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Proprietary information contained in Attachment 1 and Attachment 3
Withhold from public disclosure under 10 CFR 2.390
When separated, the cover letter, Attachment 2, Attachment 4 and Attachment 5 are non-proprietary

RS-16-057

10 CFR 50.55a

March 15, 2016

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

Braidwood Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. STN 50-456 and STN 50-457

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: Supplement to Response to Requests for Additional Information for Relief for Alternate Requirements for Repair of Reactor Vessel Head Penetrations with Nozzles Having Pressure-Retaining Partial-Penetration J-Groove Welds (RS-16-045)

References:

1) Letter from D. M. Gullott (Exelon Generation Company, LLC) to U. S. Nuclear Regulatory Commission, "Requests for Relief for Alternate Requirements for Repair of Reactor Vessel Head Penetrations with Nozzles Having Pressure-Retaining Partial-Penetration J-Groove Welds," dated September 11, 2015

2) Email from J. Wiebe (NRC) to J. Krejcie (Exelon Generation Company, LLC), Preliminary Request for Additional Information Regarding the Braidwood and Byron Requests for Relief Regarding Repair of Reactor Vessel Head Penetration J-Groove Welds, dated January 6, 2016

3) Letter from D. M. Gullott (Exelon Generation Company, LLC) to U. S. Nuclear Regulatory Commission, "Response to Requests for Additional Information for Relief for Alternate Requirements for Repair of Reactor Vessel Head Penetrations with Nozzles Having Pressure-Retaining Partial-Penetration J-Groove Welds," dated February 11, 2016

In accordance with 10 CFR 50.55a, "Codes and standards," paragraph (z)(1), Exelon Generation Company, LLC (EGC), requested relief from the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (B&PV Code) specifically related to the repair of the degraded reactor vessel closure head (RVCH) penetration nozzles and their associated partial penetration J-groove attachment welds.

Specifically, the relief request proposed to perform an alternative repair technique using the Areva Inside Diameter Temper Bead (IDTB) welding method to restore the pressure boundary of a degraded nozzle. EGC submitted this request to the NRC in Reference 1.

Subsequent to submittal of Reference 1, the NRC requested additional information to support the review of the subject relief request (Reference 2). EGC responded to the questions in the Reference 3 transmittal.

The Reference 3 transmittal included supporting documents containing proprietary information as defined by 10 CFR 2.390, "Public inspection, exemption, requests for withholding." The Reference 3 transmittal also indicated that non-proprietary versions of Attachments 2 through 5 would be transmitted at a later date. The purpose of this transmittal is to transmit the non-proprietary versions of the following documents:

1. AREVA Document #51-9252740-001, "Corrosion Evaluation of Byron Units 1 and 2 and Braidwood Units 1 and 2 IDTB Weld Repairs" (Corresponding to the Proprietary Evaluation included as Attachment 2 of Reference 3)

2. AREVA Document #51-9252709-000, "Ambient IDTB Weld Interpass Temperature Evaluation" (Corresponding to the included Proprietary Evaluation included as Attachment 5 of Reference 3)

Non-proprietary versions of Attachment 3 and Attachment 4 of Reference 3 will be sent to the NRC in a separate transmittal.

Note, this transmittal also includes revisions to the proprietary versions of the above documents as previously sent in the Reference 3 transmittal (i.e., revisions to Attachment 2 and Attachment 5 of Reference 3). Revisions to the proprietary documents were made solely for the purpose of annotating the proprietary information in brackets. No technical or other changes were made to the documents.

Areva, Inc,(Areva) as the owner of the proprietary information has executed the enclosed affidavit, which identifies that the enclosed proprietary information has been handled and classified as proprietary, is customarily held in confidence, and has been withheld from public disclosure. The proprietary information was provided to EGC by Areva as referenced by the affidavit. The proprietary information has been faithfully reproduced in the attached information such that the affidavit remains applicable. Areva hereby requests that the attached proprietary information be withheld, in its entirety, from public disclosure in accordance with the provisions of 10 CFR 2.390 and 10 CFR 9.17. The affidavit supporting the proprietary nature of the information is contained in Attachment 5. Note, the affidavit enclosed here is different from the Reference 3 Attachment 6 affidavit, updated to reflect the current revisions to the proprietary documents evaluations included in this transmittal.

There are no regulatory commitments contained within this letter.

Should you have any questions concerning this letter, please contact Ms. Jessica Krejcie at (630) 657-2816.

Respectfully,

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

Attachment 1: AREVA Document #51-9234023-002, "Corrosion Evaluation of Byron Units 1 and 2 and Braidwood Units 1 and 2 IDTB Weld Repairs" (PROPRIETARY)
Attachment 2: AREVA Document #51-9252740-001, "Corrosion Evaluation of Byron Units 1 and 2 and Braidwood Units 1 and 2 IDTB Weld Repairs" (NON-PROPRIETARY)
Attachment 3: AREVA Document #51-9245035-002, "Ambient IDTB Weld Interpass Temperature Evaluation" (PROPRIETARY)
Attachment 4: AREVA Document #51-9252709-000, "Ambient IDTB Weld Interpass Temperature Evaluation" (NON-PROPRIETARY)
Attachment 5: AREVA Inc., Affidavit for AREVA documents, March 8, 2016

ATTACHMENT 2

AREVA Document #51-9252740-001, "Corrosion Evaluation of Byron Units 1 and 2 and Braidwood Units 1 and 2 IDTB Weld Repairs"

NON-PROPRIETARY

AREVA Inc.

Engineering Information Record

Document No.: 51 - 9252740 - 001

Corrosion Evaluation of Byron Units 1 and 2 and Braidwood Units 1 and 2 IDTB Weld Repairs

Corrosion Evaluation of Byron Units 1 and 2 and Braidwood Units 1 and 2 IDTB Weld Repairs

Safety Related? ☒ YES ☐ NO

Does this document establish design or technical requirements? ☐ YES ☒ NO

Does this document contain assumptions requiring verification? ☐ YES ☒ NO

Does this document contain Customer Required Format? ☐ YES ☒ NO

Signature Block

Name and Title/Discipline	Signature	P/LP, R/LR, A-CRF, A	Date	Pages/Sections Prepared/Reviewed/ Approved or Comments
Greg Troyer Engineer II Component Engineering		P	3/8/2016	All
Ryan Hosler Supervisory Engineer Materials Engineering		R	3/8/16	All
Brian Haibach Manager Materials & RCS Structural Analysis		A	3/8/16	All

Note: P/LP designates Preparer (P), Lead Preparer (LP)
R/LR designates Reviewer (R), Lead Reviewer (LR)
A-CRF designates Project Manager Approver of Customer Required Format (A-CRF)
A designates Approver/RTM – Verification of Reviewer Independence

Project Manager Approval of Customer References (N/A if not applicable)

Name (printed or typed)	Title (printed or typed)	Signature	Date
Dave Skulina	Project Management		3/8/2016

Record of Revision

Revision No.	Pages/Sections/ Paragraphs Changed	Brief Description / Change Authorization
000	All	This content of this document is identical to 51-9234023-001, except that proprietary information is redacted.
001	All	This content of this document is identical to 51-9234023-002, except that proprietary information is redacted.

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1.0 PURPOSE

This document evaluates the potential corrosion concerns of the components and weld that establish the reactor vessel closure head (RVCH) vessel head penetration (VHP) pressure boundary following the contingency inner diameter temper-bead (IDTB) repair (i.e., “half nozzle” repair) of RVCH VHPs at Byron Units 1 and 2 and Braidwood Units 1 and 2 [1]. These corrosion concerns include corrosion of exposed low alloy steel and nickel-based weld metal. This document further justifies a conservative estimate of the general corrosion rate of the exposed RVCH low alloy steel (SA-533, Grade B, Class 1) based on a 20 month cycle (18 month operation cycle plus 2 month refueling cycle). Primary water stress corrosion cracking (PWSCC) of the remaining upper portion of the Alloy 600 nozzle is a concern but is not evaluated in this document.*

2.0 BACKGROUND

Several control rod guide mechanism (CRDM) nozzles at domestic pressurized water reactors (PWRs) have leaked via cracking attributed to PWSCC. Exelon Generation Company, LLC (Exelon) intends to perform inspections on the Byron Units 1 and 2 and Braidwood Units 1 and 2 RVCH VHPs (CRDM, reactor vessel level indication system [RVLIS], spare, and thermocouple nozzles) during an upcoming outage, and make contingency preparations for possible repairs.

Based on recent RVCH VHP repair experiences, AREVA Inc. (AREVA) has prepared contingency repair configurations for the VHPs [1]. The modification consists of roll expansion of the existing Alloy 600 nozzle to maintain nozzle engagement during and after the subsequent machining and during initial welding activities, followed by the removal of the lower portion of the existing nozzle which may contain the defect(s). The upper portion of the existing nozzle and a portion of the original pressure boundary partial penetration J-groove weld remain in place. The new pressure boundary VHP weld (Alloy 52, Alloy 52M, or Alloy 52MSS) is located in the RVCH penetration bore above the original J-groove weld. The lower portion of thermal sleeves and thermocouple column nozzles are removed to facilitate modification activities and subsequently replaced. The lower portion of the CRDM, RVLIS and Spare nozzles are not replaced. Remediation may be performed to increase the resistance to primary water stress corrosion cracking (PWSCC) in the most susceptible area of the modified configuration [1].

The IDTB repair configuration described above is shown in Figure 2-1, Figure 2-2, and Figure 2-3 [2,3]. The proposed repair configuration will leave portions of the low alloy steel inside the RVCH VHP exposed to the primary reactor coolant, Figure 2-1 and Figure 2-3.

* For Information Only: The PWSCC evaluation may be found in AREVA Documents 51-9233902 (Proprietary) and 51-9252742 (Non-Proprietary).

Figure 2-1: RVCH CRDM, RVLIS and Spare Nozzle IDTB Repair Configuration, Prior to Replacement Thermal Sleeve Attachment [2]

Figure 2-2: RVCH CRDM and RVLIS Nozzle IDTB Repair Configuration with Replacement Thermal Sleeve [2]

Figure 2-3: RVCH Thermocouple Nozzle IDTB Repair Configuration with Replacement Thermocouple Housing Extension [3]

3.0 ASSUMPTIONS

3.1 Unverified Assumptions

There are no unverified assumptions used in this analysis.

3.2 Justified Assumptions

1. A typical capacity factor of 90% is used to estimate the overall general corrosion rate for exposed low alloy steel from general corrosion rates at operating (high temperature, deaerated water) and shut down (low temperature, aerated water) conditions.

A 90% capacity factor is conservative for this calculation, [

]

2. The use of 0.0036 in/year for the general corrosion rate of the exposed low alloy steel (LAS) RVCH is justified in Section 5.1.

3.3 Modeling Simplifications

There are no modeling simplifications used in this analysis.

4.0 KNOWN OCCURRENCES OF EXPOSED CARBON/LOW ALLOY STEEL BASE METAL

The carbon or low alloy steel components in the pressurizer, reactor vessel, and the steam generator exposed to PWR reactor coolant system (RCS) are clad with either stainless steel or nickel-base alloy in order to help prevent general corrosion of the carbon or low alloy steel base metal. Throughout the operating history of domestic PWRs, there have been many cases where a localized area of the carbon or low alloy steel base metal has been exposed to the PWR primary coolant due to damage to the cladding or a repair configuration. A sampling of such instances is listed below as historical information For Information Only:

1960s Yankee-Rowe reactor vessel – Surveillance capsules fell from holder assemblies to the bottom of the vessel, releasing test specimens and other debris, leading to perforations in the cladding.

1990 Three Mile Island Unit 1 steam generator – One tube with a circumferential crack separated within the tubesheet area exposing the tubesheet material to primary coolant. (LER 289-1990-005)

1990 ANO Unit 1 pressurizer – A leak was detected at the pressurizer upper level instrumentation nozzle within the steam space in December 1990. The repair consisted of removing the outer section of the nozzle followed by welding a section of new nozzle to the outer diameter (OD) of the pressurizer. An axial crevice exists between the old and new nozzle sections, which exposes the pressurizer shell base metal to primary coolant. Nondestructive (NDE) examinations from the pressurizer OD a few years after the repair have revealed no indications of corrosion. (LER 313-1990-021)

1991 Oconee Unit 1 steam generator – A misdrilled tubesheet hole in the upper tubesheet of one of the steam generators, during plugging operation in 1991, led to exposure of a small area of unclad tubesheet to primary coolant. [Note: This area of the tubesheet has since been "patched" and is no longer exposed to coolant.]

1993 McGuire Unit 2 reactor vessel – A defect in the vessel cladding was discovered during an inspection in July 1993; the defect is believed to have occurred as a result of a pipe dropped in the vessel during construction (1975).

1993 SONGS Unit 2 hot leg nozzle – A repair to a hot leg nozzle was completed during the 1993 outage at the SONGS Unit 2. This repair consisted of replacing a section of the existing Alloy 600 nozzle with a new nozzle section fabrication from Alloy 690. A small gap exists between the two nozzle sections where the carbon steel base metal is exposed to the primary coolant.

1994 Calvert Cliffs Unit 1 pressurizer – Two leaking heater nozzles in the lower head of the pressurizer were partially removed and the penetrations were plugged in 1994. (LER 317-1994-003)

1994 St. Lucie Unit 2 pressurizer – Three pressurizer steam space nozzles were repaired with a half-nozzle technique similar to ANO Unit 1 and SONGS Unit 2. UT examinations from the pressurizer OD have revealed no indications of significant corrosion. (LER 389-1994-002)

1997 Oconee Unit 1 OTSG manway – During the end-of-cycle (EOC) 17 refueling outage, a degraded area was observed in the "bore" of the lB once through steam generator (OTSG). Subsequent inspection revealed a small circumferential damaged area to the cladding surface of the manway opening.

2001 AREVA IDTB CRDM repairs at Oconee Unit 2, Oconee Unit 3, Crystal River Unit 3, Three Mile Island Unit 1, and Surry Unit 1. (LER 270-2001-002, LER 287-2001-003, LER 302-2001-004, LER 289-2001-002, LER 280-2001-003)

2002 AREVA IDTB CRDM repairs at Oconee Unit 1 and Oconee Unit 2. (LER 269-2002-003, LER 270-2002-002)

2003 AREVA IDTB CRDM/CEDM repairs at St. Lucie Unit 2 AREVA half-nozzle repairs of South Texas Project Unit 1 bottom mounted instrument (BMI) nozzles, AREVA half-nozzle repairs of pressurizer upper level nozzles at Crystal River Unit 3. (LER 389-2003-002, LER 498-2003-003, LER 302-2003-003)

2005 Half-nozzle modification for the TMI-1 pressurizer vent nozzle.

2013 AREVA IDTB half-nozzle repairs to the Harris CRDM nozzle penetrations. (LER 400-2013-001)

In each of these instances, carbon or low alloy steel base metal was exposed to primary coolant in a localized area. Each plant returned to normal operation with the base metal exposed; in the case of Yankee-Rowe, the vessel operated for about 30 years with the base metal exposed.

5.0 CORROSION OF EXPOSED LOW ALLOY STEEL

Several types of corrosion can occur when carbon and low alloy steel base metal are exposed to primary coolant. During operating conditions, the primary coolant is deaerated and at high temperatures [

] depending on the location within the RCS. During shutdown conditions, the primary coolant temperature approaches 70°F and may become aerated and/or stagnant depending on the location within the RCS. The following sections discuss the potential corrosion mechanisms for the exposed low alloy steel base metal shown in Figure 2-1 and Figure 2-3, [

]

5.1 General Corrosion

General corrosion is defined as uniform deterioration of a surface by chemical or electrochemical reactions with the environment. Austenitic stainless steels and nickel-base alloys (e.g. wrought Type 304, Type 316 and Alloy 600 and Alloy 690 materials and their equivalent weld metals) are essentially immune to general corrosion in a PWR environment. Carbon and low alloy steels, however, may be subject to general corrosion upon exposure to the PWR environment. The general corrosion rates of carbon and low alloy steels during aerated and deaerated conditions are discussed below.

5.1.1 Oxygen Level in the Repaired Area

[

]

5.1.2 General Corrosion Rate

Many investigators have reported corrosion rates of carbon and low alloy steels in various environments [8,9,10, 11,12,13,14,15,16]. EPRI has published a handbook on boric acid corrosion [17]. This handbook summarizes the industry field experience with boric acid corrosion incidents, contains a discussion of boric acid corrosion mechanisms, and contains a compilation of prior boric acid corrosion testing and results. In Revision 1 of this handbook, the corrosion rates for low carbon and low allow steel in normal deaerated PWR primary water conditions were typically bounded by 0.001 in/yr. Revision 2 includes further testing that showed rates on the order of 0.010 in/yr; however, these subsequent tests are clearly outliers from the otherwise strong trend of corrosion rate and pH. Furthermore, these outlying points were based on on-line (electrical) measurements only

and were not validated against mass loss measurements. Excluding these points, the expected general corrosion rate remains at or below 0.001 in/yr at pH of around 7 or greater (bulk RCS fluid pH [5]).

In several studies, the corrosion rates for carbon and low alloy steels such as ASME SA-302 Grade B, ASME SA-533 Grade B (the material of the Byron Units 1 and 2 and Braidwood Units 1 and 2 RVCHs) and ASTM A212 have been observed to be similar in PWR environments [8,9,10,14,15,17].

In one evaluation, ASTM A212 carbon steel was exposed to primary coolant in aerated and deaerated conditions [18]. It was shown that under deaerated conditions (i.e., during operation), the corrosion rate depended on temperature, fluid velocity, boric acid concentration, and time [18]. At the maximum velocity tested (36 ft/sec), the corrosion rate was determined to be 0.003 inch/year (ipy) [18]. Under low flow or static conditions at 650°F, a maximum corrosion rate of 0.0009 ipy was reported. In this same study at shutdown conditions (aerated, low temperature [~70°F]), the maximum corrosion rate was determined to be 0.0015-inch for a two-month shutdown, or 0.009 ipy [18].

[

]

Combined Corrosion Rate = [ ]

= 0.0036 ipy

5.1.3 General Corrosion Rate in the Heat Affected Zone (HAZ)

[

]

5.2 Crevice Corrosion

The environmental conditions in a crevice can become aggressive with time and can cause accelerated local corrosion. Crevice corrosion testing of various combinations [

] Experiments were conducted to determine the crevice corrosion rate of low alloy steel in air saturated water. [

] Therefore, increased corrosion rates of

exposed low alloy steel in crevice conditions with dearated bulk coolant are not expected. This is supported by cases of low alloy steel in crevice conditions being exposed to primary coolant (see Section 4.0 for a partial list) without any known crevice corrosion.

Corrosion studies have examined crevice corrosion in the gaps between [

]

Crevice corrosion is not expected to be an issue for the CRDM, RVLIS and spare nozzles due to the open geometry of the final repair configuration (see Figure 2-1 and Figure 2-2). The final repair configuration for the thermocouple column will create crevice conditions between the low alloy steel in the RVCH and portions of the new nozzle (see Figure 2-3). As discussed in the preceding paragraph, crevice corrosion is not expected to be more severe than general corrosion for the low alloy steel. Furthermore, any corrosion deposits will likely plug the crevice path causing corrosion in the crevice to cease. Based on the preceding discussion, it is judged that crevice corrosion is not a concern for this repaired configuration.

5.3 Galvanic Corrosion

Galvanic corrosion may occur when two dissimilar metals in contact (i.e. coupled) are exposed to a conductive solution. The larger the potential difference between the two metals, the greater the likelihood of galvanic corrosion. Low alloy and carbon steels are more anodic than stainless steels and the nickel-base alloys and could therefore be subject to galvanic attack when coupled and exposed to reactor coolant.

Corrosion tests have been performed to determine the influence of coupling between low alloy steel and austenitic stainless steel, which has approximately the same corrosion potential as nickel-based alloys [17]. In one test, alloy steels were coupled (i.e., welded) to stainless steels exposed to high purity water for 1000 hours at 546°F in steam, steam/water, saturated water, and sub-cooled water in aerated and deaerated conditions [8]. The coupled specimen did not exhibit any accelerated rates of corrosion. Specimens made from 5% chromium steel coupled to Type 304 stainless steel were exposed to aerated water at 500°F for 85 days (~2000 hours) and 98 days (~2300 hours) with no evidence of galvanic corrosion [8]. In each of the tests described above, corrosion rates were not affected by coupling. The results of the Nuclear Regulatory Commission's (NRC's) boric acid corrosion test program have shown that the galvanic difference between ASTM A533 Grade B Steel, Alloy 600, and Type 308 stainless steel (used in reactor vessel cladding) is not significant enough to consider galvanic corrosion as a strong contributor to the overall boric acid corrosion process [20]. Based on the preceding discussion, it is judged that galvanic corrosion between the exposed RVCH low alloy steel and Alloy 600, Alloy 690, or their weld metals is not a concern for this repaired configuration.

This is supported by many cases of low alloy steel coupled to stainless steel or nickel-base alloys being exposed to primary coolant (see Section 4.0 for a partial list) without known galvanic corrosion observed.

5.4 Stress Corrosion Cracking

Stress corrosion cracking (SCC) occurs when the following three conditions are present:

- a susceptible material
- a tensile stress
- an aggressive environment

Numerous laboratory studies have been performed on carbon and low alloy steels to assess their susceptibility to SCC in various aqueous environments. Most of these studies, which have been performed on reactor materials in light water reactor (LWR) environments, have been coordinated under the auspices of the International Cyclic

Crack Growth Rate (ICCGR) Group. While this group has focused its attention on corrosion fatigue, much of the work has also been on SCC. An excellent review of the work conducted through 1990 appears in a paper by Scott and Tice [21]. The conclusion of this evaluation is that, considering the environmental conditions present in a PWR, low alloy and carbon steel will not be subject to SCC unless they experience some prolonged departure from expected normal operation conditions, which is not permitted per the water chemistry guidelines. The results of these evaluations have been supported by several more recent laboratory studies [22,23,24,25].

Extensive PWR (see Section 4.0) and BWR [25] operating experience related to low alloy steel being exposed to reactor coolant has resulted in no known occurrences of SCC of low alloy steel reactor vessel material to any significant depth. Noteworthy are SCC cracks revealed in the stainless steel cladding in charging pumps [26,27]. The interdendritic cracks, present in the cladding, were determined to have blunted at the clad/low alloy steel interface.

Based on the preceding discussion it is judged that SCC of the exposed low alloy steel RVCH is not a concern for this repaired configuration.

5.5 Hydrogen Embrittlement

It is known that the adverse effects due to hydrogen become most acute in high-strength steels, where instances of loss of ductility and delayed failure have been attributed to the presence of hydrogen at the 1 ppm or lower levels. However, hydrogen embrittlement is also possible for lower-strength steels such as SA-533 Grade B (the material for the Byron Units 1 and 2 and Braidwood Units 1 and 2 RVCHs), although higher hydrogen concentrations are required to effect a decrease in ductility similar to that of high-strength steels [28,29,30].

[ ] For A302B steel (a reactor vessel steel similar to SA-533 Grade B) irradiated to nearly 3x10^{20} n/cm^2 (E > 1 MeV), ductility was not significantly affected by the presence of up to 2 ppm hydrogen [30]. The NRC's Expanded Materials Degradation Assessment (EMDA) concludes that, based on the data available, a hydrogen level of 4 ppm and higher in the reactor vessel material could become a contributor to the overall degradation in fracture resistance of the reactor vessel [30]. [ ]

Hydrogen exists within the reactor coolant system (used as an oxygen scavenger) and can also be produced by radiolytic decomposition of the water. However, the primary source of hydrogen that could cause embrittlement is general corrosion of the exposed low alloy steel [32]. The hydrogen concentration at the exposed low alloy steel surface during normal operation could be as high [ ] [32]. This calculation assumes [ ] This is not likely given [ ]

The hydrogen concentration at shutdown temperatures (70°F to 140°F) was measured to be as high as 2 ppm for A302B steel exposed to boric acid solution in aerated and deaerated conditions for as long as 4 months. The tests showed no increase in hydrogen concentration with time. This test program showed hydrogen concentrations lower than about [ ] This is reasoned to be due to [ ]

† 1 cm^3 Hydrogen (Normal Temperature Pressure) per 100 grams of steel ~ 0.84 ppm.

The hydrogen concentration in the RVCH will be greatest at the exposed low alloy steel surface and will decrease across the thickness of the RVCH to the trace concentration of hydrogen in the steel. Thus, the average hydrogen concentration in the RVCH is less than the maximum hydrogen concentrations summarized in this section.

Based on the preceding discussion, it is judged that hydrogen embrittlement is not a concern for the exposed RVCH low alloy steel in this repaired configuration. This is supported by many cases of low alloy steel being exposed to primary coolant (see Section 4.0 for a partial list) without cracking attributed to hydrogen embrittlement observed.

6.0 CORROSION OF WROUGHT ALLOY 690 AND ITS WELD METALS

Alloy 690, Alloy 52, Alloy 52M and Alloy 52MSS are the specified RVCH VHP IDTB weld repair materials [1]. To better understand the potential corrosion concerns of Alloy 52, Alloy 52M and Alloy 52MSS, information regarding the corrosion of Alloy 690 (the associated base metal) and Alloy 152 will also be presented. The corrosion resistance of Alloy 52M and Alloy 52 MSS is expected to be similar to that of Alloy 52 and Alloy 690. The differences between Alloy 52, Alloy 52M and Alloy 52MSS are only minor alloying elements for enhanced weldability. The chromium content, which provides the corrosion resistance of the material, of Alloys 52M and 52MSS is similar to Alloy 52 and Alloy 690 [34]. The corrosion resistance of Alloy 690 and weld metals Alloy 52 and 152 has been extensively studied as a result of numerous PWSCC failures in mill annealed Alloy 600 and weld metals Alloy 82 and 182 in PWR environments. As a result of Alloy 600 and Alloy 82 and 182 failures, Alloy 690 and weld metals Alloys 52 and 152 have been chosen by the nuclear industry as the replacement materials of choice for Alloy 600 and weld metals Alloy 82 and 182 in PWRs.

A comprehensive review of testing for the use of Alloy 690 and Alloy 52 in PWR systems cites numerous investigations and tests results under a wide array of conditions, including both primary (high temperature deoxygenated water) and secondary coolant environments [35]. The first Alloy 690 steam generator went online in May 1989 with no reported tube failures due to environmental degradation [36]. To this date, there are no known Alloy 690 in-service PWSCC failures [37]. Nickel-base alloys in general have an excellent resistance to general and crevice corrosion [36,38].

The typical test conditions cited in the Alloy 690 literature review included temperatures to 689°F (365°C), dissolved oxygen levels <20 ppb, tests in doped and undoped 752°F (400°C) steam, lithium concentrations up to 20 ppm, chlorides up to 300 ppb, and various heat treatments of Alloy 690. Reverse U-bend SCC tests within the above matrix of environmental conditions produced no PWSCC in Alloy 690. No cracking was observed in high purity water containing 16 ppm oxygen at 550°F (288°C), even in a creviced situation. Only slight intergranular cracking of Alloy 690MA (mill annealed) was observed in slow strain rate testing (SSRT) in 680°F (360°C) high purity deaerated hydrogenated water. However, the same cracking was also observed in argon, so the cracking observed in high purity deaerated hydrogenated water could not be confirmed to be PWSCC. [36]

The SCC resistance of weld metals Alloy 52 and 152 was identified as unaffected by a variety of test conditions, including primary water. No cracking occurred in weld metals containing >22% chromium [36]. [

]

SCC test data comparing results between Alloy 690 and Alloy 600 is available in both aerated and deaerated high temperature water. [

]

[

]

Alloy 52M was tested (not as a part of this work scope) in accelerated corrosion conditions by testing a weld mockup that simulated nozzle safe end repairs. The testing consisted of 752°F (400°C) steam-plus-hydrogen doped with 30 ppm each of fluoride, chloride, and sulfate anions. The hydrogen partial pressure was controlled at approximately 75 kPa with a total steam pressure of 20 MPa; this environment has been previously used to accelerate the simulated PWSCC of nickel-base alloys. After a cumulative exposure of 2051 hours (equivalent to 45.6 effective full power years [EFPY]), no environmental degradation was detected on the surface of the Alloy 52M welds. Small microfissures on the surface of two of the high-strain Alloy 52M welds, stressed in tension, did not serve as initiation sites for environmental degradation, nor did they propagate during the tests. Stress corrosion cracks initiated in the also tested Alloy 182 welds in exposure times less than one-fifth the total exposure time of the Alloy 52M specimens [41]. Alloy 52MSS PWSCC susceptibility testing in simulated primary water shows crack growth rates similar to other high chromium nickel weld metal alloys (Alloy 52, Alloy 52M, Alloy 152). These studies and others [40,42] indicate that the weld metals in this repair configuration are expected to have a low susceptibility to PWSCC.

Based on the preceding discussion, it is judged that PWSCC of Alloy 690 and its weld metals (Alloy 52, Alloy 52M and Alloy 152 is not a concern for this repaired configuration.

7.0 CONCLUSIONS

This report evaluates potential corrosion concerns of the components and weld that establish the RVCH VHP pressure boundary following the contingency IDTB repair of RVCH VHPs (Figure 2-1, Figure 2-2, and Figure 2-3) at Byron Units 1 and 2 and Braidwood Units 1 and 2.

It is judged that galvanic corrosion, hydrogen embrittlement, SCC, and crevice corrosion are not a concern for the exposed RVCH low alloy steel base metal resulting from the IDTB repair process. General corrosion of the exposed RVCH low alloy steel base metal as shown in Figure 2-1 and Figure 2-3 is expected to occur. Based on industry data and AREVA's experience, the general corrosion rate is conservatively estimated to be 0.0036 inch/year based on an 18-month operation cycle plus a 2-month refueling cycle.

Wrought Alloy 690 has been shown by extensive testing and in-reactor operating experience to exhibit superior PWSCC resistance compared to other available alloys, with no known in-service failures at this time. In laboratory testing, cracking of Alloy 690 has not been observed for oxygenated conditions (up to 16 ppm) in creviced situations. However, cracking of Alloy 690 is possible under more extreme environmental conditions not found in PWRs. Alloys 52, 52M, and 52MSS have also been shown to exhibit superior PWSCC resistance compared to Alloys 82 and 182. Based on this review, Alloy 690 and its weld metals are the best available replacement materials for Alloy 600 and its weld metals in PWRs. It is judged that degradation of Alloy 690 and its weld metals from exposure to typical PWR environments is not a concern for this repair configuration.

8.0 REFERENCES

References identified with an (*) are maintained within Exelon's Records System and are not retrievable from AREVA Records Management. These are acceptable references per AREVA Administrative Procedure 0402-01, Attachment 8. See page 2 for Project Manager Approval of customer references.

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ATTACHMENT 4

AREVA Document #51-9252709-000, "Ambient IDTB Weld Interpass Temperature Evaluation"

NON-PROPRIETARY

AREVA Inc.

Engineering Information Record

Document No.: 51 - 9252709 - 000

Ambient IDTB Weld Interpass Temperature Evaluation – Non Proprietary

Safety Related? ☒ YES ☐ NO

Does this document establish design or technical requirements? ☐ YES ☒ NO

Does this document contain assumptions requiring verification? ☐ YES ☒ NO

Does this document contain Customer Required Format? ☐ YES ☒ NO

Signature Block

Name and Title/Discipline	Signature	P/LP, R/LR, A-CRF, A	Date	Pages/Sections Prepared/Reviewed/ Approved or Comments
Steven Wolbert Welding Engineer	SE WOLBERT 2/4/2016	LP		ALL
Ben Grimmett Supervisor, Welding Engineering	BB GRIMMETT 2/5/2016	LR		ALL
Dave Waskey Manager, CR&D	DE WASKEY 2/5/2016	A		ALL

Note: P/LP designates Preparer (P), Lead Preparer (LP)
R/LR designates Reviewer (R), Lead Reviewer (LR)
A-CRF designates Project Manager Approver of Customer Required Format (A-CRF)
A designates Approver/RTM – Verification of Reviewer Independence

Project Manager Approval of Customer References (N/A if not applicable)

Name (printed or typed)	Title (printed or typed)	Signature	Date
N/A	N/A	N/A	N/A

Record of Revision

Revision No.	Pages/Sections/ Paragraphs Changed	Brief Description / Change Authorization
000	All	Original issue

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1.0 PURPOSE

1.1 The purpose of this document is to demonstrate that the interpass temperatures for an ambient ID Temper Bead repair of a CRDM/CETC Nozzle Penetration does not require performance of direct temperature measurements prior to each weld pass per the method described in Ref 5.6 subsection 3(e)(2). The configuration of the repair does not allow for direct monitoring of the interpass temperature and therefore requires an evaluation to verify interpass temperature is maintained below the level required by ASME code and the welding procedure.

2.0 ASSUMPTIONS

2.1 Ref. 5.6 subsection 3(e)(2) provides the required variables for performing an interpass heat flow evaluation. All variables are either directly or indirectly utilized in the evaluation as described below:

2.1.1 **Welding heat input** – Welding heat input is used to calculate the "Q" variable in Rosenthal's 3D equation. As discussed below, the maximum allowable heat input for the production WPS (Ref. 5.1) is used in the evaluation although actual heat inputs are typically considerably lower, thus the case evaluated is conservative.

2.1.2 **Initial base material temperature** – The initial component temperature (T_0) is assumed to be less than [ ] for the case analyzed. However, if the initial component temperature is found to exceed [ ] prior to welding, alternative wait times based on maximum initial component temperature are documented in Appendix A.

2.1.3 **Configuration, thickness, and mass of the item being welded** – the configuration, thickness, and mass of the component is considered in the selection of Rosenthal's 3-D equation for a moving heat source (heating effects) and 1-D heat loss equation (cooling effects). For example, both equations assume steady-state heat flow (i.e. no transients), constant thermal properties, no heat losses from the surface, and no convective effects of the weld pool. The 1-D heat loss equation is an extremely conservative choice for cooling effects as it only assumes heat loss in one plane whereas the true application experiences cooling in all directions. However, the 1-D equation is used due to the analytical nature of the equation as well as providing added conservatism.

2.1.4 **Thermal conductivity and diffusivity of the materials being welded** – Thermal conductivity and diffusivity are used in the 1-D cooling equation and are provided by Ref. 5.5 for the base material being welded.

2.1.5 **Arc time per weld pass and delay time between each pass** – Arc time per pass is considered since steady state heat distribution is assumed in Rosenthal's 3-D equation (no transients). Delay time between each pass is considered in the 1-D cooling equation. A [ ] average time interval between subsequent

passes is very conservative as it assumes an arc-on time of [ ] compared to a typical processing and setup time of [ ]. Based on AREVA's extensive experience with CRDM nozzle repairs and supporting project documentation, a typical repair evolution is on the order of [ ]. Therefore, a [ ] wait time is conservative, but was selected as a "worst case" based on an elevated initial head temperature.

2.1.6 **Arc time to complete weld** – the arc time to complete the weld is considered by the increase in temperature per pass multiplied by the total number of passes used in the repair. Historical data of previous repairs by the same process were used to establish the total number of passes required.

2.2 Process Modeling Assumptions

2.2.1 **Larger bore diameter covered by evaluation –** a larger bore diameter is more conservative than the case analyzed due to the increased cooling time from the time to arc passes the next start position to the end of the weld.

2.2.2 **Bead width –** Values for bead width were provided from direct measurements of the smallest bead on an IDTB weld mockup using first layer parameters from the production WPS (Ref. 5.1). The smallest number was selected for conservatism as the axial offset from the previous bead would be smaller as a result.

2.2.3 **Use of location 180° away from the arc –** The use of a point 'R' 180° away from the arc in Rosenthal's equation is used due to the difference between the flat plate model and the cylinder model. In practice (see section 3.2), the next point 'R' is 270° away from the arc at weld completion. However, using 270° ([ ]) would provide a lower change in temperature without accounting for convective effects due to the cylindrical shape of the weld joint in Rosenthal's when welding in a ring configuration. Therefore, the approach is more conservative.

2.2.4 **Absorption efficiency** – Absorption efficiency is the amount of energy produced by the arc compared to the amount of energy absorbed by the base material, has been reported as 60% - 90% for the GTAW process (Ref 5.7). An absorption efficiency of 90% is used for conservatism (i.e. majority of heat is being absorbed by the base material).

2.2.5 **Bead overlap –** A bead overlap of [ ] is specified in the production WPS (Ref. 5.1).

2.3 All assumptions have been verified through literary references and/or review of prior experience with performing mockup and production IDTB repairs. There are no unverified assumptions.

3.0 ANALYTICAL EVALUATION

3.1 The interpass temperature evaluation has been performed using closed form solutions which conservatively bounds the final result. This method of solution utilizes industry accepted heat source calculations in addition to one dimensional heat transfer calculations thereby modeling the thermal conditions of the field weld in a conservative manner.

3.2 The welding process is directed by Ref. 5.1 and the final weld is shown in Ref. 5.2 and/or Ref. 5.3. [

]

The WPS identified the heat input at less than [ ].

1 BTU = 1,055 J.

The initial bore is approximately [ ] diameter and reduced to approximately [ ]. A larger bore diameter is conservative and covered by this evaluation. The head thickness is greater than [ ] and the weld is at least [ ] from the head OD.

Estimations of the total increase in temperature per weld pass can be found by two equations: Rosenthal's 3-D equation for a moving heat source (heating effects) and 1-D heat loss equation (cooling effects). From Reference 5.4 the temperature increase immediately following welding can be estimated at a given location (See Figure 1) from the heat source by Rosenthal's 3-D equation:

$$T_w - T_o = \frac{Q}{2\pi KsR} \exp\left(\frac{-U(R-x)}{2\alpha}\right)$$

Where:

Q = Welding Heat Input
K_s = Thermal Conductivity
R = Distance from moving arc to point of question
U = Arc Travel Speed
α = Thermal diffusivity
T_w-T_0 = change in temperature at point "R"

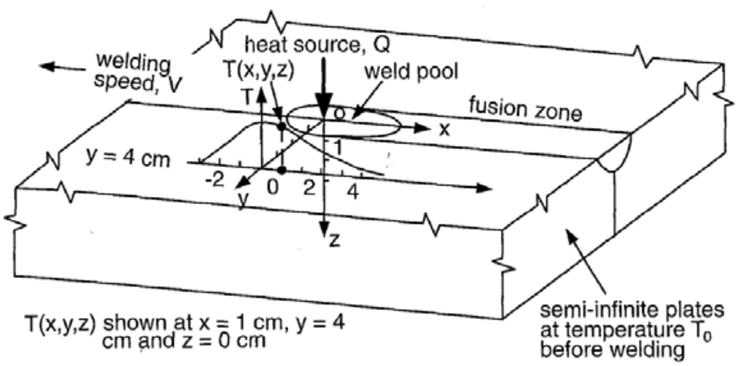

Figure 1: Rosenthal's 3D Heating Schematic (Ref. 5.4)

$$R = \sqrt{\left(x^2 + y^2 + z^2\right)}$$

Where x is the direction of travel and y, z are normal to x. In this case, the temperature at the weld centerline at a location 180° away from the arc is the point of interest 'R'. Therefore, y=0, z=0, and x = ½ the circumference or [ ].

Therefore, R – x is 0 and thus the exponent function becomes equal to 1:

$$\frac{U(R-x)}{2\alpha} = 0.000 \qquad \text{and} \qquad \exp(-0) = 1$$

Therefore, rewriting Rosenthal's equation without the exponent function:

[ ]

Note that layer one has an actual maximum heat input of [ ] for WP3/43/F43TBSC3, thus the case analyzed is conservative. Additionally, the absorption efficiency (η), or the amount of energy produced by the arc compared to the amount of energy absorbed by the base material, has been reported as 60% - 90% for the GTAW process (Ref 5.7). An absorption efficiency of 90% is used for conservatism.

Now assume that the set up for each circumferential weld pass takes [ ] on average, a very conservative model for heat loss results, and estimate the degree of cooling during set up by the analytical solution of a semi-infinite solid with conduction effects:

$$\frac{T(\chi,\tau)-T_o}{T_w-T_o} = \left[erf\left(\frac{y}{2\sqrt{\alpha\tau}}\right)\right]$$ heat loss in one (1) direction only from Reference 5.8, pg. 4-80

[ ]; where τ is time in hours

y = [ ]; where y is the distance to the area where the subsequent pass will be deposited ([ ]).

*α = 0.437 ft^2/hr ; where α is thermal diffusivity for Material Group C * Reference 5.5

$T(\chi, \tau) - T_o$ = [ ] ; the local temperature increase per pass

Based on historical data, the weld process is typically completed in less than [ ] weld passes. If the local temperature increases by [ ] per pass, the total temperature increase is conservatively estimated to be [ ] over the course of the entire weld process. Based on AREVA's extensive experience with CRDM nozzle repairs, the average weld time a typical repair evolution is on the order of [ ]. Therefore, time between subsequent weld passes will be significantly greater to allow for weld sequencing, viewing previously deposited weld passes, completing appropriate paperwork, independent verifications, routine equipment maintenance (e.g. tungsten electrode replacement) and welder attention to detail during the repair evolution such that the area will never exceed interpass temperature limits.

4.0 CONCLUSION

4.1 The Automated Temper Bead repair method is a low heat input process with small weld deposition.

4.2 Assuming an average time interval of [ ] between restarting subsequent 360° weld passes, the increase of interpass temperature per pass is low enough to ensure the interpass temperature will never be exceeded. A [ ] average time interval

between subsequent passes is very conservative as it assumes an arc-on time of [ ] compared to a typical processing and setup time of [ ]. Based on AREVA's extensive experience with CRDM nozzle repairs, a typical repair evolution is on the order of [ ]. Therefore, time between subsequent weld passes will be significantly greater to allow for weld sequencing, viewing previously deposited weld passes, completing appropriate paperwork, independent verifications, routine equipment maintenance (e.g. tungsten electrode replacement) and welder attention to detail during the repair evolution such that the area will never exceed interpass temperature limits.

4.3 In the event that the initial head temperature exceeds [ ] prior to welding, alternative wait times have been evaluated based on different initial head temperatures (Appendix A).

4.4 Direct monitoring of the interpass temperature between weld passes is not required as justified by the evaluation in section 3.0.

5.0 REFERENCES

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5.3 AREVA NP Drawing 02-9232824E, Revision 000

5.4 Welding Metallurgy by Sindo Kou
John Wiley & Sons Copyright 1987

5.5 2015 ASME Boiler and Pressure Vessel Code Section II, Part D
Table TCD for Mn – ½ Mo – ½ Ni (SA533GrB)

5.6 ASME Code Case N-638-4 « Similar and Dissimilar Metal Welding Using Ambient Temperature Machine GTAW Temper Bead Technique »

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APPENDIX A: ALTERNATIVE WAIT TIMES BASED ON INITIAL HEAD TEMPERATURE

Maximum Initial Head Temperature (°F)	Average Wait Time (min)	Temperature Change Per Pass (°F)	Total Temperature Increase (°F)	Total Process Time (hours: min)

ATTACHMENT 5

AREVA Inc., Affidavit for AREVA documents, March 8, 2016

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Morris Byram. I am Manager, Product Licensing, for AREVA Inc. (AREVA) and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA to determine whether certain AREVA information is proprietary. I am familiar with the policies established by AREVA to ensure the proper application of these criteria.

3. I am familiar with the AREVA information contained in the following AREVA Engineering Information Records: 51-9234023-002, "Corrosion Evaluation of Byron Units 1 and 2 and Braidwood Units 1 and 2 IDTB Weld Repairs," dated March 8, 2016; and 51-9245035-002, "Ambient IDTB Weld Interpass Temperature Evaluation," dated February 5, 2016, and referred to herein as "Documents." Information contained in these Documents has been classified by AREVA as proprietary in accordance with the policies established by AREVA Inc. for the control and protection of proprietary and confidential information.

4. These Documents contain information of a proprietary and confidential nature and are of the type customarily held in confidence by AREVA and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in these Documents as proprietary and confidential.

5. These Documents have been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in these Documents be withheld from public disclosure. The request for withholding of proprietary information is

made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by AREVA to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA, would be helpful to competitors to AREVA, and would likely cause substantial harm to the competitive position of AREVA.

The information in these Documents is considered proprietary for the reasons set forth in paragraphs 6(b), 6(c), and 6(d) above.

7. In accordance with AREVA's policies governing the protection and control of information, proprietary information contained in these Documents has been made available, on a limited basis, to others outside AREVA only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

Morris S. Byram

SUBSCRIBED before me this 8th

day of March, 2016.

Sherry L. McFaden
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 10/31/18
Reg. # 7079129

SHERRY L. MCFADEN
Notary Public
Commonwealth of Virginia
7079129
My Commission Expires Oct 31, 2018