AN ENGINEERING APPROACH TO ASSESS IASCC INITIATION IN AUSTENITIC STAINLESS STEELS OF LIGHT WATER REACTOR INTERNALS

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ABSTRACT

The objective of this paper is to present and discuss an extension of our stress corrosion cracking (SCC) initiation model previously developed for the unirradiated materials in light-water reactor (LWR) service to address irradiation-assisted SCC (IASCC) of austenitic stainless steels for typical reactor internals application. A brief literature review is documented on the effects of irradiation on the relevant mechanical properties and SCC susceptibility of typical austenitic steels in LWR service. The review findings support the adopted modeling approach. The extension of SCC initiation model to IASCC utilizes dpa as a measure of the irradiation damage and essentially consists of two parts: one accounts for the impact of irradiation on SCC susceptibility correlated with changes in the strength properties, and the other accounts for enhanced degradation of micro-cracking resistance and material–environment interaction at higher dpa levels. The parameters of extended model are derived from O-ring IASCC failure data on one heat of Type 316 austenitic stainless steel (flux thimble tubes) in simulated primary water environment. The model response is predicted for two other heats and compared with the data from similar O-ring tests. The IASCC model is shown to be in agreement with IASCC test data on failure times, mostly within a factor of three, and with major trends, particularly for stress, dpa, cold-work, and radiation hardening. The data limitations and model characteristics are discussed with recommendations for future testing and further assessment to address the limitations and to corroborate the model basis.

Keywords: IASCC, initiation model, yield strength, Type 316 stainless steel, hardening, dpa.

1.0 BACKGROUND AND OBJECTIVES

Stress corrosion cracking (SCC) has been recognized as an active degradation mechanism of potential significance to the long-term operation of LWRs, requiring inspections, evaluations, and disposition for several components of austenitic stainless steels as well as nickel based alloys [1, 2]. While the original material selection and design of these components were based on well accepted guidelines and conservative rules, the need for much longer terms of operation and the resulting impact of environmental factors were not fully quantified. Furthermore, major characteristics of the SCC damage morphology in service components—viz., its tightness (starting from zero crack size), tortuosity, and branching—pose a great challenge in its detection and sizing, and in applying the leak-before-break concept, for optimal decision to be made with confidence. The additional challenge for long-term operation in the case of reactor internals is the effects of irradiation that bring significant changes in their mechanical strength properties and deformation (strain-rate) response, both of which are known to be directly related to the kinetics of SCC. These practical aspects increase the importance of understanding and predicting the SCC or IASCC initiation phase including the short crack response in these components.

The types and effects of irradiation on SCC are themselves multiple and varied. These are reviewed briefly in a subsequent section with the objective of providing a rationale for the factors and considerations used in the IASCC model described in this paper. In general, with regard to IASCC, the primary type of irradiation damage (in the non-fissile materials of interest) is due to the cumulative bombardment or flux of energetic particles, typically estimated in terms of the particle fluence. The
resulting irradiation hardening leads to increased mechanical strength, reduced ductility, and increased micro-cracking susceptibility. An additional effect is related to the radiation induced changes in the grain-boundary region defects and/or chemical composition likely to affect the corrosion susceptibility. In many ways these effects are similar, but not necessarily the same at the microstructural level, to that of the influence of cold work in SCC for which a simplified model was described previously [3, 4]. For each of these influences the model has factors analogous to those for SCC susceptibility under unirradiated conditions as discussed previously [3, 4]. Therefore, it is of interest to examine the possible extension of the simplified initiation model to also address the impact of irradiation on SCC susceptibility.

The objective of this paper is to discuss the extension of our SCC initiation model previously developed for the unirradiated materials in LWR service to address the influence of irradiation on SCC of austenitic stainless steels. Presented below are the results of this model extension to IASCC and its assessment in relation to a set of data obtained from a prior comprehensive international test program.

2.0 OBSERVATIONS FROM SERVICE EXPERIENCE ON IASCC

A few observations from reported service failures due to IASCC are briefly summarized mainly from the perspective of identifying factors that seem to be more significant and/or discriminating in relation to the effect of irradiation on SCC. This summary is not intended as a detailed review of the service experience on IASCC which is available in other publications [e.g., 5–7]. In general, given the lack of a definitive physical model of IASCC and relative rarity of IASCC service failures to date, a critical assessment of factors involved in these failures should provide a useful perspective.

One of the earliest such failures, in 1960s, was related to the extensive cracking of fuel cladding made of annealed Type 304 stainless steel under boiling water reactor (BWR) conditions [8]. Neither the irradiated nor unirradiated cladding samples showed any trace of sensitization (carbide precipitation in grain boundaries), as examined by optical and electron microscopes. The IASCC failures were confined to locations of high stress, high irradiation levels, and locally high strains due to fuel–clad interaction [8]. Also, reportedly the failure rate for initially annealed cladding was not statistically different from the failure rate for initially cold-worked cladding. Later, similar IASCC of unsensitized austenitic steel cladding was also observed in the PWR environment [9]. These instances clearly show that the manifestation of SCC under irradiation over time does not require prior cold-work, sensitization, or prior grain boundary carbide precipitation, but does seem to correspond with local deformation, high stress/strain, or low stress but with high fluence, in both BWR and PWR environments.

A more recent assessment of IASCC was reported on a control assembly tube made of 0X18H10T austenitic stainless steel (similar to AISI Type 321), after 30 years of operation in the temperature range of 250–285°C, with fluence of about 3x10^{25} n/m^2 (> 0.1 MeV) just above the weld-joint area where intergranular cracking was observed [10]. However, no cracking was found in the lower base metal area that was subject to higher fluence in this case, and where radiation induced segregation (RIS) was noted. This difference in cracking response was attributed to higher tensile stress reported in the cracking location, leading to their conclusion that the stress factor dominated over the segregation/depletion factor.

3.0 IRRADIATION EFFECTS AND SCC

For typical conditions of LWR designs and operation, the types of irradiation of primary interest to the reactor vessel and internals are the high energy photons (gamma rays) and fast neutrons. At the basic level, the irradiation damage results from their interaction with lattice atoms leading to a cascade of displaced atoms which generates several pairs of interstitials and vacancies, also known as Frenkel pairs. This basic damage process is extremely fast, on the order of a few picoseconds, and includes considerable generation of local heating. That is, the energy of incident fast neutrons is rapidly consumed in a displacement spike
and a thermal spike in the material lattice, which leads to local melting and quenching, followed by short-term and long-term annealing (of these local regions) [11–13]. This local quench-annealing is similar in many ways to that often used for hardening or strengthening during material processing or component fabrication. Obviously, the character and distribution of the resulting damage is different, compared with that of the general quenching, but the type and consequence of the damage are expected to be similar. The other net effect of the above neutron interaction events is that of an increased stored energy within the lattice, as in the case of general quenching or cold-working of a non-irradiated material.

Furthermore, the differences in mobility and diffusion characteristics of the generated vacancies and interstitials lead to subsequent defect clustering (of excess vacancies and interstitials) in the form of faulted and unfaulted Frank loops [11–14], and stacking fault tetrahedra (SFT) [15]. These microstructural rearrangements cause or determine the alterations in material properties at microscopic and macroscopic scales [e.g., 12, 13]. Even such a simplified description of the irradiation process and its major effects of concern indicate the complexity and multitude of possible damage mechanisms and defects, let alone their interactions with other existing defects (dislocations, grain boundaries, precipitates, etc.) in the original and deformed material under stress. However, from an engineering applications point of view, and for performing simulations and useful accelerated testing, it is of interest to define a parameter that characterizes the severity of irradiation for its damaging effects. Since the primary drivers of the above irradiation damage are the net atom displacement type of impact and the stored energy within the material lattice, the displacement-per-atom [16], or dpa,\(^1\) is considered as the parameter to characterize the cumulative dose of irradiation. In addition to being the more physically based parameter, dpa also allows a comparison and use of data from different types of reactors or sources of radiation with widely differing energy spectra on a consistent and more uniform basis than the fluence. Therefore, the use of dpa in this work as the primary correlating variable is justified for quantifying irradiation effects on microstructure, mechanical properties, and IASCC susceptibility.

From a broader review of the literature it is apparent that the underlying mechanism of IASCC is believed to be generally the same as that of SCC; that is, irradiation acts as an accelerant of the SCC process that has been known to result from the interaction of three factors: environment (water chemistry), material (microstructure), and stress/strain (mechanics). Irradiation affects all three factors and the related processes of oxidation/passivation, material hardening, and deformation or strain-rate response. These effects are briefly reviewed below with an assessment in support of the engineering model for IASCC as an extension of the prior model of unirradiated materials that accounted for the hardening due to cold-work [3, 4, and 17]. Other independent reviews on the role of irradiation in SCC may be consulted for additional historic updates and perspectives [e.g., 5, 18–22].

The influence of reactor water chemistry in SCC has been generally, if not primarily, correlated with the electrochemical corrosion potential (ECP) as a measure of the oxidizing tendency and with the ionic conductivity of water, at operating temperature. Radiation has been known to influence the reactor water chemistry mainly via its influence on the water radiolysis that tends to increase the ECP for structural metals, although the net influence is dependent on the main chemistry control regime, such as the normal water chemistry (NWC), hydrogen water chemistry (HWC), or noble-metal water chemistry (NMWC) in BWRs, and the lithium-borated chemistry (with hydrogen overpressure) in PWR primary loops [5]. In this regard, it may be noted that the in-situ ECP and its change are expected to reach a nearly stable value under the normal operation for a specific water chemistry regime that is also tightly controlled for other system-related objectives. As such, the susceptibility and performance of these systems regarding SCC in general may not be significantly altered due to the effect of radiolysis over the time of operation.

\(^1\) dpa represents the expected number of times a lattice atom gets displaced, on average, during the irradiation. For steels under a typical LWR spectrum of neutron irradiation, 1 dpa generally corresponds to the fluence of 6.5x10\(^{24}\) n/m\(^2\) (1 MeV).
The influence of normal BWR and PWR water chemistries on IASCC susceptibility of several austenitic stainless steels (stabilized, non-stabilized, commercial, and high-purity grades) was assessed under active straining with in-reactor tubular samples (in high flux locations for 2 or 3 operating cycles) [23, 24]. From the results of these initiation type tests under active straining, it was concluded that the stainless steels behaved quite similarly, in spite of the large difference in coolant chemistry. In particular, all the austenitic stainless steel variants in both the BWR and the PWR failed at rather low strains (with the exception of a very high purity Type 348 variant); the differing alloying elements or purity levels failed to show significant difference in the IASCC response.

A review of the more recent work [25] noted that although there is an association between grain boundary chromium depletion and IASCC susceptibility in oxygenated BWR coolants (albeit not always evident in the more susceptible materials at high neutron doses), no such detrimental effect was observed in hydrogenated PWR primary water. The review also noted that evidence was obtained indicating that radiation-induced silicon enrichment at grain boundaries would negatively impact the IASCC resistance in both types of LWR coolant [25].

In an earlier assessment of IASCC for the effects of material microchemistry and its alterations, under irradiation of typical austenitic stainless steels in BWR conditions, the factors or processes known to determine the material strength or hardening were noted to have a more significant influence on IASCC than the typical compositional variations in a given class of materials [26]. Some of the more focused investigations (e.g., [27]) concerning the role of bulk concentrations of major and minor alloying elements and impurities on radiation hardening of various austenitic stainless steels also noted, within the relatively wide range of compositions examined, a significant correlation between the yield strength and hardness. That is, the strength and hardening response seem to be the more significant correlating factors for radiation hardening over a wide range of typical compositional variations in a given class of materials.

The cumulative effect of irradiation on microstructure changes the local deformation response and micro-cracking tendency which are expected to increase the susceptibility to SCC. These effects are also reflected in the resulting radiation hardening and associated changes in the material strength. The material hardening in general is a direct result of the development of internal microstructure and density, distribution of various lattice imperfections and disordering. As such, the manner in which these contributory microstructural arrangements are introduced or developed (quenching, cold-working, irradiation, precipitation, etc.) should not matter as much as the final content and distribution of the defects, since the latter will primarily determine the material hardening response generally well correlated with the resulting material strength properties. This expectation, and the influence or correlation of the resultant hardening, via yield strength, has been demonstrated in several SCC studies [e.g., 28–30]. Similarly, in the case of IASCC of austenitic steels in LWRs, the assessment of various factors and correlating parameters led to the consideration that radiation hardening plays an important role in IASCC, with the possibility that RIS may also influence the susceptibility [31]. It was also noted that the hardening response (strength and ductility) of irradiated materials was closely associated with the microstructure, and that the yield strength showed a strong correlation with IASCC response.

The fact that irradiation changes the material properties of strength, ductility, hardening, and deformation response, and that the changes are strongly correlated with, if not directly affected by, the significant microstructural changes due to irradiation, have been noted since the very early days of reactor technology [32–34], confirmed in several later works [35–39] and more recently [40–42]. While similar observations pertain to the effects of cold work in unirradiated materials, at present there is no direct correlation or measure by which the level of cold-work and that of irradiation can be compared or considered equivalent. However, the hardening behavior and deformation response are reflected, at least on an average basis, in the stress–strain behavior in an analogous manner for both the cold-worked and irradiated conditions. This observation is also well supported by test results [e.g., 40, 43, and 44].
Based on the above review and assessment of phenomenological observations it is reasonable to expect
that the original SCC initiation model \[3, 4\] would also be applicable and work as a good framework to
account for the influence of irradiation on SCC in LWR conditions. In addition to the radiation hardening,
there are other accompanying or concomitant changes in the material microstructure due to irradiation.
Also, the current mechanistic understanding and state of knowledge is only partial and qualitative at best.
As such, it is considered both imperative and useful to focus on variables such as stress, strength, and dpa,
judged to be most significant and available in practical applications for an engineering assessment.

4.0 IASCC MODEL

The model for IASCC initiation presented below is developed as an extension of the SCC initiation model
for unirradiated materials previously described \[3, 4, \text{ and } 17\]. As before, for the purposes of model
development and application, the term “initiation” refers to the development of at least one crack with a
depth of 0.4 mm to 2 mm, starting from a nominally defect-free surface. This range of peak depth is also
comparable with that noted in the IASCC failures test data used in this work. The time to crack initiation
is modeled as a function of material parameters that reflect the influence of material hardening, material
condition, water chemistry, yield strength, dpa level, applied stress and residual stress, and includes the
temperature dependence via an activation energy term.

For reference and comparison, the original semi-empirical model for SCC initiation time, \(t_i\), for
unirradiated materials \[4, 17\], is re-stated as the following simplified expression:

\[
(t_i) = \lambda_0 \cdot [(r - 1)(rE / S_y)^{1/3}] \ln[(A - z) / (S/S_y - z)] \cdot F
\]

where, \(\lambda_0\) : material/environment factor including the Arrhenius temperature dependence
\(p\) : hardening sensitivity parameter
\(A\) : micro-cracking resistance parameter
\(z\) : threshold stress severity parameter
\(S\) : effective tensile stress, including residual stress
\(S_y\) : yield strength of the material condition at room temperature
\(r\) : strength ratio = \(S_u/S_y\), where \(S_u\) is the ultimate strength
\(E\) : Young’s modulus of elasticity
\(F\) : Symbol to represent \(\ln[A]/\ln[(A-z)/(1-z)]\)

Note that \(S_u, r, \text{ and } E\) are material property inputs, \(S\) is the total stress input, and \(S/S_y\) is defined as the
stress severity. All the factors/parameters are entered in the above expression as non-dimensional
quantities, except for \(\lambda_0\) expressed in time. Although this parameter could be made non-dimensional, or
normalized by a characteristic time, the time dimension is related to the assumed Arrhenius type relation
that represents a chemical reaction process with rate expressed inversely as time.

The model parameters \(A\) and \(z\) delineate the mostly mechanical regimes of damage from the
environmental influence regime. That is, model parameters \(A\) and \(z\) can be considered as the upper and
lower limits of the stress severity, respectively, beyond which the environmental (corrosion) influence is
not sustainable due to the local stress/strain rate being too high, or too low relative to the prevailing
corrosion (repassivation/oxidation) kinetics, respectively. The parameter \(A\) represents internal micro-
cracking resistance affected by the material hardening that promotes an increase in the local stress, for
example, at intersections of grain boundaries and dislocation pileups during deformation. These two
parameters were shown to be correlated with the hardening/strength properties \[3, 4\]. The use of stress
severity, instead of stress alone, was also justified in the prior work \[3, 4\].
The model parameter $\lambda_0$ does not depend on the material hardening, but it incorporates the influence on repassivation and/or oxidation response related primarily to the environmental variables of corrosion potential and conductivity. The parameter $p$ is also related to the SCC kinetics for the material–environment system and it reflects the material deformation related influence of the SCC susceptibility.

The form of above original SCC initiation model was based on the corrosion–deformation interaction effects, as assessed by the strain rate damage mechanics (SRDM) approach [17, 45]. Since these effects are expected to be applicable under the influence of irradiation as well, this form is retained in extending it to the case of IASCC. Furthermore, the above physical interpretation of the model parameters is maintained. It is to be expected, however, that the interaction kinetics will be enhanced by the irradiation effects. Also, the material strength properties are expected to be affected by the cumulative effect of irradiation and the conditions of irradiation. As an objective, while retaining the form of original model, the proposed modifications to account for irradiation are also kept to a minimum as described below. In particular, the following modification is proposed for IASCC to account for irradiation effects on SCC susceptibility, in addition to those related through the changes in strength properties:

\[
t_i = \lambda_0 \cdot \exp\left(\frac{D}{D_0}\right) \left[\left(r - 1\right)\left(rE / S_y\right)^{1/3}\right]^p \ln\left((A - z) / \left(S / S_y - z\right)\right) \cdot F (2)
\]

where, $D$ : displacement-per-atom, as the measure of irradiation level
$D_0$ : material–environment irradiation resistance parameter (unit: dpa)
$A = (v - a_0 \cdot D) \cdot \exp(w \cdot r)$
$a_0$ : irradiation damage sensitivity parameter of $A$ (unit of $a_0$: 1/dpa)
$v, w$ : correlation constants for the micro-cracking resistance parameter $A$
$z = z_1 + z_2 \ln r$
$z_1, z_2$ : correlation constants for the threshold stress severity parameter $z$

Note that the above model for IASCC initiation reduces to the original SCC initiation model, with zero dpa ($D = 0$). That is, the model parameters $\lambda_0, p, \text{ and } z$ are the same as in the original SCC initiation model described above. The model parameters $D_0$ and $a_0$ are used as modifiers to account for the additional irradiation effects, particularly for the high dpa regime where the damaging influence is expected to be due to (i) the reduced resistance to micro-cracking and ductility, and (ii) continued RIS or reduction in the SCC resistance afforded by the material–environment interaction: these effects are associated with the original model parameters $\lambda_0$ and $A$, respectively. Therefore, the functional forms of these parameters include an explicit dependence on dpa ($D$) as described by the above IASCC model.

The change in hardening response and its impact on the IASCC susceptibility are directly reflected in the irradiated material yield strength, $S_y$, and the strength ratio, $r$, that also influences the model parameters $A$ and $z$. The parameter $z$ represents an apparent threshold stress severity level that decreases with dpa, as the strength ratio $r$ approaches 1 due to loss of strain-hardening capacity, with a lower bound $z_1$. Note that the absolute value of the apparent stress threshold itself may increase with dpa since it is also affected by the yield strength that increases with dpa (reaching a saturation level). The IASCC model parameters and assessment with IASCC data are summarized next.

5.0 IASCC DATA AND MODEL ASSESSMENT

IASCC failures data from a recently completed comprehensive set of tests using a systematic evaluation of three heats of highly irradiated Type 316 stainless steel were reviewed and used to assess the above IASCC initiation model. These tests were performed under the International IASCC Advisory Committee (IIAC) program that was previously described by Freyer, et al. [46] and earlier work by Conermann et al. [47]. The essential key features of these data sets are summarized here for context, while the original
articles [46, 47] may be consulted for additional details. The IASCC tests were performed on samples cut from in-service irradiated flux thimble tubes (FTT) of three PWR units. The steel tubes were reported to have an initial cold-work of about 15% required to achieve the initial high yield strength in the range of 483 to 621 MPa. The nominal wall thickness of the FTT was in the range of 1.22 to 1.47 mm, resulting in relatively thin-wall test sections. The tests were performed on typical O-ring samples cut from the FTT, which were loaded in a shielded autoclave with once-through circulating water at a temperature of 340°C in a simulated PWR water environment. The O-rings were essentially subject to load-control conditions, and the failure times were determined based on the time recorded for load drop and the resulting deflection of the displacement transducer monitored during the testing. The recorded time was considered to be the time to IASCC crack initiation and it was reported also as the failure time. The reported peak stresses for each test sample were obtained with detailed, elastic, finite-element analysis work of the specimen geometry. Likewise, the irradiation levels for each sample were reported based on detailed, plant-specific operating cycle parameters and time-integrated flux calculations. These conditions covered the stresses in the range of 612 to 929 MPa and the dpa range of 12 to 76 [46, 47].

The O-ring failures data are shown in Figure 1 (solid symbols) illustrating the normalized stress dependence of initiation time for dpa values ranging from 12 to 76 dpa. Note that several tests were also terminated or did not produce failures after a specific time had elapsed. These non-failures data (open symbols) are also shown in Figure 1; the stress values for these tests were in the range of 140 to 790 MPa. Note that some of the variation seen in this depiction is attributable to the fluence levels not shown for the data points and possible heat-to-heat variability. With the exception of a few points in the upper right-most corner, Figure 1 shows a general trend for increasing failure times with lower normalized stress over the wide range of fluence, especially if one allows for a very long life (an apparent threshold) below about sixty percent value of the normalized stress.

The O-ring test samples were cut from FTT of three PWR units (Beaver Valley 1, H. B. Robinson 2, and Ringhals 1), each with its own heat of Type 316 stainless steel [46, 47]. The failures data from these tests were analyzed with the above IASCC model on a heat-by-heat basis. The Ringhals data were used to determine the model parameters since this set had the largest number of data points, relatively less scatter, and wider coverage of stress and dpa levels. This determination was made with the approach as summarized in the following.

The strength-based correlation parameters $v$, $w$, and $z_2$, as well as the hardening sensitivity parameter $p$ of the model were kept the same as those derived in the prior SCC initiation work for unirradiated stainless steels [3, 17]. Furthermore, since the value of threshold parameter ($z_1 = 0.4$) was a conservative estimate in the original model for unirradiated condition, the new IASCC model was evaluated for this fixed value of 0.4 as before, as well as for its data-specific best-fit value.

For the cold-work, dose level, and temperature applicable to a test sample, the correlation from [48] provides an estimate of the expected yield strength at the test temperature and at room temperature. The ratio of these two values gives the scaling factor that is applied to the sample-specific yield strength reported at high temperature to estimate the sample-specific yield strength at room temperature ($S_y$). The correlation from [48] also provides an estimate of the expected strength ratio ($r$) at room temperature. These estimates of the room temperature yield strength ($S_y$) and the strength ratio ($r$) are used in the model estimation of initiation time.

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2 In this and other Figures, for consistency with the reported data, the stress is normalized with the irradiated material yield strength at the high temperature (340°C) for the cold-work condition and the applicable dpa levels, unless indicated otherwise.
The irradiation-specific model parameters $a_0$ and $D_0$ are independent of the strength properties. The remaining material–environment specific parameter, $\lambda_0$, is also independent of material strength but includes a temperature dependence; for the set of IASCC initiation tests the temperature was fixed. These three model parameters were considered as variables of optimization in determining their values. The model estimated initiation times were then compared with the corresponding failure times for the Ringhals data (in this estimation, data below 10-hour failure time limit were considered to have too short a time for much environmental contribution to be included in the analysis). The root-mean-square value of the deviations between data and model estimates on a log-life basis was used as the basis for optimization. The resulting value for $\lambda_0$ was 487 hours at 340°C; and with the original, conservative estimate of $z_1 = 0.4$ fixed, the values of $a_0$ and $D_0$ were determined to be 0.0021 (1/dpa) and 72.5 dpa, respectively. The data-specific best-fit value for $z_1$ was determined to be 0.5623, with the resulting estimates for $a_0$ and $D_0$ of 0.0016 (1/dpa) and 44.5 dpa, respectively.

The Ringhals IASCC failure data, test conditions, and model estimates/parameters from the above described optimization are shown in Figure 2 and Figure 3 for the case of conservatively fixed $z_1$ of 0.4 and for the data-specific high value determined as $z_1$ of 0.5623, respectively. In these figures the lines identified as $x_1$, $x_2$, and $x_3$ represent the factors of 1, 2, and 3, respectively, on the model estimates for comparison with the reported data. Both these sets of model comparisons in Figures 2 and 3 show fairly close agreement with the data, mostly within a factor of three, with an overall improvement with the higher value for $z_1$.

The heat-to-heat variation of IASCC failure data was assessed further as verification by using the above best estimate parameters (Figure 3) derived from the Ringhals FTT heat to predict the failure times in tests on O-ring samples from the other two heats of FTT taken from Beaver Valley and H. B. Robinson units. These model results, along with those for the Ringhals heat, are compared with the data in Figure 4, which shows that most points for all three heats are well within a factor of three on the model predictions. It may be noted that the data from Beaver Valley FTT heat were fairly limited and showed large scatter in the failure times: for example, two tests under practically identical conditions (340°C, 51 dpa, 15% cold-work, and same heat) with stress of 678 MPa and 676 MPa were reported to fail in 29 and 483 hours, respectively, with the model estimates very close to 107 hours in Figure 4. Likewise, two other points (the lowest and the highest in Figure 4) correspond to nearly identical test conditions but with reported failure times of 5 and 372 hours, respectively; this is almost a factor of 74, likely indicative of some anomaly or potential outlier data. There are no obvious or apparent factors that could be taken into account in any deterministic model to reduce this scatter. Also note that the data from these two non-Ringhals heats cover only a limited range of test conditions, as may be inferred from the range of model estimated time (open symbols in Figure 4). While these characteristics of the data from these two heats do not allow for a dependable model fit to them, the data are used here to compare the respective estimation based on the independent model fit derived from the Ringhals data, as shown in Figure 4.

Based on the above model assessment of heat-specific data from Ringhals FTT, and considering the larger set of Ringhals data with relatively less scatter, the dpa-specific parameters in the new IASCC initiation model were selected as the best estimate, with $D_0 = 44.5$ dpa and $a_0 = 0.0016$ (1/dpa). The value of model parameter $z_1$ seemed to lie in the range of 0.4 to 0.6, where the former was suggested as a conservative estimate in the original SCC model for the unirradiated condition, while the latter appears to be an upper bound that, however, is likely subject to the limitations or specificity of the data used in this assessment. As such, the geometric average of these two values, 0.49, is used with engineering judgment as the estimate for the threshold severity parameter $z_1$. As noted above, the strength-based correlation parameters $v$, $w$, and $z_2$, as well as the hardening sensitivity parameter $p$ were the same as derived in the prior work on unirradiated materials [3, 17]. For reference, this complete set of IASCC initiation model parameters is summarized in Table 1.
6.0 DISCUSSION AND FURTHER CONSIDERATIONS

The results presented above were based on extending the SCC initiation model for unirradiated conditions to assess IASCC initiation. In this extension, the emphasis was placed on making use of the correlations previously shown to take into account the expected increase in material susceptibility to SCC resulting from the material hardening or high strength levels due to processing. Consistent with the phenomenological description of the effects of the irradiation process on the factors necessary for SCC, the damaging influence of irradiation, as measured by dpa, is reflected in all the multiplicative factors contributing to IASCC susceptibility or time to initiation.

The resulting IASCC model addresses a wide range of cold-work, stress severity, and dpa level, including unirradiated and non-cold-work conditions. As such, the effects of these variables on time to initiation and the model trends for varied combinations can be examined more generally. This is illustrated in Figure 5 with the best estimate model parameters for Type 316 stainless steel in a simulated PWR environment at 340°C. The model response shows the expected trends in the saturation with respect to cold-work level in the case of unirradiated material (curves in Figure 5 for dpa = 0) and the additional increase in susceptibility due to irradiation at the low and high dpa values. Results in Figure 5 show a wide variation in estimated time to initiation and a notable sensitivity to the main variables, especially below the stress of about 80% of the irradiated material yield strength at temperature. It also indicates the likelihood of an apparent large scatter in this range, including sensitivity to failure versus non-failure events, and emphasizes the need for accurate control and estimation of the test variables.

The model parameters are based on the O-ring tests conducted at 340°C that is somewhat above the temperature range of in-service irradiation of the FTT samples. The latter range is expected to have contributed the highest irradiation damage within that of normal LWR operation, with the exception of void swelling likely to be significant near or above 365°C. The difference in higher test temperature relative to the irradiation temperature range may not be of much significance, but it is noteworthy. Also, the nearly constant load type of testing on the relatively thin wall samples is expected to have some impact on the failure times in these tests in that the observed times are likely to be shorter than would be obtained, for example, under constant tensile stress using a round bar of 3 to 6 mm diameter. That is, the reported initiation times in the analyzed O-rings tests are representative of the IASCC depths near or below the lower end of ~0.4 mm, versus the high end of ~2 mm intended in the engineering initiation model. As such, it is expected that the estimated model parameters in the current work represent greater conservatism with regard to the resulting initiation times.

The review of reported IASCC test data on time to failure in constant load tests (and on the extent or percent area of intergranular mode in constant extension rate tests in general) showed considerable scatter even for a single heat of material. If the scatter is real, and if it is taken also to reflect the long-term (extrapolated) response, then it is more critical to examine and understand, if not to address, the plausible underlying causes or reasons for such an extraordinary range of scatter. Also, considering the significance of strength properties to the IASCC susceptibility, the sensitivity of these properties to dpa levels below 10 dpa, and the lack of IASCC data in this range, there is a need for more IASCC data with a systematic test plan in this range of dpa. Also, going forward, it would be useful to do a critical review of conditions or root cause analyses of actual service instances where IASCC has been identified, with a view to confirm or prioritize the significant factors contributing to the manifestation of IASCC.

Over the relatively narrow range of LWR temperatures of interest (about 0.33 to 0.39 on homologous scale) the model assumes that the temperature dependence is adequately represented by the Arrhenius relation with a single value for the activation energy independent of other variables. The activation energy could not be ascertained from the IASCC data analyzed in the current work; the likely range of this apparent activation energy needs to be confirmed with additional tests/assessment.
In the current assessment for IASCC initiation response, an implicit assumption is made that any impact of irradiation on the interface (or oxide film) properties relevant to SCC is indirectly included in the material–environmental model factor. It would be useful to review and confirm the extent and relative significance of any changes in properties of the interface/oxide under irradiation versus those under non-irradiation, including those of repassivation, which are likely to be affected by material compositional changes near/within the grain boundary regions.

Although the specific data and model assessment in this work dealt with the primary water reactor (PWR) primary water environments, the model formulation is expected to be applicable in light water reactor (LWR) water environments in general; this needs to be confirmed with additional data and model application. The specific model as developed in this work gives a deterministic basis for a best estimate of the initiation response, but the formulation is considered simple enough to implement in a probabilistic assessment such as that being developed in the EPRI/DOE/NRC collaborative program on extreme low probability of rupture (xLPR) event, or using the standard methods of propagation of uncertainty.

SUMMARY AND CONCLUSION

Key observations from service failures attributed to IASCC and an assessment of the effects of irradiation damage in relation to the factors/processes responsible for manifestation of SCC were summarized. This summary supported the phenomenological basis for applicability and extension of the SCC initiation model previously validated for the case of unirradiated austenitic stainless steel and nickel-base alloys. In particular, it was noted that (a) the stress severity (i.e., the ratio of stress to yield strength) and strength properties are significant correlating factors for SCC response, with or without irradiation, (b) the material properties and deformation response are strongly dependent on the elements of microstructure that are well correlated with the major effects of irradiation, and (c) the effects of irradiation are better represented in terms of the dpa as a more fundamental or physical measure of damage.

The mechanistically based model for SCC of unirradiated materials was extended to assess IASCC initiation. The extended model was shown to be in agreement with IASCC failure time data, mostly within a factor of three, and with major trends, particularly for stress, dpa, cold-work, and radiation hardening assessed in this work. In particular, the model extension to IASCC initiation was developed and model parameters were determined for the case of Type 316 stainless steel in a typical PWR water environment at high temperature. The model results were compared with the test data under constant load conditions and up to 76 dpa. The best-fit model parameters determined from IASCC failure data on one heat of Type 316 stainless steel were used to predict the failures of two other heats tested similarly; the data from these two heats were not used in determining the model parameters. Within the limits of the analyzed data, with the exception of possible outlier data, the predicted failure times and the test data for these two heats are in reasonable agreement supporting the model validation.

In the extended model, from an engineering applications perspective, the significant effects of irradiation in enhancing the SCC susceptibility were taken into account by using (a) the mechanical strength properties affected by the irradiation hardening, and (b) two modifications for effects not covered by the strength properties alone—one for reduced ductility and micro-cracking resistance, and the other for enhanced interaction at the material–environment interface—using dpa as the correlating parameter. The IASCC model reverts to that for the original SCC model for unirradiated condition, when the dpa is zero.

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REFERENCES


Table I
IASCC initiation model parameters for Type 316 stainless steel in simulated PWR primary water environment at 340°C.

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Description</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>$D_0$</td>
<td>material–environment irradiation resistance parameter, dpa</td>
<td>44.5</td>
</tr>
<tr>
<td>$a_0$</td>
<td>irradiation damage sensitivity parameter of the micro-cracking resistance $A$, $1$/dpa</td>
<td>0.0016</td>
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<tr>
<td>$\nu$</td>
<td>pre-multiplier of the micro-cracking resistance $A$</td>
<td>0.9</td>
</tr>
<tr>
<td>$w$</td>
<td>strength sensitivity of the micro-cracking resistance $A$</td>
<td>0.166</td>
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<td>$z_1$</td>
<td>lower bound of the threshold stress severity $z$</td>
<td>0.49</td>
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<tr>
<td>$z_2$</td>
<td>strength sensitivity of the threshold stress severity $z$</td>
<td>0.16</td>
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<tr>
<td>$p$</td>
<td>hardening sensitivity parameter</td>
<td>0.13125</td>
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<tr>
<td>$\lambda_0$</td>
<td>material–environment factor including temperature dependence, hours</td>
<td>487</td>
</tr>
</tbody>
</table>

Fig. 1. IIAC O-rings test data showing general trend for the time dependence on the normalized stress observed over a wide range of estimated dpa values.
Fig. 2. Assessment of the O-ring failures data for Ringhals FTT heat and comparison with the new IASCC model fit with a lower bound threshold stress severity.

Fig. 3. Assessment of the O-ring failures data for Ringhals FTT heat and comparison with the new IASCC model best-fit with a data-specific high threshold stress severity.
Fig. 4. Comparison of O-ring failures data of Beaver Valley and H.B. Robinson heats with prediction using the best-fit IASCC model from the Ringhals FTT failures data.

Fig. 5. IASCC model characteristics showing the influence of initial cold-work level, stress severity, and irradiation (dpa) level, including unirradiated condition.