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10 CFR 50.55a

RS-14-314

October 27, 2014

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Braidwood Station, Unit 1
Facility Operating License No. NPF-72
NRC Docket No. STN 50-456

Byron Station, Units 1 and 2
Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. STN 50-454 and STN 50-455

Subject: Requests for Relief for Alternate Examination Frequency Under ASME Code Case N-729-1 for Reactor Vessel Head Penetration Welds in accordance with 10 CFR 50.55a(a)(3)(i)

References:

(1) Letter from Patrick Simpson (EGC) to U.S. NRC, "Requests for Relief for Alternate Examination Frequency Under ASME Code Case N-729-1 and from Requirements for Limited Examination of Reactor Vessel Head Penetration Welds in accordance with 10 CFR 50.55a(a)(3)(i)," dated April 2, 2009, ADAMS Accession No. ML091030444

(2) Letter from Patrick Simpson (EGC) to U.S. NRC, "Supplement of Request for Relief from Requirements for Limited Examination of Reactor Vessel Head Penetration Welds in Accordance with 10 CFR 50.55a(a)(3)(i)," dated December 17, 2009, ADAMS Accession No. ML093520172

(3) Letter from Stephen Campbell (U.S. NRC) to Charles Pardee (EGC), "Byron Station, Unit No. 2 – Relief Request I3R-16 for Reactor Pressure Vessel Head Penetration Examination Frequency (TAC No ME1066)," dated January 28, 2010, ADAMS Accession No. ML100210231

(4) Letter from J. L. Hansen (EGC) to U.S. NRC, "Third 10-Year Inservice Inspection Interval Requests for Relief for Alternative Requirements for the Repair of Reactor Vessel Head Penetrations," dated April 19, 2011, ADAMS Accession No. ML111100620

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(5) Letter from Jacob Zimmerman (U. S. NRC) to M. J. Pacilio (EGC), "Braidwood Station, Units 1 and 2 and Byron Station, Unit Nos. 1 and 2 – Relief Requests I3R09 and I3R-20 Regarding Alternative Requirements for Repair of Reactor Vessel Head Penetrations (TAC Nos. ME6071, ME6073, and ME6074)," dated March 29, 2012, ADAMS Accession No. ML120790647

(6) Letter from David M. Gullott, (EGC) to U.S. NRC, "Revision to the Third 10-Year Inservice Inspection Interval Requests for Relief for the Repair of Reactor Vessel Head Penetrations," dated September 8, 2014, ADAMS Accession No. ML14251A536

In accordance with 10 CFR 50.55a, "Codes and standards," paragraph (a)(3)(i), Exelon Generation Company, LLC (EGC), submitted relief request I3R-16 and I3R-17 for Byron Station Unit 2 (Reference 1). In Reference 2, EGC withdrew relief request I3R-17. The relief request (i.e., I3R-16) proposed an alternate examination schedule for volumetric and surface examinations to that required by 10 CFR 50.55a(g)(6)(ii)(D)(*5*) which modified Code Case N-729-1, Note (8) which requires re-inspection[1] each refueling outage instead of the Code Case N-729-1 re-inspection frequency. The relief request also requires re-inspection of Byron, Unit 2 Penetration 68 each refueling outage because it was previously repaired. In Reference 3, the NRC provided their authorization to implement Relief Request I3R-16.

Note, while Reference 3 allowed a re-inspection frequency of every other outage for Byron Station, Unit 2 penetrations (with the exception of flawed penetration number 68) for the remainder of the third 10-year Inservice Inspection (ISI) interval, the Safety Evaluation stipulated that the alternate schedule was not authorized should any additional indications of Primary Water Stress Corrosion Cracking (PWSCC) be found on the Byron Station Unit 2 Reactor Pressure Vessel (RPV) head penetration nozzles or associated J-groove welds. In the recent refueling outage of Fall 2014 (B2R18), an indication was discovered in Byron Station, Unit 2, RPV Penetration 6. The apparent cause of the indication was attributed to PWSCC. Therefore, the inspection interval approved for Byron Station, Unit 2 in Reference 3 is no longer authorized for the remainder of the third 10-year ISI interval and all Byron Station, Unit 2 penetrations are included as part of this relief request.

In accordance with 10 CFR 50.55a, "Codes and standards," paragraph (a)(3)(i), Exelon Generation Company, LLC (EGC), submitted relief requests (RRs) I3R-20 for Byron Station, Units 1 and 2, and I3R-09 for Braidwood Station, Units 1 and 2, (i.e., Reference 4). The RRs proposed an alternative repair technique using weld overlays on the reactor vessel head penetration housing and J-groove welds, using a Westinghouse embedded flaw repair method. EGC proposed the alternative for indications that may be

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[1] Note, 10 CFR 50.55a(g)(6)(ii)(D)(*5*) refers to intervals as "re-inspection" intervals while the ASME Code Case N-729-1 refers to intervals as "reexamination" intervals. For the purpose of this relief request, "re-inspection" and "reexamination" are synonymous.

encountered in the future, and that may be the result of PWSCC. In Reference 5, the NRC provided their authorization to implement Relief Requests I3R-09 and I3R-20, Revision 1 as a repair method for degradation identified in Reactor Vessel Head Penetrations.

In Reference 6, EGC submitted a Revision to the Relief Request for I3R-09 and I3R-20 (i.e., I3R-09 and I3R-20 Revision 2) which requested relief from performing surface examinations (i.e., dye penetrant (PT)) every cycle under certain conditions. EGC has reviewed the technical basis for requiring re-inspection of all nozzles each outage as required by 10 CFR 50.55a(g)(6)(ii)(D)(*5*) along with the examination results and personnel radiation exposure associated with examinations and determined that it is appropriate to relax the required re-inspection frequency. Attachment 1 provides Relief Requests Byron Station I3R-27 and Braidwood Station I3R-14 which are proposing relaxation of the re-inspection frequency as defined in 10 CFR 50.55a(g)(6)(ii)(D)(*5*) for Byron Station, Unit 1, Byron Station, Unit 2, and Braidwood Station, Unit 1 in accordance with 10 CFR 50.55a(a)(3)(i). Attachment 2 provides Electric Power Research Institute (EPRI) Report 3002003099, "Materials Reliability Program: Reevaluation of Technical Basis for Inspection of Alloy 600 PWR Reactor Vessel Top Head Nozzles (MRP-395)" which provides a technical basis for the re-inspection frequency change requested. Attachment 2 is provided based on an Outbound Copyright Release authorized for EGC by EPRI on October 24, 2014.

EGC requests approval of this proposed relief request by September 4, 2015, prior to beginning of the Bryon Station refueling outage in Fall 2015 (B1R20).

There are no regulatory commitments contained in this submittal.

If you have any questions regarding this matter, please contact Jessica Krejcie at (630) 657-2816.

Respectfully,

David M. Gullott
Manager - Licensing
Exelon Generation Company, LLC

Attachment 1: 10 CFR 50.55a Relief Requests Byron Station I3R-27 and Braidwood Station I3R-14, "Requests for Relief for Alternate Examination Frequency Under ASME Code Case N-729-1 for Reactor Vessel Head Penetration Welds *in accordance with 10 CFR 50.55a(a)(3)(i)*"

Attachment 2: EPRI Report 3002003099, "Materials Reliability Program: Reevaluation of Technical Basis for Inspection of Alloy 600 PWR Reactor Vessel Top Head Nozzles (MRP-395)," dated September 2014

bcc: Project Manager, NRR - Byron Station
Illinois Emergency Management Agency - Division of Nuclear Safety
Site Vice President - Byron Station
Vice President - Licensing and Regulatory Affairs
Regulatory Assurance Manager - Braidwood Station
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Attachment 1

10 CFR 50.55a RELIEF REQUESTS Byron Station I3R-27 and Braidwood Station I3R-14

Requests for Relief for Alternate Examination Frequency Under ASME Code Case N-729-1 for Reactor Vessel Head Penetration Welds in accordance with 10 CFR 50.55a(a)(3)(i)

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Requests for Relief for Alternate Examination Frequency Under ASME Code Case N-729-1 for Reactor Vessel Head Penetration Welds in accordance with 10 CFR 50.55a(a)(3)(i)

1.0 ASME CODE COMPONENT(S) AFFECTED

Component Numbers	Braidwood Station, Unit 1 and Byron Station, Units 1 and 2, Reactor Vessels 1RC01R (Unit 1) and 2RC01R (Unit 2)
Description:	Alternate Examination Frequency Under ASME Code Case N-729-1 for Limited Examination of Reactor Vessel Head Penetration Welds
Code Class:	Class 1
Examination Category:	ASME Code Case N-729-1
Code Item:	B4.20
Component Identification:	All reactor vessel closure head penetrations
Drawing Numbers:	Various

2.0 APPLICABLE CODE EDITION AND ADDENDA

American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, 2001 Edition, through 2003 Addenda. 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.

3.0 APPLICABLE CODE REQUIREMENT

10 CFR 50.55a(g)(6)(ii)(D)(*5*) requires that "If flaws attributed to [Primary Water Stress Corrosion Cracking] PWSCC 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."

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Byron Station, Units 1 and 2 and Braidwood Station, Unit 1 have repaired reactor vessel head penetrations due to PWSCC and therefore, currently require re-inspection[1] each refueling outage per 10 CFR 50.55a(g)(6)(ii)(D)(*5*). Byron Station, Units 1 and 2 and Braidwood Station, Unit 1 relief requests I3R-09 and I3R-20 (Reference 4) require repaired penetrations be examined and re-examined in accordance with Code Case N-729-1 including 10 CFR 50.55a(g)(6)(ii)(D) conditions as described in the reference relief requests.

As an alternative to the re-inspection frequency requirements prescribed in 10 CFR 50.55a(g)(6)(ii)(D)(*5*), frequency of examinations will be conducted in accordance with ASME Code Case N–729–1 Table 1 requirements based on Electric Power Research Institute (EPRI) Report 3002003099, "Materials Reliability Program [(MRP)]: Reevaluation of Technical Basis for Inspection of Alloy 600 PWR Reactor Vessel Top Head Nozzles (MRP-395)," dated September, 2014 (Reference 6).

4.0 REASON FOR THE REQUEST

The criteria to meet the conditions of 10 CFR 50.55a(g)(6)(ii)(D)(*5*) requires examination of closure head penetrations each refueling outage if PWSCC has been detected. Since PWSCC has been detected on Byron Station, Units 1 and 2 and Braidwood Station, Unit 1, examination of the reactor vessel closure head penetrations is required each refueling outage.

Reference 5 allowed a re-inspection frequency of every other outage for Byron Station, Unit 2 penetrations (with the exception of repaired penetration number 68) for the remainder of the third 10-year Inservice Inspection (ISI) interval. The Safety Evaluation specified that the alternate schedule was not authorized should any additional indications of PWSCC be found on the Byron Station, Unit 2 Reactor Pressure Vessel (RPV) head penetration nozzles or associated J-groove welds. In the recent refueling outage of Fall 2014 (B2R18), an indication was discovered in Byron Station, Unit 2, RPV Penetration 6. The apparent cause of the indication was attributed to PWSCC. Therefore, the inspection interval approved for Byron Station, Unit 2 in Reference 5 is no longer authorized for the remainder of the third 10-year ISI interval. This relief request also includes all Byron Station, Unit 2 penetrations (inclusive of repaired penetrations).

As an alternative to the re-inspection frequency requirements prescribed in 10 CFR 50.55a(g)(6)(ii)(D)(*5*), frequency of examinations will be conducted in accordance with ASME Code Case N–729–1 Table 1 requirements. ASME Code Case N-729-1 Table 1 Note 8 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 [Re-Inspection Years] =2.25) or every second

[1] Note, 10 CFR 50.55a(g)(6)(ii)(D)(*5*) refers to intervals as "re-inspection" intervals while the ASME Code Case N-729-1 refers to intervals as "reexamination" intervals. For the purpose of this relief request, "re-inspection" and "reexamination" are synonymous.

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refueling outage, and [Note (9)] does not apply. Additionally, repaired areas shall be examined during the next refueling outage following the repair."

Byron Station Units 1 and 2 and Braidwood Station Unit 1 RIY values range from approximately 0.420 to 0.488 per operating cycle. This results in a two cycle RIY of approximately 1. This RIY value is well under the 2.25 RIY value specified in Note 8 of Code Case N-729-1 Table 1 outlined above. Therefore, this relief request refers to a reexamination interval of every second refueling outage (i.e., every other outage) since this results in the more frequent of the normal reexamination frequency (before RIY = 2.25) or every second refueling outage as indicated in Note 8.

As described in Reference 6 Section 1, the original technical basis for the reactor vessel head inspection requirements defined in ASME Code Case N-729-1 is documented in Section 3 of Reference 7. The technical basis is supported by the Reference 8 safety assessment report and the safety assessments that it references including Reference 9. Note, the inspection requirements for top head nozzles developed on the basis of the Reference 8 safety assessment were published in Reference 7. These requirements were intended to supersede the inspection requirements of NRC Order EA-03-009, but instead, Reference 8 and Reference 7 formed the technical basis for the inspection requirements of ASME Code Case N-729-1 which replaced the NRC order as the current mandatory inspection requirements document (subject to certain conditions as listed in 10 CFR 50.55a(g)(6)(ii)(D)). The technical basis was originally developed in part based on plant experience with non-cold heads (i.e., reactor vessels operating at reactor hot-leg temperature (Thot)) experience with PWSCC.

The probabilistic fracture mechanics (PFM) analyses performed in Reference 9 used a Monte Carlo simulation algorithm to determine a probability of failure versus time for PWR vessel top heads for a set of input parameters, including operating temperature, inspection types (visual or volumetric NDE), and inspection intervals. Input into this algorithm included an experience-based time-to-leakage correlation based on a Weibull model of plant inspections, circumferential cracks, fracture mechanic analyses of various nozzle configurations containing axial and circumferential cracks, and the MRP-developed statistical crack growth rate model for Alloy 600 (Reference 10). The original technical basis for the N-729-1 inspection requirement (contained in Reference 7), concluded that a ceiling of two cycles (i.e., re-inspection/reexamination every other cycle) was a conservative approach.

Since the time of the Reference 9 analysis, indications of PWSCC have been identified in Alloy 600 CRDM nozzles in five domestic PWR cold heads (i.e., reactor vessels operating at reactor cold-leg temperature (Tcold) including Braidwood Station and Byron Station). Since the time of the Reference 2 analysis, indications of PWSCC have been identified in five domestic PWR cold heads. Therefore, it is appropriate to assess the implication of this new experience on the technical basis of the inspection frequency requirements evaluated by the MRP. The Reference 6 report evaluates the adequacy of the current inspection requirements, including the frequency of periodic volumetric or surface examinations for heads operating at Tcold, considering the recent cases of PWSCC reported in PWR cold heads. In addition, the Reference 6 report re-evaluates whether the original approach of a re-inspection interval of two 18-month fuel cycles (i.e., every other cycle) is justified for heads operating at Tcold in which PWSCC has been previously detected. The Reference 6 evaluation considers the following technical aspects: industry examination history of PWSCC, a deterministic crack growth rate analysis, a probabilistic Monte Carlo simulation analysis and also an assessment of the impact of boric acid. The Reference 6 evaluation determined that examination frequency per Code Case N-729-1 results in an acceptable level of quality and safety.

In addition to the technical aspects described above that have concluded an acceptable level of quality and safety exists with reexamination every other outage, examination of the Byron Station, Units 1 and 2 and Braidwood Station, Unit 1 reactor vessel head penetrations results in approximately 500-1000 mRem each outage. Also, EGC will continue to perform Code Case N-729-1 item number B4.10 visual examinations (VE) each refueling outage. Therefore, since the MRP evaluation has determined that examination frequency per Code Case N-729-1 (i.e., reexamination every other outage) results in acceptable level of quality and safety, EGC is requesting NRC approval to perform Code Case N-729-1 item number B4.20 examinations of the Byron and Braidwood reactor vessel closure heads without application of NRC condition 10 CFR 50.55a(g)(6)(ii)(D)(*5*).

5.0 PROPOSED ALTERNATIVE AND BASIS FOR USE

Basis for Use

Reexamination of the Byron Station, Units 1 and 2 and Braidwood Station, Unit 1 reactor vessel closure head penetrations per the Code Case N-729-1 Table 1 prescribed frequency will continue to ensure that degradation will be detected early and will not result in significantly increased probability of leakage or significantly reduce nuclear safety. Continued inspection per Code Case N-729-1 will reduce personnel exposure (i.e., approximately 500-1000 mRem) each outage an inspection is not required to be performed and will maintain an acceptable level of quality and safety. Reference 6 provides an industry report of the technical basis for this request. A summary of the report is provided below.

5.1 PWSCC Experience for Alloy 600 Reactor Vessel Closure Head Nozzles

Laboratory testing is the principal technique applied to determine relative crack growth rates for Alloy 600 wrought material. However, plant experience is a source of data that can in some cases be used to make estimates of the relative crack growth rate for comparison with statistical assessments of the laboratory crack growth rate data. 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 heads operating at temperatures significantly above Tcold (i.e., non-cold heads) was assessed with regard to implied relative crack growth rates. The first case of apparent PWSCC detected at a cold head was for the first in-service volumetric/surface examination and was associated with a weld fabrication flaw (Byron Station, Unit 2). As such, this case was not a good candidate for assessment. The crack growth rates implied by the ultrasonic examination data for the other cold head cases are consistent with the probabilistic crack growth rate inputs developed on the basis of the MRP-55 (Reference 10) assessment of laboratory crack growth rate

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data and used in the original MRP-105 (Reference 9) probabilistic assessment, as well as the current probabilistic assessment documented in Section 4 of the attached report. Furthermore, the cases in which relative crack growth rates could reasonably be inferred for non-cold heads were also consistent with the crack growth rate inputs of the probabilistic assessments. Hence, the crack growth rate assumptions of the technical basis for the N-729-1 inspection requirements remain valid in light of the CRDM nozzle inspection experience.

The findings of the top head examinations performed to date support the adequacy of the current inspection requirements (i.e., inspections every other outage), including the RIY = 2.25 interval for periodic volumetric/surface examinations:

- Since 2004, no circumferential PWSCC indications located near or above the top of the weld have been detected. These are the types of flaws that could produce a nozzle ejection were they to grow to a very large size.

- Since examinations capable of detecting flaws connected to the outer diameter (OD) surface of the nozzle tube were first applied in the early 2000's, there have been no reports of top head nozzle leakage (i.e., through-wall cracking) occurring after the time that the first in-service volumetric or surface examination was performed of all Control Rod Drive Mechanism (CRDM) or Control Element Drive Mechanism (CEDM) nozzles in a given head. The only incidence of nozzle leakage since 2004 was detected in 2010 during the first in-service inspection (after about six calendar years of operation) performed of a replacement Alloy 600 head procured from a cancelled plant. Thus, this initial examination experience is not directly relevant to the adequacy of the re-inspection interval requirement. No discernible corrosion was detected of the low-alloy steel head material during the bare- metal visual examinations of this replacement Alloy 600 head. It is noted that in late 2011 this first replacement head was replaced with a head having PWSCC-resistant nozzles.

- The volumetric or surface examinations performed on cold heads and the repeat volumetric or surface examinations performed on non-cold heads have been effective in detecting the PWSCC degradation reported in its relatively early stages, with modest numbers of nozzles affected by part-depth cracking, often located below the weld, where the nozzle tube is inside (not directly a part of) the pressure boundary.

- Five of the 20 operating cold heads with Alloy 600 nozzles have shown indications of PWSCC. This cracking was part-depth. For one of these five heads, the indication was associated with a weld fabrication defect. Hence, plant experience continues to show a very low probability of nozzle leakage for the cold heads given the examinations being performed.

5.2 *Deterministic Crack Growth Analysis*

Deterministic crack growth evaluation can be applied to assess PWSCC risks for specific components and operating conditions. In general, such deterministic evaluation quantifies the time between a certain initial condition with a known or hypothetical flaw

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size to some adverse condition such as through-wall growth, with a prescribed stability margin, etc., under a set of assumptions. This time may provide information and options for inspection intervals, mitigation, and repair. The evaluation described further in Section 3 of Reference 6 determined the following:

- The current N-729-1 volumetric examination interval (i.e., RIY = 2.25) for reactor pressure vessel head (RPVH) without previous PWSCC detection is adequate to provide sufficient opportunity for flaw detection prior to significant leakage or ejection risk.

- The examination interval for 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. For example, all calculations assume the existence of a roughly 10% through wall surface crack, among other conservatisms, and nevertheless predict times to leakage between 7 and 17 Effective Full Power Years (EFPY) at cold head temperatures.

- 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. This conclusion holds for operating temperatures bounding for the active fleet of Alloy 600 top heads.

5.3 Probabilistic Monte Carlo Simulation Analysis

The purpose of the probabilistic analysis is to quantify the risk of leakage and ejection more precisely through 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 incorporating uncertainty to reflect lack of specifics about physical variability in the RPVH PWSCC degradation process. Probabilistic predictions are in the form of event frequencies and probabilities. Based upon these predictions, RPVH examination intervals are recommended to achieve acceptable levels of leakage and ejection risk, both relative to risks predicted with currently accepted examination intervals and with respect to absolute core damage frequency limits.

The probabilistic results support the current inspection requirements (i.e., the requirements contained in ASME code case N-729-1) for Alloy 600 RPVH penetration nozzles, including for plants operating at Tcold. This probabilistic analysis is a key part of the updated technical basis of ASME Code Case N-729-1, superseding that of MRP-105, to include industry experience since 2004 and to replace the technical letter MRP 2011-034 (Reference 11) submitted to the U.S. NRC in December 2011. The key conclusions of this section, discussed further in Section 4 of Reference 6, are as follows:

- The risk of ejection is predicted to be acceptably low (below 5E-5 ejections per year per RPVH, averaged across the operating lifetime) when periodic UT examinations are performed per the RIY = 2.25 interval of ASME Code Case

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N-729-1. This is true despite taking no credit for more frequent inspections required after PWSCC detection by N-729-1 as conditioned by 10 CFR 50.55a(g)(6)(ii)(D).

- Average penetration leakage frequencies due to cracks initiating in the nozzle material are below 0.05 new leaking penetrations per year for all the cases evaluated (including cold and non-cold heads), up to and including inspection intervals of RIY=2.25.

- No leaks have occurred since the onset of complete head inspections, so the model provides a conservative evaluation of the potential for PWSCC flaws to grow without detection because the predicted Average Leakage Frequency (ALF) values are on the order of 0.02 - 0.1 leaks per head per year for non-calibrated initiation models.

- Even assuming a plant that is nominally as susceptible to PWSCC as the Alloy 600 replacement RPVH calibration case (see Reference 6 Section 4.2.2), the probabilistic analysis demonstrates that a re-inspection (i.e., UT inspection) interval based on RIY = 2.25 is sufficient to minimize the risk of leakage and ejection to acceptable levels. The RIY = 2.25 interval generally equates to an inspection interval of four or five 18-month cycles for a head operating at Tcold.

- Under various conditions, cases were run to investigate a re-inspection (i.e., UT inspection) interval of every refueling outage versus every other refueling outage. The absolute difference in average ejection frequency between these cases is generally small (e.g., less than 2E-6 for all conditions evaluated and less than 6E-7 for the conditions that did not assume a most-conservative initiation model based on the experience of the Alloy 600 replacement RPVH calibration case).

- The model sensitivity cases did not show significant deviation and support the robustness of conclusions drawn from the results.

The probabilistic Monte Carlo simulation analysis concluded that a reexamination (i.e., UT inspection) interval of one or two refueling outages for top heads operating at Tcold that have previously detected PWSCC results in an acceptably small effect on ejection and leakage risks. This conclusion is based on a) the acceptable risks achieved when no credit is given to reducing inspection intervals below RIY=2.25 once PWSCC is detected, and b) the comparable benefits achieved when using a one or two cycle re-inspection (i.e., UT inspection) interval. The probabilistic analyses conservatively assume a high likelihood that many PWSCC flaws are initiated and detected in the head over life.

5.4 Assessment of Concern for Boric Acid

The concern for boric acid corrosion of the low-alloy steel head material due to primary coolant leakage at a through-wall PWSCC flaw was considered in Section 5 of Reference 6. The concern for structural integrity of the pressure boundary directly due to circumferential PWSCC is discussed below.

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It is concluded that the current requirements for periodic VE for evidence of pressure boundary leakage (per ASME Code Case N-729-1 Table 1 item number B4.10 as conditioned by 10 CFR 50.55a(g)(6)(ii)(D)) remain valid to address the concern for potential boric acid corrosion. For heads with EDY ≥ 8 and/or previously detected PWSCC (i.e., Byron Station, Units 1 and 2 and Braidwood Station, Unit 1), the VE frequency of every refueling outage is appropriately conservative. For heads with EDY < 8 (effectively all heads with Alloy 600 nozzles operating in U.S. at Tcold) and no previously detected PWSCC (i.e., Braidwood Station Unit 2), the original basis for extending the interval to every third refueling outage or 5 calendar years, whichever is less, remains valid.

This approach is supported by the demonstrated low probability of pressure boundary leakage for heads operating at Tcold and the supplemental requirement for the VT-2 visual examination of the head under the insulation through multiple access points in outages that the VE is not completed. Given the large amounts of boric acid deposits that necessarily accompany substantial rates of boric acid corrosion, the VT-2 requirement is an effective supplement to the periodic VE examinations. It is emphasized that this conclusion is not dependent on the volumetric or surface reexamination interval for heads operating at Tcold with previously detected PWSCC being one rather than two 18-month fuel cycles. Plant experience and analyses show that the probability of leakage is low for heads operating at Tcold with previously detected PWSCC, regardless of the volumetric reexamination interval (i.e., one or two 18-month fuel cycles).

5.5 Repaired Nozzles

The analyses presented in Sections 3 and 4 of Reference 6 (i.e., the Deterministic Crack Growth Analysis and Probabilistic Monte Carlo Simulation Analysis) do not explicitly model repaired nozzles. However, as discussed below and further described in Section 6.2 of Reference 6, a reexamination interval of two 18-month cycles (i.e., every other refueling outage) in the case of previously detected PWSCC in a head operating at Tcold is also justified for the periodic Non-Destructive Examination (NDE) required for individual nozzles that have been repaired using either of the two main methods that have historically been used. These repair methods are (1) the embedded flaw repair (EFR) with application of a weld overlay on the outer nozzle and weld surfaces and (2) the "ID temper bead mid-wall repair." Note, Byron Station, Units 1 and 2 and Braidwood Station, Unit 1 have used the EFR method when repairs have been required. Currently, per approved NRC safety evaluations in response to relief requests associated with these repair methods, NDE of each repaired nozzle is performed during each refueling outage when all nozzles are examined per the volumetric or surface examination requirement of ASME Code Case N-729-1 as conditioned by 10 CFR 50.55a(g)(6)(ii)(D). The below discussion justifies that the NDE specific to repaired areas also be performed every other refueling outage in cases where an interval of two cycles is justified for the general (i.e., non-repaired nozzles) volumetric or surface examination of N-729-1 per Reference 6:

- The EFR for a flaw connected to the nozzle outer surface involves applying PWSCC-resistant weld metal (e.g., Alloy 52) over the OD of the Alloy 600 nozzle tube and the wetted surfaces of the J-groove weld, overlapping the vessel

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cladding and extending to the bottom of the nozzle, to isolate the susceptible material from primary coolant. The large majority of reactor vessel penetration nozzle PWSCC that has been detected has been located on the nozzle outer surface. Without contact with coolant, further PWSCC-induced growth is prevented. This repair is unlikely to significantly affect the stress state at the nozzle Inner Diameter (ID), and to the extent there is an effect on the stress at the ID, the squeezing of the nozzle tube by shrinkage of the weld overlay upon cooling would tend to reduce the magnitude of the tensile stress at the nozzle ID.

- Periodic reexamination (i.e., UT) on the nozzle ID, per the standard N-729-1 approach as conditioned by 10 CFR 50.55a(g)(6)(ii)(D), monitors the potential for growth of an embedded flaw originally located in the nozzle tube, or checks for growth into the nozzle tube of an embedded flaw originally located in the weld. The EFR technique has been applied in over 45 different instances throughout the world, and the flaw being repaired has never come into contact with water after repair. These repairs have been in place up to 10 years in some cases. Even in the unlikely case that the embedded flaw were to become wetted, any growth due to PWSCC would occur at a significantly reduced rate at Tcold compared to heads operating at temperatures similar to reactor hot-leg temperature (e.g., 2.8 times slower at 560°F compared to 600°F for the standard growth activation energy of 31 kcal/mole). The standard re-inspection (i.e., UT) on the nozzle ID also addresses the potential for new flaws initiating on the nozzle ID in the same manner as for an unrepaired nozzle.

In summary, reexamination every other outage is more than sufficient for periodic NDE after the embedded flaw repair based on (see Reference 6 references 53 and 54 for more information):

- The coverage of the entire outer nozzle surfaces with PWSCC-resistant material,
- The benefit of operating at Tcold for PWSCC crack growth rates, and
- The favorable plant experience, with over 45 repairs, some remaining in service for over 10 years to date.

5.6 Conclusions

Exelon proposes to perform reactor vessel head penetration examinations per Code Case N-729-1 as amended by 10 CFR 50.55a(g)(6)(ii)(D) with one exception. Instead of the re-inspection frequency described in 10 CFR 50.55a(g)(6)(ii)(D)(*5*), Exelon proposes to perform reexamination every second refueling outage (i.e., every other refueling outage). This examination frequency will be applied to all reactor vessel head penetration nozzles including those nozzles repaired using the EFR method. The technical basis report in Reference 6 supports a reexamination interval of two 18-month fuel cycles (i.e., every other refueling outage) for reactor vessel heads with previously detected PWSCC operating at Tcold.

It is noted that ASME Code Case N-729-1 specified that the re-examination interval be two fuel cycles or before RIY = 2.25, whichever is sooner, in the case of previously detected PWSCC requiring repair; however, as described above, the technical basis for this prescribed re-examination interval had not yet considered PWSCC in RPVHs

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operated at Tcold. The revised technical basis as contained in Reference 6 has been updated to include experience with Tcold heads and, in the same manner as the current technical basis, has demonstrated that reexamination performed every two cycles as currently described in in N-729-1 is a conservative approach. The revised probabilistic and deterministic calculations contained in Reference 6 are consistent with the probabilistic calculations originally performed in MRP-105 (Reference 9) which showed the RIY = 2.25 interval results in an acceptably small effect on nuclear safety, regardless of whether PWSCC has been previously detected. Additionally, EGC will continue to perform Code Case N-729-1, Item Number B4.10 visual examination each refueling outage. In summary, performing reexamination of the RPV head penetrations per Code Case N-729-1 defined frequency provides an acceptable level of quality and safety. Table 1 provides the NDE requirements that will be applied to nozzles repaired with the EFR method.

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Table 1
Embedded Flaw Repair Methods and Inspection Requirements

Repair Location in Original Component	Flaw Orientation in Original Component	Repair Method	Repair NDE Note (2)	ISI NDE Note (2)
VHP Nozzle/Tube ID	Axial or Circumferential	Seal weld	UT and Surface	UT or Surface
VHP Nozzle/Tube OD above J-groove weld	Axial or Circumferential	Note (1)	Note (1)	Note (1)
VHP Nozzle/Tube OD below J-groove weld	Axial or Circumferential	Seal weld	UT or Surface	UT or Surface
J-groove weld	Axial	Seal weld	UT and Surface, Note (3)	UT and Surface, Notes (3) and (4)
J-groove weld	Circumferential	Seal weld	UT and Surface, Note (3)	UT and Surface, Notes (3) and (4)

Notes:

(1) Repair method to be approved separately by NRC.
(2) Preservice and Inservice Inspection to be consistent with 10 CFR 50.55a(g)(6)(ii)(D), which requires implementation of Code Case N-729-1 with conditions; or NRC-approved alternatives to these specified conditions.
(3) UT personnel and procedures qualified in accordance with 10 CFR 50.55a(g)(6)(ii)(D), which requires implementation of Code Case N-729-1 with conditions. Examine the accessible portion of the J-groove repaired region. The UT plus surface examination coverage equals to 100%.
(4) Surface examination of the entire embedded flaw repair (EFR) shall be performed during each refueling outage. Surface examinations may be discontinued after the EFR or any localized welded repairs to the EFR subsequent to initial installation have been in service for two fuel cycles and the most recent examination satisfies ASME Section III, NB-5350 acceptance standards.

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6.0 DURATION OF THE PROPOSED ALTERNATIVE

The duration of the proposed alternative is for the remainder of the Byron Station, Units 1 and 2, Third Inservice Inspection Interval currently scheduled to end in July 15, 2016.

The duration of the proposed alternative is for the remainder of the Braidwood Station, Unit 1, Third Inservice Inspection Interval currently scheduled to end in July 28, 2018.

7.0 PRECEDENTS

None

8.0 REFERENCES

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2. Materials Reliability Program: Probabilistic Fracture Mechanics Analysis of PWR Reactor 2074 Vessel Top Head Nozzle Cracking (MRP-105), EPRI, Palo Alto, CA: 2004. 1007834. [NRC 2075 ADAMS Accession No. ML041680489]
3. T_{cold} RV Closure Head Nozzle Inspection Impact Assessment, EPRI, Palo Alto, CA: 2011. 2183 MRP 2011-034. [NRC ADAMS Accession No. ML12009A042]
4. Letter from David M. Gullott, EGC to USNRC, "Revision to the Third 10-Year Inservice Inspection Interval Requests for Relief for Alternative Requirements for the Repair of Reactor Vessel Head Penetration," dated September 8, 2014. [NRC ADAMS Accession No. ML14251A536]
5. Letter from Stephen Campbell (U.S. NRC) to Charles Pardee (EGC), "Byron Station, Unit No. 2 – Relief Request I3R-16 for Reactor Pressure Vessel Head Penetration Examination Frequency (TAC No ME1066)," dated January 28, 2010, [ADAMS Accession No. ML100210231]
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7. Materials Reliability Program (MRP): Inspection Plan for Reactor Vessel Closure Head Penetrations in U.S. PWR Plants (MRP-117). EPRI, Palo Alto, CA: 2004. 1007830 [freely available on www.epri.com]
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10. Materials Reliability Program: Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Thick-Wall Alloy 600 Materials (MRP-55) Revision 1, EPRI, Palo Alto, CA: 2002. 1006695. [freely available on www.epri.com]
11. Materials Reliability Program: Tcold RV Closure Head Nozzle Inspection Impact Assessment, EPRI, Palo Alto, CA: 2011. MRP 2011-034. [NRC ADAMS Accession No. ML12009A042]