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

June 26, 2017

Mr. Bryan C. Hanson
Senior Vice President
Exelon Generation Company, LLC
President and Chief Nuclear Officer (CNO)
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: BRAIDWOOD STATION, UNITS 1 AND 2; BYRON STATION, UNIT NOS. 1 AND 2; CALVERT CLIFFS NUCLEAR POWER PLANT, UNITS 1 AND 2; CLINTON POWER STATION, UNIT NO. 1; DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3; LIMERICK GENERATING STATION, UNITS 1 AND 2; NINE MILE POINT NUCLEAR STATION, UNITS 1 AND 2; PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3; QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2; R. E. GINNA NUCLEAR POWER PLANT; AND THREE MILE ISLAND NUCLEAR STATION, UNIT 1 — PROPOSED ALTERNATIVE TO ELIMINATE EXAMINATION OF THREADS IN REACTOR PRESSURE VESSEL FLANGE (CAC NOS. MF8712–MF8729 AND MF9548)

Dear Mr. Hanson:

By application dated October 31, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16306A270), Exelon Generation Company, LLC (the licensee) submitted a request in accordance with Paragraph 50.55a(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, "Codes and standards," for Braidwood Station, Units 1 and 2; Byron Station, Unit Nos. 1 and 2; Calvert Cliffs Nuclear Power Plant, Units 1 and 2; Dresden Nuclear Power Station, Units 2 and 3; Limerick Generating Station, Units 1 and 2; Nine Mile Point Nuclear Station, Units 1 and 2; Peach Bottom Atomic Power Station, Units 2 and 3; Quad Cities Nuclear Power Station, Units 1 and 2; R. E. Ginna Nuclear Power Plant; and Three Mile Island Nuclear Station, Unit 1. By letter dated April 3, 2017 (ADAMS Accession No. ML17093A883), the licensee provided additional information, revised its application, and expanded its request to include Clinton Power Station, Unit No. 1.

The proposed alternative would allow the licensee to eliminate the examination of threads in the reactor pressure vessel flange, required by Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), 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 at the facilities requested in the licensee's application, as supplemented, for the duration of the applicable 10-year inservice inspection interval, as specified in the licensee's April 3, 2017, letter, or until the NRC approves an applicable alternative in NRC Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," or other document.

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.

If you have any questions, please contact Blake Purnell at 301-415-1380 or via e-mail at Blake.Purnell@nrc.gov.

Sincerely,

David J. Wrona, Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. STN 50-456, STN 50-457, STN 50-454, STN 50-455, 50-317, 50-318, 50-461, 50-237, 50-249, 50-352, 50-353, 50-220, 50-410, 50-277, 50-278, 50-254, 50-265, 50-244, and 50-289

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

PROPOSED ALTERNATIVE TO ELEMINATE

EXAMINATION OF THREADS IN REACTOR PRESSURE VESSEL FLANGE

BRAIDWOOD STATION, UNITS 1 AND 2;

BYRON STATION, UNIT NOS. 1 AND 2;

CALVERT CLIFFS NUCLEAR POWER PLANT, UNITS 1 AND 2;

CLINTON POWER STATION, UNIT NO. 1;

DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3;

LIMERICK GENERATING STATION, UNITS 1 AND 2;

NINE MILE POINT NUCLEAR STATION, UNITS 1 AND 2;

PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3;

QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2;

R.E. GINNA NUCLEAR POWER PLANT; AND

THREE MILE ISLAND NUCLEAR STATION, UNIT 1.

EXELON GENERATION COMPANY, LLC

DOCKET NOS. STN 50-456, STN 50-457, STN 50-454, STN 50-455, 50-317, 50-318, 50-461,

50-237, 50-249, 50-352, 50-353, 50-220, 50-410,

50-277, 50-278, 50-254, 50-265, 50-244, AND 50-289

1.0 INTRODUCTION

By application dated October 31, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16306A270), Exelon Generation Company, LLC (Exelon, the licensee) submitted a request in accordance with Paragraph 50.55a(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, "Codes and standards," for Braidwood Station, Units 1 and 2; Byron Station, Unit Nos. 1 and 2; Calvert Cliffs Nuclear Power Plant, Units 1 and 2; Dresden Nuclear Power Station, Units 2 and 3; Limerick Generating Station, Units 1 and 2; Nine Mile Point Nuclear Station, Units 1 and 2; Peach Bottom Atomic Power Station, Units 2 and 3; Quad Cities Nuclear Power Station, Units 1 and 2; R. E. Ginna Nuclear Power Plant; and Three Mile Island Nuclear Station (TMI), Unit 1. The U.S. Nuclear Regulatory Commission (NRC) staff requested additional information regarding the licensee's application by e-mail dated March 3, 2017 (ADAMS Accession No. ML17062A491). By letter dated April 3, 2017 (ADAMS Accession No. ML17093A883), the licensee provided additional information, revised its application, and expanded its request to include Clinton Power Station, Unit No. 1.

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 American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), 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.

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 an 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 Examination Category B-G-1, Item No. B6.40, in Section XI of the ASME Code.

3.1.2 Applicable Code Edition and Addenda

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

PLANT	ISI INTERVAL	ASME CODE EDITION	START	END
Braidwood Station, Unit 1	3rd	2001 Edition, through 2003 Addenda	7/29/2008	7/28/2018
Braidwood Station, Unit 2	3rd	2001 Edition, through 2003 Addenda	10/17/2008	10/16/2018
Byron Station, Unit Nos. 1 and 2	4th	2007 Edition, through 2008 Addenda	7/16/2016	7/15/2025
Calvert Cliffs Nuclear Power Plant, Units 1 and 2	4th	2004 Edition	10/10/2009	6/30/2019
Clinton Power Station, Unit No. 1	3rd	2004 Edition	7/1/2010	6/30/2020
Dresden Nuclear Power Station, Units 2 and 3	5th	2007 Edition, through 2008 Addenda	1/20/2013	1/19/2023
Limerick Generating Station, Units 1 and 2	4th	2007 Edition, through 2008 Addenda	2/1/2017	1/31/2027
Nine Mile Point Nuclear Station Unit No. 1	4th	2004 Edition	8/23/2009	8/22/2019
Nine Mile Point Nuclear Station, Unit 2	3rd	2004 Edition	4/5/2008	6/15/2018
Peach Bottom Atomic Power Station, Units 2 and 3	4th	2001 Edition, through 2003 Addenda	11/5/2008	12/31/2018
Quad Cities Nuclear Power Station, Units 1 and 2	5th	2007 Edition, through 2008 Addenda	4/2/2013	4/1/2023
R. E. Ginna Nuclear Power Plant	5th	2004 Edition	1/1/2010	12/31/2019
Three Mile Island Nuclear Station, Unit 1	4th	2004 Edition	4/20/2011	4/19/2022

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 RPV flange stud holes as shown 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 RPV flanges, required by Examination Category B-G-1, Item No. B6.40, of the ASME Code, Section XI, for the duration of the current 10-year ISI intervals for each facility, or until the NRC approves an applicable alternative in NRC Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," or other document. 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 (ADAMS Accession No. ML16221A068). The licensee's submittals 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. The submittals also included information from the EPRI report regarding operating experience and potential degradation mechanisms for threads in RPV flanges.

Additionally, the licensee described maintenance activities it performs, each time the RPV closure head is removed, to detect and mitigate general degradation prior to returning the reactor to service. Specifically, the licensee stated that the threads in the RPV flange are inspected for damage, cleaned, and lubricated prior to reinstallation of the RPV studs.

3.2 NRC Staff's Evaluation

The licensee relied on the EPRI report for the technical basis for the proposed alternative to eliminate examination of threads in RPV flanges. 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 the licensee's submittals.

By letter dated January 26, 2017 (ADAMS Accession No. ML17006A109), 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 safety evaluation (SE) for the SNC authorization (SNC SE) documents the 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 eliminating the examination of threads in the RPV flange. For Exelon's proposed alternative, the 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 is applicable to the Exelon 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 regarding the volumetric examination results for threads in RPV flanges (Table 3 of the licensee's submittals). 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 threads in RPV flanges are performing their function without a credible threat to the structural integrity of the RPV flanges.

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, thermal fatigue, mechanical 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 thermal fatigue are the only potential degradation mechanisms for the threads in the RPV flange at each of the licensee's facilities. The other degradation mechanisms listed in the EPRI report 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 in the FEM were the preload on the RPV studs, internal pressure, and thermal loads.

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 Exelon's proposed alternative, the 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 Exelon 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 8 PWR units and 11 BWR units. The licensee stated that not all of its units are bounded by the geometric parameters used in the EPRI report; however, the preload stress for each unit (calculated from the geometric parameters) are bounded by the preload stress used in the EPRI report. The NRC staff considers the FEM to be acceptable for the licensee's PWR units because the FEM is based on bounding geometric parameters for a PWR design. However, the acceptability of the FEM for the licensee's BWR units requires further evaluation.

In its April 3, 2017, letter, the licensee provided the thread pitch (number of threads per inch) and depth (distance from crest to root) for each of its BWR units, and concluded that the thread pitch and depth used in the EPRI report bounds or is representative of the thread pitch and depth for its BWR units. The thread pitch for the licensee's BWR units is 8 threads per inch, which is identical to the thread pitched used in the FEM. The thread depth for the licensee's BWR units ranges from 0.06345 inches to 0.06765 inches, but the FEM assumes a thread depth of 0.06500 inches

The NRC staff determined that the differences between the thread depth for the licensee's BWR units and the FEM are too small to have any significant impact on the final results of the stress analysis and flaw tolerance analysis. 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 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 ($\sqrt{(0.06765/0.06500)} = 1.02$), which gives an increase in K_I of 2 percent. This increase is negligible compared to the limiting margin of 58 percent of the allowable K_I to the applied K_I (discussed in Section 3.2.4 of this SE). Additionally, the 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 are negligible.

The NRC staff determined that the FEM described in the EPRI report is acceptable for the licensee's BWR units because (1) the thread pitch for the licensee's BWR units is the same as used in the EPRI report and (2) the variation in thread depths for the licensee's BWR units has negligible impact in the analysis results.

Applied Loads

The licensee's April 3, 2017, letter (Attachment 2, Table 1) provided geometric parameters for each of the Exelon units and compared them to the bounding values used in the EPRI calculation of preload stress on the RPV studs. Using the updated final safety analysis reports (UFSARs) for the plants, the NRC staff verified some of the geometric parameters and the corresponding value of the calculated preload stress on the RPV studs. The staff determined that the calculated preload stress on the studs of 42,338 pounds per square inch (psi) in the EPRI analysis bounds the calculated preload stress on the studs for each of the licensee's facilities. Although TMI has an actual applied stress on the studs of 46,000 psi (Section 4.3.3.f of TMI UFSAR), the staff determined this to be acceptable as discussed in Section 3.2.4 of this SE.

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 Exelon 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 pressure-temperature limits report, as applicable, for each of the licensee's facilities.

Based on the above, the NRC staff determined that the applied loads used in the EPRI stress analysis are acceptable for the Exelon facilities.

3.2.4 Flaw Tolerance Evaluation

Section 6.2, "Flaw Tolerance Evaluation," of the EPRI report describes how the crack driving force or K_I due to the applied loads was determined. The flaw tolerance evaluation, including 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 Exelon's proposed alternative, the 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 Exelon 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 heatup 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 heatup 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 root inch (ksi√in) for the first case and 19.8 ksi√in for the second case. The maximum applied K_I of 19.8 ksi√in is less than the allowable value of 69.6 ksi√in, which is based on the RPV flange fracture toughness (K_{IC}) value at operating temperature. The K_{IC} value is from the lower bound K_{IC} curve applicable to ferritic steels (Figure A-4200-1) in Appendix A, "Analysis of Flaws," 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 flanges 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√in for the preload case to the allowable value of K_I at the temperature appropriate for the preload case. By letter dated April 3, 2017, the licensee stated that, for the preload case, TMI has the most limiting allowable value of K_I (17.0 ksi√in) for its facilities. The licensee determined this value using the analytic approximations for the K_{IC} curve in Appendix A to the ASME Code, Section XI, with a safety factor of √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√in for the preload case in the EPRI report. Therefore, consistent with the approach discussed in the SNC SE, the licensee applied a safety factor of 2 instead of √10 to the K_{IC} value. The NRC staff determined that a safety factor of 2 is appropriate for evaluating postulated flaws, as compared to a safety factor of √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 determined a limiting allowable value of K_I of 27.5 ksi√in, which is greater than the maximum applied K_I value of 17.4 ksi√in. The NRC staff verified the licensee's calculations and found them to be acceptable.

For TMI, which has the most limiting allowable value of K_I, the UFSAR states that the membrane tensile stress in the studs is approximately 46,000 psi, which is greater than the preload stress of 42,338 psi calculated by EPRI. The NRC staff determined that this slightly higher tensile stress (8.6 percent higher) in the studs does not impact the comparison of the maximum applied K_I with the limiting allowable value of K_I, because the allowable K_I (27.5 ksi√in) is 58 percent higher than the maximum applied K_I (17.4 ksi√in). Based on this, the staff determined that the threads in the RPV flange for each of the licensee's facilities are reasonably flaw tolerant at preload temperatures.

The licensee stated that some of its units have closure heads in service with one missing stud. The licensee stated that the expected increase in applied K_I (less than 2 percent) during service conditions with one inoperable stud would still be less than the allowable K_I, because the allowable K_I of 69.6 ksi√in in service (preload with heatup and pressure) is much higher than maximum applied K_I value of 19.8 ksi√in. The NRC staff agrees with this assessment because the licensee's estimated increase in applied K_I due to one stud missing is reasonable, and the allowable K_I is approximately 3.5 times greater than the applied K_I.

In the SNC SE, it states that the EPRI evaluation determined, for a postulated flaw of 0.2 inches, that the crack would grow by 0.005 inches over 80 years of reactor operation.[1] 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, "Reference Fatigue Crack Growth Curves for Carbon and Low Alloy Ferritic Steels Exposed to Air Environments (Subsurface Flaws)," in the 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 staff confirmed that these assumptions are conservative for Exelon'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 the RPV flange for each of the licensee's facilities. Therefore, the staff determined that elimination of the ASME Code-required examination of threads in the RPV flange at each of the Exelon facilities is acceptable, because the licensee has provided reasonable assurance of structural integrity of the threads in the RPV flanges 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 not perform ASME Code-required examination of threads in RPV flanges provides an acceptable level of quality and safety. Accordingly, the 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 supplemented, for the duration of the applicable 10-year ISI interval listed in Section 3.1.2 of this SE, or until the NRC approves an applicable alternative in NRC Regulatory Guide 1.147 or other document.

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.

Principal Contributor: David Dijamco, NRR/DE/EVIB

Date of issuance: June 26, 2017

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[1] The amount of crack growth was provided in an SNC letter dated October 24, 2016 (ADAMS Accession No. ML16298A049).

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