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

January 26, 2017

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

SUBJECT: VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2, AND JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 1 – ALTERNATIVE TO INSERVICE INSPECTION REGARDING REACTOR PRESURE VESSEL THREADS INFLANGE INSPECTION (CAC NOS. MF8061, MF8062, MF8070)

Dear Mr. Pierce:

By application dated August 4, 2016, as supplemented by letters dated October 24, 2016, and November 23, 2016, Southern Nuclear Operating Company, Inc. (SNC, the licensee) submitted a request for an alternative for the Vogtle Electric Generating Plant, Units 1 and 2, and Joseph M. Farley Nuclear Plant, Unit 1. The licensee proposes to eliminate the reactor pressure vessel (RPV) threads-in-flange examination requirement as an alternative to certain requirements of Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) for inservice inspection of RPV components.

Specifically, the licensee submitted the proposed alternative in accordance with the requirements of Title 10 of the *Code of Federal Regulations* 10 CFR 50.55a(z)(1) on the basis that the alternative provides an acceptable level of quality and safety.

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the subject request, and concludes that SNC has adequately addressed all of the regulatory requirements and that the proposed alternative provides an acceptable level of quality and safety. Therefore, the NRC staff authorizes the proposed alternative in accordance with 10 CFR 50.55a(z)(1). The NRC staff's safety evaluation is enclosed.

All other ASME Code, Section XI, requirements for which relief was not specifically requested and authorized herein by the NRC staff remain applicable, including the third party review by the Authorized Nuclear Inservice Inspector.

If you have any questions, please contact the Project Manager, Shawn Williams, at 301-415-1009 or by e-mail at Shawn.Williams@nrc.gov.

Sincerely,

Michael T. Markley

Michael T. Markley, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-424, 50-425, 50-348

Enclosure:
Safety Evaluation

cc w/encl: Distribution via Listserv

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

ALTERNATIVE REQUESTS VEGP-ISI-ALT-11, VERSION 2.0

AND FNP-ISI-ALT-19, VERSION 2.0

REACTOR PRESSURE VESSEL FLANGE BOLT HOLE THREAD EXAMINATION

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 1

SOUTHERN NUCLEAR OPERATING COMPANY

DOCKET NOS. 50-424, 50-425, 50-348

1.0 INTRODUCTION

By letter dated August 4, 2016 (Agencywide Documents Access and Management System (ADAMS) Package Accession No. ML16221A072), as supplemented by letters dated October 24, 2016 (ADAMS Accession No. ML16298A049), and November 23, 2016 (ADAMS Accession No. ML16328A374), Southern Nuclear Operating Company (SNC, the licensee) submitted a Request for Alternative (VEGP-ISI-ALT-11, Version 2.0) for Vogtle Electric Generating Plant (VEGP), Units 1 and 2, and an Alternative (FNP-ISI-ALT-19, Version 2.0), for Joseph M. Farley Nuclear Plant (FNP), Unit 1. The licensee proposes to eliminate the reactor pressure vessel (RPV) threads in flange examination requirement for the remainder of the third and fourth inservice inspection (ISI) intervals, respectively, as an alternative to certain requirements of Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) for ISI of RPV components.

The licensee submitted the proposed alternative in accordance with the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(1) on the basis that the alternative provides an acceptable level of quality and safety.

2.0 REGULATORY EVALUATION

The ISI of the ASME Code Class 1, 2, and 3 components is performed in accordance with Section XI of the ASME Code and applicable addenda as a way to detect anomaly and degradation indications so that structural integrity of these components can be maintained. This is required by 50.55a(g), except where specific relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). Regulations in 10 CFR 50.55a(z) states that alternatives to the requirements of paragraphs (b) through (h) of 10 CFR 50.55a or portions thereof may be used, when authorized by the Director, Office of Nuclear Reactor Regulation. A proposed alternative must be submitted and authorized prior to implementation. The applicant or licensee must demonstrate that: (1) the proposed alternative would provide an acceptable level of quality and safety; or (2) compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(g)(4), components (including supports) that are classified as ASME Code Class 1, Class 2, and Class 3 must meet the requirements, except design and access provisions and preservice examination requirements, set forth in Section XI of editions and addenda of the ASME Code, that become effective subsequent to editions specified in paragraphs (g)(2) and (3) of this section, to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during successive 120-month inspection intervals (following the initial 120-month inspection interval) must comply with the requirements in the latest edition and addenda of the ASME Code, which was incorporated by reference in 10 CFR 50.55a(a) 12 months before the start of the 120-month interval (or the optional ASME Code Cases listed in NRC Regulatory Guide (RG) 1.147, Revision 17), subject to the conditions listed in 50.55a(b).

The applicable ASME Code of record for the third 10-year interval ISI program at VEGP, Units 1 and 2 and the fourth 10-year interval ISI program at FNP, Unit 1 is the 2001 Edition through the 2003 Addenda of the ASME Code, Section XI. The third 10-year interval ISI program at VEGP, Units 1 and 2, is scheduled to conclude on May 30, 2017, and the fourth 10-year interval ISI program at FNP, Unit 1, is scheduled to conclude on November 30, 2017.

3.0 TECHNICAL EVALUATION

3.1 Licensee's Evaluation

3.1.1 Component Identification

Requests for Alternative VEGP-ISI-ALT-11, Version 2.0 for VEGP, Units 1 and 2, and FNP-ISI-ALT-19, Version 2.0 for FNP, Unit 1, apply to Examination Category B-G-1, Item No. B6.40, "Threads in Flange."

3.1.2 Applicable ASME Code Requirement

Table IWB-2500-1 requires that the RPV threads in flange are examined by a volumetric examination technique with 100 percent of the flange ligament areas examined every ISI interval. The examination area is the one-inch area around each RPV stud hole, as shown in Figure IWB-2500-12.

3.1.3 Licensee's Proposed Alternative and Basis

Instead of the ASME Code, Section XI requirements for a volumetric ultrasonic examination, SNC proposes to eliminate the RPV flange threads requirement for the remainder of the third ISI interval for VEGP, Units 1 and 2, and the remainder of the fourth ISI interval for FNP, Unit 1.

The proposed alternative is based on stress analysis and flaw tolerance analysis results documented in an Electric Power Research Institute (EPRI) report entitled, "Nondestructive Evaluation: Reactor Pressure Vessel Threads in Flange Examination Requirements" (ADAMS Accession No. ML16221A068). SNC has confirmed that VEGP and FNP units' plant-specific parameters (e.g., vessel diameter, number of studs, and inservice inspection findings) are consistent with or bounded by the EPRI report.

The stress analysis and the subsequent flaw tolerance evaluation are based on finite element method (FEM) modeling of a portion of the RPV flange with a bolt hole and threads. Geometric symmetry was used to limit the size of the generic FEM model, which has a configuration bounding the sixteen units covered by the EPRI report. The loading includes pressure, bolt preload, and thermal due to heatup/cooldown. A flaw tolerance evaluation was performed using the results of the stress analysis to determine the allowable flaw size of a postulated flaw, considering fatigue crack growth. A linear elastic fracture mechanics (LEFM) evaluation consistent with ASME Code, Section XI, IWB-3600 was performed. Specifically, applied stress intensity factors (applied Ks) at four flaw depths of a 360°, partial-through-wall circumferential flaw, are obtained using the FEM model described above with crack-tip elements. The maximum applied K value around the bolt hole circumference for each flaw depth is then extracted from the FEM results and used for the crack growth calculations. The largest postulated flaw has a depth to thickness ratio (a/t) of 0.77.

The maximum applied K for the combined pressure, preload and thermal from the FEM analysis is 19.8 ksi$\sqrt{\text{in}}$. The allowable K based on the acceptance criteria in ASME Section XI, IWB-3610 for the RPV flange material with an assumed K_{IC} of 220 ksi$\sqrt{\text{in}}$ is 69.6 ksi$\sqrt{\text{in}}$. Since the allowable K is far greater than the applied K for all postulated crack depth, up to a/t =0.77, a crack with a configuration of even a/t = 0.77 originated from the bolt hole threads will still have adequate structural integrity.

For the crack growth evaluation, an initial postulated flaw size of 0.2 in. (5.08 mm) is chosen consistent with the ASME Code, Section XI IWB-3500 flaw acceptance standards. Heatup/cooldown of 50 times per year and constant bolt preload five times per year are considered in the fatigue crack growth calculation. The resulting crack growth was calculated to be negligible, and the allowable flaw size of a/t = 0.77 will not be reached. In summary, the integrity of the component is not challenged for at least 80 years.

Based on the stress analysis and flaw tolerance evaluation results, the licensee concluded that the threads in flange component at VEGP, Units 1 and 2, and FNP, Unit 1, are very flaw tolerant and can operate for 80 years without violating ASME Code, Section XI safety margins. Therefore, the licensee concluded the threads in flange examinations can be eliminated without affecting the structural integrity of the RPV.

In addition to the analytical results from the flaw tolerance evaluation, SNC states that in each outage it uses a detailed procedure for the removal, care and visual inspection of the RPV studs and the threads in flange to protect against non-service related degradation. Care is taken to not only remove the studs, but once the studs are removed to inspect the RPV threads for damage and to install RPV stud plugs to protect threads from damage. Prior to reinstallation, the studs and stud holes are cleaned and lubricated. The studs are then replaced and tensioned into the RPV. These controlled maintenance activities provide further assurance that degradation is detected and mitigated prior to returning the reactor to service.

Based on the above, SNC requests to use the proposed alternative pursuant to 10 CFR 50.55a(z)(1) on the basis that the use of the alternative provides an acceptable level of quality and safety.

3.2 NRC Staff's Evaluation

The NRC staff has evaluated the industry's generic stress analysis and flaw tolerance evaluation documented in Section 6 of the EPRI report. The purpose of the flaw tolerance evaluation, which is based on LEFM, is to demonstrate that a postulated flaw that originated from the RPV flange bolt hole threads can grow for 80 years under the normal loading condition without endangering the structural integrity of the RPV flange.

3.2.1 The EPRI's Generic Stress Analysis and Flaw Tolerance Evaluation

The EPRI's generic stress analysis and flaw tolerance evaluation are based on the FEM where a stress model was created for stress analyses and the model was then slightly modified to insert the crack-tip elements to become the LEFM model to determine the flaw size which will meet the acceptance criteria of ASME Code, Section XI, 3600. The NRC staff examined modeling assumptions for the stress model and the LEFM model, as well as details of the LEFM methodology, to ensure that the stresses, applied Ks, and the resulting flaw depth are reasonably correct and can be used for approval determination. The NRC staff examined analysis assumptions, FEM modeling, and flaw evaluation results, and issued RAIs regarding fidelity of the model and modeling techniques. The following is an evaluation of technical issues that were identified by the NRC staff and resolution of RAIs regarding the licensee's stress analysis and flaw tolerance evaluation.

3.2.1.1 Stress Analysis

A three-dimensional FEM model based on linear elasticity was used to perform the stress analysis for the RPV flange bolt hole threads. This is acceptable considering that: (1) FEM models based on linear elasticity were commonly used for RPVs, (2) the current model has a vessel wall thickness of 16 inches with a bolt hole of 7 inches in diameter, unlikely to create a sizeable plastic zone under the normal loading for the low-alloy steel, and (3) the resulting stresses presented in the EPRI report have a maximum value around 45 ksi. The stress FEM model has symmetric boundary conditions on both symmetry faces of the model: one through the bolt hole axis and one between the bolt holes. The RPV head was simulated by fixing the top of the cladding surface axially, and the bottom of the flange is coupled axially to simulate the rest of the RPV. All these FEM modeling techniques are in line with the generally accepted industry practice.

Applied Loads

The loads that were considered in the FEM analyses are operating pressure, bolt preload, and normal heatup/cooldown transients. These are normal operating loads. The ASME Code, Section XI provides evaluation procedures and acceptance criteria for both detected flaws and postulated flaws in various components. For detected flaws such as in Appendix A, "Analysis of Flaws," and Appendix C, "Evaluation of Flaws in Piping," in addition to normal operating loads, emergency and faulted loads are also considered. For postulated flaws such as in Appendix G, "Fracture Toughness Criteria for Protection against Failure," only normal operating loads are considered. The current application of considering only normal operating loads is acceptable because it is in line with the ASME Code, Section XI evaluation principle for postulated flaws in general applications. Although there are exceptions such as Appendix K, "Assessment of Reactor Vessels with Low Upper Shelf Charpy Impact Energy Levels," and Appendix L, "Operating Plant Fatigue Assessment," for postulated flaws, they are created for special applications with specific considerations as indicated in the Appendix titles and shall not be used as guidance for general usage.

Validity of the Stresses from the FEM Analysis (RAI-5, Question 2b)

Since validity of the stresses from the stress FEM model provides confidence in the validity of the applied Ks from the LEFM FEM model, the NRC staff requested the licensee to explain the effort taken to ensure that the stresses from the stress FEM model is realistic and valid. The licensee stated, in its response dated October 24, 2016, that,

> Hand calculations were performed to verify proper hoop and axial stresses due to pressure. The nominal preload stress was verified after the application. The resultant preload thread stress concentration contour, as shown in Figure 6-5 of the EPRI report, indicated reasonable response where the top 10 or so threads assumed majority of the preload. Verification was also performed to ensure that the stresses in the bottom region of the model trend toward uniform, indicating that the model includes sufficient axial length.

Based on the above, the NRC staff determined that sufficient effort was applied to ensure validity of the stresses. Although the licensee did not include a discussion of the verification of the thermal stresses, NRC staff concluded that was acceptable considering thermal stresses only contribute a negligible amount of the total applied K. For instance, for a crack with a/t =0.55, the combined contribution to the applied K due to pressure and thermal stresses is only 3.7 percent. Therefore, the contribution due to thermal alone is less than 3.7 percent. NRC staff, therefore, concluded the response to RAI-5, Question 2b, is acceptable.

Heatup Versus Cooldown (RAI-5, Question 3, and follow-up)

Section 6.1.2 of the EPRI report provides a brief description of only the heatup transient. The licensee's response to the follow-up RAI dated November 23, 2016, further clarified that, "[d]uring the transients, heatup and cooldown produce maximum tensile and maximum compressive stress at opposite time points of the transient history, and produce the same stress range (maximum stress minus minimum stress) at each node point." This confirmed that use of heatup or cooldown has no effect on the fatigue crack growth calculation described in Section 3.1.3.

The NRC staff's concern for the selection of the heatup transient in evaluation of the margin for the allowable flaw size under the combined loading (preload + heatup + pressure) is that the stresses due to heatup that contributed to the applied K shown in Table 6-1 of the EPRI report may not be tensile. However, considering that (1) the heatup transient would produce compressive stresses towards RPV inner wall and tensile stresses towards RPV outer wall, and (2) the RPV flange bolt hole is closer to the RPV outer wall, the NRC staff concludes that using only the heatup transient in the stress analysis is conservative, and, therefore, the response to RAI-5, Question 3 is acceptable.

Based on the above, the NRC staff concludes that the licensee's stress analysis is acceptable and the resulting stresses can be used in the subsequent LEFM analysis.

3.2.1.2 LEFM Analysis

The applied K is from the LEFM FEM model directly. Since the NRC staff already accepted the stress FEM model as discussed above, use of the standard approach of inserting crack-tip elements in the stress FEM model to generate applied K is acceptable. As stated in Section 3.1.3 of this SE, the licensee used the LEFM methodology consistent with ASME Code, Section XI, IWB-3600. This means that the crack growth was calculated in accordance with Appendix A, "Analysis of Flaws." The licensee used an initial postulated flaw size of 0.2 in. (5.08 mm) in the flaw growth calculation. This is appropriate because it is consistent with the ASME Code, Section XI IWB-3500 flaw acceptance standards. Heatup/cooldown of 50 times per year and constant bolt preload five times per year for the fatigue crack growth calculation are also conservative. It should be noted that the calculated small crack growth of 0.005 inches over 80 years of operation can tolerate a great deal of uncertainty in the input variables described above. This also contributes to the NRC staff's acceptance of the licensee's flaw growth result. Lastly, the licensee used an assumed K_{Ic} of 220 ksi$\sqrt{in}$ and applied the IWB-3600 acceptance criterion with a structural factor of $\sqrt{10}$ to determine the allowable flaw size. This confirmed that a flaw 77 percent through-wall would meet the IWB-3600 acceptance criterion. The NRC staff determined in Section 3.2.1.1 of this SE that it is appropriate to apply loading associated with postulated flaws in this application. Likewise, it is also appropriate to apply the acceptance criteria for postulated flaws in this application. However, instead of using a structural factor of 2 for postulated flaws (Appendix G), the licensee used the acceptance criteria of IWB-3600 with a structural factor of $\sqrt{10}$ for detected flaws in this application. The NRC staff considers this approach conservative and is, therefore, acceptable. The LEFM FEM modeling techniques are in accordance with the industry practice and the LEFM analysis is consistent with ASME Code, Section XI. However, the following is an evaluation of technical issues that were identified by the NRC staff and resolution of RAIs regarding the LEFM analysis.

Stress Model Versus the LEFM Model (RAI-5, Question 4a and follow-up)

Section 6.2.1 of the EPRI report did not provide sufficient details for the FEM model for the applied K determination. The licensee's response to follow-up RAI-5 confirmed that the FEM model for the applied K determination is the same as the FEM model for the stress determination. The licensee also confirmed that loads applied on the two models are identical. It further stated it its response dated November 23, 2016, that,

> The purpose of performing the stress determination on the model without crack tip elements was to determine the appropriate location to insert the crack tip elements.

> Once that location is identified, the analyses were repeated using the model with crack tip elements to determine the K results. For example:
>
> - Internal pressure was applied uniformly on the inside surface of the stress and crack tip elements models.
> - Convective heatup heat transfer load was applied on the inside surface of the stress and crack tip elements models to determine temperature. Then the temperature results were imported to the corresponding stress or crack tip elements models to determine the stress or K results.

Since the licensee has provided detailed information on the stress FEM model without crack tip elements and the LEFM FEM model with crack tip elements, and how the pressure and thermal loads are applied to them, the ambiguity associated with the applied K determination has been removed. This direct applied K determination using identical FEM models with identical loading has the advantage of causing fewer modeling errors. The licensee's response to RAI-5, Question 4a and follow-up information are acceptable.

<u>Simplified FEM Model without Flange Threads (RAI-5, Question 5)</u>

The NRC staff issued RAI-5, Question 5, to request justification for use of a simplified FEM model under pressure and thermal loading conditions. In SNC's letter dated October 24, 2016, the licensee responded:

> The reactor pressure vessel (RPV) pressure and thermal transient loads were run without preload and without flange threads. The RPV pressure and thermal transient loads are global loads and they do not cause stress concentration in the threads. Since all analyses are linear elastic, the RPV pressure and thermal transient stresses are algebraically summed with the preload stresses for the crack growth iteration. A sensitivity analysis was performed with a representative number of threads included in the FEM model with internal pressure, and the results indicated that the stress distributions for models with and without the threads were very similar.

Although the licensee's sensitivity analysis is only for the pressure load, the NRC staff believes that the conclusion of very similar stresses with and without the threads applies to the thermal load also. Both pressure and thermal loads are global loads which are applied on the RPV inside surface at same distance from the bolt-hole threads. Therefore, the licensee's response to RAI-5, Question 5, is acceptable.

<u>Use of Fracture Toughness at Upper Shelf Energy Temperature (RAI-5, Question 6, and its follow-up)</u>

The NRC staff requested the licensee clarify whether it is conservative to use the fracture toughness (K_{IC}) at the upper shelf energy temperature in the flaw tolerance evaluation. The licensee's response dated October 24, 2016, states that:

> The use of the upper shelf temperature in the acceptance criteria is based on the fact that the component is at this temperature the majority of the time when the RPV is at full operating pressure. To determine the effect of lower temperatures on the

analysis, a conservative evaluation is performed by determining the maximum RT_{NDT} of the component to meet the acceptance criteria. In the analysis, the maximum K at any crack depth is about 20ksi√in. This requires a K_{IC} of 20 * √10 = 3.2 ksi√in [a misprint of 63.2 ksi√in]. Based on 2004 Edition of ASME Section XI, Appendix A, Figure A-4200-1, an RT_{NDT} of up to 70°F will not affect the results. For reference, the RT_{NDT} for the flange region is 20°F for VEGP Unit 1, 10°F for VEGP Unit 2, and 60°F for FNP Units 1 and 2.

According to Figure A-4200-1, a K_{IC} of 63.2 ksi√in will give a (T-RT_{NDT}) of 18.5 °F. Therefore, based on the RT_{NDT} values provided by the licensee, the operating temperature at the maximum applied K must be at least 38.5 °F for VEGP, Unit 1; 28.5 °F for VEGP, Unit 2; and 78.5 °F for FNP, Unit 1. VEGP, Units 1 and 2 meet these criteria because the minimum operating temperature of 60 °F per Vogtle Electric Generating Plant Pressure Temperature Limits Report [PTLR], Revision 2, dated July 1, 2005 (ADAMS Accession No. ML051870322) is higher than the calculated minimum temperatures for VEGP, Units 1 and 2. This is not true for FNP, Unit 1 because the minimum operating temperature of 60 °F per Joseph M. Farley, Units 1 & 2 PTLR, Revision 6, dated March 5, 2015 (ADAMS Accession No. ML15064A023), is lower than the calculated minimum temperature of 78.5 °F for FNP, Unit 1. The NRC staff examined this case further and found that the worst applied K reported in Table 6-1 of the EPRI report for the low temperature preload case is 17.4 ksi√in. This would require a K_{IC} of 55 ksi√in and a (T-RT_{NDT}) of 2.51 °F per ASME Section XI, Appendix A, Figure A-4200-1. Therefore, the calculated minimum temperature is 62.51 °F, more demanding than the actual operating temperature of 60 °F per FNP PTLR. The NRC staff noted, however, that the structural factor (safety factor) used in the current flaw tolerance evaluation is √10, typical for detected flaws under LEFM. Considering that the structural factor for postulated flaws in PTLR applications is only 2, the NRC staff determined that the conservatism in the structural factor alone is sufficient to offset the temperature difference of 2.51 °F.

Based on the above evaluations in Sections 3.2.1.1 and 3.2.1.2 of this SE, the NRC staff concludes that the generic stress analysis and flaw tolerance evaluation in the EPRI report are acceptable and the results (i.e., an allowable flaw 77 percent through-wall and the final size of 0.205 inch for the postulated flaw) can be used to support this request for alternative.

3.2.2 Plant-Specific Applicability of the EPRI's Generic Stress Analysis and Flaw Tolerance Evaluation to VEGP, Units 1 and 2 and FNP, Unit 1

Plant-Specific Applicability - RPV Parameters

VEGP, Units 1 and 2

The licensee provided in Table 1 of VEGP-ISI-ALT-11, Version 2.0 details of the RPV parameters for VEGP, Units 1 and 2, and compared them to the bounding values used in the EPRI's generic analysis. This table shows that the VEGP RPVs' number of studs, RPV inside diameter at stud hole, flange thickness at stud hole, and design pressure are identical to those in the generic analysis. However, the diameter of the stud for the VEGP RPVs is larger than that used in the generic analysis. These parameters are required to calculate the preload in accordance with the equation in the generic analysis. Since other key parameters are the same, the larger stud diameter of the VEGP RPVs results in lower preload per bolt. The licensee's conclusion that the stresses from the generic analysis is conservative in application to the VEGP RPVs is, therefore, acceptable to the NRC.

FNP, Unit 1

The licensee provided in Table 1 of FNP-ISI-ALT-19, Version 2.0, details of the RPV parameters for FNP, Unit 1, and compared them to the bounding values used in the EPRI's generic analysis. This table shows that the FNP RPV has a smaller RPV diameter, the same bolt diameter and design pressure, and more studs compared to those in the generic analysis. The FNP RPV flange thickness at stud hole is only 6 percent smaller than that in the generic analysis, and will not affect the preload as much as the combine effects due to the others. All these parameters are required to calculate the preload in accordance with the equation in the generic analysis. The smaller RPV diameter results in lower pressure and thermal stresses and the more number of studs results in lower preload per stud. Hence, the NRC staff accepts the licensee's conclusion that the stresses from the generic analysis would be conservative in application to FNP.

Plant-Specific Applicability – Consistency of the Preload Equation (RAI-2)

VEGP, Units 1 and 2 and FNP, Unit 1

The licensee states on Page E1-3 of the submittal dated August 1, 2016, "the preload was calculated as detailed in VEGP Units 1 and 2 RPV manual," and Page E2-3, "the preload was calculated as detailed in FNP, Unit 1 RPV manual." Considering that the preload equations in the plant-specific RPV manuals may not be the same as the preload equation in the generic analysis, RAI-2 was issued to request confirmation of consistency of the generic and plant-specific preload equations. In its letter dated October 24, 2016, the licensee stated that the preload stud stresses according to VEGP Manual and FNP Manual are less than the preload stud stress of 42,338 psi used in the generic analysis. The NRC staff, therefore, concludes the licensee's response to RAI-2 is acceptable.

Plant-Specific Applicability – Consistency of the Bolt Hole Number of Threads (RAI-4)

VEGP, Units 1 and 2, and FNP, Unit 1

Because the number of threads per foot of the flange bolt hole may be different from plant to plant so that the load on the threads will be different, the NRC staff issued RAI-4 to confirm consistency of VEGP and FNP units with the plant in the generic analysis. The licensee's response indicated that all plants considered in the EPRI report, including VEGP and FNP, use eight threads per inch configuration. The NRC staff, therefore, concludes the licensee's response to RAI-4 is acceptable.

Other RAIs which are not discussed above are primarily of a clarification nature. However, they still contribute to the NRC staff's confidence in the validity of the licensee's FEM modeling and results. Based on the evaluation of key technical elements of the generic analysis supporting the Alternatives and resolution of RAIs (Section 3.2.1), the NRC staff determined that the licensee's stress analysis and flaw tolerance evaluation are acceptable. The flaw tolerance evaluation results summarized in Table 6-1 of the EPRI report showed that the RPV flange hole is very flaw tolerant, such that a flaw can be 77 percent through-wall without exceeding the evaluation results summarized in Table 6-1 of the EPRI report showed that the RPV flange hole is very flaw tolerant, such that a flaw can be 77 percent through-wall without exceeding the ASME Code, Section XI acceptance criteria. Further, a postulated flaw of 0.2 inch is estimated to grow to only 0.205 inch for 80 years of operation. The preceding generic evaluation applies to the VEGP and FNP units based on the NRC staff's plant-specific applicability evaluation of Section 3.2.2. Considering that the postulated flaw with crack growth of 80 years is still far below the allowable flaw depth and the past examination results showing no reportable indications for the U.S. fleet of RPVs (Section 3 of the EPRI report), the NRC staff determined that the component is very flaw tolerant and Alternatives VEGP-ISI-ALT-11, Version 2.0 and FNP-ISI-ALT-19, Version 2.0 provides an acceptable level of quality and safety.

NRC Staff Conclusion

Based on NRC staff's evaluation in Section 3.2 of this SE, the NRC staff concludes that the licensee's stress analysis and flaw tolerance evaluation are acceptable and the results can be used to support the requests for alternatives. Considering that: (1) the RPV flange hole can tolerate a flaw 77 percent through-wall, (2) a postulated flaw of 0.2 inch will grow to only 0.205 inch for 80 years of operation, and (3) past examination results show no reportable indications for the U.S. fleet of RPVs, the NRC staff concludes that the requests to eliminate the RPV flange bolt hole threads requirement provides an acceptable level of quality and safety.

4.0 CONCLUSION

As set forth above, the NRC staff determines that proposed alternative provides an acceptable level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1). Therefore, the NRC staff authorizes the alternative at VEGP, Units 1 and 2, and FNP, Unit 1, until the remainder of the third ISI interval for VEGP, Units 1 and 2 and the remainder of the fourth ISI interval for FNP, Unit 1.

All other ASME Code, Section XI requirements for which relief was not specifically requested and approved in the subject requests for alternative remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: Simon Sheng

Date: January 26, 2017

If you have any questions, please contact the Project Manager, Shawn Williams, at 301-415-1009 or by e-mail at Shawn.Williams@nrc.gov.

Sincerely,

/RA/

Michael T. Markley, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-424, 50-425, 50-348

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Safety Evaluation

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