Exelon Generation

4300 Winfield Road
Warrenville, IL 60555

630 657 2000 Office

Proprietary information contained in Attachment 1
Withhold from public disclosure under 10 CFR 2.390
When separated, the cover letter and Attachments 2, 3 and 4 are Non-Proprietary

10 CFR 50.55a

RS-17-030

February 13, 2017

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

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

Subject: Supplemental Response to Request for Additional Information for Byron Station Relief Request I4R-10: Proposed Alternative Requirements for the Repair and Examination of Reactor Vessel Head Penetrations for the Fourth Inservice Inspection Interval

References:

(1) 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 13R-09 and 13R-20 Regarding Alternative Requirements for Repair of Reactor Vessel Head Penetrations (TAC Nos. ME6071, ME6073, and ME6074)," dated March 29, 2012 (ML120790647)

(2) Letter from David M. Gullott, (EGC) to U.S. NRC, "Revision to the Third 10-Year Inservice Inspection Interval Requests for Relief for Alternative Requirements for the Repair of Reactor Vessel Head Penetrations," dated September 8, 2014 (ML14251A536)

(3) Letter from Justin C. Poole, (U.S. NRC) to Bryan C. Hanson (EGC), "Byron Station, Units Nos. 1 and 2, and Braidwood Station, Units 1 and 2 – Relief from the Requirements of the ASME Code," dated January 21, 2016 (ML16007A185)

(4) Letter from David M. Gullott, (EGC) to U.S. NRC, "Relief Request for Alternative Requirements for the Repair and Examination of Reactor Vessel Head Penetrations for the Fourth Inservice Inspection Interval," dated August 16, 2016 (ML16229A250)

(5) Email from Joel Wiebe, (U.S. NRC) to Jessica Krejcie (EGC), "Preliminary Request for Additional Information (RAI) Regarding Relief Request I4R-10," dated December 8, 2016 (ML16343A252)

(6) Letter from David M. Gullott, (EGC) to U.S. NRC, "Response to Request for Additional Information for Byron Station Relief Request I4R-10: Proposed Alternative Requirements for the Repair and Examination of Reactor Vessel Head Penetrations for the Fourth Inservice Inspection Interval," dated December 29, 2016 (ML17003A274)

In Reference 1, the U. S. Nuclear Regulatory Commission (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 2, Exelon Generation Company, LLC (EGC) submitted a relief request that was applicable to the third 10-Year Inservice Inspection (ISI) interval and requested inspection frequency relief for the reactor vessel head penetrations repair weld surface examinations (i.e., dye penetrant (PT)) for Braidwood Station, Units 1 and 2 and Byron Station, Units 1 and 2. In Reference 3, the NRC approved the request for the third ISI interval for Braidwood Station and Byron Station.

In Reference 4, EGC submitted a relief request similar to the relief request that was approved in Reference 3. In Reference 5, the NRC requested additional information related to their review of Reference 4. EGC submitted a response to the Reference 5 request in Reference 6. The Reference 6 response was discussed during clarification teleconferences with the NRC on January 10, 2017 and January 13, 2017. During these clarification teleconferences it became evident that additional information was needed to supplement the Reference 6 response.

Specifically, the Reference 4 request includes a reference to Westinghouse Electric Company LLC (Westinghouse) WCAP-16401-P, Revision 0, "Technical Basis for Repair Options for Reactor Vessel Head Penetration Nozzles and Attachment Welds: Byron and Braidwood Units 1 and 2," dated March 2005. Section 3.3.2 of WCAP-16401 Revision 0 concludes that based on fatigue crack growth rate analytical results, the attachment weld region into the vessel head has at least 10 years of service life. Based on the table included in Reference 6 (page 3 of the cover letter) that noted the dates when flaws at Byron Station were repaired, it is evident that certain flaws will exceed the 10 years of service life during the Byron Station 4th ISI interval. Therefore, Westinghouse reevaluated the service life identified in section 3.3.2 of WCAP-16401 Revision 0.

Westinghouse reevaluated the service life using more realistic inputs. Specifically, the normal operating pressure was utilized instead of the design pressure and the flaw size aspect ratio was revised. This resulted in at least 40 years of service life for a hypothetical flaw that would encompass the entire attachment weld region. Based on the flaw sizes at Byron Station, the penetration nozzle attachment welds have at least a 40-year service life.

Note, Section 2 of WCAP-16401 which analyzes the service life of the weld repair penetration nozzle was not revised. Section 2.4 concludes that the penetration nozzle has at least a 20 year service life. Based on the table in Reference 6, all repaired nozzles have a service life that extends past the 4th ISI program interval which ends on July 15, 2025.

This response, including the attached revised relief request, supersedes the service life prediction in Reference 6 for the attachment weld region. The following attachments are included as part of this supplemental response:

1. WCAP-16401-P Revision 1, "Technical Basis for Repair Options for Reactor Vessel Head Penetration Nozzles and Attachment Welds: Byron and Braidwood Units 1 and 2" (Proprietary), January 2017
2. WCAP-16401-NP, Revision 1, "Technical Basis for Repair Options for Reactor Vessel Head Penetration Nozzles and Attachment Welds: Byron and Braidwood Units 1 and 2" (Non-Proprietary), January 2017
3. Application for Withholding Proprietary Information from Public Disclosure (for Attachment 1)
4. 10 CFR 50.55a Relief Request I4R-10, Revision 2, Alternative Requirements for the Repair of Reactor Vessel Head Penetrations In Accordance with 10 CFR 50.55a(z)(1)

Also enclosed (as part of Attachment 3) are the Westinghouse Application for Withholding Proprietary Information from Public Disclosure CAW-17-4539, accompanying Affidavit, Proprietary Information Notice, and Copyright Notice.

As Attachment 1 contains information proprietary to Westinghouse, it is supported by an Affidavit signed by Westinghouse, the owner of the information. The Affidavit sets forth the basis on which the information may be withheld from public disclosure by the NRC and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.390 of the NRC's regulations. Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the NRC's regulations.

The fourth interval of the Byron ISI Program began on July 16, 2016 and is scheduled to end on July 15, 2025. The attached relief request addresses potential repairs and inspections that would be performed during a refueling outage within the fourth interval and therefore, EGC requests approval of the proposed relief request by February 20, 2017, prior to the beginning of the Byron Station Unit 1 refueling outage in Spring 2017 (B1R21).

There are no regulatory commitments contained in this letter.

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

cc: Regional Administrator- NRC Region III
NRC Senior Resident Inspector- Byron Station
NRC Project Manager, NRR – Braidwood and Byron Station
Illinois Emergency Management Agency – Division of Nuclear Safety

ATTACHMENT 2

WCAP-16401-NP, Revision 1, "Technical Basis for Repair Options for Reactor Vessel Head Penetration Nozzles and Attachment Welds: Byron and Braidwood Units 1 and 2" (Non-Proprietary)

WCAP-16401-NP
Revision 1

January 2017

Technical Basis for Repair Options for Reactor Vessel Head Penetration Nozzles and Attachment Welds: Byron and Braidwood Units 1 and 2

Westinghouse

WCAP-16401-NP
Revision 1

Technical Basis for Repair Options for Reactor Vessel Head Penetration Nozzles and Attachment Welds: Byron and Braidwood Units 1 and 2

Alexandria M. Carolan*
Piping Analysis and Fracture Mechanics

January 2017

Reviewer: Anees Udyawar*
Piping Analysis and Fracture Mechanics

Approved: Benjamin A. Leber*, Manager
Piping Analysis and Fracture Mechanics

*Electronically approved records are authenticated in the electronic document management system.

Westinghouse Electric Company LLC
1000 Westinghouse Drive
Cranberry Township, PA 16066, USA

Record of Revisions

Rev	Date	Revision Description
0	April 2007	Original Issue
1	See EDMS	Revision 1 of this WCAP report is revised to extend the service life for the embedded flaw repair on the attachment welds from 10 years to 40 years. Changes are marked by revision bars in the left hand margin.

FOREWORD

This document contains Westinghouse Electric Company LLC proprietary information and data which has been identified by brackets. Coding $^{(a,c,e)}$ associated with the brackets sets forth the basis on which the information is considered proprietary. These codes are listed with their meanings in WCAP-7211 Revision 8 (September 2015), "Proprietary Information and Intellectual Property Management Policies and Procedures."

The proprietary information and data contained within the brackets in this report were obtained at considerable Westinghouse expense and its release could seriously affect our competitive position. This information is to be withheld from public disclosure in accordance with the Rules of Practice 10CFR2.390 and the information presented herein is safeguarded in accordance with 10CFR2.903. Withholding of this information does not adversely affect the public interest.

This information has been provided for your internal use only and should not be released to persons or organizations outside the Directorate of Regulation and the ACRS without the express written approval of Westinghouse Electric Company LLC. Should it become necessary to release this information to such persons as part of the review procedure, please contact Westinghouse Electric Company LLC, which will make the necessary arrangements required to protect the Company's proprietary interests.

The proprietary information in the brackets has been deleted in this report. The deleted information is provided in the proprietary version (WCAP-16401-P Rev. 1) of this report.

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1 INTRODUCTION

Leakage has been reported from the reactor vessel closure head penetration nozzles in a number of plants. This has led to requests for inspection of these regions. Inspections of the leaking penetrations indicate the presence of axial cracks that extend above and below the head penetration attachment welds. The cause of these axially oriented cracks has been determined to result from primary water stress corrosion cracking (PWSCC) that are driven by both steady state operating and residual stress. The residual stress is due to weld shrinkage and the offset geometry of the attachment weld that induces bending of the penetration nozzle. The bending also contributed to the penetration nozzle being ovalized over the attachment weld region.

As a part of the inspection and repair efforts associated with the head penetration inspection program for Byron and Braidwood Units 1 and 2, engineering evaluations were performed to support the Westinghouse embedded flaw repair method. [

]^{a,c,e} The methodology used is based on extensive analytical work completed to-date for the Westinghouse Owners Group (WOG), and a large collection of test data obtained under the sponsorship of Westinghouse, Babcock & Wilcox (B&W) and Combustion Engineering Owners groups (CEOG), as well as the Electric Power Research Institute (EPRI). The technical basis of the embedded flaw repair method is documented in WCAP-15987-P [1] and has been reviewed and accepted by the NRC. In the NRC Safety Evaluation that was incorporated in WCAP-15987-P, the NRC staff concluded that, subject to the specified conditions and limitations, the embedded flaw repair process described in WCAP-15987-P provides an acceptable level of quality and safety. The staff also concluded that WCAP-15987-P is acceptable for referencing in licensing applications.

Section XI of the ASME Code did not have a provision for welding over an existing flaw until the 1992 Edition. Those plants inspecting to an earlier edition of the Code would have to process relief requests to use the embedded flaw repair method, or update to the 1992 Edition of the Code. The repair and replacements rules of IWA-4000 in Section XI, starting with the 1992 Edition, allow weld repair over an existing flaw, provided that the flaw can be shown to be acceptable to the analytical requirements of Section XI. Engineering evaluations were performed and the results are presented in this report to provide the maximum flaw sizes that would satisfy the requirements in Section XI of the ASME Code [2] and be suitable to support the embedded flaw repair process.

Section XI repair rules allow the use of grinding to remove flaws, regardless of the edition of the Code. The only requirement is to ensure that the excavated region still meets the stress limits of the original construction code, which was Section III. Evaluations were performed and the results presented in this report address the effects of the local structural discontinuities resulting from the grinding operations performed to excavate flaws in the head penetration nozzles.

The purpose of this report is to provide plant-specific technical basis for the use of the embedded flaw repair method and to confirm that Byron and Braidwood Units 1 and Unit 2 meet the criteria for application of the embedded flaw repair process stated in Appendix C of WCAP-15987-P [1]. The results

presented in this report would enable the weld repair team to effectively determine the appropriate repair method.

In this report, the technical basis and evaluation results to support the use of the embedded flaw repair method for a flawed head penetration nozzle are provided in Section 2. The technical basis and evaluation results that support a similar application for a flawed head penetration attachment weld are provided in Section 3. Results of the evaluation providing a basis for grinding operations to excavate flaws in the head penetration nozzles are discussed in Section 4.

The purpose of Revision 1 of this WCAP report is to revise the technical basis to support the embedded flaw repair method for the attachment weld provided in Section 3. The projected life of the repair has been increased from 10 years to 40 years.

2 TECHNICAL BASIS FOR APPLICATION OF EMBEDDED FLAW REPAIR METHOD TO HEAD PENETRATION NOZZLES

This section provides a discussion on the technical basis for the use of embedded flaw repair method for a flawed head penetration nozzle. [

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Flaw evaluations for postulated planar flaws with various flaw sizes and shapes in the head penetration nozzles were performed. Based on the results of these evaluations, the largest flaw size that can be repaired using the embedded flaw repair method is determined.

2.1 Acceptance Criteria

The evaluation procedures and acceptance criteria for indications in austenitic piping are contained in paragraph IWB-3640 of ASME Section XI [2]. [

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2.1.1 Acceptance Criteria for Axial Flaws

For axial flaws, the hoop stress at limit load, σ_h, is first determined by using the following expression:

$$\sigma_h = \frac{\sigma_f}{SF_m}\left[\frac{1-\left(\frac{a}{t}\right)}{1-\left(\frac{a}{t}\right)/M_2}\right] \qquad (2\text{-}1)$$

where M_2 is the bulging factor given by:

$$M_2 = \left[1+\left(\frac{1.61}{4R_m t}\right)\ell^2\right]^{1/2} \qquad (2\text{-}2)$$

and

σ_f = $\frac{\sigma_u + \sigma_y}{2}$ (Average of Ultimate and Yield Strengths)

σ_h = PR_m/t (Hoop Stress due to Pressure P)

ℓ = Total Flaw Length

a = Flaw Depth

R_m = Mean Radius of Penetration Nozzle

t = Wall Thickness of Penetration Nozzle

SF_m = Safety Factor for membrane stress
2.7 for Level A Service Loading
2.4 for Level B Service Loading
1.8 for Level C Service Loading
1.3 for Level D Service Loading

The limits of applicability of this equation are $a/t \leq 0.75$ and $\ell < \ell_{allow}$

$$\ell_{allow} = 1.58(R_m t)^{0.5}[(\sigma_f / \sigma_h)^2 - 1]^{0.5} \tag{2-3}$$

This limit ensures that surface flaws would remain below the critical size based on the plastic collapse condition if they should grow through the wall.

2.1.2 Acceptance Criteria for Circumferential Flaws

For circumferential flaws, the axial membrane stress is calculated from internal pressure and axial components of other loads on the penetration nozzle. The axial primary membrane stress at limit load under pure membrane loading, σ^c_m, is calculated by using the following equations:

$$\sigma^c_m = \sigma_f[1 - (\frac{a\theta}{t\pi}) - \frac{2\varphi}{\pi}] \tag{2-4}$$

and

$$\varphi = \arcsin[\frac{1}{2}\left(\frac{a}{t}\right)\sin\theta]$$

where:

θ = Half Flaw Angle

σ_f = $\dfrac{\sigma_u + \sigma_y}{2}$ (Average of Ultimate and Yield Strengths)

The allowable primary membrane stress, S_t, is given by

$$S_t = \frac{\sigma^c_m}{SF_m} \tag{2-5}$$

where:

SF_m = Safety Factor for membrane stress depends on Service Level, as defined in Section 2.1.1.

2.2 METHODOLOGY

The evaluation assumes that a flaw has been detected in a penetration nozzle and that the embedded flaw repair method is used to seal the flaw from further exposure to the primary water environment. The

evaluation began with the determination of an allowable flaw size based on the acceptance criteria described in Section 2.1 for a flaw postulated in the penetration nozzle. [

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2.2.1 Geometry and Source of Data

There are many head penetration nozzles in the reactor vessel upper head. The dimensions of the penetration nozzles as well as the reactor vessel heads and the location of the nozzles are the same in all the Byron and Braidwood Units. The outermost penetration nozzles (penetrations 74-78) were selected for analysis because the stresses in the outermost penetration nozzles are in general more dominating. A schematic of a closure head penetration nozzle for a typical Westinghouse design plant is shown in Figure 2-1.

The dimensions of all the penetration nozzles are identical, with a 4.00 inch Outside Diameter (OD) and a wall thickness of 0.625 inches. The distributions of residual, transient thermal, and pressure stresses in the vessel head penetration nozzle were obtained from detailed three-dimensional elastic-plastic finite element analyses [3] for Byron and Braidwood Units 1 and 2. The through-wall stress distributions from the finite element analyses were used to determine the fatigue crack growth and the resulting allowable flaw size for the postulated flaw in the repaired penetration nozzles. [

]a,c,e The finite element model with the selected stress cuts used in the evaluation is shown in Figure 2-2. The term "stress cut" is defined as an imaginary line or plane over which stress distribution is evaluated. The regions of the head penetration that have the highest stresses are the ones in the vicinity of the attachment weld (Cuts 1 and 2 in Figure 2-2) which are the potential locations for crack initiation.

2.2.2 Loading Conditions

Thermal Transient Selection for Maximum Allowable Flaw Size Determination

The requirement for evaluating a flaw using the rules of Section XI is that the governing transients be chosen from the normal/upset conditions as well as from the emergency/faulted conditions. This is necessary because, as discussed in Section 2.1, different safety margins are used for the normal, upset, emergency, and faulted conditions. A lower safety factor is used to reflect a lower probability of occurrence for the emergency/faulted conditions.

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Thermal Transient Selection for Fatigue Crack Growth Prediction

[

]a,c,e The thermal transients that occur in the upper head region

are relatively mild, because most of the water in the head region has already passed through the core. The flow in the upper head is low compared to other regions of the reactor vessel, which mutes the effects of the operating thermal transients. The thermal transients that occur in Byron and Braidwood Units 1 and 2 are shown in Table 2-1. [

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2.2.3 Stress Intensity Factor

One of the key elements in a fracture mechanics evaluation is the determination of the crack driving force or stress intensity factor, K_I. This is based on the equations available in the literature.

Stress Intensity Factor for Surface Flaw

For a part-through wall flaw, the stress profile is approximated by a cubic polynomial as follows:

$$\sigma(x) = A_0 + A_1 x + A_2 x^2 + A_3 x^3$$

where:

x	=	The distance into the wall (inch)
σ	=	Stress perpendicular to the plane of the crack (ksi)
A_i	=	Coefficients of the cubic polynomial fit, i = 0, 1, 2, 3

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2.2.4 Allowable Flaw Size Determination

Allowable flaw sizes for axial and circumferential flaws with various aspect ratios (flaw length/flaw depth) in a penetration nozzle are calculated in accordance with the acceptance criteria discussed in Section 2.1. The thermal transients that have the [ ]a,c,e were considered in determining the allowable flaw sizes. It should be noted that these allowable flaw sizes must be adjusted to account for fatigue crack growth. Since the repaired flaws are embedded and sealed, they are not subjected to PWSCC and hence the only mechanism for sub-critical crack growth is fatigue. Adjustments to the allowable flaw sizes are based on the results from the fatigue crack growth evaluation described in Section 2.2.5.

2.2.5 Fatigue Crack Growth Prediction

The analysis procedure involves postulating various types of flaw in the penetration nozzle subject to a series of design loads. The applied loads include pressure, thermal transients and residual stresses. The governing thermal transients used for this evaluation are shown in Section 2.2.2. The cycles are distributed evenly over the entire plant design life. The stress intensity factor range, ΔK_I, which controls fatigue crack growth, depends on the geometry of the crack, its surrounding structure and the range of applied stresses in the region of the postulated crack. Once ΔK_I is calculated, the fatigue crack growth due to a particular stress cycle can be determined using a crack growth rate reference curve applicable to the material of the head penetration nozzle.

The fatigue crack growth rate (CGR) reference curve for nickel base alloys in air environment is based on the results reported in [6] and is shown below.

$$\frac{da}{dN} = CS_R \Delta K^n \tag{2-7}$$

$$C = 4.835\times10^{-14} + 1.622\times10^{-16}T - 1.490\times10^{-18}T^2 + 4.355\times10^{-21}T^3 \tag{2-8}$$

$$S_R = \left[1 - 0.82R\right]^{-2.2}$$

$$n = 4.1$$

where:

T = Average temperature of the transient (°C)

ΔK = Stress intensity factor range (MPa$\sqrt{m}$)

R = Stress Ratio (K_{min}/K_{max})

da/dN = Fatigue crack growth rate (meters / cycle)

The crack growth rate reference curve in air used in determining the fatigue crack growth into the repair weld Alloy 52 is not available. However, there are limited test data on Alloy 52 in PWR water environment and, based on these data, the crack growth rates for Alloy 52 and Alloy 600 were concluded to be the same in PWR water environment [6]. Therefore, the crack growth rate curve for Alloy 52 in air is assumed to be the same as that for Alloy 600 in air.

Once the incremental crack growth corresponding to a specific transient for a small time period is calculated, it is added to the original crack size, and the analysis continues to the next time period and/or thermal transient. The procedure is repeated in this manner until all the significant analytical thermal transients and cycles known to occur in a given period of operation have been analyzed.

2.3 FRACTURE MECHANICS ANALYSIS RESULTS

Axial and circumferential flaws found in a head penetration nozzle can be repaired using the embedded flaw repair method. A range of potential flaw sizes and shapes was investigated to thoroughly evaluate the use of embedded flaw repair.

2.3.1 Results for Allowable Flaw Sizes (Without Fatigue Crack Growth Adjustment)

Allowable Flaw Sizes for Axial Flaws

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]^{a,c,e} The allowable flaw sizes for a maximum design pressure of 2.5 ksi can be obtained as shown in Figure 2-3 for postulated inside surface axial flaws with various aspect ratios (flaw length/flaw depth). The allowable flaw sizes determined this way for the inside surface axial flaws can also be [ ]^{a,c,e} It should be noted that the allowable flaw sizes determined from Figure 2-3 must be adjusted to account for the fatigue crack growth of the repaired flaws, which are no longer subjected to stress corrosion cracking. The amount of adjustments is described in Section 2.3.2.

Allowable Flaw Sizes for Circumferential Flaws

[

]^{a,c,e} The allowable flaw sizes for a maximum design pressure of 2.5 ksi can be determined as shown in Figure 2-4 for postulated inside surface circumferential flaws with various aspect ratio (flaw length/flaw depth). The allowable flaw sizes determined this way for the inside surface flaws can also be [ ]^{a,c,e} It should be noted that the allowable flaw sizes determined from Figure 2-4 must be adjusted to account for the fatigue crack growth of the repaired flaws, which are no longer subjected to stress corrosion cracking. The amount of adjustments is described in Section 2.3.2.

2.3.2 Results for Allowable Flaw Sizes (With Fatigue Crack Growth Adjustment)

Fatigue crack growth (FCG) evaluation was performed to determine the potential crack growth for the inside surface, outside surface, and embedded flaws in a repaired penetration nozzle. It was determined that the FCG results for the outside surface flaws envelop those for the inside surface flaws and embedded flaws of comparable flaw sizes. Therefore, the FCG results for outside surface flaws are conservatively applied to the inside surface flaws as well as the embedded flaws.

Allowable Axial Flaw Sizes

Figures 2-5 and 2-6 show the fatigue crack growth prediction of the penetration nozzles for a range of flaw depths at the uphill side and downhill side, respectively. It should be noted that the total flaw depth is limited to 75% of the wall thickness in all cases except for the flaws with an aspect ratio (flaw length/flaw depth) of 10. The allowable flaw depth for a flaw with an aspect ratio of 10 is slightly over 74% of the wall thickness as shown in Figure 2-3. The maximum allowable flaw sizes accounting for fatigue crack growth in a repaired penetration nozzle can be determined from these figures by subtracting the fatigue crack growth increments shown on Figures 2-5 and 2-6 from the ASME Code allowable flaw sizes shown on Figure 2-3, for the desired period of service life. Figure 2-7 shows the maximum allowable axial flaw sizes taking into account fatigue crack growth for a 20 year period of service life.

Allowable Circumferential Flaw Sizes

Figures 2-8 and 2-9 show the fatigue crack growth prediction of the penetration nozzles for a range of flaw depths at the uphill side and downhill side, respectively. It should be noted that the total flaw depth is limited to 75% of the wall thickness in all cases. The maximum allowable flaw sizes accounting for fatigue crack growth in a repaired penetration nozzle can be determined from these figures by subtracting the fatigue crack growth increments shown on Figures 2-8 and 2-9 from the ASME Code allowable flaw sizes shown on Figure 2-4, for the desired period of service life. Figure 2-10 shows the maximum allowable circumferential flaw sizes taking into account of fatigue crack growth for a 20 year period of service life.

2.4 SUMMARY

Axial and circumferential flaws found on the inside surface or outside surface of a head penetration nozzle can be repaired using the embedded flaw repair method to seal it from the primary water environment. The maximum allowable axial and circumferential flaw sizes detected in the penetration nozzles that can be repaired using the embedded flaw repair method are shown in Figures 2-7 and 2-10 with the effects of fatigue crack growth included for a 20 year period of service life. For other periods of service life, the maximum allowable flaw sizes can be determined directly from Figures 2-5, 2-6, 2-8 and 2-9.

Table 2-1 (See also next page)

Summary of Reactor Vessel Transients for Byron and Braidwood Units 1 and 2 [7]

Normal Conditions	**Number of Occurrences**
Plant Heatup And Cooldown	200 (each)
Plant Loading and Unloading at 15% of Full Power / Minute	13,200 (each)
Step Load Increase and Decrease of 10% of Full Power	2,000 (each)
Large Step Load Decrease with Steam Dump	200
Steady State Fluctuation	infinite
Feedwater Cycling at Hot Shutdown	2,000
Loop Out of Service, Normal Loop Shutdown	80
Loop Out of Service, Normal Loop Startup	70
Unit Loading and Unloading between 0 and 15% Full Power	500 (each)
Boron Concentration Equalization	26,400
Refueling	80
Upset Conditions	
Loss of Load	80
Loss of Power	40
Partial Loss of Flow	80
Reactor Trip From Full Power	400
Inadvertent RCS Depressurization	20
Inadvertent Startup of an Inactive Loop	10
Control Rod Drop	80
Inadvertent S.I. Actuation	60
Test Conditions	
Turbine Roll Test	20
Primary Side Hydrostatic Test	10
Secondary Side Hydrostatic Test	10
Primary Side Leak Test	200
Secondary Side Lead Test	80
Tube Leakage Test	800
Emergency and Faulted Conditions	
Small Loss-of-Coolant Accident (LOCA)	5
Small Steam Break	5

Table 2-1 (Continued)

Summary of Reactor Vessel Transients for Byron and Braidwood Units 1 and 2 [7]

Emergency and Faulted Conditions (Continued)	
Complete Loss of Flow	5
Reactor Coolant Pipe Break (Large LOCA)	1
Large Steam Line Break	1
Feedwater Line Break	1
Reactor Coolant Pump Locked Rotor	1
Control Rod Ejection	1

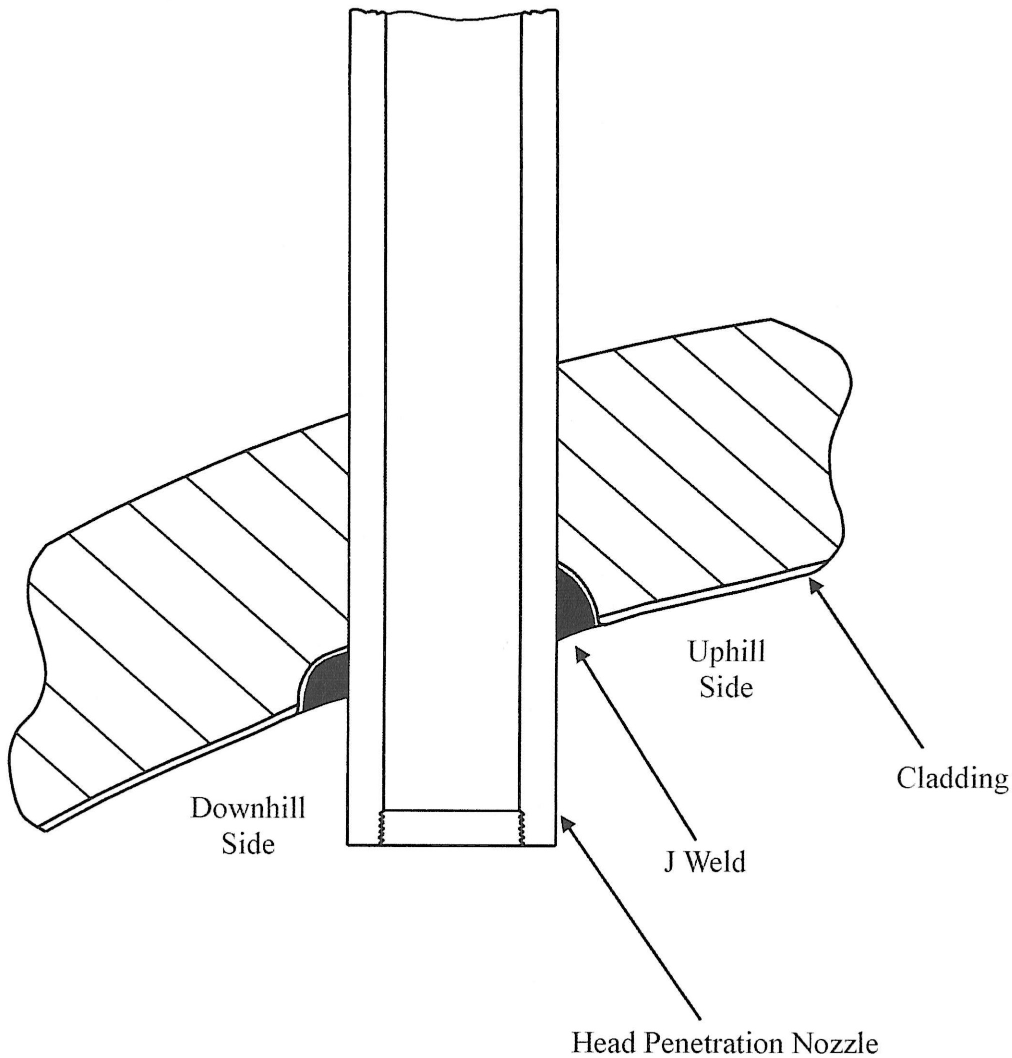


Figure 2-1 Geometry of a Typical Closure Head Penetration

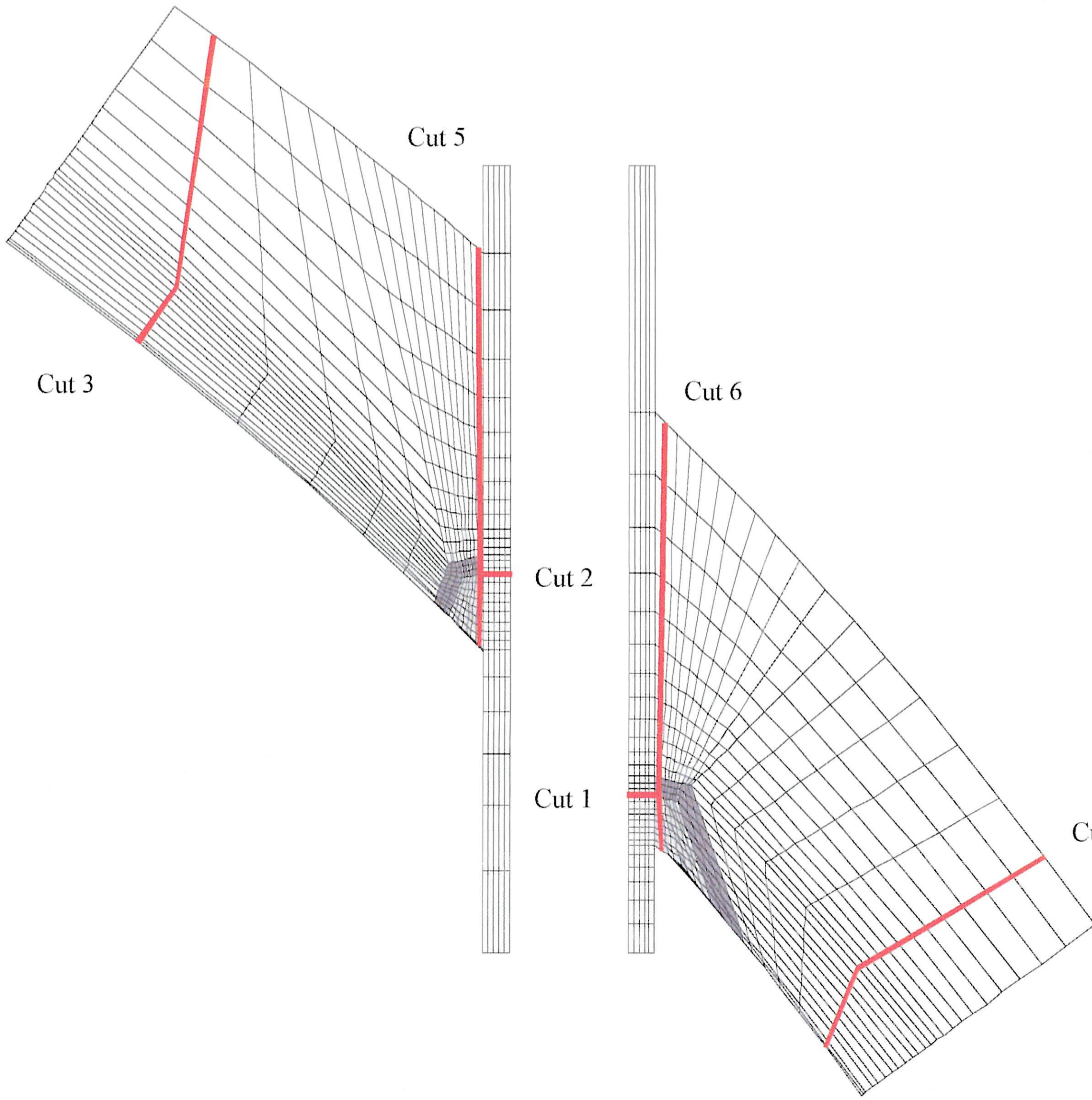


Figure 2-2 Analytical Stress Cuts Taken from the Finite Element Model

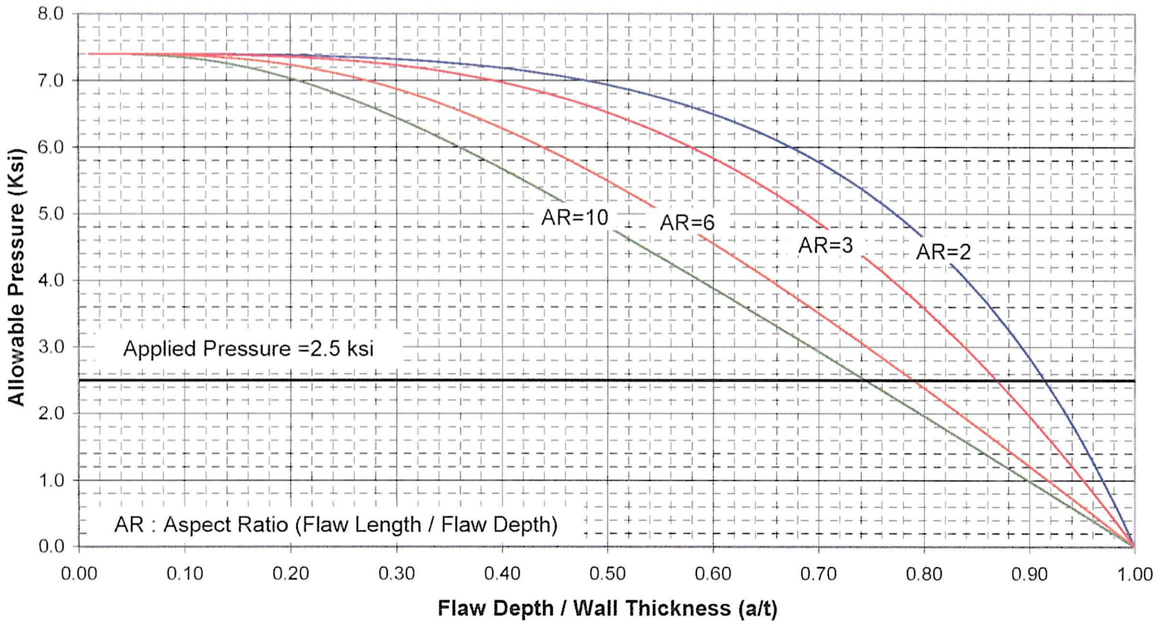


Figure 2-3 Allowable Axial Flaw Sizes In Penetration Nozzle (Without Fatigue Crack Growth)
[ ]^{a,c,e}

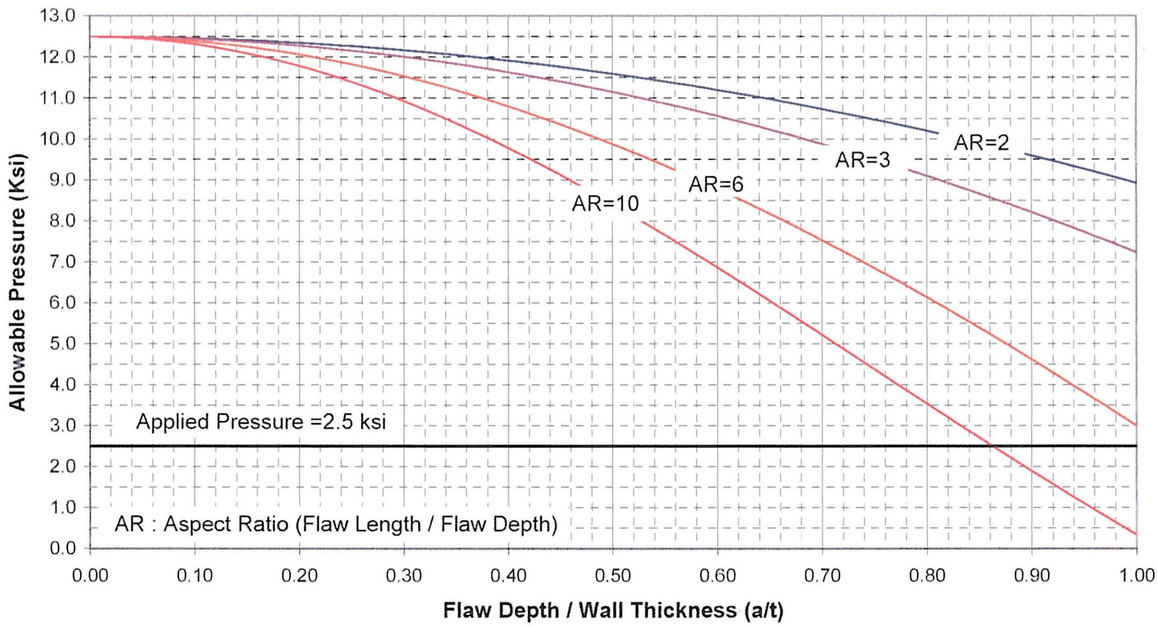


Figure 2-4 Allowable Circumferential Flaw Sizes In Penetration Nozzle (Without Fatigue Crack Growth) [ ]a,c,e

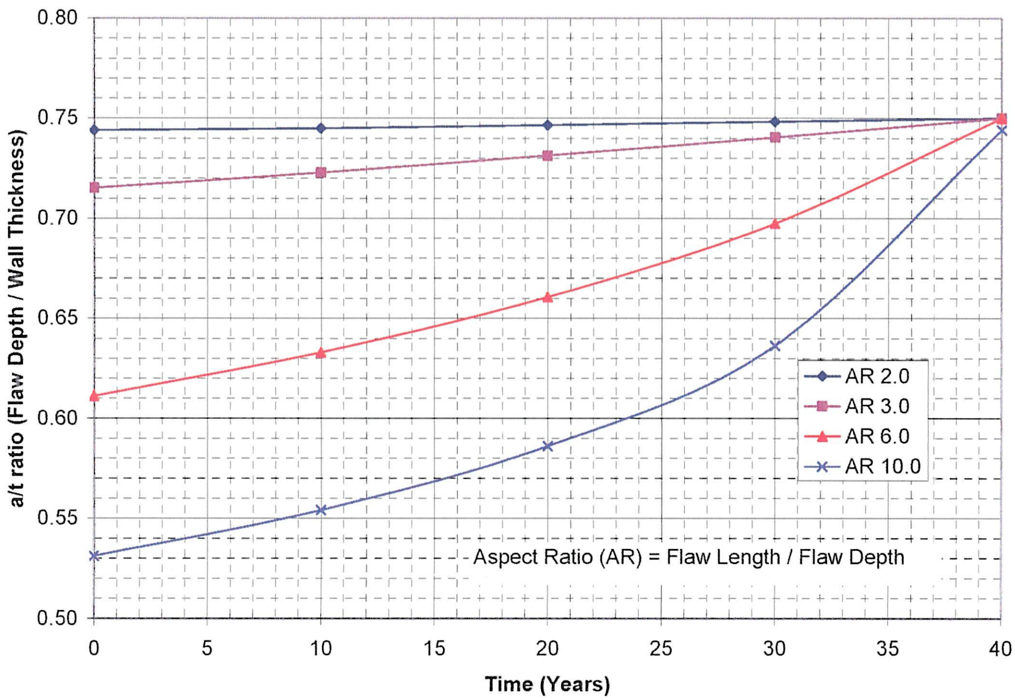


Figure 2-5 Fatigue Crack Growth Prediction for Repaired Axial Flaws in the Penetration Nozzles (Uphill side)

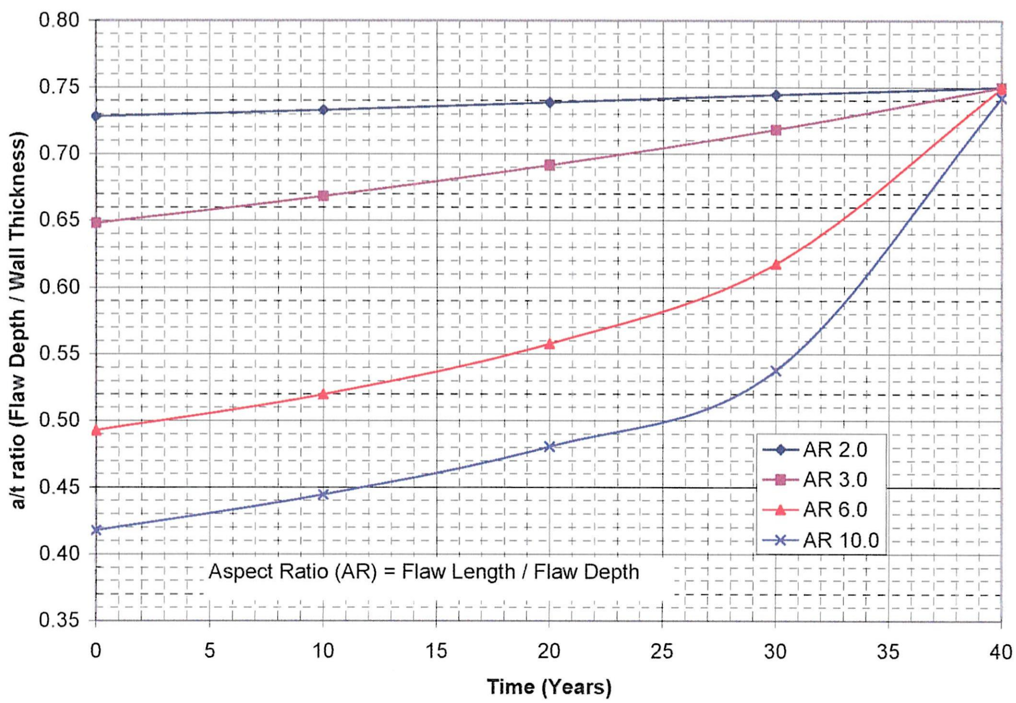


Figure 2-6 Fatigue Crack Growth Prediction for Repaired Axial Flaws in the Penetration Nozzles (Downhill side)

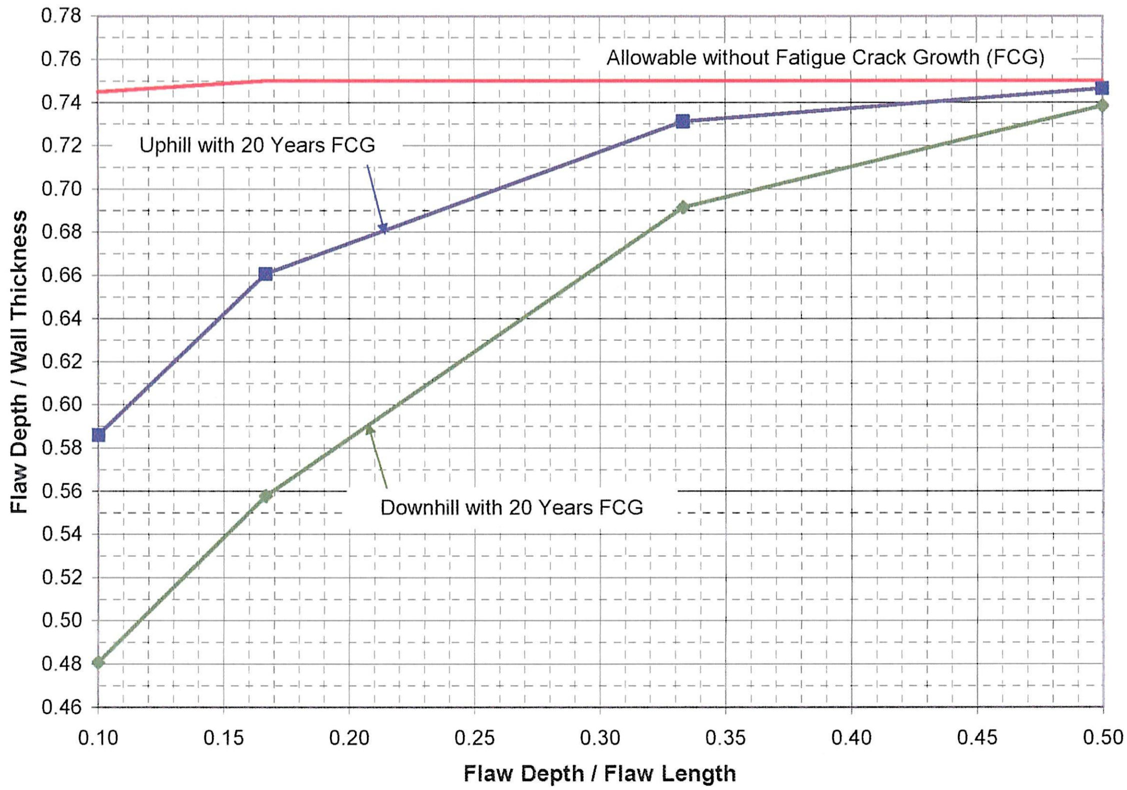


Figure 2-7 Maximum Allowable Axial Flaw Sizes in the Repaired Penetration Nozzles for 20 Years Service Life

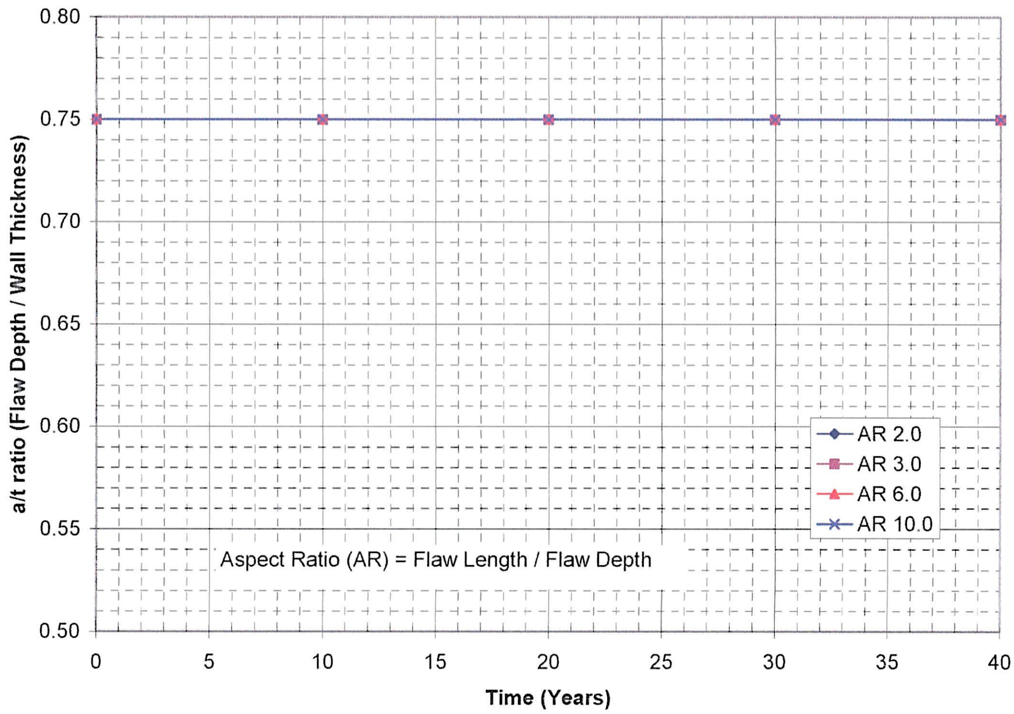


Figure 2-8 Fatigue Crack Growth Prediction for Repaired Circumferential Flaws in the Penetration Nozzles (Uphill side)

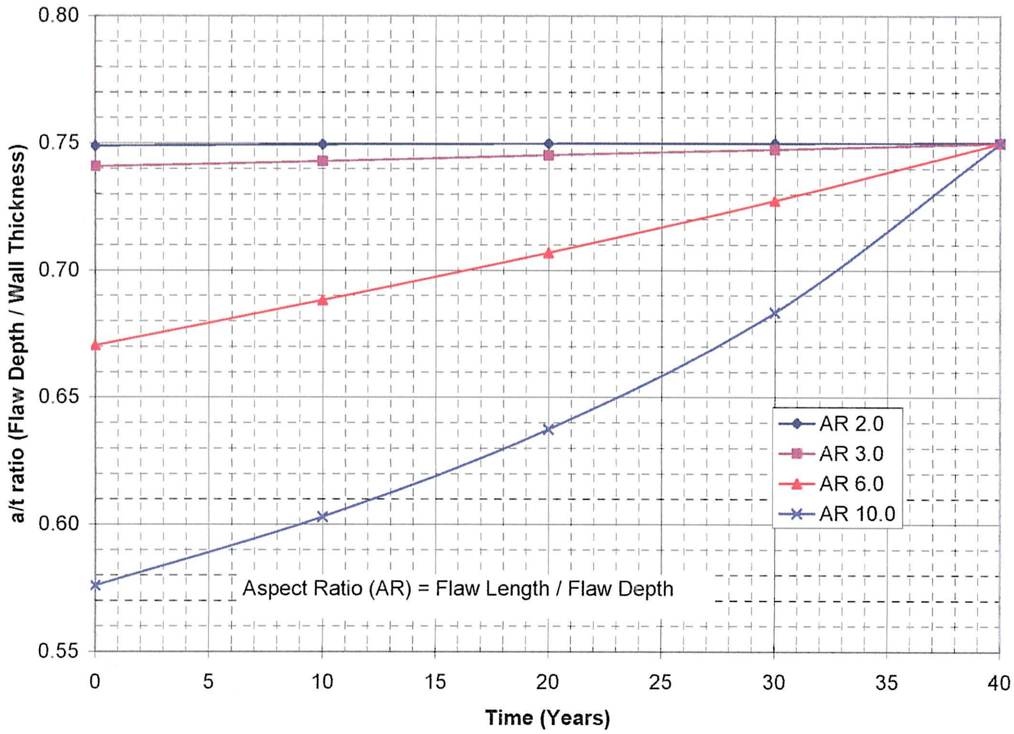


Figure 2-9 Fatigue Crack Growth Prediction for Repaired Circumferential Flaws in the Penetration Nozzles (Downhill side)

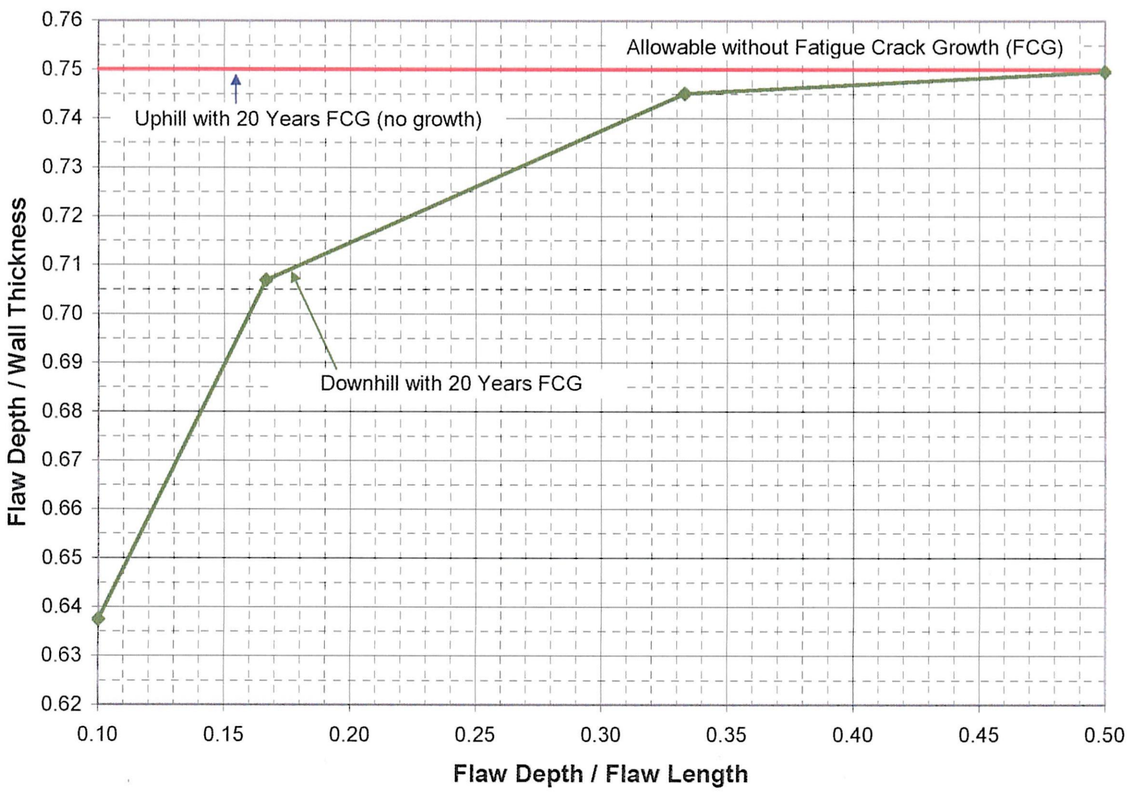


Figure 2-10 Maximum Allowable Circumferential Flaw Sizes in the Repaired Penetration Nozzles for 20 Years Service Life

3 TECHNICAL BASIS FOR APPLICATION OF EMBEDDED FLAW REPAIR METHOD TO PENETRATION NOZZLE ATTACHMENT WELDS

This section provides a discussion on the technical basis for the use of embedded flaw repair method for a flawed head penetration attachment weld. [

]a,c,e

A flaw evaluation was carried out by postulating a planar flaw of that size in the reactor vessel head. Based on the results of the evaluation, the largest flaw size that can be repaired using the embedded flaw repair method was determined.

3.1 Acceptance Criteria

3.1.1 Section XI Appendix K

The acceptance criteria and evaluation procedures used to demonstrate structural integrity of the reactor vessel closure head is contained in Appendix K of ASME Code Section XI [2]. Although the original purpose of Appendix K was to evaluate reactor vessels with low upper shelf fracture toughness, the methods are equally applicable to any region of the reactor vessel where the fracture toughness can be described with elastic plastic parameters. The head region of the reactor vessel is one of the hottest portion of the reactor vessel where the typical steady state temperature is approximately 550-620 °F. This ensures ductile behavior, and so the use of elastic-plastic methods is appropriate.

The approach to evaluating the integrity of a nuclear vessel has been developed over a ten-year period, and has been illustrated with a number of example problems [8] to demonstrate its use. The extension of this methodology to issues other than the low shelf fracture toughness issue is appropriate when service conditions (temperature) ensure ductile behavior. The extension of the Elastic Plastic Fracture Mechanics (EPFM) method to the reactor vessel head is appropriate, as discussed above.

The acceptance criteria are to be satisfied for each category of transients, namely, Service Load Level A (normal), Level B (upset), Level C (emergency) and Level D (faulted) conditions. The criteria are listed below:

$$J < J_{0.1}$$

$$\frac{\partial J}{\partial a} < \frac{dJ_R}{da}$$

J_R = J-integral resistance to ductile tearing for the material

J = Applied J-integral, with a safety factor of 1.15

$J_{0.1}$ = J-integral resistance at a ductile flaw extension of 0.1 inch

$\frac{\partial J}{\partial a}$ = Partial derivative of the applied J-integral with respect to flaw depth, a

$\frac{dJ_R}{da}$ = Slope of the J-R curve

3.1.2 Primary Stress Limits

In addition to satisfying the Section XI criteria, the primary stress limits of paragraph NB-3000 in Section III of the ASME Code must be satisfied. The effects of a local area reduction that is equivalent to the area of the postulated flaw in the vessel head attachment weld must be considered by increasing the membrane stresses to reflect the reduced cross section. The allowable flaw depth was determined by evaluating the primary stress of a spherical head with reduced wall thickness, using normal operating pressure of 2250 psia. The results show that the allowable flaw depth is bigger than all the attachment weld sizes.

3.2 METHODOLOGY

The evaluation assumed that a flaw has been detected in a penetration nozzle attachment weld and that the embedded flaw repair method is used to seal the flaw from further exposure to the primary water environment. The evaluation was performed to demonstrate that for a postulated flaw that encompassed the entire attachment weld region in the vessel head near the penetration nozzle, the flaw is stable under ductile crack growth based on the acceptance criteria described in Section 3.1. [ ]^{a,c,e} Therefore, fatigue crack growth evaluations for the postulated flaw in the reactor vessel head and the repair weld were performed to ensure structural integrity.

3.2.1 Geometry and Source of Data

There are many head penetrations in the reactor vessel upper head, and the highest stressed region of the vessel head is chosen for analysis. The distribution of residual, transient thermal, and pressure stresses in the closure head region is obtained from detailed three-dimensional elastic-plastic finite element analyses of the head penetration nozzle region [3] for Byron and Braidwood Units 1 and 2. The outermost penetration nozzle attachment weld regions were chosen for analysis because the stresses are in general highest there. The through-wall stress distributions from the finite element analyses were used to determine the fatigue crack growth and the resulting allowable flaw size for the postulated flaws in the attachment weld regions of the vessel head. The stress cuts on the reactor vessel head were selected for the analysis and the finite element model with the selected stress cuts is shown in Figure 2-2.

3.2.2 Loading Conditions

Thermal Transient Selection for Maximum Allowable Flaw Size Determination

The requirement for an evaluation of a flaw using the rules of Section XI is that the governing transients be chosen for the normal/upset conditions as well as the emergency/faulted conditions. [

]a,c,e

Thermal Transient Selection for Fatigue Crack Growth Prediction

[

]a,c,e The thermal transients that occur in the upper head region are relatively mild because most of the water in the head region has already passed through the core region. The flow in the upper head region is low compared to other regions of the reactor vessel, which mutes the effects of the operating thermal transients. The thermal transients that occur in Byron and Braidwood Units 1 and 2 are shown in Table 2-1. [

]a,c,e

3.2.3 Stress Intensity Factor

One of the key elements in a fracture mechanics evaluation is the determination of the crack driving force or stress intensity factor (K_I). This is based on the information available in the literature.

The stress intensity factors for two corner flaws emanating from the edge of a hole in a plate was taken from the data by [ ]a,c,e Use of this method requires that the stresses remote from the hole be resolved into membrane and bending stress components. The stress intensity factor can be expressed conservatively in terms of the membrane and bending stress components as follows:

[

]a,c,e This flexibility is necessary because this expression will be applied to a range of flaw shapes corresponding to different attachment

weld shapes in Byron and Braidwood Units 1 and 2. The coefficients A and B can be found in [9] for selected values of r/t, a/ℓ and a/t, where "r" is the outside radius of the penetration nozzle and "t" is the wall thickness of the reactor vessel head. For the r/t, a/ℓ and a/t values that are not shown in [9], the coefficients A and B were determined using interpolation. Since the coefficients are provided for various locations around the flaw front, [

]a,c,e

The stress intensity factors for the resulting embedded flaws due to the embedded flaw repair method were calculated based on the method given in Appendix A of Section XI. The sub-surface stress intensity factor expression can be applied to a crack approaching the surface of a component as stated in the technical basis [10]. The stress intensity factor can be expressed in terms of the equivalent membrane and bending stress components as follows:

$$K_I = \left(\sigma_m M_m + \sigma_b M_b\right)\sqrt{\pi a / Q}$$

where

σ_m, σ_b = Equivalent membrane and bending stresses, as defined in A-3200(a) of the Code [2]. (See Figure 3-4)

M_m, M_b = Correction factors for the membrane and bending stresses. The equations for the correction factors are listed in [10]

a = One-half the axis of elliptical flaw

Q = Flaw shape parameter as defined in [10]

3.2.4 Material Properties

One of the most important information on the toughness for pressure vessel and piping materials is the J-R curve of the material, where J-R stands for material resistance to crack extension, as represented by the measured J-integral value versus crack extension. Simply put, J-R curve to cracking resistance is as significant as the stress-strain curve to load-carrying capacity and ductility of a material. Both J-R curve and stress-strain curve are properties of a material.

Unfortunately, directly measured J-R curves are not generally available for a specific material of interest. Fortunately, methods that can generate such information from available data such as material chemistry, radiation exposure, temperature and Charpy V-notch energy, is now available [11]. The method provided in [11] summarizes a large collection of public test data, and fitted into multivariable model based on advanced pattern recognition technology. Separate analysis models and databases were developed for different material groups, including reactor pressure vessel (RPV) welds, RPV base metals, piping welds, piping base metals and a combined materials group.

The material resistance J-values, J_{mat}, are fitted into the following equation [11, 12]:

$$J_{mat} = (MF)C1\,(\Delta a)^{C2} \exp\left[C3\,(\Delta a)^{C4}\right]$$

where C1, C2, C3, and C4 are fitting constants, and Δa is crack extension

MF is the Margin Factor from [12]:
MF= 0.749 for Service Levels A, B and C
MF= 1.0 for Service Level D

For the RPV base metal model, the constants C1, C2, C3, and C4 are taken from Table 11 of [11]. C1, C2, C3, and C4 are complicated parameters as defined below:

$$\ln C1 = a_1 + a_2 \ln CVNp + a_3 T + a_4 \ln B_n + a_5 \phi t$$
$$C2 = d_1 + d_2 \ln C1 + d_3 \ln B_n$$
$$C3 = d_4 + d_5 \ln C1 + d_6 \ln B_n$$
$$C4 = d_7$$

where T = Temperature (°F),
B_n = Section thickness (inches).
CVNp = Charpy impact energy (ft-lbs) = 114 ft-lb from [13].
ϕt = Fluence ($x10^{18}$ n/cm^2, E>1Mev).

a_1, a_2, a_3, a_4, a_5, d_1, d_2, d_3, d_4, d_5, d_6, d_7 (briefly, a_i and d_i) are constants given in Table 11 of [11]:

a1 = -2.44
a2 = 1.13
a3 = -0.00277
a4 = 0.0801
a5 = 0.0
d1 = 0.0770
d2 = 0.116
d3 = -0.0412
d4 = -0.0812
d5 = -0.00920
d6 = -0.0295
d7 = -0.409

Neutron irradiation has been shown to produce embrittlement that reduces the toughness properties of reactor vessel ferritic steel material. The irradiation levels are very low in the reactor vessel head region and therefore the fracture toughness will not be measurably affected.

3.2.5 Applied J-Integral

For small scale yielding, $J_{applied}$ of a crack can be calculated by the Linear Elastic Fracture Mechanics (LEFM) method. A plastic zone correction must be performed to account for the plastic deformation at the crack tip. The plastic deformation ahead of the crack front is then regarded as a failed zone and the crack size is, in effect, increased. The K_I-values can be converted to $J_{applied}$ by the following equation:

$$J_{applied} = \frac{K_{ep}^2}{E'}$$

where K_{ep} is the elastically calculated K_I-value based on the plastic zone adjusted crack depth or size
$E' = E/(1-\nu^2)$ for plane strain, $E' = E$ for plane stress, E = Young's Modulus,
and ν = Poisson's Ratio.

The plastic zone size, r_p, is calculated by

$$r_p = \frac{1}{6\pi}\left(\frac{K_I}{S_y}\right)^2$$

where S_y is the yield strength of the material. Assume that the crack depth is a_o, the K_{ep} can now be calculated based on a new crack length, $a_o + r_p$. For small scale yielding, K_{ep} can be simplified as follows:

$$K_{ep} = f\ K_I$$

Where $f = \sqrt{\dfrac{(a_0 + r_p)}{a_0}}$

3.2.6 Fatigue Crack Growth Prediction

The analysis procedure involves postulating planar flaws that extend radially over the entire attachment weld cross-section in the vessel head and are subjected to a series of design loads. The loading included pressure, thermal transients, and residual stresses. The transients used for this evaluation are shown in Section 3.2.2 and the cycles are distributed evenly over the plant design life. The stress intensity factor range, ΔK_I, which controls the fatigue crack growth, depends on the geometry of the crack, its surrounding structure and the range of applied stresses in the region of the postulated crack. Once ΔK_I is calculated, the fatigue crack growth due to a particular stress cycle can be determined using a crack growth rate reference curve applicable to the material where the crack is postulated.

The crack growth rate (CGR) curves used in the analyses for the postulated flaws in the reactor vessel head are taken directly from Appendix A in the 2004 Edition of ASME Code Section XI for ferritic steels. Since the flaw is sealed from the primary water environment, the crack growth rate reference curve for the air environment is used. This curve is a function of the applied stress intensity factor range (ΔK_I) and the R ratio, which is the ratio of the minimum to maximum stress intensity factor during a thermal transient. The crack growth equation is given below:

$$\frac{da}{dN} = C_o\,(\Delta K)^{3.07}$$

where:

$\dfrac{da}{dN}$ = Crack growth rate, inches/cycle

ΔK_I = Stress intensity factor range, ksi$\sqrt{in}$
= $(K_{Imax} - K_{Imin})$

ΔK_{th} = ΔK_I threshold value, ksi$\sqrt{in}$
= $5.0\ (1 - 8.0\ R)$ for $0 \le R < 1.0$

$$C_o = 1.99 \times 10^{-10} S$$

$$S = 25.72\,(2.88 - R)^{-3.07} \text{ for } 0 \leq R < 1.0$$
$$R = K_{I\,min} / K_{I\,max}$$

The crack growth rate reference curve in air used in determining the fatigue crack growth into the repair weld Alloy 52 is not available. However, there are limited test data on Alloy 52 in PWR water environment and, based on these data, the crack growth rates for Alloy 52 and Alloy 600 were concluded to be the same in PWR water environment [6]. Therefore, the crack growth rate curve for Alloy 52 in air is assumed to be the same as that for Alloy 600 in air, which is shown in Equation 2-7 of Section 2.

Once the incremental crack growth corresponding to a specific transient for a small time period is calculated, it is added to the original crack size and the analysis continues to the next time period and/or thermal transient. The procedure is repeated in this manner until all the significant analytical thermal transients and cycles known to occur in a given period of operation have been analyzed.

3.3 FRACTURE MECHANICS ANALYSIS RESULTS

3.3.1 Results for Applied J-Integral and Material J-R Curves

The actual geometry or weld shapes of Byron and Braidwood Units 1 and 2 head penetration attachment welds [14] are shown in Table 3-1 and Figure 3-1, which forms the basis for the geometry of the postulated flaws in the attachment weld region. The stress intensity factors were calculated for the worst attachment weld shapes (lowest a/ℓ ratio), which are the downhill side welds for the outermost penetration No. 74-78. These attachment weld shapes with the lowest a/ℓ ratio were selected to bound all the other penetration nozzle attachment weld shapes in both Byron and Braidwood Units 1 and 2.

The applied J-integral values for the worst attachment weld shapes were calculated based on the method described in Section 3.2.5. The material J-R Curve was obtained as discussed in Section 3.2.4 by setting the Margin Factor (MF) to 0.749. The applied J-integral values and the material J-R Curve were tabulated in Table 3-2 and plotted in Figure 3-2. Using the acceptance criteria shown in Section 3.1, the structural integrity of the reactor vessel head with a postulated flaw that encompassed the entire attachment weld region can then be determined.

The key aspects of the structural integrity evaluation are the values of the applied J-integral versus that for the reactor vessel head material and the slope of the J-applied curve versus the slope of the material J-R curve. Figure 3-2 demonstrated the structural stability of a postulated flaw with the worst attachment weld shape. In Figure 3-2, it can be seen that for a crack extension of 0.1 inch with an initial flaw depth of 1.55 inch, the applied J-integral value is below that of the material J-R curve. In addition, the slope of the material J-R curve exceeds that of the J-applied curve. Since the acceptance criteria in Section 3.1 is met for the worst attachment weld shape, it can be concluded that structural stability is also demonstrated for all postulated flaws with the attachment weld shapes tabulated in Table 3-1.

3.3.2 Results for Fatigue Crack Growth into the Vessel Head

Fatigue crack growth was determined for postulated flaws in the reactor vessel head with attachment weld shapes on the uphill and downhill sides of the penetration nozzles that envelop all the other attachment weld shapes in both Byron and Braidwood Units 1 and 2. As shown in Figure 3-3, the predicted crack growth for the uphill side is slightly higher than that for the downhill side, but in general, the fatigue crack growth is quite small. Based on the fatigue crack growth results, stability of postulated flaws which encompassed the entire attachment region can be shown stable for at least 40 years of service life.

3.3.3 Results for Fatigue Crack Growth into the Repair Weld

[

]a,c,e The fatigue crack growth result indicates that the repaired weld can last at least 40 years of service life based on the assumed initial flaw depth.

3.4 SUMMARY

The results of the evaluation have demonstrated that the embedded flaw repair method is a viable method for repairing flaws found in the J-weld. The repair weld layer would last at least 40 years of service life regardless of the size of the flaw found in the penetration nozzle attachment weld. In addition, structural stability can also be demonstrated for at least 40 years of fatigue crack growth regardless of the size of the flaw found in the penetration nozzle attachment weld.

Table 3-1

Geometry of Byron and Braidwood Units 1 and 2 Head Penetration Attachment Welds (All dimensions in inches)

	Uphill			Downhill		
Pen. No.	ℓ	a	a/ℓ	ℓ	a	a/ℓ
1	1.61	1.86	1.155	1.61	1.86	1.155
2-5	1.56	1.98	1.269	1.73	1.75	1.012
6-9	1.56	2.03	1.301	1.81	1.70	0.939
10-13	1.56	2.05	1.314	1.85	1.68	0.908
14-17	1.58	2.11	1.335	1.98	1.62	0.818
18-21	1.59	2.13	1.340	2.02	1.61	0.797
22-29	1.59	2.15	1.352	2.07	1.59	0.768
30-37	1.62	2.21	1.364	2.22	1.55	0.698
38-41	1.66	2.26	1.361	2.40	1.50	0.625
42-49	1.67	2.28	1.365	2.46	1.49	0.606
50-53	1.69	2.30	1.361	2.53	1.47	0.581
54-61	1.72	2.34	1.360	2.68	1.44	0.537
62-65	1.80	2.44	1.356	3.45	1.65	0.478
66-73	1.83	2.46	1.344	3.60	1.63	0.453
74-78	1.89	2.54	1.344	4.11	1.55	0.377

Note : The values a (weld depth) and ℓ (weld length) are dimensions of the J-groove weld only and do not include the dimensions of the fillet weld

Table 3-2 Results of Applied J-integral and Material J-R Curve

a (inch)	J_{mat} (kip-in/in^2)	$J_{applied}$ (kip-in/in^2)
1.550	0.0000	1.2926
1.554	0.3061	1.2959
1.558	0.4933	1.2993
1.562	0.6252	1.3026
1.566	0.7281	1.3059
1.570	0.8131	1.3093
1.574	0.8857	1.3126
1.578	0.9492	1.3159
1.582	1.0058	1.3193
1.586	1.0569	1.3226
1.590	1.1036	1.3260
1.594	1.1465	1.3293
1.598	1.1863	1.3326
1.602	1.2234	1.3360
1.606	1.2581	1.3393
1.610	1.2909	1.3426
1.614	1.3218	1.3460
1.618	1.3511	1.3493
1.622	1.3790	1.3526
1.626	1.4057	1.3560
1.630	1.4311	1.3593
1.634	1.4555	1.3626
1.638	1.4789	1.3660
1.642	1.5014	1.3693
1.646	1.5231	1.3727
1.650	1.5440	1.3760
1.654	1.5642	1.3793
1.658	1.5837	1.3827
1.662	1.6027	1.3860
1.666	1.6210	1.3893
1.670	1.6388	1.3927
1.674	1.6561	1.3960
1.678	1.6730	1.3993
1.682	1.6894	1.4027
1.686	1.7053	1.4060
1.690	1.7209	1.4093
1.694	1.7361	1.4127
1.698	1.7509	1.4160
1.702	1.7653	1.4194
1.706	1.7795	1.4227

Note: Table 3-2 has been updated to revise the J-R calculation to improve accuracy; the requirements of Section 3.1.1 are met.

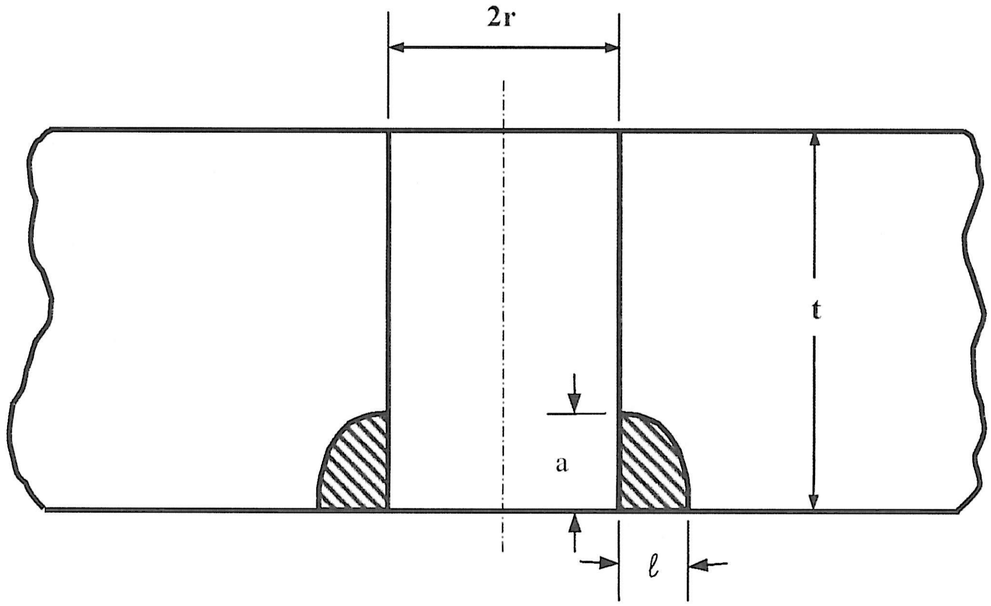
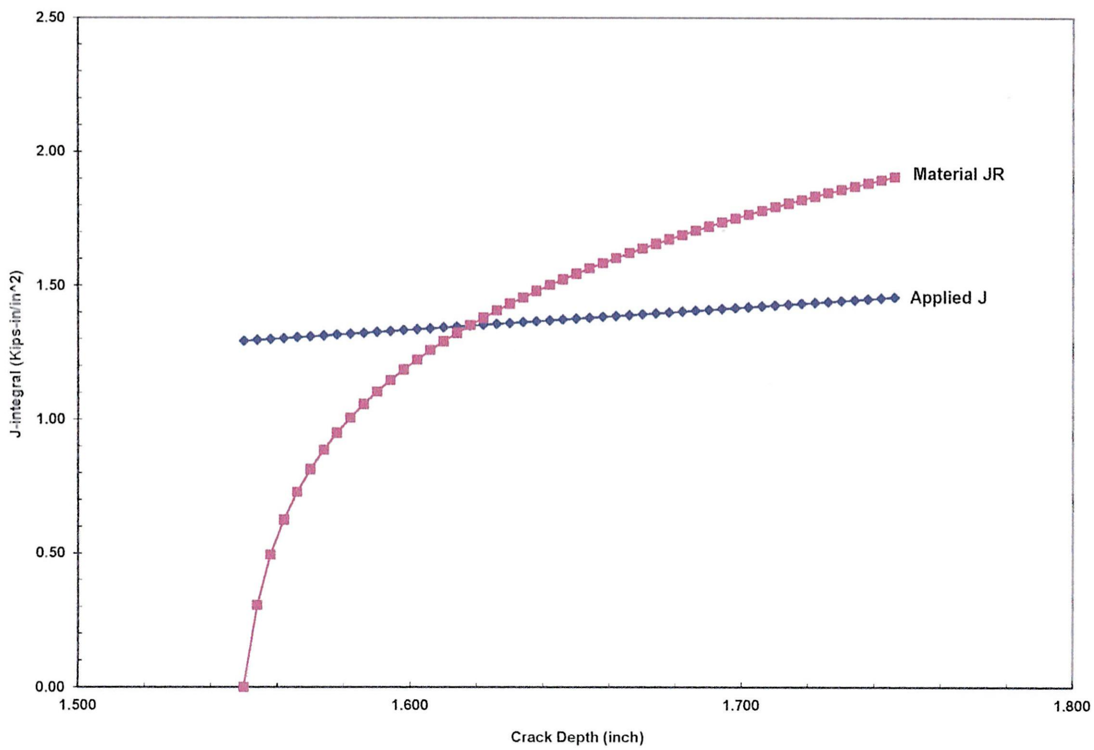


Figure 3-1 Geometry and Terminology as Applied in [ ]^{a,c,e}



Note: Figure 3-2 has been updated to revise the J-R calculation to improve accuracy; the requirements of Section 3.1.1 are met.

Figure 3-2 Comparison of the Slope of the Applied J-integral and Material J-R Curve

(Governing Transient: Loss of Load)

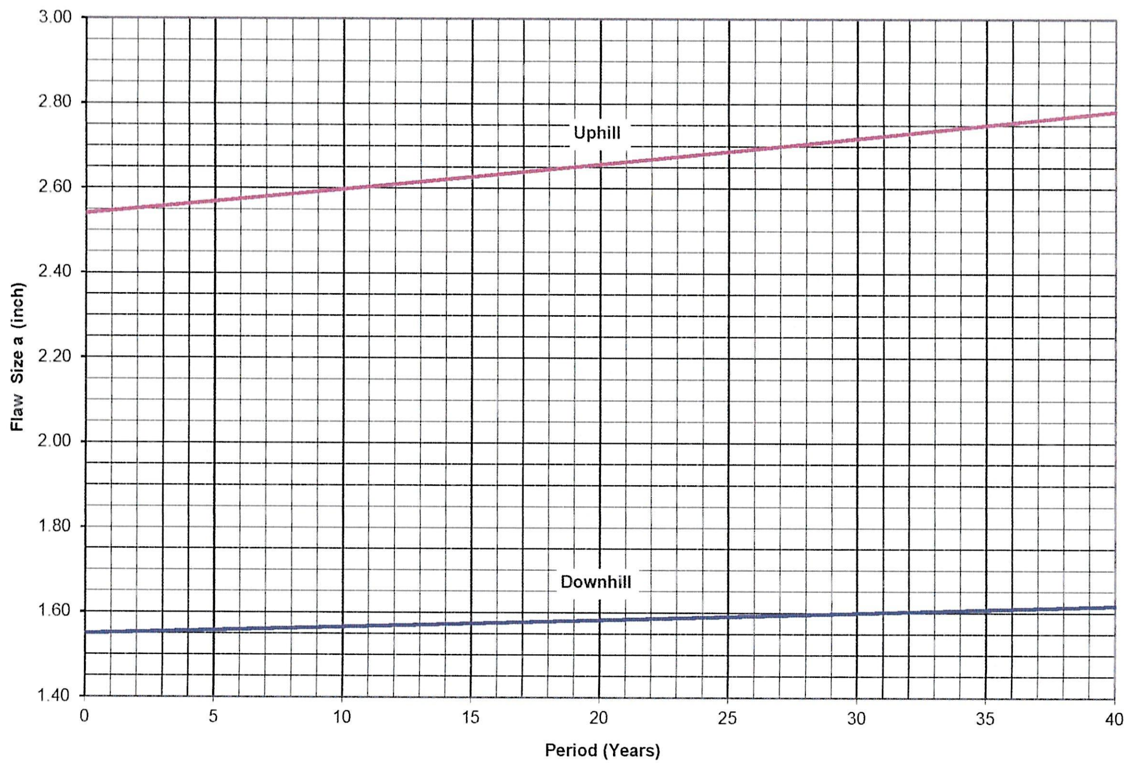


Figure 3-3 Fatigue Crack Growth Prediction in the Reactor Vessel Head with Maximum Postulated Flaws in the Attachment Weld

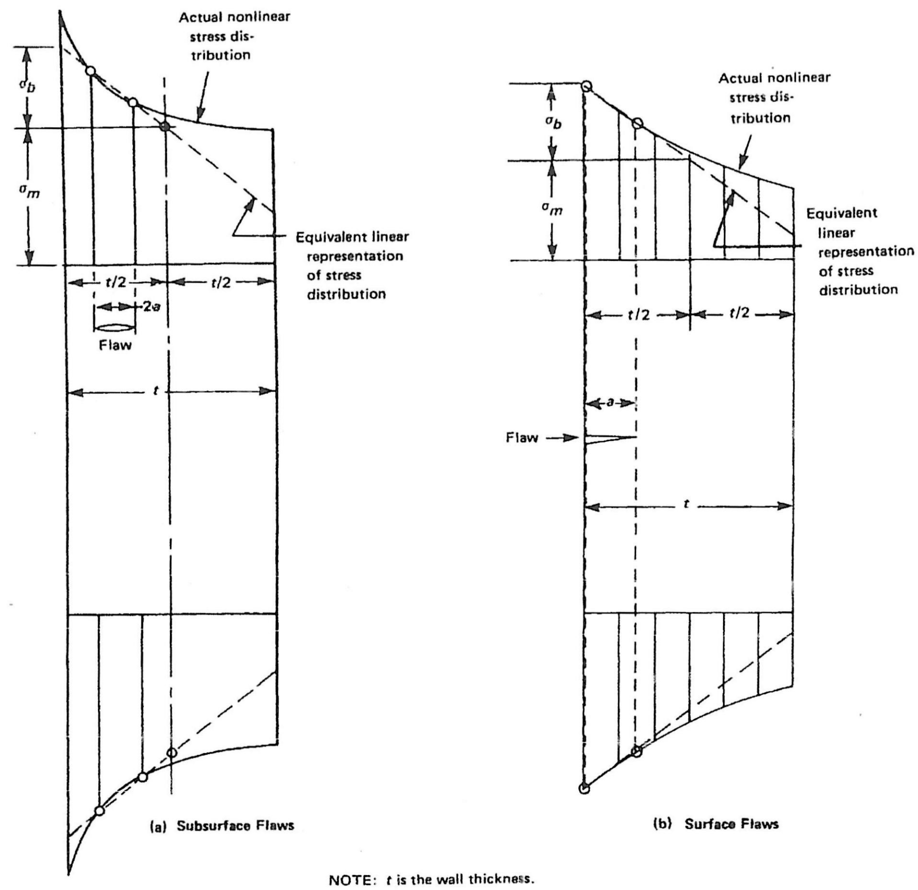


FIG. A-3200-1 LINEARIZED REPRESENTATION OF STRESSES

Figure 3-4 Linearized Representation of Through-Wall Stress Distribution

4 TECHNICAL BASIS FOR "AS-EXCAVATED" REPAIRS

4.1 Introduction

Cracks have been detected on the lower part of the reactor vessel closure head penetration nozzles in some operating plants, both foreign and domestic. Cracks have also been detected in the J-groove attachment welds. The root cause of the problem has been identified as Primary Water Stress Corrosion Cracking (PWSCC) of the Alloy 600 penetration material, or welds.

One of the repair options is to excavate the inside surface of the penetration nozzle on and around each individual crack location. Excavation will serve to remove the material that contains the crack from the penetration nozzle, thus removing the crack.

The purpose of this evaluation is to determine the maximum allowable excavation depth and geometry into the penetration nozzle inside surface, which will still meet the same ASME Code Section III stress allowable that were used in the reactor vessel stress report for Byron and Braidwood Units 1 and 2 [15]. In addition, cyclic stresses induced during normal and upset operating conditions are compared against the ASME code fatigue allowable criteria.

4.2 Technical Approach and Acceptance Criteria

A range of excavation sizes in the penetration nozzles was evaluated to determine the maximum excavation depth. The approach used here is to reconcile the stresses used in the original stress report to account for various excavation depths, and to determine the maximum excavation depth which would still meet the ASME Code Section III requirements used in the Byron and Braidwood Units 1 and 2 reactor vessel stress report [15]. The key dimensions of the head penetration nozzles are shown in Table 4-1. Other key dimensions include the minimum vessel head thickness, which is 6.625 inches, and the inside radius of the spherical head, which is 88.156 inches. The vessel heads and penetration nozzles for both Byron and Braidwood units have the same sizes.

The results presented in this section are based on the original reactor vessel stress qualification report for both Byron and Braidwood Units 1 and 2 [15]. The report provides the technical basis for the original design compliance with the Section III Code requirements. The effects of excavation in the regions of interest will be reflected in higher membrane stresses, and the limiting excavation depths will be those which meet the code stress limits.

The design loads include mechanical, thermal, and other external loads. The effect of other external loads in this region is negligible, because the moments are taken out at the location where the penetration nozzle exits the vessel head. Thus, the effects of the seismic and pipe break loads would be negligible at the location of interest.

The ASME Section III acceptance criteria used in the Byron and Braidwood Units 1 and 2 reactor vessel stress report are shown in Table 4-2. In addition, the local primary stress plus average primary stress was also calculated and shown to satisfy the Section III Code requirements. The loads and material properties used are the same as those used in the original reactor vessel stress report.

Results are provided for the following locations of possible excavation:

1. Penetration nozzles at and above the attachment weld
2. Penetration nozzles below the attachment weld

4.3 Evaluation Results for the "As-Excavated" Repairs

At and Above the Attachment Weld Region

For the penetration nozzle excavation analysis, a 360-degree grind-out was conservatively considered, resulting in a simple thickness reduction in the nozzle. There was no limitation in length. Reducing the nozzle thickness increases the primary stress, and has a small effect on the secondary stress. It was assumed that any grinding performed would have a 3:1 taper or greater in order to minimize any stress concentration effects and any sharp corners or notches resulting from the grinding operations should be eliminated.

For the penetration nozzle, the required minimum wall thickness of the nozzle was calculated to be 0.375 inch. So for a nominal wall thickness of 0.625 inch, the allowable depth of grinding is 0.250 inch. This amounts to a grinding depth of 40 percent of the penetration nozzle nominal wall thickness.

Below the Attachment Weld Region

The region of the penetration nozzles below the attachment weld is not part of the pressure boundary, so there are few restrictions on grinding in this region. Also there are no net pressure loads here, so there are no restrictions on the minimum wall thickness. In an extreme case, the entire penetration nozzle below the weld could be removed. If grinding is performed in this region, care should be taken to ensure that no sharp corners or notches are created and that the potential for loose part is minimized. The slope of the grinding should also maintain a 3:1 taper or greater.

4.4 SUMMARY

Allowable excavation depths have been determined for Byron and Braidwood Units 1 and 2 reactor vessel head penetration nozzles. The approach used here is to reconcile the stresses used in the original stress report to account for various excavation depths, and to determine the maximum excavation depth which would meet the ASME Code Section III requirements.

For excavation on penetration nozzles at or above the attachment region, grinding can be justified to a depth of 40 percent of the nominal wall thickness with a minimum required wall thickness of 0.375 inch. For excavation on penetration nozzles below the attachment weld region, there are no excavation depth limits, as this region is within the pressure boundary. In either case, grinding should have a 3:1 taper or greater in order to minimize any stress concentration effects. Sharp corners or notches resulting from the grinding operations should be eliminated.

Table 4-1 Key Dimensions of Head Penetration Nozzles [14]	
Inside Diameter (in.)	**Outside Diameter (in.)**
2.75	4.00

Table 4-2 Results of Structural Qualification for "As-Excavated" Repairs (Housing Penetration Wall Thickness Reduced to 0.375 inch)

Category	Stress Intensity (ksi)	Allowable (ksi)	Fatigue Usage Factor
P_L+P_b+Q[a]	< 69.9	$3.0\ S_m = 69.9$	< 1.0 [b]
P_m	20.5	$1.5\ S_m = 35.0$	---

[a] Criterion from [15]

[b] Allowable Fatigue Usage Factor is 1.0

5 SUMMARY AND CONCLUSIONS

Engineering evaluations were performed to support the repair efforts associated with the reactor vessel head penetration inspection program for Byron and Braidwood Units 1 and 2.

The technical basis for the use of the embedded flaw repair method if indications or flaws are found in the head penetration nozzle is provided in Section 2. Axial and circumferential flaws found on the inside surface or outside surface of a head penetration nozzle can be repaired using the embedded flaw repair method to seal the flaws from the primary water environment. The maximum allowable axial and circumferential flaw sizes detected in the penetration nozzles that can be repaired using the embedded flaw repair method are shown in Figures 2-7 and 2-10 with the effects of fatigue crack growth included.

The technical basis for the use of the embedded flaw repair method if indications or flaws are found in the head penetration attachment welds is provided in Section 3. The results of the evaluation have demonstrated that the embedded flaw repair method is a viable method for repairing flaws found in the J-weld. The repair weld layer would last at least 40 years of service life regardless of the size of the flaw found in the penetration nozzle attachment weld. In addition, structural stability can also be demonstrated for at least 40 years of fatigue crack growth regardless of the size of the flaw found in the penetration nozzle attachment weld.

The evaluations which address the effects of local structural discontinuities resulting from the grinding operations performed to excavate flaws in the head penetration nozzle are provided in Section 4. For excavation on penetration nozzles at or above the attachment weld region, grinding can be justified to a depth of 40 percent of the nominal wall thickness with a required minimum wall thickness of 0.375 inch. For excavation on penetration nozzles below the attachment weld region, there are no excavation depth limits, as this region is within the pressure boundary. In either case, grinding should have a 3:1 taper or greater in order to minimize any stress concentration effects. Sharp corners or notches resulting from the grinding operations should be eliminated.

Plant specific technical basis for the use of the embedded flaw repair method is presented in this report and it is concluded that Byron and Braidwood Units 1 and Unit 2 meet the criteria for application of the embedded flaw repair process stated in Appendix C of WCAP-15987-P [1].

6 REFERENCES

1. [ ][a,c,e]

2. ASME Code Section XI, "Rules for Inservice Inspection of Nuclear Plant Components," 2004 Edition.

3. Fleming, M., et al. "CRDM Nozzle Transient Stress Analysis: Byron & Braidwood Reactor Vessel Heads," Dominion Engineering, Inc. Report R-8705-00-1, Revision 0, December 2004.

4. Brown, C. M., and Mills, W. J., "Fracture Toughness, Tensile and Stress Corrosion Cracking Properties of Alloy 600, Alloy 690, and Their Welds in Water," in Proceedings of Corrosion 96, Paper 90.

5. [ ][a,c,e]

6. NUREG/CR-6721, ANL-01/07, "Effects of Alloy Chemistry, Cold Work, and Water Chemistry on Corrosion Fatigue and Stress Corrosion Cracking of Nickel Alloys and Welds," April 2001.

7. [ ][a,c,e]

8. "Development of Criteria for Assessment of Reactor Vessels with Low Upper Shelf Fracture Toughness," Welding Research Council Bulletin 413, July 1996.

9. [ ][a,c,e]

10. Marston, T. U. et al., "Flaw Evaluation Procedures: ASME Section XI," Electric Power Research Institute Report EPRI-NP-719-SR, August 1978.

11. E. D. Eason, J. E. Wright, E. E. Nelson, "Multivariable Modeling of Pressure Vessel and Piping J-R Data," NUREG/CR-5729, MCS 910401, RF, R5, May 1991.

12. Regulatory Guide 1.161, "Evaluation of Reactor Pressure Vessel with Charpy Upper-Shelf Energy Less Than 50 ft-lb."

13. Reference Documents for Charpy Impact Energy for Byron and Braidwood Units 1 and 2.

    a) [ ][a,c,e]

    b) [ ][a,c,e]

c) [

]a,c,e

d) [

]a,c,e

14. Reference Drawings for Byron and Braidwood Units 1 and 2:

a) B&W Drawing No. 184573E, Revision 4, "Closure Head Assembly". (Byron 1)
b) B&W Drawing No. 184574E, Revision 6, "Closure Head Sub-Assy Sheet#1". (Byron 1)
c) B&W Drawing No. 184577E, Revision 2, "Control Rod Mech. Housing". (Byron 1)
d) B&W Drawing No. 185282E, Revision 0, "Closure Head Assembly". (Byron 2)
e) B&W Drawing No .185283E, Revision 1, "Closure Head Sub-Assy Sheet#1". (Byron 2)
f) B&W Drawing No. 185286E, Revision 1, "Control Rod Mech. Housing". (Byron 2)
g) B&W Drawing No. 185313E, Revision 0, "Closure Head Assembly". (Braidwood 1)
h) B&W Drawing No. 185314E, Revision 0, "Closure Head Sub-Assy Sheet#1". (Braidwood 1)
i) B&W Drawing No. 185317E, Revision 0, "Control Rod Mech. Housing". (Braidwood 1)
j) B&W Drawing No. 185344E, Revision 0, "Closure Head Assembly". (Braidwood 2)
k) B&W Drawing No. 185345E, Revision 0, "Closure Head Sub-Assy Sheet # 1". (Braidwood 2)
l) B&W Drawing No. 185348E, Revision 0, "Control Rod Mech. Housing". (Braidwood 2)

15. [

]a,c,e

ATTACHMENT 3

Application for Withholding Proprietary Information from Public Disclosure (for Attachment 1)

Westinghouse

Westinghouse Electric Company
1000 Westinghouse Drive
Cranberry Township, Pennsylvania 16066
USA

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Direct tel: (412) 374-4643
Direct fax: (724) 940-8560
e-mail: greshaja@westinghouse.com

CAW-17-4539

January 31, 2017

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: WCAP-16401-P, Revision 1, "Technical Basis for Repair Options for Reactor Vessel Head Penetration Nozzles and Attachment Welds: Byron and Braidwood Units 1 and 2" (Proprietary)

The Application for Withholding Proprietary Information from Public Disclosure is submitted by Westinghouse Electric Company LLC ("Westinghouse"), pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Nuclear Regulatory Commission's ("Commission's") regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-17-4539 signed by the owner of the proprietary information, Westinghouse. The Affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying Affidavit by Exelon.

Correspondence with respect to the proprietary aspects of the Application for Withholding or the Westinghouse Affidavit should reference CAW-17-4539 and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 3 Suite 310, Cranberry Township, Pennsylvania 16066.

J A Gresham

James A. Gresham, Manager
Regulatory Compliance

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

I, James A. Gresham, am authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC ("Westinghouse") and declare that the averments of fact set forth in this Affidavit are true and correct to the best of my knowledge, information, and belief.

Executed on: 1/31/17

J A Gresham

James A. Gresham, Manager
Regulatory Compliance

(1) I am Manager, Regulatory Compliance, Westinghouse Electric Company LLC ("Westinghouse"), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Nuclear Regulatory Commission's ("Commission's") regulations and in conjunction with the Westinghouse Application for Withholding Proprietary Information from Public Disclosure accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitute Westinghouse policy and provide the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(iii) There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iv) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, is to be received in confidence by the Commission.

(v) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(vi) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in WCAP-16401-P, Revision 1, "Technical Basis for Repair Options for Reactor Vessel Head Penetration Nozzles and Attachment Welds: Byron and Braidwood Units 1 and 2" (Proprietary), dated January 2017, for submittal to the Commission, being transmitted by Exelon letter. The proprietary information as submitted by Westinghouse is that associated with Westinghouse's request for NRC approval of WCAP-16401, and may be used only for that purpose.

(a) This information is part of that which will enable Westinghouse to provide technical justification to support the repair options for the reactor vessel head penetration nozzles and attachment welds at Byron and Braidwood Units 1 and 2.

(b) Further, this information has substantial commercial value as follows:

(i) Westinghouse plans to sell the use of similar information to its customers for the purpose of providing technical justification to support the repair options for the reactor vessel head penetration nozzles and attachment welds.

(ii) Westinghouse can sell support and defense of industry guidelines and acceptance criteria for plant-specific applications.

(iii) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and non-proprietary versions of a document, furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the Affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

COPYRIGHT NOTICE

Letter for Transmittal to the NRC

The following paragraphs should be included in your letter to the NRC Document Control Desk:

Enclosed are:

1. WCAP-16401-P, "Technical Basis for Repair Options for Reactor Vessel Head Penetration Nozzles and Attachment Welds: Byron and Braidwood Units 1 and 2" (Proprietary)

2. WCAP-16401-NP, "Technical Basis for Repair Options for Reactor Vessel Head Penetration Nozzles and Attachment Welds: Byron and Braidwood Units 1 and 2" (Non-Proprietary)

Also enclosed are the Westinghouse Application for Withholding Proprietary Information from Public Disclosure CAW-17-4539, accompanying Affidavit, Proprietary Information Notice, and Copyright Notice.

As Item 1 contains information proprietary to Westinghouse Electric Company LLC ("Westinghouse"), it is supported by an Affidavit signed by Westinghouse, the owner of the information. The Affidavit sets forth the basis on which the information may be withheld from public disclosure by the Nuclear Regulatory Commission ("Commission") and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.390 of the Commission's regulations.

Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse Affidavit should reference CAW-17-4539 and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 3 Suite 310, Cranberry Township, Pennsylvania 16066.

ATTACHMENT 4

10 CFR 50.55a Relief Requests I4R-10, Revision 2, Alternative Requirements for the Repair of Reactor Vessel Head Penetrations In Accordance with 10 CFR 50.55a(z)(1)

10 CFR 50.55a RELIEF REQUEST I4R-10
Revision 2

Request for Relief
Alternative Requirements for the Repair of Reactor Vessel Head Penetrations
In Accordance with 10 CFR 50.55a(z)(1)

1.0 ASME CODE COMPONENT(S) AFFECTED

Component Numbers	Byron Station, Units 1 and 2, Reactor Vessels 1RC01R (Unit 1) and 2RC01R (Unit 2)
Description:	Alternative Requirements for the Repair of Reactor Vessel Head Penetrations (VHPs) and J-groove Welds
Code Class:	Class 1
Examination Category:	ASME Code Case N-729-1
Code Item:	B4.20
Identification:	Byron Units 1 and 2, VHP Numbers 1 through 78, (P-1 through P-78) Previous repairs (I3R-14): Unit 2, P-68[1] (I3R-19): Unit 1, P-31, P-43, P-64, and P-76[1] (I3R-20): Unit 2, P-6[1]
Drawing Numbers:	Various

2.0 APPLICABLE CODE EDITION AND ADDENDA

Inservice Inspection and Repair/Replacement Programs: American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, 2007 Edition, with the 2008 Addenda. Examinations of the VHPs 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.

Code of Construction [Reactor Pressure Vessel (RPV)]: ASME Section III, 1971 Edition through Summer 1973 Addenda.

[1] This relief request includes Inservice Inspection (ISI) examination requirements for repairs previously completed in accordance with I3R-14, I3R-19, and I3R-20 in the previous inspection interval.

10 CFR 50.55a RELIEF REQUEST I4R-10
Revision 2

3.0 APPLICABLE CODE REQUIREMENT

IWA-4000 of ASME Section XI contains requirements for the removal of defects from and welded repairs performed on ASME components. The specific Code requirements for which use of the proposed alternative is being requested are as follows:

ASME Section XI, IWA-4421 states:

Defects shall be removed or mitigated in accordance with the following requirements:

- *(a) Defect removal by mechanical processing shall be in accordance with IWA-4462.*
- *(b) Defect removal by thermal methods shall be in accordance with IWA-4461.*
- *(c) Defect removal or mitigation by welding or brazing shall be in accordance with IWA-4411.*
- *(d) Defect removal or mitigation by modification shall be in accordance with IWA-4340.*

Note that use of the "Mitigation of Defects by Modification" provisions of IWA-4340 is prohibited per 10 CFR 50.55a(b)(2)(xxv).

For the removal or mitigation of defects by welding, ASME Section XI, IWA-4411 states, in part, the following.

> *Welding, brazing, fabrication, and installation shall be performed in accordance with the Owner's Requirements and ... in accordance with the Construction Code of the item...*

The applicable requirements of the Construction Code required by IWA-4411 for the removal or mitigation of defects by welding from which relief is requested are as follows.

Base Material Defect Repairs:

For defects in base material, ASME Section III, NB-4131 requires that the defects are eliminated, repaired, and examined in accordance with the requirements of NB-2500. These requirements include the removal of defects via grinding or machining per NB-2538. Defect removal must be verified by a Magnetic Particle (MT) or Liquid Penetrant (PT) examination in accordance with NB-2545 or NB-2546, and if necessary to satisfy the design thickness requirement of NB-3000, repair welding in accordance with NB-2539.

ASME Section III, NB-2539.1 addresses removal of defects and requires defects to be removed or reduced to an acceptable size by suitable mechanical or thermal methods.

ASME Section III, NB-2539.4 provides the rules for examination of the base material repair welds and specifies they shall be examined by the MT or PT methods in accordance with NB-2545 or NB-2546. Additionally, if the depth of the repair cavity exceeds the lesser of 3/8-inch or 10% of the section thickness, the repair weld shall be examined by the radiographic method in accordance with NB-5110 using the acceptance standards of NB-5320.

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Revision 2

Weld Metal Defect Repairs (This applies to the CRDM penetration J-Groove weld.)

ASME Section III, NB-4450 addresses repair of weld metal defects.

ASME Section III, NB-4451 states; that unacceptable defects in weld metal shall be eliminated and, when necessary, repaired in accordance with NB-4452 and NB-4453.

ASME Section III, NB-4452 addresses elimination of weld metal surface defects without subsequent welding and specifies defects are to be removed by grinding or machining.

ASME Section III, NB-4453.1 addresses removal of defects in welds by mechanical means or thermal gouging processes and requires the defect removal to be verified with MT or PT examinations in accordance with NB-5340 or NB-5350 and weld repairing the excavated cavity. In the case of partial penetration welds where the entire thickness of the weld is removed, only a visual examination is required to determine suitability for re-welding.

As an alternative to the requirements above, repairs will be conducted in accordance with the appropriate edition/addenda of ASME Section III and the alternative requirements, based on WCAP-15987-P, Revision 2-P-A, "Technical Basis for the Embedded Flaw Process for Repair of Reactor Vessel Head Penetrations," December 2003, (Refer to Reference 1, hereafter known as WCAP-15987-P).

4.0 REASON FOR THE REQUEST

Exelon Generation Company, LLC (EGC) will conduct examinations of the reactor Vessel Head Penetrations (VHPs) in accordance with Code Case N-729-1, as amended by 10 CFR 50.55a. Flaw indications that require repair may be found on the VHP tube material and/or the J-groove attachment weld(s) on the underside of the reactor vessel head.

Relief is requested from the requirements of ASME Section XI, IWA-4411 to perform permanent repair of future defects that may be identified on the VHP's and/or J-groove attachment weld(s) in accordance with the rules of the ASME Section III Construction Code as described in this relief request.

Specifically, relief is requested from:

- The requirements of ASME Section III, NB-4131, NB-2538, and NB-2539 to eliminate and repair defects in materials.
- The requirements of ASME Section III, NB-4450 to repair defects in weld metal.

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Revision 2

5.0 PROPOSED ALTERNATIVE AND BASIS FOR USE

5.1 Proposed Alternative

EGC proposes to use the less intrusive embedded flaw process (Reference 1) for the repair of VHP(s) as approved by the NRC (Reference 2) as an alternative to the defect removal requirements of ASME Section XI and Section III.

5.1.1 The criteria for flaw evaluation established in 10 CFR 50.55a(g)(6)(ii)(D), which specifies the use of Code Case N-729-1, will be used in lieu of the "Flaw Evaluation Guidelines" specified by the NRC Safety Evaluation for WCAP-15987-P (Refer to Reference 5).

5.1.2 Consistent with WCAP-15987-P, Revision 2-P-A methodology, the following repair requirements will be performed.

1. Inside Diameter (ID) VHP Repair Methodology

   a. An unacceptable axial flaw will be first excavated (or partially excavated) to a maximum depth of 0.125 inches. Although this depth differs from that specified in WCAP-15987-P, the cavity depth is not a critical parameter in the implementation of a repair on the ID surface of the VHP. The goal is to isolate the susceptible material from the primary water (PW) environment. The purpose of the excavation is to accommodate the application of at least two (2) weld layers of Alloy 52 or 52M, which is resistant to Primary Water Stress Corrosion Cracking (PWSCC), to meet that requirement. The depth specified in WCAP-15987-P is a nominal dimension and the depth needed to accommodate three weld layers while still maintaining the tube ID dimension. Since two (2) weld layers will be applied, less excavation is required and only 0.125 inches of excavation is necessary. The shallower excavated cavity for 2 weld layers would mean a slightly thinner weld, which would produce less residual stress.

      The excavation will be performed using an Electrical Discharge Machining (EDM) process to minimize VHP tube distortion. After the excavation is complete, either an ultrasonic test (UT) or surface examination will be performed to ensure that the entire flaw length is captured. Then a minimum of 2 layers of Alloy 52 or 52M weld material will be applied to fill the excavation. The expected chemistry of the weld surface is that of typical Alloy 52 or 52M weldment with no significant dilution. The finished weld will be conditioned to restore the inside diameter and then examined by UT and surface examination to ensure acceptability.

   b. If required, unacceptable ID circumferential flaw will be either repaired in accordance with existing code requirements; or will be partially excavated to reduce the flaw to an acceptable size, examined by UT or surface examination, inlaid with Alloy 52 or 52M, and examined by UT and surface examination as described above.

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Revision 2

2. Outside Diameter (OD) VHP and J-groove Weld Repair Methodology

   a. An unacceptable axial or circumferential flaw in a tube below a J-groove attachment weld will be sealed off with an Alloy 52 or 52M weldment. Excavation or partial excavation of such flaws is not necessary. The embedded flaw repair technique may be applied to OD axial or circumferential cracks below the J-groove weld because they are located away from the pressure boundary, and the proposed repair of sealing the crack with Alloy 690 weld material would isolate the crack from the environment as stated in Section 3.6.1 of the NRC Safety Evaluation for WCAP-15987-P.

   b. Unacceptable radial flaws in the J-groove attachment weld will be sealed off with a 360 degree seal weld of Alloy 52 or 52M covering the entire weld. Excavation or partial excavation of such flaws is not necessary.

   c. If EGC determines an excavation is desired (e.g., boat sample), then

      - The excavation will be filled with Alloy 52 or 52M material.

      - It is expected that a portion of the indication may remain after the boat sample excavation; however, a surface examination will be performed on the excavation to assess the pre-repair condition.

      - Depending on the extent and/or location of the excavation, the repair procedure requires the Alloy 52 or 52M weld material to extend at least one half inch outboard of the Alloy 82/182 to stainless steel clad interface.

   d. Unacceptable axial flaws in the VHP tube extending into the J-groove weld will be sealed with Alloy 52 or 52M as discussed in Item 5.1.2.2.a above. In addition, the entire J-groove weld will be sealed with Alloy 52 or 52M to embed the axial flaw. The seal weld will extend onto and encompass the portion of the flaw on the outside diameter of the VHP tube.

   e. For seal welds performed on the J-groove weld, the interface boundary between the J-groove weld and stainless steel cladding will be located to positively identify the weld clad interface to ensure that all of the Alloy 82/182 material of the J-groove weld is seal welded during the repair.

   f. The seal weld that will be used to repair an OD flaw in the nozzles and the J-groove weld will conform to the following.

      - Prior to the application of the Alloy 52 or 52M seal weld repair on the RPV clad surface, at least three beads (one layer) of ER309L stainless steel buffer will be installed 360° around the interface of the clad and the J-groove weld metal.

      - The J-groove weld will be completely covered by at least three (3) layers of Alloy 52 or 52M deposited 360° around the nozzle and over the ER309L stainless steel buffer. Additionally, the seal weld will extend onto and

10 CFR 50.55a RELIEF REQUEST I4R-10
Revision 2

encompass the outside diameter of the penetration tube Alloy-600 material by at least one half inch.

- The VHP tube will have at least two (2) layers of Alloy 52 or 52M deposited over the flaw on the VHP tube, extending out at least one half inch beyond the flaw, or to the maximum extent allowed by the nozzle geometry (e.g., limited length of the VHP tube).

g. Nondestructive examinations of the finished seal weld repair (i.e., Repair NDE) and during subsequent outages (i.e., ISI NDE) are summarized in the table below.

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 embedded flaw repair (EFR) shall be performed to ensure the repair satisfies ASME Section III, NB-5350 acceptance standards. The frequency of examination shall be as follows:

a. Perform surface examination during the first and second refueling outage after installation or repair of the EFR.

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Revision 2

b. When the examination results in 4.a above verify acceptable results then re-inspection of the EFR will be continued at a frequency of every other refueling outage. If these examinations identify unacceptable results that require flaw removal, flaw reduction to acceptable dimensions or welded repair the requirements of 4.a above shall be applied during the next refueling outage.

5.1.3 J-Groove Weld ISI NDE Requirements

Note 4 permits a reinspection frequency of every other cycle when the surface examination results of the EFR are verified to be acceptable for two consecutive cycles after the original installation or repair of the EFR. Westinghouse Report LTR-PSDR-TAM-14-005, Revision 3 (Reference 9) provides the technical bases for reducing surface examination requirements for J-groove weld repairs. This technical justification includes a detailed review of PT examination history, review of potential causes of PT indications in EFRs, and the use of crack resistant alloys in the EFR. The EFR is a robust design that is resistant to PWSCC. EFR installation, examination, and operational history indicate that the EFR performs acceptably. Examination and removed sample history indicate that the flaws identified shortly after installation of EFR weld material were due to embedded weld discontinuities and not due to service induced degradation. With inspection of the EFR every other cycle of operation, the nozzles are adequately monitored for degradation by ultrasonic examination methods similar to the nozzles without EFR repairs.

EGC projects that the reduction of the PT examination of nozzles would result in a dose savings of approximately 0.4 to 0.7 REM per nozzle examination. The historical radiation dose associated with these examinations is presented in Reference 9, Table 2.

The proposed changes to the inservice examination requirements assure that the EFR repaired nozzles are adequately monitored through a combination of volumetric and surface examinations throughout the life of the installation at a frequency approved by the NRC, thus ensuring the EFR repaired nozzles will continue to perform their required function.

5.1.4 Reporting Requirements and Conditions on Use

EGC will notify NRC of the Division of Component Integrity or its successor of changes in indication(s) or findings of new indication(s) in the penetration nozzle or J-groove weld beneath a seal weld repair, or new linear indications in the seal weld repair, prior to commencing repair activities in subsequent outages.

5.2 Technical Basis for Proposed Alternative

As discussed in WCAP-15987-P, the embedded flaw repair technique is considered a permanent repair. As long as a PWSCC flaw remains isolated from the Primary Water (PW) environment, it cannot propagate. Since an Alloy 52 or 52M weldment is considered highly resistant to PWSCC, a new PWSCC flaw should not initiate and grow through the Alloy 52 or 52M seal weld to reconnect the PW environment with the embedded flaw. Structural integrity of the affected J-groove weld and/or nozzle will be

maintained by the remaining unflawed portion of the weld and/or the VHP. Alloy 690 and Alloy 52/52M are highly resistant to stress corrosion cracking, as demonstrated by multiple laboratory tests, as well as over twenty years of service experience in replacement steam generators.

The residual stresses produced by the embedded flaw technique have been measured and found to be relatively low because of the small seal weld thickness. This implies that no new flaws will initiate and grow in the area adjacent to the repair weld. There are no other known mechanisms for significant flaw propagation in the reactor vessel closure head and penetration tube region since cyclic loading is negligible, as described in WCAP-15987-P. Therefore, fatigue driven crack growth should not be a mechanism for further crack growth after the embedded flaw repair process is implemented.

The thermal expansion properties of Alloy 52 or 52M weld metal are not specified in the ASME Code. In this case the properties of the equivalent base metal (Alloy 690) should be used. For Alloy 690, the thermal expansion coefficient at 600 degrees F is 8.2E-6 in/in/degree F as found in Section II part D. The Alloy 600 base metal has a coefficient of thermal expansion of 7.8E-6 in/in/degree F, a difference of about 5 percent. The effect of this small difference in thermal expansion is that the weld metal will contract more than the base metal when it cools, thus producing a compressive stress on the Alloy 600 tube or J-groove weld. This beneficial effect has already been accounted for in the residual stress measurements reported in the technical basis for the embedded flaw repair, as noted in the WCAP-15987-P.

WCAP-16401-P, Revision 1 (Reference 3) provides the plant-specific analysis performed for Byron and Braidwood Stations using the same methodology as WCAP-15987-P. This analysis provides the means to evaluate a broad range of postulated repair scenarios to the reactor vessel head penetrations and J-groove welds relative to ASME Code requirements for allowable flaw size and service life. Based on Reference 3, a service life of at least twenty (20) years was determined for flaws in the VHP nozzles and a service life of at least forty (40) years was determined for flaws in the J-groove attachment welds.

The above proposed embedded flaw repair process is supported by applicable generic and plant specific technical bases, and is therefore considered to be an alternative to Code requirements that provides an acceptable level of quality and safety, as required by 10 CFR 50.55a(z)(1).

6.0 DURATION OF THE PROPOSED ALTERNATIVE

The duration of the proposed alternative is for the Byron Station Units 1 and 2, Fourth Inservice Inspection Interval currently scheduled to end in July 15, 2025.

7.0 PRECEDENTS

In Reference 8, 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.

10 CFR 50.55a RELIEF REQUEST I4R-10
Revision 2

In Reference 11, the NRC provided their authorization to implement Relief Requests I3R-09 and I3R-20, Revision 2 during the Third ISI Interval. The Fourth ISI Interval Relief Request utilizes the same approach that was previously approved under the January 21, 2016 Safety Evaluation for Byron Station Units 1 and 2.

8.0 REFERENCES

1. Westinghouse WCAP-15987-P, Revision 2-P-A, "Technical Basis for the Embedded Flaw Process for Repair of Reactor Vessel Head Penetrations," December 2003

2. Letter from H. N. Berkow (U. S. NRC) to H. A. Sepp (Westinghouse Electric Company), "Acceptance for Referencing – Topical Report WCAP-15987-P, Revision 2, 'Technical Basis for the Embedded Flaw Process for Repair of Reactor Vessel Head Penetrations,' (TAC NO. MB8997)," dated July 3, 2003

3. Westinghouse WCAP-16401-P, Revision 1, "Technical Basis for Repair Options for Reactor Vessel Head Penetration Nozzles and Attachment Welds: Byron and Braidwood Units 1 and 2," January 2017

4. Letter LTR-NRC-03-61 from J. S. Galembush (Westinghouse Electric Company) to Terence Chan (U. S. NRC) and Bryan Benney (U.S. NRC), "Inspection of Embedded Flaw Repair of a J-groove Weld," dated October 1, 2003

5. Letter from R. J. Barrett (U. S. NRC) letter to A. Marion (Nuclear Energy Institute), "Flaw Evaluation Guidelines," dated April 11, 2003

6. American Society of Mechanical Engineers Boiler and Pressure Vessel 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"

7. Letter from R. Gibbs (U. S. NRC) to C. M. Crane (EGC), "Byron Station, Unit No. 2 – Relief Request I3R-14 for the Evaluation of Proposed Alternatives for Inservice Inspection Examination Requirements (TAC No. MD5230)," dated May 23, 2007 (ML071290011)

8. 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, (ML120790647)

9. Westinghouse Report LTR-PSDR-TAM-14-005, Revision 3, "Technical Basis for Optimization or Elimination of Liquid Penetrant Exams for the Embedded Flaw Repair," dated May 2015

10. Letter from J. Zimmerman (U.S. NRC) to M. Pacilio (EGC), "Byron Station, Unit No. 1 – Inservice Inspection Relief Request I3R-19: Alternative Requirements for the

10 CFR 50.55a RELIEF REQUEST I4R-10
Revision 2

Repair of Reactor Vessel Head Penetrations (TAC Nos. ME5877 and ME5948)," dated February 1, 2012 (ML112990783)

11. Letter from Justin C. Poole, (U.S. NRC) to Bryan C. Hanson (EGC), "Byron Station, Units Nos. 1 and 2, and Braidwood Station, Units 1 and 2 – Relief from the Requirements of the ASME Code," dated January 21, 2016 (ML16007A185)