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

December 10, 2014

Mr. Michael J. Pacilio
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. 1 - RELIEF FROM THE REQUIREMENTS OF THE ASME CODE TO EXTEND THE REACTOR VESSEL INSERVICE INSPECTION INTERVAL (TAC NO. MF3596)

Dear Mr. Pacilio:

By letter to the U.S. Nuclear Regulatory Commission (NRC) dated March 10, 2014 (Agencywide Documents Access and Management System Accession No. ML14069A559), Exelon Generation Company, LLC submitted a request for Byron Station, Unit No. 1, for the use of alternatives to certain American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, requirements.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(a)(3)(i), 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, 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(a)(3)(i). Therefore, the NRC staff authorizes use of the proposed alternative until the expiration of the facility operating license or, if the license renewal application is approved, until the end of 2025.

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.

If you have any questions, please contact Joel S. Wiebe at 301-415-6606.

Sincerely,

Travis L. Tate

Travis L. Tate, Chief
Plant Licensing III-2 and
Planning and Analysis Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-456

Enclosure:
Safety Evaluation

cc w/encl: Distribution via ListServ

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST NO. I3R-23 REGARDING REACTOR PRESSURE VESSEL WELDS

EXELON GENERATION COMPANY, LLC

BYRON STATION, UNIT NO. 1

DOCKET NO. 50-454

1.0 INTRODUCTION

By letter dated March 10, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14069A559), Exelon Generation Company, LLC (Exelon or the licensee) proposed an alternative to the inservice inspection (ISI) interval requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, Paragraph IWB-2412, "Inspection Program B," for inservice examination of the reactor pressure vessel (RPV) welds required by Table IWB-2500-1, Examination Categories B-A, "Pressure Retaining Welds in Reactor Vessel," and B-D, "Full penetration Welded Nozzles in Vessels." The licensee's proposed alternative would extend the third 10-year ISI interval from 10 to 20 years in order to allow for the deferral of the subject RPV weld examinations.

The third 10-year ISI interval for Byron, Unit No. 1, began on July 1, 2006, and is scheduled to end on July 15, 2016. The licensee's proposed alternative would allow for the deferral of the subject RPV examinations until 2025. The U.S. Nuclear Regulatory Commission (NRC) staff reviewed Request for Alternative I3R-23 pursuant to Section 50.55a(a)(3)(i) of Title 10 of the *Code of Federal Regulations* (10 CFR) to determine whether the licensee's proposed alternative to the ISI interval requirements of the ASME Code, Section XI, will provide an acceptable level of quality and safety.

Specifically, pursuant to 10 CFR 50.55a(a)(3)(i), 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

2.1 Regulations and Guidance

In accordance with 10 CFR 50.55a(g)(4), the licensee is required to perform an 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.

Enclosure

For the third 10-year ISI interval at Byron, Unit No. 1, 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(a)(3) states, in part, that the Director of the Office of Nuclear Reactor Regulation may authorize an alternative to the requirements of 10 CFR 50.55a(g). There are two justifications for an alternative to be authorized. First, per 10 CFR 50.55a(a)(3)(i), the licensee must demonstrate that the proposed alternative would provide an acceptable level of quality and safety. For the second possible justification for an alternative to be authorized, described in 10 CFR 50.55a(a)(3)(ii), the licensee must show that following the ASME Code requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," describes general procedures acceptable to the NRC staff for calculating the effects of neutron irradiation 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 Examination Categories B-A and B-D components consists of volumetric and surface 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 Examination Categories B-A and B-D components. Specifically, WCAP-16168-NP-A, Revison 2, took data associated with three different pressurized-water reactor (PWR) plants (referred to as the pilot plants), one designed by each of the three main vendors (Westinghouse, Combustion Engineering (CE), and Babcock and Wilcox (B&W)) for PWR nuclear power plants in the United States, and performed studies on these pilot plants to justify the proposed extension of the ISI interval for the Examination 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) methodology 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)" (ADAMS Accession No. ML070860156). The PWROG analyses incorporated the effects of fatigue crack growth and ISI examination histories. Design basis transient data was used as inputs to the fatigue crack growth evaluation. The effects of ISI examination histories 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 re-evaluation underlying 10 CFR 50.61a, "Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," heretofore identified as the alternate PTS rule.

From the results of the studies, 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. Their conclusion from the results for the pilot plants 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 parameters of 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, 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's conclusion in this latter SE indicates 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 order to demonstrate 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 it's supporting material properties at the end of the period in which the alternative is requested to extend the ISI interval from 10 to 20 years. The 95th percentile $TWCF_{TOTAL}$ must be calculated using the methodology in NUREG-1874. The Charpy transition temperature for the RPV, RTMAX-X, and the shift in the Charpy transition temperature produced by irradiation defined at the 30 ft-lb energy level, ΔT30, must be calculated using the methodology documented in 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 that are considered to significantly contribute to fatigue crack growth.

(3) Licensees must report the results of prior ISI of the RPV welds and the proposed schedule for the deferred RPV weld exams for the 20-year ISI interval. The 20-year inspection interval is the maximum interval allowed per WCAP-16168-NP, Revision 2. In its request for an alternative, each licensee shall identify the years in which the deferred inspections will be performed. The dates provided in licensees' requests must be within plus or minus one refueling cycle of the dates identified in the revised implementation plan provided to the NRC in PWROG Letter OG-10-238, dated July 12, 2010 (ADAMS Accession No. ML11153A033).

(4) Licensees with B&W plants must (a) verify that the fatigue crack growth for 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 RTMAX-FO values exceeding 240 °F must submit a plant-specific evaluation to extend the inspection interval for the ASME Code, Section XI, Examination Category 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 TECHNICAL EVALUATION

3.1 The Licensee's Proposed Alternative

The licensee proposed to defer the third 10-year ISI interval RPV weld examinations for Byron, Unit No. 1, from 2015 until 2025. The subject RPV weld examinations are required by the ASME Code, Section XI, Table IWB-2500-1, Examination Categories B-A and B-D.

The licensee requested deferral of the Byron, Unit No. 1, third 10-year ISI interval examinations for the following RPV examination categories and item numbers from Table IWB-2500-1 of the ASME Code, Section XI:

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-D	B3.90	Nozzle-to-Vessel Welds

The licensee stated that, in accordance with 10 CFR 50.55a(a)(3)(i), an alternate ISI interval for the subject components is requested on the basis that the current interval can be extended based on a negligible change in risk when compared to the risk criteria specified in RG 1.174. The methodology used to demonstrate the acceptability of extending the inspection interval for the Examination Category B-A and B-D RPV welds is contained in the WCAP-A. This methodology uses the calculated $TWCF_{TOTAL}$ as a measure of the risk of RPV failure. The licensee addressed the plant-specific information discussed in Section 2.4 of this SE as follows:

(1) The licensee provided detailed $TWCF_{TOTAL}$ calculations along with critical input parameters (neutron fluence and material properties) for demonstrating that the embrittlement of the Byron, Unit No. 1, RPV is bounded by the Westinghouse pilot plant analysis. The $TWCF_{TOTAL}$ value was calculated as 2.30 x 10-14 events per year; this value is less than the value of 1.76 x 10-08 events per year for the Westinghouse pilot plant study in the WCAP-A.

(2) The licensee stated that frequency of the Byron, Unit No. 1, RPV limiting design basis transients are bounded by the frequencies identified in the PWROG fatigue analysis.

(3) The licensee provided the results of the previous RPV inspection for Byron, Unit No. 1. The next RPV inspection, which is currently scheduled to occur prior to the end of the third 10-year ISI interval in 2015, would be deferred until 2025 in accordance with the WCAP-A methodology. The licensee noted that the proposed examination date of 2025 is consistent with the latest revised implementation plan schedule for Byron, Unit No. 1, in PWROG letter OG-10-238.

Plant-Specific Information Items (4), (5), and (6) were not addressed in the licensee's application. Based on its evaluation of plant-specific information items (1), (2), and (3) above, the licensee concluded that the Byron, Unit No.1 , RPV is bounded by the Westinghouse pilot plant analysis in the WCAP-A, and therefore, the use of this proposed alternative will provide an acceptable level of quality and safety. Accordingly, the licensee requested that the NRC authorize Request for Alternative I3R-23, pursuant to 10 CFR 50.55a(a)(3)(i).

3.2 NRC Staff Evaluation

The NRC staff reviewed Request for Alternative I3R-23 to determine whether Byron, Unit No. 1, is bounded by the Westinghouse pilot plant study performed in the WCAP-A. The staff conducted its review by performing an independent evaluation of Plant-Specific Information Items (1), (2), and (3), as provided in the licensee's submittal. The staff also reviewed Plant-Specific Information Items (4), (5), and (6) to verify that they are not applicable to Byron, Unit No. 1, and, therefore, do not require a plant-specific evaluation.

Regarding Plant-Specific Information Item (1), the NRC staff reviewed the licensee's statement in Table 2 of the submittal that the plant-specific value for $TWCF_{TOTAL}$ (2.30 x 10-14 events per year) is bounded by the $TWCF_{TOTAL}$ of 1.76 x 10-08 events per year from the Westinghouse pilot plant study in the WCAP-A. Table 4 of the submittal provides the details of the $TWCF_{TOTAL}$ calculation for the Byron, Unit No. 1, RPV. The staff performed an independent calculation to verify that the licensee's $TWCF_{TOTAL}$ value was correctly determined in accordance with NUREG-1874, as required by the WCAP-A. The staff also verified that the RTMAX-X and ΔT30 values were calculated using the methodology in RG 1.99, Revision 2, consistent with the WCAP-A. Additionally, the staff reviewed the unirradiated material properties used for the $TWCF_{TOTAL}$ calculation against those used as the basis for the current reactor coolant system pressure-temperature (P-T) limit curves. These P-T limits are established for 32 effective full-power years (EFPY) in a P-T limits report (PTLR), which was submitted to the NRC by letter dated January 23, 2007 (ADAMS Accession No. ML070240261). The staff determined that the unirradiated material property inputs for the RPV beltline materials are consistent with those documented in the PTLR.

The $TWCF_{TOTAL}$ value was calculated using a bounding neutron fluence that is projected out to 57 EFPY. This neutron fluence significantly bounds the period of deferral for the subject RPV weld examinations. The 57 EFPY neutron fluence was calculated using methodologies that have been approved by the NRC, as described in WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," May 2004 (ADAMS Accession No. ML050120209), and WCAP-16083-NP-A, Revision 0, "Benchmark Testing of the FERRET Code for Least Squares Evaluation of Light Water Reactor Dosimetry," May 2006 (ADAMS Accession No. ML061600256). These methodologies conform to RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." Therefore, the licensee's neutron fluence input to the $TWCF_{TOTAL}$ calculation is acceptable.

Based on the above assessment, the NRC staff finds that the $TWCF_{TOTAL}$ value for Byron, Unit No. 1, is bounded by the WCAP-A results, and therefore, the licensee has satisfactorily addressed Plant- Specific Information Item (1) for Byron, Unit No. 1.

Regarding Plant-Specific Information Item (2), the NRC staff reviewed the licensee's statement in Table 2 of the submittal that the frequency and severity of the design basis transients for Byron, Unit No. 1, are bounded by the Westinghouse pilot plant study in the WCAP-A. The Westinghouse pilot plant study in the WCAP-A analyzed fatigue crack growth based on seven heatup and cooldown cycles per year; this was determined to be the bounding design basis transient frequency and severity for fatigue crack growth for the Westinghouse design. The staff reviewed the licensee's statement in Table 2 of the submittal against the information in Section 5.2 of the final safety analysis report (FSAR) pertaining to the number of heatup and cooldown cycles analyzed for the design of the reactor coolant system for the 40 year licensed operating term. The staff verified that the cumulative number of heatup and cooldown cycles for 40 years, as listed in the FSAR, corresponds to less than seven heatup and cooldown cycles per year from the WCAP-A. Based on its review of this information, the staff determined that there is adequate assurance that the number of heatup and cooldown cycles for Byron, Unit No. 1, will remain less than that assumed for the Westinghouse pilot plant study in the WCAP-A. Therefore, the staff finds that the licensee has satisfactorily addressed Plant-Specific Information Item (2) for Byron, Unit No. 1.

Regarding Plant-Specific Information Item (3), the NRC staff reviewed the information provided in Table 3 of the licensee's submittal pertaining to the results of previous inspections of the RPV Examination Category B-A and B-D welds. Table 3 states that two 10-year ISIs have been performed to date, and the RPV volumetric examinations that were performed during the second 10-year ISI interval met the performance demonstration requirements of Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems," of the 1995 Edition of the ASME Code, Section XI, with 1996 Addenda, as modified by 10 CFR 50.55a(b)(2)(xiv – xvi). The staff finds this information acceptable because paragraph (e), "Examination and Flaw Assessment Requirements," of the alternate PTS rule, 10 CFR 50.61a, requires that volumetric examinations performed in support of demonstrating the applicability of the rule (which is the basis for the risk-informed analysis of the WCAP-A) must use procedures, equipment, and personnel that have been qualified in accordance with Appendix VIII of the ASME Code, Section XI, as specified in 10 CFR 50.55a(b)(2)(xv). The staff's evaluation of the licensee's second 10-

year ISI interval RPV beltline volumetric examination results, relative to the acceptance criteria of 10 CFR 50.61a, are discussed below.

The licensee stated in Table 3 that there were two indications identified in the RPV beltline region circumferential weld during the second 10-year ISI interval. The licensee indicated that both indications are acceptable based on the flaw acceptance criteria specified in Table IWB-3510-1 of the ASME Code, Section XI. The licensee noted that one of these indications is located in the adjacent nozzle shell forging material within the inner one-tenth or one inch of the RPV wall thickness. Therefore, in accordance with 10 CFR 50.61a(e), this indication must be evaluated against the scaled limits for the allowable number of flaws in plates and forgings specified Table 3, "Allowable Number of Flaws in Plates and Forgings," of 10 CFR 50.61a. The licensee determined that, based on the measured through-wall extent (TWE) for the flaw, it meets the scaled limits for the maximum number of allowable flaws in forging material.

The NRC staff reviewed the licensee's assessment of the inspection results for the Byron, Unit No. 1, RPV and determined that these results meet the examination and flaw assessment requirements of 10 CFR 50.61a because the one indication that is located within the inner one inch or 10 percent of the RPV shell thickness is bounded by the 10 CFR 50.61a, Table 3, scaled acceptance criteria for the maximum number of allowable flaws in forgings, based on the measured TWE for the indication. Therefore, the staff finds that the volumetric examination results for the Byron, Unit 1, RPV are bounded by the requirements of 10 CFR 50.61a, and the risk-informed analyses of the WCAP-A are applicable.

Plant-Specific Information Item (3) states that the deferred RPV weld examination dates provided in licensees' requests to implement this alternative must be within plus or minus one refueling cycle of the dates identified in the revised implementation plan provided to the NRC in PWROG letter OG-10-238. Furthermore, the 20-year inspection interval is the maximum interval allowed for the subject RPV welds per the WCAP-A methodology. The NRC staff confirmed that the licensee's proposed examination date of 2025 corresponds to the revised implementation plan schedule in PWROG letter OG-10-238, and this date ensures that the subject RPV examinations will occur no later than 20 years from the start of the third ISI interval. Therefore, the licensee's proposed examination date of 2025 is technically acceptable. However, the current 40-year operating license for Byron, Unit No. 1, is set to expire on October 31, 2024. A license renewal application (LRA), submitted by letter dated May 29, 2013 (ADAMS Accession No. ML13155A387), proposes to extend the facility operating license for Byron, Unit 1 for an additional 20 years. This LRA is currently under NRC review. Therefore, pending approval of the LRA, this alternative would remain in effect until the expiration of the facility operating license or, if the LRA is approved, until the end of calendar year 2025.

Based on the acceptable second 10-year ISI interval volumetric examination results, as discussed above, and the acceptable deferred RPV weld examination date of 2025, the NRC staff finds that the licensee has satisfactorily addressed Plant-Specific Information Item (3) for Byron, Unit No. 1.

Regarding Plant-Specific Information Item (4), the NRC staff noted that this item is only applicable to B&W plants since this is the only plant design for which plant-specific verification of the fatigue crack growth for all design basis transients is necessary. Byron, Unit No. 1, is a Westinghouse plant, and it has already been established in the WCAP-A that the fatigue crack growth corresponding to seven heatup/cooldown cycles per year is bounding for all design basis transients, per the Westinghouse pilot plant study. Therefore, the staff finds that Plant-Specific Information Item (4) is not applicable to Byron, Unit No. 1, and no plant-specific evaluation is required for this item.

Regarding Plant-Specific Information Item (5), the NRC staff noted that the RTMAX-FO value for Byron, Unit 1, is less than 240 °F. This item is only applicable to plants with RTMAX-FO values exceeding 240 °F (regardless of the number of cladding layers). Therefore, the staff finds that Plant-Specific Information Item (5) is not applicable to Byron, Unit 1, and no plant-specific evaluation is required for this item.

Regarding Plant-Specific Information Item (6), the NRC staff finds that this item is not applicable to Byron, Unit No. 1, because the licensee's application requested deferral of the subject RPV weld examinations for only the third 10-year ISI interval.

Based on its review of the licensee's submittal, and its findings regarding Plant-Specific Information Items (1), (2), (3), (4), (5), and (6), as documented above, the NRC staff has determined that the licensee adequately demonstrated that the Byron, Unit No. 1, RPV is bounded by Westinghouse pilot plant study from the WCAP-A. Consequently, the licensee has demonstrated that the proposed alternative will provide an acceptable level of quality and safety, and it meets the guidance provided by RG 1.174, Revision 1, for risk-informed decisions.

4.0 CONCLUSION

The NRC staff has reviewed the subject request and concludes, as set forth in the enclosed SE, 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(a)(3)(i). Therefore, the staff authorizes use of the proposed alternative until the expiration of the facility operating license or, if the license renewal application is approved, until the end of 2025.

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: C. Sydnor

Date of issuance: December 10, 1024

If you have any questions, please contact Joel S. Wiebe at 301-415-6606.

Sincerely,

/RA/

Travis L. Tate, Chief
Plant Licensing III-2 and
Planning and Analysis Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-456

Enclosure:
Safety Evaluation

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