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ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Edwin I. Hatch Nuclear Plant Units 1&2
Joseph M. Farley Nuclear Plant Units 1&2
Vogtle Electric Generating Plant Units 1&2
Proposed Alternative to Use ASME Code Case N-752,
Risk-Informed Categorization and Treatment for Repair/ Replacement
Activities in Class 2 and 3 Systems, Section XI, Division 1

Ladies and Gentlemen:

Pursuant to 10 CFR 50.55a(z)(1), Southern Nuclear Operating Company (SNC) requests the Nuclear Regulatory Commission's (NRC) authorization of a proposed alternative GEN-ISI-ALT-2024-01 to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components."

Authorization is requested to use the alternative requirements of 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 Boiler and Pressure Vessel Code, Section XI, IWA-1000, IWA-4000, and IWA-6000 requirements.

Use of the proposed alternative is based upon 10 CFR 50.55a(z)(1) which states that the licensee must demonstrate that the proposed alternative would provide an acceptable level of quality and safety. The proposed alternative and the basis for the proposed alternative are provided in the Enclosure to this letter.

SNC requests NRC authorization of the proposed alternative within one year of acceptance.

No regulatory commitments are contained in this submittal. If you have any questions, please contact Ryan Joyce at 205.992.6468.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on the 27th day of June, 2024

Respectfully,

Jamie Coleman

Jamie M. Coleman
Regulatory Affairs Director

Enclosure: Proposed Alternative GEN-ISI-ALT-2024-01

Cc: Regional Administrator, Region II
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RTYPE: CGA02.001

Edwin I. Hatch Nuclear Plant - Units 1 and 2
Joseph M. Farley Nuclear Plant - Units 1 and 2
Vogtle Electric Generating Plant - Units 1 and 2

Enclosure to NL-24-0143

Proposed Alternative GEN-ISI-ALT-2024-01

10 CFR 50.55a Request No. GEN-ISI-ALT-2024-01
Implementation of Code Case N-752
Farley, Vogtle, and Hatch Nuclear Plants Units 1 and 2
Proposed Alternative in Accordance with 10 CFR 50.55a(z)(1)
Acceptable Level in Quality and Safety

1. American Society of Mechanical Engineers (ASME) Code Component(s) Affected

This request applies to 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 [a] as defined by the Owner.
- 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.

a. NUREG-0800, Section 3.6.2 provides a method for defining this scope of piping.

2. Applicable Code Edition and Addenda

The applicable Code Editions and Addenda for the current Inservice Inspection (ISI) intervals at each station are as specified below for each applicable Southern Nuclear Operating Company (SNC) site.

Plant	ISI Interval	ASME Section XI Edition/Addenda (Reference 8.1)	Interval Start	Interval Scheduled End
Farley Nuclear Plant (FNP), Units 1 and 2	5	2007 Edition/2008 Addenda [b, c]	12/01/2017	11/30/2027
Hatch Nuclear Plant (HNP), Units 1 and 2	5	2007 Edition/2008 Addenda [b, c]	01/01/2016	12/31/2025
Vogtle Electric Generating Plant (VEGP), Units 1 and 2	4	2007 Edition/2008 Addenda [b, c]	05/31/2017	05/30/2027

b. Per Regulatory Issue Summary (RIS) 2004-12 (Reference 8.2), Letter GEN-ISI-ALT-2020-02 (Reference 8.3) and authorization (Reference 8.4), the NRC staff concluded that the use of subparagraph IWA-4540(b) of the 2017 Edition of the ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, is acceptable for HNP Units 1&2 and FNP Units 1&2. Similarly, per letter VEGP-ISI-ALT-04-06 (Reference 8.5) and authorization (Reference 8.6), the NRC staff concluded that the use of subparagraph IWA-4540(b) of the 2017 Edition of the ASME B&PV Code, Section XI, is acceptable for VEGP Units 1&2. Therefore, as the applicability of this alternative, the Code of Record for IWA-4540(b) is the 2017 Edition of the ASME Section XI.

c. Per letter GEN-ISI-ALT-2023-01 (Reference 8.7) and authorization (Reference 8.8), the NRC staff concluded that the use of IWA-6211(a) through IWA-6211(e), IWA-6220, IWA-6230, and Mandatory Appendix II of the 2019 Edition of the ASME B&PV Code, Section XI, is acceptable for HNP Units 1&2, FNP Units 1&2 and VEGP Units 1&2. Therefore, as the applicability of this alternative, the Code of Record for IWA-6211(d), IWA-6211(e), and IWA-6220 is the 2019 Edition of ASME Section XI.

3. Applicable Code Requirement

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) [d] requires Owners to possess or obtain an arrangement with an Authorized Inspection Agency (AIA).
- IWA-1400(j) [d] requires Owners to perform repair/replacement activities in accordance with written programs and plans.
- IWA-1400(n) [d] 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-6211(d) and (e), specify Owner reporting responsibilities such as preparing Form NIS-2, Owner's Repair/Replacement Certification Record.
- IWA-6220 repeats the IWA-4150 requirements that a Repair/Replacement Plan be prepared for all repair/replacement activities, requires Form NIS-2 be completed, provides the required timing for completion of Form NIS-2, identifies certification requirements for Form NIS-2, and includes the requirement for maintaining an index of Repair/Replacement Plans.
- 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.

d. Code Case N-752 is based on the 2017 Edition of ASME Section XI while SNC's Code of record for HNP, FNP and VEGP 1&2 is the 2007 Edition/2008 Addenda, except as noted in Section 2 of this request. Below is a cross reference for affected code paragraphs:

- IWA-1400(g), (k), and (o) in the 2017 Edition are IWA-1400(f), (j), and (n) in the 2007 Edition/2008 Addenda.
- IWA-6211(f) and IWA-6212 in the 2017 Edition do not exist in or apply to the 2007 Edition/2008 Addenda.

4. Reason for Request

SNC currently performs ASME Section XI Repair/Replacement Activities at HNP Units 1&2, FNP Units 1&2 and VEGP Units 1&2 in accordance with a deterministic Repair/Replacement Program based on the 2007 Edition/2008 Addenda of ASME Section XI. Repair/Replacement Program requirements apply to procurement, design, fabrication, installation, examination, and pressure testing of items within the scope of ASME Section XI. Repair/replacement activities include welding, brazing, defect removal, metal removal using thermal processes, rerating, and removing, adding, or modifying pressure-retaining items or supports. Repair/replacement activities are performed in accordance with SNC's 10 CFR 50,

Appendix B Quality Assurance (QA) Program and the ASME Section XI Code. In applying a deterministic approach to repair/replacement activities, a safety class (e.g., ASME Class 2 or 3) is assigned to every component within a system based on system function; the same treatment requirements are then applied to every component within the system without considering the risk associated with the probability that a specific item or component may or may not be functional at a time when needed.

Alternatively, a probabilistic approach to regulation enhances and extends the traditional deterministic approach by allowing consideration of a broader set of potential challenges to safety, providing a logical means for prioritizing these challenges based on safety significance, and allowing consideration of a broader set of resources to defend against these challenges. In contrast to the deterministic approach, Probabilistic Risk Assessment (PRA) addresses credible initiating events by assessing the event frequency. Mitigating system reliability is then assessed, including the potential for common cause failures. The probabilistic approach to regulation is an extension and enhancement of traditional regulation by considering risk in a comprehensive manner. In 2004, the NRC adopted a new Section 50.69 of 10 CFR relating to risk-informed categorization and treatment of structures, systems, and components (SSCs) for nuclear power plants (Reference 8.9). This new section permits power reactor licensees to implement an alternative regulatory framework with respect to "special treatment" (treatment beyond normal industrial practices) of low safety significant (LSS) SSCs. In May 2006, the NRC staff issued Regulatory Guide (RG) 1.201, “Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance, For Trial Use,” Revision 1 (Reference 8.10). RG 1.201 endorses a categorization method, with conditions, for categorizing active SSCs described in Nuclear Energy Institute (NEI) 00-04, “10 CFR 50.69 SSC Categorization Guideline.”

SNC is not requesting NRC approval to implement 10 CFR 50.69 in this request. As noted below in Section 5.2.C, SNC is already approved to implement 10 CFR 50.69 at Hatch, Farley, and Vogtle Units 1 and 2, and this alternative request does not change any obligations with regards to these existing 10 CFR 50.69 programs. Instead, SNC is proposing to implement the risk-informed categorization and treatment requirements of ASME Code Case N-752 when performing repair/replacement activities on Class 2 and 3 pressure-retaining items or their associated supports. Code Case N-752, approved by the ASME in July 2019, employs a comprehensive categorization process requiring input from both a PRA model and deterministic insights. This approach will enable evaluation, categorization, and implementation of alternative treatments for resolution of emergent issues in segments of piping having low safety significance. Use of Code Case N-752 will also allow SNC to identify, and more clearly focus engineering, maintenance, and operations resources on, critical components with high safety-significance, thus, enabling SNC to make more informed decisions and increase the safety of the plant.

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

Pursuant to 10 CFR 50.55a(z)(1), SNC proposes to implement ASME Code Case N-752, without exception, as an alternative to the ASME Code requirements specified in Section 3. ASME 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 as defined in Section 1. This requested implementation includes the categorization of passive SSCs (e.g., piping) and implementation of 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 will not be applied to the parts/components associated with the active function. Code Case N-752 may be applied on a system basis or on individual items within selected systems. Code Case N-752 does not apply to Class 1 items.

The use of this proposed alternative is requested on the basis that requirements in Code Case N-752 will provide an acceptable level of quality and safety.

5.1 <u>Overview of Code Case N-752</u>

Code Case N-752 provides for risk-informed categorization and treatment requirements for performing repair/replacement activities on Class 2 and 3 pressure retaining items or their associated supports. ASME Code Case N-752 is not applicable to the following:

- Class CC and MC items.
- Piping within the break exclusion region [> 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 PWRs from the SG, including the SG, to the outer containment isolation valve.

ASME 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 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. 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 8.20), as supplemented by Entergy. The NRC approved relief request ANO2-R&R-004, Revision 1, in a safety evaluation dated April 22, 2009 (Reference 8.21). 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.

5.2 Basis for Use

The information below is provided as a basis or justification for SNC's proposed alternative to implement the risk-informed categorization and treatment requirements of Code Case N-752 on Class 2 and 3 pressure retaining items or the associated supports as delineated in Section 1.

A. Application to Individual Items Within a System

The risk-informed methodology of Code Case N-752 may be applied on a system basis or on individual items within selected systems. Paragraph -1100 of Code Case N-752 states: “This Case may be applied on a system basis, including all pressure retaining items and their associated supports, or on individual items categorized as low safety significant (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 Code Case N-752 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 ranking 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.

B. Categorization Process

The categorization process of Code Case N-752 is delineated in Appendix I of the Code Case. This categorization process is technically identical to the process approved by the NRC under Relief Request ANO2-R&R-004, Revision 1 (Reference 8.20), which, in turn, is based on founding principles in EPRI Report TR-112657, Revision B-A, “Revised Risk-Informed Inservice Inspection Evaluation Procedure,” and the categorization process of Code Case N-660, but with improvements and lessons learned from trial applications. A comparison of the ANO2-R&R-004, Revision 1 categorization methodology to that of Code Case N-752 was provided by Entergy in Enclosure Attachment 1 of Precedent 7.1.

The ASME Code Case N-752 risk-informed categorization evaluation is performed by an Owner-defined team that includes members with expertise in PRA, plant operations, system design, and safety or accident analysis. The risk-informed categorization process is based on the conditional consequence of failure, given that a postulated failure has occurred. A consequence category for each piping segment or component is determined via a failure modes and effects analysis (FMEA) and impact group assessment. The FMEA considers pressure boundary failure size, isolability of the break, indirect effects, initiating events, system impact or recovery, and system redundancy. The results of the FMEA for each system, or portion thereof, are partitioned into core damage impact groups based on postulated piping failures that (1) cause an initiating event, (2) disable a system/train/loop without causing an initiating event, or (3) cause an initiating event and disable a system/train/loop.

Failures are also evaluated for their importance relative to containment performance. In addition, the consequence rank is reviewed and adjusted to reflect the pressure boundary failure's impact on plant operation during shutdown and on the mitigation of external events. Credit may be taken for plant features and operator actions to the extent these would not be adversely affected by failure of the piping segment or component under consideration.

Consequence evaluation results are ranked as High, Medium, Low, or None (no change to base case). Piping segments/components ranked as High by the consequence evaluation process are considered HSS and require no further review. Piping segments/components ranked as Medium, Low, or None by the consequence evaluation shall be determined to be HSS or LSS by evaluating the additional categorization considerations or conditions outlined in paragraph I-3.4.2(b) of ASME Code Case N-752. If any of these conditions are not met, then HSS shall be assigned. If all conditions are met, then LSS may be assigned. If LSS is assigned, the categorization process shall verify that there are sufficient margins to account for uncertainty in the engineering analysis and supporting data. If sufficient margin exists, then LSS should be assigned. If sufficient margin does not exist, then HSS shall be assigned.

C. PRA Technical Adequacy

Appendix I, Section I-3.2 of Code Case N-752 requires that the plant-specific PRA shall be assessed to confirm it is applicable to the safety significant categorization of Code Case N-752 including verification of assumptions on equipment reliability for equipment not within the scope of the ASME Code Case.

EDWIN I. HATCH NUCLEAR PLANT UNITS 1&2

The Hatch PRA model was previously used in the approval of Risk-Informed Completion Time [ML22297A146] and 10 CFR 50.69 [ML20077J704].

The full scope peer review of the Plant Hatch Full-Power Internal Events (including Internal Flooding) (FPIE) Probabilistic Risk Assessment (PRA) was performed in November 2009. The peer review team issued a total of 25 unique findings. In April 2017, an Appendix X Fact and Observation (F&O) Closure Independent Assessment (IA) was performed to review SNC's disposition of the finding-level F&Os received from the November 2009 peer review. This closure review was performed in accordance with the requirements described in NEI 05-04 (Reference 8.11), and in particular, Appendix X of this report (Reference 8.12). The NEI methodology for performing F&O closure reviews was accepted by the US Nuclear Regulatory Commission (References 8.13 & 8.14), with a number of clarifications and specific documentation expectations noted. Upon completion of the IA, of the 25 total reviewed findings, only 3 were left OPEN and 1 left “Partially Closed”. Two F&Os were specific to internal flooding technical element and are discussed below. The other two F&Os relate to the accident sequence and human reliability technical elements. These two findings have since been resolved by performing the recommendations from the IA team. See Table 1 for the disposition of these findings with respect to this application. The other 21 reviewed findings were closed by the IA team. Given the resolution of the remaining findings, the Hatch Internal Events PRA is of adequate technical capability to support Code Case N-752.

The Hatch Internal Flooding PRA was the subject of a full scope peer review in November 2009. A subsequent Appendix X F&O closure was also performed to address open F&Os in April 2017. Two F&Os related to internal flooding technical element remained, one open and one partially closed. In response to the resolution of these two F&Os, SNC determined that a focused scope peer review (FSPR) was triggered due to significant changes in risk insight resulting from an update to the internal flooding PRA model. A FSPR was performed in October 2019 to address the supporting requirements that are associated with the responses to these two findings. As a result of the FSPR, no finding-level F&Os were generated. Given there are no remaining open finding-level F&Os, the Hatch Internal Flooding PRA is of adequate technical capability to support Code Case N-752.

Table 1

Peer Review F&Os Not Closed

Supporting Requirement(s)	Finding	Disposition	Impact to Code Case N-752 Implementation
AS-B3, AS-C2	Reviewed the AS Notebook. Generally, the discussion of the accident sequence modeling is adequate. However, the following information was not present: (1) Discussion of environmental conditions associated with sequences. (2) Interface between accident sequences and plant damage states. <u>Basis</u>: SR AS-B3 requires the identification of phenomenological conditions expected from each accident sequence. <u>Possible Resolution</u>: Include additional detail for each accident sequence. Particularly, there was no mention of the generation of harsh environments affecting temperature, pressure, debris, water levels or humidity that could impact the success of the system. <u>Per 2017 App. X Finding Closure Review</u>: The detail required by this finding has been added to the accident sequence notebook. The sequence descriptions have a discussion of Environmental Conditions. However, no information has been provided on why no environmental impact. One Accident	Original F&O Disposition: In AS Notebook Subsections 3.#.3, Environmental Conditions have been evaluated for each accident sequence. In subsections 3.#.5, each sequence has a detailed description of the characteristics, which also describes the interface between accident sequences and the plant damage states. 2017 Appendix X Finding Closure Disposition: In AS Notebook Section 3.0, a discussion was added that the Hatch PRA does not typically rely on equipment or operator actions in an area where a severe environment is expected. A discussion on environmental considerations for the station blackout (SBO) sequence where fire water is used was added to the documentation.	Based on the Hatch Equipment Qualification Program, equipment located in potentially harsh environment conditions, including inside containment, are expected to perform its safety function when exposed to normal, abnormal, and accident environment. For all other areas, the models do not credit use of equipment in the area for events that cause adverse environmental events, such as Interfacing System Loss of Coolant Accident (ISLOCA) events and steam line breaks outside containment. The Internal flooding analysis evaluates the susceptibility of components to spray and flooding separately. This

Table 1

Peer Review F&Os Not Closed

Supporting Requirement(s)	Finding	Disposition	Impact to Code Case N-752 Implementation
	Sequence that does have an adverse impact, is based upon an assumption of equipment qualifications. No listing of Environmental considerations is provided for SBO sequences with uses of Fire Water. No discussion about access or other issues.		finding is a documentation issue; therefore, there is no impact to the implementation of Code Case N-752.
HR-G6	Consistency check did not include comparison of Human Error Probabilities (HEPs) in regards to scenario context, plant history, procedures, operational practices, and experience. <u>Basis</u>: Section 8.3 of the Human Reliability Analysis (HRA) notebook evaluated consistency of HEPs based on the ratio of execution time and recovery time to check the reasonableness of the derived probability. This checks the reasonableness of the derivation of the probability. However, there was no check of consistency based on scenario context, plant history, procedures. <u>Possible Resolution</u>: Compare HEPs and determine if the HEP values, relative to each other, are representative of applicable	The internal events HRA documentation was revised to incorporate a better consistency analysis. A discussion on requirements from NUREG-1792 and feasibility requirements from NUREG-1921 to be used for internal events Human Failure Events (HFEs) were added to the HRA notebook. HFEs and their HEP were reviewed relative to each other to check their reasonableness given the scenario context including plant procedures, plant history, operational practices and experiences and documented in the HRA notebook.	The documentation associated with this issue has been revised. There is no impact to the implementation of Code Case N-752.

Table 1

Peer Review F&Os Not Closed

Supporting Requirement(s)	Finding	Disposition	Impact to Code Case N-752 Implementation
	scenario context, plant history, procedures, operational practices, and experience. <u>Per 2017 App. X Finding Closure Review</u>: Reviewed Hatch Human Reliability Analysis Update dated Oct. 26, 2009. The evidence presented in the HRA notebook does not appear to be adequate, comparing with the documentation for a similar BWR plant. The model owner at the time of the Peer Review felt that the brief comparison of action versus overall timing in Section 8.3 of the HRA notebook was adequate to comply with HR-G6 and that no additional information needed to be added. Because of this approach there were no revisions made to the HRA notebook after the peer review and there are no added words or amplifying discussion to further support this view in the calculation.		

JOSEPH M. FARLEY NUCLEAR PLANT UNITS 1&2

The Farley PRA model was previously used in the approval of Risk-Informed Completion Time [ML19175A243] and 10 CFR 50.69 [ML21137A247].

The full scope internal events (including internal flooding) PRA peer review was subject to a self-assessment and a full scope peer review conducted in March 2010. A finding closure review was conducted on the internal events (including internal flooding) in October 2018. Closed findings were reviewed and closed using the process documented in Appendix X (Reference 8.12). The results of this review have been documented and are available for NRC audit. All finding level F&Os were closed out.

In November-December 2019, a focused-scope peer review of the Farley internal events, internal flooding, fire, and seismic PRAs against applicable requirements of the ASME/ANS PRA standard (Reference 8.15) was conducted of the following PRA model upgrades:

- Reactor Coolant Pump Shutdown Seal Model (applicable to both internal events and internal flooding PRA models)
- Main Control Room Abandonment (applicable to fire PRA model only)
- FLEX modeling with FLEX HRA (applicable to all PRA models)

The FLEX HRA portion of the review was performed using the Integrated Human Event Analysis System (IDHEAS) Method based on Electric Power Research Institute (EPRI) FLEX Human Reliability Analysis (HRA) Report 3002013018 (Reference 8.16). As a result of this focused-scope peer review, a total of seven F&Os were generated, all of which were “Suggestion” type F&Os. The above demonstrates that the Farley Internal Events and Internal Flooding PRA is of adequate technical capability to support Code Case N-752.

VOGTLE ELECTRIC GENERATING PLANT UNITS 1&2

The Vogtle 1&2 PRA model was previously used in the approval of Risk-Informed Completion Time [ML15127A669] and 10 CFR 50.69 [ML14237A034].

The full scope internal events (including internal flooding) PRA peer review was performed in May 2009 against ASME RA-Sb-2005 (Reference 8.17) and any clarifications and qualifications provided in the NRC endorsement of the Standard contained in Revision 1 to RG 1.200 (Reference 8.18). A gap assessment to the 2009 ASME Standard (Reference 8.19) was completed in support of the Risk-Informed Completion Time License Amendment Request [ML12258A055].

In July 2019, an Appendix X Facts and Observations (F&O) closure review of the F&Os received from the 2009 full scope peer review was performed to address open F&Os. All finding level F&Os were closed using the Appendix X process (References 8.12 and 8.13). During the same month, a focused scope peer review (FSPR) was also performed as there were some model changes (performed as part of the scheduled model maintenance) to the Vogtle 1&2 Internal Events (including Internal

Flooding) PRA model that met the guidance for classification as “upgrades”. These “upgrades” were self-identified by SNC. Seven finding level F&Os were issued. In November 2019, An Appendix X F&O closure review was performed with a scope of the seven finding level F&Os. All seven finding level were closed.

In 2020, the internal flooding PRA (IFPRA) model was split off from the internal events PRA model to create a separate, stand-alone IFPRA. As part of the IFPRA model maintenance performed during the same year, SNC self-identified changes that constitute a change in methodology, which was not previously applied to the Vogtle IFPRA and as such not covered in previous peer reviews. Also, some of the revisions involve changes in the PRA scope and/or capability that impact the significant accident sequences, significant accident progression sequences, or risk insights. Per the ASME/ANS PRA Standard requirements, these types of changes are considered as PRA model upgrades, and thus a focused-scope peer review was performed in December 2020 resulting in 17 finding level and 12 suggestion level F&Os.

An Appendix X closure review of the F&Os from the IFPRA focused-scope peer review was performed in December 2021. All F&Os from the IFPRA FSPR were closed out using the Appendix X process (References 8.12 and 8.13). Given there are no remaining open F&Os, the Vogtle 1 & 2 Internal Events and Internal Flooding PRAs are of adequate technical capability to support Code Case N-752.

D. Feedback and Process Adjustment

SNC 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. SNC 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).

E. Treatment Requirements for LSS Items

Code Case N-752 exempts LSS items, which have been categorized as LSS in accordance with the code case, from having to comply with the repair/replacement requirements of ASME Section XI. Exempted ASME Code requirements for LSS items are outlined in Section 3, above. In lieu of these requirements, Code Case N-752, Paragraph -1420 requires the Owner to define alternative treatment requirements which confirm with reasonable confidence that each LSS item remains capable of performing its safety-related functions under design basis conditions. These Owner treatment requirements must address or include all the provisions stipulated in Paragraphs -1420(a) through (j) of the code case. This approach to treatment is consistent with RISC-3 treatment requirements specified in 10 CFR 50.69(d)(2).

To comply with the above, SNC will develop and/or revise existing procedures and documents to define treatment requirements for performing repair/replacement activities on LSS items in accordance with the Code Case N-752. Defined treatment requirements address design control, procurement, installation, configuration control, and corrective action. These procedures and documents also include provisions which address/implement the following requirements:

1. Administrative controls for performing these repair/replacement activities.

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

3. 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.

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

5. 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 as permitted by the licensing basis.

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

7. Performance of repair/replacement activities, and associated non-destructive examination (NDE), 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.

8. 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.

9. 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. See paragraph 5.2.E.11, below, for additional details.

10. Implementation of Code Case N-752 does not negate or affect SNC commitments to regulatory and enforcement authorities having jurisdiction at HNP Units 1&2, FNP Units 1&2 and VEGP 1&2.

11. Periodic ISI and inservice testing (IST) of LSS items at HNP Units 1&2, FNP Units 1&2 and VEGP 1&2 will be performed as follows:

- ISI of LSS pressure retaining items or their associated supports will be performed in accordance with the site's ISI program implemented in accordance with 10 CFR 50.55a.
- IST of pumps and valves that have been classified as LSS will be performed in accordance with the site's IST program implemented in accordance with 10 CFR 50.55a.
- IST of snubbers that have been classified as LSS will be performed in accordance with the site's Snubber Testing program implemented in accordance with 10 CFR 50.55a.
- Inspections of LSS items performed under other plant programs, such as the Flow Accelerated Corrosion and Microbiologically Induced Corrosion programs, will continue to be performed under those programs for the site.

12. Adverse conditions identified in LSS components will be entered in the SNC corrective action program, which satisfies 10 CFR 50 Appendix B criteria for corrective action. 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 SSCs under 10 CFR 50.36, "Technical Specifications", adverse conditions will be addressed within the timeline of the limiting conditions of operability, or the necessary action statements will be performed. 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. The SNC corrective action process takes appropriate actions to monitor, investigate, and/or correct undesired conditions with the level of emphasis and effort commensurate with the risk and significance of the issue. Finally, this approach to corrective action of LSS items is consistent with the NRC position on corrective action of RISC-3 SSCs as specified in 10 CFR 50.69(d)(2)(ii).

13. As permitted by Code Case N-752, SNC intends to implement the exemption from IWA-1400(f) and IWA-4000 applicable to utilization of an Authorized Inspection Agency (AIA) and Authorized Nuclear Inservice Inspector (ANII) when performing repair/replacement activities on LSS items. In lieu of ANII inspection services, SNC believes that its proposed treatment requirements, as described herein, provide reasonable confidence that LSS systems and items remain capable of performing their safety-related functions when repair/replacement activities are performed without the inspection services of an ANII. It should also be noted that the exemption of ANII services is not unique to Code Case N-752. Utilization of ANII inspection services is already exempt by ASME Section XI for certain items and activities such as small items (IWA-4131) and rotation of items for testing or preventative maintenance (IWA-4132). Finally, exemption of AIA/ANII services for this code case application is consistent with the NRC's position on risk-informed programs as specified in 10 CFR 50.69(b)(1)(v).

14. Code Case N-752 paragraph -1420 allows LSS items to be exempt from the requirements of certain ASME Section XI, including subparagraph IWA-1400(n) and article IWA-4000. However, Code Case N-752 does not allow exemption

from ASME Section XI subparagraph IWA-1400(n) if compliance with 10 CFR 50 Appendix B or NQA-1 is required at the Owner's facility, as is the case for SNC's nuclear facilities. SNC's Quality Assurance Program requirements, currently applicable to Farley, Hatch, and Vogtle Units 1 & 2, are described in the SNC Quality Assurance Topical Report (QATR). Changes to the QATR are subject to the regulatory change control requirements of 10 CFR 50.54(a)(3). Accordingly, SNC intends to amend QATR Section 1.1 “Scope / Applicability” in accordance with 10 CFR 50.54(a)(3) to include the following:

> For SNC nuclear sites having received NRC authorization to use the alternative repair/replacement categorization and treatment requirements of ASME Code Case N-752 in lieu of the corresponding sections of ASME Section XI, as referenced in 10 CFR 50.55a Codes and Standards, treatment of safety-related structures, systems, and components identified as low safety significant (LSS) Class 2 and 3 SSCs in accordance with ASME Code Case N-752 is not required to meet the requirements of this manual. Instead, treatment of these LSS SSCs is performed 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 basis for the SNC QATR change is established in the precedent identified in Section 7.2 of this alternative request and in accordance with 10 CFR 50.54(a)(3)(ii), which establishes that a quality assurance alternative or exception approved by an NRC safety evaluation is not considered a reduction in QA Program commitments provided the bases of the NRC approval are applicable to the licensee's facility. Consistent with the precedent in Section 7.2, under the amended QATR, SNC will define alternative treatment requirements that confirm with reasonable confidence that each Class 2 and 3 LSS SSC will remain capable of performing its safety-related function under design-basis conditions. In doing so, SNC will use current QA Program processes and procedures with additional controls for the treatment of Class 2 and 3 LSS components to reasonably assure continued capability and reliability of the design-basis function(s). This includes confirming, with reasonable confidence, that changes to the configuration, design, material, fabrication, examination, and testing requirements used to support repair/replacement activities on Class 2 and 3 LSS SSCs are performed in accordance with SNC's existing design change process and addressing in SNC's corrective action program any condition that may prevent a LSS SSC from performing its design-basis function. For the procurement of Class 2 and 3 LSS components as non-safety-related for repair/replacement activities in accordance with ASME's Code Case N-752, supplemental procurement requirements will be specified, and additional controls will be implemented as appropriate to provide reasonable assurance that Class 2 and 3 LSS SSCs will remain capable of performing their safety-related function under design basis conditions. Such controls include conducting receipt inspections using qualified inspection personnel consistent with SNC's procurement requirements and prohibiting suppliers of Class 2 and 3 SSCs and subparts from making design changes or changes to the procurement order without prior SNC approval. Using these existing QA Program processes and alternative treatment requirements, SNC believes that the implementation of

ASME Code Case N-752 will provide reasonable assurance that each Class 2 and 3 LSS SSC remains capable of performing its design-basis function, and the SNC QATR will continue to provide an acceptable level of quality and safety.

15. As permitted by Code Case N-752, SNC intends to implement the exemptions from IWA-1400(j) and IWA-4000 applicable to repair/replacement programs and plans. In lieu of these ASME Section XI administrative controls, SNC will establish Owner defined administrative controls as required by paragraph -1420(a) of Code Case N-752. SNC will utilize its existing work management processes for planning and documenting the performance of repair/replacement activities and supplement those process requirements as necessary to comply with Code Case N-752. These controls will confirm, with reasonable confidence, that repair/replacement activities on LSS items are performed in accordance with work instructions that have been appropriately, planned, reviewed, and implemented. It should also be noted that the exemption of Repair/Replacement Plans as required by IWA-1400(j) and IWA-4150 is not unique to Code Case N-752. Repair/Replacement Plans are already exempt by ASME Section XI for certain items and activities such as small items (IWA-4131) and rotation of items for testing or preventative maintenance (IWA-4132). Finally, the exemption of ASME Section XI programs and plans and the alternative use of Owner-defined administrative requirements on LSS items is consistent with the NRC's position on risk-informed programs as specified in 10 CFR 50.69(b)(1)(v).

16. As permitted by Code Case N-752, SNC intends to implement the exemption on IWA-4000 applicable to repair/replacement activities. Article IWA-4000 of the ASME Section XI Code specifies administrative, technical, and programmatic requirements for performing repair/replacement activities on pressure retaining items and their supports. As specified in IWA-4110(b), repair/replacement activities "include welding, brazing, defect removal, metal removal by thermal means, rerating, and removing, adding, and modifying items or systems. These requirements are applicable to procurement, design, fabrication, installation, examination, and pressure testing of items within the scope of this Division." In lieu of these IWA-4000 requirements, SNC will perform repair/replacement activities on LSS items in accordance with an Owner defined program that complies with paragraph -1420 of ASME Code Case N-752. The SNC program will utilize existing SNC processes such as those applicable to procurement, design, re-rating, fabrication, installation, modifications, welding, defect removal, metal removal by thermal processes and supplement those process requirements as necessary to comply with Code Case N-752. SNC believes this program will confirm, with reasonable confidence, that LSS items remain capable of performing their safety-related functions under design basis conditions. Finally, the exemption of IWA-4000 requirements and the alternative use of Owner-defined treatment requirements for LSS items is consistent with the NRC's position on risk-informed programs as specified in 10 CFR 50.69(b)(1)(v) and (d)(2).

17. As permitted by Code Case N-752, SNC intends to implement the documentation exemptions on IWA-6211(d), IWA-6211(e), and IWA-6350. These ASME Section XI paragraphs address preparation and retention of

various ASME Section XI records such as Form NIS-2, IWA-4160 verification of acceptability evaluations, IWA-4311 evaluations, Repair/Replacement Plans, and reconciliation documentation. In lieu of these ASME Section XI forms and evaluations, the following repair/replacement activity records shall be retained in accordance with SNC's Owner-defined program for performing repair/replacement activities on LSS items.

- Repair/replacement activity documentation.
- Evaluations of LSS items that do not comply with requirements of the applicable Construction Code, standard, specification, and/or design specification. See also paragraph 5.2.E.12 above.
- Evaluations and documentation of design and configuration changes including material changes.

In addition to the above, SNC will also revise applicable HNP Units 1&2, FNP Units 1&2 and VEGP 1&2 licensing basis documents (e.g., Safety Analysis Report), as appropriate, to identify systems, subsystems, or individual items that have been categorized as LSS and address alternative treatment requirements. Changes to licensing basis documents will be performed in accordance with 10 CFR 50.59.

F. Conclusion

ASME Code Case N-752 specifies requirements for performing risk-informed categorization and treatment for performing repair/replacement activities on Class 2 and 3 pressure retaining items or associated supports. The ASME Code Case N-752 categorization process provides a comprehensive methodology for determining the safety significance of items – HSS or LSS. This categorization process is technically identical to that approved by the NRC under relief request ANO2-R&R-004, Revision 1 (Reference 8.20). Repair/replacement activities performed on items determined to be HSS as well as uncategorized must continue to comply with their applicable nuclear special treatment requirements (e.g., Quality Assurance requirements, Repair/Replacement per ASME Section XI requirements, etc.). Repair/replacement activities performed on LSS items may comply with alternative treatment requirements that are defined by the Owner but must comply with all provisions of paragraph -1420 of ASME Code Case N-752. SNC's proposed treatment requirements, as described herein, meet these criteria, and provide reasonable confidence that LSS systems and items remain capable of performing their safety-related functions under design basis conditions. Finally, categorization and treatment requirements of Code Case N-752 applicable to repair/replacement activities are consistent with NRC requirements specified in 10 CFR 50.69.

6. Duration of Proposed Alternative

The duration of this proposed alternative for each SNC sites' current ASME Section XI 10-year inservice inspection interval, as shown below:

Plant	ISI Interval	ASME Section XI Edition/Addenda (Reference 8.1)	Interval Start	Interval Scheduled End
FNP, Units 1 and 2	5	2007 Edition/2008 Addenda	12/01/2017	11/30/2027
HNP, Units 1 and 2	5	2007 Edition/2008 Addenda	01/01/2016	12/31/2025
VEGP, Units 1 and 2	4	2007 Edition/2008 Addenda	05/31/2017	05/30/2027

7. Precedent

1. Entergy Operations, Inc., Arkansas Nuclear One Units 1 and 2 Request for Relief No. EN-20-RR-001, submitted May 27, 2020 (ML20148M343), approved May 19, 2021 (ML21118B039).
2. Entergy Operations, Inc., Arkansas Nuclear One, Units 1 and 2 - Reduction of Commitment to the Entergy Operation's Quality Assurance Program Manual Description (ML20300A324), Approved May 19, 2021 (ADAMS Accession No. ML21132A279).
3. Duke Energy Carolinas, LLC., Oconee Nuclear Station, Units 1, 2, and 3, Authorization of Alternative to use RR-22-0174, Request for Alternative in Accordance with 10 CFR 50.55a(z)(1) to Use ASME Code Case N-752, "Risk-Informed Categorization and Treatment for Repair/Replacement Activities in Class 2 and 3 Systems Section XI, Division 1" Submitted July 27, 2022 (ML22208A031), and approved December 13, 2023 (ML23262A967).
4. Entergy Operations, Inc., Grand Gulf Nuclear Station, Unit 1; River Bend Station, Unit 1; and Waterford Steam Electric Station, Unit 3; "Relief Request Number EN-RR-22-001 – Proposed Alternative to Use ASME Code Case N-752, Risk-Informed Categorization and Treatment for Repair/ Replacement Activities in Class 2 and 3 Systems, Section XI, Division 1" Submitted June 30, 2022 (ML22181B114), and approved May 30, 2024 (ADAMS Package No. ML24151A236)

8. References

1. American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Section XI, 2007 Edition with the 2008 Addenda.
2. RIS 2004-12, Clarification on use of Later Editions and Addenda to the ASME OM Code and Section XI (ML042090436).
3. SNC Letter, GEN-ISI-ALT-2020-02, to NRC, "Request to Use a Provision of a Later Edition of the ASME Boiler and Pressure Vessel Code, Section XI," dated August 19, 2020 (ML20232D191).
4. Safety Evaluation by The Office of Nuclear Reactor Regulation, "Request to Use a Provision of a Later Edition of The American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI," dated August 26, 2020 (ML20234A332).

5. SNC Letter, VEGP-ISI-ALT-04-06, to NRC, “Request to Use a Provision of a Later Edition of the ASME Boiler and Pressure Vessel Code, Section XI,” dated July 31, 2020 (ML20213C712).
6. Safety Evaluation by The Office of Nuclear Reactor Regulation, “Request to Use a Provision of a Later Edition of The American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI,” dated August 6, 2020 (ML20216A399).
7. SNC Letter, GEN-ISI-ALT-2023-01, “Request to Use a Provision of a Later Edition of The American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI,” dated April 12, 2023 (ML23102A123).
8. Safety Evaluation by The Office of Nuclear Reactor Regulation, “Request to Use a Provision of a Later Edition of The American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI,” dated August 2, 2023 (ML23202A112).
9. 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, And Components for Nuclear Power Reactors," USNRC, 69 FR 68047, Nov. 22, 2004.
10. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.201, Revision 1, "Guidelines for Categorizing Structures, Systems, And Components in Nuclear Power Plants According to Their Safety Significance," dated May 2006.
11. Nuclear Energy Institute NEI 05-04, "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard," Revision 3, November 2009.
12. Nuclear Energy Institute NEI 05-04/07-12/12-06 Appendix X, “Close Out of Facts and Observations (F&Os)," February 2017 (ML17086A431).
13. Letter from J. Giitter and M.J. Ross-Lee (US NRC) to G. Krueger (NEI), “U.S. Nuclear Regulatory Commission Acceptance on Nuclear Energy Institute Appendix X To Guidance 05-04, 07-12, And 12-13, Close-Out of Facts And Observations (F&Os),” May 2017 (ML17079A427).
14. Memorandum from NRC Risk-Informed Steering Committee (US NRC) to S. Rosenberg (US NRC), “U.S. Nuclear Regulatory Commission Staff Expectations for An Industry Facts and Observations Independent Assessment Process," April 2017 (ML17097A275).
15. ASME/ANS RA-S-2009, Addenda to ASME/ANS RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, February 2009.
16. Electric Power Research Institute (EPRI) Report 3002013018, "Human Reliability Analysis (HRA) for Diverse and Flexible Mitigation Strategies (FLEX) and Use of Portable Equipment: Examples and Guidance," November 30, 2018.
17. RA-Sb-2005, Addenda to ASME RA-S-2002 Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications, American society of Mechanical Engineers, New York, New York, December 2005.
18. Regulatory Guide 1.200, Revision 1, An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities, January 2007.
19. ASME/ANS RA-Sa-2009, Addenda to ASME/ANS RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, February 2009.

20. Entergy Letter to NRC dated April 17, 2007, "Request for Alternative ANO2-R&R-004, Revision 1, Request to Use Risk-Informed Safety Classification and Treatment for Repair/Replacement Activities in Class 2 and 3 Moderate Energy Systems," (ML071150108) as supplemented by letters dated August 6, 2007 (ML072220160), February 20, 2008 (ML080520186), and January 12, 2009 (ML090120620).

21. Safety Evaluation Report (SER) by the Office of Nuclear Reactor Regulation "Approval of Request for Alternative ANO2-R&R-004, Revision 1, Request to Use Risk-Informed Safety Classification and Treatment for Repair/Replacement Activities in Class 2 and 3 Moderate and High Energy Systems," dated April 22, 2009 (ML090930246).