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

September 15, 2016

Mr. C. R. Pierce
Regulatory Affairs Director
Southern Nuclear Operating Co., Inc.
P.O. Box 1295, Bin 038
Birmingham, AL 35201-1295

SUBJECT: VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2 – REQUEST FOR ADDITIONAL INFORMATION (CAC NOS. MF8061 AND MF8062)

Dear Mr. Pierce:

By letter dated August 4, 2016, Southern Nuclear Operating Company, Inc. (SNC) proposed an inservice inspection alternative for the Vogtle Electric Generating Plant (VEGP), Units 1 and 2. SNC proposes to eliminate the reactor pressure vessel threads in flange examination requirement for the remainder of the inservice inspection intervals.

The U.S. Nuclear Regulatory Commission staff has determined that additional information is needed to continue its review. We request that SNC respond to the enclosed request for additional information at its earliest opportunity.

If you have any questions, please contact me at 301-415-1493 or by e-mail at Robert.Martin@nrc.gov.

Sincerely,

Bob Martin

Bob Martin, Senior 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:
Request for Additional Information

REQUEST FOR ADDITIONAL INFORMATION

ALTERNATIVE REQUEST VEGP-ISI-ALT-11

REACTOR PRESSURE VESSEL FLANGE BOLT HOLE THREAD EXAMINATION

VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

DOCKET NOS. 50-424 AND 50-425

By application dated August 4, 2016 (Agencywide Documents Access and Management System Package Accession No. ML16221A072), Southern Nuclear Operating Company, Inc. (SNC) proposed Alternative VEGP-ISI-ALT-11, Version 2.0, for the Vogtle Electric Generating Plant (VEGP), Units 1 and 2. Alternative VEGP-ISI-ALT-11, Version 2.0, proposes to eliminate the reactor pressure vessel (RPV) threads in flange examination requirement for the remainder of the fourth inservice inspection (ISI) intervals as an alternative to certain requirements of Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code for ISI of RPV components. The U.S. Nuclear Regulatory Commission (NRC) staff is reviewing the submittal and has determined that additional information is needed as follows.

VEGP-ISI-ALT-11, Version 2.0 (Enclosure 1 of the Application)

RAI-1

Table 1 compares the VEGP, Units 1 and 2, plant-specific data with the values used in the Electric Power Research Institute (EPRI) Nondestructive Evaluation Report (the EPRI report). Please provide the plant-specific heatup rate for the NRC staff to compare with the assumed 100 degrees Fahrenheit (°F)/hour heatup rate that was used in the EPRI report.

RAI-2

Page E1-3 states, "The preload was calculated as detailed in VEGP Units 1 and 2 RPV manual." This statement is then followed by the calculation of the bounding preload taken from the EPRI report. Please provide the preload that was calculated in the VEGP, Units 1 and 2, RPV manual and applied to both units to date. In addition, does the operating experience of VEGP, Units 1 and 2, indicate, for various reasons, that the bolt-up contingencies exceeded 1.1?

Enclosure

RAI-3

Figure 4 (referenced on page E1-4 of the application) is missing. Please confirm that the missing figure is meant to be Figure 6-8 of the EPRI report, or supply the missing figure.

EPRI Nondestructive Evaluation Report (Enclosure 3 of the Application)

RAI-4

Table 2-1 of the EPRI report summarized threads in flange components at seven domestic pressurized-water reactor plants. The number of threads per foot for each reported unit is not listed. Please provide this information and confirm whether this parameter is approximately the same among different units (including the VEGP units). If not, please assess the impact of variation of this parameter on the generic analyses (stress analysis plus fracture mechanics analysis) in the EPRI report.

RAI-5

To better assess the results from the finite element method (FEM) model reported in the EPRI report, please clarify the following:

1. Symmetry of both faces of the FEM model implies that the peaks and valleys of the threads are concentric circles. Please assess the impact of neglecting the forces due to the fact that these circular planes are not perpendicular to the thread axis.

2. Section 6.1.1 states, "To simulate the RPV head, the top of the cladding surface is fixed in the axial direction, and the bottom of the flange is coupled axially to simulate the rest of the vessel."

    - Explain how the loads due to pressure and thermal transients are applied to the model. Are they applied uniformly to the bottom and inner surfaces of the model?

    - Explain the validation that was taken to ensure that the FEM model could generate realistic stress results, considering assumptions regarding the axial extent of the model, the boundary conditions not described in the quote above, and the load distribution (uniform, linear, or higher order) over the surfaces.

3. Section 6.1.2 states, "This transient typically consists of a steady 100°F/hour ramp up to the operating temperature." Please explain why this heatup transient is more limiting than the cooldown transient.

4. Section 6.2.1 did not provide sufficient details for the FEM model for the applied stress intensity factor (K) determination. Please:

   - Confirm that the FEM model for the applied K determination is the same as the FEM model for the stress determination, except for the crack tip elements as shown in Figure 6-8.

   - Explain what "ellipsoidal flaw shape around the circumference of the flange" means. (Question 1 seems to imply that flaws are of circular shapes.)

5. Section 6.2.1 states, "When preload is not being applied, the bolt, bolt threads, and flange threads are not modelled." The NRC staff noted that the loads that were used to generate results in Table 6-1 all include preload. Please provide information for the cases that were simulated without preload and without flange threads, and explain how the results from these simulations (without preload and without flange threads) affect the results reported in Table 6-1.

6. Justify use of the fracture toughness (K_{IC}) at the upper shelf energy temperature in the linear elastic fracture mechanics analysis, considering that the critical combination of applied K and K_{IC} may occur at a much lower temperature.

7. Section 6.2.3 states, "The resulting crack growth as calculated by pc-CRACK is negligible." Please provide this value.

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Bob Martin, Senior Project Manager
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Docket Nos. 50-424 and 50-425

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