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

December 3, 2015

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 NOS. 1 AND 2, AND BRAIDWOOD STATION, UNIT 1 - RELIEF FROM THE REQUIREMENTS OF THE ASME CODE (CAC NOS. MF1530, MF1531, AND MF1532

Dear Mr. Hanson:

By letter dated October 27, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14302A343), Exelon Generation Company, LLC (the licensee) submitted a request to the Nuclear Regulatory Commission (NRC) for the use of alternatives to certain American Society of Mechanical Engineers Boiler and Pressure Vessel Code Case N-729-1 for reactor head penetration weld requirements at Byron Station, Unit Nos. 1 and 2, and Braidwood Station, Unit 1.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(a)(3)(i) (retitled paragraph 50.55a(z)(1) by 79 FR 65776, dated November 5, 2014), 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 licensee has not adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(a)(3)(i) (retitled paragraph 50.55a(z)(1) by 79 FR 65776, dated November 5, 2014).

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

Sincerely,

Justin C. Poole, Acting Branch Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No(s). STN 50-456,
STN 50-454 and STN 50-455

Enclosure:
Safety Evaluation

cc w/encl: ListServ

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST NOS. I3R-27 AND I3R14 REGARDING

REACTOR VESSEL HEAD PENETRATIONS

EXELON GENERATION COMPANY, LLC

BYRON STATION, UNIT NOS. 1 AND 2, AND BRAIDWOOD STATION, UNIT 1

STN 50-456, STN 50-454 AND STN 50-455

1.0 INTRODUCTION

By letter dated October 27, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14302A343), Exelon Generation Company, LLC (the licensee) submitted a request to the Nuclear Regulatory Commission (NRC) for the use of alternatives to certain American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (B&PV Code) Case N-729-1 for reactor head penetration weld requirements at Byron Station, Unit Nos. 1 and 2, and Braidwood Station, Unit 1.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(a)(3)(i) (retitled paragraph 50.55a(z)(1) by 79 FR 65776, dated November 5, 2014), 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

In this relief request, the licensee proposes to use alternatives to the requirements of 10 CFR 50.55a(g)(6)(ii)(D), which specifies the use of ASME Section XI Code Case N-729-1.

Alternatives to requirements under 10 CFR 50.55a(g) may be authorized by the NRC pursuant to 10 CFR 50.55a(a)(z)(1) or 10 CFR 50.55a(a)(z)(2). In proposing alternatives or requests for relief, the licensee must demonstrate that: (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.

The NRC staff has reviewed and evaluated the licensee's request pursuant to 10 CFR 50.55a(a)(z)(1).

Enclosure

3.0 TECHNICAL EVALUATION

3.1 The Licensee's Alternative

The specific components involved are the Class 1, Code Item B4.20 reactor vessel closure head penetrations.

The Byron Station, Unit Nos. 1 and 2, and Braidwood Station, Unit 1, use the 2001 Edition with 2003 Addenda of the ASME B&PV Code, Section XI. Examinations of the reactor vessel closure head penetrations are performed in accordance with 10 CFR 50.55a(g)(6)(ii)(D), which specifies the use of Code Case N-729-1, with conditions. 10 CFR 50.55a(g)(6)(ii)(D)(5) requires that "if flaws attributed to (PWSCC [primary water stress corrosion cracking] have been identified, whether acceptable or not for continued service under Paragraphs -3130 or -3140 of ASME Code Case N-729-1, the re-inspection interval must be each refueling outage instead of the re-inspection intervals required by Table 1, Note (8) of ASME Code Case N-729-1."

The licensee proposes to use the re-inspection interval required by Table 1, of Code Case N-729-1, which for the licensee, would be every second refueling outage for all reactor vessel head penetration nozzles, including nozzles repaired using the embedded flaw repair method.

The licensee states that the proposed alternative would be used for the remainder of the Byron Station, Units 1 and 2, Third 10-year Inservice Inspection Interval that is scheduled to end on July 15, 2016 and the remainder of the Braidwood Station, Unit 1, Third Inservice Inspection Interval that is scheduled to end July 28, 2018.

PWSCC has been detected on Byron Station, Units 1 and 2, and Braidwood Station, Unit 1, reactor vessel closure head penetrations and, in order to meet the conditions of 10 CFR 50.55a(g)(6)(ii)(D)(5), examination is required each refueling outage. The licensee has proposed to use the ASME Code Case N-729-1, Table 1 requirements as an alternative to the examination frequency, which states:

"If flaws have been previously detected that were unacceptable for continued service in accordance with -3123.3 or that were corrected by a repair/replacement activity of -3132.2 or -3142.3(b), the reexamination frequency is the more frequent of the normal reexamination frequency (before RIY = 2.25) or every second refueling outage. Additionally, repaired areas shall be examined during the next refueling outage following the repair."

The licensee based acceptability of using these requirements on the report, "Materials Reliability Program: Reevaluation of Technical Basis for Inspection of Alloy 600 PWR Reactor Vessel Top Head Nozzles (MRP-395)." A summary of the licensee's basis for relief is as follows:

PWSCC Experience for Alloy 600 Reactor Vessel Closure Head Nozzles

Plant PWSCC experience for reactor vessel top head nozzles was assessed for cases in which meaningful crack growth rate data could be developed. Plant inspection experience for both cold heads and non-cold heads was assessed with regard to laboratory crack growth rates. The crack growth rates estimated by ultrasonic test (UT) examination data for cold head cases are consistent with the probabilistic crack growth rate inputs determined in MRP-395. The licensee states that the crack growth rate assumptions of the technical basis for the N-729-1 inspection requirements remain valid due to the control rod drive mechanism nozzle inspection experience.

Deterministic Crack Growth Analysis

The licensee stated that deterministic crack growth evaluation can be applied to assess PWSCC risks for specific components and operating conditions. The deterministic evaluation in MRP-395 determined the following:

- The current N-729-1 volumetric examination interval without previous PWSCC detection is adequate to provide sufficient opportunity for flaw detection prior to significant leakage or ejection risk
- The examination interval for reactor pressure vessel head (RPVH) operating at cold leg temperatures with previously detected PWSCC may be extended from the currently required interval of each refueling outage to every other refueling outage without introducing significant added risk of leakage or ejection
- The N-729-1 examination interval of each refueling outage for non-cold Alloy 600 heads with previously detected PWSCC is considered effective for limiting risks of leakage and ejection while not being overly conservative

Probabilistic Monte Carlo Simulation Analysis

The purpose of the probabilistic analysis is to quantify the risk of leakage and ejection through a comprehensive simulation of the PWSCC degradation process, including the introduction of a PWSCC initiation model. The probabilistic evaluation replaces many of the conservatisms of the deterministic evaluation with best estimates, and incorporates uncertainty to reflect lack of specifics about physical variability in the RPVH PWSCC degradation process. MRP-375 concludes that the risk of ejection is predicted to be acceptably low when periodic UT examinations are performed per the RIY = 2.25 interval of Code Case N-729-1. This conclusion is based on the acceptable risks achieved when no credit is given to reducing inspection intervals below RIY = 2.25 once PWSCC is detected and also the comparable benefits achieved when using a one or two cycle re-inspection interval.

Assessment of Concern for Boric Acid

The licensee proposes to continue bare metal visual examinations every outage as is currently required by N-729-1, as conditioned by 10 CFR 50.55a(g)(6)(ii)(D) to address the concern for potential boric acid corrosion.

3.2 NRC Staff Evaluation

The Byron and Braidwood units' reactor pressure vessel (RPV) upper heads are required to be inspected in accordance with 10 CFR 50.55a(g)(6)(ii)(D). The RPV upper heads at each of these units operate at near cold leg operating temperatures. This would allow the volumetric inspection frequency to be every third or fourth refueling outage. However, since three of the units had previously identified PWSCC in the nozzles and/or welds of their RPV upper heads, 10 CFR 50.55a (g)(6)(ii)(D) requires volumetric inspections for each nozzle (both repaired and unrepaired) at each refueling outage for heads at the three plants that have experience cracking. The licensee is requesting relief from the volumetric inspection requirement each outage, to allow volumetric inspection every other outage.

During its review of the proposed alternative, the NRC staff identified several technical issues which are critical to the NRC staff's evaluation of the licensee's proposed alternative. These issues are addressed below and include:

a. Prior issuance of a similar proposed alternative for Byron[1]
b. Results of deterministic crack growth analyses
c. Results of probabilistic crack growth analyses
d. Implications of boric acid corrosion

Prior issuance of a similar proposed alternative for Byron[1]

As a basis to support the acceptability of its current proposed alternative, the licensee noted that the NRC staff had authorized a similar alternative, i.e., inspection of the nozzles every other outage, in 2010. This authorization was in response to the identification of nozzle cracking at Byron in 2007. The NRC staff notes that the licensee is correct in that the NRC staff did authorize such an alternative for Byron in 2010 (ADAMS Accession No. ML120790647). However, the NRC staff notes that at the time the alternative was authorized, Byron was the only cold head plant which had experienced cracking, the cracking consisted of a single nozzle, the time frame in which it was observed, and that a follow up inspection had been conducted which did not identify any additional cracking. The NRC staff also notes that approval of the alternative was terminated (as a condition of the original relief) when additional cracking was detected at Byron in 2014. The NRC staff further notes that since its original authorization of Byron's proposed alternative, additional cracking has been observed at Byron 1, Braidwood 2, VC Summer, and Shearon Harris. These plants all operate at approximately the same temperature and have nozzles manufactured by the same manufacturer.

Based on the above analysis and operating experience which has occurred since the NRC staff's authorization of Byron's proposed alternative, the NRC staff finds that the licensee's arguments do not justify relaxation of the inspection schedule.

Results of deterministic crack growth analyses

As part of its basis for the acceptability of its proposed alternative the licensee conduced deterministic crack growth analyses. The licensee proposed that these analyses demonstrate that inspections of the nozzles every other outage are sufficient to provide opportunity for flaw detection prior to significant leakage or ejection risk.

In its review of the licensee's proposed alternative, the NRC staff evaluated the licensee's deterministic crack growth analyses. The NRC staff finds the model and the data upon which

[1] No similar proposed alternative for Braidwood was previously requested nor approved because nozzle cracking was not previously observed at Braidwood and at the time Byron was thought to be an outlier.

the model is based to be lacking because the model predicts crack growth only based on temperature and time at temperature. No other factors are included in the model. Since essentially all cracking of cold head nozzles has occurred in nozzles made by one manufacturer, it is a significant limitation that the model fails to consider all variables which influence cracking. The NRC position is that the time at temperature model is not a bounding analysis and does not address material properties of the penetration material or weld process characteristics that may enhance PWSCC initiation or crack growth rates.

Based on the above analysis, the NRC staff finds that the licensee's arguments do not justify relaxation of the inspection schedule.

Results of probabilistic crack growth analyses

A probabilistic analysis, described in MRP-395, has been used by the licensee as a basis to support its proposed alternative. In reviewing this analysis the NRC noted that a number of input variables were based on engineering judgement because of a lack of data. The NRC also noted that the model is based on a Weibull distribution. A Weibull distribution is a probability distribution named after Swedish mathematician Waloddi Weibull. If the Weibull distribution uses time-to-failure as an input, the distribution output provides a failure rate that, in theory, could be used to predict future failures. The NRC staff finds that this distribution does not completely address unknown factors that may be identified in the future such as material property factors and fabrication and weld process issues and, as such, cannot address future potential degradation susceptibilities. In support of this finding, the NRC staff notes that this distribution continues to evolve with time and the accumulation of operating experience. From 2003 to 2006, the time at temperature model predicted that it would be very unlikely that a nozzle or weld in a cold leg temperature RPV upper head would experience PWSCC. In 2007 the model was adjusted to account for the identification of cracking at Byron, Unit 2, a cold head plant. In 2010 the model was again adjusted due to the identification of cracking at Davis-Besse. Between 2010 and 2015 cold leg temperature RPV upper heads were inspected for a second time. PWSCC was identified in 5 of the 20 remaining cold leg temperature heads. These findings required a third revision of the model. The NRC staff has observed that, due at least in part to the evolution of the Weibull distribution, the probabilistic approach proposed by the licensee as a basis for its proposed inspection interval is likely to require additional future revisions due to future inspection findings.

Based on the above analysis including the licensee's use of engineering judgement in the probabilistic analysis and the continuing evolution of the Weibull distribution of operating experience, the NRC staff finds that the licensee's arguments do not justify relaxation of the inspection schedule. Therefore, the NRC staff continues to find that the inspection requirement during each refueling outage is a necessary defense in depth requirement to provide adequate assurance of structural integrity of the reactor coolant pressure boundary once PWSCC has been identified.

Assessment of Concern for Boric Acid

The NRC staff finds that the proposal to continue bare metal visual examinations every outage, as is currently required by N-729-1, as conditioned by 10 CFR 50.55a(g)(6)(ii)(D), to address the concern for potential boric acid corrosion is acceptable as part of a comprehensive inspection program.

Additional Observations

Although not specifically addressed in the licensee's basis for its proposed alternative, the NRC notes that its requirement to inspect heads in which cracking has occurred every outage has been in place since 2003. During that time the NRC staff notes that the ASME code has found the NRC's position to be persuasive as the Code adopted the NRC's required inspection interval in ASME Code Case N-729-4.

The NRC staff also notes that its requirement to inspect nozzles in heads in which cracking has been experienced has been very effective in detecting cracking prior to leakage and/or structural failure. In support of this observation the NRC staff notes that at one cold leg temperature plant, Shearon Harris, PWSCC has been found in one or more nozzles each of the past three refueling outages, the latest being in the spring of 2015. The requirement to inspect during each outage allowed the Shearon Harris licensee to identify cracking in the reactor coolant pressure boundary prior to leakage.

Summary

Given the operating history of cracking in cold head plants since 2007, the apparent influence of material specific issues and fabrication and weld processing issues which are not addressed in the time/temperature cracking model, the evolution of the Weibull model to predict cracking, and the long history of successful application of every outage inspections in detecting nozzle cracking prior to leakage, the NRC staff does not find a sufficient basis to relax the current required volumetric inspection frequency of each refueling outage at the Byron and Braidwood Units.

4.0 CONCLUSION

As set forth above, the NRC staff determines that the proposed alternative does not provide an acceptable level of quality and safety. Accordingly, the NRC staff concludes that the licensee has not adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1). Therefore, the NRC staff does not authorize the proposed alternative for the third ISI interval at Braidwood, Unit 1, and Byron, Units 1 and 2.

Principal Contributors: MAudrain
JCollins

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

Sincerely,

/RA/

Justin C. Poole, Acting Branch Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No(s). STN 50-456,
STN 50-454 and STN 50-455

Enclosure:
Safety Evaluation

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***NLO subject to correcting and changing inconsistent sentence that Commission is authorized.**
JSW response: The consistent sentence was eliminated in its entirety as it is unnecessary to make that find in a denial – JSW 11/25/15