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Plasma-Surface Interaction, Scrape-Off Layer and Divertor Physics: Implications for ITER

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Plasma-surface interaction, scrape-off layer and divertor physics: Implications for ITER

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Abstract: The work of the ITPA SOL/divertor group is reviewed and implications for ITER discussed. Studies of near SOL gradients have revealed a connection to underlying turbulence models. Analysis of a multi-machine database shows that parallel conduction gradients near the separatrix scale as major radius. New SOL measurements have implicated low-field side transport as driving parallel flows to the inboard side. The high-n nature of ELMs has been elucidated and new measurements have determined that they carry ~10-20% of the ELM energy to the far SOL with implications for ITER limiters and the upper divertor. Analysis of ELM measurements imply that the ELM continuously loses energy as it travels across the SOL – larger gaps should reduce surface loads. The predicted divertor power loads for ITER disruptions has been reduced as a result of finding that the divertor footprint broadens during the thermal quench and that the plasma can lose up to 80% of its thermal energy before the thermal quench (not true for VDEs or ITBs). On the other hand predictions of power loading to surfaces outside the divertor have increased. Disruption mitigation through massive gas puffing has been successful at reducing divertor heat loads but estimates of the effect on the main chamber walls indicate 10s of kG of Be could be melted/mitigation. Estimates of ITER tritium retention have reduced the amount retained/discharge although the uncertainties are large and tritium cleanup may be necessary every few days to weeks. Long-pulse studies have shown that the fraction of injected gas that can be recovered after a discharge decreases with discharge length. The retention rate on the sides of tiles appears to ~ 1-3% of the ion flux to the front surface for C tiles and ~100x less for Mo tiles. T removal techniques are being developed based on surface heating and surface ablation although ITER mixed materials will make T removal more difficult. The use of mixed materials gives rise to a number of potential processes – e.g. reduction of surface melting temperatures (formation of alloys) and reduction of chemical sputtering. Advances in modelling of the ITER divertor and flows have enhanced the capability to match experimental data and predict ITER performance.

1. Introduction

The interaction of plasma with surrounding Plasma Facing Component (PFC) surfaces will have considerable impact on the performance of fusion plasmas, the lifetime of PFCs, and the retention of tritium in next step Burning Plasma Experiments (such as ITER). This review of work by the International Tokamak Physics Activity (ITPA) SOL/Divertor group shows the considerable progress achieved in the last several years. This group has also recently written an extensive review of work in this area [1].

2. Transport

Recent results have changed our understanding of steady-state transport in the Scrape-Off-Layer (SOL). Studies of the density and temperature profiles just outside the separatrix (within one density scale length, λ_n - 'near' SOL) [2] have shown that the local pressure gradient, ∇P, is proportional to the local ν^*. Moreover, connection to the underlying turbulence which determines the gradients is made: at fixed values of normalized collisionality parameter (characterized as the 'diamagnetic parameter', α_d), electron pressure gradients in the near SOL increase with plasma

Figure 1: Ratio of Te e-folding length to machine size plotted vs normalized separatrix density.

current squared, holding the MHD ballooning parameter, α_{MHD}, constant. A parallel analysis of ASDEX-Upgrade density and temperature scale lengths in the region of the separatrix during ELMy H-modes [3] showed that the temperature decay length, λ_T, is two times shorter than λ_n, corresponding to $\eta_e = \lambda_n/\lambda_T \approx 2$. A multi-machine database [4] of n_e & T_e profiles in the region of the separatrix has been analyzed yielding the T_e profile e-folding length λ_T (averaged over ELMs) scaling roughly as major radius, Figure 1. The corresponding conduction power flow scale length, $\lambda_q = 2\lambda_T/7$, predicted for ITER would then be of order 5-7 mm. Power spreading in the divertor region generally leads to broader power flux profiles on the divertor surface. Farther from the separatrix ('far' SOL) the density gradients are weaker leading to substantial densities [5, 6]. The far SOL transport is convective ($v_r \sim$ 100-500 m/s) during L-mode [7, 8] as well during and between ELMs [8]. Initial dimensionless scaling studies imply that radial convective the velocity in ITER should be similar to that observed in today's tokamaks leading to similar wall ion fluences and resultant surface erosion, impurity sources, and recycling. The emerging picture is that instabilities in the near SOL give rise to plasma filaments (along B) or 'blobs' that are ejected and carry plasma radially into the far SOL (100-500 m/s) [1] leading to the observed broadened profiles ('shoulders') and convective characteristics. The difficult process of including an approximation of such turbulent transport into predictive, fluid models of the divertor and SOL, is being addressed [9, 10].

Parallel flows in the SOL play an important role in impurity transport, ideally sweeping impurities to the divertor before they can diffuse into the core plasma. Flows also play a central role in the co-deposition of D with carbon in present day tokamaks and are associated with concerns for T retention in ITER. New inner wall probe flow measurements (Fig. 2) have demonstrated a pressure imbalance between the high- & low-field SOL regions [11]. High-field Mach numbers can reach 1. It is proposed that the strong ballooning transport at the low-field side drives the measured poloidal pressure imbalance which, in turn, drives flows along B to the inner edge [11-13]. Initial implementation in codes has allowed a much better match to experiment [14].

Figure 2: Parallel flow measurements in the SOL made by Langmuir probes. S ranges from 0 at the outer divertor to 1 at the inner divertor. Positive Mach number (flow) is flow towards the inner divertor.

3. Edge Localized Modes (ELMs)

ELMs also raise concerns for main chamber PFCs. Filamentary plasma structures protrude from the pedestal [15] and then propagate
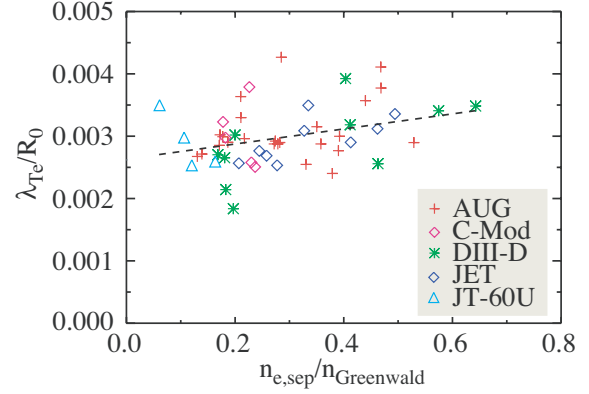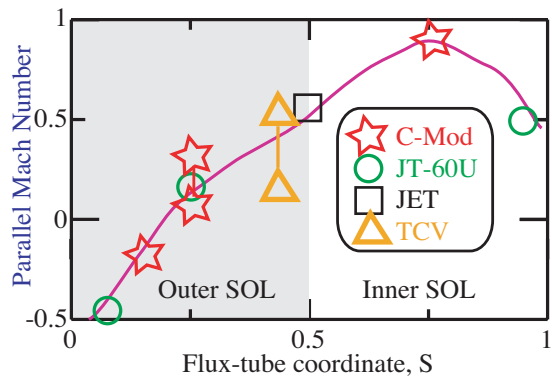
radially until striking a material surface, Fig.3 [16]. The filaments, which carry up to 20% of the total ELM energy, exhibit a toroidal mode number n~4-20, similar to the mode structure from stability analysis at ELM onset [17, 18]. Heat flux striations outboard of the divertor strike-point have been observed as a result of this transport [19]. The filaments range in size up to 10 cm in diameter and propagate radially at $v_r \leq 1$ km/s. The decay of the density and temperatures in the filament is consistent with a model of parallel transport that drains the filament to the divertor as it propagates radially [20]. Outboard limiter measurements of $T_i \sim$ several hundred eV imply that T_i drops more slowly than T_e as the filament travels radially [21]. A further implication of high outboard limiter T_i is that the energy in the filament is primarily carried by ions; Sputtering yields are high per incident ion.

Localized ELM filament heat loading can be considerable with up to 1% of the total ELM energy deposited in a 10-20 cm^2 area [22] in a very short period, 100-200 μsec. Averaged over a number of ELMs, the ELM-derived deposited heat flux has a more continuous profile over main chamber PFCs. During H-mode, and with no external gas puffing, 80% of the particle flux to the outboard midplane limiter arrives during ELMs, even though the ELM duration is a small fraction of the time between ELMs [5]. At high density the fraction of limiter particle flux remains high at 50% though the ELMs are smaller and the radial flux between ELMs becomes much larger. Additional detailed poloidal measurements of ELM flux to main chamber PFCs are needed.
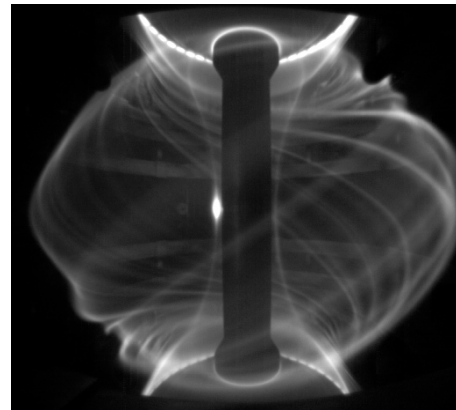

Figure 3: Image of MAST ELMs showing the filamentary nature.

The main chamber ELM flux represents some serious concerns for ITER. The high instantaneous heat flux could melt the beryllium PFCs there. While replacement of Be with higher-Z materials would reduce the risk of melting it would give rise to concerns for high-Z contamination of, and radiation in, the core. Identification and implementation of small ELM regimes and control techniques could offer considerable help by reducing the ELM flux to the main chamber walls.

4. Disruptions

The first-wall location of the rapid energy deposition during disruptions is a large concern for the ITER PFC's. ITER has significantly higher (~x10) energy density than present devices that can cause transient overheating damage to the surface of PFC materials, threatening their long-term viability for power handling. Additionally, ITER presently employs three different PFC materials at various locations and each material will respond differently to the energy pulse. Previously, it was assumed that most of the energy dissipation would be in the divertor region. However, measured energy deposition during disruptions can reach as high as 50% of the thermal and magnetic plasma energy at first-wall regions outside of the divertor [1]. This high heat flux is linked to two effects: 1) the broadening of the divertor heat load footprint by up to a factor of 10 during the MHD-induced thermal quench, and 2) an isotropic deposition of poloidal magnetic energy due to radiation in the current quench when the plasma is typically cold (T < 10 eV). The fraction of thermal energy dissipated in the divertor during disruptions appears to be reduced on average as the stored thermal energy density in the plasma increases, Fig. 4. This new insight lessens concerns that the divertor must absorb all the thermal energy of the plasma, but at the same time raises concerns for the main chamber PFCs.

Disruption mitigation techniques are being developed and tested in order to alleviate thermal and electromagnetic damage to PFCs on a variety of tokamaks with positive implications for ITER [1]. However, an additional concern for the main chamber PFCs has arisen regarding

the intentional distribution of plasma thermal energy by radiation following massive impurity injection. Radiation models and extrapolation from present mitigation experiments suggest that even completely uniform distribution of the ~350 MJ of thermal energy in the ITER Q=10 target plasma may cause melting of the beryllium first wall (tungsten/carbon are not melted/ablated) [23]. This would produce a distributed ~10's micron thick molten layer over ~ 800 m^2, constituting several 10's of kg of Be; however the stability and movement of this molten layer remains uncertain.

Another important insight gained is that for many types of disruptions the plasma thermal energy content is often reduced as the disruption approaches. A clear example is that of density limit disruptions in which, prior to the thermal quench, the original plasma thermal energy is typically reduced through degradation in energy confinement (e.g. reversion to L-mode) [24]. In such a manner, a significant fraction of the initial stored thermal energy (up to 80%) is safely dissipated before the rapid thermal quench, a positive development for all PFCs. The exceptions to this behavior (example circled in Fig. 4) are disruptions triggered by ideal-like β limits, such as for Internal Transport Barrier (ITB) plasmas, and disruptions caused by a Vertical Displacement Event (VDE) [1]. In these cases there is no significant reduction in the plasma thermal energy before the disruption, making these disruption types of the highest concern for ITER operations. ITB-type disruptions tend to have large fractional dissipation in the divertor, while VDE disruptions typically lose their thermal energy to the main-wall, and therefore may require different avoidance and/or mitigation strategies [24].
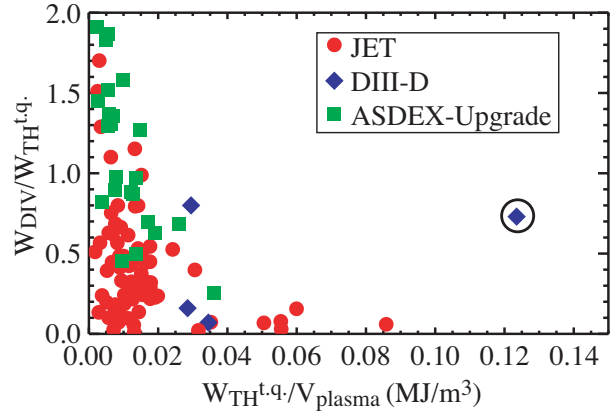

Figure 4: Ratio of plasma energy reaching the divertor to the thermal energy (at the thermal quench) plotted vs thermal quench energy (normalized by machine size). The circled point corresponds to an ITB disruption.
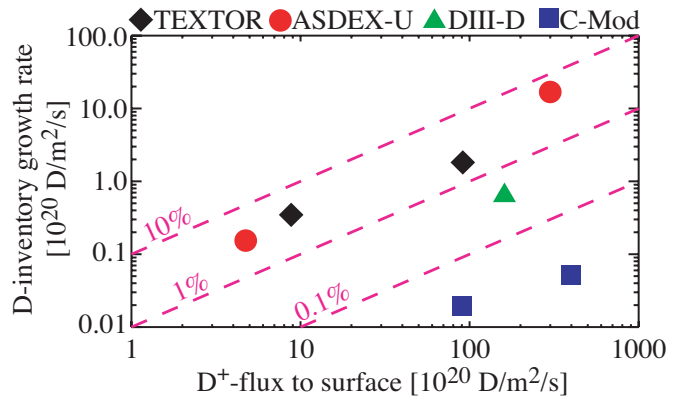
5. Tritium retention

The retention of tritium is a serious concern for ITER given that 100g is to be injected each discharge and the in-vessel limit may be as low as 350g [25]. Our current understanding is that T retention is dominated by chemically sputtered molecules, including D and C (e.g. methane), which are then are co-deposited on surfaces facing low-Te plasmas and areas shadowed from direct plasma contact. Present retention fractions of the injected fuel are in the range of 3 to 30% [1]. Extrapolating current D/T retention experience to ITER (->1.6 – 27 g/shot) indicates the need for a significant tritium removal effort after 10-200 reference discharges [1]. Unfortunately, the large uncertainties of this extrapolation, coupled together with the current selection of ITER PFC materials (Be, C and W) leads us to conclude that we cannot properly predict the T retention in ITER. Overall, D retention in tokamaks with high-Z PFCs appears lower than for carbon, but still higher than scaled from previous non-tokamak, or laboratory, experiments [1, 26].

Figure 5: D retention dependence on front surface ion flux. Data are from carbon tiles (Mo in C-Mod) & boronized surfaces.

Currently-operating tokamaks have of order 10^4 tiles with estimates of the

contribution to overall D/T retention corresponding to tile sides in the range 15-30% [1]. The ITER deisgn utilizes castellated tiles and macrobrush armour thus increasing the number of such surfaces dramatically to 10^6 making the total side and front surface areas comparable. Through a cross-tokamak tile side retention study (Fig. 5 [27, 28]) we have found that tile side D retention is ~ 1% of the ion flux to the corresponding front surface for carbon PFC dominated devices while, for the fully high-Z PFC device included, the tile side retention is significantly reduced. Elevated surface temperatures above 200°C also lead to a reduction of the D-retention by a factor of 10 [28]. Therefore, key parameters for the reduction of T retention will be a low C-concentration in the plasma and elevated tile temperatures.

The wealth of existing T retention operational experience is derived from short pulse tokamaks. In recent years experiments on a number of devices have revealed common long-pulse effects: During the pulse, in a first phase (from ~5s in JET [29] up to ~100 s in Tore Supra [30]), the retention rate decreases, becoming constant in the following second phase. Although the D retention in the second phase represents only a small fraction of the recycling flux (1-5%), it is a significant fraction of the injected flux (50-80%). The only exception where the retention rate is low, or negative, are discharges with low fuelling rates, such as long L modes performed in JET [29], or discharges with saturated walls, such as repetitive long H modes performed in JT-60U [31].

The amount of gas recovered after each discharge is independent of pulse length for long pulses. Therefore, the gas retention integrated over many discharges is negligible when performing short pulses, while a significant inventory builds up for long pulses. 10-20% retention of the injected fuel is typical for both short and long-pulse discharges. Co-deposition of deuterium with carbon can explain the magnitude of long-pulse retention observed in the JET DTE1 campaign [32], but is insufficient for Tore Supra [33] and JT60U [31]. A second T retention process, bulk diffusion deep into porous CFC has been postulated. Laboratory experiments, coordinated through ITPA, confirm deep penetration of D into CFC [34] as well as in Mo [26]. Such deep diffusion would impact the viability of de-tritiation methods to be used in ITER. The main concern in terms of the T inventory buildup remains codeposition as its scales linearly with fluence.

Given the large uncertainties in ITER T retention and the potential to halt operation to remove T it is clear that efficient T removal techniques are required. Initial efforts on TFTR and JET were slow (~10^{-3} g/hour) [1]. Assuming of order 100 g of T needs to be removed overnight the required T removal rate would be ~ 10 g/hour. The T removal methods currently being developed and examined fall loosely into 2 types [35, 36]: 1) heat the surface such that the a-C:T bond is broken and the T liberated (scanning laser or radiative plasma termination) or; 2) remove the mixed C/T layer completely through ablation (heating using lasers or flash-lamps) or oxidation. Although such techniques have not been tested widely the removal rates in some cases approach that needed for ITER [36, 37]. The above T recovery methods must be compatible with the ITER toroidal field and not lead to additional problems (e.g. dust or compromise following discharges). It is likely that no single T removal scheme will be sufficient and, depending on the retention rate, different methods will be applied at different times.

6. Materials

In contrast to current tokamaks ITER plans to have Be, C and W PFC surfaces, thereby minimizing C usage and, potentially, T retention. New studies indicate that such a mix
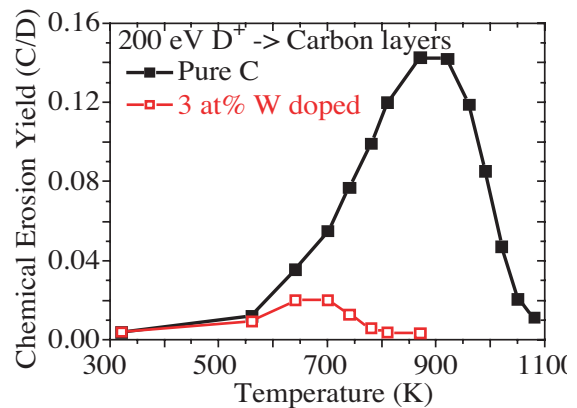

Figure 6: Temperature dependence of the chemical erosion of pure carbon compared to W-doped carbon layers.

of materials creates new effects: First, an intermixture of different materials, alloys or compounds may be formed with strongly different thermo-mechanical properties. An example is that in the temperature range 800-1200°C Be and W alloys are formed which leads to reduced melting temperatures (Be_2W: 2250°C, $Be_{12}W$: <1750°C, $Be_{22}W$: <1300°C) [38-40]. In divertor conditions formation of thick alloy layers may be prevented by Be sublimation and erosion [41]. Secondly, surface alloy and compound formation (Be_2C, WC) will also influence the tritium retention capabilities of metals [42, 43, 41]. Such surface compounds may retain more hydrogen than pure metals, or even act as out-diffusion barriers for implanted hydrogen isotopes, thus increasing bulk retention or permeation. Lastly, modification of the C surface through buildup of metal or carbide precipitates there (e.g. through simultaneous incidence of Be and D/T ions) can reduce carbon chemical erosion [44]. Because carbon materials doped with metals typically show reduced chemical erosion yields [45] it is possible that better graphite PFCs (Figure 6 [46]) could be developed. It is recognised that material mixing gives rise to both potential risks and opportunities for plasma-wall interaction issues. Further research programs have been started both in laboratory experiments and in fusion devices [47]

Most fusion reactor studies prefer tungsten over carbon due to its advantages with respect to erosion and neutron damage [48-50] as well as the level of tritium retention which is predicted to be lower by orders of magnitude [51] than with carbon PFCs. It is clear that prior to operation of any fully-ignited fusion reactor the compatibility of high-Z PFCs with high performance operation, without low-Z coatings, needs to be demonstrated in ITER. Recently, both Alcator C-Mod (entirely molybdenum PFC tiles) and ASDEX-Upgrade (most recently 85% W-coated carbon PFCs) have addressed the question of the role of boronization in minimizing high-Z impurity levels in the core plasma [52-55]. Boronization dramatically lowers the core concentration of W or Mo (by a factor of 10-100) and medium Z impurities, lowering radiation in the core and improving the quality of H-modes. The erosion of the boron layers, which appears to be localized to a small fraction of the full PFC area, leads to a recovery of the core Mo/W levels back to pre-boronization equilibrium over 10's of shots. The erosion mechanism during ICRF heating appears to be an RF enhancement of the surface sheath ('sheath-rectification') that then accelerates impurity ions (C^{4+}, B^{3+}) leading to physical sputtering at the top of the outer divertor (C-Mod [52]) and the ICRF guard limiters (ASDEX-Upgrade [54]). For NB-heated plasmas fast ions drift out of the core and impact the limiter again leading to W physical sputtering [56]. In general the B erosion, and subsequent W/Mo erosion, is faster for ICRF-heated plasmas as opposed to NB or Ohmic H-modes. C-Mod results indicate that the tungsten planned for ITER is in a location that may be heavily eroded during ICRH-heated discharges. AUG results raise concerns over a Be outer limiter in ITER. However, during NB-heated plasmas the W levels appear to be lower with less of an effect on the core plasma. Further experiments on the 2 tokamaks will clarify the implications for a fully W ITER.

7. The ITER divertor and collisional neutrals

In predicting ITER performance it is clear that we must have confidence that codes are able to accurately predict plasma performance in present devices. While current plasma/neutral models have reproduced divertor pressures in neutral kinetic regime common to most current tokamaks, there has been difficulty in the ITER limit where neutrals are fluid-like (C-Mod). Recent interpretive modelling efforts have employed the Onion Skin Model (OSM) - EIRENE combination where a number of new physics processes have been implemented – neutral viscosity, neutral-ion collisions, and Lyman-alpha photon trapping [57]. With these modifications, C-Mod's ITER-like neutral pressures and other divertor characteristics are reproduced. At the highest densities recombination is the source of 80% of the neutrals in the divertor. The same code improvements have been included in the B2-EIRENE code [58, 59]. When the model is applied to ITER nonlinear neural effects cause the Private Flux Region neutral pressure operational range to shift to ~ 2x higher values [58] as compared to previous, linear neutral, studies. The introduction of divertor Lyman-alpha trapping, although modifying

the local plasma parameters, does not affect the ITER divertor performance with respect to the power and particle handling [59]. The benchmarking of the new code physics against C-Mod as well as the continued testing for ITER [60] is ongoing.

In parallel with the above testing of the code physics against experiment a program of code-code benchmarking has been followed (EDGE2D to SOLPS to UEDGE). This process has brought to fore the different physics assumptions across codes (e.g. treatment of kinetic electrons, neutral and ion flux limiters, the Bohm Mach condition at the divertor surfaces and the treatment of drifts in the core plasma), some of which can lead to significant differences in the model output. Current plans are for more comparisons of the codes with each drift process turned on one at a time.

8. Summary

Advances in measurement and modelling capabilities have been instrumental in advancing our understanding of SOL & divertor physics. 10-20% of the ELM energy reaches into the far SOL locally raising concerns for limiters and the secondary divertor. Disruption divertor power deposition estimates have been lowered while that for the main chamber has been raised. Disruption mitigation utilizing massive gas injection reduces divertor loadings but increases main chamber loadings such that Be melting may be a concern. Estimates of T retention have been reduced. However the level of T retention predicted for ITER will probably still mandate interruptions in operation to remove T. Tile side D/T retention appears proportional to the ion flux impacting the front surface with levels of order a few % for C and much lower for Mo. Many T removal techniques are being developed. Depending on the T retention rate and principal locations a number of techniques will be needed at various times in ITER operation. The use of the combination of Be, W and C for ITER tiles should reduce T retention (through reduced carbon usage) and has the potential to reduce chemical sputtering. The surface alloying of Be with W reduces its melting temperature.

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