



15.2. Plasma Material Interactions

(Session Organizers: Y. Hirooka, N. Noda, C.H. Wu;

Invited Speakers: S. Takamura, T. Haasz, Yu.V. Martynenko, A. Hassanein)

Plasma-facing materials mixing and mixed material properties

1. Session organization

Upon agreement among the session organizers and invited speakers, this session on plasma–material interactions (PMI) focussed on materials mixing and mixed material property issues. Materials mixing as a result of erosion and redeposition is virtually unavoidable if two or more different plasma-facing materials are used in a single magnetic fusion device, but mixed material properties are unknown. These issues have attracted considerable attention from the PMI-research community, since the advent of the ITER divertor cassette design in which beryllium, carbon and tungsten are employed as components, closely positioned to one another. Although as such, ITER generated the interest originally, as shown in Table 1, one finds many potential opportunities for materials mixing in fusion devices presently in operation as well as those under design study.

First, Y. Hirooka of UCSD gave a keynote address on the roles of plasma impurities in materials erosion and mixing. Then, the invited speakers made short comments on the physics of impurity transport and mixed material properties. These speakers were Professor S. Takamura of Nagoya University, Professor T. Haasz of Toronto University, Dr Yu.V. Martynenko of Kurchatov Institute, Dr A. Hassanein of Argonne Nat. Lab. Also, the session organizers, N. Noda of NIFS and C. Wu of IPP-Garching gave short talks. Then, the session was opened for general discussion for a while, soliciting opinions from the audience. The speakers will be referred to only by their surnames in the remainder of this summary.

These presentations may be summarized as follows (not in order of deliveries):

1. Takamura gave a tutorial talk about the basic physics involved in impurity transport.
2. Hirooka described critical conditions under which plasma impurity deposition occurs.
3. Haasz overviewed his database on doped-graphite erosion and hydrogen retention.
4. Noda described the trapping and release of hydrogen by boronized coatings.
5. Martynenko presented the data on Ti-doped graphite erosion and hydrogen retention.
6. Wu reviewed recent efforts on Si-doped materials development and characterization.
7. Hassanein predicted the behavior of mixed materials under disruption conditions.

2. Physics of impurity transport

Takamura gave a tutorial talk on the basic physics of impurity transport, a subject hard to find the opportunity to learn about in the fusion reactor materials research community. Particularly, the knowledge about ionized particle flows in the toroidal confinement system helps us predict where and how far impurities can be transported along as well as across the magnetic field.

In the case of impurity flow in the background DT plasma along the magnetic field, which is often referred to as ‘parallel transport’, the equation of motion of impurities has been derived by Braginski [1] in the following manner:

$$F = -\frac{\partial p_z}{\partial s} + n_z Z_e E + C_e n_z \frac{\partial T_e}{\partial s} + C_i n_z \frac{\partial T_i}{\partial s} + m_z n_z \frac{v_p - v_z}{\tau_z}. \quad (1)$$

Because the expertise of most of the readers is expected to be materials science or related areas, details regarding Eq. (1) are not discussed here. However, it is important to describe physical meaning of the right-hand terms. The first term is the force due to pressure gradient, the second term is due to electric field, the third and fourth are the thermal force and the last one is the friction force. Depending on the way this force is applied, impurities may change their flow direction along with the magnetic field, which is referred to as ‘impurity flow reversal’. For the radiative divertor operation, for example, the friction force must be maintained effective

Table 1
Possible cases of materials mixing in fusion devices

Devices	Limiter (first wall protection)	Divertor	Fuel
<i>Future devices</i>			
ITER	Be	W + C	D + T
Starlite	V	W	D + T
NSTX	C (+B) ^a	C (+B) ^a	D
<i>Existing devices</i>			
JET	Be + C	Be + C	D + T
TFTR	C (+B + Li) ^a		D + T
TEXTOR	C (+B + Si) ^a		D
ASDEX-U	C (+B) ^a	W + C (+B) ^a	D
ALCATOR C-Mod.	Mo (+B) ^a		D
JT-60U	C (+B)	B ₄ C + C (+B) ^a	D
LHD, DIII-D	C (+B) ^a	C (+B) ^a	H, D
W-AS7	B-doped C (+B) ^a		D

^a Wall conditioning coatings.

Remark: Residual gas impurities containing carbon and/or oxygen and also unprotected first wall materials such as stainless steel can participate in materials mixing.

such that impurities will not be transported too far in the upstream, leaving the divertor region.

Impurities can flow across the magnetic field, i.e., ‘cross-field transport’, which is just as important as the parallel transport in understanding the dynamics of plasma-facing materials mixing. To understand how these two transport processes are related, let us consider boundary layers between the core plasma, scrape-off layer and plasma-facing components, as shown in Fig. 1. First, we define the characteristic cross-field diffusion length for ionized impurities as follows:

$$\lambda_m = \sqrt{\frac{D_{\perp} L_c}{v_{\parallel}}}, \quad (2)$$

where D_{\perp} is the perpendicular diffusivity, L_c is the magnetic field line connection length between the points of our concern and v_{\parallel} is the parallel velocity. Notice that cross-field diffusion is related to the time taken by impurity particles to go around the system on the magnetic field line, i.e., the connection length. One finds a very

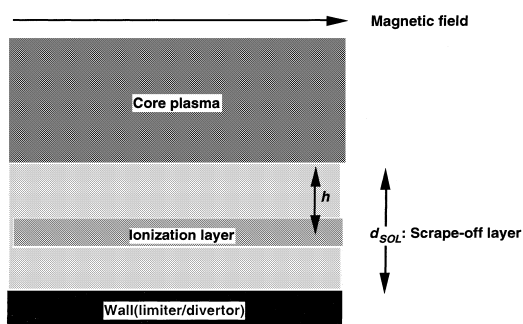


Fig. 1. Schematic illustration of plasma boundaries.

similar expression in materials science, \sqrt{Dt} , which gives the characteristic diffusion length.

In the following impurity transport analysis, it is important to emphasize that impurities are all ionized to begin with. Here, let us first consider the case where materials are ejected, via sputtering, and then ionized in the scrape-off layer. Predicted destinations of these impurities due to cross-field transport are:

1. transported to the divertor if $\lambda_m < h$; and
2. transported into the core if $\lambda_m > h$.

Next, we assume that all impurities are in the core before cross-field transport occurs. Then, transport destination criteria for these impurities are:

1. transported to the divertor if $\lambda_m < d_{\text{SOL}}$; and
2. transported to the first wall if $\lambda_m > d_{\text{SOL}}$.

Let us examine the following two cases using the above-mentioned criteria: where divertor impurities go, and where first wall impurities go. Taking an ITER-relevant configuration as example [2], we assume that $D_{\perp} = 0.2 \text{ m}^2/\text{s}$, $T_e = 50 \text{ eV}$ at mid-plane and 10 eV at divertor plate, $v_{\parallel} = 0.3 C_s$ where C_s is the sound velocity, $d_{\text{SOL}} = 1.5 \text{ cm}$ at mid-plane 6.5 cm at divertor plate, and $L_c = 20 \text{ m}$ from X-point to divertor, 50 m from X-point to mid-plane and 80 m from divertor to mid-plane. Under these conditions, the characteristic diffusion length for carbon is $\lambda_m = 3.3 \text{ cm}$ which is smaller than the SOL length, $d_{\text{SOL}} = 6.5 \text{ cm}$. One predicts in this case, that the eroded carbon from the divertor will be ionized in the scrape-off layer and then transported crossing the magnetic field to tungsten components, i.e., divertor materials mixing. If $L_c = 80 \text{ m}$ is the case, $\lambda_m = 6.6 \text{ cm}$, meaning that carbon eroded from the divertor can be transported to the first wall, i.e., first wall-divertor materials mixing. For beryllium eroded from the first wall, assuming $L_c = 50 \text{ m}$, we obtain $\lambda_m = 3.2 \text{ cm}$, which,

again, is smaller than $d_{\text{SOL}} = 6.5$ cm. Therefore, beryllium is likely to be transported to the divertor region.

From these examples, one expects that materials mixing in the divertor region will involve not only materials eroded from the local components but also those from the first wall region. The question is what will happen to the surfaces subjected to the bombardment of these ionized impurities simultaneously with DT-fuel particles? More specifically, it is important to investigate whether or not these impurities will affect host materials erosion behavior. And, if they do enhance or reduce materials erosion, under what conditions will that happen? These were the issues discussed in the keynote address by Hirooka.

3. Redeposition of plasma impurities

In his keynote address, Hirooka reviewed several examples to demonstrate the effects of plasma impurities on materials erosion and mixing observed in the recent PISCES experiment and he then described first-order modeling analysis of these observations. Details can be found in his paper in part A of these proceedings [3]. For completeness, a brief summary of his talk is presented here. Reviewed in the first example was the observation that film deposition occurs when beryllium is bombarded by deuterium plasmas contaminated with carbon impurities such as those in the form of C_xH_y , a phenomenon referred to as ‘carbon-poisoning’ [4]. As a result, beryllium erosion is reduced due to the shielding effect of the resultant film, which is encouraging from the erosion lifetime viewpoint. In fact, reduced beryllium erosion has been observed in JET, which is also attributed to carbon poisoning [5]. However, this means that beryllium components can act as a large-area substrate for tritium co-deposition with carbon impurities, a serious safety issue. This question will be answered later in this summary.

From the zero-dimensional materials balance analysis, the following condition must be met for impurity film deposition to occur under simultaneous DT-particle bombardment:

$$\sum \left(\frac{\Gamma_i}{\Gamma_{D,T}} \right) > \frac{s Y_{D,T \rightarrow i}}{(v_i - s_s Y_{i \rightarrow i})}. \quad (3)$$

This criterion needs an adjustment for carbon impurity cases, particularly with a low-Z substrate. For example, carbon film deposition on beryllium was observed only when the surface temperature was above 500 or so [4]. This is because as observed in a separate experiment [6], outward diffusion of beryllium occurs to penetrate carbon deposits, which then reduces the carbon chemical erosion rate due to impurity catalysis. Therefore, reduced erosion yield data must be used for the numerator of the right-hand side of Eq. (3). In other words, at el-

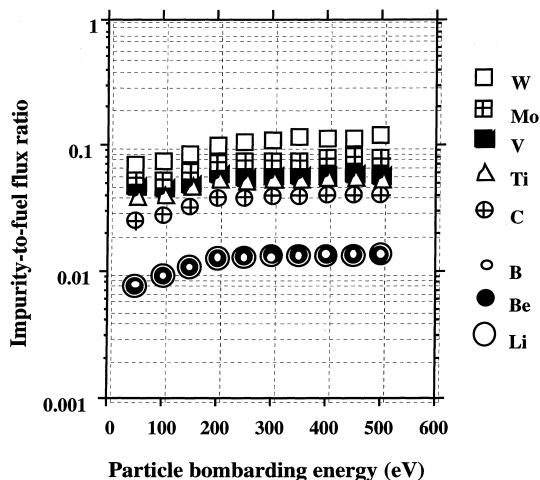


Fig. 2. Critical carbon contents for film deposition under simultaneous deuterium plasma bombardment at normal incidence (reproduced from Ref. [3]).

evated temperatures beryllium becomes more susceptible to carbon film deposition. This two-step materials mixing, i.e., impurity deposition followed by diffusion, must be considered if carbon and beryllium are used for plasma-facing components. Although experimental confirmation is needed, one expects similar behavior for other low-Z materials such as boron and lithium. These cases may be related to the wall behavior under boronization and lithium-conditioning. To summarize, the carbon impurity film deposition criteria are shown in Fig. 2.

Let us consider the possible consequences of impurity deposition. For example, if a beryllium host is covered completely with carbon, the surface should act like carbon. As soon as this occurs, as shown in Fig. 2, the critical concentration for impurity deposition increases by about a factor of 3. Therefore, carbon deposits will be eroded until beryllium is exposed again. These erosion and deposition processes may repeat themselves at rather high frequencies. Needless to say, experimental investigation is necessary to prove this hypothesis.

One would then raise the question ‘what happens to the host material if carbon film deposition does not occur?’. This case has also been demonstrated in the recent PISCES experiments on tungsten bombarded with oxygen-contaminated deuterium plasmas. These data are partially reproduced in one of the invited papers in part A of these proceedings [7]. Clearly, oxygen is not a condensable impurity so that there is no such case as film deposition. Nonetheless, these data clearly demonstrate that non-condensable impurities will enhance host materials erosion. More details on the modeling of impurity-enhanced erosion of tungsten are presented in Hirooka’s contributed paper in these proceedings [3].

Let us examine another case on impurity-enhanced erosion, related to ITER. As pointed out in Takamura's talk, carbon impurities from the divertor as well as beryllium impurities from the first wall can be transported to the tungsten components in the divertor. For a tungsten substrate, however, a 10% carbon contamination is necessary for film deposition, as shown in Fig. 2. For non-normal incidence, to be more relevant to the reactor situation, the minimum carbon concentration becomes significantly higher [8]. This is because in Eq. (3), the trapping coefficient decreases and the self-sputtering yield increases. For example, the critical concentration exceeds 50% for beryllium impurities bombarding tungsten. One would not expect this level of impurity contamination in the host DT-plasma unless under some off-normal condition. This means that tungsten components are likely to suffer from impurity-enhanced erosion in exchange for surface cleanliness.

4. Mixed material properties

Because plasma-facing materials mixing takes place in an operating fusion device, it is impossible to prescribe materials composition and structure for the sake of making specimens for ex situ laboratory experiments. On the other hand, it is also true that the existing confinement devices are operating to carry out their machine-specific missions, not necessarily pertinent to materials research. Furthermore, regardless of machine time availability, it is virtually impossible to prepare well-controlled specimens in a fusion device where a variety of plasma-material interactions take place, including even disruptions. The DIII-D team has been making efforts on this aspect using the DiMES facility, a materials erosion and deposition probe [9]. Also recently, similar probe work has been done in TFTR [10]. Nonetheless, materials mixed in DIII-D and TFTR have their own characteristics due to machine configuration and due to the host material that covers the majority of the interior surface. The issue here is that it is extremely difficult to predict precisely the ways that materials will be mixed with one another and how mixed materials will interact with edge plasmas in a reactor under design.

Having pointed this out as the basic principle, one can still make first-order predictions by extrapolating from the conventional wisdom. The session organizers believe that the data review on doped materials erosion and hydrogen retention data will give us a hint as to in which way the future research should be directed. Based on these thoughts agreed among the session organizers, the following presentations were invited at this discussion session.

First, Haasz reviewed the data from his experiments on erosion and hydrogen retention. His database covers a variety of 'hypothetical' mixed materials consisting of

carbon as the base material in the form of graphites and C-C composites and impurities including B, Si, Ti, Ni, W, TiB₂, etc. It is widely known that these impurities enhance the recombinative desorption of implanted hydrogen and thus reduce chemical sputtering [11]. Therefore, these data are not reproduced in this summary.

Caution must be taken here because it is well known in the HTGR community, but probably not in the fusion community that metallic impurities on graphites can significantly enhance oxidation [12]. This is a serious safety issue for the operation of a fusion reactor if graphite is used as the plasma-facing material to cover a large fraction of the interior area. This is because at an accidental vacuum loss, high-temperature oxidation of graphite due to water vapor molecules in the air leads to the formation of hydrogen, which can then be mixed with oxygen to explode. Unfortunately, graphite oxidation reaction is exothermic, meaning a self-maintaining reaction. These oxidation-enhancing impurities include Ni, Ti and W [12]. After these screenings, only B and Si seem to remain as the possible dopants to reduce oxidation as well as chemical sputtering of graphites. If radiation-enhanced sublimation is included in this argument as well, B appears to be the sole survivor as an effective dopant.

Regarding the use of boron in a fusion reactor, however, one must take into account the fact that boron-10 absorbs thermal neutrons due to the reaction: $^{10}\text{B}(n, \alpha)\text{Li}^7$. The recoil alpha and lithium particles will generate lattice damage. This will require an isotope-tailoring procedure to enrich boron-11. Interestingly, boron-10 is a material, referred to as a 'burnable poison', in need for the control rods in BWRs. Therefore, if in the end boron-11 is chosen as the dopant in graphite or as the wall-conditioning agent (e.g., boronization), the enrichment cost can be shared between the fission and fusion power industries.

Turning to the hydrogen retention data, Haasz presented several examples showing the effect of materials mixing. Included in these examples are B, Si, Ti, and W as dopants. As shown in Fig. 3, reproduced from Ref. [13], the hydrogen retention in doped graphites is generally larger than that in pure graphite. At fluences around 10^{24} D ions/m², for example, Si-doped graphite absorbs about three times more deuterium than pure graphite.

It is important to mention here that in these data, the deuterium retention increases as bombarding fluence increases even for pure graphite [14]. At the end of the first 1000-s discharge in ITER, which is probably of the order of 10^{26} D ions/m² at the divertor plate, the difference will be about a factor of 6. This prediction does not seem to support Si-doping of graphite although clearly, further experimental work is necessary. In contrast, B-doping seems to have almost no effect on

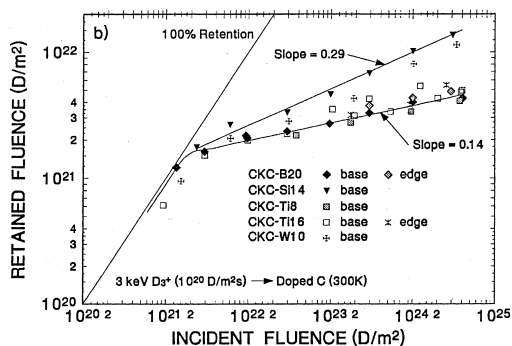


Fig. 3. Deuterium retention in doped graphites (reproduced from Ref. [13]).

hydrogen retention. This is consistent with the observation that the re-emission of deuterium from boronized graphites is enhanced, relative to pure graphite, to reduce chemical erosion [15].

Related to boron effects, Noda presented hydrogen absorption and desorption data from the recent experiments [16,17]. The important findings are summarized as follows. From the TDS data, hydrogen trapped in boronized coatings is found to be released almost completely at temperatures below around 400°C, significantly lower than in the case of graphite. This allows us to predict that if boronized first walls are maintained at these elevated temperatures, the tritium retention will be significantly reduced. Furthermore, if boronized coatings are pre-exposed to oxygen, hydrogen retention is further reduced.

Earlier in this summary, an increase in tritium inventory was predicted if carbon film formation occurs on beryllium. Related to this point, hydrogen retention in carbon deposits on beryllium at elevated temperatures has recently been measured to be less than one would expect, due to the reaction: $[CH_n]_m + 2Be \Rightarrow Be_2C + (m/2) H_2$ [18]. This is a great relief to the ITER community. Experiments are under way to investigate if a similar reaction can take place with a high- Z material as the substrate [19]. In any case, hydrogen retention in carbon deposits may decrease, depending on the substrate and surface temperature. Although thermodynamic data may give us some predictions, as to what substrate material works well, further experimental work is necessary.

Following Noda's comments, Martynenko talked about the recent observations on the deuterium retention in Ti-doped graphite. Details can be found in his paper presented at the conference [20]. Used in his experiments are the steady-state plasma device, LENTA, and the disruption simulator, MKT. The disruption-simulated plasma bombardment can release a large fraction of previously implanted deuterium. This means that disruptions provide the heat that helps release

deuterium rather than an increase in deuterium retention. He also claims that the surface topography change induced by disruption-simulated plasma bombardment enhances subsequent steady-state plasma erosion.

Wu presented the data on material characterization on Si-doped CFC, developed for ITER with the silicon concentration ranging between 8 and 10 at.%. Predictably, the chemical erosion yield of this material is found to be about a factor of 2–3 lower than that of pure graphite. Also reported was the observation that at elevated temperatures ranging up to 1500°C, the surface of Si tends to be depleted whereas the opposite is true in the subsurface layer. This subsurface silicon enrichment has been found to be as high as 50 at.%. More details can be found in his contributed paper in part A of these proceedings [21]. As to hydrogen retention, however, data are clearly contradictory: reduced retention has been observed in Wu's experiments [22] whereas the opposite is reported by Haasz, as described earlier in this summary (see Fig. 3). The materials quality control seems to be a key issue on the development of doped graphites.

Finally in the presentation session, Hassanein predicted with confidence that mixed materials on the surface will probably not make any substantial differences in the behavior under ITER-related disruption conditions, based on his modeling work on pure materials including carbon, beryllium and tungsten.

5. Open session and summary

After these invited talks, the session was then opened for general discussion to have inputs from the audience. As usual (?), however, something unexpected happened. Because the use of actual fusion devices for mixed materials experiments would require tremendous amount of resource, the session organizers and invited speakers believe that it would be best to re-visit the existing data such as erosion or hydrogen retention and continue on the basic laboratory experiments on 'hypothetical mixed materials' such as doped graphites. In fact, this was the idea behind organizing the session. Prior to ICFRM-8, a technical consultants' meeting was held at IAEA [23]. The attendees include Hirooka and Haasz. At this meeting, too, the use and extension of existing databases was unanimously adopted as the baseline strategy for mixed material properties research. Surprisingly, however, most of the opinions raised from the audience were to run actual fusion devices to generate actual mixed materials, a fundamental disagreement. Obviously, there was not enough time to find the middle ground. As a result, the session did not reach a consensus as to how to address and resolve materials mixing issues. Nonetheless, the session organizers believe that this discussion has at least provided an opportunity for brainstorming

and stimulating interest among the scientists in the PMI community.

Acknowledgements

Special thanks go to Professor S. Takamura of Nagoya University for making extensive efforts to illustrate the physics of impurity transport in layman's terms to the fusion reactor materials community. Useful comments by Professor K. Ashida of Toyama University on the hydrogen retention properties in mixed materials are greatly appreciated.

References

- [1] S.I. Braginski, *Sov. Phys. JETP* 6 (1958).
- [2] R. Aymar et al., *J. Nucl. Mater.* 258–263 (1998) 56.
- [3] Y. Hirooka, *J. Nucl. Mater.* 258–263 (1998) 1045.
- [4] Y. Hirooka et al., *J. Nucl. Mater.* 230 (1996) 173.
- [5] H. Guo et al., *J. Nucl. Mater.* 241–243 (1997) 385.
- [6] K. Ashida, K. Watanabe, *Fusion Eng. Design* 37 (1997) 307.
- [7] N. Yoshida, Y. Hirooka, *J. Nucl. Mater.* 258–263 (1998) 173.
- [8] Y. Hirooka, *Fusion Eng. Design* 37 (1997) 299.
- [9] D. Whyte et al., *J. Nucl. Mater.* 241–243 (1997) 660.
- [10] Y. Hirooka et al., Deposition of lithium on a plasma edge probe in TFTR, Presented at the 13th International Conference on Plasma–Surface Interactions in Controlled Fusion Devices, San Diego, 18–22 May 1998.
- [11] J. Roth et al., *J. Nucl. Mater.* 191–194 (1992) 45.
- [12] E.A. Heintz, W.E. Parker, *Carbon* 4 (1966) 473.
- [13] A. Haasz, J.W. Davis, *J. Nucl. Mater.* 232 (1996) 219.
- [14] J.W. Davis et al., *J. Nucl. Mater.* 176&177 (1990) 992.
- [15] Y. Hirooka et al., *J. Nucl. Mater.* 176&177 (1990) 473.
- [16] N. Noda et al., *J. Nucl. Mater.* 220–222 (1995) 623.
- [17] N. Noda et al., *J. Nucl. Mater.* 266–269 (1999) 234.
- [18] K. Ashida et al., *J. Nucl. Mater.* 241–243 (1997) 1060.
- [19] K. Ashida et al., *J. Nucl. Mater.* 266–269 (1999) 434.
- [20] Yu.V. Martynenko et al., Ti-containing graphite erosion and D-retention under long stationary and subsequent power pulse plasma action, presented at 8th Int. Conf. on Fusion Reactor Materials, Sendai, Japan, Oct. 26–31 1997.
- [21] M. Balden et al., *J. Nucl. Mater.* 258–263 (1998) 740.
- [22] C.H. Wu et al., Proceedings of the 19th Symposium of Fusion Technology, Lisbon, Portugal, 16–20 September 1996, pp. 327–330.
- [23] IAEA Consultants' Meeting on 'Status of Plasma Induced Erosion Data for Fusion Reactor Materials', 21–22 August 1997, Vienna.