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

SAFETY EVALUATION BY THE OFFICE OF NEW REACTORS

RELATED TO REQUEST FOR ALTERNATIVE:

APPLICATION OF VT-1 VISUAL EXAMINATION METHODOLOGY

FOR PRESERVICE INSPECTION OF REACTOR VESSEL NOZZLE

INNER RADIUS SECTIONS (VEGP 3&4-PSI-ALT-07)

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MEAG POWER SPVM, LLC

MEAG POWER SPVJ, LLC

MEAG POWER SPVP, LLC

CITY OF DALTON, GEORGIA

VOGTLE ELECTRIC GENERATING PLANT UNITS 3 AND 4

DOCKET NOS. 52-025 AND 52-026

1.0 INTRODUCTION

By letter dated July 6, 2017, as supplemented by letters dated December 8, 2017, March 8, 2018, and August 31, 2018 (Accession Nos. ML17192A125, ML17342A826, ML18067A287, and ML18243A540 (publicly available) and ML18243A541 (non-publicly available)), respectively, in the U.S. Nuclear Regulatory Commission's (NRC) Agencywide Documents Access and Management System), Southern Nuclear Operating Company, Inc. (SNC), requested NRC approval of an alternative to the requirements of American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, Table IWB-2500-1, Examination Category B-D, Item B3.100 for Vogtle Electric Generating Plant (VEGP), Units 3 and 4. The proposed alternative would allow SNC to perform a VT-1 visual examination for the preservice inspection of the inner radius sections of the reactor vessel inlet, outlet, and direct vessel injection (DVI) nozzles in lieu of the volumetric examination required by ASME Code, Section XI, Table IWB-2500-1, Examination Category B-D, Item B3.100.

2.0 REGULATORY EVALUATION

Section 50.55a(g)(3)(ii) of Title 10 of the *Code of Federal Regulations* (10 CFR) requires that ASME Code Class 1 components (including their supports) in nuclear power facilities whose construction permits were issued after 1974 meet the preservice examination requirements set forth in either the edition and addenda of Section III or Section XI of the ASME Code that was applied to the construction of the component, or in subsequent editions and addenda that have been incorporated by reference into 10 CFR 50.55a. Per 10 CFR 50.55a(z), alternatives to the requirements of 10 CFR 50.55a may be used when authorized by the Director, Office of New Reactors. In proposing alternatives, the licensee must demonstrate that: (1) the proposed alternative would provide an acceptable level of quality and safety; or (2) compliance would result in hardship or unusual difficulty without a compensating increase in quality and safety.

3.0 TECHNICAL EVALUATION

3.1 SNC's Alternative

The components affected by this request are the ASME Code Class 1 inlet, outlet, and DVI nozzles in the VEGP, Units 3 and 4 reactor vessel upper shells. Each reactor vessel has four 22 inch inner diameter inlet nozzles, two 31 inch inner diameter outlet nozzles, and two 6.81 inch inner diameter DVI nozzles (8 inch schedule 160 pipe connections) designed in accordance with ASME Code, Section III, Subsection NB (16 total nozzles, 8 for Unit 3 and 8 for Unit 4). The nozzles are fabricated of SA-508, Grade 3, Class 1 ferritic steel forgings clad on the inner diameter surface. The first cladding layer is Type 309L stainless steel and the subsequent cladding layers are Type 308L stainless steel.

The ASME Code of Record for the preservice inspection of VEGP, Units 3 and 4 is the 2007 Edition, including the 2008 Addenda, of ASME Code, Section XI. The inspection requirements for ASME Code Class 1 components are provided in ASME Code, Section XI, Subsection IWB. In accordance with ASME Code, Section XI, Table IWB-2500-1, Examination Category B-D, Item B3.100, volumetric (e.g., ultrasonic) examination is the required examination method for the reactor vessel nozzle inner radius section.

SNC proposed to perform a VT-1 visual examination for the preservice inspection of the inner radius sections of the reactor vessel inlet, outlet, and DVI nozzles for VEGP, Units 3 and 4 in lieu of the volumetric examination required by ASME Code, Section XI, Table IWB-2500-1, Examination Category B-D, Item B3.100. The proposed VT-1 visual examination method will use an underwater camera system attached to a submersible and will be performed in accordance with ASME Code, Section XI, as conditioned by 10 CFR 50.55a. The proposed VT-1 visual examination will cover essentially 100 percent of the examination volume as defined in ASME Code, Section XI, Figure IWB-2500-7(b). SNC also stated that all stress analysis locations, including the limiting stress cuts in the nozzle inner radius sections of the reactor vessel inlet, outlet, and DVI nozzles, evaluated for fatigue and flaw tolerance will be examined during the preservice inspection using the proposed VT-1 visual examination. SNC noted that the service-induced flaw mechanisms (fatigue) for the inner radius sections of the reactor vessel nozzles will be associated with the inner diameter surface of the cladding and further noted that VT-1 visual examinations are sufficient to detect such mechanisms well before the nozzle suffers degradation of its structural integrity.

While SNC stated that the proposed VT-1 visual examination will satisfy the preservice inspection requirements of the inner radius sections of the reactor vessel inlet, outlet, and DVI nozzles for VEGP, Units 3 and 4, they also proposed to perform supplemental inspections to include a liquid penetrant testing (PT) examination and a manual ultrasonic testing (UT) examination. The supplemental PT examination will be performed prior to the proposed VT-1 visual examination, and the supplemental UT examination will be performed prior to the proposed VT-1 visual examination but after the supplemental PT examination. The supplemental PT examination will be performed in accordance with ASME Code, Section XI, Subsection IWA-2222. The supplemental UT examination will be performed in accordance with ASME Code, Section XI, Appendix III. SNC stated that the supplemental UT examination will not meet the ASME Code, Section XI, Appendix VIII performance demonstration requirements because it is not warranted for a one-time exam when failure mechanisms are readily identified using visual techniques. The supplemental UT examination will be performed from the inner diameter surface using two opposing circumferential beam directions. Specifically, dual focused 70-degree transmit-receive longitudinal wave transducers with acoustic focusing at or near the clad-to-base metal interface will be used. SNC stated that the supplemental PT and UT examination reports will be included in the preservice inspection documentation package.

SNC stated that it is their intent to adopt ASME Code, Section XI, Code Case N-648-1, "Alternative Requirements for Inner Radius Examinations of Class 1 Reactor Vessel Nozzles," which allows licensees to perform a VT-1 visual examination for inservice inspection of reactor vessel nozzles other than boiling water reactor (BWR) feedwater nozzles and operational BWR control rod drive return line nozzles in lieu of the volumetric examination required by ASME Code, Section XI, Table IWB-2500-1, Examination Category B-D, Item B3.100. SNC further stated that performing a VT-1 visual examination for the preservice inspection of the inner radius sections of the reactor vessel inlet, outlet, and DVI nozzles for VEGP, Units 3 and 4 will align the preservice inspection with future planned inservice inspections.

SNC's proposal to extend performing a VT-1 visual examination in lieu of the volumetric examination required by the ASME Code to the preservice inspection of the inner radius sections of the reactor vessel inlet, outlet, and DVI nozzles for VEGP, Units 3 and 4 includes meeting the following requirements which SNC states are consistent with the Code Case N-648-2 requirements:

- Surface M-N on ASME Code, Section XI, Figures IWB-2500-7(a) through (d) will be examined using a surface examination method and shall meet the ASME Code, Section III fabrication acceptance standards at least once after the hydrostatic test.
- The volume O-P-Q-R on ASME Code, Section XI, Figures IWB-2500-7(a) through (d) will be examined using a manual volumetric examination method and shall meet the ASME Code, Section XI acceptance standards at least once after the hydrostatic test (Note: This is an additional requirement that is not in Code Case N-648-2).
- The appropriate surface is prepared in accordance with ASME Code, Section XI, IWA-2200(b) for application of future volumetric examinations.
- The fabrication examination history for nozzle inner radius region will be reviewed.
- The nozzles will be verified to meet the requirements in ASME Code, Section III, Nonmandatory Appendix G.

SNC noted that prior to 2001, volumetric examinations of reactor vessel nozzle inner radius sections for operating pressurized water reactors (PWR) detected no recordable flaw indications; and following 2001, enhanced VT-1 visual examinations (resolution capability of distinguishing a 1-mil wire) of operating PWR reactor vessel nozzle inner radius sections detected no recordable flaw indications. SNC also noted that no fatigue cracking has been reported throughout the operating history of PWR reactor vessel nozzle inner radius sections either ultrasonically or visually. The following fabrication examinations were performed for the VEGP, Units 3 and 4 reactor vessel inlet, outlet, and DVI nozzles and SNC stated that no recordable indications were detected.

- Magnetic particle (MT) examination of the external and accessible internal surfaces of the nozzles per ASME Code, Section III, Subarticle NB-2500.
- UT examination of the nozzles per ASME Code, Section III, Article NB-5000. SNC noted that these examinations were enhanced as required by the Westinghouse material specification for SA-508 forging materials.
- MT examination of the base metal surface of the nozzles prior to deposition of the stainless steel cladding per ASME Code, Section III, Subsubarticle NB-5120.
- PT examination of the deposited stainless steel cladding surface.
- UT examination for lack of bond of the deposited stainless steel cladding per the Westinghouse fabrication specification.
- UT examination of the nozzle inner radius sections performed after completion of welding and the intermediate heat treatment but before the post-weld heat treatment, and conducted from the inside and outside diameter surfaces. SNC noted that these examinations are required by the Westinghouse fabrication specification.
- PT examination of stainless steel cladding surface of the nozzle inner radius sections performed after the vessel hydrostatic test per the Westinghouse fabrication specification.
- UT examination of the nozzle inner radius sections performed after the vessel hydrostatic test and conducted from the inside and outside diameter surfaces per the Westinghouse fabrication specification.

SNC asserted that the VEGP, Units 3 and 4 reactor vessel inlet, outlet, and DVI nozzles are expected to have the same excellent reliability as the current operating PWR fleet because the:

- Nozzles are fabricated in the same manner (nozzle forgings);
- Material properties of the nozzles, including yield strength, ultimate strength, and fracture toughness, are the same or better;
- Nozzle geometries are similar in that there are no welds in the nozzle inner radius section;
- Water chemistry and PWR environment are similar; and
- Pressure stresses in the nozzle inner radius sections are similar to the pressure stresses in the nozzle inner radius sections of operating vessels of similar size and geometry.

While the pressure stresses are similar, SNC stated that the stresses from the thermal transients would be different because the AP1000 design includes more transients overall due to the passive cooling design of the plant and has more transient cycles due to the AP1000 being designed for a 60-year life.

SNC stated that the stresses were determined by a finite element analysis (FEA) in accordance with ASME Code, Section III, Subarticle NB-3200 for the inner radius sections of the reactor vessel inlet, outlet, and DVI nozzles at limiting stress cuts in the nozzle inner radius sections.

The ASME Code, Section III, Subarticle NB-3200 evaluation also included cumulative fatigue usage factor (CUF) calculations.

SNC's evaluation of the AP1000 standard plant CUFs for the reactor vessel inlet, outlet, and DVI nozzles were calculated considering the full set of applicable AP1000 design basis loads, transients, and associated cycles for a 60-year design life using the ANSYS Fatigue Module. The AP1000 standard plant CUFs for the reactor vessel inlet and outlet nozzles were calculated using a customized ANSYS macro that provides for fatigue analysis options not available in the ANSYS Fatigue Module. [

]. The AP1000 standard plant CUF for the DVI nozzles was calculated using the ANSYS Fatigue Module without a customized ANSYS macro. [

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SNC stated that the CUFs at the inner radius sections of the reactor vessel inlet, outlet, and DVI nozzles for the AP1000 standard plant were all calculated to be below the acceptable ASME Code, Section III, Subparagraph NB-3222.4 limit of 1.0. The AP1000 standard plant CUF value for the DVI nozzles showed less margin with respect to the ASME fatigue limit of 1.0 because it was calculated using a more conservative approach. Therefore, SNC recalculated a VEGP-specific CUF value for the DVI nozzles using a calculation method more consistent with the method used to calculate the AP1000 standard plant CUF values for the inlet and outlet nozzles. [

]. SNC calculated AP1000 standard plant CUF values for the reactor vessel inlet and outlet nozzles of [ ] and [ ], respectively. SNC calculated a VEGP-specific CUF value for the reactor vessel DVI nozzles of [ ].

SNC stated that the AP1000 reactor vessel was evaluated for its ability to protect against non-ductile failure in accordance with ASME Code, Section III, Appendix G using the stresses from the FEA, and that the reactor vessel inlet, outlet, and DVI nozzle regions were included in that evaluation. SNC also stated that the AP1000 reactor vessel nozzles are in compliance with ASME Code, Section III, Appendix G because the evaluation showed that the inner radius

sections of the reactor vessel nozzles have sufficient fracture toughness to meet the requirements of the ASME Code for vessel integrity for the design life of the plant.
SNC used the stresses from the FEA to perform flaw tolerance evaluations. SNC determined the allowable end of evaluation period flaw sizes (depth within the underlying base metal) for the inner radius sections of the reactor vessel inlet, outlet, and DVI nozzles for the AP1000 plant design using linear elastic fracture mechanics (LEFM) for Level A/B/Test conditions and Level C/D conditions. SNC stated that the LEFM was performed in accordance with ASME Code, Section XI, Subsection IWB-3600 and is very conservative because it does not take into account the ductile behavior of the nozzle material. The Level A/B/Test conditions were determined to be more limiting than the Level C/D conditions. The LEFM for Level A/B/Test conditions resulted in average allowable end of evaluation period flaw sizes for the inner radius sections of the reactor vessel inlet, outlet, and DVI nozzles of 1.01 inches, 1.04 inches, and 0.36 inch, respectively.

SNC also determined the allowable end of evaluation period flaw sizes using elastic plastic fracture mechanics (EPFM) which provides a more realistic fracture assessment considering the resistance to crack extension of the ductile nozzle material. SNC stated that the EPFM was performed using ASME Code, Section XI, Code Case N-749, "Alternative Acceptance Criteria for Flaws in Ferritic Steel Components Operating in the Upper Shelf Temperature Range, Section XI, Division 1," only for Level A/B/Test conditions (more limiting than Level C/D conditions). Code Case N-749 is conditionally accepted in Regulatory Guide (RG) 1.147, Revision 18, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1." The condition imposes limits on the upper shelf transition temperature. SNC stated that they met the NRC condition on the use of Code Case N-749 because the temperature of all the transients evaluated were greater than the upper shelf transition temperature above which the EPFM method must be applied. The EPFM for Level A/B/Test conditions resulted in average allowable end of evaluation period flaw sizes for the inner radius sections of the reactor vessel inlet, outlet, and DVI nozzles of 6 inches, 5 inches, and 4.75 inches, respectively.

SNC performed fatigue crack growth analyses to determine the maximum initial flaw size (depth within the underlying base metal) that will not grow beyond the allowable end of evaluation period flaw size within the life of the plant (60 years) considering Level A/B/Test conditions using both LEFM and EPFM. SNC also performed fatigue crack growth analyses to determine the maximum initial flaw size for a 10-year period (consistent with the 10-year inspection interval) using LEFM only. SNC used the crack growth law for ferritic steels not susceptible to environmentally assisted cracking in ASME Code, Section XI, Code Case N-643-2, "Fatigue Crack Growth Rate Curves for Ferritic Steels in PWR Water Environment, Section XI, Division 1." For a 60 year period, the LEFM for Level A/B/Test conditions resulted in average maximum initial flaw sizes for the inner radius sections of the reactor vessel inlet, outlet, and DVI nozzles of 0.725 inch, 0.780 inch, and 0.331 inch, respectively. For a 60 year period, the EPFM for Level A/B/Test conditions resulted in average maximum initial flaw sizes for the inner radius sections of the reactor vessel inlet, outlet, and DVI nozzles of 4.369 inches, 3.342 inches, and 4.392 inches, respectively.

SNC committed to apply the allowable flaw length criteria of ASME Code, Section XI, Table IWB-3512-1, with flaw aspect ratio (*a/l*) limited to 0.5 and the flaw depth/component thickness ratio (*a/t*) limited to 2.5 percent, as the VT-1 visual examination acceptance standard (NRC condition on use of Code Case N-648-1). Applying this allowable flaw length criteria, SNC determined the VT-1 visual examination acceptance standard for the reactor vessel inlet, outlet, and DVI nozzles as 0.392 inch, 0.384 inch, and 0.144 inch, respectively. SNC stated that a VT-1 visual examination finding exceeding the acceptance standards would result in repair/replacement, reexamination, and regulatory review in accordance with ASME Code, Section XI, Subsections IWB-3113 and IWB-3114 to ensure fitness for service. SNC further stated that the VT-1 visual examination acceptance standards are more stringent than the maximum initial flaw lengths for the reactor vessel inlet, outlet, and DVI nozzles. For example, using an *a/l* value of 0.5 and a cladding thickness of 0.22 inch, SNC calculated a corresponding maximum initial flaw length on the cladding surface of 1.14 ((0.351 inch + 0.22 inch)/0.5) inches for the most limiting maximum initial flaw size of 0.351 inch deep within the underlying DVI nozzle base metal for a 10 year period (LEFM), which is greater than the VT-1 visual examination acceptance standard of 0.144 inch for the reactor vessel DVI nozzles.

Based on the above, SNC stated that the proposed alternative provides an acceptable level of quality and safety in accordance with 10 CFR 50.55a(z)(1).

3.2 Staff Evaluation

Section 50.55a(g)(3)(ii) of 10 CFR requires that ASME Code Class 1 components (including their supports) in nuclear power facilities whose construction permits were issued after 1974 meet the preservice examination requirements set forth in either the edition and addenda of Section III or Section XI of the ASME Code that was applied to the construction of the component, or in subsequent editions and addenda that have been incorporated by reference into 10 CFR 50.55a. Requirements for the preservice and inservice inspection of ASME Code Class 1 components are provided in ASME Code, Section XI, Subsection IWB. The reactor vessel inlet, outlet, and DVI nozzles are classified as ASME Code Class 1, therefore, the requirements of ASME Code, Section XI, Subsection IWB, must be applied. ASME Code, Section XI, Table IWB-2500-1, Examination Category B-D, Item B3.100 requires that the reactor vessel nozzle inner radius sections be volumetrically examined. Mandatory Appendix I of ASME Code, Section XI requires that the procedures, equipment, and personnel used for UT examination of the nozzle inside radius section be qualified by performance demonstration in accordance with ASME Code, Section XI, mandatory Appendix VIII, Supplement 5 or 7. Appendix I also requires that the nozzle inside radius section be examined in two opposing circumferential directions.

To provide a baseline for subsequent inservice examinations, ASME Code, Section XI, Subsection IWB-2200 requires that all examinations required by ASME Code, Section XI, Table IWB-2500-1 (with the exception of Examination Category B-P and the VT-3 examination of the internal surfaces of Categories B-L-2 and B-M-2), be performed prior to initial plant startup.

To satisfy the preservice inspection requirements, SNC proposed to perform a VT-1 visual examination of the inner radius sections of the reactor vessel inlet, outlet, and DVI nozzles for VEGP, Units 3 and 4 in lieu of the volumetric examination required by ASME Code, Section XI, Table IWB-2500-1, Examination Category B-D, Item B3.100 in order to align the preservice inspection with future planned inservice inspections. SNC intends on performing VT-1 visual examinations for the inservice inspections of the inner radius sections of the reactor vessel inlet, outlet, and DVI nozzles for VEGP, Units 3 and 4 in accordance with ASME Code, Section XI, Code Case N-648-1 in RG 1.147.

The ASME Code requirement to inspect the nozzle inner radius sections by volumetric examination was in response to cracking discovered in a non-nuclear vessel. The primary degradation mode in reactor vessel nozzles is thermal fatigue which produces surface breaking flaw indications along the nozzle's inner radius section. In the 1970s, thermal fatigue cracking was discovered in the inner radius section of feedwater nozzles and control rod drive return line nozzles at BWRs. No thermal fatigue cracking has been reported for the inner radius sections of the reactor vessel nozzles in the current operating PWR fleet. In part based on this good inspection history, ASME Code, Section XI, Code Case N-648-1 was developed which allows licensees to perform a VT-1 visual examination for inservice inspection of reactor vessel nozzles other than BWR feedwater nozzles and operational BWR control rod drive return line nozzles in lieu of the volumetric examination required by ASME Code, Section XI, Table IWB-2500-1, Examination Category B-D, Item B3.100. Code Case N-648-1 applies only to inservice inspection and is conditionally accepted in RG 1.147, Revision 18. The condition is that the allowable flaw length criteria of ASME Code, Section XI, Table IWB-3512-1, with limiting assumptions on the flaw aspect ratio be used.

The technical basis for Code Case N-648-1 included a flaw tolerance evaluation using LEFM that resulted in allowable end of evaluation period flaw sizes larger than 3 inches for the nozzle inner radius sections of the operating fleet (PWRs and BWRs). In addition, the technical basis stated that the CUF is less than 0.1 for all the nozzle inner radius sections for the operating fleet. A CUF less than 0.1 means the operating fleet has a very low probability of initiating and propagating fatigue cracks during service.

SNC noted that their proposed alternative aligns with the Code Case N-648-2 requirements which apply to both preservice and inservice inspection. In the proposed rule, "Approval of American Society of Mechanical Engineers' Code Cases," published on August 16, 2018 (83 FR 40685), the NRC proposed one condition on use of Code Case N-648-2. The proposed condition is that Code Case N-648-2 "shall not be used to eliminate the volumetric preservice inspection (PSI) examinations and shall not be used to eliminate the preservice or inservice volumetric examination of plants with a combined operating license pursuant to 10 CFR part 52, or a plant that receives its operating license after October 22, 2015." The statement of considerations (SOC) for the proposed rule stated that the bases for the proposed condition is that a preservice volumetric examination of the inner radius section of all reactor vessel nozzles should be performed for comparison with future volumetric examinations, if indications of flaws are found; and new reactor designs have no inspection history or operating experience to support eliminating the periodic volumetric examination of the inner radius section of reactor vessel nozzles. The SOC also stated that "eliminating the volumetric preservice or inservice examinations is predicated on good operating experience for the existing fleet."

NRC conditions on a code case do not preclude a licensee from submitting an alternative similar to the conditioned code case. As previously discussed, in accordance with 10 CFR 50.55a(z), the licensee must demonstrate that: (1) the proposed alternative would provide an acceptable level of quality and safety; or (2) compliance would result in hardship or unusual difficulty without a compensating increase in quality and safety. Staff has reviewed the information submitted by SNC to support their assertion that the proposed alternative provides an acceptable level of quality and safety at VEGP, Units 3 and 4. As part of its site-specific review, the staff evaluated the applicability of the good operating experience for similarly designed reactor vessel nozzles for the current operating PWR fleet to the VEGP, Units 3 and 4 reactor vessel inlet, outlet, and DVI nozzles. Specifically, staff looked at design, materials, water chemistry, stresses, fatigue and flaw tolerance evaluations, and the proposed VT-1 visual examination. This site-specific review for VEGP, Units 3 and 4 does not make any general determinations with regards to Code Case N-648-2.

Design, Materials, and Water Chemistry

VEGP, Units 3 and 4 reactor vessel inlet and outlet nozzles are of similar size and geometry of similarly designed reactor vessel inlet and outlet nozzles of the current operating PWR fleet. However, VEGP, Units 3 and 4 reactor vessel DVI nozzles differ in geometry and size than the reactor vessel inlet and outlet nozzles for both VEGP, Units 3 and 4 and the current operating PWR fleet. The inner diameter of VEGP, Units 3 and 4 reactor vessel DVI nozzles (6.81 inches) is smaller and includes a 4 inch venturi for the purpose of limiting high-pressure blowdown flow. In addition, the inner radius section thickness of VEGP, Units 3 and 4 reactor vessel DVI nozzles is thinner. Like the current operating PWR fleet, VEGP, Units 3 and 4 reactor vessel inlet, outlet, and DVI nozzles do not have a weld in the inner radius section of the nozzles.

Similar to the current operating PWR fleet, VEGP, Units 3 and 4 reactor vessel inlet, outlet, and DVI nozzles are fabricated of SA-508, Grade 3, Class 1, ferritic steel forgings clad on the inner diameter surface with multiple stainless steel cladding layers.

The water chemistry that VEGP, Units 3 and 4 reactor vessel inlet, outlet, and DVI nozzles will be exposed to follows the same Electric Power Research Institute's water chemistry guidelines as the current operating PWR fleet. In comparison to some operating PWRs, the water chemistry may be better for VEGP, Units 3 and 4 due to following a more recent version of the industry guidelines.

Stresses and Fatigue and Flaw Tolerance Evaluations

SNC stated that the pressure stresses in the AP1000 nozzle inner radius sections are similar to operating vessels of a similar size and geometry, however, the stresses from the thermal transients differ for the AP1000 design because it includes more transients overall due to the passive cooling design of the plant and has more transient cycles due to it being designed for a 60-year life. Given that the stresses from the thermal transients are different, staff also reviewed the fatigue and flaw tolerance evaluations for the AP1000 reactor vessel inlet, outlet, and DVI nozzles.

In accordance with ASME Code, Section III, Subarticle NB-3200, the reactor vessel inlet, outlet, and DVI nozzles for the AP1000 standard plant were evaluated using FEA. The stresses determined by the FEA for the inner radius sections of the AP1000 reactor vessel inlet, outlet, and DVI nozzles were determined at limiting stress cuts in the nozzle inner radius sections. The staff finds this acceptable because the stress analysis locations were where the highest stresses will be in the inner radius sections of the AP1000 reactor vessel inlet, outlet, and DVI nozzles. The AP1000 standard plant CUF calculations were included in the ASME Code, Section III, Subarticle NB-3200 evaluation.

The AP1000 standard plant CUFs for the reactor vessel inlet, outlet, and DVI nozzles were calculated considering the full set of applicable AP1000 design basis loads, transients, and associated cycles for a 60-year design life using the ANSYS Fatigue Module. The AP1000 standard plant CUFs for the reactor vessel inlet and outlet nozzles were calculated using a customized ANSYS macro that provides for fatigue analysis options not available in the ANSYS Fatigue Module. The fatigue analysis options provided more realistic results which reduced conservatisms in the AP1000 standard plant CUF calculations for the reactor vessel inlet and outlet nozzles by [ ]. Conservatisms in the AP1000 standard plant CUF calculations for the reactor vessel inlet and outlet nozzles were further reduced by [ ]. The AP1000 standard plant CUF for the DVI nozzles was calculated using the ANSYS Fatigue Module without a customized ANSYS macro. Therefore, the fatigue analysis options were not available to reduce conservatisms in the AP1000 standard plant CUF calculation for the DVI nozzles. In addition, the AP1000 standard plant CUF calculation for the DVI nozzles did not [ ].

Even though the CUFs at the inner radius sections of the reactor vessel inlet, outlet, and DVI nozzles for the AP1000 standard plant were all calculated to be below the acceptable ASME Code, Section III, Subparagraph NB-3222.4 limit of 1.0, SNC recalculated a VEGP-specific CUF value for the DVI nozzles using a calculation method more consistent with the method used to calculate the AP1000 standard plant CUF values for the inlet and outlet nozzles. The VEGP-specific CUF value for the DVI nozzles was recalculated using the [ ]. Therefore, the VEGP-specific CUF for the reactor vessel DVI nozzle is more realistic than the AP1000 standard plant CUF for the reactor vessel DVI nozzle due to the reduction in conservatisms. The staff finds the calculation methods used to calculate the AP1000 standard plant CUFs for the reactor vessel inlet, outlet, and DVI nozzles; and the VEGP-specific CUF for the reactor vessel DVI nozzles acceptable because they used an NRC-approved fatigue analysis tool and meet the ASME Code.

SNC calculated AP1000 standard plant CUF values for the reactor vessel inlet and outlet nozzles of [ ] and [ ], respectively, and calculated an VEGP-specific CUF value for the reactor vessel DVI nozzles of [ ]. While the AP1000 standard plant CUF values for the inlet and outlet nozzles and the VEGP-specific CUF value for the DVI nozzles are slightly above the CUF value reported for the operating fleet, the CUF values still indicate that the nozzles have a low probability of initiating and propagating fatigue cracks during service. In addition, the staff recognizes that the AP1000 standard plant CUF values for the inlet and outlet nozzles and the VEGP-specific CUF value for the DVI nozzles were calculated for 60 years, and, therefore, would be closer to the CUF values for the operating fleet if calculated for 40 years like the operating fleet.

SNC used the stresses from the FEA to perform flaw tolerance evaluations. SNC determined the allowable end of evaluation period flaw sizes (depth within the underlying base metal) for the inner radius sections of the reactor vessel inlet, outlet, and DVI nozzles for the AP1000 plant design using LEFM for Level A/B/Test conditions and Level C/D conditions in accordance with ASME Code, Section XI, Subsection IWB-3600; and using EPFM for only Level A/B/Test conditions (more limiting than Level C/D conditions) in accordance with ASME Code, Section XI, Code Case N-749, as conditioned in 10 CFR 50.55a. SNC stated that the EPFM provides a more realistic fracture assessment considering the resistance to crack extension of the ductile nozzle material. The LEFM for Level A/B/Test conditions resulted in average allowable end of evaluation period flaw sizes for the inner radius sections of the reactor vessel inlet, outlet, and DVI nozzles of 1.01 inches, 1.04 inches, and 0.36 inch, respectively. The EPFM for Level A/B/Test conditions resulted in average allowable end of evaluation period flaw sizes for the inner radius sections of the reactor vessel inlet, outlet, and DVI nozzles of 6 inches, 5 inches, and 4.75 inches, respectively. The staff agrees that the EPFM provides more realistic allowable end of evaluation period flaw sizes because it considers the ductile nozzle material's resistance to crack extension. The staff finds the flaw tolerance evaluations performed for the inner radius sections of the reactor vessel inlet, outlet, and DVI nozzles for the AP1000 plant design acceptable because they meet 10 CFR 50.55a and the ASME Code.

The LEFM flaw tolerance results for the inner radius sections of the reactor vessel inlet, outlet, and DVI nozzles for the AP1000 plant design showed tolerance for smaller flaws than the LEFM flaw tolerance results for the operating fleet (greater than 3 inches). However, the staff recognizes that the flaw tolerance results for the AP1000 reactor vessel inlet, outlet, and DVI nozzles were calculated for 60 years, and, therefore, would be closer to the flaw tolerance results for the operating fleet if calculated for 40 years like the operating fleet. It should be noted that the inner radius thicknesses for the AP1000 inlet and outlet nozzles and operating fleet inlet and outlet nozzles are similar, and therefore would have similar flaw sizes. While the inner radius thickness for the DVI nozzles is thinner than the operating fleet, the corresponding flaw size that can be tolerated would be smaller, but would have similar margin when considering the flaw size depth to inner radius thickness.

Proposed VT-1 Visual Examination

SNC proposed to perform a VT-1 visual examination for the preservice inspection of the inner radius sections of the reactor vessel inlet, outlet, and DVI nozzles for VEGP, Units 3 and 4 in lieu of the volumetric examination required by ASME Code, Section XI, Table IWB-2500-1, Examination Category B-D, Item B3.100. The proposed VT-1 visual examination method will use an underwater camera system attached to a submersible and will be performed in accordance with ASME Code, Section XI, as conditioned by 10 CFR 50.55a. The proposed VT 1 visual examination will cover essentially 100 percent of the examination volume as defined in ASME Code, Section XI, Figure IWB-2500-7(b). In addition, SNC stated that all stress analysis locations, including the limiting stress cuts in the nozzle inner radius sections of the reactor vessel inlet, outlet, and DVI nozzles, evaluated for fatigue and flaw tolerance will be examined during the preservice inspection using the proposed VT-1 visual examination.

SNC committed to apply the allowable flaw length criteria of ASME Code, Section XI, Table IWB-3512-1, with *a/l* limited to 0.5 and *a/t* limited to 2.5 percent, as the VT-1 visual examination acceptance standard (NRC condition on use of Code Case N-648-1). Applying this allowable flaw length criteria, SNC determined the VT-1 visual examination acceptance standard for the reactor vessel inlet, outlet, and DVI nozzles as 0.392 inch, 0.384 inch, and 0.144 inch, respectively. A VT-1 visual examination finding exceeding the acceptance standards would result in repair/replacement, reexamination, and regulatory review in accordance with ASME Code, Section XI, Subsections IWB-3113 and IWB-3114 to ensure fitness for service. Based on the flaw tolerance evaluations, the VT-1 visual examination acceptance standards for the reactor vessel inlet, outlet, and DVI nozzles are smaller than the maximum initial flaw lengths for the reactor vessel inlet, outlet, and DVI nozzles, even for the conservative LEFM. The staff finds this acceptable because flaws smaller than the maximum initial flaw lengths for the reactor vessel inlet, outlet, and DVI nozzles will be repaired/replaced, reexamined, and reviewed in accordance with ASME Code, Section XI, Subsections IWB-3113 and IWB-3114 to ensure fitness for service.

Performing the VT-1 visual examination in accordance with ASME Code, Section XI, as conditioned by 10 CFR 50.55a, demonstrates that SNC's examination methodology (i.e., personnel and equipment) is capable of 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). Therefore, it is reasonable to conclude that SNC's proposed VT-1 visual examination methodology is capable of detecting flaws smaller than the maximum initial flaw lengths for the reactor vessel inlet, outlet, and DVI nozzles, even for the more conservative LEFM; and flaws with lengths smaller than the acceptance standards for the reactor vessel inlet, outlet, and DVI nozzles.

The staff did not specifically review the acceptability of the supplemental PT and manual UT examinations because SNC did not state that they would contribute towards satisfying the preservice inspection requirements for the VEGP, Units 3 and 4 reactor vessel inlet, outlet, and DVI nozzles. However, staff agrees that the combination of the completed fabrication examinations and performance of the supplemental PT and manual UT examinations will provide additional assurance that no flaws exist on the cladding inner diameter surface of the nozzles prior to performing the proposed VT-1 visual examination.
Based on the review described above, staff finds that SNC has demonstrated that the proposed alternative provides an acceptable level of quality and safety because the VEGP, Units 3 and 4 reactor vessel inlet, outlet, and DVI nozzles:

- Are fabricated in the same manner, including no weld in the inner radius section, and with the same materials as the nozzles for the operating fleet.
- Will be exposed to water chemistry that is the same or better than the operating fleet.
- Will have, similar to the operating fleet, a low probability of initiating and propagating fatigue cracks during service based on the calculated CUF values.
- Demonstrate tolerance for flaws within an acceptable margin to the flaw tolerance for the operating fleet.
- Will be inspected with a VT-1 visual examination performed in accordance with ASME Code, Section XI, as conditioned by 10 CFR 50.55a, that will cover essentially 100 percent of the examination volume as defined in ASME Code, Section XI, Figure IWB-2500-7(b). The examination volume includes the limiting stress cut locations for the inner radius sections of the reactor vessel inlet, outlet, and DVI nozzles.

- Will be inspected with a VT-1 visual examination method capable of detecting the type and sizes of flaws expected.

4.0 CONCLUSION

As set forth above, staff determines that the proposed alternative to the requirements of the 2007 Edition, including the 2008 Addenda, of ASME Code, Section XI, Table IWB-2500-1, Examination Category B-D, Item B3.100 provides an acceptable level of quality and safety. Accordingly, staff concludes that SNC has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1), and is in compliance with the ASME Code requirements. Therefore, staff authorizes PSI-ALT-07 for the preservice inspection at VEGP, Units 3 and 4. All other requirements of ASME Code, Section XI, and 10 CFR 50.55a for which an alternative has not been specifically requested and authorized, remain applicable.

5.0 REFERENCES

1. VEGP 3&4-PSI-ALT-07, "Request for Alternative: Application of VT-1 Visual Examination Methodology for Preservice Inspection of the Reactor Vessel Nozzle Inner Radius Sections," dated July 6, 2017 (ADAMS Accession No. ML17192A125).

2. "Request for Additional Information (Draft), Request for Alternative No. 1, Reactor Vessel Nozzle Inner Radius Sections," dated October 30, 2017 (ADAMS Accession No. ML17303A270).

3. VEGP 3&4-PSI-ALT-07S1, "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," dated December 8, 2017 (ADAMS Accession No. ML17342A826).

4. Request for Additional Information, dated February 23, 2018 (ADAMS Accession No. ML18054A672).

5. VEGP 3&4-PSI-ALT-07S2, "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," dated March 8, 2018 (ADAMS Accession No. ML18067A287).

6. Request for Additional Information, dated April 4, 2018 (ADAMS Accession No. ML18094B066).

7. VEGP 3&4-PSI-ALT-07S3, "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," dated August 31, 2018 (ADAMS Accession Nos. ML18243A540 (publicly available) and ML18243A541 (non-publicly available).

8. Vogtle Electric Generating Plant, Units 3 and 4, Updated Final Safety Analysis Report, dated August 11, 2017 (ADAMS Accession No. ML17172A218).