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Fatigue life predictions for irradiated stainless steels considering void swellings effects

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ABSTRACT

The objective of this study is to estimate fatigue life of irradiated austenitic stainless steels types 304, 304L, and 316, which are extensively used as structural alloys in the internal elements of nuclear reactors. These reactor components are typically subjected to a long-term exposure to irradiation at elevated temperature along with repeated loadings during operation. Additionally, it is known that neutron irradiation can cause the formation and growth of microscopic defects or swellings in the materials, which may have a potential to deteriorate the mechanical properties of the materials. In this study, uniaxial fatigue models were used to predict fatigue properties based only on simple monotonic properties including ultimate tensile strength and Brinell hardness. Two existing models, the Bäumel–Seeger uniform material law and the Roessle–Fatemi hardness method, were employed and extended to include the effects of test temperature, neutron irradiation fluence, irradiation-induced helium and irradiation-induced swellings on fatigue life of austenitic stainless steels. The proposed models provided reasonable fatigue life predictions compared with the experimental data for all selected materials.

1. Introduction

With a growing demand to reduce greenhouse gas emissions, the focus of energy generation sources has shifted from fossil fuel-based electrical production to those that provide low emission, inexpensive, and reliable electricity such as nuclear power reactors. It was reported that approximately 19% of the total electrical supply in the United States in 2014 was generated from 104 commercial nuclear reactors at 62 nuclear power plant sites in operation nationwide [1].

While nuclear reactors are typically designed with an operational life of 40 years, their lives can be extended to 20 additional years or more with provisions for Licensing Renewal [2]. To ensure the safe operation of existing reactors beyond their initial design lives, understanding the long-term structural integrity of reactors is a major concern. One of the prominent issues related to failures in nuclear power components is attributed to material degradation due to aggressive environment conditions, and mechanical stresses. For instance, reactor core support components, such as fuel claddings, are under prolonged exposure to an intense neutron field from the fission of fuel and operate at elevated temperature under cyclic (i.e., fatigue) loadings caused by start-up, shut-down, and unscheduled SCRAM (emergency shut-down) [3]. The fluctuations in loadings typically occur a few hundred to a thousand times during the life of the vessels [4]. Pressurizer, steam separators, pumps, steam generator shells, piping, etc., are among the nuclear reactor components subjected to fatigue damage during operation [3].

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Nomenclature

A_o, A_1	Ultimate tensile strength coefficient of a specific stainless steel at 330°C for a given dose.
b	Fatigue strength exponent
c	Fatigue ductility exponent
d	Radiation dose in displacements per atom (dpa)
d_o	Normalized radiation dose in displacements per atom (dpa)
$\dot{d}$	Radiation dose rate (dpa/s)
Δd	Radiation dose increment
$\mathbf{E}$	Energy
E	Modulus of elasticity
F	Correction factor in void swelling equation
HB	Brinell hardness
K_{void}	Void swelling exponent
N_{void}	Void swelling constant
$2N_f$	Reversals to failure
r_{cw}	Percent cold-work
S	Volumetric swelling strain
S'	Volumetric stress-free swelling rate
ΔS	Incremental swelling strain
T	Test temperature
ε_f'	Fatigue ductility coefficient
$\frac{\Delta\varepsilon}{2}$	Total strain amplitude
$\frac{\Delta\varepsilon_e}{2}$	Elastic strain amplitude
$\frac{\Delta\varepsilon_p}{2}$	Plastic strain amplitude
σ_f'	Fatigue strength coefficient
σ_U	Ultimate tensile strength
$\Delta\overline{\sigma}$	von Mises effective stress range

It is known that failure due to fatigue is the dominant mechanical failure mode for most machinery and structural components [5]. This type of failure is associated with crack initiation and growth that eventually lead to fracture. The majority of fatigue failures are unexpected and generally occur under the cyclic loading with peak values significantly less than the safe loads estimated from the static fracture analysis [6]. Additionally, exposure to high fluence neutron radiation can lead to microscopic defects that result in material hardening and embrittlement, which significantly affects the physical and mechanical properties of the materials, resulting in further reduction in fatigue life of reactor structural components.

The strain–life (ε–N) method for fatigue life estimation is one of the classical approaches that can be applied in both low-cycle and high-cycle fatigue regimes. In this method, the total strain amplitude, $\frac{\Delta\varepsilon}{2}$, can be separated into elastic and plastic strain components, and related to reversals to failure, $2N_f$, by the Coffin–Manson relationship as:

$$\frac{\Delta\varepsilon}{2} = \frac{\Delta\varepsilon_e}{2} + \frac{\Delta\varepsilon_p}{2} = \frac{\sigma_f'}{E}\left(2N_f\right)^b + \varepsilon_f'\left(2N_f\right)^c \tag{1}$$

where $\frac{\Delta\varepsilon_e}{2}$ is the elastic strain amplitude and $\frac{\Delta\varepsilon_p}{2}$ is the plastic strain amplitude. In Eq. (1), σ_f', b, ε_f', and c are fatigue strength coefficient, fatigue strength exponent, fatigue ductility coefficient, and fatigue ductility exponent, respectively. Although the most accurate approach to obtain the fatigue behavior of a given material is to perform the experimental-based determination of fatigue parameters, comprehensive fatigue experiments are usually costly and time consuming. Furthermore, fatigue tests of materials or components subjected to irradiation require specialized instruments and extensive amounts of time. The tests are also extremely expensive and can involve significant radiation exposure to the test personnel. Therefore, the majority of research in the literature involving irradiated materials has been focused on studying the changes in tensile and fracture toughness properties, which is less expensive and can be obtained within a short timeframe. Only limited experimental studies have been carried out to investigate the effects of radiation on fatigue properties of reactor structural materials.

A number of semi-empirical relations have been proposed to correlate uniaxial fatigue behavior of unirradiated metallic materials to their tensile properties. Among these methods, Bäumel–Seeger uniform material law [7] and Roessle–Fatemi hardness methods [8] have been shown to provide good approximations of fatigue parameters for various types of metals [9–11]. Both methods are extended in this work to estimate the fatigue behavior of irradiated stainless steels.

Bäumel and Seeger [7] were among the first to consider unalloyed and low-alloy steels separately from aluminum and titanium alloys. In their work [7], the empirical relations to approximate the strain amplitude-life curve, known as a uniform material law,