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

February 25, 2019

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, RELIEF FROM THE REQUIREMENTS OF THE ASME CODE (EPID L-2018-LLR-0118)

Dear Mr. Hanson:

By letter dated August 30, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18248A060), Exelon Generating Company, LLC (the licensee) submitted a request to the U.S. Nuclear Regulatory Commission (NRC) for the use of alternatives to certain American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, requirements at Byron Station, Unit No. 2.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(2), the licensee requested to use the proposed alternative on the basis that the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The NRC staff has reviewed the subject request and concludes, as set forth in the enclosed safety evaluation, that Exelon has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(2). Therefore, the NRC staff authorizes the use of relief request I4R-16 for the remainder of the fourth inservice inspection interval at Byron Station, Unit No. 2, currently scheduled to end on July 15, 2025.

All other requirements of ASME Code, Section XI, for which relief was not specifically requested and authorized by the NRC staff remain applicable, including the third party review by the Authorized Nuclear In-service Inspector.

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

Sincerely,

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

Docket No. STN 50-455

Enclosure:
Safety Evaluation

cc: Listserv

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SAFETY EVALUATION BY THE OFFICE NUCLEAR REGULATION

RELIEF REQUEST I4R-16 REGARDING

ALTERNATIVE FOLLOW-UP INSPECTIONS FOR

REACTOR PRESSURE VESSEL HEAD PENETRATION NOZZLES

EXELON GENERATION COMPANY, LLC

BYRON STATION, UNIT NO. 2

DOCKET NO. 50-455

1.0 INTRODUCTION

By letter dated August 30, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18248A060), Exelon Generating Company, LLC (the licensee) requested approval for alternative follow-up inspections of peening-applied reactor vessel head penetration nozzles for the fourth inservice inspection (ISI) interval of Byron Station, Unit 2.

The inspection requirements for the penetration nozzles are specified in paragraph (g)(6)(ii)(D) of 10 *Code of Federal Regulations* (10 CFR) 50.55a, "Codes and Standards." As documented in the NRC safety evaluation dated September 19, 2017 (ADAMS Accession No. ML17249A241), the application of water jet peening and associated inspection requirements were approved for these nozzles based on the guidance in Electric Power Research Institute report, Materials Reliability Program (MRP)-335, Revision 3-A, "Materials Reliability Program: Topical Report for Primary Water Stress Corrosion Cracking Mitigation by Surface Stress Improvement." With respect to the follow-up inspection, the mitigated nozzles are required to be inspected during the second (N+2) refueling outage (RFO) following the peening mitigation. Specifically, the 70 nozzles mitigated during RFO B2R19 are required to be inspected during RFO B2R21 (spring 2019) and the 9 nozzles mitigated during RFO B2R20 are required to be inspected during RFO B2R22 (fall 2020).

Pursuant to 10 CFR 50.55a(z)(2), the licensee proposed that the follow-up inspections of the 70 nozzles will be conducted during RFO B2R22 in alignment with the follow-up inspections of the 9 nozzles.

2.0 REGULATORY EVALUATION

Components (including supports) that are classified as ASME Code Class 1, Class 2, and Class 3 must meet the requirements in 10 CFR 50.55a(g)(4), "Inservice Inspection Standards Requirement for Operating Plants," throughout the service life of a boiling- or pressurized-water-reactor (BWR or PWR). The exception is the 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 10 CFR 50.55a, which are incorporated by reference in paragraph (a)(1)(ii) of 50.55a, to the extent practical within the limitations of design, geometry, and materials of construction of the components.

Pursuant to 10 CFR 50.55a(g)(6)(ii), the U.S. Nuclear Regulatory Commission (NRC) may require the licensee to follow an augmented ISI program for systems and components for which the NRC deems that added assurance of structural reliability is necessary.

Pursuant to 10 CFR 50.55a(g)(6)(ii)(D), "Reactor Vessel Head Inspections," the NRC requires licensees of PWRs to augment their ISI of the reactor vessel head with ASME Code Case N-729-4, "Alternative Examination Requirements for PWR Reactor Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds, Section XI, Division 1," with conditions.

10 CFR 50.55a(z)(2) states, in part, that alternatives to the requirements of 10 CFR 50.55a(g) may be used when authorized by the NRC if the licensee demonstrates compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Based on the above and subject to the following technical evaluation, the NRC staff finds that regulatory authority exists for the licensee to request the use of an alternative and the NRC to authorize the proposed alternative.

3.0 TECHNICAL EVALUATION

3.1 ASME Code Components Affected

The subject components are ASME Code Class 1, reactor vessel head penetration nozzles that have pressure-retaining partial-penetration J-groove welds. These nozzles are ASME Code Case N-729-4, Item B4.20, components fabricated with Alloy 600/82/182 materials. Water jet peening (also called cavitation peening) was applied on the nozzles for mitigation of potential primary stress corrosion cracking (PWSCC) in accordance with the guidance in MRP-335, Revision 3-A. As discussed above, the application of the peening process on the subject nozzles was approved in the NRC staff safety evaluation (SE) dated September 19, 2017. Since the reactor vessel head of Byron Station, Unit No. 2, operates at cold-leg temperature (547 – 561 °F (degree Fahrenheit)), the head is categorized as a T_{cold} head or cold head.

3.2 Applicable ASME Code Edition and Addenda

The current code of record for the fourth ISI interval of Byron Station, Unit No. 2, is the 2007 Edition through the 2008 Addenda of ASME Code, Section XI. Examinations of the subject nozzles are performed in accordance with 10 CFR 50.55a(g)(6)(ii)(D), which specifies the use of ASME Code Case N-729-4 with conditions.

3.3 Applicable Code Requirements

ASME Code Case N-729-4 addresses inspection requirements for reactor vessel head penetration nozzles, as conditioned by 10 CFR 50.55a(g)(6)(ii)(D). The regulation in 10 CFR 50.55a(g)(6)(ii)(D)(1) requires in part that holders of operating licenses or combined licenses for PWRs as of or after August 17, 2017, shall implement the requirements of ASME

Code Case N-729-4 instead of ASME Code Case N-729-1, subject to the conditions specified in paragraphs (g)(6)(ii)(D)(2) through (4) of 10 CFR 50.55a, by the first RFO starting after August 17, 2017.

As discussed above, the NRC SE dated September 19, 2017, approved the application of peening on the subject nozzles for mitigation of potential PWSCC, pursuant to 10 CFR 50.55a(z)(1). With respect to the follow-up inspections after the peening, the NRC staff's SE granted Byron Station, Unit No. 2, relief from conducting the first RFO inspection (N+1 inspection) after the peening application that is specified in MRP-335, Revision 3-A, Table 4-3, Note (11)(b). As a result, the follow-up inspection required for the subject nozzles is the second RFO inspection (N+2 inspection) following the peening application, as documented in the September 19, 2017, SE. In addition, the licensee is required to perform ISI on the subject nozzles every 10 years (i.e., every ISI interval).

3.4 Reason for Request

The relief request (RR) pertains to the follow-up inspections for the penetration nozzles including the associated welds. A summary of the licensee's reason for the request is included below.

During the water jet peening application at Byron Station, Unit No. 2, in the spring 2016 (RFO B2R19), 70 nozzles were successfully peened, but 9 nozzles did not receive complete peening in accordance with the performance criteria of MRP-335, Revision 3-A. The affected nozzles were 8 control rod drive mechanism (CRDM) nozzles (nozzle Nos. 67, 68, 69, 71, 73, 74, 75, and 77) and the vent line nozzle. Subsequently, the licensee peened these 9 nozzles during the fall 2017 RFO (B2R20), which completed the water jet peening on all 79 reactor vessel head nozzles. As discussed above, the NRC SE dated September 19, 2017, allowed the licensee not to conduct the first RFO follow-up inspection (N+1 inspection). Therefore, the licensee is currently required to perform the second RFO follow-up inspection (N+2 follow-up inspection) on the subject nozzles.

In this RR, the licensee proposed that for the 70 nozzles peened in spring 2016, the second RFO volumetric examination (N+2 inspection) be postponed by one cycle to the third RFO to align with the follow-up volumetric examination for the 9 nozzles peened in fall 2017. Approval of this request would allow for aligning the timing of the follow-up volumetric examination of all 79 nozzles to a single RFO (B2R22 of fall 2020), thereby, reducing personnel containment entries, risk of working in a locked high radiation area (LHRA), and total personnel collective radiation dose.

In its determination of the hardship and level of quality and safety, the licensee considered the following factors: (1) radiological dose and industrial safety concerns; (2) assessments based in part on deterministic analysis results in MRP-335, Revision 3-A, for N+3 follow-up inspection timing; and (3) operating experience of unmitigated CRDM nozzles.

3.5 Proposed Alternative and Its Basis

The licensee requested that as an alternative to the requirements of 10 CFR 50.55a(g)(6)(ii)(D), a single follow-up inspection is proposed to be conducted in the third (N+3) RFO for the 70 nozzles that were peened in the spring 2016 outage. This alternative allows that the follow-up inspections of all 79 nozzles (including the associated welds) are conducted during a single

RFO (B2R22 in the fall of 2020). Table 1 summarizes the proposed follow-up inspection schedule in this RR.

Table 1. Proposed Follow-up Inspection Schedule

Refueling Outage	Outage Number	Water Jet Peening and Follow-Up Inspection Activities
Spring 2016	B2R19	Peening of 70 nozzles was completed
Fall 2017	B2R20	Peening of the remaining 9 nozzles was completed
Spring 2019	B2R21	Proposed alternative: The follow-up inspection of the 70 nozzles is postponed to RFO B2R22 (Fall 2020).
Fall 2020	B2R22	Proposed alternative: Follow-up inspections of all the 79 nozzles

The basis of the proposed alternative is summarized below. Performance of the follow-up examinations in two separate outages results in hardships that are not compensated by a corresponding increase in safety or quality. Synchronization of the follow-up inspection timing for all 79 nozzles would have the benefits of reducing the number of personnel containment entries, risk of working in a LHRA, and total personnel collective radiation dose.

The increase in dose is estimated to be approximately 240 to 272 millirem (mrem) that includes approximately 80 mrem to set-up and demobilize equipment and approximately 160 to 192 mrem due to testing activities such as tool change-out and expected probe failure changes. This dose estimate is based on historical data but can be higher if tool breakdowns or issues occur requiring additional personnel entry, which is inconsistent with industry as-low-as-reasonably-achievable (ALARA) practices.

In addition, the proposed alternative to perform the follow-up inspection for the 70 nozzles in the fall 2020 outage (N+3 outage for the 70 nozzles) to align inspection for all 79 nozzles is also supported by the assessments and supplemental evaluations. The following references provide supporting technical evaluations for structural integrity of the subject nozzles:

- MRP-335, Revision 3-A, "Materials Reliability Program: Topical Report for Primary Water Stress Corrosion Cracking Mitigation by Surface Stress Improvement," November 2016 (ADAMS Accession No. ML16319A282)
- Technical Note TN-4069-00-01, Revision 0, "MRP-335 R3-A Matrix of Deterministic Crack Growth Calculations for T_{cold} Reactor Vessel Top Head Nozzles Evaluated for Alternative Peening Follow-up Volumetric Examination Timing," August 2018 (included as Attachment 2 to the licensee's letter dated August 30, 2019)
- Technical Note TN-4069-00-02, Revision 0, "Experience for Unmitigated CRDM Nozzles in U.S. PWRs Evaluated for Margin Against Leakage Considering Additional PWSCC Growth if Indications Had Remained in Service," August 2018 (included as Attachment 3 to the licensee's letter dated August 30, 2019)

The additional 18 months for an N+3 follow-up inspection has the advantage of allowing more time for potential shallow pre-existing flaws to grow and become more readily detectable at the time of the follow-up inspection. Considering that ultrasonic testing is not qualified to detect shallow flaws extending less than 10 percent through the nozzle wall, the N+3 follow-up inspection would be more effective in addressing slow-growing flaws prior to implementing the long-term 10-year ISI interval.

The experience for unmitigated heads in the U.S. operating at T_{cold} (including that for Byron Station, Unit No. 2, prior to the 2016 peening application) shows that even without taking credit for the peening surface stress improvement, through-wall cracking and leakage are unlikely to occur during the alternative N+3 cycle for the follow-up inspection within the 54-month (4.5-year) time period after peening.

For example, a 2016 pressure vessel and piping (PVP) conference paper evaluated in detail the PWSCC indications detected in 25 reactor vessel head nozzles in T_{cold} heads by that time, all in the area of the toe of the J-groove weld on the nozzle outer diameter (Reference: "Deterministic Technical Basis for Re-Examination Interval of Every Second Refueling Outage for PWR Reactor Vessel Heads Operating at T_{cold} with Previously Detected PWSCC," Proceedings of the ASME 2016 PVP Conference, PVP2016-64032, Copyright 2016 by ASME). Through an extension of the assessment of plant experience in the PVP paper, the evaluation per Technical Note TN-4069-00-02 demonstrates that there is a substantial margin against growth upward to the nozzle annulus and against consequential leakage for the N+3 inspection (i.e., 54 months after the peening for the units with nominal 18-month fuel cycles as is the case with Byron Station, Unit No. 2).

In accordance with MRP-335, Revision 3-A, and the corresponding NRC SE (ADAMS Accession No. ML16208A485), the licensee performs a bare metal visual examination of each nozzle for evidence of pressure boundary leakage every RFO. This requirement ensures in the unlikely event of through-wall cracking prior to the time of an alternative N+3 follow-up volumetric examination, that the through-wall cracking is identified in a timely fashion.

In accordance with 10 CFR 50.55a and ASME Code Case N-729-4 with conditions, the demonstrated leak path assessment examination is also required whenever a volumetric examination is performed (including the examinations immediately prior to peening as part of the pre-peening baseline inspection). These assessments also provide defense-in-depth to identify leakage through both the J-groove weld and nozzle base metal. Recent industry operating experience with a leaking CRDM penetration nozzle affected by cracking of the J-groove weld supports the effectiveness of the demonstrated leak path assessment examination as an early indication of leakage (Reference: Licensee Event Report 2018-001-00, "Penetration Indications Discovered During Reactor Pressure Vessel Head Inspection," Docket No. 50-247, dated May 21, 2018 (ADAMS Accession No. ML18149A126)).

3.6 Duration of the Proposed Alternative

The duration of the proposed alternative is for the fourth ISI interval which is scheduled to end on July 15, 2025.

3.7 NRC Staff Evaluation

The licensee provided the following basis for hardship associated with the follow-up inspections during the two RFOs without the implementation of proposed synchronization to a single RFO follow-up inspection (RFO B2R22 of fall 2020).

- Additional occupational radiation exposure is required due to entry inside containment. The increase in dose is estimated to be approximately 240 to 272 mrem based on historical data. This estimated dose can be higher if tool breakdowns or issues occur requiring additional personnel entry, which is inconsistent with industry ALARA practices.

- Combining two inspections to one inspection as proposed reduces risk of industrial accidents. The fewer number of containment entries and potential entries to LHRA decreases the potential for industrial safety risks.

- In addition, the potential for increased contamination exposure exists due to entries inside containment and entry to LHRA.

The NRC staff finds that the licensee adequately identified the basis for hardship that involves the additional occupational radiation doses, potential for increases in industrial accident risks and potential for increases in contamination exposure.

As discussed above, the NRC staff's SE dated September 19, 2017, approved that for the follow-up inspections after the peening, the first RFO follow-up inspection (N+1 inspection) may not be conducted on the subject 79 nozzles. Therefore, the NRC staff evaluation below focused on the proposed alternative addressed in this RR that the N+2 inspection for the 70 nozzles (including the J-groove welds) is shifted to N+3 timing (RFO B2R22) in order to align with the follow-up inspection for the 9 nozzles (peened during RFO B2R20).

Leakage from the penetration nozzle can involve circumferential cracking and resulting ejection of a penetration nozzle from the head. This could cause a small break loss of coolant accident. The leakage could cause boric acid corrosion of the low alloy steel material that compromises the thickness of the head. Boric acid corrosion rates of low alloy steel could be up to 6 inches/year under very severe conditions (NUREG/CR-6875, Boric Acid Corrosion of Light Water Reactor Pressure Vessel Materials, J.-H. Park, O. K. Chopra, K. Natesan, and W. J. Shack; July 2005). After sufficient corrosion occurs, a small or medium break loss-of-coolant accident could occur. To prevent such significant degradation in reactor vessel heads and penetration nozzles, 10 CFR 50.55a requires an inspection program for these components, including volumetric examinations and bare metal visual examinations. The NRC staff also recognizes that due to the cold-leg head temperature of the reactor vessel head at Byron Station, Unit No. 2, the crack growth rates for a circumferential flaw growth to cause nozzle ejection would be sufficiently longer than the leakage time during one operating cycle. The NRC staff further notes that the licensee applied peening on the subject nozzles in accordance with MRP-335, Revision 3-A, such that PWSCC initiation has been effectively mitigated in the components.

In addition, the licensee confirmed that a bare metal visual examination is performed on each nozzle for evidence of pressure boundary leakage every RFO in accordance with MRP-335, Revision 3-A. The NRC staff finds that the visual examination is an effective defense-in-depth inspection. While it cannot proactively identify future leakage through the reactor coolant pressure boundary, the frequency of examination, each outage, reasonably addresses the consequences of such leakage.

The NRC staff also notes that technical specification (TS) 3.4.13 of Byron Station, Unit No. 2, requires operational leakage monitoring, which includes containment sump monitoring and containment atmosphere radioactivity monitoring (TS 3.4.15). Given the licensee's peening mitigation and hardship, the NRC staff finds that a bare metal visual examination each outage, when coupled with operational leakage monitoring, provides reasonable assurance of structural integrity over the inspection extension period of the licensee's proposed alternative (i.e., one cycle). In addition, if any leakage is identified, the nozzle would be required to be repaired by the licensee, and would not impact the basis of this technical analysis.

As discussed above, the NRC staff finds that the bare metal visual examination, along with operational leakage monitoring, provides reasonable assurance that the integrity of the penetration nozzles and adjacent reactor vessel head areas is maintained without significant degradation and that any potential leakage will be detected in a timely manner prior to significant degradation in the reactor coolant pressure boundary components.

In addition, the licensee provided technical information regarding crack growth calculations and evaluations that are described in the MRP-335, Revision 3-A; TN-4069-00-01, Revision 0; and TN-4069-00-02, Revision 0. The crack growth analyses are based on conservative assumptions and industry-wide crack size measurement data for T_{cold} heads (operating at 547 – 561 °F). The licensee's analysis includes a matrix of deterministic PWSCC crack growth calculations. The matrix considers various crack growth cases that involve different hypothetical initial crack sizes, crack aspect ratios, operating temperatures of T_{cold} heads and severity levels of stress profiles. The crack growth analysis discusses the effectiveness of follow-up volumetric examination timings after peening (i.e., N+1, N+2, and N+3 timings) to prevent pressure boundary leakage of the nozzles. The analysis for inspection timing effectiveness further estimates the growth of hypothetical, shallow PWSCC cracks that may exist in the base metal of the nozzle at the time of peening and would be too shallow to be reliably detected during pre-peening baseline inspection. Such a shallow crack depth is less than approximately 10 percent of the nozzle wall thickness.

The licensee's evaluation indicates that both the N+2 and N+3 inspection schedules result in the virtually identical low fraction of crack growth cases that would cause nozzle leakage. These limited cases are based on very conservative assumptions (such as use of 95th percentile crack growth rate per MRP-55). The NRC staff finds that these hypothetical leakage cases could be detected in ISIs subsequent to the follow-up inspection after peening such as bare metal visual examinations performed every RFO.

As discussed above, the NRC staff finds the following basis for acceptance of the proposed alternative in its review: (a) hardship exists due to the additional occupational radiation doses, potential for increases in industrial accident risks, and potential for increases in contamination exposure if the two follow-up inspections (RFOs B2R21 and B2R22) are not aligned into a single RFO inspection (B2R22) and (b) the bare metal visual examination performed each RFO, along with operational leakage monitoring, provides reasonable assurance that any leakage of the subject nozzles and reactor vessel head will be detected and managed in a timely manner prior to significant degradation of the components.

4.0 CONCLUSIONS

The NRC staff determines that the proposed alternative provides reasonable assurance of the integrity of the subject components and that complying with the requirement would result in hardship or unusual difficulty without a compensating increase in the 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)(2). Therefore, the NRC staff authorizes the use of the proposed alternative in RR I4R-16 at Byron Station, Unit No. 2, for the fourth 10-year ISI interval that is scheduled to end on July 15, 2025.

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

Principal Contributor: S. Min

Date of Issuance: February 25, 2019

SUBJECT: BYRON STATION, UNIT NO 2, RELIEF FROM THE REQUIREMENTS OF THE ASME CODE (EPID L-2018-LLR-0118) DATED FEBRUARY 25, 2019

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