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Application of the FRI crack growth model for neutron-irradiated stainless steels in high-temperature water of a boiling water reactor environment

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A B S T R A C T

This study considered a methodology to reflect the effect on material properties by neutron irradiation onto the FRI model of the crack growth rate (CGR) of stress corrosion cracking developed at Tohoku University. Yield strength and strain hardening exponent were evaluated by a tensile test for irradiated stainless steels, and the relationship between mechanical property and strain to fracture of the oxide film was derived. Effect of the radiation-induced segregation on CGR was also discussed. The experimental tendencies for the CGR to increase with dose and almost saturate above 3 dpa were well replicated by the model calculations.

1. Introduction

Stress corrosion cracking (SCC) in high-temperature water is a major concern for boiling water reactor (BWR) components. In the case of reactor internals (RINs), SCC susceptibility of their constituent materials is influenced by neutron irradiation which affects material properties. Irradiation assisted stress corrosion cracking (IASCC) is SCC phenomenon caused by neutron irradiation of austenitic stainless steels [1–4], and it is necessary to consider the IASCC susceptibility for RINs. SCC incidents have been observed in the core shroud of welds [5,6]. It is unclear if the cause of these incidents is IASCC or not, however, neutron irradiation influences propagation of a crack after its initiation. Fracture resistance of RINs is lowered as the crack propagates; therefore, evaluation of crack growth rate (CGR) of irradiated stainless steels is indispensable to ensure the integrity of RINs such as the core shroud in BWRs.

Neutron-irradiated stainless steels show higher CGRs than unirradiated stainless steels since material properties vary with irradiation. Many studies have reported that CGR increases with the yield stress of the tested material [7,8]. Neutron irradiation changes the mechanical properties of the material and increases the yield stress. Consequently, radiation hardening accelerates CGR of irradiated stainless steels. Another possible accelerating factor is radiation-induced segregation (RIS) such as Cr depletion at a grain boundary (GB). Cr depletion at a GB decreases corrosion resistance and it may affect the SCC crack growth behaviour. These changes of material properties by neutron irradiation should be considered in the SCC CGR evaluation.

In order to evaluate the SCC CGR, predictive models of crack growth have also been developed. The slip dissolution/oxidation model by Ford and Andresen [9–11] is a prototype model describing SCC propagation in an oxygenated aqueous system. In the slip dissolution/oxidation mechanism, cracks propagate by localized grain boundary dissolution or oxidation resulting from repeated breakdown of the protective film at the crack tip. Relating the CGR to the oxidation charge density during rupture/repassivation of oxide films, they derived a relationship between the CGR and the strain rate at the crack tip. The PLEDGE model [12] is an empirical model based on the Ford-Andresen model for irradiated stainless steels. The FRI model [13–15] developed by Shoji et al. employs a relationship between the CGR and the strain rate at the crack tip defined by Ford and Andresen model. The FRI model determines the strain rate from the strain field distribution ahead of the growing crack tip. The input parameters of the FRI model relate to material properties such as corrosion properties and mechanical properties. The model parameters relevant to mechanical properties determine the strain rate at the crack tip, and the parameters characterizing the oxidation current are related to corrosion resistance. Accordingly, the FRI model is applicable for irradiated stainless steels if the mechanical properties and the corrosion resistance influenced by neutron irradiation can be taken into account in the input parameters. Many CGR data for irradiated stainless steels [16–21] have been acquired recently, and

Abbreviations: BWR, boiling water reactor; CGR, crack growth rate; CT, compact tension; EPR, electrochemical potentiokinetic reactivation; FRI, fracture and reliability research institute at Tohoku University; GB, grain boundary; IASCC, irradiation assisted stress corrosion cracking; JMTR, Japan Materials Testing Reactor; NISA, Nuclear and Industrial Safety Agency; PLEDGE, plant life extension diagnosis by General Electric; RIN, reactor internal; RIS, radiation induced segregation; SCC, stress corrosion cracking; SS, stainless steel

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Nomenclature

a	Crack length
B_1	Intermediate variable as defined by Eq. (2)
B_2	Intermediate variable as defined by Eq. (3)
C	Constant in the stress-strain relation defined by Gao and Hwang
$\frac{da}{dt}$	Crack growth rate
E	Young's modulus of the metal
F	Faraday's constant
i_o	Bare surface oxidation current density
K	Stress intensity factor
K_{ISCC}	SCC threshold stress intensity factor
M	Atomic weight of the material
m	Exponent of the oxidation current decay curve
n	Strain hardening exponent defined by Gao and Hwang
r_o	Characteristic distance at which the strain rate is defined
t	Time
t_o	Time for the onset of oxidation current decay
z	Change in valence due to oxidation
β	Dimensionless constant for crack tip plastic strain
ε	True strain
ε_f	Strain to fracture of the oxide film
ε_y	Yield strain
λ	Dimensionless constant for crack tip plastic strain
ρ	Atomic density of the metal
σ	True stress
σ_{UTS}	Ultimate tensile strength
σ_y	Yield stress

the FRI model calculation is expected to give results consistent with them.

In order to assess the CGR of components in accordance with the latest CGR data, development of the predictive model which can consider variation of the material properties with the irradiation is valuable. If such a predictive model is prepared, SCC CGR in components of operating BWRs can be evaluated more quantitatively considering not only the neutron fluence but also the resultant material properties. This paper discusses a methodology to reflect the change of material properties due to neutron irradiation in the input parameters of the FRI model. As for the influence of the material properties from irradiation on the CGR, the latest CGR data were reviewed and analysed. Handling procedures for input parameters were proposed taking into account these analyses, and CGRs were calculated and compared with experimental data.

2. Crack growth rate equation

Fig. 1 shows the schematic of the oxidation current during the crack propagation by the slip dissolution/oxidation mechanism. The terms m, i_0, and t_0 are parameters characterizing the oxidation current decay curve. t_f is the periodic of the oxide film rupture and it is determined by the strain rate at the crack tip and the strain to fracture of the oxide film. In the FRI model, the strain rate at the crack tip is derived using the strain distribution of Gao and Hwang [22] ahead of the growing crack tip for the plane-strain condition. Gao and Hwang equation contains tensile properties explicitly, consequently, input parameters yield stress (σ_y) and strain hardening exponent(n) require the data for irradiated stainless steels to reflect the effect of radiation hardening. The oxidation current decay curve as shown in Fig. 1 is likely affected by Cr depletion at GB caused by RIS. This is another point of consideration for the effects of neutron irradiation on the CGR model.

On the other hand, the FRI model has mathematical singularity for the exponent of the decay curve $m = 1$. In order to avoid this numerical instability, modified CGR equations were derived for different load conditions by Hashimoto and Koshiishi [23]. As many CGR data for irradiated stainless steels have been obtained by using CT specimens with a constant load condition, the equation for constant-load condition is used for the CGR calculation in this study. In the constant-load condition, CGR (da/dt, where a is crack length, t is time) is expressed as follows:

$$\frac{\mathrm{d}a}{\mathrm{d}t} = \frac{\varepsilon_f}{B_1 t_0}\left[\frac{B_1 B_2}{1 - m + mB_1B_2}\right]^{\frac{1}{1-m}}, \tag{1}$$

and B_1 and B_2 are given by the next equations.

$$B_1 = \left(\frac{\partial\varepsilon}{\partial K}\frac{\partial K}{\partial a} - \frac{\partial\varepsilon}{\partial r}\right)_{r=r_0} = \beta\cdot\frac{\sigma_y}{E}\frac{n}{n-1}\left(\frac{2}{K}\frac{\partial K}{\partial a} + \frac{1}{r_0}\right)\left\{\ln\left(\frac{\lambda}{r_0}\left(\frac{K}{\sigma_y}\right)^2\right)\right\}^{\frac{1}{n-1}} \tag{2}$$

$$B_2 = \frac{M}{z\rho F}\cdot\frac{i_0 t_0}{\varepsilon_f} \tag{3}$$

Here, M and ρ are the atomic weight and density of the metal respectively, and z is the change in valence due to oxidation. F is the Faraday constant and ε_f is the strain to fracture of the oxide film. The terms m, i_0, and t_0 are the exponent of the decay curve, the bare surface oxidation current density, and the time for the onset of current decay respectively. β and λ are dimensionless constants relating to crack tip plastic strain, σ_y is the yield stress, E is Young's modulus of the metal, n is the strain hardening exponent in Gao and Hwang's definition, K is the stress intensity factor, and r_0 is the characteristic distance at which the strain rate is defined.

3. Influence of material properties on CGR

3.1. CGR data survey

CGR data are surveyed in this section, to clarify how the material property changes induced by irradiation affect CGRs. Data that meet the following conditions are selected.

- CGRs of austenitic stainless steels sampled from BWR RINs or irradiated in test reactors, such as Japan Materials Testing Reactor (JMTR) and Halden Reactor, at BWR operating temperatures.
- CGRs measured with CT specimens under constant load conditions.
- CGRs measured in post irradiation examinations at BWR operating temperatures and in the simulated BWR normal water chemistry

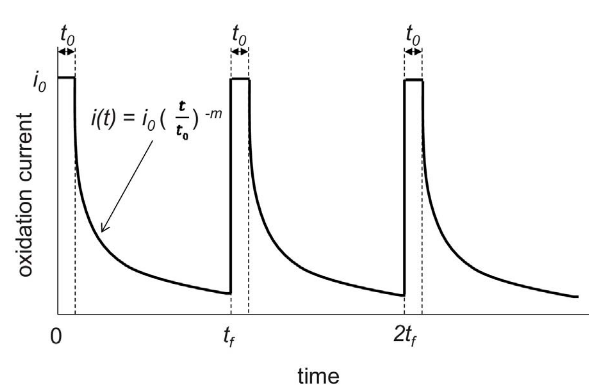

Fig. 1. Schematic of the oxidation current.