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U.S. Nuclear Regulatory Commission
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Southern Nuclear Operating Company
Vogtle Electric Generating Plant Units 3 and 4
Response to Request for Additional Information Regarding Application of VT-1 Visual Examination Methodology for Preservice Inspection of the
Reactor Vessel Nozzle Inner Radius Sections (VEGP 3&4-PSI-ALT-07S2)

Ladies and Gentlemen:

By letter dated July 6, 2017, Southern Nuclear Operating Company (SNC) submitted a request for an alternative in accordance with 10 CFR 50.55a for preservice inspection of the Reactor Vessel Nozzle Inner Radius Sections [ML17192A125]. On October 30, 2017, the Nuclear Regulatory Commission (NRC) staff issued a draft request for additional information (RAI) [ML17303A270]. The responses to the RAI questions were provided on December 8, 2017 [ML17342A826]. The NRC provided a second draft RAI on February 6, 2018 and additional clarification was provided during the February 15, 2018 public meeting. On February 23, 2018, the RAI was issued [ML18054A672]. The responses to the RAI questions are included in Enclosure 4.

The supplemental information provided in this letter does not impact the scope or conclusions of the original alternative.

This letter contains no regulatory commitments. This letter has been reviewed and determined not to contain security related information.

Should you have any questions, please contact Mr. Corey Thomas at (205) 992-5221.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 8th of March 2018.

Respectfully submitted,

Brian H. Whitley
Director, Regulatory Affairs
Southern Nuclear Operating Company

Enclosures: 1) - 2) (previously submitted with the original code alternative, VEGP 3&4-PSI-ALT-07, in SNC letter ND-17-1121)

3) (previously submitted in VEGP 3&4-PSI-ALT-07S1, in SNC letter ND-17-2032)

4) Response to Request for Additional Information Regarding Application of VT-1 Visual Examination Methodology for Preservice Inspection of the Reactor Vessel Nozzle Inner Radius Sections (VEGP 3&4-PSI-ALT-07S2)

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**Southern Nuclear Operating Company**

**ND-18-0279**

**Enclosure 4**

**Vogtle Electric Generating Plant (VEGP) Units 3 and 4**

**Response to Request for Additional Information Regarding Application of VT-1 Visual Examination Methodology for Preservice Inspection of the Reactor Vessel Nozzle Inner Radius Sections (VEGP 3&4-PSI-ALT-07S2)**

**(This Enclosure consists of 4 pages, including this cover page)**

The following are questions provided by the NRC Staff on February 23, 2018 (ADAMS Accession Number ML18054A672) regarding the review of Southern Nuclear Operating Company (SNC) Proposed Alternative VEGP 3&4-PSI-ALT-07 in accordance with 10 CFR 50.55a(z)(1) - Application of VT-1 Visual Examination Methodology for Preservice Inspection of the Reactor Vessel Nozzle Inner Radius Sections (ADAMS Accession Number ML17303A270) submitted on October 30, 2017 and Response to Request for Additional Information (RAI) Regarding Application of VT-1 Visual Examination Methodology for Preservice Inspection of the Reactor Vessel Nozzle Inner Radius Sections (VEGP 3&4-PSI-ALT-07S1, ADAMS Accession Number ML17342A826) submitted on December 8, 2017. Responses are provided below for each RAI question.

**RAI Question 11:**

In RAI Question 1.a, the NRC staff requested confirmation that the alternative examinations will be performed in accordance with the ASME Code as conditioned in § 50.55a of Title 10 of the *Code of Federal Regulations* (10 CFR). The NRC staff requests confirmation that the proposed remote VT-1 visual examination will be performed in accordance with ASME Code, Section XI, as conditioned in 10 CFR 50.55a, which requires that a visual examination performed instead of an ultrasonic examination has a magnification that has a resolution sensitivity to resolve 0.044 inch (1.1 mm) lower case characters without an ascender or descender (*e.g.*, a, e, n, v).

**SNC Response to RAI Question 11:**

The proposed remote VT-1 visual examination will be performed in accordance with ASME Code, Section XI, as conditioned in 10 CFR 50.55a, which requires that a visual examination performed instead of an ultrasonic examination has a magnification that has a resolution sensitivity to resolve 0.044 inch (1.1 mm) lower case characters without an ascender or descender (*e.g.*, a, e, n, v).

**RAI Question 12:**

In RAI Question 1.b.ii, the NRC staff requested a description of the alternative examinations. The NRC staff requests confirmation that the proposed remote VT-1 visual examination will cover essentially 100 percent of the examination volume for the inlet, outlet, and DVI nozzles as defined in ASME Code, Section XI, Figure IWB-2500-7(b). If not, then provide sketches or some other means detailing the examination volume for the inlet, outlet, and DVI nozzles that will be covered by the proposed remote VT-1 visual examination.

**SNC Response to RAI Question 12:**

The proposed remote VT-1 visual examination will cover essentially 100 percent of the ASME Code, Section XI, Figure IWB-2500-7(b) section M-N surface area for the inlet, outlet, and DVI nozzles.

**RAI Question 13:**

In the letter dated December 8, 2017, SNC stated that the stresses at the nozzle corner region for the Advanced Passive 1000 (AP1000) are similar to the operating fleet. In order to determine whether the operational experience from the current operating fleet applies, the NRC staff requests the following information:

a. The design stress state and operating stresses at the nozzle corner region from the AP1000 reactor pressure vessel design report for each inlet, outlet, and DVI nozzle. In addition, these stresses should specifically include any thermal cycling and stratification loading for each inlet, outlet, and DVI nozzle.

b. An evaluation that compares the design stress state and operating stresses at the nozzle corner region for each inlet, outlet, and DVI nozzle of the AP1000 provided in (a) above to the design stress state and operating stresses in the nozzle corner region of similar designed nozzles for the current operating fleet. This should include a discussion of why each inlet, outlet, and DVI nozzle is similar to the operating fleet nozzle designs.

**SNC Response to RAI Question 13:**

The pressure stresses in the AP1000 nozzle corner regions are similar to operating vessels of a similar size and geometry. However, the stresses from the thermal transients would be different for the AP1000 design. It is for this reason that the fracture analyses documented in Section 5.0 of the Alternative Request (ML17192A125) were performed using AP1000 specific stress data. The fracture evaluation results discussed in the technical basis for Code Case N-648-1 (Reference 1) were calculated using stresses in the nozzle corner regions for the operating fleet. Thus, the fracture analyses results for the AP1000 provided in the Alternative Request are directly comparable to those for the operating fleet. Conversely, the stress information requested by the NRC staff is not easily obtained, nor would it provide a better comparison than the fracture analyses results already provided. Stress data by itself would not provide context or acceptance criteria without also performing the facture analyses to determine acceptable flaw sizes considering the operating life of the plant. Therefore, the following text which further explains the fracture analyses results documented in the Alternative Request (ML17192A125), is provided as an alternative to the data requested by the NRC staff:

The technical basis for Code Case N-648-1 (Reference 1) included a fracture evaluation that demonstrated tolerance for flaws greater than 3 inches in the reactor vessel inner radii for the operating fleet with a 40 year design life. This result provided assurance that visual examination methods would be sufficient for detecting flaws within the inner radii that are well below the acceptable size calculated using ASME Section XI methods. Likewise, the analyses presented in Section 5.0 of the Alternative Request (ML17192A125) reached a similar conclusion: that the AP1000 nozzle inner radii are tolerant of flaws greater than 3 inches in depth, considering a 60 year design life (see Table 2 of ML17192A125). The main difference between the two analyses is that the allowable flaw sizes for the operating fleet were calculated using linear elastic fracture mechanics (LEFM) only, but the analyses results presented for the AP1000 used both LEFM and elastic plastic fracture mechanics (EPFM) methods. Both calculations were performed using ASME Section XI methodology.

The fracture evaluation guidelines of ASME Section XI, Appendix A (Reference 2) were used in the initial approach for evaluation of flaw tolerance for AP1000 nozzle corners. The Appendix A

methods are based on LEFM and are more conservative than EPFM methods such as those provided in Section XI Appendix K or Code Case N-749 (Reference 3). This more conservative LEFM approach is similar to the calculation methods used to determine the allowable flaw sizes for the operating fleet in (Reference 1). The LEFM results in the Alternative Request are smaller compared to the 3 inch allowable flaw size calculated for the operating fleet. As shown in Table 2 of the Alternative Request, the fatigue crack growth (FCG) results show that the maximum initial flaw sizes that will not grow beyond the allowable end of evaluation period flaw sizes are less than 1 inch considering 10 and 60 year operating periods. Although the allowable flaw sizes are less than 3 inches, they are considered to be acceptable because they are less than the flaw size that may have been placed into service considering the acceptance criteria of NB-2545 for magnetic partial examination (0.094 inch depth – see Section 5 of the Alternative Request).

Because the LEFM flaw evaluations produced notably smaller allowable flaws for the AP1000 nozzle designs than for the operating fleet, a second set of calculations were performed using the EPFM method provided in Code Case N-749. The Code Case is a valid approach for evaluating the AP1000 nozzles for flaw tolerance as discussed in the Alternative Request and previous RAI responses. The EPFM fracture analysis results listed in Table 2 of the Alternative Request show that the allowable flaw sizes for the inlet, outlet and DVI nozzles are all over 3 inches, which is the same conclusion reached in (Reference 1) for the operating fleet nozzles. This conclusion also provides assurance that flaws large enough to be of concern within the AP1000 nozzle corner regions would be detected by the VT-1 examination proposed in the Alternative Request.

Given this discussion, the NRC staff is requested to consider the results presented in Table 2 of the Alternative Request (ML17192A125) as a comparison to the results presented in the technical basis for Code Case N-648-1 (Reference 1) in lieu of the stress data requested in the RAI. The results in Table 2 of the Alternative Request (ML17192A125) provide a direct comparison of ASME Section XI fracture mechanics results for the operating fleet and the AP1000. The final conclusion reached is that both the operating fleet and the AP1000 design are tolerant of flaws greater than 3 inches in depth within the reactor vessel inner radii regions, which provides assurance that any flaws detected by visual examination would be well below the acceptable size based on ASME Section XI fracture evaluation methods.

References:

1. W. H. Bamford, D. Kurek, B. A. Bishop, N. Closky, G. Stevens, L. Becker, S. Ranganath, "Technical Basis for Elimination of Reactor Vessel Nozzle Inner Radius Inspections," 2001 Pressure Vessel and Piping Conference Proceeding, American Society of Mechanical Engineers.

2. ASME Boiler and Pressure Vessel Code, Section XI, 2007 Edition with 2008 Addenda, "Rules for Inservice Inspection of Nuclear Power Plant Components," American Society of Mechanical Engineers.

3. ASME Boiler and Pressure Vessel Code, Case N-749: Alternative Acceptance Criteria for Flaws in Ferritic Steel Components Operating in the Upper Shelf Temperature Range Section XI, Division 1, March 16, 2012.