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Recent progress of hydrogen isotope behavior studies for neutron or heavy ion damaged W

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HIGHLIGHTS

- This paper reviews recent results pertaining to hydrogen isotope behavior in neutron and heavy ion damaged W.
- Accumulation of damage in W creates stable trapping sites for hydrogen isotopes, thereby changing the observed desorption behavior.
- The distribution of defects throughout the sample also changes the shape of TDS spectrum.
- Experimental results show that production of Re by nuclear reaction of W with neutrons reduces the density of trapping sites, though no remarkable retention enhancement is observed.

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ABSTRACT

This paper reviews recent results pertaining to hydrogen isotope behavior in neutron and heavy ion damaged W. Accumulation of damage in W creates stable trapping sites for hydrogen isotopes, thereby changing the observed desorption behavior. In particular, the desorption temperature shifts higher as the defect concentration increases. In addition, the distribution of defects throughout the sample also changes the shape of TDS spectrum. Even if low energy traps were distributed in the bulk region, the D diffusion toward the surface requires additional time for trapping/detrapping during surface-to-bulk transport, contributing to a shift of desorption peaks toward higher temperatures. It can be said that both of distribution of damage (e.g. hydrogen isotope trapping sites) and their stabilities would have a large impact on desorption. In addition, transmutation effects should be also considered for an actual fusion environment. Experimental results show that production of Re by nuclear reaction of W with neutrons reduces the density of trapping sites, though no remarkable retention enhancement is observed.

1. Introduction

Due to the high tritium solubility in graphite, tungsten (W) will be the primary plasma-facing material candidate for fusion. It is reported that lower hydrogen isotope retention and higher thermal conductivity are favorable qualities for usage of W in fusion reactor [1–4]. However, recent studies show that tritium dynamics will be clearly controlled by the characteristics of W material, the damage profiles introduced by neutrons and other energetic particles, accumulation of impurities like He and/or C on the W surface, and transmutation of W by neutrons.

The fundamental characteristics of the W material clearly control the damage profiles. Tanabe has reviewed the behavior of W material characteristics; for further details we refer the reader to Ref. [5]. Vacancies become mobile above 800 K and enhance grain growth, whereas impurities tend to segregate at grain boundaries. These processes weaken the bonding between the grains and cause a significant increase of ductile to brittle transition temperature (DBTT). Ultimately, this potentially limits the utilization of W as a structural material [5]. In addition, the size and direction of grains have a large impact on hydrogen isotope retention and diffusion.

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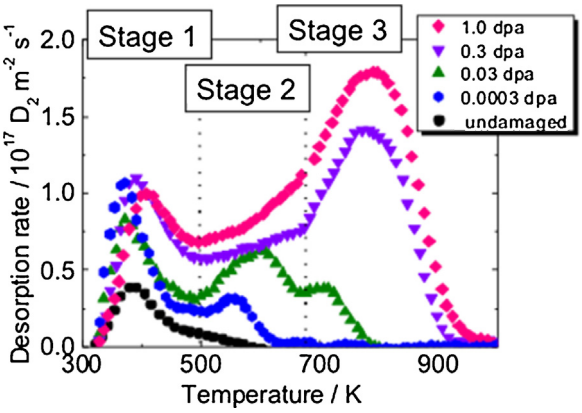

Fig. 1. D_2 TDS spectra for Fe^{2+}-damaged W with various damage concentrations [8]. (1 keV D_2^+ with 1.0×10^{22} D m^{-2}).

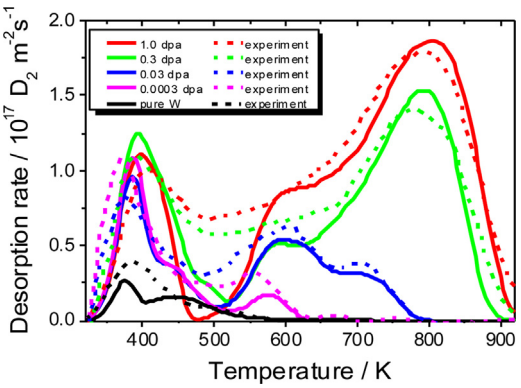

Fig. 2. D_2 TDS simulation results for Fe^{2+}-damaged W [8].

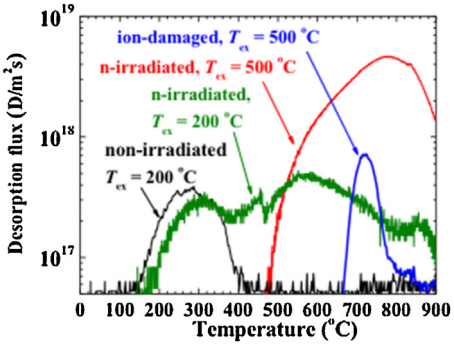

Fig. 3. D_2 TDS spectra for 0.025 dpa neutron-damaged W, 0.5 dpa Fe^{2+}-damaged W and non-irradiated W. [15].

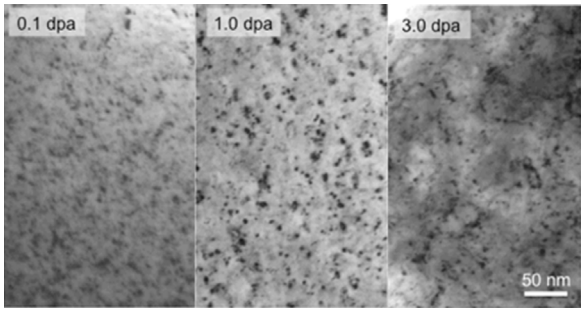

Fig. 4. TEM images (dislocation loops) for 0.1, 1.0 and 3.0 dpa damaged W. [8].

Hasegawa has reported about the major damage profiles within W after neutron irradiation by JOYO, HFIR and JMTR [6]. The main defects produced by neutron irradiation will be dislocation loops at lower damage concentration, but the accumulation of defects clearly changes the nature of material; voids will be the major form of damage in W under such conditions. This microstructure change clearly controls hydrogen isotope behavior, and various trapping sites with different trapping energies will be produced by neutron irradiation.

The production of He by nuclear reaction of T with D will also modify the surface nature due to nucleation of He bubbles and accompanying damage evolution within the near-surface region. In our previous paper, He^+ implantation induced higher deuterium retention even if the He^+/D^+ flux ratio was low [7]. Under such circumstances, the main D_2 desorption stage occurred at temperatures less than 700 K. A large amount of D desorption between 400 and 700 K was found, indicating that He^+ irradiation would introduce defects and its retention would prevent the D diffusion toward the depth and/or D desorption. The precipitation of C would also change the hydrogen isotope behavior and stable trapping site was formed at higher temperature side.

The interaction of irradiation-induced defects and hydrogen isotopes is envisioned to be one of the most prominent problems for plasma-facing materials, particularly because the damage produced by neutrons will be distributed throughout the sample. Therefore, this paper reviews recent research activities about hydrogen isotope behavior for neutron and/or heavy ion damaged W.

2. Damage concentration dependence

The damage profile is known to be changed based on the concentration of defects. As mentioned in the introduction, the dislocation loops are expected to be the dominant form of defect at lower damage concentrations. However, the density of dislocation loops nearly saturates at 0.1 dpa. Above this damage concentration, vacancies and voids are introduced and hence, the hydrogen isotope retention profile is dramatically changed. Fig. 1 shows the D_2 TDS spectra for Fe^{2+} damaged W with damage concentrations ranging from 0.0003 dpa to 1.0 dpa [8]. For the un-damaged W, D_2 desorption stages were concentrated at lower temperatures, less than 500 K. However, as displacement damage was introduced, a second desorption stage (Stage 2) was found at temperatures between 500 K to 700 K. Above 0.03 dpa, additional desorption stage (Stage 3) was found at the temperature more than 700 K. The desorption temperatures for both of Stages 2 & 3 shifted toward higher temperatures as damages are accumulated. The hydrogen isotope diffusion and trapping (HIDT) simulation was applied to aid in the interpretation of these D desorption behaviors. The simulation results shown in Fig. 2 reveal that the D_2 desorption behavior can be accurately simulated using three trapping sites with energies of 0.65 eV, 1.25 eV and 1.55 eV, which were almost consistent with previous reports, corresponding to the desorption of D trapped by dislocation loops, vacancies and voids, respectively [8–14]. Major desorption stage for W with lower damage concentration less than 0.03 dpa was concentrated on lower temperature, which can be attributed to the desorption of D trapped by dislocation loops and/or surface. The Transmittance electron microscope showed the saturation of dislocation loops was observed as shown in Fig. 4 [8]. As the damage concentration increased, additional large desorption stages were found at higher temperature side, indicating that the vacancies and voids would work as a major D trapping site. In addition, HIDT simulation has revealed that the shift of desorption temperatures for Stage 2 & 3 can be correlated with the density of these trapping sites. In the case of higher damage concentration, D diffusion toward the surface would be delayed due to repetition

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