# Interaction of Low Energy Deuterium Plasma With Tungsten and Nano-Structured Tungsten Coatings: from Laboratory Data to Tokamak

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The objective of this work is to compare the deuterium retention in bulk tungsten and tungsten coatings exposed to deuterium plasma in well-defined laboratory conditions and complicate tokamak conditions and make reliable prediction for future fusion reactors.

**Keywords:** nano-structured tungsten coatings, deuterium plasma, radiation-induced damage

## 1 INTRODUCTION

Particle retention in tokamak walls is a key issue for long time discharges in future thermonuclear fusion reactors. Tungsten (W), its alloys and coatings are used as plasma-facing materials in present tokamaks as ASDEX Upgrade (AUG) and JET and selected to be used in future fusion devices. Prediction of fuel retention in plasma-exposed wall materials for ITER and future fusion reactors depends on the knowledge of the fundamental retention mechanisms in these materials. Laboratory experiments on plasma-W interaction allow a detailed study of hydrogen behavior in tungsten in well-defined experimental conditions. A comparison of laboratory data with tokamak data using same materials gives us a platform for under-standing of key parameters govern tritium retention in present tokamaks and reliable extrapolations towards future fusion devices.

## 2 EXPERIMENTAL

In the present work, two types of bulk polycrystalline W (with grains oriented parallel, WP, and perpendicular, WI, to the incident plasma) and two types of W coatings (vacuum plasma sprayed and nanostructured tungsten) were used. Dense nanostructured tungsten coatings were produced on fine grain graphite by Combined Magnetron Sputtering and Ion Implantation (CMSII-W) at the Romanian Association MEdC [1]. Those samples were exposed to low energy well-defined deuterium (D) plasma generated by electron-cyclotron resonance (ECR) plasma source. Experiments were performed at different sample temperatures from 320 to 700 K. The D retention in each sample was subsequently analyzed by nuclear reaction analysis (NRA) for the depth profiling up to 6 μm. Total D retention was found by thermal desorption spectroscopy (TDS). Laboratory data were compared with same materials used as outer divertor target in AUG [2].

## 3 RESULTS and DISCUSSION

Fig. 1 shows an example of the D retention in CMSII-W coating obtained from TDS and NRA after exposure in AUG [2] and by TDS after plasma exposure in lab conditions.
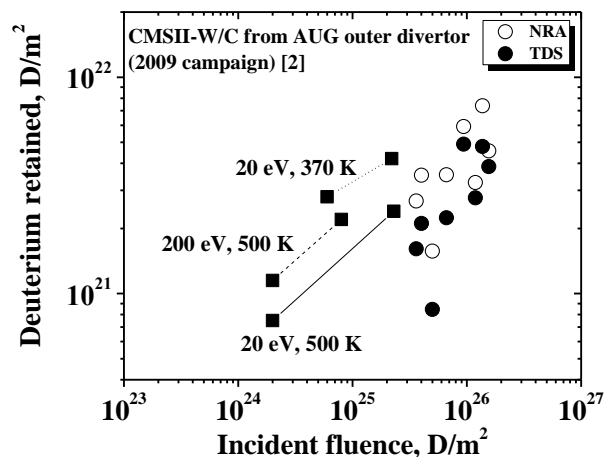


*Fig. 1. The D retention in CMSII-W coating (10 μm) on carbon substrate after plasma exposure in AUG (circles) and in lab (squares) as a function of incident D fluence.*

As it was point out in [2], surface temperature of AUG samples during a discharge is 400 - 500 K in most cases. The lab data for 20 eV and 500 K are in a very good agreement with AUG data.

The D concentration in W layers measured by NRA was considerably higher than that one in polycrystalline W in both laboratory [3] and AUG [2] experiments because W coatings have high intrinsic defect density [3]. It was shown that the D concentration in CMSII-W coating is approximately 0.5 at.% up to 6 µm in depth [3] with increasing of the deuterium concentration in the interlayer between W coating and substrate. Consequently, the D retention in the interlayer between different materials can be a concern. The D concentration in polycrystalline W is about $10^{-2}$ at.% at 6 µm. The D retention in bulk W with grains elongated parallel to the incident plasma shows lower D retention than for W with grains elongated perpendicular. It is reasonable to suggest that such difference is due to different migration ability of D along and across to the grain elongation.

Fig. 2 shows the D retention in bulk WI and WP as well as in CMSII-W film obtained from TDS data in lab experiments versus AUG data [2] as a function of the incident fluence. The substantial fraction of D in CMSII-W coating is trapped only in 10 µm of coating [3], whereas the D can migrate deeper into the bulk in the case of polycrystalline W. This is a reason that the D retention at AUG shows only slightly higher D retention in CMSII-W film compared to that one in a bulk W at high fluences $>10^{26}$ D/m$^2$ as one can see on Fig. 2. Moreover, the D retention in bulk W has a maximum at around 500 K [3]. Whereas the D retention decreases monotonically with increasing of the temperature in the case of CMSII-W. At this temperature of 500 K, the D retention in polycrystalline W can be only by a factor of two less compared to that one in CMSII-W film. The D retention in CMSII-W film is considerably higher (orders of
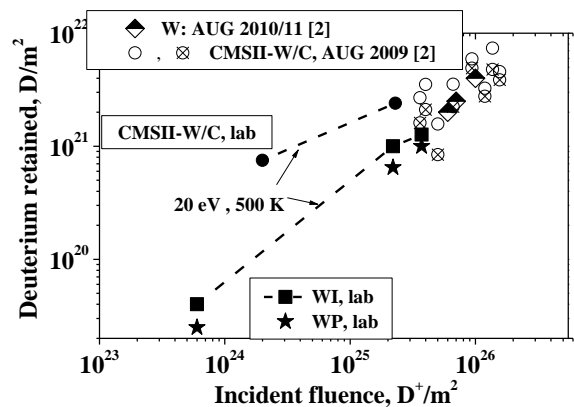


*Fig. 2. The D retention in polycrystalline WI and WP and CMSII-W coating (10 µm) on carbon substrate after plasma exposure in AUG [2] and in lab as a function of incident D fluence.*

magnitude) than in a bulk W at elevated temperatures >600 K.

The incident deuterium flux in lab plasma exposure was ~$10^{20}$ D/m$^2$s while the flux in AUG was ~$10^{22}$ D/m$^2$s [2]. In spite of difference in the incident flux, the overall D retention behaviour in W materials exposed to the D plasma at AUG experiments was reasonably in the range of predictions based on laboratory studies.

Calculations using a diffusion-trapping model showed a good agreement with the laboratory data and with the observations at AUG and, therefore, were used for extrapolations to future fusion devices. In contrast to the present tokamak, W will be subjected to neutron (n) irradiation in future fusion devices. To simulate n-like damage at low dpa (displacement per atom), self-ion irradiation was used. The radiation-induced defects were produced by irradiation of targets with 20 MeV W ions up to 1 dpa as described in [4]. The damaging rate was ~$2\times10^{-4}$ dpa/s (approximately 0.5 dpa per hour). This radiation-induced damage creates additional trapping sites for deuterium and significantly increases the D inventory in polycrystalline W but only slightly increases the D inventory in W coatings. Fig. 3 shows the D concentration at radiation-induced

defects as a function of irradiation dose for W coating and bulk W. Due to high initial concentration of intrinsic defects in CMSII-W, the irradiation with self-ions does not result in a significant increase of the total density of defects (open circles are D concentration in undamaged CMSII-W and closed circles are D concentration at radiation-induced defects). In contrast, pre-irradiation of polycrystalline W with self-ions results in an increase of the D retention by orders of magnitude. At irradiation dose of 0.5 dpa, the same concentration of radiation-induced trapping sites for deuterium in different W grades was found independent on the initial density of defects. Therefore, the structure of the W material will play an important role in the D retention in the beginning of the plasma operation in the future fusion device, whereas after several hundreds pulses, the D retention will not be depend on the W grades.
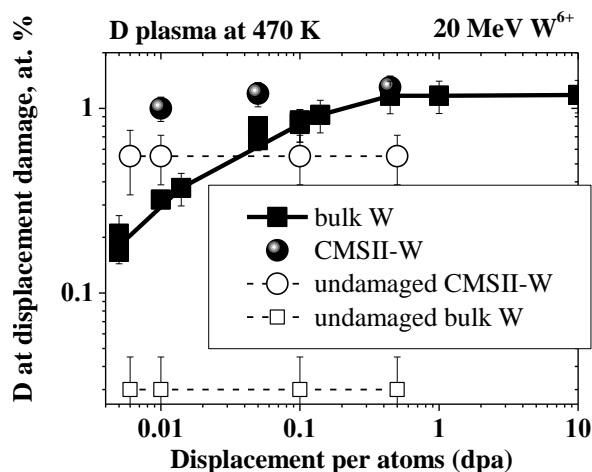


*Fig. 3. The D concentration at radiation-induced defects in CMSII-W coating (10 $\mu$m) on carbon substrate (circles) and in bulk W (squares) pre-irradiated with 20 MeV W ions at different doses and subsequently exposed to 20 eV D plasma at 470 K. D concentrations in undamaged (0 dpa) W materials are shown as open symbols.*

## 4    CONCLUSION

The D retention in nano-structural W coatings and bulk W exposed to AUG shows reasonable agreement with laboratory deuterium plasma experiments. On the basis of present experimental results, extrapolation of the tritium retention in polycrystalline bulk W and dense nano-structural W coating to ITER and DEMO conditions taking into account effect of 14 MeV neutrons can be done.

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