B. H. Whitley
Director
Regulatory Affairs

Southern Nuclear Operating Company, Inc.
42 Inverness Center Parkway
Birmingham, AL 35242
Tel 205.992.7079
Fax 205.992.5296

December 8, 2017

Docket Nos.: 52-025
52-026

ND-17-2032
10 CFR 50.55a

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

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-07S1)

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]. A clarification call was held in a public meeting on November 16, 2017 to provide clarification on the RAI questions. The responses to the RAI questions are included in Enclosure 3.

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 December 2017.

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) 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-07S1)

cc:

Southern Nuclear Operating Company / Georgia Power Company
Mr. S. E. Kuczynski (w/o enclosure)
Mr. M. D. Rauckhorst
Mr. D. G. Bost (w/o enclosure)
Mr. M. D. Meier (w/o enclosure)
Mr. D. H. Jones (w/o enclosure)
Mr. D. L. McKinney (w/o enclosure)
Mr. T. W. Yelverton (w/o enclosure)
Mr. B. H. Whitley
Mr. J. J. Hutto
Mr. C. R. Pierce
Ms. A. G. Aughtman
Mr. D. L. Fulton
Mr. M. J. Yox
Mr. J. Tupik
Mr. W. A. Sparkman
Ms. A. C. Chamberlain
Ms. A. L. Pugh
Mr. J. D. Williams
Document Services RTYPE: VND.LI.L00
File AR.01.02.06

Nuclear Regulatory Commission
Mr. W. Jones (w/o enclosure)
Ms. J. Dixon-Herrity
Mr. C. Patel
Ms. J. M. Heisserer
Mr. B. Kemker
Mr. G. Khouri
Ms. S. Temple
Ms. V. Ordaz
Mr. T.E. Chandler
Ms. P. Braxton
Mr. T. Brimfield
Mr. C. J. Even
Mr. A. Lerch

State of Georgia
Mr. R. Dunn

Oglethorpe Power Corporation
Mr. M. W. Price
Mr. K. T. Haynes
Ms. A. Whaley

Municipal Electric Authority of Georgia
Mr. J. E. Fuller
Mr. S. M. Jackson

Dalton Utilities
Mr. T. Bundros

Westinghouse Electric Company, LLC
Mr. L. Oriani (w/o enclosure)
Mr. G. Koucheravy (w/o enclosure)
Mr. M. Corletti
Mr. M. L. Clyde
Ms. L. Iller
Mr. D. Hawkins
Mr. J. Coward

Other
Mr. S. W. Kline, Bechtel Power Corporation
Ms. L. A. Matis, Tetra Tech NUS, Inc.
Dr. W. R. Jacobs, Jr., Ph.D., GDS Associates, Inc.
Mr. S. Roetger, Georgia Public Service Commission
Ms. S. W. Kernizan, Georgia Public Service Commission
Mr. K. C. Greene, Troutman Sanders
Mr. S. Blanton, Balch Bingham
Mr. R. Grumbir, APOG
NDDocumentinBox@duke-energy.com, Duke Energy
Mr. S. Franzone, Florida Power & Light

Southern Nuclear Operating Company

ND-17-2032

Enclosure 3

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-07S1)

(This Enclosure consists of 10 pages, including this cover page)

The following are questions provided by the NRC Staff 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) on October 30, 2017. Responses are provided below each Request for Additional Information (RAI) question.

<u>RAI Question 1:</u>

1. Pursuant to § 50.55a(g)(3)(ii) of Title 10 of the *Code of Federal Regulations* (10 CFR), ASME Code Class 1, 2, and 3 components (including supports) must meet the PSI requirements set forth in ASME Code Section III or Section XI. ASME Code Section XI, Table IWB-2500-1, Examination Category B-D, Item B3.100, requires a PSI volumetric examination of the reactor vessel nozzle inner radius sections. Your proposed PSI examinations include a liquid penetrant testing (PT) examination prior to the VT-1 visual examination, and an ultrasonic testing (UT) examination prior to the VT-1 but after the PT. Please provide the following information regarding your proposed PSI examinations:

   a. 10 CFR 50.55a provides conditions on ASME Code Section XI examinations including that UT examinations shall be performed by qualified procedures and personnel in accordance with ASME Code Section XI, Appendix VIII (performance based). Confirm that the examinations will be performed in accordance with the ASME Code as conditioned in 10 CFR 50.55a.

   b. If any of the examinations will not be performed in accordance with the ASME Code as conditioned in 10 CFR 50.55a, then provide the following information:

      i. Discuss how the performance of the examinations will not meet the ASME Code and your basis for not performing the examinations in accordance with the ASME Code.

      ii. Describe the alternative(s) to the examination(s) not being performed in accordance with the ASME code as conditioned in 10 CFR 50.55a, including a clear discussion of how the alternative(s) will meet 10 CFR 50.55a(z)(1) by providing an acceptable level of quality and safety.

      iii. Discuss the impacts and acceptability concerning flaw characterization and sizing since there will be no performance-based examination baseline data for comparison purposes if subsequent ISI examination (UT) is necessary when indications are found during the VT-1 inspection.

SNC Response to RAI Question 1(a):

The proposed alternative does not provide provisions to perform examinations in accordance with the ASME Code as conditioned in 10 CFR 50.55a. The proposed alternative allows a VT-1 examination in lieu of the UT requirement of ASME Section XI IWB-2500, Examination Category B-D, Item B3.100. A PT and UT are also proposed to supplement the visual examination. The proposed UT to be performed is in accordance with ASME Section XI, Appendix III; however, the proposed UT does not meet ASME Section XI, Appendix VIII performance demonstration requirements for ultrasonic examination systems.

SNC Response to RAI Question 1(b)(i):

The proposed UT to be performed is in accordance with ASME Section XI, Appendix III; however, the proposed UT does not meet ASME Section XI, Appendix VIII performance demonstration requirements for ultrasonic examination systems. The logic for the VT-1 visual examination is that service-induced flaw mechanisms (fatigue) will be associated with the inner diameter (ID) surface of the cladding, and that the VT-1 examinations are sufficient to detect such mechanisms occurring at the ID surface well before the nozzle experiences degradation of its structural integrity. Performance of an Appendix VIII examination for these nozzles is not warranted for a one time exam when failure mechanisms are readily identified using visual techniques.

SNC Response to RAI Question 1(b)(ii):

This technical basis addresses a VT-1 visual examination approach that includes a deterministic fracture assessment similar to that performed as a basis for Code Case N-648-1. The code alternative provides an acceptable level of quality and safety in accordance with 10 CFR 50.55a(z)(1) because the VT-1 examinations are sufficient to detect service-induced flaw mechanisms (fatigue) occurring at the ID surface well before the nozzle experiences degradation of its structural integrity.

A fracture assessment was performed to determine the maximum initial flaw size that will not grow beyond the allowable end of evaluation period flaw size for the life of the plant (60 years) considering Level A/B/Test conditions which were limiting in comparison to the Level C/D/Test. The allowable end of evaluation period flaw sizes (depths) for the AP1000® inlet, outlet, and direct vessel injection (DVI) nozzles were determined using both linear elastic fracture mechanics (LEFM) and elastic plastic fracture mechanics (EPFM) methods.

For the most limiting case, the DVI nozzle using the LEFM analysis, the acceptance criteria for the VT-1 examinations using the requirements of ASME Section XI are much more limiting than the governing initial flaw depth for each nozzle and would not allow a flaw length that results in an unacceptable flaw depth during the examination period without performance of repair/replacement and regulatory review in accordance with ASME Section XI, IWB-3113 and IWB-3114. For example, for the limiting DVI Nozzle Case, the initial limiting flaw size that would grow to the limiting flaw depth of 0.358″ over a 10-year period is 0.351″. Using a 0.5 flaw depth to length ratio, this corresponds to a 1.14″ flaw length on the surface of the cladding ((0.351″ + 0.22″) / 0.5), given a 0.22″ cladding thickness. The proposed ASME Section XI Table IWB-3512-1 acceptance criteria of 0.144″ for the maximum allowable flaw length detected during the VT-1 is much more stringent than supported by the fracture mechanics analysis of 1.14″ in length.

The VT-1 acceptance criteria are conservative for the limiting LEFM case; however, it is important to note that more realistic EPFM results show that a flaw over 3″ in depth for each reactor vessel nozzle can be tolerated for a 60-year plant life.

Examinations performed during fabrication of the nozzle forgings include magnetic particle in accordance with ASME Section III and UT in accordance with ASME Section V, Article 7 and Article 5. These examinations ensure that examination surfaces have been appropriately prepared for the application of future volumetric examinations. Following deposition of the cladding, a PT examination was performed using the acceptance standards of ASME Code Section III, NB-5350. Following intermediate heat treatment, a UT in accordance with the examination procedure requirements of ASME Section V, using the acceptance standards of Section XI, IWB-3512 was performed with no recordable indications. Following the hydrostatic test, a PT examination was performed using the acceptance standards of ASME Code Section III, NB-5350, with no relevant indications (i.e., an indication greater than 1/16″ long). These examinations, in addition to the proposed UT and visual examinations that are proposed in this alternative, provide assurance that existing flaws are limited in length and flaws do not exist on the cladding inner diameter surface prior to implementing a VT-1 examination. These examinations provide the bases for using the postulated flaw size of 0.16″ (this depth correlates to a 0.32″ flaw length) in the ASME Code Section III, Appendix G analysis. Each reactor vessel satisfies ASME Section III, Appendix G requirements.

SNC Response to RAI Question 1(b)(iii):

The purpose of the examination of nozzle inner radii is to detect fatigue cracking due to operation and service conditions of the component. Note that no fatigue cracking has been reported throughout the operating history of PWR commercial nuclear power plants in nozzle inner radii either ultrasonically or visually. The ability to visually detect fatigue cracks has been demonstrated successfully with probability of detection of 80% or greater.

This data is supported most recently via the joint round robin conducted by the industry Electric Power Research Institute (EPRI) and the research arm of the NRC, Pacific Northwest National Laboratory (PNNL), in a 3-phase joint project. The types of cracking the round robin was attempting to detect were much more challenging than fatigue cracking (Intergranular Stress Corrosion Cracking (IGSCC) & Irradition Assisted Stress Corrosion Cracking (IASCC)) due to the inherent morphology of fatigue cracks versus IGSCC or IASCC. The round robin and previous operating experience clearly demonstrate that detection of cracking is primarily dependent upon the crack opening which would be greater for fatigue cracks.

Visual examination for critical reactor vessel components is routinely conducted industry wide via the Boiling Water Reactor Vessel Internals Program (BWRVIP) for BWRs and Materials

Reliability Program (MRP) for Pressurized Water Reactors (PWRs). The VT-1 method as well as EVT-1 and VT-3 are applied for these critical reactor components.

For the most limiting case, the DVI nozzle using the LEFM analysis, the acceptance criteria for the VT-1 examinations using the requirements of ASME Section XI are much more limiting than the governing initial flaw depth for each nozzle and would not allow a flaw length that results in an unacceptable flaw depth during the examination period without performance of repair/replacement and regulatory review in accordance with ASME Section XI, IWB-3113 and IWB-3114. Based on the above, VT-1 examinations of the nozzle inner radii are adequate for detection of fatigue cracks should they be present. If conditions change for the NRC approved version of N-648 for operating plants in the future, SNC will evaluate measures that will need to be taken at that time.

RAI Question 2:

The request notes that you plan to adopt Code Case N-648-1, “Alternative Requirements for Inner Radius Examinations of Class 1 Reactor Vessel Nozzles,” for future ISI, however, even though you do not reference Code Case N-648-2, it appears that your technical basis for application of VT-1 for PSI is based on that code case. Code Case N-648-1 applies only to ISI and is conditionally approved for use by the NRC in Regulatory Guide 1.147, “Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1.” Code Case N-648-2 applies to both PSI and ISI, and is currently being reviewed by the NRC staff for incorporation into Regulatory Guide 1.147 based on comments provided during ASME Code meetings. Please confirm that you are basing application of VT-1 for PSI on Code Case N-648-2 and discuss how you have addressed potential NRC staff comments regarding use of this code case for new reactors with no previous operating experience.

SNC Response to RAI Question 2:

The requested alternative is consistent with the requirements of Code Case N-648-2, but N-648-2 was not specifically referenced, as this is not allowed during the period when the staff is reviewing N-648-2 for acceptance. The NRC staff comments on N-648-2 are addressed in the alternative. The first comment by the staff on N-648-2 was in 2013 to ASME Code Committees, and asked if the conditions in Regulatory Guide 1.147 for N-648-1 were included. The alternative addresses the NRC condition on N-648-1 by utilizing the ASME Section XI Table IWB-3512-1 acceptance criteria, and goes beyond the requirements of Code Case N-648-2, in that a manual UT is performed as part of the pre-service examinations. The second comment by the staff on N-648-2 was in 2013 as well, which requested that a plant specific flaw tolerance be performed for the AP1000 nozzle at inner radius corner. The current alternative satisfies this request by providing the flaw tolerance evaluation for the AP1000 nozzles.

In addition to the above discussion, it is noted that the nozzles to which the alternative applies are fabricated from nozzle forgings, in the same manner in which the nozzles now in service in operating plants were fabricated. The material properties (yield strength, ultimate strength, and fracture toughness) for the AP1000 nozzles are the same or better as compared to the operating fleet. The geometries are also similar, in that there are no welds in the region of the nozzle corner. Furthermore, the stresses at the nozzle corner region for the AP1000 are similar to the operating fleet; nevertheless, a plant specific stress analysis evaluation was performed for the AP1000 nozzle corner regions based on finite element analysis. The stresses were then used to perform a plant specific flaw tolerance evaluation based on ASME Section XI (as described in the alternative request) and design basis ASME Section III Appendix G evaluation to demonstrate the structural integrity of the nozzle corner with presence of a large postulated flaw.

The water chemistry and PWR environment for the AP1000 are also similar to the operating fleet. The AP1000 chemistry requirements follow the latest provided EPRI water chemistry requirements, and over time the requirements have become stricter due to the advances in instrumentation and their sensitivities. Nevertheless, the AP1000 water chemistry ranges (such as pH, boron concentration, conductivity, dissolved hydrogen and oxygen) are similar to that of the operating fleet. In general, lack of oxygen in the water chemistry precludes general corrosion and wastage in the carbon steel during normal operating conditions (where primary water chemistry is controlled and which generally represents about 90% of the plant lifetime). During shutdown conditions, any potential for corrosion is prevented with the presence of stainless steel cladding which is layered over the carbon steel base material.

Volumetric Examination

The reason volumetric exams were required in the nozzle corner region in Section XI was the history of degradation experienced in fossil plants, where welds are present in the nozzle corner region. Such geometry does not exist in the nozzles of interest here.

Examination techniques for this region have been perfected over the years. Regulatory Guide 1.150 stimulated improvement in examinations of the clad to base-metal interface many years ago. The same techniques have been used for more than 25 years for nozzle inner radius examinations performed from the bore. Capability demonstrations for the clad to base-metal interface have been conducted at the EPRI Nondestructive Examination (NDE) Center since 1983. These demonstrations were performed primarily for the belt-line region. However, the same techniques are used for both the vessel belt-line and the nozzle from the inside surface.

Performance Demonstration Initiative (PDI) demonstrations were initiated in 1994, for Appendix VIII Supplements 4 and 6. Vendors performing these PDI demonstrations found that few if any changes were required to achieve high success rates for the clad to base-metal interface, Supplement 4.

Five inspection vendors and more than 50 personnel have completed Appendix VIII Supplement 4, clad to base-metal demonstrations. In this time no individual, even those who failed the test, failed to detect cracks larger than approximately 0.25″. Sizing capability was also very good. The mean sizing error was 0.12″ root mean square (RMS). Sizing errors for the lead personnel, who normally make acceptability decisions, were even better, at 0.08″ RMS.

Figure 1 depicts the expected rejection probability as a function of flaw size, documented in Reference 1. Correct rejection probability considers the detection capability and the sizing capability for flaws. For example, as shown in Figure 1, the probability of detecting and rejecting a flaw 0.25″ into the base material is equal to or greater than 90%.

Examinations using modern technology have been performed industry-wide since 1989, and these examinations were compiled in Table 1 of Reference 1. The table shows that over 550 exams had been completed as of the year 2000, with no indications found. In addition to these examinations, some 2500 examinations had been completed by that time, using earlier technologies, with no indications ever being discovered.

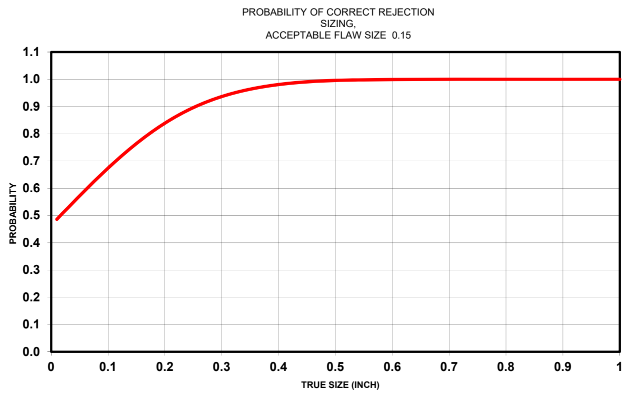

Figure 1. Probability of Correct Rejection Sizing, Acceptable Flaw Size 0.15 (Reference 1)

Table 1. Inspection Results Using Modern Technology		
Inspection Agency	**Number of Nozzles Inspected**	**Indications**
Westinghouse	210	0
IHI – Southwest	196	0
Framatome Technologies	148	0
Total	554	0

In summary, the nozzles covered by this alternative are fabricated in the same manner as those which have been in service now for typically over 40 years, and therefore are expected to have the same excellent reliability. In addition to the multiple exams conducted during the fabrication process, a manual UT exam will be performed to ensure that the nozzles are free of indications.

RAI Question 3:

The request notes that your elastic plastic fracture mechanics method followed the guidelines of Code Case N-749, "Alternative Acceptance Criteria for Flaws in Ferritic Steel Components Operating in the Upper Shelf Temperature Range." This code case is not currently approved for use, however, conditions on its use were proposed in the proposed rule published in the *Federal Register* on March 2, 2016 (81 FR 10780). Please discuss how you have addressed the proposed conditions.

SNC Response to RAI Question 3:

Justification for Use of Code Case N-749

The Elastic-Plastic Fracture Mechanics (EPFM) method used in the AP1000 inner radii analyses follows the guidance of Code Case N-749 which is similar to the procedures for ferritic components provided in Appendix K of ASME Section XI and Regulatory Guide 1.161. Code Case N-749 is applicable to ferritic steel components operating in the upper shelf temperature range and uses similar methods as Appendix K but with more conservative safety factors. A review of the through-wall temperatures for the limiting transient events revealed that the nozzle material is well above the upper shelf temperature for level A/B[1] events and the fracture toughness of the nozzle material is relatively high. Therefore, the procedures of Code Case N-749 were used to calculate a more appropriate end of evaluation period flaw size for Level A/B conditions as a supplement to LEFM (Linear Elastic Fracture Mechanics) results. The technical basis for the alternative acceptance criteria of Code Case N-749 is discussed in detail within PVP2012-78190 (Reference 2).

How the Conditions in the Federal Register were Addresses

The proposed rule in the Federal Register imposes a limit on the upper shelf transition temperature for use of Code Case N-749. When considering EPFM (Elastic Plastic Fracture Mechanics), the upper shelf temperature is limited to T_c = 154.8°F + 0.82 x RT_{NDT}. Therefore, the temperature of the transients evaluated at the most limiting time step should be greater than this upper shelf temperature in order to use the elastic plastic fracture mechanics methodology. In the case of the AP1000 nozzles, the nil ductility transition temperature is conservatively set to 10°F based on the design specification limit. The AP1000 nozzles evaluated are all above the beltline region and there is no significant irradiation embrittlement. Therefore, the upper shelf temperature per the proposed rule would be T_c =163°F. The limiting stresses intensity factor values for the AP1000 nozzles occur at temperatures well above T_c when the EPFM guidelines are considered in the evaluation. Stress intensity factors occurring during transient events at lower temperatures are not significant and do not factor into the final limiting flaw size calculation, and are covered based on the LEFM methodology.

[1] Note that the EPFM method was only used for level A/B conditions. The LEFM method was used for Level C/D conditions.

References

1. W. H. Bamford et al., 'Technical Basis for Elimination of Reactor Vessel Nozzle Inner Radius Inspections,' Proceedings of ASME 2001 Pressure Vessels and Piping Conference, Atlanta, GA.

2: H.L. Gustin, R.C. Cipolla, S.X. Xu, D.A. Scarth, Proceedings of the ASME 2012 Pressure Vessel & Piping Conference, PVP2012-78190, "Alternative Acceptance Criteria for Flaws in Ferritic Steel Components Operating in the Upper Shelf Temperature Range."