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

December 26, 2017

Mr. Steven Capps
Senior Vice President
Nuclear Corporate
Duke Energy
526 South Church Street, EC-07H
Charlotte, NC 28202

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 1; CATAWBA NUCLEAR STATION, UNIT NO. 2; SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1; MCGUIRE NUCLEAR STATION, UNIT NOS. 1 AND 2; OCONEE NUCLEAR STATION, UNIT NOS. 1, 2 AND 3; AND H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 – ALTERNATIVE TO INSERVICE INSPECTION REGARDING REACTOR PRESSURE VESSEL THREADS IN FLANGE INSPECTION (CAC NOS. MF9513 - MF9521; EPID L-2017-LLR-0019)

Dear Mr. Capps:

By letter dated March 29, 2017, as supplemented by letter dated August 9, 2017, Duke Energy Carolinas, LLC and Duke Energy Progress, LLC (the licensee) submitted a request in accordance with Part 50, Section 55a(z)(1) of Title 10 of the *Code of Federal Regulations* (10 CFR) for a proposed alternative to the requirements of 10 CFR 50.55a and the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code) for Brunswick Steam Electric Plant, Unit No. 1; Catawba Nuclear Station, Unit No. 2; Shearon Harris Nuclear Power Plant, Unit 1; McGuire Nuclear Station, Unit Nos. 1 and 2; Oconee Nuclear Station, Unit Nos. 1, 2, and 3; and H. B. Robinson Steam Electric Plant, Unit No. 2. The August 9, 2017, letter revised the request in the March 29, 2017, letter and supersedes the March 29, 2017, letter in its entirety.

The proposed alternative would allow the licensee to eliminate the examination of threads in the reactor pressure vessel flange, required by ASME Code, Section XI, Examination Category B-G-1, Item Number B6.40, at each of the requested facilities. Specifically, pursuant to 10 CFR 50.55a(z)(1), the licensee requested to use the alternative on the basis that it will provide an acceptable level of quality and safety.

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the subject request and concludes, as set forth in the enclosed safety evaluation, 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NRC staff authorizes the use of the proposed alternative for the facilities requested in the licensee's application, as superseded, for the duration of the applicable 10-year inservice inspection intervals.

If you have any questions, please contact the Duke Fleet Project Manager, Dennis Galvin at 301-415-6256 or via e-mail at Dennis Galvin@nrc.gov.

Sincerely,

for

Undine Shoop, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-325, 50-414, 50-400,
50-369, 50-370, 50-269,
50-270, 50-287, and 50-261

Enclosure:
Safety Evaluation

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

PROPOSED ALTERNATIVE TO ELIMINATE

EXAMINATION OF THREADS IN REACTOR PRESSURE VESSEL FLANGE

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1;

CATAWBA NUCLEAR STATION, UNIT 2;

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1;

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2;

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT 2

DUKE ENERGY CAROLINAS, LLC AND DUKE ENERGY PROGRESS, LLC

DOCKET NOS. 50-325, 50-414, 50-400, 50-369,

50-370, 50-269, 50-270, 50-287, 50-261

1.0 INTRODUCTION

By letter dated March 29, 2017, as supplemented by letter dated August 9, 2017 (References 1 and 2), Duke Energy Carolinas, LLC and Duke Energy Progress, LLC (the licensee) submitted Relief Request 17-GO-001 in accordance with Part 50, Section 55a(z)(1) of Title 10 of the *Code of Federal Regulations* (10 CFR) for a proposed alternative to the requirements of 10 CFR 50.55a and the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code) for Brunswick Steam Electric Plant, Unit No. 1; Catawba Nuclear Station, Unit No. 2; Shearon Harris Nuclear Power Plant, Unit 1; McGuire Nuclear Station, Unit Nos. 1 and 2; Oconee Nuclear Station, Unit Nos. 1, 2, and 3; and H. B. Robinson Steam Electric Plant, Unit No. 2. By electronic correspondence dated July 11, 2017 (Reference 3), the U.S. Nuclear Regulatory Commission (NRC) staff issued requests for additional information (RAIs) regarding the licensee's application. By letter dated August 9, 2017 (Reference 2), the licensee issued Relief Request 17-GO-001, Revision 1, which supersedes the March 29, 2017, letter (Reference 1) in its entirety and contains responses to the NRC staff's RAIs.

The proposed alternative would allow the licensee to eliminate the examination of threads in the reactor pressure vessel (RPV) flange required by Examination Category B-G-1, Item No. B6.40, in Section XI, "Rules for Inservice Inspection (ISI) of Nuclear Power Plant Components," of the ASME Code for the 10-year ISI interval at each of the requested facilities listed in the application and shown in Section 3.1.2 of this safety evaluation (SE). Specifically, pursuant to 10 CFR 50.55a(z)(1), the licensee requested to use the alternative on the basis that it will provide an acceptable level of quality and safety.

2.0 REGULATORY EVALUATION

The regulations in 10 CFR 50.55a(g)(4) state, in part, that ASME Code Class 1, 2, and 3 components (including supports) must meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in Section XI of the applicable editions and addenda of the ASME Code to the extent practical within the limitations of design, geometry, and materials of construction of the components. The threads in the RPV flange are categorized as ASME Code Class 1 components. Therefore, per 10 CFR 50.55a(g)(4), ISI of these threads must be performed in accordance with Section XI of the applicable edition and addenda of the ASME Code.

The regulations in 10 CFR 50.55a(z) state, in part, that alternatives to the requirements in paragraphs (b) through (h) of 10 CFR 50.55a may be authorized by the NRC if the licensee demonstrates that: (1) the proposed alternative provides an acceptable level of quality and safety, or (2) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

3.0 TECHNICAL EVALUATION

3.1 Licensee's Request

3.1.1 ASME Code Components Affected

For each requested facility, the proposed alternative applies to threads in the RPV flange subject to ASME Code, Section XI, Examination Category B-G-1, Item No. B6.40.

3.1.2 Applicable Code Edition and Addenda

The licensee identified the applicable ASME Code editions and addenda for each facility as shown in the table below. In addition, the table shows the applicable 10-year ISI interval, including the start and end dates.

PLANT	ASME CODE, SECTION XI EDITION OF RECORD	ISI INTERVAL	START	END
Brunswick Steam Electric Plant, Unit 1	2001 Edition, through 2003 Addenda	Fourth	5/11/2008	5/10/2018
Catawba Nuclear Station, Unit 2	2007 Edition, through 2008 Addenda	Fourth	8/19/2015	2/24/2026
Shearon Harris Nuclear Power Plant, Unit 1 (1)	2001 Edition, through 2003 Addenda	Third	5/2/2007	5/1/2018
McGuire Nuclear Station, Units 1 and 2	2007 Edition, through 2008 Addenda	Fourth	12/1/11 (Unit 1) 7/15/14 (Unit 2)	11/30/21 (Unit 1) 12/14/24 (Unit 2)
Oconee Nuclear Station, Units 1, 2, and 3	2007 Edition, through 2008 Addenda	Fifth	7/15/2014	7/15/2024
H. B. Robinson Steam Electric Plant, Unit 2	2007 Edition, through 2008 Addenda	Fifth	7/21/2012	7/30/2021

Note (1): In the March 29, 2017, application for Shearon Harris Nuclear Power Plant, Unit 1, the licensee requested the proposed alternative for two consecutive 10-year ISI intervals: the remainder of the third interval and the entire fourth interval. By letter dated August 9, 2017 (Reference 2), the licensee superseded the original request for Shearon Harris Nuclear Power Plant, Unit 1 by proposing to use the alternative for only the remainder of the third 10-year ISI interval in response to the NRC staff's RAI-1.

3.1.3 Applicable Code Requirement

The licensee has requested an alternative to the examination requirements in Examination Category B-G-1, Item No. B6.40, which is listed in Table IWB-2500-1, "Examination Categories" of the ASME Code, Section XI. This item requires the licensee to perform, every ISI interval, a volumetric examination of all the threads in the RPV flange stud holes as indicated in Figure IWB-2500-12, "Closure Stud and Threads in Flange Stud Hole," of the ASME Code, Section XI.

3.1.4 Licensee's Proposed Alternative and Basis for Use

The licensee is proposing to eliminate the examination of threads in the RPV flange, as required by Examination Category B-G-1, Item No. B6.40, of the ASME Code, Section XI, for the 10-year ISI intervals of each facility listed in Section 3.1.2 of this SE. The licensee's request is based on an evaluation by the Electric Power Research Institute (EPRI) documented in EPRI Technical Report No. 3002007626 (EPRI report), "Nondestructive Evaluation: Reactor Pressure Vessel Threads in Flange Examination Requirements," dated March 2016 (Reference 4). Relief Request 17-GO-001, Revision 1 included information from the EPRI report regarding the generic stress analysis and flaw tolerance evaluation, with additional plant-specific information to demonstrate applicability of the EPRI results. Additionally, Relief Request 17-GO-001, Revision 1 included information from the EPRI report regarding operating experience and potential degradation mechanisms for the threads in the RPV flange.

3.2 <u>NRC Staff's Evaluation</u>

The licensee relied on the EPRI report for the technical basis for the proposed alternative to eliminate examination of the threads in the RPV flange. The NRC staff focused its evaluation of the proposed alternative on the deterministic stress analyses and flaw tolerance evaluation in the EPRI report, but also considered operating experience and potential degradation mechanisms. Each of these topics were discussed in the EPRI report and in Relief Request 17-GO-001, Revision 1.

By letter dated January 26, 2017 (Reference 5), the NRC staff authorized Southern Nuclear Operating Company, Inc. (SNC) to use a similar alternative at Vogtle Electric Generating Plant, Units 1 and 2, and Joseph M. Farley Nuclear Plant, Unit 1, which was also based on the generic stress analysis and flaw tolerance evaluation in the EPRI report. Section 3.2.1 of the SE for the SNC authorization (SNC SE) documents the NRC staff's evaluation of the EPRI report and concludes that EPRI's generic stress analysis and flaw tolerance evaluation are acceptable, and the results can be used to support the elimination of the RPV threads in flange examination. For Duke Energy's proposed alternative, the NRC staff relied on this previous evaluation and focused on the plant-specific RPV flange thread information to determine if EPRI's generic stress analysis and flaw tolerance evaluation are applicable to the Duke Energy facilities.

3.2.1 Operating Experience

The EPRI report included the results of a survey of U.S. nuclear reactors taken in 2015 and early 2016 of the volumetric examination results for threads in RPV flanges (Table 4 "Summary of Survey Results – US Fleet" of Enclosure 2 of Relief Request 17-GO-001, Revision 1). The survey included 33 boiling-water reactor (BWR) units and 61 pressurized-water reactor (PWR) units. The total number of examinations for all 94 units is 10,662 with no reportable indications. The NRC staff finds that these survey results offer ample supporting evidence that the threads

in the RPV flange are performing their function without a credible threat to the structural integrity of the RPV flange.

3.2.2 Potential Degradation Mechanisms

Section 5, "Evaluation of Potential Degradation Mechanisms," of the EPRI report provides an evaluation of the susceptibility of the threads in the RPV flange to the following degradation mechanisms: general corrosion, galvanic corrosion, de-alloying corrosion, velocity phenomena, crevice corrosion, pitting, intergranular attack, corrosion fatigue, stress corrosion cracking, mechanical fatigue, thermal fatigue, mechanical wear, creep, and stress relaxation. The EPRI report concluded that the only potential degradation mechanisms applicable to the threads in the RPV flange are mechanical and thermal fatigue. To address the potential for mechanical or thermal fatigue, the licensee referred to the generic stress analysis and flaw tolerance analysis in the EPRI report.

The NRC agrees that mechanical and/or thermal fatigue are the only potential degradation mechanisms for the threads in RPV flange at the licensee's facilities. The other degradation mechanisms listed in the EPRI report (e.g., stress corrosion cracking and creep) are not credible degradation mechanisms for the threads in the RPV flange because they are not in contact with the reactor coolant and they are not in the operating temperature range where metal creep can occur.

3.2.3 Stress Analysis

Section 6.1, "Stress Analysis," of the EPRI report describes the determination of stresses at the critical location in the threads in the RPV flange. These stresses were used as input into the flaw tolerance evaluation, which is discussed in Section 3.2.4 of this SE. The stress analysis was performed using a three-dimensional, symmetric finite element model (FEM) of a portion of the threads in the RPV flange, RPV shell immediately below the flange, and a symmetric half of an RPV stud. Geometric parameters, such as number of RPV studs, stud diameter, RPV inside diameter, and flange thickness at the threads, were used to create the FEM. The loads applied to the FEM were the preload on the RPV studs, internal pressure, and thermal loads due to heatup and cooldown transients.

In the SNC SE, the NRC staff concluded that the generic EPRI stress analysis is acceptable and that the resulting stresses can be used in the subsequent flaw tolerance evaluation. For Duke Energy's proposed alternative, the NRC staff relied on its previous evaluation and conclusion regarding the generic EPRI stress analysis, and focused on the plant-specific RPV flange thread information to determine the applicability of the generic stress analysis for the Duke Energy facilities.

Finite Element Model

As discussed in the EPRI report, bounding geometric parameters were used to create an FEM. The EPRI report states that the PWR design (as opposed to a BWR design) was used as a representative geometry for the FEM because of its higher design pressure and temperature. The licensee's request is for eight PWR units and one BWR unit. The licensee listed the relevant geometric parameters of threads in RPV flange of the nine Duke Energy units and compared them with the geometric parameters used in FEM in the EPRI report. The NRC staff reviewed the comparison and determined that the geometric parameters of the FEM in the EPRI report represents the geometric parameters of the threads in RPV flange of the nine Duke

Energy units. However, the NRC staff was concerned that the thread geometry of the one BWR unit could be significantly different than the thread geometry of the PWR used in the FEM in the EPRI report. Therefore, by electronic correspondence dated July 11, 2017 (Reference 3), the NRC staff requested in RAI-2 how the PWR thread geometry (specifically, the thread pitch [number of threads per inch] and depth [distance from crest to root]) bounds or is representative of a BWR thread geometry. By letter dated August 9, 2017 (Reference 2), the licensee provided the thread pitch and depth of the one BWR unit and concluded that the thread pitch and depth used in the EPRI report bounds or is representative of the one Duke Energy BWR unit. The NRC staff reviewed the licensee's comparison. The thread pitch of the BWR unit is eight threads per inch, identical to the thread pitched used in the PWR FEM in the EPRI report. The thread depth of the BWR unit is 0.06766 inch, compared to 0.06500 inch for the PWR FEM in the EPRI report.

The NRC staff determined that the difference between the thread depth for the one BWR unit and the PWR FEM in the EPRI report is too small to have any significant impact on the final results of the stress analysis and flaw tolerance analysis. The NRC staff made this determination as follows. By conceptualizing a thread as a crack, where thread depth is equivalent to crack depth, the increase in the stress intensity factor (K_I) due to the deeper threads can be quantified. K_I is the crack driving force and is proportional to stress, geometric shape factor, and the square root of crack depth. For the same stress and geometric shape factor, the increase in K_I due to the deeper threads can be calculated by the square root of the ratio of the deeper thread depth to the thread depth used in the EPRI report ($\sqrt{(0.06766/0.06500)} = 1.02$), which gives an increase in K_I of 2 percent. This increase is negligible compared to the limiting margin of 55 percent of the allowable K_I to the applied K_I (discussed further in Section 3.2.4 of this SE). Additionally, the NRC staff noted that the EPRI analysis has enough conservatisms, such as large postulated flaw sizes, that the small increase in K_I due to the deeper threads is negligible. Therefore, RAI-2 is resolved.

The NRC staff determined that the PWR FEM described in the EPRI report is acceptable for the licensee's BWR unit because (1) the thread pitch for the licensee's BWR unit is the same value used in the EPRI report, and (2) the difference in the thread depth of the licensee's BWR unit has negligible impact in the analysis results.

Applied Loads

Table 2, "Comparison Duke Energy Plant Parameters to Bounding Values Used in Analysis," of Enclosure 2 of Relief Request 17-GO-001, Revision 1 included geometric parameters for each of the Duke Energy units and compared them to the bounding values used in the EPRI calculation of preload stress on the RPV studs. The NRC staff verified the geometric parameters the licensee provided against docketed information (if available), such as the plant updated final safety analysis reports (UFSARs), and the values of the calculated preload stress for each of the nine Duke Energy units. The NRC staff determined that the 42,338 pounds per square inch (psi) preload stress used in the EPRI analysis bounds the calculated preload stress for each of the licensee's facilities. However, the NRC staff was concerned that the calculated preload stress for each facility could be lower than the *actual* preload stress in the RPV studs for each facility, and that therefore the 42,338 psi preload stress in the EPRI report may not be bounding. Therefore, by electronic correspondence dated July 11, 2017 (Reference 3), the NRC staff requested the licensee in RAI-3 to confirm that the actual preload stress applied to the RPV studs of each of the nine Duke Energy units is equal to or less than 42,338 psi. By letter dated August 9, 2017 (Reference 2), the licensee stated that except for two of the units, the actual preload stress applied to the RPV studs is less than 42,338 psi. For the two excepted

units, the NRC staff calculated that there could be up to 4.7-percent increase in preload stress compared to the value used in the EPRI report. The NRC staff noted, however, referring to the licensee's response to RAI-4, that the two excepted units were not the limiting units with regards to the value of applied K_I compared to the allowable K_I. As mentioned in the discussion of the FEM, even for the limiting unit, the allowable K_I is 55-percent higher than the applied K_I (discussed further in Section 3.2.4 of this SE). Therefore, the NRC staff determined that although the increase in preload stress for the two excepted units could be up to 4.7 percent, the threads in RPV flange of the two excepted units would still be flaw tolerant because the applied K_I due to the higher preload stress would remain below the allowable K_I. Therefore, RAI-3 is resolved.

The stress analysis in the EPRI report evaluated reactor heatup, but not a reactor cooldown. In the SNC SE, the NRC staff found that the use of heatup or cooldown has no effect on the fatigue crack growth calculation (evaluated in Section 3.2.4 of this SE for the Duke Energy units) because it would produce the same stress range in the calculation. The EPRI thermal transient analysis assumed a 100 degrees Fahrenheit per hour heatup rate for the reactor coolant until the operating temperature was reached. The heatup rate is acceptable because it is greater than or equal to the maximum allowed reactor coolant heatup rate specified in the technical specifications or UFSAR, as applicable, of each of the nine Duke Energy units.

Based on the above, the NRC staff determined that the applied loads used in the EPRI stress analysis are acceptable for the nine Duke Energy units.

3.2.4 Flaw Tolerance Evaluation

Section 6.2, "Flaw Tolerance Evaluation," of the EPRI report describes how the applied K_I was determined. The flaw tolerance analysis, which includes the crack growth analysis, is based on the principles of linear elastic fracture mechanics. The stresses in the region of the root of the threads in the FEM were used to determine the critical location based on the largest tensile axial stress. A flaw was simulated by inserting crack tip elements in the FEM originating from this critical location, which enabled K_I to be determined. The flaw was modeled around the critical thread and orientated such that the axial stresses act normal to the face of the flaw. Four flaw depths were modeled to determine the variation of K_I with flaw depth, and the maximum applied K_I was compared to the maximum value allowed by subarticle IWB-3600, "Analytical Evaluation of Flaws," of the ASME Code, Section XI. A flaw growth evaluation was then performed with a postulated initial flaw size at the root of the critical thread to show that the structural integrity of the threads in the RPV flange was not compromised for 80 years of plant life.

In the SNC SE, the NRC staff documented its conclusion that the generic EPRI flaw tolerance evaluation is acceptable. For Duke Energy's proposed alternative, the NRC staff relied on its previous evaluation and conclusion, and focused on the plant-specific RPV flange thread information to determine the applicability of the generic flaw tolerance analysis to the Duke Energy facilities.

The generic EPRI flaw tolerance evaluation included simulations of a postulated flaw of four sizes inserted into the FEM to determine K_I due to preload, internal pressure, and heat-up transient. The maximum applied K_I around the postulated flaw was determined for each flaw depth for two load cases: (1) preload only and (2) preload with heat-up and pressure. The first case occurs during tensioning of the RPV bolts, and the second case occurs during reactor heatup to operating temperature and pressure. The EPRI report identified a maximum applied K_I of 17.4 kilopounds per square inch-square root inch (ksi$\sqrt{in}$) for the first case and 19.8 ksi$\sqrt{in}$

for the second case. The maximum applied K_I of 19.8 ksi$\sqrt{in}$ is less than the allowable value of 69.6 ksi$\sqrt{in}$, which is based on the RPV flange fracture toughness (K_{IC}) value in the upper shelf temperature range, which includes operating temperature. The K_{IC} value is from the lower bound K_{IC} curve applicable to ferritic steels in Appendix A to the ASME Code, Section XI. Since the maximum applied K_I is less than the allowable value, the NRC staff determined that the threads in RPV flange are reasonably flaw tolerant at operating temperatures.

The EPRI report does not include a comparison of the maximum applied K_I value of 17.4 ksi$\sqrt{in}$ for the preload case to the allowable value of K_I at the temperature appropriate for the preload case. Therefore, by electronic correspondence dated July 11, 2017 (Reference 3), the NRC staff requested the licensee in RAI-4 to provide a comparison of K_I with $K_{IC}/\sqrt{10}$ for the preload case for the most limiting RPV flange threads of the nine Duke Energy units. By letter dated August 9, 2017 (Reference 2), the licensee stated that for the preload case, H. B. Robinson Steam Electric Plant, Unit No. 2 has the most limiting allowable value of K_I (17.0 ksi$\sqrt{in}$). The licensee determined this value using the lower bound K_{IC} curve in Appendix A to the ASME Code, Section XI, with a safety factor of $\sqrt{10}$. The licensee noted that this limiting allowable value of K_I is slightly less than the maximum applied K_I value of 17.4 ksi$\sqrt{in}$ for the preload case in the EPRI report and referred to the SNC SE, which used a safety factor of 2 instead of $\sqrt{10}$ to calculate the allowable value of K_I. The NRC staff determined that a safety factor of 2 is appropriate for evaluating postulated flaws, as compared to a safety factor of $\sqrt{10}$ for detected flaws, and is consistent with the analytical procedures used for establishing pressure-temperature limit curves. With a safety factor of 2, the licensee calculated a limiting allowable value of K_I of 26.97 ksi$\sqrt{in}$, which is greater than the maximum applied K_I value of 17.4 ksi$\sqrt{in}$. The NRC staff verified the licensee's calculations and found them to be acceptable. Therefore, RAI-4 is resolved. Accordingly, the NRC staff determined that the threads in RPV flange are reasonably flaw tolerant at preload temperatures.

The SNC SE stated that for a postulated flaw of 0.2 inches from the root of thread, the crack would grow by 0.005 inch over 80 years of reactor operation. The 0.005 inch crack growth was provided in a letter dated October 24, 2016 (Reference 6) during the review of the SNC request. The NRC staff concluded in the SNC SE that this amount of crack growth was acceptable. For the current evaluation, the NRC staff determined this crack growth length is bounding, using the fatigue crack growth curves in Figure A-4300-1 in ASME Code, Section XI, Appendix A. The crack growth evaluation in the EPRI report also assumed 50 reactor heatup/cooldown cycles per year and 5 bolt preloads per year. The NRC staff confirmed that these assumptions are conservative for Duke Energy's facilities.

3.2.5 Technical Conclusion

The NRC staff determined that the licensee has demonstrated that the deterministic stress analysis and flaw tolerance evaluation in the EPRI report are bounding for the threads in RPV flange for each of the licensee's facilities. Therefore, the NRC staff determined that elimination of the ASME Code-required examination of threads in the RPV flange at the Duke Energy facilities is acceptable because the licensee has provided reasonable assurance of structural integrity of the threads in RPV flange without these examinations for the duration of the applicable 10-year ISI interval listed in Section 3.1.2 of this SE.

4.0 CONCLUSION

As set forth above, the NRC staff determined that the licensee's proposed alternative to eliminate the ASME Code-required examination of threads in the RPV flange for the applicable 10-year ISI interval 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 use of the proposed alternative for the facilities requested in the licensee's application, as superseded, for the duration of the applicable 10-year ISI interval listed in Section 3.1.2 of this SE.

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

5.0 REFERENCES

1. Donahue, J., Duke Energy, letter to U.S. Nuclear Regulatory Commission, "Proposed Alternative in Accordance with 10 CFR 50.55a(z)(1) for Examination of ASME Section XI, Examination Category B-G-1, Item Number B6.40, Threads in Flange (17-GO-001)," March 29, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17088A846).

2. Donahue, J., Duke Energy, letter to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information (RAI) Regarding 10 CFR 50.55a(z)(1) Proposed Alternative to ASME Section XI Threads In Flange Examination (17-GO-001)," August 9, 2017 (ADAMS Accession No. ML17221A305).

3. Galvin, D. J., U.S. Nuclear Regulatory Commission, e-mail to Zaremba, A., Duke Energy, "Duke Energy Fleet RAIs - Alternative for Reactor Pressure Vessel Flange Threads Examination (MF9513 to MF9521)," July 11, 2017 (ADAMS Accession No. ML17192A484).

4. Electric Power Research Institute (EPRI), EPRI Technical Report No. 3002007626, "Nondestructive Evaluation: Reactor Pressure Vessel Threads in Flange Examination Requirements," March 2016 (ADAMS Accession No. ML16221A068).

5. Markley, M. T., U.S. Nuclear Regulatory Commission, letter to Pierce, C. R., Southern Nuclear Operating Company, Inc., "Vogtle Electric Generating Plant, Units 1 and 2, and Joseph M. Farley Nuclear Plant, Unit 1 - Alternative to Inservice Inspection Regarding Reactor Pressure Vessel Threads in Flange Inspection (CAC Nos. MF8061, MF8062, MF8070)," January 26, 2017 (ADAMS Accession No. ML17006A109).

6. Pierce, C. R., Southern Nuclear Operating Company, Inc., letter to U.S. Nuclear Regulatory Commission, "Joseph M. Farley Nuclear Plant, Unit 1 [and] Vogtle Electric Generating Plant, Units 1 and 2 - Response to Request for Information Regarding Proposed Inservice Inspection," October 24, 2016 (ADAMS Accession No. ML16298A049).

Principal Contributor: David Dijamco

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ADAMS Accession No. ML17331A086 *by memo (ML17292A381)

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