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Mechanical properties of neutron-irradiated model and commercial FeCrAl alloys☆

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ABSTRACT

The development and understanding of the mechanical properties of neutron-irradiated FeCrAl alloys is increasingly a critical need as these alloys continue to become more mature for nuclear reactor applications. This study focuses on the mechanical properties of model FeCrAl alloys and of a commercial FeCrAl alloy neutron-irradiated to up to 13.8 displacements per atom (dpa) at irradiation temperatures between 320 and 382 °C. Tensile tests were completed at room temperature and at 320 °C, and a subset of fractured tensile specimens was examined by scanning electron microscopy. Results showed typical radiation hardening and embrittlement indicative of high chromium ferritic alloys with strong chromium composition dependencies at lower doses. At and above 7.0 dpa, the mechanical properties saturated for both the commercial and model FeCrAl alloys, although brittle cleavage fracture was observed at the highest dose in the model FeCrAl alloy with the highest chromium content (18 wt %). The results suggest the composition and microstructure of FeCrAl alloys plays a critical role in the mechanical response of FeCrAl alloys irradiated near temperatures relevant to light water reactors.

1. Introduction

Development of FeCrAl alloys for nuclear power applications was originally pursued by General Electric (GE) Corporation in the 1960s with the objective of developing oxidation-resistant fuel elements that would lead to better thermal efficiency, higher burnup, and robust containment of fission products in nuclear reactor environments of steam, air, and carbon-dioxide [1]. A significant amount of work was completed toward developing these alloys for high-temperature nuclear power applications, including studies of fuel-clad compatibility, high-temperature exposure in both air and steam, aging, welding/formability, and radiation effects, among several other topics [1–5]. Furthermore, development of both wrought and powder metallurgy processed FeCrAl alloys was pursued. Eventually, FeCrAl alloys lost favor in this high-temperature materials program, as the alloys developed at the time did not exhibit adequate high-temperature mechanical properties for structural applications [6] and were prone to embrittlement [4].

The original properties that made FeCrAl alloys attractive for use in high-temperature nuclear reactor applications are now also drawing attention to their development for accident-tolerant-fuel (ATF) applications [7]. FeCrAl alloys have shown outstanding high-temperature steam oxidation resistance [8–11], a key characteristic for developing systems with enhanced safety margins during design-basis and beyond-design-basis accident scenarios for light water reactor (LWR) cladding applications [12]. ATF applications such as LWR cladding would involve lower expected service temperatures than those studied by GE in considering FeCrAl alloys for use in high-temperature nuclear reactors. Lower

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operating temperatures and modern processing routes could overcome the deficiencies described within GE's alloy development program in the 1960s. Hence, FeCrAl alloys are currently undergoing rapid advancement. Of primary interest is developing an alloy or set of alloys that perform well in both pressurized water reactor and boiling water reactor environments during normal operation while exhibiting strong steam oxidation resistance during accident scenarios. Current efforts have shown a balance must be achieved between FeCrAl materials properties, including high-temperature mechanical strength, oxidation resistance, formability, thermal stability, creep strength, and radiation tolerance.

Radiation tolerance assessment of FeCrAl alloys is of primary importance, as neutron radiation exposure can result in significant deviations in the mechanical properties of the material compared with as-received or even aged conditions. In high-chromium (Cr) ferritic alloys, these deviations typically entail significant hardening and embrittlement and hence limit the service life in a nuclear reactor. As noted, the early high-temperature materials program completed several radiation tolerance evaluations, including irradiation at 50 °C to a neutron fluence of 1×10^{19} n/cm^2 (E > 1 MeV) (dpa was not reported) of a FeCrAl alloy with 15 wt % Cr. The alloy showed significant changes in the ductile-to-brittle transition temperature and radiation-induced hardening [5]. Building on this, Field et al. conducted mechanical tests on neutron-irradiated model FeCrAl alloys of varying compositions to 1.8 dpa at 382 °C, which showed similar behavior with significant hardening and a composition dependence in the hardening driven by the formation of the Cr-rich α' phase [13]. Additional work has been completed on commercial alloys, such as the study of Ahmedabadi and Was in which the Kanthal Advanced Power Metallurgy Technology (APMT) alloy was proton-irradiated to 5 dpa at 360 °C and tested using constant-extension-rate tensile tests; it demonstrated good stress corrosion cracking resistance and reduction of area [14]. Other than these studies, few details exist for the mechanical response of neutron- or ion-irradiated FeCrAl alloys.

Furthermore, little attention has been paid to whether the issues identified in the high-temperature materials program in the 1960s have been remedied through modern design and fabrication processes. A critical need exists to develop a robust and expansive mechanical properties database on as-received and neutron-irradiated FeCrAl alloys before their deployment as an ATF technology in commercial LWRs.

In this study, investigations were performed to assess the mechanical properties of advanced oxidation-resistant FeCrAl alloys after neutron irradiation from beginning-of-life to expected near end-of-life dpa conditions at a nominal temperature of 320 °C using accelerated testing in a materials test reactor. The same model alloys used in the previous study [13] and the powder metallurgy–derived, commercially available FeCrAl alloy Kanthal APMT were investigated. Tensile tests of sub-size tensile specimens and post-test fractography were used to determine key mechanical properties, including yield stress, tensile ductility, and reduction of area. The results and analysis within this study represent continuing steps toward developing a database of mechanical properties and radiation tolerance of FeCrAl alloys for nuclear power production applications, including LWR ATF cladding.

2. Materials and methods

Materials for irradiation and subsequent tensile testing included four model FeCrAl alloys with varying Cr and aluminum additions and the commercially available FeCrAl alloy Kanthal APMT. The model alloys studied are designated as F1C5AY, B125Y, B154Y-2, B183Y-2 and have been the subject of several other studies as well [13,15–17]. In the case of the "B-series" alloys, the first two numbers in the alloy designation refers to the nominal target Cr composition while the third number refers to the nominal Al composition, e.g. B125Y has a nominal Cr content of 12 wt% and Al content of 5 wt%. The F1C5AY model alloy had a nominal content of 10 wt% Cr and 5 wt% Al. The compositions of all alloys studied were determined using inductively coupled plasma optical emission spectroscopy (ICP-OES), the results of which are presented in Table 1.

Model FeCrAl alloys were manufactured using standard wrought alloy practices, including arc melting of pure element feedstocks and pre-alloyed aluminum-yttrium specimens. Thermomechanical treatments of the model FeCrAl alloys were conducted using hot forging, rolling, and heat treatment, according to conditions prescribed by Yamamoto et al. [12]. Ten percent cold-work was applied to the model alloys before final sample machining. The as-received microstructures of the model FeCrAl alloys were dominated by dislocation networking and dislocation cell structures [13]. Kanthal APMT, an alloy produced by powder metallurgy techniques including oxide dispersion, was studied alongside these wrought model FeCrAl alloys.

Details for the sample geometries and lower-dose irradiation conditions for the model alloys have been provided in previous work [13,16,17]. All specimens used in those studies and within this study were prepared by a single vendor using wire electric discharge machining to make dog-bone, sheet-type SS-J2 specimens (gage size 5.0 × 1.2 × 0.5 mm). Six specimens per alloy were irradiated for each dose-temperature condition. Neutron irradiations were completed in the central flux trap of the High Flux Isotope Reactor (HFIR) from 0.3 dpa to 13.8 dpa. Irradiations of capsules at 0.3 dpa and 0.8 dpa were completed using the hydraulic tube facility at HFIR; all other capsules were irradiated in static positions.

Neutron flux, and hence dose rate, varied depending on the axial locations of the capsules within the central flux trap of HFIR. Table 2 provides the details of each irradiation condition. Target irradiation temperatures were designed to be 320 °C. The temperatures shown in Table 2 were determined from performing dilatometric analysis of passive silicon carbide (SiC) thermometry samples contained within each irradiation capsule. The dilatometric analysis was conducted up to a maximum temperature of 600 °C at a constant ramp rate of 1 °C/min and a cooling rate of 2.5 °C/min using a Netzsch 402 CD dilatometer. Analysis of individual specimens was completed using the methodology outlined by Campbell et al. [18]. The median temperature derived from the dilatometric analysis was used as the nominal irradiation temperature. Errors for the temperatures in Table 2 are reported as one standard deviation of the mean from at least three separate, randomly selected SiC thermometry specimens from the same irradiation capsule.

Tensile tests on the SS-J2 specimens in the as-received and irradiated state were performed on an Instron universal test machine. Tests were completed using shoulder loading with a crosshead speed of 0.0055 mm/s, resulting in a nominal strain rate of ~10^{-3} s^{-1}. Owing to the unavailability of a contact or non-contact extensometer at the time of the test, all engineering strains were determined from the digitally recorded crosshead separation. Engineering stress was calculated based on the digitally recorded load and measured thickness and width of the gage region before irradiation. Therefore, the calculations here assume negligible linear or volumetric swelling in the specimens due to neutron irradiation. This assumption seems within reason, as Little and Stow showed no swelling in a neutron-irradiated FeCrAl alloy near the temperatures studied here [19]. Room-temperature tests (24 °C) were performed in air, and high-temperature tensile tests (320 °C) were performed in high vacuum (<1×10^{-5} torr). Only a single high-temperature tensile test was performed per alloy for all irradiation conditions