Status of Plasma Facing Material Studies and Issues toward DEMO

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This review paper presents status of research activity of plasma facing materials, mostly tungsten and critical issues towards DEMO reactors. A helium effect on tungsten surface morphology and its impact on fusion reactors and a pulsed heat effect to tungsten are briefly summarized. For DEMO, effects of steady-state operation and heavy neutron irradiation are important subjects to investigate. Present understandings on these are briefly summarized. Finally, issues of helical system towards DEMO will be discussed.

Keywords: plasma facing material, graphite, tungsten, helium effect, neutron effect, deposition control, surface protective coating

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1. Introduction

Plasma facing materials in divertors in fusion reactors are subject to high heat load up to ~10 MW/m^2. To withstand this heat flux, only materials with high thermal conductivity and high melting (sublimation) points can be used. Tungsten and CFC (Carbon Fiber Composite) graphite are the sole candidates. Both materials, however, have concerns: for CFC enhanced erosion of graphite by chemical sputtering, for tungsten cooling of fusion plasma by core accumulation in the burning plasma.

In ITER, serious discussion on the choice of plasma facing materials for divertor is going on. The safety issues are the most critical such as keeping in-vessel tritium retention below the administrative limit (presently set at 700 g [1]) and also amount of dusts (especially dusts on hot surfaces) should be below the limit [1]. In ITER, coolant of water will be used with its inlet temperature of about 100 °C, leading to relatively low wall surface temperature (200~300 °C) except for high heat flux region. Under this temperature condition, any material will potentially contain non-negligible amount of tritium even for metallic materials. It is believed that the use of tungsten is the best choice with regard to these viewpoints, since carbon materials will keep significant amount of tritium in codeposition layers with high T/C ratio (up to 0.4 for plasma facing side).

For DEMO, several operation conditions are essentially different from those of ITER such as steady-state operation (up to several years), high temperature walls (more than 500 °C), and high fluence neutron dose (more than 10 dpa). Under these conditions, tritium retention problem will be probably eased, while neutron effects at elevated temperature will become critical. Tritium permeation from the plasma facing surface to the coolant will need to be properly evaluated.

In this paper, material issues of tungsten and CFC graphite for ITER and DEMO will be reviewed and critical issues for DEMO reactors will be presented.

2. Basic Properties of Graphite and Tungsten

Graphite has been widely used for many magnetic confinement devices, and gives excellent plasma performance and new confinement regime because of less impact on plasma confinement than metallic materials. CFC graphite has also an excellent feature as a divertor material such as non-melting feature and high thermal shock resistance. However, erosion by plasma ion bombardment is quite large due to chemical sputtering. In addition, redeposition layers contain tritium, which would be dominated for in-vessel T retention. Therefore, CFC can probably be used only in the first phase of ITER. Although wall temperature could be high enough to neglect T retention in graphite in DEMO, hydrocarbon transport to remote area, leading to thick deposition with T retention, is serious concerns. In addition, heat shock resistance of CFC materials (NB31) may not be very high under repetitive heat pulse irradiation [2]. For NB31, pitch fibers are arranged perpendicular to a plasma facing surface. The fibers were broken at 10~100 µm from the top, which will be eventually released as dust particles and lead to enhanced erosion.

Tungsten has also good performance under high heat flux plasma exposure because of a high melting point and low sputtering by fuel ions. However, there is quite a concern because of a melting feature and brittleness such as low temperature brittleness, recrystallization brittleness, and neutron irradiation brittleness. Once tungsten melts, material strength is greatly reduced. After solidification,

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yield strength is greatly reduced and internal stresses is generated during a solidification process. These changes will cause cracking and destruction of high heat flux components in the worst case. Helium effects are very important and critical to use tungsten as plasma facing materials. Details will be shown in Sec. 3. The neutron effect will not be a big concern for ITER since neutron fluence to wall materials is not high (up to about 1 dpa for tungsten). But for a steady-state rectors such as DEMO, it would be the most critical issue for tungsten divertor. Details will be shown in Sec. 5.2.

3. Helium Effects

Recently, helium effects on tungsten have attracted increasing attention in terms of material degradation [3], leading to exfoliation and grain ejection (dust formation). Helium atoms have high trapping energy with point defects (4.0 eV-4.4 eV) in tungsten, while hydrogen atoms have much lower binding energy with point defects (~1.4 eV for a single vacancy, ~2.1 eV for a void). Therefore, He atoms are hardly detrapped from these defects even at elevated temperatures. In addition, when tungsten temperature exceeds recrystallization temperature (1300 °C), helium and defect complexes becomes mobile and tend to agglomerate to form, so-called, helium bubbles. Below this temperature range (more than about 800 °C), nanoscopic structure is formed [4, 5]. The typical nanostructure is shown in Fig. 1 [5] for sintered tungsten (purity ~99.99 %) with layered microstructure. The thickness of nanostructure increases with square root of time, indicating diffusion-like behavior.

There are several disadvantages for He bubble- or nano-structure on tungsten. At first, it can be pointed out that the bubble structure in the subsurface region significantly reduce thermal diffusivity, leading to melting and evaporation of surface layer by transient heat loads. Secondly, dust formation associated with enhanced erosion would take place by the He effects. At elevated temperature, He bubbles diffuse into the bulk of tungsten and tend to be trapped at grain boundaries. He bubbles along the grain boundaries reduce adhesion between the grains, which are easily ejected by the effects of thermal stress or internal stress caused by hydrogen isotopes and/or helium containment. Figure 2 shows the ejected grain particles on the tungsten sample surface. This tungsten sample was exposed to He plasma at 1,600 K to the fluence of 9×10^{25} m^{-2} at first, followed by deuterium plasma exposure at 550 K to the fluence of 2.5×10^{25} m^{-2}. Tungsten used in this experiment was sintered with laminar grain structure with the purity of 99.95 %. It was recrystallized during He plasma exposure and He bubbles with the diameters less than 1 μm were densely formed along grain boundaries. Therefore, the grain ejection in this case took place probably due to reduced adhesion between grains by He bubble accumulation.

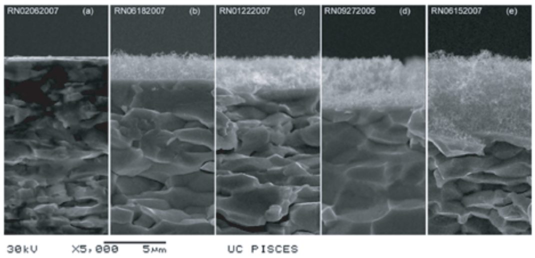

Fig. 1 He plasma induced nanostructure on tungsten [5].

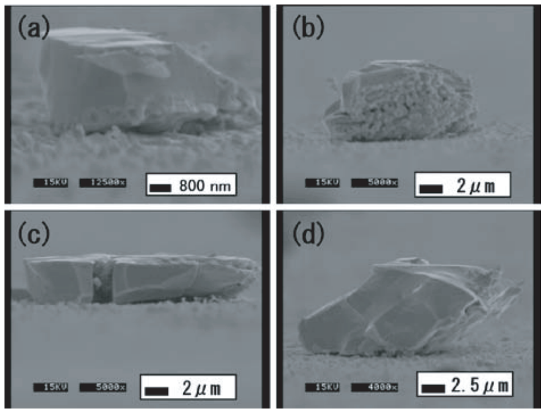

Fig. 2 Grain ejection of tungsten surface after deuterium plasma exposure at 550 K subsequent to He plasma pre-exposure at 1,600 K. (D. Nishijima *et al.* [6]).

Therefore, it is important to study formation conditions, effects to core plasma, and suppression technique (if necessary) for He bubbles and nanostructure in fusion reactor environments.

Recent studies have shown that He bubble layers act as a hydrogen isotope diffusion barrier. By this effect, reduction of deuterium retention and suppression of blistering take place [7], see Fig. 3. In this experiment, sintered tungsten (purity ~99.99 %) with laminar grain structure was used. In ITER, since wall temperature except for near strike points is low (200~300 °C), this effect greatly affects T retention in a tungsten wall material. For DEMO, T retention in wall materials will not be the issue but T permeation to coolant tubes is a matter of concern. The barrier effect of He bubbles could greatly reduce T permeation. According to the reference [7], this effect is effective at least up to the temperature of 450 °C (723 K). Over this temperature, more studies are needed to understand T behavior in first walls of DEMO.

4. Pulsed Heat Load Effects

For more than 20 years, many good confinement modes of core plasmas have been found in tokamak devices and are the keys to achieve economical fusion reactors. One of them is H-mode, which has transport barriers near the edge plasma (so called pedestal). This mode, however, is known to be accompanied by repeated energy and

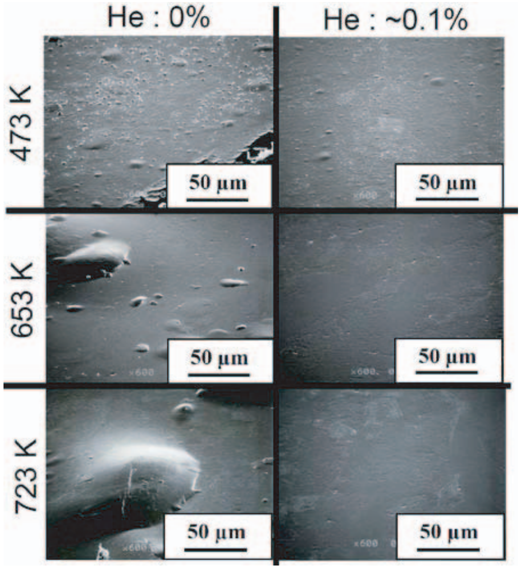

Fig. 3 Temperature dependence of surface morphology of ion irradiated W by 1 keV H, C, and He mixed ion tungsten [7].

particle ejection, so-called ELM (Edge Localized Mode). For ITER, a pulse length and a heat load of Type I ELMs were predicted to be ~0.2 ms and 0.5-1.2 MJ/cm^2, respectively [8, 9]. This Type I ELM pulse can raise the surface temperature of tungsten above the melting point of tungsten (3422 °C). Once tungsten melts, grain growth and significant reduction of yield strength will occur, leading to crack formation and dust generation. Therefore, it is believed that the mitigation of the ELM pulse energy is of great importance for fusion reactors.

Recently it has been pointed out that even under non-melting conditions the repetitive ELM pulse effects could be serious [10]. Repetitive heat pulse cause surface expansion and contraction alternately, which would cause metal fatigue and cracking. Particle induced processes, mainly due to helium ions, could enhance this effects. More studies will be needed to comprehensively understand this effect and to avoid serious effect in fusion reactors for tungsten walls.

So far, no similar repetitive heat pulse to wall materials like ELM's in tokamak has not been found in the helical system. But it is noted than ELM is associated with pedestals near the separatrix, which appears as a result of improved plasma confinement, known as H-mode. So in the future, there still remain some possibility to obtain good confinement regimes with ELM-like edge plasma behavior in helical system. In this case, repetitive heat pulse must be reduced or suppressed to an acceptable level.

5. Issues for Reactor Environment

5.1 Steady-state operation

Plasma duration in present magnetic confinement devices are limited to a few hours. Especially for high performance plasmas with the fusion energy gain factor around 1, the duration is limited to an order of seconds. On the other hand, the discharge duration of DEMO reactors will be an order of several months. There still remains a significant gap between the present and next step device, even ITER, and DEMO.

There are several time constants in terms of plasma wall interaction. Wall saturation for fuel atoms (hydrogen isotopes) is one of the important time constant. If wall saturation does not take place, walls always suck tritium, eventually the tritium wall retention exceeds accepted level (~700 g for ITER). In addition, wall pumping of tritium reduces usable tritium produced in blankets, leading that requirements of TBR (Tritium Breeding Ratio) should be raised in blankets. According to present reactor design study [11], TBR is very marginal compared with the required value (~1.1). Therefore, wall pumping should be terminated in an acceptable short operation duration.

In JT-60U, plasma performance under these wall-saturated conditions have been investigated [12]. The wall saturation time in this case is an order of minutes. This is acceptable because it is much shorter than expected operation time of DEMO. In Tore Supra, however, wall saturation has not been observed for 2 min discharge and wall pumping continued at least up to the cumulative discharge time of 5 hours [13]. In these devices, wall materials were graphite and the relatively low first wall temperature (less than 300 °C).

In DEMO, in the case of metallic wall materials (tungsten as a leading candidate), implanted tritium will diffuse into the bulk to be trapped at intrinsic or neutron induced trapping sites or permeate to the rear surface or interface with structural materials (low activation materials such as RAF, vanadium alloys, SiC as present candidates). For metallic structural materials, tritium will permeate through to reach coolants. On the other hand, for ceramic materials such as SiC, since this is a strong diffusion barrier of tritium, tritium will not permeate to the coolants. If the wall temperature is high enough, tritium will not accumulate in metallic armor materials and permeate to the coolants, which are the most desirable situation in terms of tritium retention. But if the coolant temperature is not very high (ex. 300 °C of water), tritium could accumulate around coolant tubes. This will not only increase tritium retention but also will affect deterioration of material properties. In addition, as was mentioned in the section 3, helium bubble layers tend to work as tritium diffusion barrier. If there are two diffusion barriers both on the plasma facing side (ex. He bubble layer) and on the rear side (ex. SiC/SiC composite), tritium tends to be confined in wall materials between these diffusion barriers, which increase tritium retention and probably deteriorate the wall materials. Therefore, in DEMO reactors, issues of tritium implanted from the plasma facing side are closely related to the design of divertor and blankets. This issue, however, have not been studied so far, and will be one of the most important R&D

subjects for blanket development.

As was described, tritium retention in codeposited layers is a matter of concern for ITER. Even in DEMO, since erosion of wall materials does not have a clear limit, formation of codeposition layers would continue during plasma operation. It is known that hydrocarbon molecular radicals have low sticking coefficients on high temperature walls. These molecules are transported far from plasma facing walls through exhaust ducts. Even if the surface temperature of in-vessel components are high enough for these radicals not to stick, there are low temperature surface somewhere in remote area (vacuum pumps etc.). Hydrocarbon radicals will deposit on these surface and produce T retained deposition layers. This is also a concern related to the use of carbon contained walls for steady-state reactors. To solve this issue, complete understandings of transport of hydrocarbon radicals and effective removal methods of T retained deposit must be needed.

Degradation of wall materials under steady-state conditions also needs to be investigated. There have been quite a few studies for the effects of plasma exposure to wall materials. Ion fluence of these studies, however, are limited up to $10^{28}\,\mathrm{m}^{-2}$, while ion fluence to divertor plates in fusion reactors will reach $10^{31}\,\mathrm{m}^{-2}$ in a year. At present, no plasma device can simulate wall materials under this fluence condition, and there are even no plans for it. We need to make a strategy for the development of reliable wall materials under very high fluence conditions.

5.2 Heavy neutron irradiation

As was pointed out, one of the most significant differences between ITER and DEMO is neutron fluence. In fact, ITER will provide a test bed of 14 MeV neutron irradiation for materials and components. Its fluence, however, is much lower than that in DEMO due to low duty plasma operation. ITER will be able to provide the average neutron fluence of about $0.3\,\mathrm{MWm}^{-2}$ year. On the other hand, neutron fluence to wall materials of fusion reactors would reach about $10\,\mathrm{MWm}^{-2}$ year [14].

Fusion neutrons (14 MeV) will have several effects on wall materials. Radiation damages produced by elastic collision with lattice atoms. These damages will result in hardening, swelling and some other material degradation. In addition, transmutation of materials needs to be taken care of due to very high fluence in DEMO. For example, some of tungsten isotopes are transmuted to Re, then Os [15]. Thermal conductivity of tungsten contained with Re was studied by Fujitsuka *et al.* [16]. Their study showed that thermal conductivity of tungsten decreases with Re concentration. Tungsten with 10at % Re has lower thermal conductivity than pure tungsten by about 30 % at 1000 K. Under fusion neutron irradiation, this composition would be reached in about 2 years of operation. The other heavy-atom transmutation effects have not been known well.

In addition, (n, α) reaction will produce He atoms which appears with the neutron energy more than 10 MeV (tungsten). This means fission neutron (less than a few MeV) cannot cause this reaction. As already mentioned, helium could cause deteriorating effects on metals due to the formation of He bubbles. Therefore, definitely we need some facilities other than fission reactors or dedicated experiments to examine transmuted He effects on tungsten bulk material property.

New tungsten material with the resistance to neutron irradiation is being developed by Kurishita *et al.* [17]. This new material, UFG-W (Ultra Fine Grained W) with TiC dispersoids, has much smaller grain sizes (less than submicrometer) which greatly improve embrittlement of ordinary tungsten. This material has also desirable feature under high flux plasma exposure environment. For ordinary tungsten, high flux plasma exposure produces blisters [18], but UFG-W did not show blisters up to the fluence of about $10^{26}\,\mathrm{m}^{-2}$ [19]. In addition, D retention is not higher than that in ordinary tungsten (stress relieved tungsten with the grain size of a few micrometer). Although it is necessary to examine at higher fluence conditions, UFG-W clearly has some advantages over ordinary tungsten as plasma facing materials for ITER and DEMO.

For CFC graphite, neutron effects would be very serious. Most important effects are reduction of thermal conductivity and dimensional change [20]. Reduction of thermal conductivity changes appear even in ~0.01 dpa. The reduction is larger at lower temperatures. Over about 1000 °C, reduction is small because of the annealing of radiation damage. On the other hand, dimensional change takes place during annealing process of damage, which makes new graphitic plane and expand graphite crystal along c-axis [21]. For carbon fibers, neutron irradiation leads to shrinkage in the direction parallel to the fibers and to swelling in the perpendicular direction. Since this process increases with the increase in temperature, the most serious effects of dimensional change would appear in CFC graphite tiles at strike points of divertor. Although we do not have database under reactor relevant high fluence conditions, this effect could be inevitable and the most serious problem for the use of CFC in DEMO.

5.3 Strategy needs for DEMO

Design of tokamak based DEMO device has been carried out by several research groups [11, 22]. Handling of divertor heat load is always an issue. The heat load of $10\,\mathrm{MW/m}^2$ to the divertor is a typical standard of a design parameter. In terms of high heat flux technology, development of high heat flux components for DEMO has more limitation than ITER. Coolant tubes for ITER can be made of copper alloys (ex. CuCrZr alloy for ITER) due to high thermal conductivity. But this alloy is subject to hardening under heavy neutron irradiation. Therefore, under DEMO environment, the other materials need to be examined. One of the candidate materials in the JAEA design is RAF [23].

Design of water cooled tungsten monoblock divertor with the RAF coolant tube can handle the maximum heat flux of 13 MW/m^2, while 25 MW/m^2 can be handled by the module with the Cu coolant tube. Heat removal capability for the plasma facing components in DEMO and the commercial reactors must have some tolerance for safety operation and material degradation during long term operation. Therefore, heat removal capability of the divertor with the abovementioned RAF cooling tube would not be enough for the 10 MW/m^2 heat flux. The heat flux to the divertor plates in DEMO should be substantially reduced in comparison with ITER as long as solid materials would be used.

In the roadmap shown in ITER home page [24], the DEMO reactor construction will start just before the second DT operation phase in ITER and the operation phase 1 in DEMO will start before the end of the ITER operation. IFMIF will be employed simultaneously with the ITER operation, which provide the opportunity to select and optimize blanket structural materials. For plasma facing components, however, there is no plan to make a selection and qualification test of the components for the steady-state operations of DEMO.

The test conditions of divertor components are very complicated. They should include high heat flux irradiation up to 10-20 MW/m^2, high fluence irradiation of 14 MeV neutron up to about 10 MW·year/m^2, and high fluence D/T and He (5~10 %) plasma irradiation up to the fluence of about 10^{31} m^{-2}. Combination of these mixed irradiation is extremely important. For example, thermal stress caused by temperature gradient would be closely related to neutron irradiation creep. Neutron damage and helium bubbles would strongly affect hydrogen isotope and helium behavior, and its effect on material degradation in tungsten.

This combination test should be done before the installation of divertor modules in steady-state operation of DEMO. The relevant facilities (ideas) are CTF (Component Test Facility, steady-state magnetic confinement plasma for a volume neutron source), IFMIF with a high density plasma device, and the use of the operation phase 1 of DEMO. In any case, we need to seriously consider the strategy for R&D and a qualification test of divertor modules for DEMO.

6. Towards Helical Reactors

Helical reactors also have similar requirements as tokamak reactors in terms of plasma wall interaction. The important issues are avoidance of impurity accumulation in the core plasmas, and power and particle (He) control to the divertor. LHD type helical reactors already have several advantages over tokamak devices such as no major disruption associated with current quenching and natural divertor configuration with edge ergodic layers [25]. Since the connection length between divertor plates and X points is shorter in helical devices than tokamak devices, the role of the ergodic layers is very important to control impurity influx to the core plasma. Kobayashi *et al.* showed that the edge surface layer plays an important role in impurity retention, where the friction force significantly dominates over the thermal force in LHD [26]. In a short pulse discharge (an order of seconds), experimental data proved that this layer effectively blocked wall impurities from penetrating into the core plasma. In the future, investigation on impurity behavior in this ergodic layer for much longer time scale is needed.

Power and particle control (He ash exhaust) is another important issue in helical system towards DEMO. As was mentioned before, it is better to reduce divertor heat load to much less than 10 MW/m^2 for realistic solution for solid divertor system. According to the reactor design FFHR [25], divertor heat flux of 1.6 to 2.3 MW/m^2 was a design parameter. This number is very attractive in terms of heat removal. In general, as particle flux to the divertor plate is low, neutral pressure near evacuation slot is also low, leading to reduction of He exhaust efficiency. The important issue is to achieve compatibility of low heat flux to the divertor plate and high He exhaust efficiency. Appropriate divertor design would be a key to find optimization of these.

7. Conclusion

Although there still remain several important issues on plasma facing components for ITER, there will be more challenging issues towards DEMO because of steady-state operation and high neutron dose. Feasibility study and development of relevant tungsten materials under steady-state fusion reactor environments must be pursued. For these purposes, we need clear strategy for the development of plasma facing components.

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