Enclosure
Relief Request 52 Proposed Alternative in Accordance with
10 CFR 50.55a(a)(3)(i)

ATTACHMENT 2

Corrosion Evaluation for Palo Verde Unit 3 Reactor Vessel BMI Nozzle
Modification

A
AREVA

AREVA NP Inc.

Engineering Information Record

Document No.: ___51___ - __9214650__ - __000___

Corrosion Evaluation for Palo Verde Unit 3 Reactor Vessel BMI Nozzle Modification – Non-Proprietary Version

∆ AREVA

Corrosion Evaluation for Palo Verde Unit 3 Reactor Vessel BMI Nozzle Modification – NonProprietary Version

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

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000	All	Original Release. The corresponding proprietary version is in AREVA document 51-9213061-001.

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1.0 PURPOSE

AREVA NP has designed a modification for the Palo Verde Unit 3 (PV-3) reactor vessel bottom mounted instrument (BMI) nozzle [1]. The purpose of this document is to evaluate the potential for material degradation for the repair configuration. The evaluation will focus on 1) the low alloy steel reactor vessel bottom head exposed due to the repair, 2) the new pressure boundary Ni-base alloy materials, and 3) the affected stainless steel materials (i.e., guide tubes near the modification and the potential coupling, and socket welds). For the exposed low alloy steel, types of potential material degradation that will be considered include general corrosion, crevice corrosion, stress corrosion cracking (SCC), hydrogen embrittlement, and galvanic corrosion. The evaluation of the new Ni-based alloy components will focus primarily on potential primary water stress corrosion cracking (PWSCC) susceptibility. The focus of the evaluation of the affected stainless steel materials will be potential SCC susceptibility. The corrosion evaluation will be performed for a life that will span the remainder of the 60-year license (~34 years, but 40 years will be used for conservatism).

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2.0 ASSUMPTIONS

Justified Assumptions

1. For the AREVA process of calculating the corrosion rate (Section 5.2.3), [

]

Assumptions Requiring Verification

1. PV-3 will be shut down [   ] of the time for the remainder of the plant licensed operation.
2. [

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3. It is assumed that PV-3 will conduct appropriate plant chemistry reviews and demonstrate that a sufficient level of hydrogen overpressure has been implemented for the reactor coolant system, and that the contaminant concentrations in the reactor coolant have been typically maintained at levels below 5 ppb dissolved oxygen, 150 ppb for halide ions, and 150 ppb for sulfate ions [

].

4. It is assumed that, during the outage in which the modification is performed, the licensee adopting this evaluation's arguments regarding the potential for stress corrosion cracking (SCC) of the exposed low alloy steel will need to review their plant-specific reactor coolant system coolant chemistry histories over the last two operating cycles for PV-3, and confirm that these conditions have been met over the last two operating cycles.

3.0   BACKGROUND

During the 2013 inspection of the Alloy 600 BMI nozzles on the outside surface of the reactor vessel bottom head at PV-3, potential reactor coolant leakage was identified in the annulus between the in the reactor vessel bottom head penetration and the BMI nozzle [1]. This leakage was observed in at least one penetration.

The original BMI nozzle configuration (Figure 3-1) [4] consists of an Alloy 600 nozzle, which is welded to the reactor vessel bottom head inside surface with Alloy 82/182 weld material [1].

AREVA will utilize the half-nozzle approach to modify the BMI nozzle configuration at the Penetration #3 location, as shown in Figure 3-2 [4]. The process will consist of removing the existing in-core instrument guide tube to BMI nozzle weld, removing a portion of the existing BMI nozzle below the reactor vessel bottom head, the machine application of an Alloy 52M temper bead weld pad on the outer surface of the reactor vessel bottom head around the penetration, replacing the removed portion of the BMI nozzle with an Alloy 690 nozzle [5], attaching it to the weld pad with a J-groove partial penetration weld, and attaching the in-core instrument guide tube with a socket weld using Alloy 52M weld material [1,4]. The guide tube may also be severed below the BMI nozzle to facilitate modification activities (for the purpose of this corrosion evaluation, it is assumed that it will be necessary to sever the lower portion of the guide tube). During reinstallation, the guide tube segment will be installed and the two tube ends will be joined with a socket welded coupling [1,4]. The guide tube segment above the coupling will be new material (Type 316), the guide tube segment below the coupling will be the original guide tube (Type 316), the coupling material will be new (F316), and the coupling socket welds will be ER316/316L [1].

Figure 3-1: Original BMI Nozzle Penetration #3 Configuration [1,4]

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Figure 3-2: Modified BMI Nozzle Penetration #3 Configuration [1,4]

4.0   INDUSTRY 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 pressurized water reactor (PWR) reactor coolant system (RCS) primary coolant are clad with either stainless steel or nickel-base alloy in order to prevent 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 RCS primary coolant due to damage to the cladding or repair configuration. Several such instances are listed below:

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 – Several tubes were separated within the tubesheet due to SCC, and repaired via explosive expansion. The repair exposed a small area in the tubesheet penetrations to primary coolant. (LER 289-1990-005)

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

1991   Oconee Unit 1 steam generator – A miss-drilled tubesheet hole in the upper tubesheet of one of the steam generators, during a 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 fabricated from Alloy 690. An axial gap approximately [   ] wide exists between the two nozzle sections where the carbon steel base metal is exposed to the primary coolant. The hot leg nozzle containing this repair was removed and the exposed carbon steel examined.

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   Saint 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 1B once through steam generator (OTSG). Subsequent inspection revealed [   ] circumferential damaged area to the cladding surface of the manway opening.

2001   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, 287-2001-003, 302-2001-004, 289-2001-002, 280-2001-003)

2002   CRDM/CEDM repairs at Millstone Unit 2, Oconee Unit 1 and Oconee Unit 2. (LER 269-2002-003, 270-2002-002)

2003   AREVA NP IDTB CRDM/CEDM repairs at St. Lucie Unit 2 and Millstone Unit 2, AREVA NP half nozzle repairs of STP-1 bottom mounted instrument nozzle, AREVA NP half nozzle repairs of pressurizer lower level nozzles at Ringhals Unit 4 and Crystal River Unit 3. (LER 389-2003-002, 498-2003-003, 302-2003-003)

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

2008   AREVA NP performed an IDTB half-nozzle modification to the Calvert Cliffs Unit 1 pressurizer heater penetrations (LER 317-2008-001)

2013   AREVA NP performed an IDTB half-nozzle repairs to the Harris reactor vessel head 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 roughly 30 years with the base metal exposed.

5.0   CORROSION OF LOW ALLOY STEEL EXPOSED TO RCS

Several types of corrosion can potentially occur when carbon and low alloy steel base metal are exposed to primary coolant. During operation, the primary coolant design temperature is 650°F [1]. [

] During shutdown, the primary coolant temperature is at 70-100°F and may become aerated. The following subsections discuss the potential corrosion mechanisms for the exposed low alloy steel base metal identified in Figure 3-2.

5.1   Methodology

The AREVA methodology used to evaluate the potential for corrosion of the exposed low alloy steel due to this modification is as follows:

- Wastage
  - [

                                                      ]

- General Corrosion
  - [

                             ]

- Crevice Corrosion
  - [

                                              ]

- Galvanic Corrosion
  - [

           ]

- Stress Corrosion Cracking
  - [

                                              ]

5.2 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, Alloy 600, and Alloy 690 materials and their equivalent weld metals) are virtually immune to general corrosion from exposure to PWR primary coolant. Carbon and low alloy steels, however, may be subject to general corrosion upon exposure to primary coolant. The general corrosion rates of carbon and low alloy steels during aerated and deaerated reactor coolant conditions are discussed below.

5.2.1 General Corrosion Data

Many studies [6,7,8,9,10,11,12,13,14] have reported corrosion rates of carbon and low alloy steels in high temperature water. In many studies, the corrosion rates for carbon and low alloy steels have been observed to be similar; the corrosion data [7,8 ,11,13,14] includes carbon and low alloy steels such as ASME SA-212 and SA-302 Grade B. The material of the PV-3 reactor vessel bottom head [   ] is essentially equivalent to SA-302 Grade B, except that [   ] also includes some nickel. Nickel has no deleterious effect on the general corrosion of low alloy steel, so the referenced corrosion data is considered applicable to the PV-3 reactor vessel bottom head. The Electric Power Research Institute (EPRI) has also compiled a handbook [15] on boric acid corrosion. 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.

One evaluation was completed in response to the Yankee-Rowe incident (noted in Section 4.0); in the evaluation, A-212 carbon steel was exposed to primary coolant in aerated and deaerated conditions [16]. It was shown that under deaerated conditions (i.e., during operation), the corrosion rate depended on temperature, fluid velocity, boric acid concentration, and time. At the maximum velocity tested (36 ft/sec.), the corrosion rate was determined to be 0.003 ipy (inch per year) – the maximum corrosion rate reported [16]. Under low flow or stagnant conditions at 650°F, a maximum corrosion rate of 0.0009 ipy was reported. In the same study under 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 [16]. The primary coolant at the location of interest is expected to be [

] during both operation and shutdown periods, resulting in [   ] general corrosion rates.

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5.2.2   Pressure Boundary Leakage (Wastage)

For the primary coolant inside the modified BMI nozzle penetration there is no mechanism for boric acid to concentrate due to the new pressure boundary material removing the leak path. [

] As indicated in the preceding section, test data of low alloy steel exposed to borated water similar to primary coolant shows very slow corrosion rates relative to wastage, which can be greater than one inch/year [15]. [

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5.2.3   General Corrosion Rate (AREVA Process)
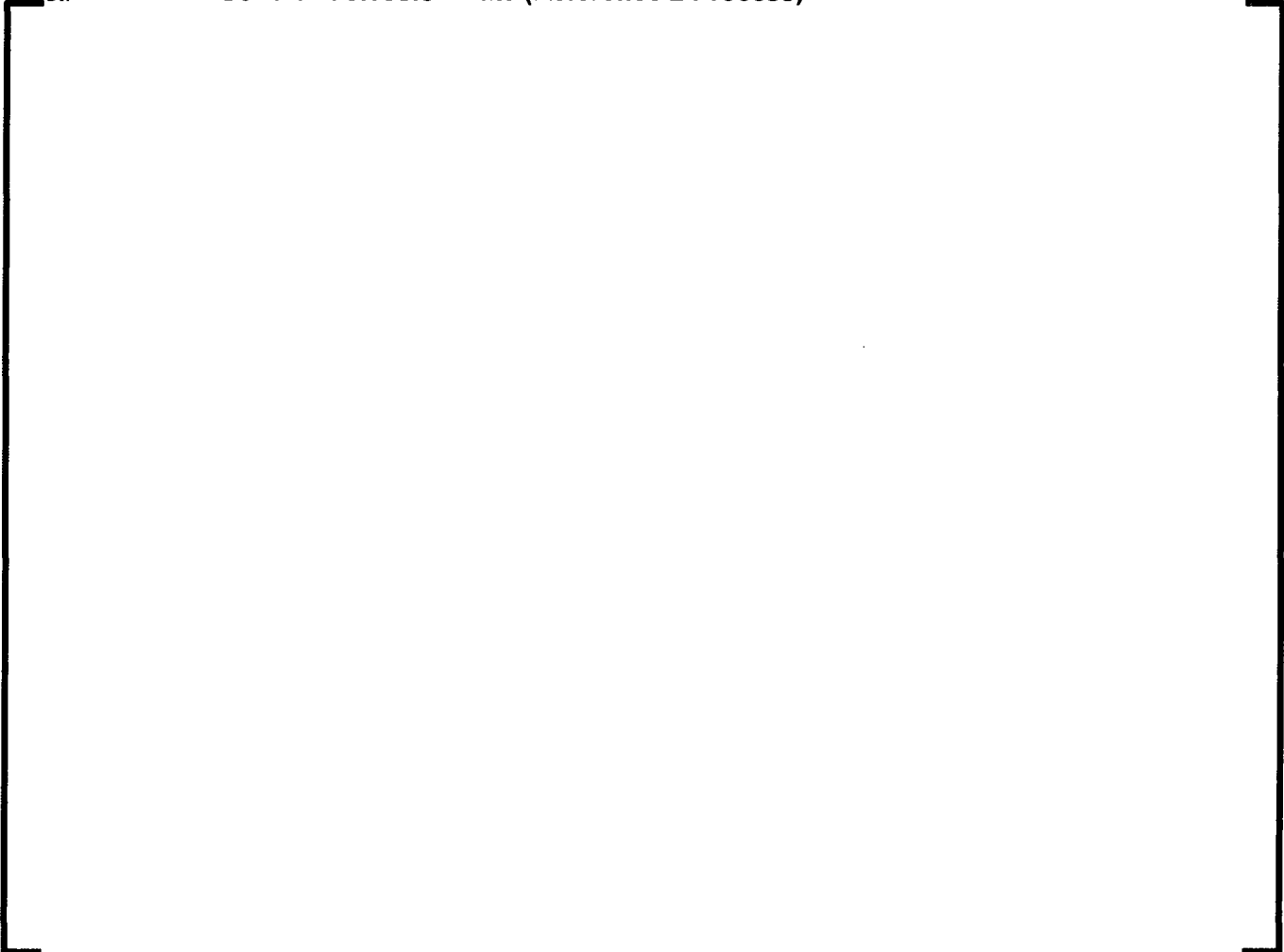
[

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Table 5-1:   [   ] **Average Corrosion Rate** [   ]

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5.2.4   **General Corrosion Rate (Reference 2 Process)**

5.2.5   **Recommended General Corrosion Rate**

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5.2.6 Long Term General Corrosion
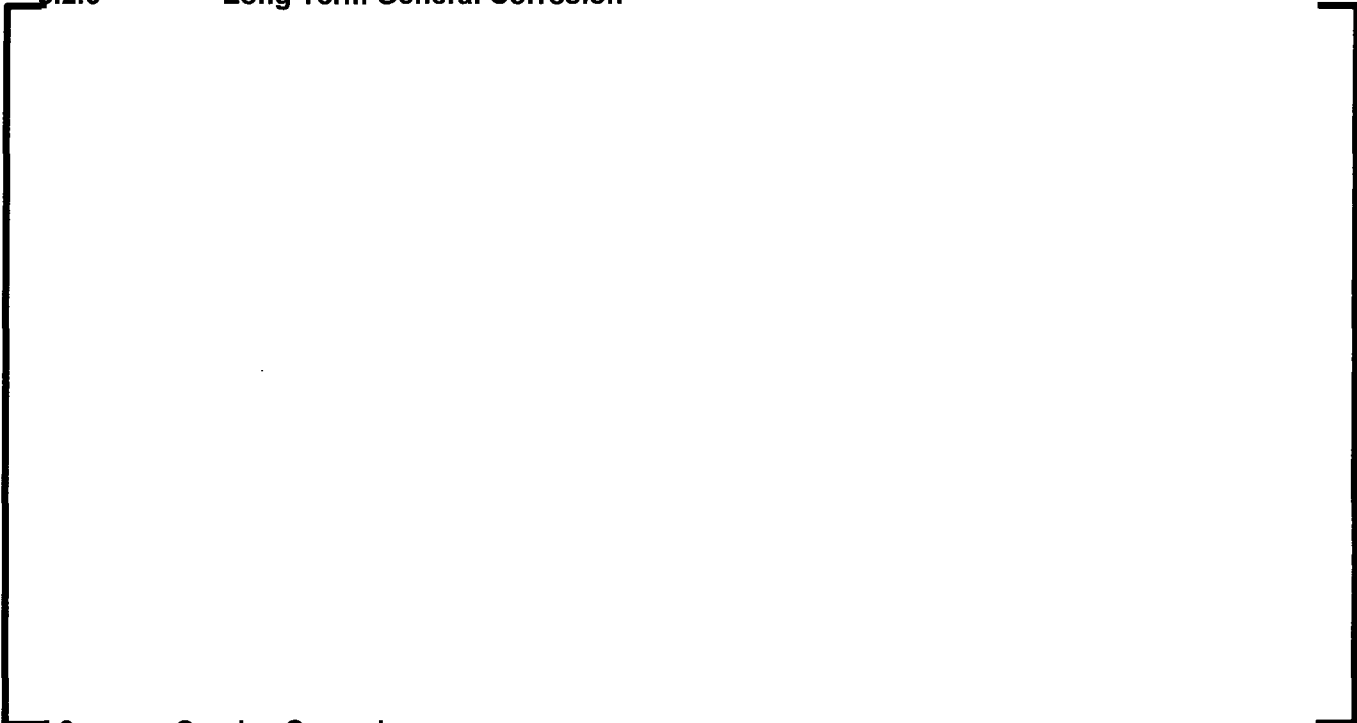
5.3 Crevice Corrosion

The environmental conditions in a crevice can become aggressive with time and can cause accelerated local corrosion. Experiments were conducted to study crevice corrosion of low alloy steel. The results showed no evidence of increased corrosion in the crevices for both aerated and deaerated conditions (i.e., crevice corrosion could not be distinguished from general corrosion) [11,16]. [

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5.4 Galvanic Corrosion

Galvanic corrosion may occur when two dissimilar metals in contact are exposed to a conductive solution. The larger the potential difference between the two metals, the greater the likelihood of galvanic corrosion. Low alloy steel is more anodic than nickel-based weld metals (Alloy 52M) and could therefore be subject to galvanic attack when coupled and exposed to reactor coolant, as shown in Figure 3-2.

Several corrosion tests were performed to determine the influence of coupling between low alloy and austenitic stainless steel, which has approximately the same corrosion potential as nickel-based alloys. 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 [6]. 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 [14]. In each of the tests described above, corrosion rates were not affected by coupling.

The results of the NRC's boric acid corrosion test program at Argonne National Laboratory have shown that the galvanic difference between A533 Grade B Steel, Alloy 600, and 308 stainless steel (used in reactor pressure vessel cladding) is not significant enough to consider galvanic corrosion as a strong contributor to the overall boric acid corrosion process [18].

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5.5 Stress Corrosion Cracking

Stress corrosion cracking (SCC) can occur only when the following three conditions are present: (1) a susceptible material, (2) a tensile stress, (3) and 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 over the past two decades has also been on SCC. An excellent review of the work conducted through 1990 appears in a paper by Scott and Tice [19]. A more recent review of the relevant laboratory work and field experience appears in a report prepared for the CE Owners Group [20]. The conclusions of both evaluations are the same, i.e., 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 [21,22,23,24].

Extensive PWR [   ] and BWR [24] 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 pressure vessel material to any significant depth. Noteworthy are SCC cracks revealed in the stainless steel cladding in charging pumps [25,26]. The interdendritic cracks, present in the cladding, were determined to have blunted at the clad/low alloy steel interface.

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5.6 Hydrogen Embrittlement

Hydrogen embrittlement in low alloy steels results from excessive amounts of hydrogen in a metal's crystal lattice. This type of damage is a mechanical/environmental failure process, which usually occurs in combination with residual or applied stresses. Hydrogen embrittlement is observed most often in plastically deformed metals or alloys in high pressure hydrogen environments and is characterized by ductility losses and lowering of the fracture toughness. High pressure hydrogen environments are not typical of PWR systems and are defined as an environment with approximately 5,000-10,000 psi [27]. Hydrogen exists within the reactor coolant system (used as an oxygen scavenger) and is expected to accumulate at locations such as the top of the pressurizer. Hydrogen-induced mechanical property changes are more pronounced in high-strength low-toughness steels. Low-strength high-toughness steels such as SA-212, SA-302, or SA-533 are little affected by hydrogen in this manner [6]. In the case of a PWR reactor vessel under normal operation, the corrosion rate and diffusion rate of hydrogen through the vessel wall are such that the maximum hydrogen concentration in the steel of the lower head is approximately 0.3 ppm [6]. In either case, the quantity of hydrogen that may accumulate at locations within the coolant system is not expected to induce hydrogen embrittlement in materials at those locations.

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6.0 PWSCC SUSCEPTIBILITY OF ALLOY 690 AND ALLOYS 52M

Alloy 52M is a modified version of Alloy 52 for improved weldability, but this variation is similar to Alloy 52 in terms of the PWSCC resistance properties discussed below [28]. The corrosion resistance of Alloy 690 and weld metal Alloys 52 and 152 has been extensively studied as a result of numerous PWSCC failures in mill annealed Alloy 600 and weld metals Alloys 82 and 182 in primary water environments. As a result of Alloy 600 and Alloys 82 and 182 failures, Alloy 690 and Alloys 52 and 152 have been chosen by the nuclear power industry as the replacement material of choice for Alloy 600 and Alloy 82 and 182 components and welds in PWRs.

Reference 29 provides a comprehensive review of testing for the use of Alloy 690 and Alloy 52 in PWRs, citing numerous investigations and test results under a wide array of conditions, including both primary (high temperature de-oxygenated water) and secondary coolant environments. The first Alloy 690 steam generator went on-line in May 1989 with no reported failures of tubes due to environmental degradation as of an August 1997 literature review [30]. To this date there are no known reports of Alloy 690 in-service PWSCC failures.

The typical test conditions cited in the Alloy 690 literature review included temperatures up to 365°C (689°F), dissolved oxygen levels < 20 ppb, in doped and undoped 400°C (752°F) 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 288°C (550°F), even in a creviced situation. Only slight intergranular cracking of Alloy 690MA (mill annealed) was observed in slow strain rate testing (SSRT) in 360°C (680°F) high purity deaerated hydrogenated water. However, the same cracking was also observed in tests under argon, so the cracking observed in high purity deaerated hydrogenated water could not be confirmed to be PWSCC [30].

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 [30]. CERT (constant extension rate test) data is available for specimens fabricated from weld mockups made from Alloys 690, Alloy 82 and Alloy 152 and tested under normal and faulted (addition of 150 ppb chloride) primary water conditions. The water chemistry was typical primary water: 2.2 ppm Li, 1000 ppm B, 6.5 pH, > 20 cc/kg hydrogen. Neither the weld mockups nor the CERT specimens were stress relieved after the Alloy 152 weld prior to testing. The test temperature was 343°C (650°F), with tests performed at a constant strain rate of either $1x10^{-6}$ or $5x10^{-8}$ sec^{-1}. No evidence of SCC failures was observed on any specimens tested at the faster strain rates, only dimpled rupture fracture surfaces indicating ductile failure were reported. The results of the slower strain rate (~10^{-8} sec^{-1}) tests did show evidence of SCC failure in the Alloy 82 weld metal, but not in Alloy 690 or Alloy 152. These tests demonstrate the high resistance of Alloy 690 and Alloy 152 weld metal under the conditions of high stress, which include residual stresses from welding, machining, and high plastic straining of the test specimens [31].

SCC test data comparing results between Alloy 690 and Alloy 600 is available in both aerated and deaerated high temperature water. Test specimens were made from a creviced double U-bend geometry, and were tested for 48 weeks at 316°C (601°F). Various Alloy 690 material conditions were tested, including MA, MA + thermal treatment (TT), MA followed by solution annealing (SA), cold working (cold rolled 40%), and gas tungsten arc welding (GTAW) welded specimens with "matching" filler metal. In tests with an environment of a minimum of 6 ppm dissolved oxygen and pH of 10, the control alloys, including Alloy 600 and Alloy 800 readily cracked, whereas Alloy 690 showed no cracking. Additional tests were carried out under deaerated conditions (~ 20 ppb O_2) where Alloy 690 showed no cracking, while Alloy 600 in the MA + cold roll heat treated condition cracked. Consistent with other investigators previously cited, Alloy 600TT showed a resistance to cracking. Additional tests were carried out at 360°C (680°F) in deaerated water with a pH of 10 as part of the same study. Alloy 690 material test conditions included SA (1010°C [1850°F] and 1120°C [2048°F]) and SA+TT (1120°C [2048°F]) + TT (675°C [1247°F]). Again, no cracking of Alloy 690 was reported [32].

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7.0 SCC SUSEPTIBILITY OF TYPE 316 GUIDE TUBE NEAR ALLOY 52M SOCKET WELD

Stress corrosion cracking requires three synergistic elements to occur: 1) sustained tensile stress, 2) an aggressive (corrosive) environment and 3) a susceptible material. In the case of stainless steel in a reactor environment, aerated water is considered an aggressive environment. Austenitic stainless steels are generally susceptible to SCC in elevated temperature environments where impurities such as halogens (e.g., chlorides and fluorides) and/or dissolved oxygen are present. Cracking can occur as intergranular stress corrosion cracking (IGSCC) or transgranular stress corrosion cracking (TGSCC) under these conditions.

Original Configuration

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[

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Modified Configuration

Based on industry experience, socket welds can be subject to fatigue. This mechanism is not considered environmental degradation and, therefore, is outside the scope of this evaluation.

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8.0 SCC SUCEPTIBILITY OF COMPONENTS NEAR STAINLESS STEEL COUPLING

As shown in Figure 3-2 Detail G, the original guide tube segment (Type 316) will be joined to the new guide tube segment (Type 316) using a coupling (F316) and two socket welds (ER316/316L).

<u>Guide Tube and Coupling Material Adjacent to Coupling Socket Welds</u>

<u>Coupling Socket Welds</u>

9.0 CONCLUSIONS

The information presented above describes the potential corrosion mechanisms that may affect the PV-3 BMI nozzle penetration modification. Based on this evaluation, the modification is found acceptable (as detailed below).

Per industry experience, socket welds can be subject to fatigue. This mechanism is not considered environmental degradation and, therefore, is outside the scope of this evaluation.

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2. [

   ]

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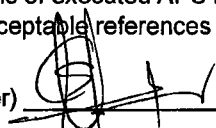
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Maya Chandrashekhar (Project Manager) _____ 11|6|13

APPENDIX A: MATERIAL LOSS DUE TO DISSOLVED OXYGEN CONSUMPTION

A.1 **Results Summary**

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A.2 **Introduction**

Corrosion Evaluation for Palo Verde Unit 3 Reactor Vessel BMI Nozzle Modification – NonProprietary Version

Figure A-1: Estimated [   ] Region near Location A

A.3   Calculation of Material Lost

Corrosion Evaluation for Palo Verde Unit 3 Reactor Vessel BMI Nozzle Modification – NonProprietary Version

Table A-1: [   ]

Table A-2: [   ]