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

May 14, 2020

Ms. Cheryl A. Gayheart
Regulatory Affairs Director
Southern Nuclear Operating Co., Inc.
3535 Colonnade Parkway
Birmingham, AL 35243

SUBJECT: VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2 – AUDIT PLAN FOR RELIEF REQUEST INSERVICE INSPECTION ALTERNATIVE VEGP-ISI-ALT-04-04 (EPID L-2019-LLR-0109)

Dear Ms. Gayheart:

By letter dated December 11, 2019 (Agencywide Documents and Access Management System (ADAMS) Accession No. ML19347B105), Southern Nuclear Operating Company (SNC, the licensee), submitted to the U.S. Nuclear Regulatory Commission (NRC), a proposed alternative to the inservice inspection (ISI) requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) for the steam generator (SG) main steam outlet nozzle-to-vessel welds and SG feedwater nozzle-to-vessel welds and nozzle inside radius (NIR) sections of the Vogtle Electric Generating Plant (Vogtle), Units 1 and 2.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Paragraph 50.55a(z)(1), the licensee proposed to increase the ISI interval for the subject components to 30 years, from the current ASME Code Section, Section XI requirement of 10 years. The regulation 10 CFR 50.55a(z)(1) requires the licensee to demonstrate that the proposed alternative provides an acceptable level of quality and safety. The licensee included in its submittal non-proprietary Electric Power Research Institute (EPRI) Report No. 3002014590, "Technical Bases for Inspection Requirements for PWR Steam Generator Feedwater and Main Steam Nozzle-to-Shell Welds and Nozzle Inside Radius Sections," April 2019 (see letter dated December 11, 2019, ADAMS Accession No. ML19347B107) as the technical basis for the proposed alternative. In its request, the licensee included an applicability evaluation of the EPRI technical basis report to Vogtle, Units 1 and 2.

The EPRI technical basis report contains probabilistic fracture mechanics (PFM) analyses that the licensee is citing, in part, as the technical basis for extending the ISI interval for the subject components from 10 to 30 years. The PFM analyses were performed using the software, **PR**obabilistic **O**pti**M**ization of **I**n**S**p**E**ction (**PROMISE**) Version 1.0, developed by Structural Integrity Associates to analyze the SG nozzle-to-vessel welds and NIR sections.

The NRC staff determined it would need to audit PROMISE to verify if it properly implements PFM principles and that it has undergone adequate verification and validation.

The NRC staff has prepared the enclosed audit plan for **PROMISE** per the NRC Office of Nuclear Reactor Regulation (NRR) Office Instruction LIC-111, Revision 1 (ADAMS Accession No. ML082900195).

NRR's Office Instruction LIC-111 states that a regulatory audit is a planned, licensed or regulation-related activity that includes the examination of primarily non-docketed information with the intent to gain understanding, to verify information, or to identify information that will require docketing to support the basis of the licensing or regulatory decision.

The audit will take place virtually for approximately two days between June 29 through July 1, 2020.

If you have any questions, please contact me at (301) 415-3100 or via email at John.Lamb@nrc.gov.

Sincerely,

/RA/

John G. Lamb, Sr. Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-424 and 50-425

Enclosure: Audit Plan

cc: Listserv

SUBJECT: VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2 – AUDIT PLAN FOR RELIEF REQUEST INSERVICE INSPECTION ALTERNATIVE VEGP-ISI-ALT-04-04 (EPID L-2019-LLR-0109) DATED MAY 14, 2020

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OFFICE	DORL/LPL2-1/PM*	DORL/LPL2-1/LA*	DORL/LPL2-1/BC*	DORL/LPL2-1/PM*
NAME	JLamb*	KGoldstein*	MMarkley*	JLamb
DATE	5/7/2020	5/13/2020	5/13/2020	5/14/2020

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REGULATORY AUDIT PLAN

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FOR THE **PROMISE** VERSION 1.0

PROBABILISITIC FRACTURE MECHANICS SOFTWARE

REGARDING RELIEF REQUEST VEGP-ISI-ALT-04-04

VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-424 AND 50-425

EPID NO. L-2019-LLR-0109

1.0 BACKGROUND

By letter dated December 11, 2019 (Agencywide Documents and Access Management System (ADAMS) Accession No. ML19347B105), Southern Nuclear Operating Company (SNC, the licensee), submitted to the U.S. Nuclear Regulatory Commission (NRC), a proposed alternative to the inservice inspection (ISI) requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) for the steam generator (SG) main steam outlet nozzle-to-vessel welds and SG feedwater nozzle-to-vessel welds and nozzle inside radius (NIR) sections of the Vogtle Electric Generating Plant (Vogtle), Units 1 and 2.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Paragraph 50.55a(z)(1), the licensee proposed to increase the ISI interval for the subject components to 30 years, from the current ASME Code Section, Section XI requirement of 10 years. The regulation 10 CFR 50.55a(z)(1) requires the licensee to demonstrate that the proposed alternative provides an acceptable level of quality and safety. The licensee included in its submittal non-proprietary Electric Power Research Institute (EPRI) Report No. 3002014590, "Technical Bases for Inspection Requirements for [Pressurized-Water Reactor] PWR Steam Generator Feedwater and Main Steam Nozzle-to-Shell Welds and Nozzle Inside Radius Sections," April 2019 (see letter dated December 11, 2019, ADAMS Accession No. ML19347B107) as the technical basis for the proposed alternative. In its request, the licensee included an applicability evaluation of the EPRI technical basis report to Vogtle, Units 1 and 2.

The EPRI technical basis report contains probabilistic fracture mechanics (PFM) analyses that the licensee is citing, in part, as the technical basis for extending the ISI interval for the subject components from 10 to 30 years. The PFM analyses were performed using the software, **PR**obabilistic **O**pti**M**ization of **I**n**S**p**E**ction (**PROMISE**) Version 1.0, developed by Structural Integrity Associates to analyze the SG nozzle-to-vessel welds and NIR sections.

2.0 REGULATORY AUDIT BASIS

The NRC Office of Nuclear Reactor Regulation (NRR) Office Instruction LIC-111 (ADAMS Accession No. ML082900195) states that a regulatory audit is a planned, licensed or regulation-related activity that includes the examination of primarily non-docketed information

with the intent to gain understanding, to verify information, or to identify information that will require docketing to support the basis of the licensing or regulatory decision.
The NRC staff does not have similar software to verify the PFM analyses performed with **PROMISE** Version 1.0. Therefore, the staff determined it needs to examine the **PROMISE** Version 1.0 software to verify that it properly implements PFM principles and has undergone sufficient Verification and Validation (V&V) per the software quality assurance program.

3.0 REGULATORY AUDIT SCOPE OR METHODOLOGY

The focus of the NRC staff's audit is on the PFM software, **PROMISE** Version 1.0. The information the NRC staff will audit is provided in the next section.

4.0 INFORMATION AND OTHER MATERIAL NECESSARY FOR THE AUDIT

The NRC staff will review the following information on **PROMISE** Version 1.0. If SNC or its contractors identify other relevant documents or analyses during the audit, the NRC staff will also review those documents or analyses.

1. V&V documents

    a) Show the listing of V&V cases/tests that were conducted in the software V&V plan (SVVP) and V&V reports (references 73 and 74 of the EPRI technical basis report). What were the validation activities? Just benchmarking?
    b) Show the V&V documents for the nozzle corner crack stress intensity factors (SIFs) from TIFFANY software and how these SIFs are input into **PROMISE** Version 1.0.
    c) Show user's manual (typically called for by software quality assurance guidance).

2. Technical aspects of **PROMISE** Version 1.0

    a) *Uncertainties and sampling method*
        i. Clarify if one realization is equivalent to one nozzle.
        ii. Table 8-7 of the EPRI technical basis report indicates that there were 10 million aleatory realizations and 1 epistemic realization? How would results change if a combination of aleatory and epistemic realizations were implemented?
        iii. Show an input deck to clarify which parameters are random and which parameters are constant.
        iv. Sampling loop and time loop
            1. Explain and show how the sampling and time loops are implemented. For each realization, does the code loop through time (say for one month or one year), then proceed through the number of realizations (say 10 million), then report the probability of failure for that time step?
            2. How is crack depth (and length) updated for the next and subsequent time steps?
    b) Flaw distribution
        i. To understand the flaw distribution model used in **PROMISE** Version 1.0, show the values of depth and length (and density? or is density a single density instead of a *distribution* of densities?) for several realizations.
        ii. Section 8.2.4.3.5 of EPRI technical basis report: explain "single flaw in each case."

iii. Section 8.2.4.3.6 of EPRI technical basis report: why did the PVRUF [Pressue Vessel Research User's Facility] distribution end up with slightly lower probability of failure value? It was expected to result in higher probability of failure value. Show a comparison of the PVRUF and Marshall flaw distribution inputs. (Section 8.2.2.2 of the report states that PVRUF was slightly more conservative than the Marshall.)
iv. Show SIA [Structural Integrity Associates, Inc.] calculation package 1700313.301 for derivation of depth and length.
v. Clarify: Page 8-31 of the EPRI technical basis report mentions "a distribution derived by SI[A] based on data from NUREG/CR-6817." Is this distribution the same as Equation 8-1 of the report? Can you change the depth parameter in Equation 8-1?

c) *Probability of Detection (POD) and Inservice Inspection (ISI)*
i. The EPRI technical basis report does not discuss the interaction between POD [probability of detection] curve and ISI. Show how POD and ISI are implemented into the **PROMISE** Version 1.0 code by showing the software requirements for this feature and running some example problems. Refer to Sections 8.2.2.3 and 8.2.2.8 of the EPRI technical basis report.

d) *Fracture toughness*
i. Discuss fracture toughness and the standard deviation used in the base and sensitivity cases discussed in the EPRI technical basis report. Show the runs for FEW-P3A for the base case and those in the sensitivity study for fracture toughness.

e) *Stress input*
i. Show how stresses (due to internal pressure, thermal transients, residual stress) are input into **PROMISE** Version 1.0.
ii. For thermal transients, were the maximum and minimum stress distributions selected, or were SIFs developed throughout the time history of a transient and the maximum and minimum SIFs selected and input into the code? Show this calculation.
iii. Generally speaking, compared to the BWRVIP-05 [Boiling Water Reactor Vessel and Internals Program] work for RPV [Reactor Pressure Vessel] beltline welds for a low temperature isothermal pressure transient, the probability of failure results for the subject components in the EPRI technical basis report are at least 3 orders of magnitude lower. Again, overall this makes sense given that (per the EPRI report) the temperature of the applicable transients is in the upper range of ASME K_{IC} curve. But what about heatup/cooldown in which temperatures can go quite low? Show the SIF or stress history with the temperature history to ensure maximum SIF or stress does not occur at low temperatures.

f) *Convergence*
i. Show a plot of failure probability versus number of realizations for a low probability case and a high probability case.

g) *Modeling dependencies*
i. Show model dependencies between the inputs, if any, considered in the sensitivity studies in Section 8.2.4.3 of the EPRI technical basis report (example: interaction between POD and ISI).

h) Code execution and input deck
i. Show code execution and input deck for one or two cases presented in the EPRI technical basis report. Clarify which parameters are random and which are constant.

3. Cases reported in the EPRI technical basis report

    a) In addition to the benchmark cases shown in Table 8-5 of the EPRI technical basis report, show a benchmark run with VIPERNOZ for the FEW-P3A base case in Table 8-9.
    b) Show underlying calculations and/or calculation packages for the base cases in Tables 8-8 and 8-9 of the EPRI technical basis report.
    c) Show calculation of values in Table 8-26 of the EPRI technical basis report (probability of leak for 80 years) from the plot in Figure 8-12 (probability of leak versus time).
    d) Show underlying calculations and/or calculation packages for the cases in Table 8-28 or Table 8-29 of the EPRI technical basis report.

5.0 TEAM ASSIGNMENTS

The audit team consists of Mr. John Lamb, Senior Project Manager for Vogtle, Units 1 and 2, in the Division of Operating Reactor Licensing; Dr. David Rudland, Senior Technical Advisor in the Division of New and Renewed Licenses; and Mr. David Dijamco, Technical Reviewer in the Vessels and Internal Branch, Division of New and Renewed Licenses.

6.0 LOGISTICS

The audit will occur via teleconference or face-to-face meeting (see request in the next section) for approximately two days between on **June 29 through July 1, 2020.** The NRC staff plans to hold a brief entrance meeting at the beginning of the audit and a brief exit meeting at the close of the audit.

7.0 SPECIAL REQUESTS

The NRC staff requests the licensee and its contractors to have the information listed in Section 4 readily available and accessible for the NRC staff's review. The NRC staff also requests that cognizant licensee and contractor staff be available to answer any questions from the NRC staff and to track NRC staff questions during the audit. Because of the circumstances surrounding COVID-19, the NRC staff requests that the licensee and its contractors to be flexible in how they will present the audit materials. Given the current situation, presenting the audit materials through a teleconferencing application with capability for sharing content, such as WebEx or Skype, is the best method for all parties. The NRC staff, however, requests that the licensee and its contractors remain open to a face-to-face meeting if the circumstances allow it and if deemed desirable by all parties.

8.0 DELIVERABLES

An audit summary will be prepared within 90 days of the completion of the audit. If information evaluated during the audit is needed to support a regulatory decision, the NRC staff will identify it in a formal request for additional information (RAI). The NRC staff will provide the RAIs to the licensee in separate docketed correspondence.