

# Implications of PMI and wall material choice on fusion reactor tritium self-sufficiency

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## Abstract

Tritium self-sufficiency is a critical issue for the production of nuclear fusion energy. Here we quantify the impact of co-deposition of eroded wall material and fuel on the tritium particle balance in a hypothetical reactor system. The expected ITER plasma parameters and geometry are used to estimate the amount eroded material from a full tungsten, beryllium or carbon device. Measured D concentrations in co-deposits are extrapolated to the wall temperature expected in future reactors and used along with these eroded flux estimates to determine the net loss probability of tritium from the device due to co-deposition with fuel species. The use of liquid divertor surfaces is also considered with the amount of tritium residing in the recirculating liquid estimated. The general conclusion, from a tritium self-sufficiency viewpoint, is that one should avoid low-Z materials that readily form hydrogen bonds, in favor of high-Z non-hydride forming materials.

## Introduction

Harnessing nuclear fusion power production will most likely initially involve the energy release from the D-T nuclear fuel cycle. The D-T fuel cycle is attractive because it offers the highest reaction cross-section at a lower fuel temperature [1]. Unfortunately, it also involves using tritium as a fueling agent for sustaining the reaction and there is a limited external supply of tritium available. The ability of a fusion powered reactor to generate sufficient tritium for its own use (as well as generating sufficient tritium for the start-up of a subsequent reactor) is, therefore, one of the most fundamental go-or-no-go questions that fusion researchers must successfully address after the demonstration of burning plasma physics in ITER.

In fusion reactor designs, tritium is generated by harnessing the energy released as high-energy neutrons by the fusing of deuterium and tritium ions. These neutrons interact with lithium contained within the blankets systems surrounding the burning plasma to generate tritium as a by-product of the  $\text{Li}(n,\alpha)\text{T}$  nuclear reactions [2]. Because each neutron generated in the core

burning plasma consumed one T nucleus, and there are inevitable losses of T elsewhere in the breeding and fuel processing system, each fusion neutron must create on average more than one tritium atom in the blanket in order to achieve tritium sustainability. This tritium multiplication factor is called the Tritium Breeding Ratio (TBR) and must be above unity for D-T based fusion energy to be a viable energy source.

Neutronics calculations of the TBR [3-4] for various reactor designs have been conducted taking into account the influence of structural materials, blanket cutouts (for heating, pumping and diagnostic access), neutron multiplication (by e.g. Be) and other factors associated with the blanket system. Aside from the attenuation of the neutron flux by the plasma-facing armor, surprisingly little effort has been dedicated to including the effects of plasma-material interactions on the TBR. Recently, a study was published taking into account the loss of tritium due to implantation and trapping of ions [5] in the plasma-facing tungsten armor that is necessary to protect the blanket systems from the environment of the confined plasma. That work showed that the maximum allowable trapping probability of fuel in the material surrounding the plasma was about  $10^{-6}$  or smaller, implying that plasma-wall interaction effects on tritium self-sufficiency need to be carefully considered. Motivated by this earlier work, this paper examines the impact of armor material erosion and subsequent tritium co-deposition on fuel self-sufficiency for a variety of proposed wall material choices.

## I. Calculations

We employ the same particle balance model used earlier [5, 6] to estimate a necessary TBR. Briefly, this model balances the tritium burn rate in the core plasma and the loss rate of particles trapped in the first wall material with the rate of additional tritium injection necessary to maintain steady-state operation. This model implicitly assumes that tritium removed through the divertor pumping system is not lost, but that the pumped tritium will be recovered with 100% efficiency and eventually re-injected into the plasma. In addition, any tritium that diffuses into the coolant channels is also considered recoverable. In essence the model is simply calculating the amount of additional tritium that needs to be supplied to the fusion reactor to overcome the loss rate due to trapping of tritium in inaccessible regions of the reactor and the amount burned in the fusion process. In steady-state, the rate of tritium injection into the device must equal the loss rate, due to both burning in the plasma and trapping in the surrounding material, such that;

$$\dot{M}_T^{\text{inj}} = \dot{M}_T^{\text{burn}} + (1 - R) \dot{M}_T^{\text{wall}}, \quad (1)$$

where  $R$  is the recycling coefficient.  $R$  thereby encompasses the kinetically reflected fraction of incident particles, as well as the fraction impinging on the wall that eventually effuses back out at thermal energies.  $\dot{M}_T^{\text{inj}}$  is the rate at which T is injected into the plasma and  $\dot{M}_T^{\text{wall}}$  is the rate at

which T impacts on the first wall.  $\dot{M}_T^{burn}$ , the rate at which tritium is burned in the core plasma can be written as;

$$p_{burn}\eta_{fuel}\dot{M}_T^{inj} = \dot{M}_T^{burn} \quad (2)$$

Where  $p_{burn}$  is the probability that an injected tritium nucleus is burned before escaping the plasma and  $\eta_{fuel}$  is the core fueling efficiency. Following the derivation in [6], we arrive at the requirement that the probability of trapping tritium in the materials surrounding the plasma,  $p_{loss}$ , must follow in order to have  $TBR > 1$ ;

$$p_{loss} \ll (TBR-1)(1-R)(p_{burn}\eta_{fuel}/(1-p_{burn}\eta_{fuel})) \quad (3)$$

in order to achieve tritium self-sufficiency, where  $TBR > 1$  and  $R < 1$ . In contrast to [5], here we take the term on the left hand side of the inequality,  $p_{loss}$ , to now include both the loss probability of tritium due to direct trapping of implanted ions (which is what was studied in [5]), as well as the loss probability of tritium due to co-deposition,  $p_{codep}$ .

$$p_{loss} = p_{trapped} + p_{codep} \quad (4)$$

Following the choices in [5], we estimate the TBR as 1.05, the recycling coefficient as 0.99 – 0.999, the fraction of tritium burned as 0.05, the fueling efficiency as 20-30% and solve for  $p_{loss}$ , which (as in [5]) results in the estimation of less than  $10^{-6}$  to  $10^{-7}$  of the incident particle fluence can afford to be lost either via permanent trapping in the wall material or lost via trapping within co-deposits which cannot be subsequently removed. If the loss probability exceeds this limit due to the choice of wall material, then it will not be possible to breed enough tritium to achieve tritium self-sufficiency in a fusion reactor using the D-T fuel cycle.

Clearly we then need to develop an additional loss term to be added to the trapping probability due to implantation to account for the probability of co-deposition of fuel with eroded wall material. This term is estimated as follows.

The rate of co-deposition,  $R_{codep}$ , is;

$$R_{codep} = R_e * 0.5C_T = Y*\Gamma * 0.5C_T \quad (5)$$

where,  $R_e$  is the erosion rate given by the sputtering yield,  $Y$ , multiplied by the ion flux,  $\Gamma$ , to each element of the material surrounding the burning plasma.  $C_T$  is the concentration of tritium fuel contained in the co-depositing material. The factor of 0.5 is included to account for both deuterium and tritium fuel assumed to be co-depositing in equal concentrations. The global co-deposition probability,  $p_{codep}$ , is simply  $R_{codep}$  divided by the total flux of ions striking the wall.

The sputtering yield as a function of incident particle species and energy has been tabulated for most materials [7]. The other terms in Eq. [5] are treated as follows. In order to estimate the

incident particle species and energies, we use the latest estimates obtained using SOLPS [8] modeling of high-power ITER D/T operation. Here we must make the assumption that such modeling will give us a reasonable estimate for a reactor since such detailed flux and energy profiles are not available for any DEMO reactor designs. The SOLPS modeling output can be easily coupled to the WalldYN [9] material transport code that is used to model expected eroded material migration in ITER. It should be pointed out that we do not use the complete WalldYN transport code, but rather simply use it to give us the total eroded flux from various areas, both first wall and divertor, around the burning plasma. This eroded material is then assumed to co-deposit with fuel with a concentration  $C_T$  at some location within the vessel. The distribution of temperatures of the various plasma-facing surfaces are taken from [10] with the minimum surface temperature of any plasma facing surface set to 600°C to account for the higher operating temperatures expected in future reactors. These co-deposits are then taken to permanently sequester the fuel from the tritium processing system.

Next, we must estimate the concentration of fuel contained within these co-deposits,  $C_T$ . Since we do not know the exact locations of co-deposition within the vessel, we assume a uniform temperature across all plasma-facing surfaces where co-deposits can be formed (i.e. in regions where the co-deposits are not re-eroded by plasma ion fluxes). Co-deposits are usually formed at locations with low power fluxes from the plasma, therefore, the assumption of a uniform temperature is justified. Extensive experimental work has been done for estimating the fuel concentration in co-deposits for the three ITER-relevant materials (W, Be and C) [11]. However, the scaling laws developed from the experimental data are strictly only valid over the temperature range studied, in this case from room temperature to 300°C. Since it is generally assumed that a fusion reactor will need to operate at higher temperature than 300°C, for enhanced thermal efficiency, we simply extrapolate the measured results for co-deposition concentration to 600°C, resulting in the semi-log plot of  $C_T$  vs temperature for a variety of wall materials shown in Figure 1.

The result obtained using Eq. [5] is the total accumulation probability of tritium in co-deposited material which is independent of fluence (or operational time). This represents a tritium loss term that is in addition to the ion implantation trapping limits expressed by equation 3. The total probability of tritium lost from the system to the surrounding materials is then the sum of  $p_{trapped}$  and  $p_{codep}$  and it is this sum that must be below  $10^{-6}$  to  $10^{-7}$  (as determined in [5]) to permit a D-T fusion reactor to achieve tritium self-sufficiency. This requirement stems strictly from the PMI perspective and does not include any other loss terms of tritium from the remainder of the fusion reactor system. Including such additional losses would then lower this upper limit further.

## II. Results

We are now positioned to estimate total fuel loss probability due to PMI effects for a variety of possible plasma-facing materials. To simplify each subsequent calculation, it is assumed that all

the solid plasma-facing surfaces are composed of the same material, although in principle it is also possible to define different materials for use in different locations in WallDYN.

First, we will extend the results cited in [5] utilizing tungsten as the plasma-facing material. For the experimental results shown in [5], the rate of tritium trapped due to ion implantation into the wall material steadily decreased with increasing wall fluence. This is due to the fact that the implanted species have to diffuse from the implantation zone to unoccupied trapping sites deeper within the bulk of the material, resulting in a trapped fuel inventory (integrated over the zone from the surface into the bulk of the material) that increases as the square root of time (as has been observed experimentally for plasma exposed tungsten [12]). For fixed ion flux, the ion fluence increases linearly in time, and so the trapping probability of incident fuel particles, given as the ratio of the trapped inventory to the incident fluence, therefore decreases with increasing fluence.

Based on this argument, the probability of fuel being trapped via ion implantation into the wall material will eventually become small enough to satisfy equation 3 above. However, in a steady-state reactor device, the erosion rate of the plasma-facing material and the co-deposition concentration are time independent, so eventually the co-deposition loss channel will begin to dominate the fuel particle loss rate. Thus for devices that operate for a long enough period of time,  $p_{\text{codep}}$  provides us with the dominant tritium loss term for use in the tritium particle balance model of the TBR.

Figure 2 shows the results from [5], where the retention data with increasing fluence was obtained at 340°C. We recently measured retention in the same tungsten material at 600°C; this data point is added in Figure 2. Since there is no reason to suspect that the fluence dependence of retention varies with exposure temperature, we can produce a similar square root of time dependence that was used in [5] at 340°C but now intersecting the new 600°C data point. This dependence is shown as the dotted line in Figure 2. As mentioned previously, for successful tritium self-sufficiency, the tritium loss rate is required to be less than  $10^{-6}$  to  $10^{-7}$  of the incident flux as indicated in Figure 2. As can be seen in the figure, the tritium loss rate at 600°C due to implantation of fuel species into tungsten drops below this value near a fluence of  $1 \times 10^{23} \text{ m}^{-2}$ .

Next, we include the tritium loss rate due to co-deposited fuel trapped within redeposited tungsten. Due to the low erosion rate of tungsten, together with the low  $C_T$  for the co-depositing tungsten, the trapping probability for co-deposited fuel in tungsten, given by the ratio of the tritium co-deposition rate,  $p_{\text{codep}}$ , to the ion flux, is estimated to be  $2 \times 10^{-11}$ . Tritium retention in a tungsten device operating at 600°C will therefore continue to be dominated by trapping within the bulk armor until a fluence of over  $3 \times 10^{30} \text{ m}^{-2}$  is achieved, after which co-deposition will begin to dominate and the tritium loss probability would then saturate at about  $2 \times 10^{-11}$  of the incident particle flux.

We could equally well have performed our calculations using a wall temperature of 340°C, in which case the erosion rate of wall material would not change, but the co-deposition concentration,  $C_T$ , would increase by approximately a factor of 5, as shown in Figure 1. This value is also plotted in Figure 2 to provide a sense of the influence of the operating temperature on the requirements for the TBR. At 340°C, tungsten would still permit tritium self-sufficiency, but with a smaller margin to accommodate other tritium loss terms or for any unjustified assumptions we have made in these estimations.

Similar WallDYN calculations have also been made for different solid plasma-facing materials. In the case of beryllium, retention due to ion implantation quickly saturates at around a fluence of  $2 \times 10^{22} \text{ m}^{-2}$  [13]. This is beneficial for tritium self-sufficiency because the loss probability due to trapping in the wall decreases linearly with increasing fluence. Unfortunately, using the sputtering yield data for beryllium and the  $C_T$  from Figure 1, the tritium trapping probability in co-deposits with beryllium at 600°C is approximately  $2 \times 10^{-5}$ . Figure 3 plots the tritium retention probability as a function of fluence and shows that the ion implantation trapping probability decreases with fluence until co-deposition begins to dominate at a fluence of approximately  $2 \times 10^{25} \text{ m}^{-2}$ . As can be seen in Figure 3, because of co-deposition the tritium loss probability for a device with all beryllium plasma-facing material will never get below the required  $10^{-6}$  to  $10^{-7}$  range, making it impossible for such a steady-state device to achieve tritium self-sufficiency.

Figure 4 shows the results of similar calculations obtained using carbon as the plasma-facing material. In the case of carbon, retention versus fluence data could not be found so only the co-deposition rate is plotted as a flat line. Carbon exhibits a chemical erosion rate which adds to its physical sputtering rate as depicted in Figure 1 of [11]. If the operating temperature of reactor can be kept high enough this chemical erosion term disappears and the total erosion rate equals the physical sputtering rate. Two horizontal lines are depicted in Figure 4, one including the chemical erosion term and one using only physical sputtering (i.e. turning off chemical erosion in WallDYN). Values for  $C_T$  for carbon are taken from Figure 1. Not surprisingly, including the chemical erosion term increases the co-deposition rate with tritium, but even in the case of no chemical erosion of carbon from the plasma-facing material, the co-deposition probability is too large to allow tritium self-sufficiency.

Other low-Z plasma facing materials, such as boron and silicon, do not have co-deposition concentration data available, but since both elements form stable chemical compounds with hydrogen in this temperature range, one would expect their tritium co-deposition concentration to be similar to that of carbon (indeed, it is this factor that likely makes them useful low-recycling coatings in today's physics-focused experiments). The physical sputtering yields of both elements are within a factor of 2-3 of carbon at roughly 100 eV [7], so the co-deposition probability of either is likely in the range between the two values plotted for carbon in Figure 4. Again, the results point towards serious impacts on tritium self-sufficiency.

Not surprisingly, these results closely parallel previous studies [11] on tritium accumulation within the ITER vessel from a safety standpoint. However, imposing a site limit based on safety is open to discussion and is, therefore, in principle a somewhat arbitrary limiting value. Examining the issue of tritium accumulation in the device from the tritium self-sufficiency standpoint provides a more concrete definition of what tritium loss rate is acceptable and what rate simply makes the prospect of fusion power impossible.

A final low-Z material that is getting attention for use in magnetic confinement devices is lithium. In the case of a full flowing conformal liquid wall, the issues of erosion and co-deposition begin to lose meaning, at least from the tritium self-sufficiency point of view. Erosion is still critical from a burning plasma purity viewpoint, but since any deposition would be into a flowing wall material, we would no longer be concerned with trapping or co-deposition with fuel species. Instead one would be concerned with the tritium content in the flowing liquid and the extraction efficiency required to permit real-time recirculation of the liquid metal. Without sufficient tritium extraction efficiency, entrained tritium would begin to build up in the recirculating liquid metal and eventually the resulting large tritium inventory would lead to significant safety concerns. However, the practicality of a fully conformal liquid wall is an open question, so in the following we will confine ourselves to approximating a flowing liquid divertor surface combined with a solid refractory metal first wall, as perhaps being a more practical approach.

WallDYN again provides the sputtering and evaporation rates for a fully-toroidal flowing liquid lithium system located at both the inner (0.30 m poloidal extent) and outer (0.40 m poloidal extent) divertor locations. For simplicity, a full tungsten wall is assumed throughout the remainder of the reactor. Neglecting any plasma changes due to the low-recycling at the strike point locations, we again use the same plasma scenario used in the previous parts of this paper [from 8]. In this case, we need to estimate how much of the eroded Li flux is redeposited on the flowing liquid surface and how much eroded Li flux escapes the divertor region and may co-deposit with fuel in some other location within the reactor.

Again the hypothetical first-wall surface temperature is defined to be 600°C for power conversion efficiency as was done in the solid surface calculation already presented. The divertor surface temperature profile is also assumed to be the same as the solid surface temperature [10] used previously. This is, of course, not correct, as the solid divertor surface would be expected to be at a higher temperature than would be expected while using a flowing surface, but it can provide a worst-case type scenario for material lost from the liquid (i.e. sputtering + evaporation). No splashing of the flowing liquid is included and the complete removal of all the liquid having flowed through the divertor region as assumed.

The maximum loss rate of lithium from the divertor target (at either the inner or outer divertor strike point) is about  $4 \times 10^{26} \text{ m}^{-2}\text{s}^{-1}$ , or roughly 0.01 m/s. However, estimates show that the vast majority of the eroded lithium is redeposited back on the flowing surface and only about  $2 \times 10^{20}$

Li/s ultimately escapes from the flowing surface to deposit elsewhere outside the divertor. The evaporation rate of Li from a surface at 600°C is approximately  $2 \times 10^{23}$  atoms/m<sup>2</sup>s. When this evaporation flux is taken over even a small part of the wall area, the result greatly exceeds the rate of Li escape from the divertor region, and therefore the issue of first-wall co-deposition of T with the escaping Li is not an issue. Assuming all the surfaces accessible to the migrating Li remain at or above 600°C, all the Li escaping the divertor region should evaporate quickly from the hot W walls and eventually find its way, as an impurity in the plasma, back into the divertor. Thus co-deposition losses on the first wall should be negligible; we note parenthetically that trapping of implanted fuel ions in the refractory metal wall [5] would still be relevant.

We now turn our attention to the fate of T trapped within the flowing liquid Li in the divertor region. Li is known to retain close to 100% of an incident flux of hydrogen ion isotopes [14]. If this is the case in a flowing lithium divertor surface, we can estimate the amount of tritium which would be embedded in the recirculating liquid volume based on different purification, or recirculation, times. It has been estimated that a 3 GW fusion power plant will require 0.5 g of tritium injected per second [15]. Assuming a recirculation time of one hour [16] would imply that the total amount of tritium contained in the lithium at any time is about 1.8 kg. The amount of tritium contained in the flowing liquid will be decreased, or increased, depending on the time chosen for effective lithium de-tritiation (one hour in our assumption).

One could also envision using an alternate liquid material, such as gallium, or tin, but the databases for these materials do not exist for calculations using WallDYN. It seems reasonable to expect somewhat smaller escape fractions from the divertor for these heavier elements. However, the evaporation rates of both Ga and Sn are significantly less (more than six orders of magnitude) than that of lithium at 600°C, so one would again need to be concerned with whether co-deposition effects are important. This would require a full WallDYN simulation to determine the migration and deposition patterns on the escaping flux and is presently not possible. Both Ga and Sn would again offer high-recycling divertor operation, so the issue with purification of the recirculating liquid would not be as critical.

#### IV. Conclusion

We have attempted to quantify the role of plasma-material interactions on one of the fundamental issues associated with nuclear fusion energy production - namely tritium self-sufficiency of future reactors. While the calculations presented have made a variety of assumptions, some of which bring along substantial uncertainties, we believe that one can still gain valuable insights from these calculations.

First, for solid wall reactor designs, erosion associated with low-Z plasma-facing materials is a serious concern. Their large erosion rates, coupled with the propensity for these low-Z materials to form hydrogen bonds with the redeposited atoms, results in the accumulation of large

quantities of tritium in co-deposits, and appears to make them unacceptable for use in burning plasma devices. One could devise an engineering solution to the problem of tritium in co-deposits, but removing tritium from co-deposits usually involves activity in the absence of the plasma. Abrasion of the co-deposits [17], or baking the material at temperature higher than the formation temperature [18] are the typical approaches. Clever engineering to force the co-deposits to form only in certain removable locations could also be imagined, but all of the activities will require downtime of the reactor and drive up the cost. A high-Z solid wall material, demonstrated here with tungsten, which does not form strong hydrogen bonds, appears to offer the most favorable approach, and appears to have a low enough overall loss probability,  $p_{\text{loss}}$ , to permit tritium self-sufficiency.

Flowing liquid wall materials might offer some benefits in that they remove any trapped tritium from the plasma confinement region and thus allow for extraction of tritium from the liquid while the reactor is still operating. However, if the liquid material entrains a large amount of the ions incident on the surface, the rate of extraction of tritium from the liquid metal becomes a critical parameter since that parameter determines the tritium inventory sequestered in the recirculating system at all times. Our analysis shows that full T removal timescales of about 1 hour must be achieved in order to limit the T inventory to  $\sim 1-2$  kg.

Admittedly this work has required a variety of assumptions, but it provides a first attempt to include the effects of PMI into the calculations of the tritium fuel cycle. A precise determination of the absolute values of tritium loss probabilities obtained in this exercise is not the goal of this investigation. Rather one can notice trends in different types of materials. For example, most low-Z materials erode at a high rate and tend to form strong bonds with hydrogenic species. These materials present the most challenge from a tritium self-sufficiency point of view. On the other hand, high-Z materials typically offer lower sputtering rates and if the choice is made of a material that does not readily form bonds with hydrogen, then such materials appear to be the preferred choice from a fuel self-sufficiency viewpoint. More refined models, which use actual reactor design values and expected plasma parameters, are needed and these PMI models should be included along with the neutronics calculations that are performed on reactor designs to determine the TBR of specific design choices. Even with our model's shortcomings, it is clear that future reactor materials need to be drawn from a group of materials that offer low erosion and which do not exhibit a strong affinity for hydrogen at the temperatures to be used for plasma-facing materials in a burning plasma environment. If liquid Li divertor solutions are to be seriously considered, then particular attention must be paid to the timescale required to remove all of the T from the Li.

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## References

- [1] J. M. Dawson, Advanced Fusion Reactors, in E. Teller (Ed.): Fusion, Magnetic Confinement, Volume 1, Part B (Academic Press, New York 1981).
- [2] R. Dierckx, Nucl. Instru. & Methods 107(1973)397.
- [3] M. Harb , L. El-Guebaly, A. Davis, P. Wilson, E. Marriott, J. Benzaquen & FESS-FNSF Team, “3-D Neutronics Assessment of Tritium Breeding Capacity and Shielding of Tokamak-Based Fusion Nuclear Science Facility”, Fusion Science and Technology, 72:3 (2017) 510-515,
- [4] P. Pereslavitsev, C. Bachmann and U. Fischer, “Neutronic analyses of design issues affecting the tritium breeding performance in different DEMO blanket concepts”, Fusion Engr. & Design. 109-111 Part B (2016) 1207.
- [5] G. R. Tynan et al., Nucl. Mater. & Energy (2017).
- [6] J.L. Barton , Diffusion, Trapping, and Isotope Exchange of Plasma Implanted Deuterium in Ion Beam Damaged Tungsten, University of California, San Diego, 2016 .
- [7] W. Eckstein, Sputtering Yields, in: R. Behrisch, W. Eckstein (Eds.): Sputtering by Particle Bombardment IV, Top. Appl. Phys. 110 (Springer, Berlin, Heidelberg 2007).
- [8] S.W. Lisgo, A. Kukushkin, R.A. Pitts, D. Reiter, J. Nucl. Mat. 438 (2013) S580.
- [9] K. Schmid, K. Krieger, S.W. Lisgo, G. Meisl, S. Brezinsek, J. Nucl. Mat., 463 (2015) 66.
- [10] A. Kukushkin A. 2008 ITER Report ITER-D-27-TKC6.
- [11] J. Roth et al., J. Nucl. Mater. 390-391 (2009)1.
- [12] R. P. Doerner, M. J. Baldwin, T. C. Lynch and J. Yu, Nucl. Mater. & Energy 9 (2016) 89.
- [13] R. A. Anderl et al., J. Nucl. Mater. 273 (1999) 1.
- [14] M. J. Baldwin et al., Nucl. Fusion 42(2002)1318.
- [15] M. Ono, R. Majeski, M. A. Jaworski et al., Nucl. Fusion 57(2017)116056.
- [16] D. N. Ruzic, private communication.
- [17] E. Tsitrone et al., Nucl. Fusion 49 (2009) 075011.
- [18] G. De Temmerman et al., Nucl. Mater. & Energy (2017).

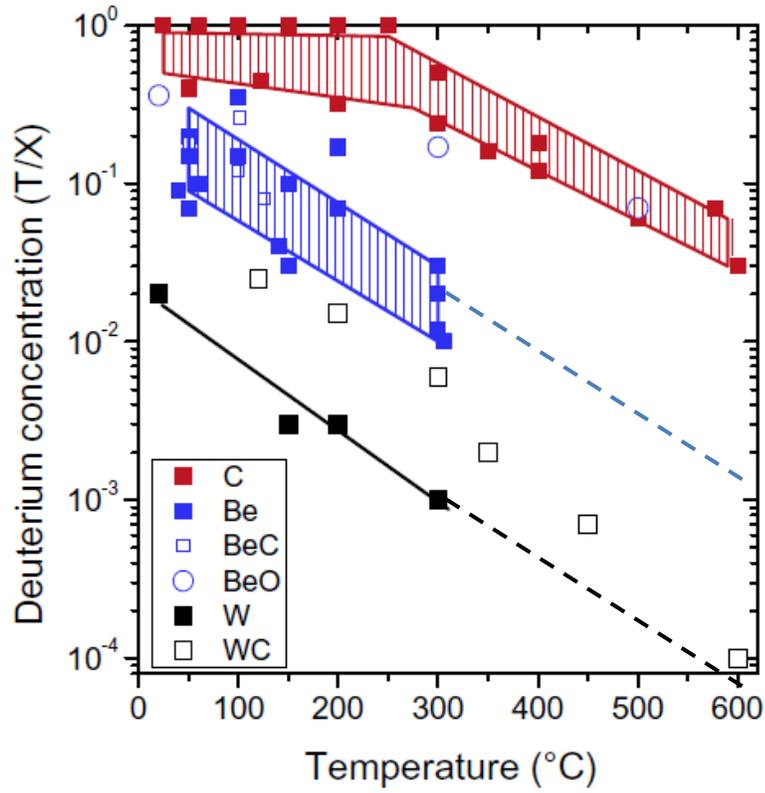


Figure 1 – Extrapolation of measurements deuterium concentration in various material co-deposits [11] to an envisioned first wall operating temperature of 600°C.

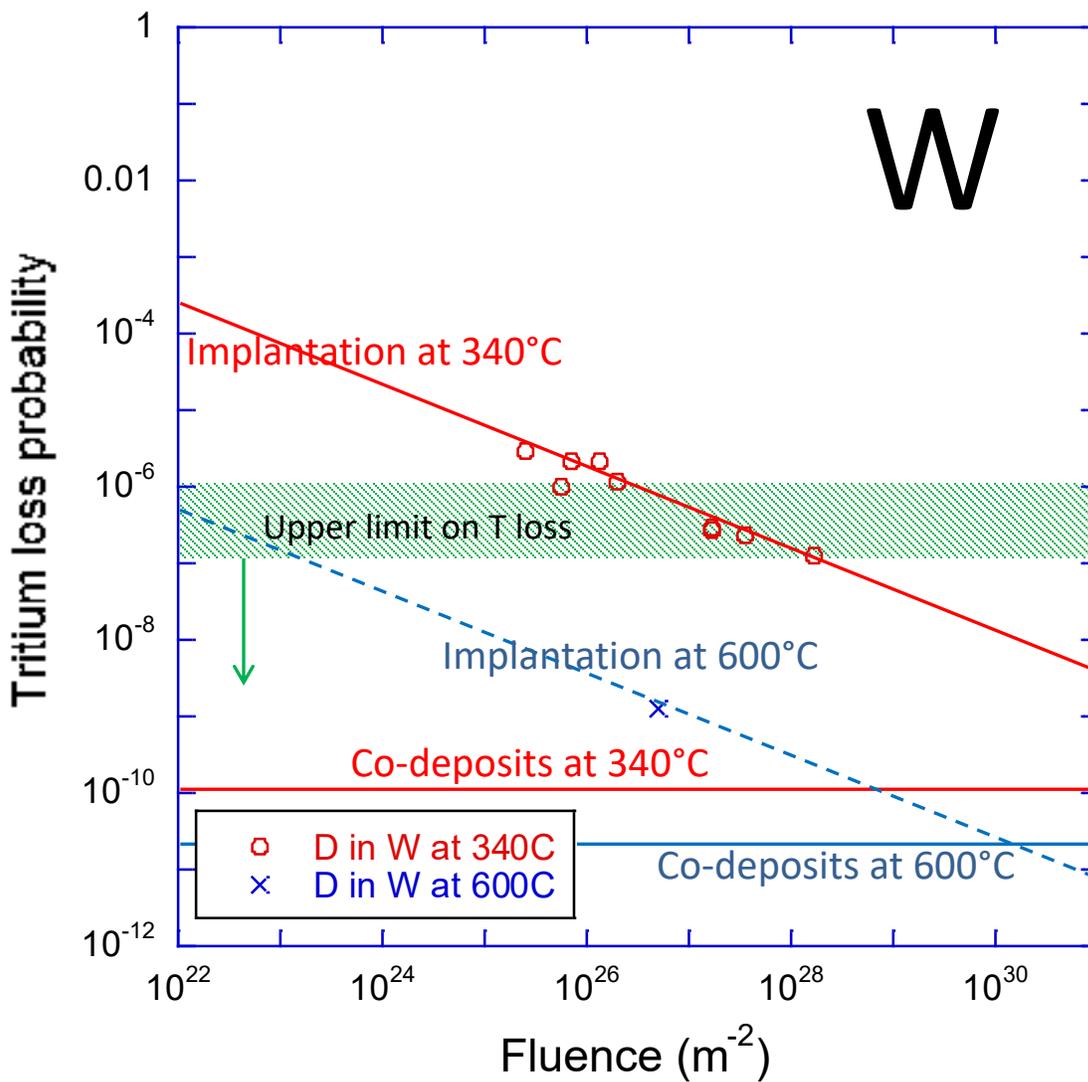


Figure 2 – Retained tritium probability due to implantation and co-deposition at two different temperatures (error bars associated with the data points correspond to the size of the symbols). Calculated upper limit on retained tritium probability for tritium self-sufficiency is also shown.

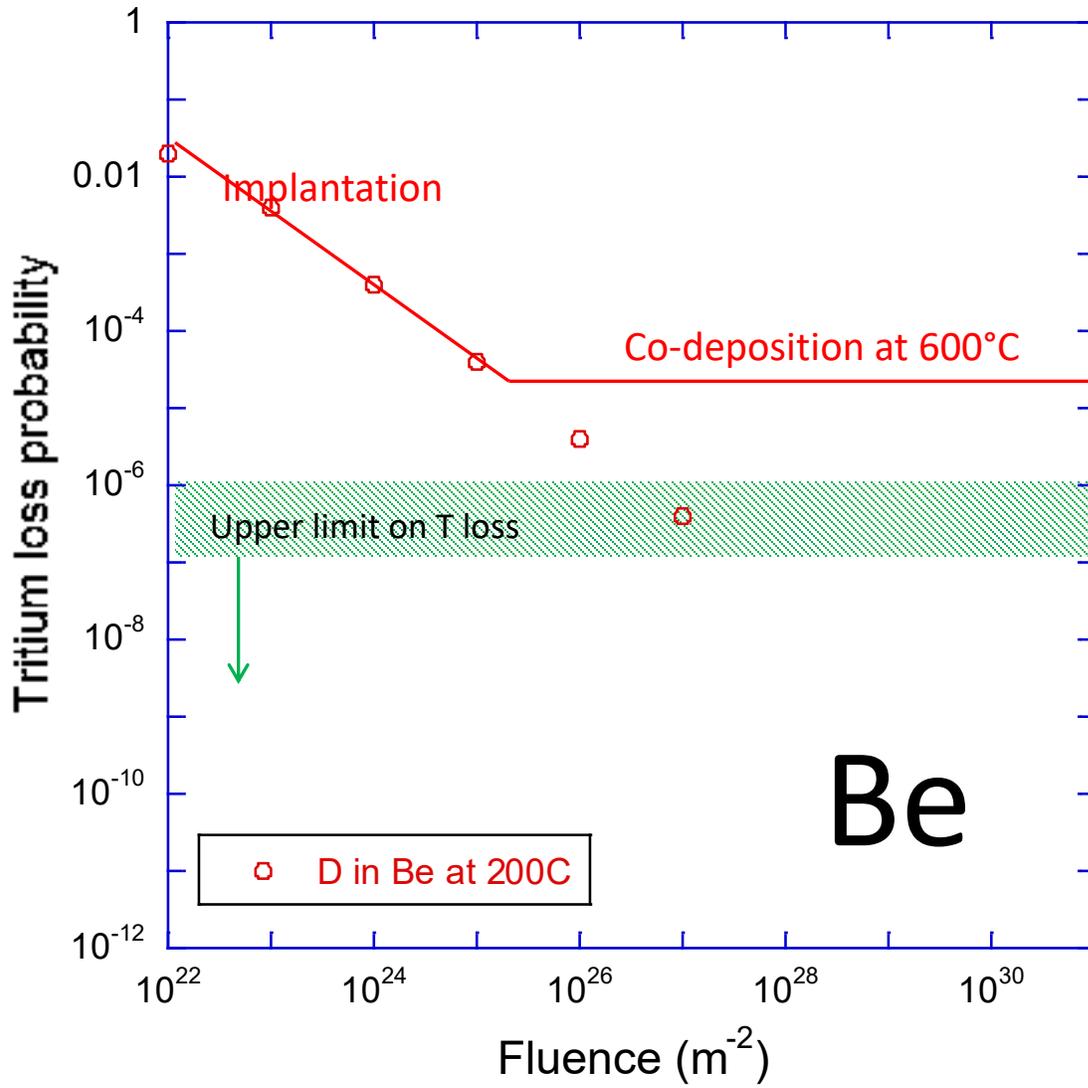


Figure 3 – Value of tritium loss probabilities due to implantation (experimental data at 200°C from [13]) and calculated value due to co-deposition (at 600°C). Loss probabilities are above those required to achieve tritium self-sufficiency.

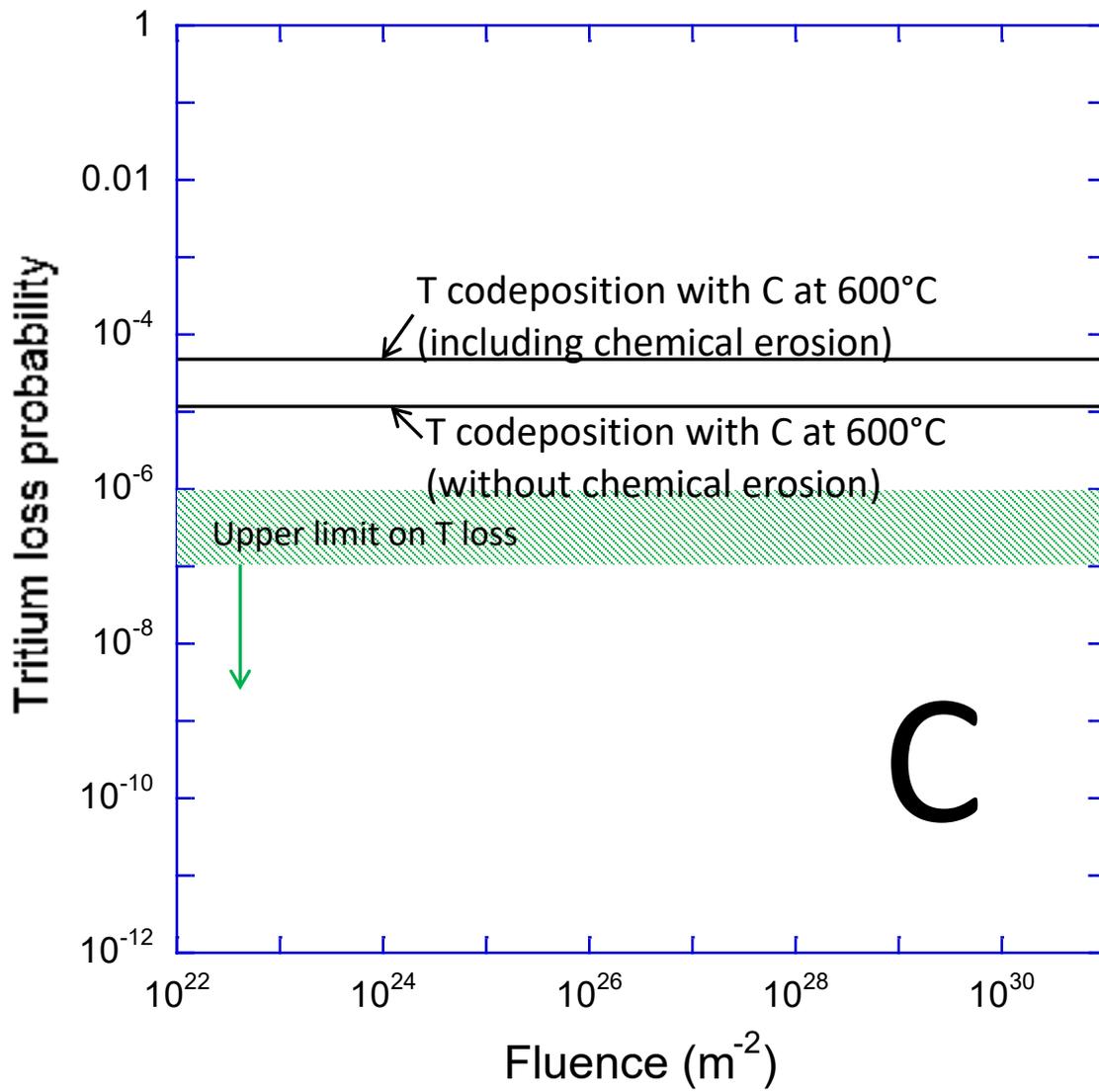


Figure 4 – Value of tritium co-deposition probability with carbon at  $600^\circ\text{C}$ , including chemical erosion of carbon plasma-facing material and without chemical erosion source term.