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

December 7, 2015

Mr. Bryan C. Hanson
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2 – RELIEF REQUEST FOR EXTENSION OF VOLUMTERIC EXAMINATION INTERVAL FOR REACTOR VESSEL HEADS WITH ALLOY 690 NOZZLES (CAC NOS. MF5829 AND MF5830)

Dear Mr. Hanson:

By letter dated January 8, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15009A035), as supplemented by letter dated July 13, 2015 (ADAMS Accession No. ML15201A067), Exelon Generation, LLC (the licensee), submitted relief request (RR ISI-024) to the U.S. Nuclear Regulatory Commission (NRC) for relief from the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section XI, associated with the examination frequency requirements of Code Case N-729-1 at the Calvert Cliffs Nuclear Power Plant, Units 1 and 2 (Calvert Cliffs).

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(z)(1), the licensee requested to use proposed alternatives, to the examination frequency of ASME Code Case N-729-1, on the basis that the alternative examination provides an acceptable level of quality and safety.

As set forth above, the NRC staff has determined that the alternative method proposed by the licensee in RR ISI-024 will provide an acceptable level of quality and safety for the examination frequency requirements of the reactor vessel closure head. 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 one time use of RR ISI-024 at Calvert Cliffs Nuclear Power Plant, Units 1 and 2, for the duration up to and including the Spring 2022 refueling outage (Unit 1) and the Spring 2023 refueling outage (Unit 2) which will occur in the fifth ten-year inservice inspection interval.

All other requirements of the ASME Code, Section XI, and 10 CFR 50.55a(g)(6)(ii)(D) 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.

If you have any questions, please contact Alex Chereskin by phone at 301-415-2549, or by e-mail at Alexander.Chereskin@nrc.gov.

Sincerely,

Travis L. Tate

Travis Tate, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-317 and 50-318

Enclosure:
Safety Evaluation

cc w/enclosure: Distribution via Listserv

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST ISI-024 – INSPECTION OF REACTOR VESSEL

CLOSURE HEAD NOZZLES IN ACCORDANCE WITH ASME

CODE CASE N-729-1 AS CONDITIONED BY 10 CFR 50.55a

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2

CALVERT CLIFFS NUCLEAR POWER PLANT, LLC

EXELON GENERATION COMPANY, LLC

DOCKET NOS. 50-317 AND 50-318

1.0 INTRODUCTION

By letter dated January 8, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15009A035), as supplemented by letter dated July 13, 2015 (ADAMS Accession No. ML15201A067), Exelon Generation, LLC (the licensee), requested relief from the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section XI, associated with the examination frequency requirements of Code Case N-729-1 at the Calvert Cliffs Nuclear Power Plant, Units 1 and 2 (Calvert Cliffs).

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(z)(1), the licensee requested to use proposed alternatives, to the examination frequency of ASME Code Case N-729-1, on the basis that the alternative examination provides an acceptable level of quality and safety.

2.0 REGULATORY EVALUATION

Section 50.55a(g)(6)(ii) of 10 CFR states, in part, that "The Commission may require the licensee to follow an augmented inservice inspection [ISI] program for systems and components for which the Commission deems that added assurance of structural reliability is necessary."

Section 50.55a(g)(6)(ii)(D), of 10 CFR states, in part, that "All licensees of pressurized water reactors shall augment their inservice inspection program with ASME Code Case N-729-1, [Alternative Examination Requirements for PWR [Pressurized Water Reactor] Reactor Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds, Section XI, Division 1,] subject to conditions specified in paragraphs (g)(6)(ii)(D)(2) through (6) of this section."

Enclosure

In this request, the licensee, has requested relief from the examination frequency required by Code Case N-729-1 and has, therefore, also requested relief from 10 CFR 50.55a(g)(6)(ii)(D).

Section 50.55a(a)(z) of 10 CFR states that alternatives to the requirements of paragraph (g) of 10 CFR 50.55a may be used, when authorized by the U.S. Nuclear Regulatory Commission (NRC), if the licensee demonstrates (1) "...the proposed alternatives would provide an acceptable level of quality and safety..." or (2) "...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 and the Commission to authorize the proposed alternative requested by the licensee.

3.0 TECHNICAL EVALUATION

3.1 Components Affected

The affected components are ASME Class 1, Reactor Vessel Closure Head (RVCH) Penetration Nozzles and partial penetration welds, which are fabricated from Inconel SB-167 (Alloy 690) UNS N06690. The nozzle J-groove welds are fabricated from ERNiCrFe-7 (UNS N06052) and ENiCrFe-7 (UNS W86152), 52/152 weld materials. The original RVCH which contained penetration nozzles which were manufactured with Alloys 600/82/182 materials, were replaced in Units 1 and 2 with new RVCH using Alloys 690/52/152 material for the penetration nozzles during the refueling outages (RFOs) that returned Units 1 and 2 to operation in Spring 2006 and Spring 2007, respectively.

3.2 Inservice Inspection Interval

The proposed duration for Unit 1 is for the fourth and part of the fifth ten-year ISI interval up to and including the Spring RFO that is scheduled to commence in 2022. The proposed duration for Unit 2 is for the fourth and part of the fifth ten-year ISI interval up to and including the Spring RFO that is scheduled to commence in 2023.

3.3 ASME Code of Record

The ASME Section XI Code of Record for the fourth 10-year ISI interval is the 2004 Edition with no Addenda.

3.4 ASME Code and/or Regulatory Requirements

Section 50.55a(g)(6)(ii)(D) of 10 CFR requires, in part, that licensees shall augment their ISI program with ASME Code Case N-729-1, subject to the conditions specified in paragraphs (2) through (6) of 10 CFR 50.55a(g)(6)(ii)(D). ASME Code Case N-729-1, Table 1, Inspection Item B4.40 requires volumetric/surface examinations be performed within one inspection interval (nominally 10 calendar years) of its inservice date for a replaced RVCH. The required volumetric/surface examinations would thus have to be completed by spring 2016 for Unit 1 and spring 2017 for Unit 2 in order to fulfill the requirements of ASME Code Case N-729-1.

3.5 Proposed Alternative

The licensee proposes to delay the next required inspection for a period not to exceed 6 years for Units 1 and 2. The licensee proposes to accomplish the inspection in accordance with ASME Code Case N-729-1 and 10 CFR 50.55a(g)(6)(ii)(D) during the 2022 and RFOs for Units 1 and 2, respectively.

3.6 Licensee's Basis for Use of the Proposed Alternative

The licensee's basis for use of the proposed alternative is based primarily on three topics of consideration. The first topic addresses the concept that the inspection interval in Code Case N-729-1 is based on primary water stress corrosion cracking (PWSCC) crack growth rates for Alloy 600/82/182. The second topic addresses a bare metal visual examination conducted on the licensee's replacement RVCH in 2012. The third topic addresses a plant specific factor of improvement (FOI) analysis conducted by the licensee.

In addressing its first basis for use of the proposed alternative, the licensee asserts that the inspection intervals contained in ASME Code Case N-729-1 for alloy 600/82/182 are based on Re-inspection years (RIY) equal to 2.25 and that this value is based on PWSCC crack growth rates as defined in the 75th percentile curve contained in MRP 55, "Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Thick-Wall Alloy 600 Material", and MRP 115, "Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Alloy 82, 182, and 132 Welds." The licensee further asserts that the PWSCC crack growth rates of alloy 690/52/152 are significantly lower than those of alloy 600/82/182 and, therefore, merit a longer inspection interval. The licensee bases that assertion on: (a) the lack of cracking in other 690 components such as steam generators and pressurizers in the approximately 20 years that alloy 690 has been in service in these components; (b) the failure to observe cracking in inspections already performed in replacement heads (9 of 40 replacement heads have been examined which includes heads which operate at higher temperatures than the head under consideration); (c) the similarity of the inspected heads to the head under consideration regarding configuration, manufactures, design and operating conditions; and (d) laboratory test data for alloys 690/52/152 as contained in MRP-375, "Technical Basis for Reexamination Interval Extension for Alloy 690 PWR Reactor Vessel Top Head Penetration Nozzles."

In addressing its second basis for use of the proposed alternative, the licensee stated that a bare metal visual examination was performed in 2014 for Unit 1 and in 2011 for Unit 2 on the replacement RVCHs in accordance with ASME Code Case N-729-1, Table 1, item B4.30. This visual examination was performed by VT-2 qualified examiners on the outer surface of the RVCHs including the annulus area of the penetration nozzles. This examination did not reveal any indications of nozzle leakage (e.g. boric acid deposits) on the surface or near a nozzle penetration. Also, the licensee did not propose any alternative examination processes to those required by ASME Code Case N-729-1, as conditioned by 10 CFR 50.55a(g)(6)(ii)(D). The Visual (VT-2) examinations and acceptance criteria as required by item B4.30 of Table 1 of ASME Code Case N-729-1 are not affected by this request and will continue to be performed on a frequency not to exceed every 5 calendar years.

In addressing its third basis for use of the proposed alternative, the licensee made a plant specific calculation of the required FOI in the crack growth rate of Alloy 690/52/152 as compared to the crack growth rate of alloy 600/82/182. In making this calculation the licensee used the actual temperature of the head and conservatively assumed that calendar years were equal to effective full power years. Based on this calculation, the licensee determined that an improvement factor of 7.6 (for both Units) was required to meet the proposed and desired inspection interval of 16 calendar years for Units 1 and 2. The licensee then proposed that because the required FOI (7.6) was smaller than the FOI of 20 which bounded most of the MRP 375 data for alloy 690/52/152, the use of a FOI of 7.6 would not result in a reduction in safety and was, therefore, justified.

The licensee stated that their analysis showed significant margin to ensure that Alloy 690 nozzle base and Alloy 52/152 weld materials used in the replacement RVCHs provide for a reactor coolant system pressure boundary where the potential for PWSCC has been shown by analysis and by years of positive industry experience to be remote. As such, the licensee proposed that the technical basis sufficient to ensure public health and safety by extending the inspection frequency of the RVCH nozzle from a maximum of 10 years to a new maximum of 16 years for Units 1 and 2.

3.7 NRC Staff Evaluation

In evaluating the technical sufficiency of the licensee's proposed alternative, i.e., a one-time extension of the volumetric/surface examination interval contained in ASME Code Case N-729-1 from 10 years to not longer than 16 years, the NRC staff considered each of the three aspects of the licensee's basis for use of the proposed alternative. The NRC staff found that the technical basis included by the licensee provided sufficient information for the NRC staff to review the proposed alternative.

Due to concerns about PWSCC, many pressurized water reactor (PWR) plants in the United States and overseas have replaced RVCHs containing Alloy 600/182/82 nozzles with heads containing Alloy 690/152/52 nozzles. The inspection frequencies developed in Code Case N-729-1 for RVCH penetration nozzles using Alloy 600/182/82 were developed based, in part, on those material's crack growth rate equations documented in MRP-55 and MRP-115. The licensee's primary technical basis for the proposed alternative is to present crack growth rate data for the new more crack resistant materials, Alloy 690/152/52, and demonstrate a FOI of these materials versus the older Alloy 600/82/182 materials. This improvement factor would then provide the basis for the extension of the inservice inspection frequency requested by the licensee in their proposed alternative.

The NRC staff did not validate all of the data used by MRP-375; however, the NRC staff review relies upon Alloy 690/152/52 crack growth rate data from two NRC contractors: Pacific Northwest National Laboratory (PNNL) and Argonne National Laboratory (ANL). This data is documented in a data summary report and can be found under ADAMS Accession Number ML14322A587. The NRC confirmatory research generally supports the contention that the crack growth rate of alloy 690/52/152 is more crack resistant but differs from the MRP-375 data in some respects.

The PNNL and ANL data summary report includes crack growth rate data up to approximately 20% cold work based on the observation of local strains in welds and weld dilution zone data.

However, the NRC staff did not consider the weld dilution zone data in its assessment. This is because the limited weld dilution zone data that is currently available has shown higher crack growth rates than are commonly observed for alloy 690/152/52 material. The high crack growth rates in weld dilution zones may be due to the reduced chromium present in these areas. The NRC staff chose to exclude the weld dilution zone data from this analysis due to the limited number of data points available, the variability in results, and due to the limited area of continuous weld dilution for flaws to grow through. For example, in the case of the highest measured crack growth rates, a flaw would have to travel in the heat affected zone of a j-groove weld along the low alloy steel head interface. It is not fully apparent to the NRC staff how accelerated crack growth in very small areas of weld dilution zone would result in a significantly increased probability of leakage or component failure during a relatively short extension of the required inspection interval. Exclusion of these data may be reevaluated as additional data become available; a better understanding of the existing data is obtained; or if a longer extension of the inspection interval is requested. Therefore, the NRC staff finds that the impact of these weld dilution zone crack growth rates on the change in volumetric inspection frequency, as requested by the licensee's proposed alternative, is not considered to be relevant for this specific relief request.

In evaluating the licensee's second basis for use of the proposed alternative, the NRC staff finds that the past bare metal visual examination on the head under consideration is a reasonable means to demonstrate the absence of leakage through the nozzle/J-groove weld prior to the time the examination was conducted. The NRC staff also finds that performance of future bare metal visual examinations in accordance with the code case is adequate to demonstrate the absence of leakage at or prior to the time the examinations are conducted. Finally, the NRC staff finds that the proposed alternative's frequency for bare metal visual examinations in conjunction with the new frequency of volumetric examinations is sufficient to provide reasonable assurance of the structural integrity of the RVCHs.

In evaluating the licensee's third basis for use of the proposed alternative, the NRC staff found that the licensee's calculated improvement factor of 7.6, to support an extension of the ASME Code Case N-729-1 inspection frequency of 2.25 RIY to 16 calendar years, was acceptable by NRC staff calculation. The NRC staff also found that the application of an FOI of 7.6 to the 75th percentile curves in MRP 55 and 115 bounded essentially all of the NRC data included in the PNNL and ANL data summary report. Therefore, the NRC staff found that this analysis supports the concept that a volumetric inspection interval for the RVCH of not more than 16 calendar years does not pose a higher risk than that associated with an alloy 600/182/82 RVCH inspected at intervals of 2.25 RIY. Hence, the NRC staff found the licensee's technical basis to be acceptable.

Therefore, the NRC finds that the proposed alternative provides an acceptable level of quality and safety as required by 10 CFR 50.55a(z)(1).

4.0 CONCLUSION

As set forth above, the NRC staff has determined that the alternative method proposed by the licensee in RR ISI-024 will provide an acceptable level of quality and safety for the examination frequency requirements of the RVCHs. 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 one time use of RR ISI-024 at Calvert Cliffs for the duration up to and including the Spring 2022 RFO (Unit 1) and the Spring 2023 RFO (Unit 2) which will occur in the fifth ten-year ISI inspection interval.

All other requirements of the ASME Code, Section XI, and 10 CFR 50.55a(g)(6)(ii)(D) 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: Margaret T. Audrain

Date: December 7, 2015

If you have any questions, please contact Alex Chereskin by phone at 301-415-2549, or by e-mail at Alexander.Chereskin@nrc.gov.

Sincerely,

/RA/

Travis Tate, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-317 and 50-318

Enclosure:
Safety Evaluation

cc w/enclosure: Distribution via Listserv

DISTRIBUTION:

PUBLIC	LPLI-1 R/F	RidsNrrDorlLplI-1
RidsNrrDeEpnb	RidsAcrsAcnw_MailCTR	RidsNrrLAKGoldstein
RidsNrrPMCalvertCliffs	RidsRgn1MailCenter	MAudrain, NRR

ADAMS Accession No.: ML15327A367 ***via dated e-mail**

OFFICE	NRR/DORLLPLI-1/PM	NRR/DORL/LPLI-1/LA	NRR/DE/EPNB/BC*	NRR/DORL/LPLI-1/BC
NAME	AChereskin	KGoldstein	DAlley	TTate
DATE	12/03/2015	12/03/2015	11/04/2015	12/07/2015

OFFICIAL RECORD COPY