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

September 29, 2021

Mr. David P. Rhoades
Senior Vice President
Exelon Generation Company, LLC
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: R. E. GINNA NUCLEAR POWER PLANT – ISSUANCE OF RELIEF REQUEST I6R-06 ALTERNATE INSPECTION FOR REACTOR VESSEL INTERNALS (EPID L-2020-LLR-0146)

Dear Mr. Rhoades:

By letter dated November 11, 2020 (Agencywide Documents and Access Management System (ADAMS) Accession No. ML20316A026) with supplements dated May 12, 2021 and August 12, 2021 (ADAMS Accession No. ML21132A079 and ML21224A284, respectively), Exelon Generation Company (the licensee) requested relief from the inspection requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI at the R.E. Ginna Nuclear Power Plant (Ginna).

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* Part 50 (10 CFR 50), 50.55a(z)(1), the licensee submitted Relief Request (RR) I6R-06 on the basis that the proposed alternative would provide an acceptable level of quality and safety. The licensee requested to use the requirements of paragraph IWB-2420(c) of the ASME Code, Section XI, 2017 Edition in lieu of paragraph IWB-2420(b) of the 2004 Edition and 2013 Edition for the successive inspection of the 270-degree lower radial support clevis insert.

The NRC staff determines 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 Relief Request I6R-06 for the fifth and sixth 10-year Inservice Inspection (ISI) intervals.

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

Enclosure

If you have any questions, please contact the Ginna Project Manager, V. Sreenivas, at 301-415-2597 or V.Sreenivas@nrc.gov.

Sincerely,

James G. Danna

Digitally signed by James G. Danna
Date: 2021.09.29 09:35:53 -04'00'

James G. Danna, Chief
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-244

Enclosure:
Safety Evaluation

cc: Listserv

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST I6R-06

ALTERNATE INSPECTION FOR REACTOR VESSEL INTERNALS

R. E. GINNA NUCLEAR POWER PLANT

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-244

1.0 INTRODUCTION

By letter dated November 11, 2020 (Agencywide Documents and Access Management System (ADAMS) Accession No. ML20316A026) with supplements dated May 12, 2021, and August 12, 2021 (ADAMS Accession No. ML21132A079 and ML21224A284, respectively), Exelon Generation Company (the licensee) requested relief from the inspection requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI at the R. E. Ginna Nuclear Power Plant.

Pursuant to Title 10 of the *Code of Federal Regulations* Part 50 (10 CFR 50), 50.55a(z)(1), the licensee submitted Relief Request (RR) I6R-06 on the basis that the proposed alternative would provide an acceptable level of quality and safety. Specifically, the licensee requested to use the requirements of paragraph IWB-2420(c) of the ASME Code, Section XI, 2017 Edition in lieu of paragraph IWB-2420(b) of the 2004 Edition and 2013 Edition for the successive inspection of the 270-degree lower radial support clevis insert.

The licensee's two analytical evaluations demonstrate that the degraded clevis insert does not impact the qualification of the reactor vessel equipment. Therefore, the NRC staff finds that it is acceptable for the licensee to use paragraph IWB-2420(c) of the ASME Code, Section XI, 2017, Edition in lieu of paragraph IWB-2420(b) of the 2004 Edition and 2013 Edition for the fifth and sixth 10-year inservice inspection (ISI) intervals.

2.0 REGULATORY EVALUATION

Pursuant to 10 CFR 50.55a(g)(4), "Inservice inspection standards requirement for operating plants," ASME Code Class 1, 2, and 3 components (including supports) must meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in the ASME Code, Section XI 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 first 10-year interval and subsequent intervals 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(a)(1)(ii) 18 months prior to the start of the 120-month interval, subject to the conditions listed in 10 CFR 50.55a(b)(2), "Conditions on ASME BPV Code, Section XI."

Paragraph 10 CFR 50.55a(g)(4)(iv) states that "Inservice examination of components and system pressure tests may meet the requirements set forth in subsequent editions and addenda that are incorporated by reference in paragraph (a) of this section [i.e., 10 CFR 50.55a(a)], subject to the conditions listed in paragraph (b) of this section [i.e., 10 CFR 50.55a(b)], and subject to Commission approval. Portions of editions or addenda may be used, provided that all related requirements of the respective editions or addenda are met."

Paragraph 10 CFR 50.55a(z) states that "Alternatives to the requirements of paragraphs (b) through (h) of this section [10 CFR 50.55a] or portions thereof may be used when authorized by the Director, Office of Nuclear Reactor Regulation. A proposed alternative must be submitted and authorized prior to implementation. The applicant or licensee must demonstrate that: 1) *Acceptable level of quality and safety*. The proposed alternative would provide an acceptable level of quality and safety; or (2) *Hardship without a compensating increase in quality and safety*. Compliance with the specified requirements of this section 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 Nuclear Regulatory Commission (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 Component(s) Affected

The affected component is the 270-degree lower radial support clevis insert which is part of the core support structure. This ASME Code Class 1 component is classified as Examination Category B-N-3 with Item Number B13.70 as specified in Table IWB-2500-1 of the ASME Code, Section XI.

3.2 Applicable Code Edition and Addenda

The 2004 Edition, No Addenda of the ASME Code, Section XI is the code of record for the fifth ISI interval which started on January 1, 2010, and ended on December 31, 2019.

The 2013 Edition of the ASME Code, Section XI, is the code of record for the sixth ISI interval which started on January 1, 2020, and will end on December 31, 2029.

The licensee stated that the fifth ISI interval was extended by approximately four months as allowed by Paragraph IWA-2430(d)(2) of the 2004 Edition, No Addenda of the ASME Code, Section XI to perform reactor vessel internal inspections for Examination Categories B-N-1, B-N-2, and B-N-3, which are specified in Table IWB-2500-1 of the ASME Code, Section XI during the Spring 2020 refueling outage. The licensee further stated that this interval extension only applied to Examination Categories B-N-1, B-N-2, and B-N-3.

3.3 Applicable Code Requirements

The 2004 Edition or the 2013 Edition of the ASME Code, Section XI, IWB-2420(b), "Successive Inspections," states that "If a component is accepted for continued service in accordance with IWB-3132.3 or IWB-3142.4, the areas containing flaws or relevant conditions shall be reexamined during the next three inspection periods listed in the schedule of the inspection program of IWB-2400."

The 2004 Edition of the ASME Code, Section XI, IWB-2420(c), "Successive Inspections," states that "If the reexaminations required by IWB-2420(b) reveal that the flaws or relevant conditions remain essentially unchanged for three successive inspection periods, the component examination schedule may revert to the original schedule of successive inspections."

The 2013 Edition of the ASME Code, Section XI, IWB-2420(c), "Successive Inspections," states that "If the reexaminations required by IWB-2420](b) reveal that the flaws or relevant conditions remain essentially unchanged, or that the flaw growth is within the growth predicted by the analytical evaluation, for three successive inspection periods, then the component examination schedule may revert to the original schedule of successive inspections or the inspection interval defined by the analytical evaluation, whichever is limiting."

The 2004 Edition of the ASME Code, Section XI, IWB-3142.4, "Acceptance by Analytical Evaluation," states that "A component containing relevant conditions is acceptable for continued service if an analytical evaluation demonstrates the component's acceptability. The evaluation analysis and evaluation acceptance criteria shall be specified by the Owner. A component accepted for continued service based on analytical evaluation shall be subsequently examined in accordance with IWB-2420(b) and (c)."

The 2013 Edition of the ASME Code, Section XI, IWB-3142.4, "Acceptance by Analytical Evaluation," states that "A component containing relevant conditions is acceptable for continued service if an analytical evaluation demonstrates the component's acceptability. The evaluation analysis and evaluation acceptance criteria shall be specified by the Owner. A component accepted for continued service based on analytical evaluation shall be subsequently examined in accordance with IWB- 2420(b) and IWB-2420(c)."

The 2017 Edition of the ASME Code, Section XI, IWB-2420(c), "Successive Inspections," states that "If a component is accepted for continued service in accordance with IWB-3142.4, successive examinations shall be performed, if determined necessary, based on an evaluation by the Owner. The evaluation shall be documented and shall include the cause of the relevant condition, if known. If the cause of the relevant condition is unknown or if the relevant condition has previously occurred, successive examinations shall be performed during each successive inspection period until the relevant condition remains essentially unchanged from the previous inspection."

3.4 Reason for Request

The licensee requested the use of the provision of IWB-2420(c) from the 2017 Edition of the ASME Code, Section XI, which permits successive examinations to be performed, if determined necessary, based on an evaluation by the owner. The licensee proposed to use this provision in lieu of IWB-2420(b) from the 2004 Edition and 2013 Edition of the ASME Code, Section XI, which requires the area containing relevant conditions to be reexamined

during the next three inspection periods, regardless of the evaluation by the owner. The licensee stated that all related requirements of the 2004 Edition or the 2013 Edition, as applicable, will be maintained.

The licensee proposed to use the provisions of IWB-2420(c) of the 2017 Edition of the ASME Code, Section XI, to define the successive inspection requirements for the 270-degree lower radial support clevis insert that was accepted by analytical evaluation performed in accordance with IWB-3142.4 of the 2004 Edition of the ASME Code, Section XI, during the Spring 2020 refueling outage. The licensee stated that use of IWB-2420(c) of the 2017 Edition may prevent the need to perform successive inspections of the 270-degree lower radial support clevis insert which would require a full core offload during the Fall 2021 outage.

3.5 Proposed Alternative

In lieu of successive examinations per IWB-2420(b) of the ASME Code, Section XI, 2004 Edition or 2013 Edition, the licensee proposed to use IWB-2420(b) of the ASME Code, Section XI, 2017 edition to address the successive inspection requirements of the 270-degree lower radial support clevis insert of the core support structure.

3.6 Basis for Use

During the G1R42 refueling outage in the Spring 2020, the licensee found that the 270-degree lower radial support clevis insert was disengaged from the clevis and radially displaced. The licensee stated that the probable cause of the clevis insert displacement was primary water stress corrosion cracking (PWSCC) of the clevis bolts. The licensee further stated that the displacement resulted in a loss of insert retention. The licensee explained that a combination of flow-induced vibrations and differential thermal expansion during plant cooldowns reduced the interference fit allowing the observed radial displacement to occur. The licensee accepted the as-found clevis insert for the next operating cycle based on an analytical evaluation performed in accordance with IWB-3142.4 of the 2004 Edition of the ASME Code, Section XI. The licensee determined that the as-left configuration of the four clevis inserts can maintain the core support function to meet design requirements. By letter dated February 19, 2021 (ADAMS Accession No. ML21055A020), the licensee submitted the evaluation of the degraded clevis insert for one fuel cycle as shown in the 2020 Owner's Activity Report (G1R42 OAR-1) in accordance with IWB-3144(b) of the 2004 Edition.

The licensee stated that IWB-2420(b) of the 2004 Edition of the ASME Code, Section XI, requires that if a component is accepted for continued service in accordance with IWB-3142.4, the areas containing relevant conditions shall be reexamined during the next three ISI periods. The licensee further stated that reexamination of the 270-degree lower radial support clevis insert would require a full core offload and removal of the core barrel resulting in an extended outage and additional dose each of the next three ISI periods. The licensee explained that implementing IWB-2420(c) of the 2017 Edition of the ASME Code, Section XI, may prevent the need to perform successive inspections provided acceptable evaluation results as required by the ASME Code are obtained to justify continued operation.

For the fifth ISI interval, the licensee proposed the use of IWB-2420(c) of the 2017 Edition of the ASME Code, Section XI in lieu of IWB-2420(b) of the 2004 Edition of ASME Section XI to define successive inspections of the degraded clevis insert that was accepted by analytical evaluation during the Spring 2020 outage for one operating cycle as shown in G1R42 OAR-1.

For the sixth ISI interval, the licensee proposed to use IWB-2420(c) of the 2017 Edition of the ASME Code, Section XI in lieu of IWB-2420(b) of the 2013 Edition of the ASME Code Section XI to define the successive inspections of the degraded clevis insert that was accepted by analytical evaluation as documented in letters dated May 12, and August 12, 2021.

Analytical Evaluation of Degraded Clevis Insert

The licensee has performed an analytical evaluation to justify continued operation with the degraded clevis insert for one cycle as documented in G1R42 OAR-1. In addition, the licensee has performed an analytical evaluation to justify continued operation with the degraded clevis insert for the sixth 10-year ISI interval as documented in its letters dated May 12, and August 12, 2021. Below is a discussion of the analytical evaluation for the 10-year ISI interval which covers the one-cycle evaluation.

The licensee analyzed three major topics -- (1) wear degradation, (2) loading on the related components using the reactor equipment system model (RESM), and (3) the revised loadings as compared with the loadings in the existing design analysis. The licensee stated that a new design analysis will not be required if the new applied loads are bounded by the original analysis.

The licensee determined the maximum wear on the degraded clevis insert that will occur over the next 10 years of operation. The licensee considered two cases to evaluate wear at the degraded clevis insert: (1) best estimate wear using the relative motion and contact conditions, and (2) limiting wear based on the limiting assembly gaps and assuming this to be wear volume. The licensee used the best estimate wear approach for the 10-year ISI interval. The licensee used the self-limiting wear case to justify for the one-cycle continued operation.

For the wear evaluation, the licensee used the Archard equation [a mathematical model to describe sliding wear based on the asperity contact] and vibrational motion of the core barrel to determine expected wear rate over the period of ten years. The licensee stated that the applied loading between the radial key and the clevis insert decays with operating time. Additionally, due to the loss of material from the wear condition, additional load decay will occur. To model the maximized wear, the licensee assumed the following: (1) worst case initial interference was assumed between the clevis insert and the radial key; (2) the clevis insert was assumed to follow with the core barrel motion, no slip between the clevis insert and radial key; and (3) one removal and reinstallation of the core barrel during the 10-year cycle was assumed. Also, the licensee considered core barrel movements based on core barrel vibrations, flow induced vibrations of the clevis insert, and thermal heat-up/cool-down cycles.

The licensee evaluated the vertical wear (loss of clevis insert flange thickness) assuming sliding between the clevis insert flange and the support block. The licensee compared flange material loss to the flange thickness to determine if the flange will be worn through and lose vertical retention. The licensee stated that results show that the material loss at the flange is less than the flange thickness and is acceptable for a 10-year operating period.

After the wear assessment was completed, the licensee input the wear results into the RESM to determine the loading outputs to use in the downstream analyses of the reactor vessel, internals, and interconnected equipment. After the revised loadings were calculated via the RESM, the licensee compared the revised loading for flow induced vibration, operating basis earthquake, safe shutdown earthquake, and loss of coolant accidents to the original design

basis analyses. The licensee considered the following components in its evaluation--reactor internals, nuclear fuel, reactor pressure vessel, reactor coolant loop piping, reactor pressure vessel supports, and reactor pressure vessel closure head equipment.

The licensee explained that the maximum wear used in the RESM analysis is the wear determined by the best estimate approach versus the more limiting, overly conservative, self-limiting wear. The licensee stated that the revised RESM is equivalent to the original and other industry RESMs except that provisions have been added to vary the gaps at the clevis inserts. The licensee developed the following loadings and applied them to the RESM: vertical steady state loads, seismic accelerations (safe shutdown earthquake and operating basis earthquake) from the containment building at the reactor vessel nozzle supports, LOCA input loading, and flow-induced vibration forcing functions. Lastly, the licensee compared the revised loading with the loading used in the design analysis and found that the original design basis analysis is still valid.

The licensee concluded that its analytical evaluation meets the requirements of ASME Code, Section XI IWB-3142.4 to demonstrate that the observed degradation and expected ongoing degradation does not impact the qualification of the reactor vessel equipment for the sixth 10-year ISI interval.

3.6 Duration of Proposed Alternative

The licensee stated that the proposed alternative will remain in effect for the remainder of the sixth Inservice Inspection Interval.

3.7 NRC Staff Evaluation

Paragraph 10 CFR 50.55a(g)(4)(iv) allows the use of subsequent editions and addenda, and portions thereof, incorporated by reference in 10 CFR 50.55a(a), subject to the limitations and modifications listed in 10 CFR 50.55a(b), and subject to Commission approval. The NRC has approved the use of the 2017 Edition of the ASME Code, Section XI in 10 CFR 50.55a. Paragraph 10 CFR 50.55a does not impose conditions on paragraph IWB-2420(c) of the 2017 Edition of the ASME Code, Section XI. Therefore, in terms of regulation, the NRC staff finds it acceptable for the licensee to use IWB-2420(c) of the ASME Code, Section XI, 2017 Edition in lieu of IWB-2420(b) of the 2004 Edition or the 2013 Edition.

To evaluate the adequacy of the proposed alternative, the NRC staff divided IWB-2420(c) of the ASME Code, Section XI, 2017 Edition into three sub-provisions: Provision (1) If a component is accepted for continued service in accordance with IWB-3142.4, successive examinations shall be performed, if determined necessary, based on an evaluation by the owner.
Provision (2) The evaluation shall be documented and shall include the cause of the relevant condition, if known. Provision (3) If the cause of the relevant condition is unknown or if the relevant condition has previously occurred, successive examinations shall be performed during each successive inspection period until the relevant condition remains essentially unchanged from the previous inspection.

With respect to the above Provision (1), the NRC determines that the licensee evaluated whether successive examinations are necessary in accordance with IWB-3142.4 of the 2004 Edition of the ASME Code, Section XI for the fifth and sixth 10-year ISI interval. The NRC staff determined that the licensee has appropriately evaluated the wear of the clevis insert and its impact on various core support structures based on the revised loading. The NRC staff further

determined that the licensee has considered necessary loads such as earthquake loads and flow-induced vibration loads. The licensee's results showed that its analytical qualification meets the requirements of ASME Code, Section XI, IWB-3142.4 to demonstrate the observed degradation and expected ongoing degradation does not impact the qualification of the reactor vessel equipment for the last cycle of the fifth 10-year ISI interval and for the sixth 10-year ISI interval. Therefore, the NRC staff finds that the licensee's evaluation satisfies provisions of IWB-3142.4 of the 2004 Edition of the ASME Code, Section XI and that the licensee has satisfied above Provision (1) of IWB-2420(c) of the ASME Code, Section XI, 2017 Edition.

With respect to the above Provision (2), the NRC staff notes that the licensee's evaluations are documented in the 2020 Owner's Activity Report (G1R42 OAR-1) and in its letters dated May 12, 2021, and August 12, 2021. In addition, the NRC staff finds that the licensee's two evaluations identified primary water stress corrosion cracking and wear as the cause of the degradation. Therefore, the NRC staff finds that the licensee has satisfied Provision (2) in Paragraph IWB-2420(c) of the ASME Code, Section XI, 2017 Edition.

The NRC staff determines that the above Provision (3) does not apply to Ginna because the licensee has identified the cause of the clevis insert degradation.

Based on the above discussion, the NRC staff determines that the licensee has satisfied Paragraph IWB-2420(c) of the ASME Code, Section XI, 2017 Edition.

The NRC staff recognizes the disadvantages of performing three successive examinations of the subject clevis insert. If examinations are performed, the licensee would need to perform a full core offload and remove the core barrel resulting in additional dose each of the next three ISI periods without sufficient increase in safety. Removing the core barrel may cause additional wear on the degraded clevis insert which could be detrimental to the integrity of the subject clevis insert. Therefore, the NRC staff determines that there are risk and negative consequence associated with three successive examinations of the subject clevis insert.

In summary, the NRC staff finds that it is acceptable for the licensee to use Paragraph IWB-2420(c) of the ASME Code, Section XI, 2017 Edition in lieu of per IWB-2420(b) of the 2004 Edition and 2013 Edition for the fifth and sixth 10-year ISI intervals because the licensee's two analytical evaluations demonstrate that the degraded clevis insert does not impact the qualification of the reactor vessel equipment for the fifth and sixth 10-year ISI intervals.

4.0 CONCLUSION

As set forth above, the NRC staff determines that the licensee's 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 Relief Request I6R-06 at Ginna Nuclear Power Plant for the fifth and sixth 10-year ISI intervals.

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

Principal Contributor: J. Tsao, NRR
D. Widrevitz, NRR

Date: September 29, 2021

SUBJECT: R. E. GINNA NUCLEAR POWER PLANT – ISSUANCE OF RELIEF REQUEST I6R-06 ALTERNATE INSPECTION FOR REACTOR VESSEL INTERNALS (EPID L-2020-LLR-0146) DATED SEPTEMBER 29, 2021

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ADAMS Accession No.: ML21250A382 *by e-mail

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