Exelon Generation
4300 Winfield Road
Warrenville, IL 60555

www.exeloncorp.com

Exelon.
Nuclear

RS-09-132

10 CFR 50.55a

October 14, 2009

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

Byron Station, Unit 2
Facility Operating License No. NPF-66
NRC Docket No. STN 50-455

Subject: Supplement to 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)

References:

(1) Letter from P. R. Simpson (Exelon Generation Company, LLC) 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

(2) Letter from M. David (U.S. NRC), "Summary of September 2, 2009, Post-Submittal Public Meeting with Exelon Generation Company, LLC, to Discuss Byron Station, Unit No. 2, Relief Request for Alternate Examination Frequency Under ASME Code Case N-729-1 for Reactor Vessel Head Penetration Welds (TAC No. ME1066)," dated September 8, 2009

In Reference 1, Exelon Generation Company, LLC, (EGC) requested in accordance with 10 CFR 50.55a, "Codes and standards," paragraph (a)(3)(i), NRC approval of relief requests I3R-16 and I3R-17 for the Byron Station, Unit 2. These relief requests are for the third 10-year interval inservice inspection (ISI) interval.

As documented in Reference 2, EGC met with the NRC on September 2, 2009, to discuss the technical justification for the reliefs submitted in Reference 1. During a conference call on September 24, 2009, with EGC, the NRC provided feedback regarding relief request I3R-16 requested in Reference 1 in light of the information shared during the September 2, 2009 meeting. EGC agreed to supplement Reference 1 to address the NRC feedback by October 23, 2009.

Attachments 1 and 2 of this letter address the NRC's feedback and supersede Attachments 1 and 2 of Reference 1. Attachments 3, 4, and 5 of Reference 1 are unaffected by the supplemental information and are not being included with this submittal.

EGC requests approval of these relief requests by April 2, 2010, in support of the Byron Unit 2 spring 2010 refueling outage (B2R15).

There are no regulatory commitments contained in this letter. If you have any questions concerning this letter, please contact Ms. Lisa A. Schofield at (630) 657-2815.

Respectfully,

Patrick R. Simpson
Manager – Licensing

Attachments:

1. 10 CFR 50.55a Relief Request I3R-16 Revision 1
2. 10 CFR 50.55a Relief Request I3R-17 Revision 1

ATTACHMENT 1

10 CFR 50.55a Relief Request I3R-16

Revision 1

10 CFR 50.55a RELIEF REQUEST I3R-16 Revision 1
Attachment 1

Request 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. <u>ASME CODE COMPONENT(S) AFFECTED</u>

Code Class:	1
Reference:	ASME Code Case N-729-1
Item Number:	B4.10 and B4.20
Description:	B4.10, Head with UNS N06600 nozzles and UNS N06082 or UNS W86182 partial penetration welds, and UNS N06600 nozzles and B4.20, UNS N06082 or UNS N06082 or UNS W86182 partial-penetration welds in head
Drawing Numbers:	185282E Revision 1, 185283E Revision 1, and 185286 Revision 2

2. <u>APPLICABLE CODE EDITION AND ADDENDA</u>

The current code of record for the Byron Station Unit 2 Inservice Inspection (ISI) Third Ten-Year Interval is the ASME Section XI Code, 2001 Edition through the 2003 Addenda, as augmented by ASME Code Case N-729-1, "Alternative Examination Requirements for PWR Reactor Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds Section XI, Division 1," as amended and noticed in the Federal Register (73 FR 52730, September 10, 2008).

Upon implementation, ASME Code Case N-729-1 superseded the First Revised NRC Order EA-03-009 (i.e., Reference 8.1).

3. <u>APPLICABLE CODE REQUIREMENT</u>

Note (4) of ASME Code Case N-729-1, Table 1, "Examination Categories," states:

> If EDY <8 and no flaws unacceptable for continued service under –3130 or –3140 have been detected, the reexamination frequency may be extended to every third refueling outage or 5 calendar years, whichever is less, provided an IWA-2212 VT-2 visual examination of the head is performed under the insulation through multiple access points in outages that the VE is not completed. This IWA-2212 VT-2 visual examination may be performed with the reactor vessel depressurized.

Note (8) of ASME Code Case N-729-1, Table 1, "Examination Categories," states:

> If flaws have previously been detected that were unacceptable for continued service in accordance with –3132.3 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, and [Note (9)] does not apply.

10 CFR 50.55a RELIEF REQUEST I3R-16 Revision 1
Attachment 1
(Page 2 of 8)

Additionally, repaired areas shall be examined during the next refueling outage following the repair.

However, as identified in a final rule action published in 73 FR 52730, dated September 10, 2008, 10 CFR 50.55a(g)(6)(ii)(D)(*5*) has added the following condition modifying ASME Code Case 729-1, Note (8):

> If flaws attributed to 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.

4. **REASON FOR REQUEST**:

As part of the NRC required (Reference 8.1) reactor vessel upper head penetration nozzle weld volumetric examinations conducted using ultrasonic testing (UT) during Byron Unit 2 refueling outage B2R13 (i.e., spring 2007), an indication was detected in Penetration 68 that was suggestive of primary water stress corrosion cracking (PWSCC). Subsequent dye penetrant examination (PT) of the J-groove weld surface revealed a linear indication and a rounded indication as shown in Figure 1 below.

Figure 1: Dye penetrant examination results of Penetration 68 showing the rounded and the linear indications.

In order to determine the source of the detected indications, a specimen (more commonly referred to as a boat sample) of the J-groove weld and underlying penetration tube material was removed for examination. The boat sample contained the linear indication detected by UT and PT as well as a portion of a subsurface defect.

During the B2R13 refueling outage, Exelon Generation Company (EGC) submitted relief request I3R-14 (Reference 8.2) requesting authorization to repair the indication and the boat sample excavation site by an embedded flaw weld overlay process. The NRC subsequently approved this alternate repair request (Reference 8.3). The overlay repair covered the entire J-groove partial penetration weld area and the lower portion of the nozzle with PWSCC-resistant material.

10 CFR 50.55a RELIEF REQUEST I3R-16 Revision 1
Attachment 1

In September 2008, ASME Code Case N-729-1 was incorporated into 10 CFR 50.55a(g)(6)(ii)(D) and superseded Order EA-03-009 as the governing rule for Reactor Pressure Vessel (RPV) upper head penetration nozzle examinations. 10 CFR 50.55a(g)(6)(ii)(D)(5) states that if flaws attributed to PWSCC have been identified, 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.

In addition, with the detection of PWSCC, Item B4.10 now requires a visual examination to be completed each outage.

While the destructive examination of the boat sample confirmed the presence of PWSCC, the final report of the destructive examination concluded that the origin of the PWSCC was attributed to welding defects from the original fabrication and not solely due to exposure to the bulk operating environment.

Therefore, relief is requested from the requirements of 10 CFR 50.55a(g)(6)(ii)(D)(5) with a proposed alternative extent of examination of all nozzles, with the exception of Penetration 68, at a frequency of once every second refueling outage or four calendar years, whichever is less, and a bare metal visual examination of the RPV head each refueling outage. Penetration 68 will undergo volumetric, surface, and visual examinations each refueling outage as required by N-729-1.

This request is proposed for the remainder of the Byron Unit 2 Third ISI Inspection Interval beginning with refueling outage B2R15 in spring 2010.

5. <u>PROPOSED ALTERNATIVE AND BASIS FOR USE</u>:

EGC proposes to continue the required volumetric and surface examinations of 10 CFR 50.55a(g)(6)(ii)(D); however, an alternative inspection frequency is proposed based on the uniqueness of the occurrence of PWSCC in Penetration 68. Specifically, EGC is proposing to perform volumetric and surface examinations of all penetrations as identified by Table 1 of ASME Code Case N-729-1 at a frequency of once every second refueling outage or four calendar years, whichever is less, except for Penetration 68, which will be volumetrically, surface, and visually examined each refueling outage. In addition, bare metal visual examinations of the RPV head will occur each refueling outage. The proposed alternative examination frequencies provide an acceptable level of quality and safety.

ASME Code Case N-729-1 inservice examination methods and frequency are determined using parameters to characterize the susceptibility for PWSCC crack initiation and the potential for crack propagation. The parameters, effective degradation years (EDY) and reinspection years (RIY), are functions of time and temperature, and are normalized to a reference temperature of 600°F. EDY indicates the susceptibility to crack initiation, and RIY is an indicator of the potential for crack propagation. At the time of refueling outage B2R13, the EDY value for Unit 2 was 2.2, which is considered to be very low. Based on the low value of EDY, Byron Unit 2 was not expected to have initiated PWSCC by B2R13.

EGC had previously completed the volumetric examinations of Byron Unit 1 and Braidwood Unit 1 and 2 prior to the scheduled volumetric examination of Byron Unit 2 in

10 CFR 50.55a RELIEF REQUEST I3R-16 Revision 1
Attachment 1

B2R13. These reactor heads are the same vintage and design as the Byron Unit 2 head. During the examinations of the other three units, there were no indications of any PWSCC flaws that would have forecast the results for Byron Unit 2. The Byron Unit 2 RPV head is considered a T-cold head (~553°F) due to the reactor coolant system (RCS) cold leg bypass flow into the upper head region. Further, industry experience from Europe predominately pertained to RPV heads with higher temperatures than Byron Unit 2, and therefore, a higher susceptibility to initiation of PWSCC.

Byron Unit 2 Penetration 68 is the only one of over 1400 domestic, cold head RPV nozzles inspected to have found an indication of PWSCC. Statistically, the likelihood of Byron Unit 2, or any low susceptibility head, finding PWSCC-initiated indications at the refueling outage subsequent to the baseline inspection has been shown to be less than 1% and is independent of the indication found in Penetration 68. An extensive review of domestic and international industry operating experience with rounded surface indications on RPV head penetrations was conducted (Reference 8.4). The conclusion reported from the other industry examples stated that the PWSCC associated with the indication had crack growth propagating from the wetted surface toward the interior.

Since the probability of initiating PWSCC in Byron Unit 2 was very low, the decision was made during B2R13 to remove the detected indication and perform a destructive examination. A complete destructive metallurgical failure analysis was completed by Exelon PowerLabs to help determine the cause of the indication found, and the results are documented in Reference 8.6. The boat sample contained a portion of the axial indication identified by the UT and PT exams. A post-removal PT exam of the excavation site uncovered an angled, subsurface linear defect that intersected the original axial indication; the subsurface defect was partially captured by the boat sample. The orientation of the boat sample with respect to the penetration tube and the J-groove weld can be seen in Figure 2.

The crack-growth surface of the weld exhibited characteristics typical of both PWSCC and hot cracking. Lack of fusion between weld passes, which was parallel to the fusion line, could be seen on the crack-growth surface of the weld, and within the weld there were several cracks that were connected to the lack of fusion defect. Based on the general characteristics of the weld defects, interdendritic weld separations, direction of crack branching, and local ductile tearing, it was concluded that the primary direction of propagation within the weld was toward the wetted surface of the boat sample. These characteristics suggest the PWSCC did not initiate from the wetted surface of the boat sample. "Detail A" of Figure 2 illustrates the results of the UT exam and the location of the defect (i.e., weld defects and PWSCC cracking) that was partially captured in the boat sample and partially remained in the penetration tube. The spatial relationship between these features and the rounded indication that was not captured in the boat sample can also be seen in Figure 2.

Although the PWSCC did not initiate from the J-groove weld surface, the ingress of primary water to the subsurface weld defects is attributed to the rounded surface indication identified by the PT surface examination of the J-groove weld. The source of the original rounded indication on the weld wetted surface cannot be determined. However, based on the presence of welding defects in the boat sample, the most probable cause is considered a welding imperfection that was not detectable or an indication that was considered acceptable per the fabrication inspection requirements.

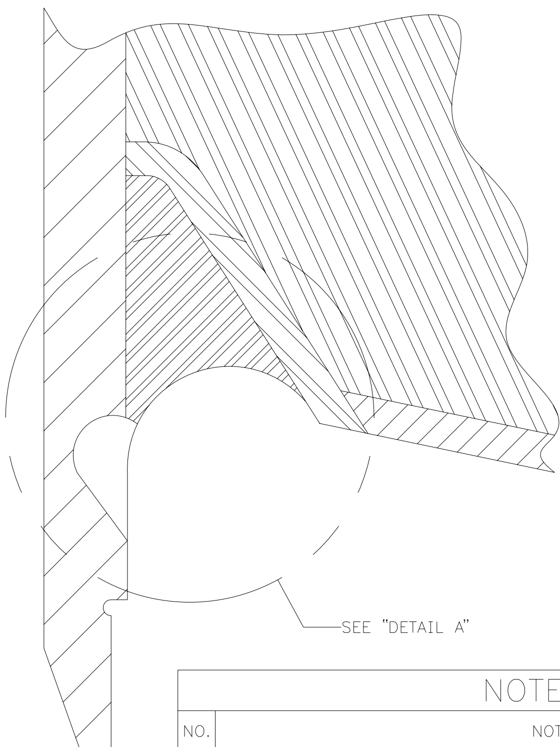
10 CFR 50.55a RELIEF REQUEST I3R-16 Revision 1
Attachment 1

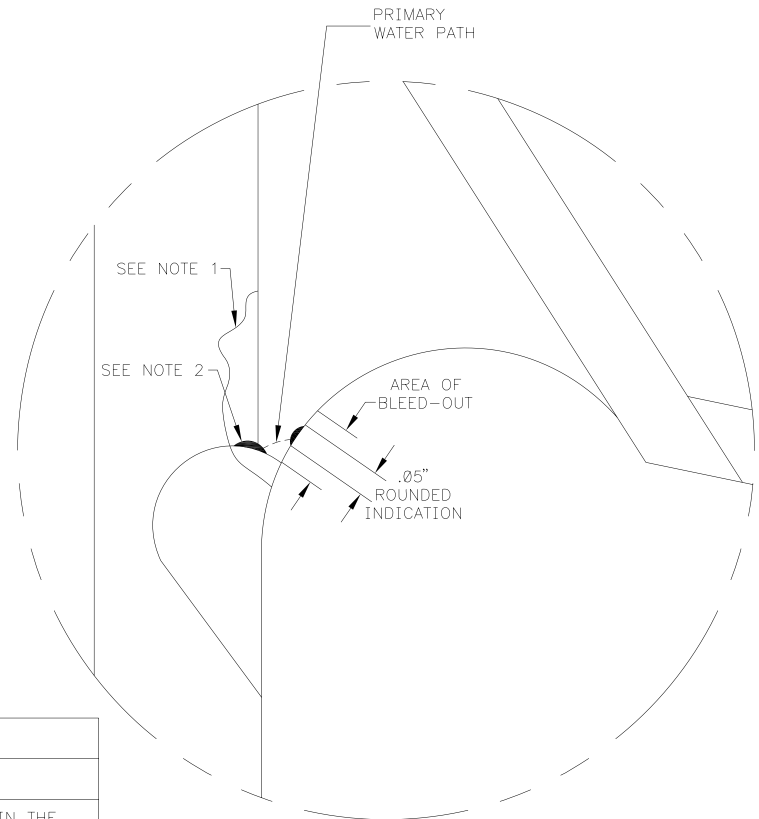
The Byron Unit 2 reactor head was fabricated by Babcock & Wilcox in 1977; the applicable code was the 1971 Edition of the ASME Section III, Summer 1973 Addenda. Per the 1971 edition of ASME Section III, NB-5000, an isolated 3/16" (0.1875") rounded dye penetrant indication would be considered acceptable. At 0.050", the rounded indication on the J-groove weld surface of Penetration 68 was well within this limitation. The subsurface volume of the rounded indication, as correlated to the size of the dye penetrant bleed area, would have been sufficient to be incident upon the network of interconnected weld defects.

At the time of B2R13, the presence of PWSCC in Byron Unit 2 was unexpected based on the low value of EDY. The conclusion drawn in the boat sample failure analysis report is that the PWSCC was not the result of exposure of the Alloy 600 tube material or the Alloy 182 weld material to the bulk primary water environment; rather, the premature initiation of PWSCC is attributed to a series of weld defects that created a conducive crevice corrosion environment in the high-stress region of the J-groove weld.

To illustrate the uniqueness of the PWSCC event at Byron Unit 2, an assessment of PWSCC in Penetration 68 has been conducted by employing probabilistic and structural reliability tools (Reference 8.4). The results showed that the probability of having the observed 50% through-wall flaw in Penetration 68 after 20 years of service were three orders of magnitude below those expected for flaw initiation and growth due to PWSCC in Byron Unit 2 and that the observed flaw did not occur in the most likely Control Rod Drive Mechanism (CRDM) penetration location in Byron Unit 2. The conclusion of the probabilistic calculations is that the PWSCC observed in Penetration 68 of Byron Unit 2 was not due to normal flaw initiation and growth by PWSCC in the Alloy 600 penetration tube material, and additional conditions, such as the weld defects identified in the boat sample metallurgical analysis, were needed to initiate and grow a PWSCC flaw in Penetration 68.

In accordance with NRC Order EA-03-009, which was the governing document in the fall of 2008, volumetric, bare metal visual, and surface examinations were performed on Byron Unit 2 during refueling outage B2R14 as required due to the discovery of PWSCC during the previous refueling outage (i.e., B2R13). The results showed no indications of PWSCC by volumetric examination, no evidence of boric acid on the RPV head, and no indications on the surface of Penetration 68 by dye penetrant examination.





NOTES	
NO.	NOTE
1.	SKETCH OF THE ORIGINAL ULTRASONIC INDICATION IDENTIFIED IN THE WESTINGHOUSE UT REPORT. THE INSPECTION IDENTIFIED A Ø.520" LONG x Ø.326" DEEP AXIAL, EXTERNAL INDICATION IN THE PENETRATION TUBE NEAR THE J-GROOVE ELEVATION. THE Ø.326" DEPTH CORRESPONDED TO APPROXIMATELY 52% OF THE Ø.625" TUBE THICKNESS.
2.	THIS REGION CONTAINED THE WELD DEFECTS AND THE SUBSURFACE DEFECTS THAT WERE CONTAINED IN THE BOAT SAMPLE.

Figure 2: Schematic representation of the removed boat sample illustrating the location of the rounded and linear defects with respect to the rounded surface indication that was not captured by the boat sample.

10 CFR 50.55a RELIEF REQUEST I3R-16 Revision 1
Attachment 1

Based on a Weibull analysis of cold head RPV penetrations and low-temperature bottom head penetration inspection and failure data (Reference 8.4), the probability of repair of one or more RPV penetrations at Byron Unit 2 during B2R14 in fall 2008 was approximately 1%. The examinations performed during B2R14 supported this analysis, since no flaws were detected and no repairs were required.

In addition to the destructive examination of the boat sample and the probabilistic and Weibull analyses of RPV penetration inspection results, a fracture mechanics analysis was performed to study the PWSCC crack growth behavior of the flaw in Penetration 68 (Reference 8.5). The analysis was conducted using the available stress distribution data, the industry-approved PWSCC crack growth formulation, and a standard fracture mechanics algorithm.

The results of the analysis showed that a minimum of six eighteen-month fuel cycles would be required to propagate a postulated flaw positioned in the middle of the J-groove weld at the point of the highest tensile stress to the top of the J-groove weld such that a leak path would be established. Therefore, at a minimum, an additional six fuel cycles would be necessary to establish a leak path if an undetected flaw was left in service. Further, the results of the analysis demonstrated that initiation of a flaw would require extenuating circumstances such as the fabrication defect found in Penetration 68, and the likelihood of detection of such a defect in other nozzles would be very high.

In an effort to forestall the initiation of PWSCC, zinc injection was implemented at Byron Unit 2 in Operating Cycle 12 following the spring 2004 refueling outage (B2R11). The purpose of zinc addition is to provide some mitigation of PWSCC with an ancillary benefit of radiation field reduction. Since the start of zinc addition in December 2004 through completion of operating Cycle 14 in October 2008, Byron 2 has accumulated 125.4 parts-per-billion-months (ppb-mos). Zinc addition has resumed in the current Operating Cycle 15, and plans call for the injection of zinc in future Byron 2 operating cycles with an average zinc concentration of 5 ppb. However, the decision to resume zinc addition as well as the levels and rates of injection for future cycles is dependent on the results of each cycle's chemistry and core data review.

In conclusion, based on the results of the boat sample examination, the probabilistic fracture mechanics study, and the crack growth analysis, EGC proposes that an acceptable alternative is a volumetric examination and surface examination frequency of every second refueling outage or four calendar years, whichever is less, and a bare metal visual examination of the RPV head each refueling outage. Penetration 68 will undergo volumetric, surface, and visual examinations each refueling outage as required by N-729-1.

6. <u>DURATION OF THE PROPOSED ALTERNATIVE</u>:

The duration of the proposed alternative is for Byron Station Unit 2 Third Inservice Inspection Interval currently scheduled to end in 2016.

7. <u>PRECEDENT</u>:

None

10 CFR 50.55a RELIEF REQUEST I3R-16 Revision 1
Attachment 1

8. REFERENCES:

8.1 NRC Order EA-03-009, "Issuance Of First Revised NRC Order (EA-03-009) Establishing Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors," dated February 20, 2004

8.2 Letter from D. M. Benyak (Exelon Generation Company, LLC) to U. S. NRC, "Byron Station Unit 2 Inservice Inspection Relief Request I3R-14: Alternative Requirements for the Repair of a Reactor Vessel Head Penetration," dated April 13, 2007

8.3 Letter from R. Gibbs (U. S. NRC) to C. M. Crane (Exelon Generation Company), "Byron Station, Unit No. 2 – Relief Request I3R-14 for Evaluation of Proposed Alternatives for Inservice Inspection Examination Requirements (TAC No. MD5230)," dated May 23, 2007

8.4 Exelon Nuclear Generating Company Nuclear Engineering Department Document AM-2007-11, "Byron Unit 2 – Technical Basis For Reactor Pressure Vessel Head Inspection Relaxation," Revision 1, dated September 27, 2007

8.5 Exelon Nuclear Generating Company Nuclear Engineering Department Document AM-2007-006, "Evaluation of Crack Growth of a Postulated Flaw in Byron Unit 2 CRDM Nozzles by Primary Water Stress Corrosion Cracking," Revision 0, dated July 4, 2007

8.6 Exelon PowerLabs Project BYR-48053, "Metallurgical Evaluations of a 'Boat' Sample from the #68 CRDM Penetration on Byron Unit 2," dated May 23, 2007

ATTACHMENT 2

10 CFR 50.55a Relief Request I3R-17

Revision 1

10 CFR 50.55a RELIEF REQUEST I3R-17 Revision 1
Attachment 2

Request for Relief from Requirements for Limited Examination of Reactor Vessel Head Penetration Welds In Accordance with 10 CFR 50.55a(a)(3)(i)

1. ASME CODE COMPONENT(S) AFFECTED:

Code Class:	1
Reference:	ASME Code Case N-729-1
Item Number:	B4.20
Description:	UNS N06082 Nozzles and UNS N06082 or UNS W86182 partial-penetration welds in head (Byron Station Unit 2 has seventy-nine (79) reactor pressure vessel (RPV) head penetration nozzles comprised of fifty-five (55) penetration tubes with thermal sleeves, twenty-three (23) locations without thermal sleeves, and one (1) vent penetration nozzle.)
Drawing Numbers:	185282E Revision 1, 185283E Revision 1, and 185286 Revision 2

2. APPLICABLE CODE EDITION AND ADDENDA:

The current code of record for the Byron Station Unit 2 Inservice Inspection (ISI) Third Ten-Year Interval is the ASME Section XI Code, 2001 Edition through the 2003 Addenda, as augmented by ASME Code Case N-729-1, "Alternative Examination Requirements for PWR Reactor Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds Section XI, Division 1," as amended and noticed in the Federal Register (73 FR 52730, September 10, 2008).

Upon implementation, ASME Code Case N-729-1 superseded the First Revised NRC Order EA-03-009.

3. APPLICABLE CODE REQUIREMENT:

Table 1, "Examination Categories," of ASME Code Case N-729-1 defines the examination requirements using Figure 2, "Examination Volume for Nozzle Base Metal and Examination Area for Weld and Nozzle Base Metal." Note (5) of Table 1 states:

> …If the examination area or volume requirements of Fig. 2 cannot be met, the alternative requirements of Appendix I shall be used and the evaluation shall be submitted to the regulatory authority having jurisdiction at the plant site.

10 CFR 50.55a RELIEF REQUEST I3R-17 Revision 1
Attachment 2

10 CFR 50.55a(g)(6)(ii)(D)(6) states:

> Appendix I of ASME Code Case N-729-1 shall not be implemented without prior NRC approval.

Further, ASME Code Case N-729-1, Figure 2, identifies the examination volume or surfaces as:

> a = 1.5 in. (38 mm) for Incidence Angle, θ, $\leq$ 30 deg and for all nozzles $\geq$ 4.5 in. (115 mm) OD or 1 in. (25 mm) for Incidence Angle, θ, > 30 deg; or to the end of the tube, whichever is less

where "a" is the length of the penetration nozzle beyond the J-weld.

4. REASON FOR REQUEST:

Due to the physical configuration and limitations of the examination equipment associated with fifteen RPV penetration nozzles, the full examination volume required by ASME Code Case N-729-1 Table 1 cannot be achieved for Item B4.20.

For Byron Station Unit 2, the bottom of each RPV head penetration nozzle includes a threaded region approximately 1.00 inch long on the outside diameter along with a chamfered area at the inside diameter which extends approximately 0.76 inches from the bottom of the penetration tube (see Figure 1). The chamfered surface is machined at a 20° angle.

The distance from the top of the thread relief to the bottom of the fillet of the J-groove weld, identified as "A" in Figure 1, varies based on location of the penetration in the RPV head. These distances are generally longer for penetrations at "inboard" locations and become progressively shorter for penetrations located farther away from the center of the RPV head. At the fifteen subject penetration nozzles (i.e., numbers 33, 34, 39, 42, 44, 45, 51, 52, 53, 55, 56, 58, 63, 69 and 71) the configuration is such that the distance "a" (as defined in ASME Code Case N-729-1, Figure 2) below the lowest point of the J-groove weld toe cannot be fully interrogated.

Table 1 contains information specific to the fifteen penetrations for which relief is being requested. The values for CRDM penetration hoop stress distributions at a point where the operating stress levels are less than 20 ksi tension (i.e., 20 ksi Line) were extrapolated from the associated graphs contained in Appendix A of Topical Report WCAP-16394-P, Revision 0, "Structural Integrity Evaluation of Reactor Vessel Upper Head Penetrations to Support Continued Operation: Byron and Braidwood Units 1 and 2," (Reference 8.1) dated February 2005. Reference 8.1 was submitted to the NRC as part of Reference 8.2.

10 CFR 50.55a RELIEF REQUEST I3R-17 Revision 1
Attachment 2
(Page 3 of 6)

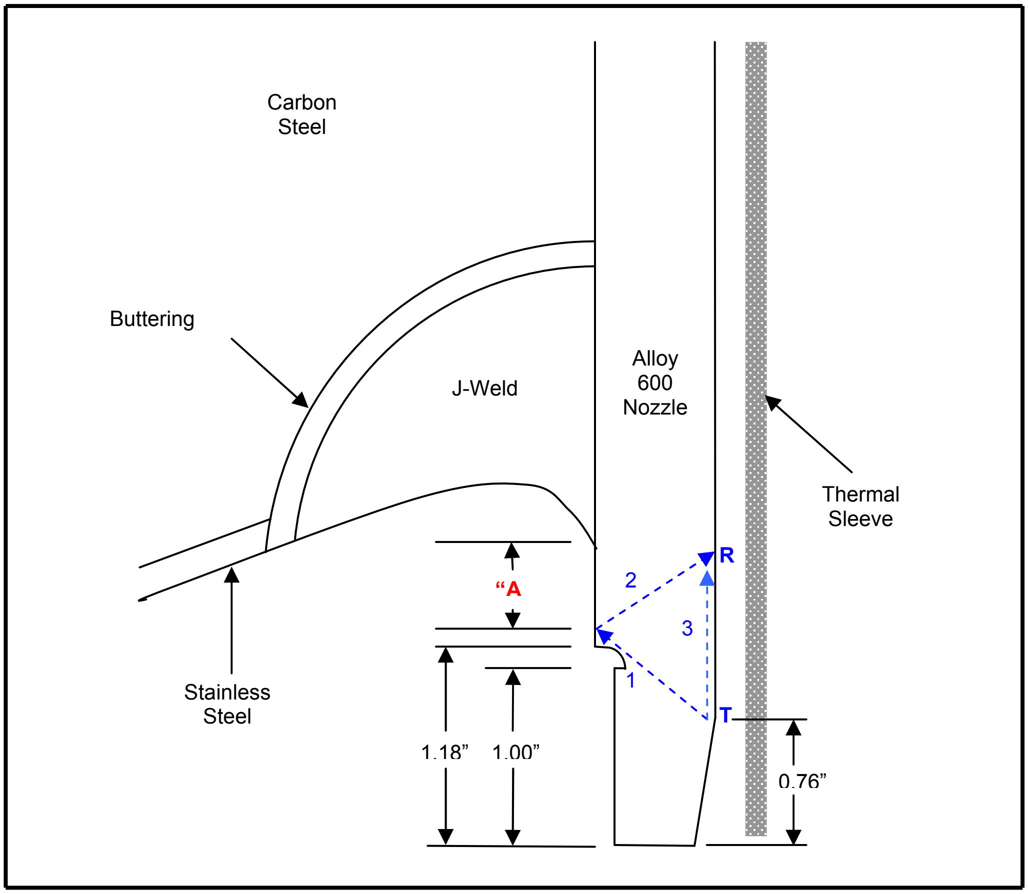


Figure 1
Illustration of Volumetric Examination Coverage on Byron Station Unit 2 Penetration Geometry (Including General Dimensions) at 0 Degrees

10 CFR 50.55a RELIEF REQUEST I3R-17 Revision 1
Attachment 2

Table 1
Penetrations with Limited Examination Volume
(Shaded Areas Do Not Meet N-729-1 Requirements)

Penetration Number	Angle (Degrees)	B2R13 Inspection Coverage (Inches Below Weld)	20 Ksi Line (Inches below J-Groove Weld)			
			Uphill Side		Downhill Side	
			ID	OD	ID	OD
33	29.3	0.920	1.85	.61	.45	.92
34	29.3	0.880	1.85	.61	.45	.92
39	32.9	0.800	1.85	.61	.45	.92
42	34.1	0.880	2.9	.62	.62	.46
44	34.1	0.840	2.9	.62	.62	.46
45	34.1	0.800	2.9	.62	.62	.46
51	35.2	0.960	2.9	.62	.62	.46
52	35.2	0.880	2.9	.62	.62	.46
53	35.2	0.880	2.9	.62	.62	.46
55	37.4	0.920	2.9	.62	.62	.46
56	37.4	0.640	2.9	.62	.62	.46
58	37.4	0.840	2.9	.62	.62	.46
63	42.8	0.760	2.9	.62	.62	.46
69	43.8	0.800	3.0	.60	.60	.46
71	43.8	0.960	3.0	.60	.60	.46

10 CFR 50.55a RELIEF REQUEST I3R-17 Revision 1
Attachment 2

5. PROPOSED ALTERNATIVE AND BASIS FOR USE

Exelon Generation Company, LLC, (EGC) proposes to define the lower boundary of the inspection volume for the effected RPV head penetration nozzles as: "the volume of the penetration tube extending from a distance "a" (as defined in ASME Code Case N-729-1, Figure 2) above the J-groove weld down to the lowest elevation that can be practically inspected." EGC proposes to use Appendix I of ASME Code Case N-729-1 to validate the alternative inspection volume. As originally submitted as Reference 8.2, the basis for establishing the frequency of examination and required coverage requirements used a similar methodology that is consistent with Appendix I of ASME Code Case N-729-1.

Testing of portions of the nozzle significantly below the J-groove weld is not significant to the phenomena of concern. The phenomena that are of concern are leakage through the J-groove weld and circumferential cracking in the nozzle above the J-groove weld.

In all cases, the measured coverage is adequate to allow Byron Station Unit 2 to continue to operate prior to the hypothetical flaws reaching the J-groove weld. In accordance with 10 CFR 50.55a(g)(6)(ii)(D)(5) requirements, the next examination required for the Byron Station Unit 2 RPV penetrations would be completed prior to flaw propagation into J-groove welds. In addition, the flaw propagation studies are aligned with the examination interval proposed in Relief Request I3R-16 contained in Attachment 1 of this submittal.

Control Rod Drive Mechanism (CRDM) Penetration 68 was previously included in the population of nozzles addressed in Reference 8.2; however, EGC proposes to perform volumetric and surface examinations of Penetration 68 each refueling outage as the result of the occurrence of PWSCC in this penetration. During B2R13 in spring 2007, an embedded flaw repair was made to Penetration 68 after the removal of a boat sample. A weld overlay of a PWSCC-resistant material (i.e., Alloy 52/152) was applied to the outer surface of the penetration tube, the J-groove weld, and an area of the RPV head extending one-half inch past the J-groove weld.

In conclusion, based on the probabilistic fracture mechanics study and the crack growth analysis, a volumetric examination frequency of every second refueling outage or four calendar years, whichever is less, provides an acceptable level of quality and safety for all penetrations, except Penetration 68, which will continue to be inspected at each refueling outage. Examination frequency and inspection requirements for Penetration 68 are discussed in detail in Attachment 1 of this submittal.

6. DURATION OF PROPOSED ALTERNATIVE:

The duration of the proposed alternative is for the remainder of the Byron Station Unit 2 Third Ten-Year ISI Interval currently scheduled to end in 2016.

10 CFR 50.55a RELIEF REQUEST I3R-17 Revision 1
Attachment 2

7. PRECEDENT:

Letter from R. Gibbs (U. S. NRC) to C. Pardee (Exelon Generation Company), "Byron Station, Unit No. 2 – Relaxation of the First Revised Order EA-03-009 (TAC No. MD6638)," dated February 7, 2008

8. REFERENCES:

8.1 WCAP-16394-P, Revision 0, "Structural Integrity Evaluation of Reactor Vessel Upper Head Penetrations to Support Continued Operation: Byron and Braidwood Units 1 and 2," February 2005

8.2 Letter from J. A. Bauer (Exelon Generation Company, LLC) to U. S. NRC, "Relaxation Request for First Revised Order EA-03-009 Establishing Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors," dated March 31, 2006

8.3 Letter from R. Gibbs (U. S. NRC) to C. Pardee (Exelon Generation Company), "Byron Station, Unit No. 2 – Relaxation of the First Revised Order EA-03-009 (TAC No. MD6638)," dated February 7, 2008