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

March 15, 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 – REQUEST FOR RELIEF FROM THE REQUIREMENTS OF THE AMERICAN SOCIETY OF MECHANICAL ENGINEERS BOILER AND PRESSURE VESSEL CODE (ASME CODE) (CAC NOS. MF8191 AND MF8192)

Dear Mr. Hanson:

By letter dated July 21, 2016, as supplemented by letters dated November 21, 2016, and February 15, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML16203A081, ML16327A061 and ML17046A225, respectively), Exelon Generation Company, LLC (EGC or the licensee) proposed an extension of the inservice inspection (ISI) interval requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, for Braidwood Station (Braidwood), Units 1 and 2.

Specifically, in lieu of the requirements of the ASME Code, Section XI, IWB-2412, "Inspection Program B," Relief Request (RR) I3R-17 proposed an alternative pursuant to Section 50.55a(z)(1) of Title 10 of the *Code of Federal Regulations* (10 CFR) to extend the ISI interval for examinations of the reactor pressure vessel (RPV) welds (Category B-A) as well as the nozzle-to-vessel welds and inner radius sections (Category B-D) from 10 to 20 years 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, as set forth in the enclosed safety evaluation, that the 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 use of the proposed alternative in RR I3R-17 until the end of the extended third interval, which is July 27, 2028, for Braidwood, Unit 1, and October 15, 2028, for Braidwood, Unit 2. Further, the NRC staff determined that the alternative examination dates for Categories B-A and B-D components are acceptable: Spring 2027 for Unit, 1 and Fall 2027 for Unit 2.

All other requirements of the ASME Code, Section XI, for which relief has not been specifically requested remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

If you have any questions, please contact the Senior Project Manager, Joel S. Wiebe at 301-415-6606 or via e-mail at Joel.Wiebe@nrc.gov.

Sincerely,

Kimberly J. Green for

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

Enclosure:
Safety Evaluation

cc w/encl: Distribution via ListServ

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST I3R-17 REGARDING

THIRD TEN-YEAR INSERVICE INSPECTION PROGRAM INTERVAL

EXELON GENERATION COMPANY, LLC

BRAIDWOOD STATION, UNITS 1 AND 2

DOCKET NOS. 50-456 AND 50-457

1.0 INTRODUCTION

By letter dated July 21, 2016, as supplemented by letters dated November 21, 2016, and February 15, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML16203A081, ML16327A061 and ML1717046A225, respectively), Exelon Generation Company LLC (EGC or the licensee) proposed an extension of the inservice inspection (ISI) interval requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, for Braidwood Station (Braidwood), Units 1 and 2. Relief Request (RR) I3R-17 is for both units and is contained in the Attachment to the July 21, 2016, letter.

Specifically, in lieu of the requirements of the ASME Code, Section XI, IWB-2412, "Inspection Program B," Relief Request I3R-17 proposed an alternative pursuant to Section 50.55a(z)(1) of Title 10 of the *Code of Federal Regulations* (10 CFR) to extend the ISI interval for examinations of the reactor pressure vessel (RPV) welds (Category B-A) as well as the nozzle-to-vessel welds and inner radius sections (Category B-D) from 10 to 20 years.

The current third 10-year interval ends on July 28, 2018, for Unit 1 and October 16, 2018, for Unit 2.

1.1 REGULATORY EVALUATION

2.1 Regulations and Guidance

In accordance with 10 CFR 50.55a(g)(4), the licensee is required to perform ISI of ASME Code Class 1, 2, and 3 components and system pressure tests during the first 10-year interval and subsequent 10-year intervals that comply with the requirements in the latest edition and addenda of Section XI of the ASME Code, incorporated by reference in 10 CFR 50.55a(b), subject to the limitations and modifications listed therein.

For the third 10-year ISI intervals at Braidwood, the code of record for the inspection of ASME Code Class 1, 2, and 3 components is the 2001 Edition through the 2003 Addenda of the ASME Code, Section XI. The regulation in 10 CFR 50.55a(z) states, in part, that alternatives to the requirements of paragraphs 10 CFR 50.55a(b) through (h) or portions thereof may be used when authorized by the Director, Office of Nuclear Reactor Regulation, or Director, Office of New Reactors, as appropriate. There are two justifications for an alternative to be authorized: (1) *Acceptable level of quality and safety,* and (2) *Hardship without a compensating increase in quality and safety.*

Regulatory Guide (RG) 1.99, Revision (Revision) 2, "Radiation Embrittlement of Reactor Vessel Materials," describes general procedures acceptable to the staff for calculating the effects of neutron radiation embrittlement of the low-alloy steels currently used for light-water-cooled RPVs.

RG 1.174, Revision 1, "An Approach For Using Probabilistic Risk Assessment In Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," describes a risk-informed approach, acceptable to the NRC, for assessing the nature and impact of proposed licensing basis changes by considering engineering issues and applying risk insights.

RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," describes methods and assumptions acceptable to the staff for determining the RPV neutron fluence.

2.2 Background

The ISI of Categories B-A and B-D components consists of visual and ultrasonic examinations intended to discover whether flaws have initiated, whether pre-existing flaws have extended, and whether pre-existing flaws may have been missed in prior examinations. These examinations are required to be performed at regular intervals, as defined in Section XI of the ASME Code.

2.3 Summary of WCAP-16168-NP-A, Revision 2

In June 2008, the Pressurized Water Reactor Owners Group (PWROG) issued the NRC-approved topical report WCAP-16168-NP-A, Revision 2, "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval" (ADAMS Accession No. ML082820046), which is in support of a risk-informed assessment of extensions to the ISI intervals for Categories B-A and B-D components. Specifically, WCAP-16168-NP-A, Revision 2 performed studies on three different pressurized-water reactor (PWR) plants (referred to as the pilot plants): Westinghouse, Combustion Engineering (CE), and Babcock and Wilcox (B&W) for PWR nuclear power plants in the United States to justify the proposed extension of the ISI interval for Categories B-A and B-D components from 10 to 20 years.

The analyses in WCAP-16168-NP-A, Revision 2 used probabilistic fracture mechanics (PFM) approach and inputs from the work described in NUREG-1806, "Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61)" (ADAMS Accession No. ML061580318) and NUREG-1874, "Recommended Screening Limits for Pressurized Thermal Shock (PTS)" (ML070860156). The PWROG analyses incorporated the effects of fatigue crack growth and ISI. Design basis transient data was used as input to the fatigue crack growth evaluation. The effects of ISI were modeled consistent with a previously-approved PFM code in WCAP-14572-NP-A, "Westinghouse Owners Group Application of Risk-

Informed Methods to Piping Inservice Inspection" (ADAMS Accession Nos. ML012630327, ML012630349, and ML012630313). These effects were considered in the PFM evaluations, using the Fracture Analysis of Vessels - Oak Ridge (FAVOR) computer code (ADAMS Accession No. ML042960391). All other inputs were identical to those used in the PTS risk re-evaluation underlying 10 CFR 50.61a.

From the results of the studies for the pilot plants, the PWROG concluded that the ASME Code, Section XI, 10-year inspection interval for Categories B-A and B-D components in PWR RPVs can be extended to 20 years. This conclusion was considered to apply to any plant designed by the three vendors as long as the critical, plant-specific parameters (defined in Appendix A of WCAP-16168-NP-A, Revision 2) are bounded by the pilot plants.

2.4 Summary of the July 26, 2011, NRC Safety Evaluation (SE) for WCAP-16168-NP-A, Revision 2

The original SE in WCAP-16168-NP-A, Revision 2 that was published in 2008, was superseded by the July 26, 2011, SE (ADAMS Accession No. ML111600303) to address the PWROG's request for clarification of the information needed in applications utilizing WCAP-16168-NP-A, Revision 2. The NRC staff concluded in the July 26, 2011, SE that the methodology presented in WCAP-16168-NP-A, Revision 2 is consistent with RG 1.174, Revision 1 and is acceptable for referencing in requests to implement alternatives to ASME Code inspection requirements for PWR plants in accordance with the limitations and conditions in the SE. In addition to showing that the subject plant parameters and inspection history are bounded by the critical parameters identified in Appendix A in WCAP-16168-NP-A, Revision 2, the licensee's application must provide the following plant-specific information:

(1) Licensees must demonstrate that the embrittlement of their RPV is within the envelope used in the supporting analyses. Licensees must provide the 95th percentile total through-wall cracking frequency ($TWCF_{TOTAL}$) and the supporting material properties at the end of the period in which the relief is requested to extend the ISI from 10 to 20 years. The 95th percentile $TWCF_{TOTAL}$ must be calculated using the methodology in NUREG-1874. The RT_{MAX-X} and the shift in the Charpy transition temperature produced by irradiation defined at the 30 ft-lb energy level, ΔT_{30}, must be calculated using the latest revision of RG 1.99 or other NRC-approved methodology.

(2) Licensees must report whether the frequency of the limiting design basis transients during prior plant operation are less than the frequency of the design basis transients identified in the PWROG fatigue analysis as significant contributors to fatigue crack growth.

(3) Licensees must report the results of prior ISI of RPV welds and the proposed schedule for the next 20-year ISI interval. The 20-year inspection interval is a maximum interval. In its request for an alternative, each licensee shall identify the years in which future inspections will be performed. The dates provided must be within plus or minus one refueling cycle of the dates identified in the implementation plan provided to the NRC in PWROG letter OG-10-238 (ADAMS Accession No. ML11153A033).

(4) Licensees with B&W plants must: (a) verify that the fatigue crack growth of 12 heat-up/cool-down transients per year that was used in the PWROG fatigue analysis

bound the fatigue crack growth for all of its design basis transients and (b) identify the design bases transients that contribute to significant fatigue crack growth.

(5) Licensees with RPVs having forgings that are susceptible to underclad cracking and with $RT_{MAX\text{-}FO}$ values exceeding 240 °F must submit a plant-specific evaluation to extend the inspection interval for ASME Code, Section XI, Categories B-A and B-D, RPV welds from 10 to a maximum of 20 years because the analyses performed in the WCAP-A are not applicable.

(6) Licensees seeking second or additional interval extensions shall provide the information and analyses requested in Section (e) of 10 CFR 50.61a.

WCAP-16168-NP-A, Revision 3, which contains this latter SE for WCAP-16168-NP-A, Revision 2, was issued in October 2011 (ADAMS Accession No. ML11306A084, referred to as the WCAP-A in the rest of this SE).

3.0 PROPOSED ALTERNATIVES FOR BRAIDWOOD, UNITS 1 AND 2

3.1 Description of Proposed Alternatives

In RR I3R-17, the licensee proposed to defer the ASME Code required Categories B-A and B-D weld ISI for Braidwood, Units 1 and 2 until 2027, plus or minus one refueling outage, consistent with the schedule proposed in the revision to PWROG letter OG-10-238.

3.2 Components for Which Relief is Requested

The affected components are the subject plant RPVs and their interior attachments and core support structures. The following examination categories and item numbers from IWB-2500 and Table IWB-2500-1 of the ASME Code, Section XI, are addressed in RR I3R-17:

Exam Category	Item Number	Description
B-A	B1.11	Circumferential Shell Welds
B-A	B1.21	Circumferential Head Welds
B-A	B1.30	Shell-to-Flange Weld
B-D	B3.90	Nozzle-to-Vessel Welds
B-D	B3.100	Nozzle Inner Radius Section

3.3 Basis for Proposed Alternatives

The licensee stated that the methodology used to demonstrate the acceptability of extending the inspection intervals for examination of Categories B-A and B-D items, based on negligible change in risk, is contained in the WCAP-A. The critical parameters for demonstrating that the Westinghouse pilot plant analysis in the WCAP-A is applicable to Braidwood, Units 1 and 2, are shown in Table 2 of the submittal. This methodology used the estimated TWCF as a measure of the risk of RPV failure, and it was demonstrated that the inspection interval for the affected components can be extended from 10 to 20 years, meeting the change in risk guidelines in RG 1.174. The licensee addressed the plant-specific information discussed in Section 2.4 of this SE as follows:

(1) The total TWCFs were calculated as 2.09E-16 event/year for Braidwood, Unit 1, and 2.04E-15 event/year for Braidwood, Unit 2, which are less than the value of 1.76E-08 for the Westinghouse pilot plant in the WCAP-A.

(2) The frequencies of the Braidwood RPVs' limiting design basis transients are bounded by the frequency of 7 heatup/cooldown cycles/year for the PWROG fatigue analysis in the WCAP-A.

(3) The results of the previous RPV inspections for the Braidwood units are provided, which confirm that satisfactory examinations have been performed on the Braidwood RPVs. The RPV examinations currently scheduled for 2018 for Braidwood, Units 1 and 2 will be deferred until 2027, plus or minus one refueling cycle, consistent with the latest implementation plan in PWROG letter OG-10-238, dated July 12, 2010.

The licensee did not address plant-specific information items (4), (5), and (6) because they do not apply to Braidwood, Units 1 and 2. Since Braidwood units are bounded by the corresponding pilot plant parameters, the licensee concluded that application of the WCAP-A methodology for Braidwood, Units 1 and 2 is acceptable.

3.4 Duration of Proposed Alternatives

The request is applicable to Braidwood, Units 1 and 2, ISI program for the extended third interval with the next ASME Categories B-A and B-D RPV weld inspections scheduled for 2027 for both units.

4.0 STAFF TECHNICAL EVALUATION

The licensee followed the guidance in Appendix A in WCAP-16168-NP-A, Revision 2, and provided in Table 2 of the submittal, a summary of the critical parameters for demonstrating applicability of the bounding analysis to Braidwood, Units 1 and 2. These critical parameters are the subject of the three plant-specific information items that are identified in Section 2.4 of this SE. Therefore, the NRC staff's evaluation is focused on the three plant-specific information items addressed by the licensee.

The NRC staff first reviewed Plant-Specific Information 1 regarding TWCFs. In Table 2 of the relief request under "Through-Wall Cracking Frequency (TWCF)," it was stated that the calculated total TWCFs are 2.09E-16 events per year for Unit 1 and 2.04E-15 events per year for Unit 2. These TWCF values were obtained by the licensee using the WCAP-A methodology with inputs from Tables 4-1 (Unit 1) and 4-2 (Unit 2) of the relief request. Tables 4-1 and 4-2 used RG 1.99, Revision 2, Position 1.1 to calculate ΔT_{30} (shift + unirradiated nil-ductility-transition reference temperature RT_{NDT}) at the end of extended operation for 57 effective full power year (EFPY) for all RPV beltline materials for the Braidwood units. The NRC staff confirmed that the neutron fluence at 57 EFPY, chemistry factors, and the unirradiated RT_{NDT} values for all RPV materials are consistent with their corresponding values in the license renewal application (LRA) dated May 29, 2013 (ADAMS Accession No. ML13155A421) that was approved in 2015 for the Braidwood units. The NRC staff independently verified that the licensee's calculated ΔT_{30} values and the resulting TWCFs for both units, as identified above, are accurate. Based on these verified results, the NRC staff determined that they are acceptable for this application because the licensee's calculated total TWCFs for both units are several orders of magnitude lower than the value of 1.76E-08 for the Westinghouse pilot plant in the WCAP-A. Hence, the NRC staff determined that the licensee has addressed Plant-Specific

Information 1 satisfactorily and confirmed that the embrittlement of the Braidwood RPVs is within the envelope used in the Westinghouse pilot plant analysis.

The NRC staff then reviewed Plant-Specific Information 2 regarding the frequency of the limiting design basis transients. In Table 2 of the relief request under "Frequency and Severity of Design Basis Transients," the licensee stated that the heatup/cooldown cycles per year for Braidwood, Units 1 and 2, are bounded by the WCAP-A. The NRC staff found that this information is consistent with the 60-year cycle projection for reactor coolant system transients listed in Table 4.3.1-4 of the LRA. Therefore, the NRC staff determined that the licensee has addressed Plant-Specific Information 2 satisfactorily and confirmed that, regarding design basis transients, the WCAP-A methodology is applicable to the Braidwood units. Also, the Braidwood, Units 1 and 2, RPVs have a single-layer cladding on the inside like the assumption used in the WCAP-A analysis.

Lastly, the NRC staff reviewed Plant-Specific Information 3 regarding the results of prior ISI of RPV welds and the proposed schedule for the next 20-year ISI interval. Tables 3-1 (Unit 1) and 3-2 (Unit 2) in the relief request contains additional information pertaining to previous RPV inspections and the schedule for future ones. Specifically, Tables 3-1 and 3-2 indicated that the three indications for Braidwood, Unit 1, and two for Braidwood, Unit 2, are acceptable per Table IWB-3510-1 of Section XI of the ASME Code. To verify flaw acceptance, the NRC staff issued a request for additional information. In its response dated November 21, 2016, the licensee provided dimensions for the five indications. The NRC staff performed evaluations using Table IWB-3510-1 parameters based on these flaw and component thickness dimensions and confirmed that all five flaws have met the acceptance criteria of this table.

Regarding the 10 CFR 50.61a requirements on allowable flaws within the inner 1/10th of the RPV wall thickness, the licensee stated that one indication associated with Braidwood, Unit 2, is within the thickness range and is required to be evaluated. This indication is characterized as an embedded circumferential flaw, 0.1 inches from the cladding-metal interface, 2.1 inches long, and 0.125 inch in through-wall extent. In its submittal dated July 21, 2016, the licensee provided Table 3-2, showing "scaled maximum number of forging flaws" and the number of detected flaws at various ranges of wall depth. Since the derivation of the scaled maximum number of flaws based on plant-specific information was not given, the NRC staff requested additional information. In its response dated November 21, 2016, the licensee provided the inspection area for Braidwood, Unit 2, RPV. Based on this information, the NRC staff confirmed that the licensee developed the scaled maximum number of forging flaws from the table in 10 CFR 50.61a appropriately. The licensee compared this information with the number of relevant detected flaws associated with Braidwood, Unit 2, RPV and demonstrated that this 10 CFR 50.61a requirement is satisfied.

Because both ASME Code, Section XI and 10 CFR 50.61a requirements regarding detected flaws are met, the NRC staff determined that the licensee has addressed the first part of Plant-Specific Information 3 satisfactorily.

The licensee proposed to conduct the next inspection for Braidwood, Units 1 and 2 in 2027 (+/- one refueling outage). However, if this RR is approved, the third ISI interval will be extended to July 27, 2028, for Unit 1 and October 15, 2028, for Unit 2. As such, the proposed inspection date of 2027, plus or minus one refueling outage, could potentially be beyond the extended interval. The NRC staff requested additional information for clarification. In its response dated November 21, 2016, the licensee clarified that, "EGC confirms that the subject weld examinations will be performed no later than the completion of the Spring 2027 refueling

outage (A1R26) for Unit 1; and Fall 2027 refueling outage (A2R26) for Unit 2." Since the RPV inspections will be conducted prior to the proposed ISI extension dates of July 27, 2028, for Unit 1 and October 15, 2028 for Unit 2, the NRC staff determined that the licensee has addressed the second part of Plant-Specific Information 3 adequately.

In summary, the NRC staff has reviewed the licensee's submittal and the responses to the NRC staff's requests for additional information supplementing the relief request. In addition, the NRC staff performed independent calculations to verify the input data and output results in Tables 4-1 and 4-2 of the RR. The difference between the licensee's and staff's calculated 95th percentile $TWCF_{TOTAL}$ is insignificant. With this information, the NRC staff concluded that the 95th percentile $TWCF_{TOTAL}$ value in Table 2 of the RR is bounded by the WCAP-A results. Consequently, the licensee has demonstrated that the proposed alternative will provide an acceptable level of quality and safety and meets the guidance provided by RG 1.174, Revision 1 for risk-informed decisions.

5.0 CONCLUSION

The NRC staff has completed its review of RR I3R-17 for Braidwood, Units 1 and 2. The staff concludes that increasing the ISI interval for Categories B-A and B-D components from 10 to 20 years will result in no appreciable increase in risk. This conclusion is based on the fact that the plant-specific information provided by the licensee is bounded by the data in the WCAP-A and the request meets all the conditions and limitations described in the NRC SE in WCAP-A. Therefore, RR I3R-17 provides an acceptable level of quality and safety, and the alternatives can be authorized for Categories B-A and B-D components pursuant to 10 CFR 50.55a(z)(1) until the end of the extended third interval, which is July 27, 2028, for Braidwood, Unit 1, and October 15, 2028, for Braidwood, Unit 2. Further, the NRC staff determined that the alternative examination dates for Categories B-A and B-D components are acceptable: Spring 2027 for Unit, 1 and Fall 2027 for Unit 2.

All other requirements of the ASME Code, Section XI, for which relief has not been specifically requested remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

Principle Reviewer: Simon Sheng

Date: March 15, 2017

SUBJETC: BRAIDWOOD STATION, UNITS 1 AND 2 – REQUEST FOR RELIEF FROM THE REQUIREMENTS OF THE AMERICAN SOCIETY OF MECHANICAL ENGINEERS BOILER AND PRESSURE VESSEL CODE (ASME CODE) (CAC NOS. MF8191 AND MF8192) DATED MARCH 15, 2017

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