 SSCs identified as LSS in accordance with Code Case N-752-1. The licensee stated that the treatment of Class 2 and 3 SSCs identified as LSS "in accordance with existing QA Program procedures and processes which include supplemental controls to ensure the capability and reliability of the SSCs design basis function." The alternate treatments for LSS components are specified in the submittal dated March 15, 2023. 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 QATR without prior NRC approval provided the bases of the approval are applicable to the licensee's facility. According to the licensee, it concluded that the change to the NextEra QATR is not considered a reduction in QA Program commitments because of the change is consistent with a changed approved by the NRC for ANO (ML21132A279) and the bases of the NRC approval of the ANO change are applicable to Turkey Point, St. Lucie, Point Beach, and Seabrook.

The NRC staff issued a safety evaluation approving a proposed change to the Quality Assurance Program Manual (QAPM) at ANO under 10 CFR 50.54(a)(4) with specific QA requirements under 10 CFR Part 50, Appendix B (although less rigorous than previously applied at ANO under Appendix B) for safety-related Class 2 and Class 3 components categorized as LSS when implementing Code Case N-752 at ANO. The NRC approval of the changes to the QAPM at ANO are based on the specific QA requirements for safety-related LSS Class 2 and Class 3 components when implementing Code Case N-752-1 documented in the Entergy's submittals for ANO dated October 26, 2020, April 5, 2021, and April 30, 2021. 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-752-1 in lieu of the corresponding sections of ASME Section XI. Further, alternate 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 NRC staff reviewed the proposed change to the Entergy's QAPM and concluded that the proposed alternative, as described above, still met the requirements of Appendix B, which includes the alternate treatment requirements of Code Case N-752 and the use of current safety-related procedures to address program elements of the treatment requirements of the LSS items.

The NRC staff confirmed that the changes to the QATR proposed by NextEra Energy are consistent with the changes approved by the NRC staff to Entergy's QAPM as documented in the safety evaluation dated May 19, 2021 (ML21132A279), therefore, it is not considered a reduction in commitment in accordance with 10 CFR 50.54(a)(3)(ii). The NRC staff concluded that there is reasonable assurance that the licensee's QATR continues to meet the requirements of Appendix B to 10 CFR Part 50 as described above.

3.6 NRC Staff Conclusion

Based on the information provided, the NRC staff finds, with reasonable assurance, that the St. Lucie, Turkey Point, Seabrook Station and Point Beach PRAs reflect the as-built, as-operated plants to support the safety significance categorization of FRR-23-01, 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.

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 RI-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 request, 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

As set forth above, the NRC staff determines that the proposed alternative provides an acceptable level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1). Therefore, the NRC staff authorizes the proposed alternative for the remainder of the fifth and the entirety of the sixth inservice inspection intervals at Turkey Point, the remainder of the fifth inservice inspection interval at St. Lucie, Unit 1, the remainder of the fourth and the entirety of the fifth inservice inspection intervals at St. Lucie, Unit 2, the remainder of the sixth inservice inspection intervals at Point Beach, and the remainder of the fourth inservice inspection interval at Seabrook.

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.

In its request, the mentioned in several places that risk-informed categorization process described in Code Case N-752-1 is "consistent" with the risk-informed categorization process described in 10 CFR 50.69. For the licensee's facilities that have been approved to use a risk-informed categorization process under 10 CFR 50.69, this authorization does not change any obligations with regards to the 10 CFR 50.69 risk-informed categorization process for SSCs.

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Date: June 12, 2024

SUBJECT: SAINT LUCIE NUCLEAR PLANT, UNITS 1 AND 2; TURKEY POINT NUCLEAR PLANT, UNITS 3 AND 4; SEABROOK NUCLEAR PLANT; AND POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2 - PROPOSED ALTERNATIVE FRR 23-01 TO USE ASME CODE CASE N-752-1, "RISK-INFORMED CATEGORIZATION AND TREATMENT FOR REPAIR/REPLACEMENT ACTIVITIES IN CLASS 2 AND 3 SYSTEMS SECTION X1, DIVISION 1" (EPID L-2023-LLR-0009) DATED JUNE 12, 2024

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