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FOREWORD

This is the thirty-sixth in a series of semiannual technical progress reports on fusion materials science activities supported by the Fusion Energy Sciences Program of the U.S. Department of Energy. This report focuses on research addressing the effects on materials properties and performance from exposure to the neutronic, thermal, and chemical environments anticipated in the chambers of fusion experiments and energy systems. This research is a major element of the national effort to establish the materials knowledge base of an economically and environmentally attractive fusion energy source. Research activities on issues related to the interaction of materials with plasmas are reported separately.

The results reported are the product of a national effort involving a number of national laboratories and universities. A large fraction of this work, particularly in relation to fission reactor irradiations, is carried out collaboratively with partners in Japan, Russia, and the European Union. The purpose of this series of reports is to provide a working technical record for the use of program participants, and to provide a means of communicating the efforts of fusion materials scientists to the broader fusion community, both nationally and worldwide.

This report has been compiled and edited under the guidance of R. L. Klueh and Teresa Roe, Oak Ridge National Laboratory. Their efforts, and the efforts of the many persons who made technical contributions, are gratefully acknowledged.

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effect, which has been reported for non-stoichiometric SiC fiber composites. This difference was discussed from the viewpoints of thermal residual stress, process-induced damages, fiber surface features, and interfacial bonding and friction. A preliminary conclusion is drawn that a thin PyC interphase, as low as ~25nm, will be generally beneficial for both fast fracture and lifetime limiting properties in stoichiometric SiC-based composites.

Oxide dispersion-strengthened (ODS) steels are being developed and investigated for nuclear fission and nuclear fusion applications in Japan, Europe, and the United States. In addition, commercial ODS products are available and have been used in niche applications. Microstructural and mechanical properties studies have been conducted at Oak Ridge National Laboratory and elsewhere on various commercial and experimental ODS steels. Tensile and creep properties have been obtained and collected from literature and commercial sources. These data are compared to show the differences and similarities of different ODS steels, and observations are explained in terms of the microstructures of the steels.

Static lithium compatibility tests have been conducted on vanadium alloys at 800°C.
Mass gains have been detected that are not consistent with prior static Li exposures of unalloyed vanadium and are difficult to interpret based on the impurity levels in the lithium specified by the manufacturer. A second test gave more reasonable results. Some of the mass gain could be attributed to N and C uptake from the Li but this should be countered by O loss from the alloy. Post exposure analysis of the specimens indicated a surface layer rich in O, C and Ca but no indication for the embrittlement observed.

collapsed by direct interaction with gliding screw dislocations. Although Wirth et al. pointed out that truncation of SFT before interaction with gliding dislocations was an important factor for the SFT collapse [1], the present experimental results clearly showed that truncation of SFT was not a crucial factor for the collapse mechanism. An SFT collapsed in a unique way: only the base portion divided by the gliding plane of dislocation annihilated, while the apex portion remained intact. Judging from the fact that similar collapse process was observed in recent computer simulations using molecular dynamics codes [2], this collapse process is a major one. However, it is still unclear whether this is only one process for SFT collapse, because this collapse process leaves a small SFT as a remnant, whereas SFTs are completely swept out in the dislocation channels actually observed in the deformation microstructure of irradiated fcc metals [3-12]. Further research works will be required for fully understanding of dislocation channeling, which is believed to be a key for ductility reduction of irradiated metals [3].

Atomic-scale modelling permits detailed simulation of the interactions between moving dislocations and particular obstacles. Such simulations should be of large enough scale to simulate a realistic dislocation density and obstacle spacing, and correctly treat long-range self-interaction between dislocation segments. Results obtained with a 2 nm void and two different atomic-scale techniques based on (1) a periodic array of dislocations (PAD) and (2) Green's function boundary conditions (GF) in alpha-Fe are presented. Static, zero temperature, simulations have been carried out with incrementally increasing strain until dislocation overcomes the obstacle. It is concluded that both techniques reproduce the same critical resolved shear stress (CRSS), and similar void and dislocation modifications are observed.

Formation energies, binding energies, and the migration of interstitial He atoms at and near the center of an a/2<111>{110} edge dislocation in α-Fe are determined using molecular dynamics and conjugate gradient relaxation methods. Results are compared as a function of the distance of the interstitial He atoms from the center of the dislocation and the amount of excess volume around the dislocation. Interstitial He atoms have negative binding energy on the compression side of the dislocation and strong positive binding energy on the tension side. Even at low temperatures, interstitial He atoms in the vicinity of the dislocation easily migrate to positions near the center of the dislocation, where they form crowdion interstitials with binding energies in excess of 2 eV.