Comparison of Boiling Water Reactor and Pressurized Water Reactor Experience with Cracking of Austenitic Stainless Steel

Prepared for EPRI by:

Joshua K. McKinley
Warren H. Bamford
Andrew M. Ruminski
Thomas L. Meikle
John L. McFadden
Westinghouse Electric Company

PURPOSE

This Enclosure transmits a comparison of boiling water reactors (BWRs) and pressurized water reactors (PWRs). The focus of this comparison is the experience with stress corrosion cracking (SCC) of austenitic stainless steel alloys in the primary coolant of each reactor design and the fundamental chemistry differences that contribute to those differences. This comparison supports the current Nuclear Regulatory Commission (NRC) safety evaluation of the MRP-227, Revision 1 PWR Reactor Vessel Internals Inspection and Evaluation Guidelines.

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1 Introduction and Background

The core shroud in the boiling water reactor (BWR) design and the core barrel in the pressurized water reactor (PWR) design are very similar, in that they are large-diameter, thick-walled cylinders which are largely used to guide the coolant flow through the internals. They are even constructed of essentially the same materials, austenitic stainless steel plate, rolled and welded into a cylinder. Although the function, fabrication, and materials of construction are similar for the two configurations, service experience at PWRs has been very different from that at BWRs. Nonetheless, from a regulatory viewpoint, it could appear to be expedient, albeit technically inaccurate, to apply the more extensive degradation experience with BWRs to PWR components. The goal of this white paper is to clearly identify the differences between the two designs, and to argue that these two components should be treated differently.

Technical arguments will be provided to identify the theoretical basis for the significant differences in cracking susceptibility. Additionally, the service experience in PWRs and BWRs will be reviewed to demonstrate the actual differences observed. These two approaches provide a complementary argument to support different treatments of these BWR and PWR components.

2 Theoretical Basis for the Observed Difference in Degradation between BWRs and PWRs

The large differences between BWRs and PWRs in environmental cracking operating experience clearly indicate some underlying difference in the parameters affecting degradation. The observed cracking in the austenitic stainless steel components of BWRs has been due to either stress corrosion cracking (SCC) or, in the case of irradiated components, irradiation-assisted SCC (IASCC) [1]. PWR austenitic stainless steel components have also experienced SCC and IASCC, but to a much lesser extent [1] [2].

Comparison of the SCC and IASCC behavior in these two reactor designs requires evaluation of the parameters relevant to these cracking mechanisms. Any type of SCC requires three conditions to be met at the same time: 1) a susceptible material 2) subjected to adequate tensile stress 3) in an aggressive enough environment (See Figure 2-1 [3]). Most of these can be dispositioned as being equivalent between the BWR and PWR designs, since both types are light water reactors operating in similar environmental regimes. The biggest difference is the hydrogen overpressure, which scavenges effectively all the oxygen from the PWR water, and virtually eliminates the possibility of SCC in that environment.

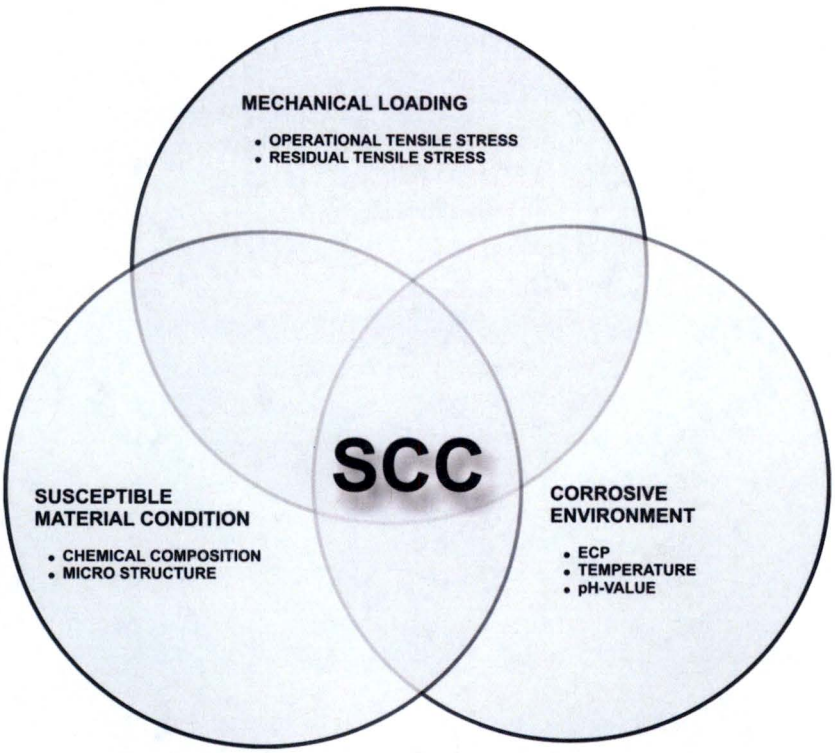

Figure 2-1: Synergistic effects which together lead to SCC

Comparison of Parameters Affecting SCC and IASCC

The three pre-requisites for SCC are shown in Figure 2-1:

- Mechanical loading (tensile)
- Susceptible material
- Corrosive environment

Each of these areas summarizes multiple possible conditions that could impact the occurrence of SCC in a PWR or BWR. Some of these are listed on the circles in Figure 2-1, but those listed on the figure are not exhaustive.

Mechanical Loading

Mechanical loading must be tensile to create SCC or IASCC. Four types of stress are typically present in a PWR or BWR:

- Gravity stresses – due to deadweight loads
- Design mechanical stresses – due to bolt preload, spring forces
- Operational stresses – due to fluid flow, temperature differences, gamma heating
- Residual stresses – due to welding, grinding, or cold working

From a very high level, the basic designs of PWRs and BWRs are the same. Energy is extracted from neutron irradiation using water and transformed into electricity by making steam. The temperature ranges and pressures are similar enough that the same types of materials and similar overall shape and layout of the reactors can be used. Austenitic stainless steel materials are used for most of the reactor vessel internals components of both designs [1] [4] [5], so the same limits on mechanical design parameters are applicable.

The similarities in the two design types mean that both PWRs and BWRs have components subjected to significant deadweight loads and both have components subjected to either bolt preload or spring forces. Operationally, both design types have significant fluid flow, temperature differences, and gamma heating. The gamma heating in certain components of a PWR will actually be higher than that in a BWR due to the higher radiation dose rate on internals components like baffle plates and former plates [3] [4] [6]. Finally, both designs contain the potential for residual stress in large structural welds or components which may have experienced surface grinding or cold work during fabrication.

Based on these potential sources of mechanical loading, it can be concluded that the BWR and PWR designs are essentially equivalent in the potential to have tensile stresses that lead to SCC or IASCC. The higher dose rate in certain localized areas of the PWR may lead to those components having slightly higher susceptibility. Overall, there are plenty of locations in each reactor design with enough tensile stress to support environmental cracking. This has been borne out in the operating experience to date [1] [2] [7, Appendix A], and discussed in Sections 3 and 4 of this report.

Susceptible Material

The focus of this white paper is on evaluating the differences between the environmental cracking experienced in the austenitic stainless steel components of the BWR and PWR reactor vessel internals. Austenitic stainless steel makes up the majority of the components in both reactor designs. Most of the components are fabricated from Type 304, with Types 316 and 347 making up nearly all of the rest [4] [5]. The materials specified for both reactor design types are governed by Section II of the ASME Boiler and Pressure Vessel Code [8]. Thus, large differences in material composition, heat treatment, and mechanical properties are not expected.

Cold work has a strong detrimental effect on the resistance to SCC of austenitic stainless steels in both BWRs and PWRs [2] [9]. Bulk cold work has been limited by the requirement to keep yield strength below 90 ksi under Code Case N-60-6 [10] and Regulatory Guide 1.84 [11]. All nuclear equipment vendors have processes in place to limit the cold work present in operating plant components. Such limitations are standard practice, and the ASME Boiler and Pressure Vessel Code also contains requirements to limit the amount of cold work in components. The presence of cold work in PWR reactor vessel internals was also summarized in report PWROG-15105-NP [12].

ASME Section III Division I Subsection NH [13] provides the following guidance regarding cold forming of reactor components subsequent to solution anneal and thermal treatment and is applicable to the fabrication of PWR core barrels and BWR core shrouds:

NH-4212 Effects of Forming and Bending Processes

The rules of this paragraph shall supplement those of NB-4212 and NB-4213. Any process may be used to form or bend pressure retaining materials, including weld metal, provided that the requirements of the subparagraphs below are met.

(a) Post fabrication heat treatment [in accordance with (b) below] of materials which have been formed during fabrication, shall be required unless one of the following conditions are met.

(1) Maximum fabrication induced local strains[10] do not exceed 5%[11], regardless of the service temperature.

(2) Written technical justification shall be provided in the Design Report for not performing heat treatment, subsequent to straining, or for the use of an alternate heat treatment procedure, to that specified in (b) below for fabrication induced strains greater than 5%.

Notes:

10. *Strain is defined as the maximum local fiber elongation or contraction per unit length; and where more than one strain increment occurs (e.g., biaxiality or reversed bending), it shall be the sum of the absolute values of all the strain increments.*
11. *Strain resulting from final straightening operations performed on materials furnished in the solution annealed or heat treated condition need not be included in the computation of strain.*

The commentary in note 11 above regarding straightening would apply to the as-received plate to fabricate the core barrel, and not to the final as-welded component. As such, manufacturers would need to limit maximum bending strain to 5%. The strain requirement should be calculated based on the actual post-weld distortion values, but is not expected to approach this 5% limit.

In addition, material specifications that govern the use of stainless steel materials generally refer to ASME B&PV Code Section II material specifications with the requirement that material is to be furnished in the solution-annealed condition. These specifications also typically require supplemental thermal treatments for base metal materials with cold work.

Cold straightening of base metal materials is generally prohibited. Cold bending operations performed on these materials are required to be followed by a stress relieving heat treatment. Repairs performed on these materials are generally followed by surface conditioning to remove residual surface tensile stresses resulting from the repair. It is also expected that field fit-up and auxiliary processes that could introduce cold work would only introduce minor increases, with the most significant increases in cold work being local and limited to the material surface [12].

Limiting cold work based on these requirements substantially reduces the risk for cold work-induced SCC [3]. This only leaves potential fabrication-induced cold work as a possible concern. This would include such things as surface repairs and temporary supports welded to the surface and removed after transport. There is evidence that repairs and temporary supports are both applicable to the BWR reactor vessel internals, where alignment lugs and transport "spiders" were welded to the core shroud for use during transport and installation [14]. Similar records have not been found for PWR reactor vessel internals; however, the lack of cracking events in PWRs has not provided motivation for an exhaustive search of fabrication records. Logically, one would expect that supports may have been required during the transport of large PWR internals components like the core barrel and that surface repairs involving grinding were almost certainly used.

Sensitization of austenitic stainless steel materials used in BWRs was a cause for early cracking observed in components exposed to reactor coolant [1]. Sensitized materials are also expected in the older PWR plant components, but experience to date has shown that the low oxygen environment of PWR primary coolant has not caused SCC due to sensitization [1].

Based on the similarity in the materials used and the general design and fabrication similarities, it can be concluded that there are no general material trends that explain the differences in BWR and PWR material degradation.

Corrosive Environment

The general lack of significant differences between BWR and PWR designs when considering stress and material leaves the corrosive environment in the reactor coolant as the likely culprit for the more aggressive environmental cracking behavior observed in BWRs. The effects of coolant chemistry have been studied at length for both reactors, and several parameters govern the impact of SCC or IASCC:

- Temperature
- Presence of impurities
- Electrochemical potential and the presence of hydrogen or oxygen
- pH and the presence of buffering agents
- Neutron irradiation dose (IASCC only)

The propensity for SCC and IASCC are directly related to the temperature, increasing with higher temperature and decreasing with lower temperature [1] [15]. This effect is the same for both PWRs and BWRs. When evaluating the potential for crack growth, a representative temperature of 288°C (550°F) can be used for a BWR, while a representative temperature of 325°C (617°F) can be used for PWR primary water [15]. In reality, the temperature varies throughout the coolant system in both cases, but this difference illustrates that PWR reactor vessel internals components are generally exposed to higher temperature in the hottest parts of the system than BWR components. This temperature difference results in a calculated factor of 5.6 higher crack growth rate for PWR conditions in the hottest locations over BWR conditions with hydrogen additions [15].

The reactor coolant in both BWRs and PWRs is well-controlled for potential impurities. The BWRVIP BWR water chemistry guidelines govern the purity of BWR coolant [16] and the EPRI PWR Primary

Water Guidelines govern the purity of PWR coolant [17]. Both of these guidelines control the presence of cation impurities like chloride and sulfate. The BWR guidance also has control parameters for coolant conductivity and guidance on other parameters that should be measured, such as dissolved metals, electrochemical potential (ECP), oxygen, and hydrogen. The PWR guidance includes additional control parameters for the measurement of fluoride, lithium, hydrogen, and oxygen. The guidelines provide a detailed technical basis for the selection of each of the required control parameters, which are typically aimed at reducing radiation fields or avoiding environmental cracking and corrosion. Earlier plant operation may not have targeted or achieved the tight control levels for impurities specified in the modern guidance, as shown for earlier BWR operation [18]. During prior operation, BWRs and PWRs could have also experienced off-normal chemistry for brief periods due to unexpected conditions like resin ingress. Modern management of the water chemistry has trended toward significantly better control, as shown for BWRs in [18]. Management according to the guidelines reduces the overall likelihood of SCC and IASCC due to impurities.

The BWR and PWR water chemistry guidelines [16] [17] provide guidance for the presence of dissolved oxygen and hydrogen in the reactor coolant. Oxygen is an oxidizing agent and hydrogen acts as a reducing agent. Practically, this means that metals will be more likely to form oxides and hydroxides in an aqueous environment with excess oxygen, for example Fe_2O_3 or $Fe(OH)_2$. In an environment with excess hydrogen, the less oxidized compounds and the base metals will be stabilized. The driving force for oxidation or reduction can be measured directly through the ECP of the system. A more positive ECP indicates a more oxidizing environment, and a more negative ECP indicates more reduction. In a typical BWR coolant environment, the threshold for SCC or IASCC is approximately -230 millivolts (mV) versus a standard hydrogen electrode (SHE) [1] [16] [19]. PWRs operate with between 25 and 50 cc of H_2/kg of H_2O (at standard temperature and pressure [STP]) [17], which keeps the ECP near approximately -770 mV [20]. This typical ECP in PWR primary water is well below the threshold level and far below the ECP of a BWR running with normal water chemistry. To address the issues associated with the high ECP of normal water chemistry (NWC), BWRs have gradually moved into using other chemistry management approaches. There are three broad BWR chemistry management approaches: NWC, hydrogen water chemistry (HWC), and noble metal chemistry.

The first type of chemistry operation in BWRs was NWC, and HWC and noble metal chemistry were added as options later to address specific issues [16]. Within the past decade, NWC was still used at some plants [21]; though all of the U.S. domestic BWRs had converted to HWC or later methods of ECP control [22]. Under NWC operation, a BWR uses high-purity water with no solid, liquid, or gas purposely added to the coolant [16]. Neutron radiation causes radiolysis of the water, creating oxidizing species that raise the ECP of NWC coolant to approximately +200 to +250 mV_{SHE} [23]. Issues with SCC led to the introduction of HWC at many plants. The addition of 0.3 to 2.0 ppm of hydrogen to the feedwater achieves SCC mitigation with some margin past the ECP threshold value, typically reducing the ECP by 500 mV_{SHE} [23]. The downside of HWC is that it causes a significant increase in the main steam line radiation and it also requires significantly more hydrogen addition than one would calculate based on what is needed at the metal surface. These issues led to the creation of noble metal chemistry [16]. Noble metal chemistry increases the efficacy of the hydrogen additions by treating the coolant system surfaces with a catalyst that lowers the activation energy for the hydrogen to oxygen

recombination reactions. This lowers the amount of hydrogen addition required and as a bonus lowers the radiation activity in the main steam line of a BWR.

EPRI published an extensive review of IASCC crack growth in [15]. This report showed that IASCC crack growth rates in austenitic stainless steel materials under light water reactor conditions can be divided into two categories. BWR NWC conditions made up one category and exhibited higher crack growth rates. BWR HWC and PWR chemistry conditions were lumped together in the other category and together exhibited lower crack growth rates. Similar conclusions are expected to apply to both the initiation of cracking and SCC without irradiation effects.

BWRs and PWRs manage the pH of the reactor coolant quite differently. BWR coolant is high-purity water with no deliberate additions to control pH [16], because the boiling action would result in a lack of chemistry control within the reactor core. PWR coolant is high-purity water with additions of boron as boric acid and lithium as lithium hydroxide [17]. The boron is used as part of the core reactivity control and lithium is added to maintain the pH neutrality of the coolant [1]. The pH of BWR coolant at temperature is approximately 5.65 and that of PWR coolant is controlled to around 7.2 [24]. Testing has been performed at multiple boron and lithium concentrations for PWR coolant resulting in the finding that at low ECP there is little impact from these changes on SCC crack growth [24]. Some recent testing has found that high lithium levels (at least by 8 ppm Li and greater) can lead to increased IASCC initiation rates [25]. This level of Li is higher than that typically used by PWRs, but determination of the effect of lithium concentrations between 2 and 8 ppm has not been completed.

Neutron irradiation dose has an impact on the potential for IASCC degradation. IASCC initiation testing in PWR primary water conditions has shown a strong effect of increasing irradiation dose in decreasing the stress required for cracking when at low doses (< 10 displacements per atom [dpa]) followed by a smaller effect at higher doses [3]. PWR components near the core experience much higher radiation doses than similar BWR components by approximately a factor of 10 [3]. Thus, from a radiation damage standpoint, the PWR components are expected to be more impacted by IASCC. However, components in both reactor types are subject to the strong dose dependence below 10 dpa, and the PWR components that experience radiation beyond 10 dpa do not experience the same strong increase in susceptibility with the additional dose.

Distinguishing which corrosive environment is more aggressive based on the foregoing discussions is difficult because of the presence of competing effects. The PWR environment has higher temperature and dose, which would lead to a higher propensity for SCC or IASCC. A BWR running with NWC conditions would have a higher likelihood of SCC and IASCC due to the excess oxygen. A BWR running with HWC implemented has been shown to have an IASCC crack growth rate behavior similar to that seen with PWR chemistry and about 4.6 times less than that in NWC [15]. Previous results have shown this to be as high as a factor of 10, while for sensitized unirradiated material the benefit of a lower ECP can be a factor of 20 to 50 [22]. Constant extension rate tensile testing of samples fabricated from the same heat of Type 304 austenitic stainless steel in both BWR NWC and PWR primary water conditions showed that the NWC caused IASCC but the PWR environment did not [20]. Impurity contents within the allowable values from the BWR or PWR water chemistry guidelines should not cause higher cracking risk, but the water purity during past operation is not well-known. The other parameters,

including factors affecting tensile stresses, pH and buffering agents, and material variables, are not expected to differentiate between BWRs and PWRs.

However, the evidence points to the strong impact of the oxidizing environment used in BWRs operating with NWC. This operating regime has a stronger impact on the likelihood of SCC or IASCC than do the higher temperature and dose experienced by materials in PWRs. PWRs have used low ECP for the entirety of their operation, whereas BWRs have operated for varying lengths of time with ECP levels above the thresholds for IASCC or SCC. These periods with higher ECP would have allowed the initiation and growth of cracks in the BWRs. PWRs have not experienced such cracking.

3 BWR Cracking Experience Summary

Intergranular SCC is a commonly observed phenomenon in BWR core shrouds. It was first observed in 1990 at the Mühleberg site in Switzerland [26]. As of 2013, the cracks being monitored in the core shroud of Mühleberg ranged from one inch to over two feet in length and had depths approaching one inch. In response to this, the original equipment manufacturer (OEM) issued an information letter to all owners of BWRs designed by the OEM, recommending a visual examination of the shroud circumferential welds [27]. The NRC responded to the early observations of cracking at several plants by issuing Generic Letter 94-03 to request that all licensees inspect their BWR core shrouds no later than the next scheduled refueling outage [28]. By the mid-1990s, the cracking had progressed to the point that several plants had installed tie rods to bolster the integrity of the core shroud, in the presence of circumferential cracks [26] [27]. By the late 1990s, a significant percentage of operating BWRs had installed tie rods [14] [29].

The tie rods relied on the integrity of the vertical welds of the shroud to maintain their key purpose, but, by 1993 cracking was being observed in the vertical welds as well [29]. Multiple visual and UT exams were conducted after this observation, and these cracks have been observed to progress slowly. Inspection and evaluation guidelines for the BWR core shroud were developed and documented in BWRVIP-76-A [30], which was submitted to the NRC for review, and Revision 1 received a safety evaluation in 2014 [31]. This supported establishment of up to a ten year re-inspection interval for these welds. To date, plants have been able to continue operation with the observed flaws despite increases in the number and size of the flaws.

As these inspections progressed, the number of flaws observed has increased and flaw dimensions have changed. Figure 3-1 illustrates the various types of cracking that have been observed [14]. Cracks oriented along the welds have been commonly observed and were the primary concern related to core shroud integrity when SCC was first identified in the early 1990s. More recently, "off-axis" cracking (i.e., cracks that propagate approximately perpendicular to the associated weld, see Figure 3-1) has become more of a concern as inspections have appeared to reveal changes to the off-axis cracks along with new or deeper than expected cracks. These changes observed during inspections could be due to improvements in inspection techniques and equipment rather than additional crack growth or initiation [14]. Analyses of the BWR core shroud indicate that the off-axis flaws are likely to grow through-wall but are not likely to grow into long cracks due to the stress distribution predicted around the weld [32]. These analytical results are consistent with the majority of the reported off-axis flaws.

Correlation studies have shown that off-axis cracking is most strongly related to plant design, neutron fluence, and shroud fabricator [14]. The location of most of the reported off-axis cracks has been the core shroud beltline region spanning the beltline welds H3 and H6a. Factors introduced during construction such as cold work due to manipulation of the shroud during fit-up or local material changes due to construction supports and alignment lugs that were welded to the core shroud and then removed could have contributed to the cracking. The correlation with neutron fluence is driven by the location of the cracking in beltline region welds (Figure 3-2) and the occurrence of the cracking in older units with more exposure time (Figure 3-3). Some of these correlated factors could be confounded with fabricator or design differences. The cause of this off-axis cracking remains under active investigation, but analytical

results indicate that off-axis cracks should remain short and do not have a significant impact on core shroud structural integrity [32]. To date, this has been confirmed by the results of field inspections.

Many other BWR internals components have also shown evidence of environmental cracking; though the cracking in core shrouds is one of the most widespread [26] and is the most relevant in a comparison to the PWR core barrel. The experience summarized above shows that stainless steel welds operating in BWR environments are rather highly susceptible to stress corrosion cracking. Neutron irradiation appears to increase this susceptibility. The experience for the BWR core shrouds does show evidence that much of the cracking initiated early in BWR plant life due to fabrication factors and the initial exposure to the oxidizing NWC environment. As will be seen in Section 4 below, the same materials operating in PWR environments have not displayed the same level of susceptibility and did not experience the early life degradation.

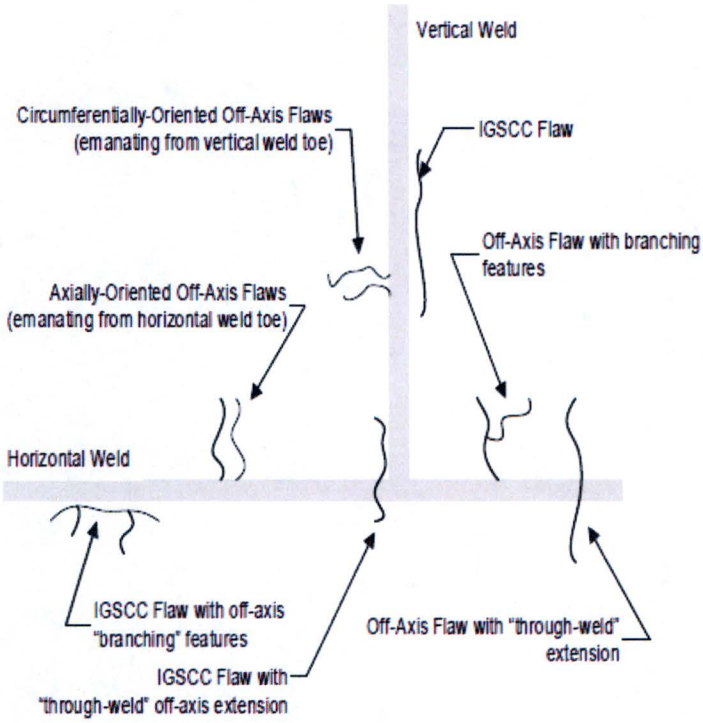

Figure 3-1: Locations and Orientations of BWR Shroud Cracking [14]

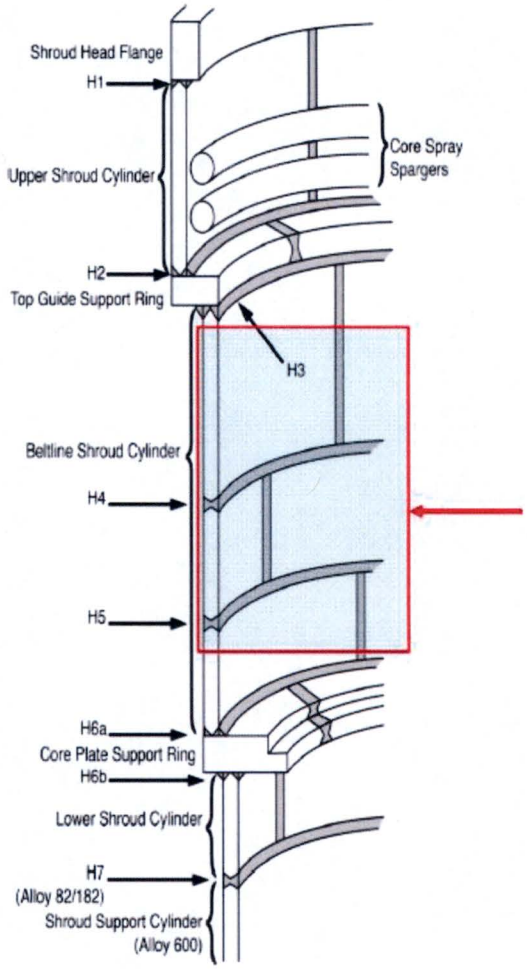

Figure 3-2: Location of BWR Shroud Off-axis Cracking—nearly all off-axis cracking is between weld numbers H3 and H6a, as shown by the arrow. [14]

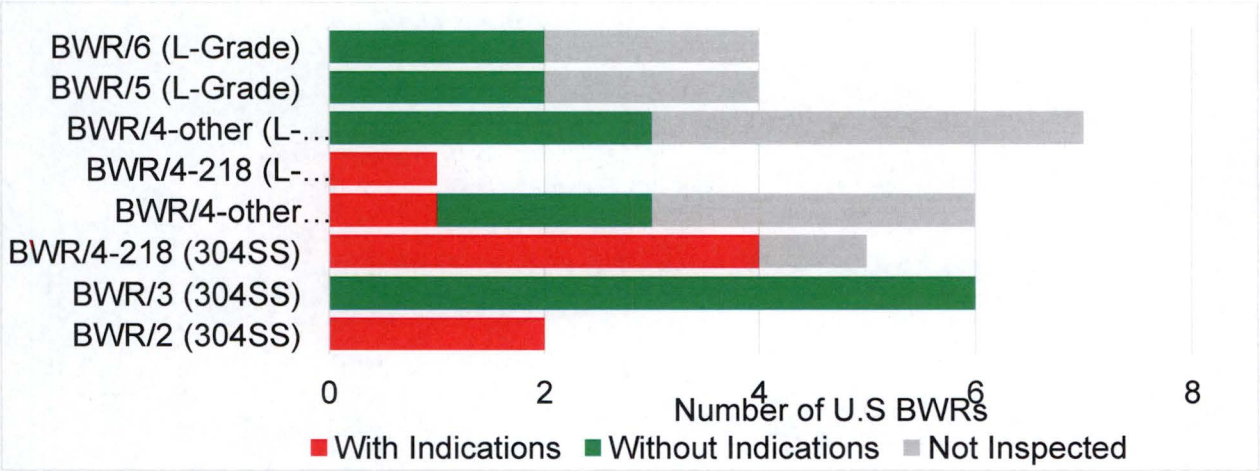

Figure 3-3: Occurrence of off-axis cracking by Plant Design [14]

4 PWR Cracking Experience Summary

Unlike the BWR core shroud experience, observations of cracking in PWR core barrels have been very limited. Two incidents were summarized in the operating experience section of MRP-227-A [7] which occurred in Combustion Engineering-designed PWRs and were due to flow-induced vibration caused failures in the thermal shield. These two events occurred early in plant life, and led to extensive cracks in the core barrels located in the base metal away from the welds. The problem was solved by removing the thermal shield, which eliminated the driving force for the cracks, and drilling holes at the ends of the cracks to blunt the tips and prevent further growth. These cracks have remained in service for over thirty years with no further degradation.

Since the implementation of MRP-227-A [7], multiple enhanced visual (EVT-1) examinations have been conducted on the core barrel welds of both Westinghouse and Combustion Engineering-designed plants [33]. These examinations have included unirradiated welds subject to SCC, such as the upper core barrel flange weld, and irradiated welds in the core beltline that are susceptible to IASCC, such as the Westinghouse lower girth weld. Relevant indications have only been observed in one of those inspections to date. That inspection was of a Combustion Engineering-designed PWR core support barrel and included multiple indications observed by the EVT-1 inspection and confirmed by supplemental volumetric examinations. Plant-specific engineering evaluations in response to this operating experience in accordance with WCAP-17096-NP-A [34] approved methodologies are currently underway. All of the other MRP-227-A inspections of core barrels to date have resulted in no relevant indications.

Historical experience with SCC in austenitic stainless steel exposed to PWR primary water environments was reviewed by Hall and Bamford [35] in 2003 and by Ilevbare et al [2] in 2010. Together, these studies provide extensive documentation of the fact that austenitic stainless steel in PWR primary water environments are not subject to the high incidence of SCC experienced by BWRs. These studies found that 83% of the SCC events occurred in low-flow, occluded, or stagnant locations in the primary system, which are where off-normal chemistry conditions are likely to persist [2]. Only about 17% of reported cracking events occurred in free-flow conditions where normal bulk coolant chemistry is expected conditions, and all of these events were associated with severe cold work. As detailed in Section 2, cold work has been specifically limited in austenitic stainless steel components, which limits the possibility of cold-work induced SCC.

Unlike the experience with BWR core shrouds and other components, environmentally-induced cracking has been very limited in PWR austenitic stainless steel components. This is due to the differences in reactor coolant chemistry between BWR and PWR designs. As described in Section 2, the hydrogen overpressure effectively scavenges all oxygen from the primary coolant system, which creates a low ECP environment in PWR primary water and has prevented SCC degradation of the core barrel to date. Continued operation with a low ECP environment is expected to be an effective strategy for mitigating future SCC in PWR core barrels.

5 Summary and Conclusions

This report has provided a comparison of PWR and BWR Core barrels and shrouds, which are large-diameter austenitic stainless steel structures whose primary purpose is to direct the coolant flow through the reactor vessel core region. The mechanical design criteria and temperatures of operation for these two structures are very similar, but there are major differences in the reactor coolant water chemistry to which they are exposed.

The primary difference is reflected in the electrochemical potential, which measures whether or not the environment is more oxidizing or more reducing. The PWR primary water environment is the most reducing case at approximately -770 mV_{SHE}. This is at least several hundred mV lower than BWR HWC environments and nearly 1000 mV lower than the ECP in a BWR NWC regime. Operation below approximately -230 mV_{SHE} results in mitigation of SCC and IASCC. The beneficial effects of a reducing environment have been explained in Section 2 of this report. The service experience supports these conclusions as well and was reviewed in Sections 3 and 4.

BWR core shrouds, as well as other internals components fabricated from austenitic stainless steels, have experienced significant levels of environmentally-induced cracking, while little cracking has been observed to date in PWR environments and mostly in bolting. Multiple detailed inspections under the requirements of MRP-227 have been completed on PWR core barrels, and only one recent inspection has discovered relevant indications. Since PWRs will operate for their entire licensed lifetimes with low ECP in the primary coolant system due to hydrogen additions, the likelihood of SCC or IASCC initiating in the core barrel is expected to be low.

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