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

November 8, 2016

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: BYRON STATION, UNIT NO. 2 – REQUEST I4R-08, RELIEF FROM THE REQUIREMENTS OF THE ASME CODE (CAC NO. MF7695)

Dear Mr. Hanson:

By letter dated April 15, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16106A116), as supplemented by letter dated September 29, 2016 (ADAMS Accession No. ML16273A201), Exelon Generation Company, LLC (Exelon) submitted relief request I4R-08 to the U.S. Nuclear Regulatory Commission (NRC), requesting approval to extend American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code, Section XI, 2007 Edition with the 2008 Addenda-required volumetric examination of specific Byron Station, Unit No. 2, reactor pressure vessel (RPV) full penetration pressure retaining Examination Category 8-A and B-D welds fourth inspection interval from 10 to 20 years.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(1), the licensee requested to use the proposed alternative on the basis that the alternative provides an acceptable level of quality and safety.

The NRC staff has reviewed the subject request and concludes, as set forth in the enclosed safety evaluation (SE), that Exelon has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1). The NRC staff concludes that Byron, Unit No. 2, RPV is bounded by WCAP-16168-NP-A, "Risk-Informed Extension of Reactor Vessel In-Service Inspection Interval," Revision 3, dated October, 2011 (ADAMS Accession No. ML11306A084), and the request met all of the provisions set forth in WCAP-16168-NP-A and as described in the NRC staff's July 26, 2011, SE included in WCAP-16168-NP-A. Therefore, the proposed alternative provides an acceptable level of quality and safety and the requested interval extension from 10 to 20 years (from no later than 2017 to no later than 2027), is approved.

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

Sincerely,

G. Edward Miller, Acting Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No.: STN 50-455

Enclosure:
Safety Evaluation

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST I4R-08 REGARDING VOLUMETRIC EXAMINATION OF

REACTOR VESSEL FULL PENETRATION PRESSURE RETAINING EXAMINATION

CATEGORY 8-A AND B-D WELDS

EXELON GENERATION COMPANY, LLC

BYRON STATION, UNIT NO. 2

DOCKET NO.STN 50-455

1.0 INTRODUCTION

By letter dated April 15, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16106A116), as supplemented by letter dated September 29, 2016 (ADAMS Accession No. ML16273A201), Exelon Generation Company, LLC (Exelon) submitted relief request I4R-08 to the U.S. Nuclear Regulatory Commission (NRC or Commission), requesting approval to extend American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, 2007 Edition with the 2008 Addenda-required volumetric examination of specific Byron Station, Unit No. 2, reactor pressure vessel (RPV) full penetration pressure retaining Examination Category 8-A and B-D welds fourth inspection interval from 10 to 20 years.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(1), the licensee requested to use the proposed alternative on the basis that the alternative provides an acceptable level of quality and safety.

2.0 REGULATORY EVALUATION

As required by 10 CFR 50.55a(g), inservice inspection (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. 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. This authorization has been delegated to the Division of Operating Reactor Licensing Branch Chiefs. 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.

Enclosure

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 the 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 Section 50.55a(b)].

RG 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," dated May 1988 (ADAMS Accession No. ML003740284) describes general procedures acceptable to the NRC 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" (ADAMS Accession No. ML023240437), 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.

By letter dated May 8, 2008 (ADAMS Accession No. ML081060045), the NRC staff issued a final safety evaluation (SE) which found that Topical Report WCAP-16168-NP, Revision 2 (the WCAP), "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval," is acceptable for referencing in licensing applications for pressurized-water reactors (PWRs) designed by Westinghouse Electric Company (Westinghouse), Combustion Engineering, Inc., and Babcock and Wilcox, Inc. (B&W). The WCAP was developed to support a risk-informed assessment of extensions to the ISI intervals for ASME Code, Section XI, Examination Category B-A and B-D components, from 10 to 20 years using data from three different PWR plants (referred to as the pilot plants) representing each of the vendors.

The analyses in the WCAP used probabilistic fracture mechanics tools 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)," March 1, 2007 (ADAMS Accession No. ML070860156). The Pressurized-Water Reactor Owners Group (PWROG) analyses incorporated the effects of fatigue crack growth and ISI data. Design basis transient data was used as an input for the fatigue crack growth evaluation. The effects of ISI data were modeled consistently with the previously-approved probabilistic fracture mechanics codes contained 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 inputs into the evaluations performed with the "Fracture Analysis of Vessels - Oak Ridge" (FAVOR) computer code. All other inputs were identical to those used in the PTS risk re-evaluation underlying 10 CFR 50.61a, "Alternative Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events."

The PWROG concluded, as a result of these studies, that the ASME Code, Section XI, 10-year ISI interval for Examination Category B-A and B-D components in PWR RPVs can be safely extended from 10 to 20 years. This conclusion, based on the results from the pilot plant analyses, was considered to apply to any plant designed by the three PWR vendors represented in the pilot plant study, as long as certain critical plant-specific criteria (defined in Appendix A of the WCAP) are bounded by the analysis for the applicable pilot plant.

The NRC staff issued a second SE dated July 26, 2011 (ADAMS Accession No. ML111600303), superseding the initial SE in the WCAP, which addressed the PWROG's request for clarification of the information needed in applications utilizing the WCAP. In this SE, the staff concluded that the methodology presented in the WCAP is consistent with the guidance provided in 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 specified in the SE. In addition to showing that the subject plant is bounded by the pilot plants/parameters identified in Appendix A in the WCAP, licensees that submit a request for an alternative based on the WCAP need to submit 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 it's 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 total TWCF ($TWCF_{95\text{-}TOTAL}$) must be calculated using the methodology in NUREG-1874. The $RT_{MAX\text{-}X}$[1] and the shift in the Charpy transition temperature produced by irradiation defined at the 30 ft-lb energy level, ΔT_{30} [the shift in the Charpy transition temperature produced by irradiation defined at the 30 ft-lb energy level], must be calculated using the methodology documented in the latest revision of RG 1.99 or other NRC-approved methodology. RTMAX-X is the material property which characterizes the reactor vessel's resistance to fracture initiating from flaws in plates ($RT_{MAX\text{-}PL}$), forgings ($RT_{MAX\text{-}FO}$), axial welds and circumferential welds ($RT_{MAX\text{-}AW/CW}$).

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 that are considered to significantly contribute 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

[1] $RT_{MAX\text{-}X}$ is a material property, which characterizes the RVs resistance to fracture initiating from flaws in welds, plates, and forgings. The method of determining $RT_{MAX\text{-}X}$ is described in Sections (f) and (g) of 10 CFR 50.61a, "Alternate fracture toughness requirements for protection against pressurized thermal shock."

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 bounds 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-FO} values exceeding 240 °F must submit a plant-specific evaluation to extend the inspection interval for ASME Code, Section XI, Category B-A and B-D RPV welds from 10 to a maximum of 20 years because the analyses performed in the WCAP 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 the SE for the WCAP, was issued in October 2011 (ADAMS Accession No. ML11306A084, referred to as the WCAP-A, in the rest of this SE).

The licensee has requested an alternative to the ASME Code requirements pursuant to 10 CFR 50.55a(z)(1). Byron, Unit No. 2, fourth 10-year ISI interval is based on the ASME Code, Section XI, 2007 Edition with the 2008 Addenda.

Based on the above, and subject to the following technical evaluation, the NRC staff finds that regulatory authority exists for the licensee to request, and the Commission to authorize, the alternative proposed by the licensee. The end date for the current Byron, Unit No. 2, fourth 10-year interval ISI program is July 15, 2025.

3.0 TECHNICAL EVALUATION

3.1 The Licensee's Proposed Alternative

The licensee proposes to defer the ASME Code required Categories B-A and B-D weld ISI for Byron, Unit No. 2, until 2027. The proposed ISI date is consistent with the schedule proposed in the PWROG Letter OG-10-238 (ADAMS Accession No. ML11153A033), the latest NRC-staff reviewed implementation plan for the PWROG plants.

The affected component is the Byron Unit No. 2, RPV. The following examination categories and item numbers from IWB-2500 and Table IWB-2500-1 of the ASME Code, Section XI, are addressed in this request:

Examination 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-A	B1.40	Head-to-Flange Weld

Examination Category	Item Number	Description
B-D	B3.90	Nozzle-to-Vessel Welds
B-D	B3.100	Nozzle Inside Radius Section

Additionally, Table 1 of the licensee's submittal lists the component identification (exam area) for each specific weld.

The basis for the proposed alternative is WCAP-A. Plant-specific parameters for Byron, Unit No 2, are summarized in Tables 1 – 4 of Attachment 4 to the licensee's letter dated April 15, 2016. The format of the information is patterned after that found in Appendix A of the WCAP-A. All of the critical parameters listed in Tables 1, 2, and 3 of the enclosure to the submittal are bounded by the WCAP-A Westinghouse pilot plant.

3.2 NRC Staff Evaluation

The NRC reviewed the licensee's proposal to extend the Byron, Unit No. 2, ISI interval in order to determine whether the licensee met the risk-informed criteria set forth in the WCAP-A for a Westinghouse plant. By showing that Byron, Unit No. 2, is bounded by the Westinghouse pilot plant analysis with respect to the six criteria discussed in Section 3.2 of this SE, the licensee would have a sufficient technical basis for extending the ISI in accordance with the provisions of the WCAP-A. Byron, Unit No. 2, RPV has a single layer cladding and is bounded by the Westinghouse pilot plant basis.

Table 4 of the licensee's submittal provided the TWCF of the limiting forging and circumferential weld, along with the parameters necessary to perform the calculations. The licensee utilized the methodology provided in RG 1.99, Revision 2, to calculate the shift in the Charpy transition temperature produced by irradiation defined at the 30 ft-lb (feet/pound) energy level, ΔT_{30}. The licensee reported that the $TWCF_{95\text{-}TOTAL}$ for Byron, Unit No. 2, was 3.75×10^{-16} per year. The NRC staff performed an independent calculation to confirm the licensee's reported value and found a $TWCF_{95\text{-}TOTAL}$ of 3.81×10^{-16} per year. Ultimately, the differences in the values are negligible and both calculations are well below the Westinghouse pilot plant bounding value of 1.76×10^{-8} per year; therefore, the staff finds the $TWCF_{95\text{-}TOTAL}$ for Byron, Unit No. 2, acceptable.

With regard to the frequency and severity of design basis transients, the licensee was required to show that Byron, Unit No. 2, heatup/cooldown transients are bounded by that of the Westinghouse pilot plant basis (7 heatup/cooldown cycles per year). Table 4.3.1-1 of Byron, Unit No. 2's, License Renewal Application shows that there had been 64 heatups and 63 cooldowns through March 31, 2012, correlating to approximately 2.5 cycles per year, which is less than the 7 cycles per year limit. Therefore, the NRC staff finds that the frequency of the limiting design basis transients during prior plant operation are less than the frequency of the Westinghouse design basis transients identified in the PWROG fatigue analysis.

The licensee stated that two complete 10-year ISIs have been performed on Byron, Unit No. 2, to date. During the most recently completed ISI, two recordable indications were discovered during ultrasonic examination of the Intermediate Shell to Lower Shell Circumferential weld. Specifically, these indications were found 3.7" [in] and 5.2" below the weld centerline in the base material of the Lower Shell Forging identified as item No. 5 in Table 4 of Attachment 4 to the licensee's submittal. As detailed in the licensee's September 29, 2016, letter both flaws are axially oriented planar subsurface indications. The first indication, which lies further from the weld, is 1.1" long, has a through-wall extent of 0.22" in an area with a thickness of 8.63" and lies 0.41" from the inner diameter surface of the RPV. The indication closer to the weld is 0.6" long, has no through-wall indication, and lies 0.26" from the outer diameter surface. Both of the indications were found to be acceptable per Table IWB-3510-1 of the ASME Code, Section XI, however, they both lie within the inner $1/10^{th}$ or inner 1 inch of the reactor vessel wall. Table 3 of the alternate PTS rule, 10 CFR 50.61a, defines the maximum allowable number of flaws per 1000 square-inches of inside surface area in the inspection volume that lie within given through-wall extent (TWE [in.]) minimums and maximums. Based on the volumetric examination area, the licensee provided a scaled table containing the maximum number of flaws that can be found in the forgings based on their TWEs as well as the number of axial and circumferential forging flaws that were identified. Both of the flaws were circumferential forging flaws and were found to lie within 0.125 and 0.375 inches; one of the flaws also lies within the TWE range of 0.175 and 0.375 inches. For Byron Unit 2, the allowable number of flaws to be identified in these length ranges are 30 and 8, respectively. Therefore, the licensee has verified that the indications found are acceptable per the requirements of 10 CFR 50.61a.

Byron, Unit No. 2, fourth ten year ISI of the reactor vessel full penetration pressure-retaining Examination Category B-A and B-D welds is scheduled to be performed during the summer 2017 refueling outage, preceding the end of the fourth 10-year ISI interval, which is scheduled for July 15, 2025. The licensee proposed to perform the fourth volumetric examination during the fifth ISI interval for Byron Unit 2 no later than 2027. The proposed date is consistent with the PWROG letter OG-10-238 and the NRC staff finds the date acceptable. However, it must also be noted that the application of WCAP-A allows for an extension of the current ISI interval from 10 to 20 years, not to solely defer the required volumetric examinations to the next interval. Therefore, consistent with WCAP-A, the examinations should be conducted in accordance with the ASME Code, Section XI, requirements that are applicable during the fourth 10-year ISI interval.

Byron, Unit No. 2, is a Westinghouse plant; therefore, the fourth criterion related to the bounding fatigue crack growth for all design basis transients and identification of design basis transients that contribute to significant fatigue crack growth in B&W plants is not applicable.

The fifth criterion stated in Section 3.2 requires that plants with forgings that are susceptible to underclad cracking and with RT_{MAX-FO} values exceeding 240 °F must submit a plant-specific evaluation to extend the inspection interval because the analyses performed in the WCAP-A are not applicable. Since the RT_{MAX-FO} value calculated does not exceed 240 °F, this criterion is not applicable to this plant.

Lastly, the licensee is not currently seeking additional interval extensions, so the sixth and final criterion is not applicable.

In summary, the NRC staff concludes that the licensee's submittal demonstrates that the RPV for Byron, Unit No. 2, is bounded by the Westinghouse limitations set forth in the WCAP-A and the associated SE from the NRC staff. The NRC staff further concludes that the licensee adequately confirmed that the Byron, Unit No. 2, RPV meets all of the applicable criteria set forth in the WCAP-A.

4.0 CONCLUSION

The NRC staff has reviewed the subject request and concludes, as set forth above, that Exelon has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1). The NRC staff concludes that the Byron, Unit No. 2, RPV is bounded by WCAP-16168-NP-A, "Risk-Informed Extension of Reactor Vessel In-Service Inspection Interval," Revision 3, dated October 2011 (ADAMS Accession No. ML11306A084), and that the request met all of the provisions set forth in WCAP-16168-NP-A and as described in the NRC staff's July 26, 2011, SE, included in WCAP-16168-NP-A. Therefore, the proposed alternative provides an acceptable level of quality and safety and the requested interval extension from 10 years to 20 years (from no later than 2017 to no later than 2027), is authorized. Therefore, the examination of the Category B-A and B-D components for Byron, Unit No. 2, shall be conducted prior to the end of the extended fourth interval in accordance with the appropriate ASME Code requirements associated with the fourth 10-year ISI interval.

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

Principal Contributor: Austin Young

Date of issuance: November 8, 2016

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

Sincerely,

/RA/

G. Edward Miller, Acting Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No.: STN 50-455

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ADAMS Accession No. ML16306A366 *via e-mail

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