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Multi machine scaling of fuel retention in 4 carbon dominated tokamaks

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Abstract

In order to benchmark predictions for the in vessel tritium inventory in ITER, a survey of fuel retention measured in 4 carbon dominated tokamaks (TEXTOR, ASDEX Upgrade in the 2002-2003 carbon configuration, Tore Supra and JET) was performed, showing retention rates from ~1 g D/h in TEXTOR (L mode, limiter machine) up to ~6-12 g D/h in AUG (H mode, divertor machine). A simple scaling used for ITER predictions is applied for comparison with experimental values : 1) estimate of wall fluxes, 2) estimate of the gross carbon erosion, 3) estimate of the net erosion/redeposition assuming a redeposition fraction and 4) estimate of the retention rate using D/C ratio scalings. The validity of each step is discussed, showing that this approach yields the right order of magnitude, but tends to

underestimate the experimental values unless a high wall flux, a low local redeposition fraction and/or a high D/C ratio are used.

JNM Keywords: Plasma-Materials interactions (P0500), Carbon (C0100), Hydrogen and hybrids (H0400), First Wall Materials (F0400), Redeposition (R0900)

PSI-19 Keywords: carbon based materials, deuterium inventory, erosion and deposition, ITER, retention

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1. Introduction

Fuel retention in plasma facing components (PFCs) is a crucial issue for next step fusion devices, where the in vessel tritium (T) inventory will be limited for safety reasons. A collaborative effort has been started to model the in vessel fuel inventory in ITER [1], showing the dominant contribution of codeposition with carbon for the initial configuration (carbon divertor, tungsten baffles and beryllium first wall). In order to benchmark the methodology used for ITER predictions, this paper presents as a first step a survey of fuel retention in 4 carbon (C) dominated tokamaks, both in limiter and divertor configuration : TEXTOR, ASDEX Upgrade (AUG) in the 2002-2003 carbon dominated phase, Tore Supra (TS) and JET, while contributions from other tokamaks willing to participate will be included in future work. This study is intended as a test of the applied methodology, and not as a prediction for fuel retention in ITER, since ITER is not a full carbon device.

2. Experimental retention rates in carbon dominated tokamaks

Data on deuterium (D) retention rates are derived from a literature survey for the tokamaks involved, both from particle balance and post mortem analyses (see [2] for a discussion on the discrepancy found between both methods). A range for D retention rates is given for each device, for different plasma conditions and/or from the uncertainties on experimental measurements.

Incident particle fluxes on the main PFCs are also estimated, in order to be scaled with the experimental retention rates. Selecting which PFC (divertor/limiter versus main chamber) is the most relevant for retention studies is still a subject of discussion. Moreover, it is worth noticing that values found in the literature for wall fluxes are scarce, and with large uncertainties. For TS and TEXTOR, incident fluxes on the limiter, identified to be the main erosion source, are given. In divertor machines, the main chamber is thought to be the main source of erosion and subsequent redeposition in the divertor, while divertor fluxes can play an important role in the redeposition processes. In contrast with AUG, the main chamber particle flux in JET is estimated to be significantly lower than the divertor flux (see [13]). It is also found to be on the low side compared to main chamber fluxes in AUG, despite the more compact size of AUG, which could be linked to different plasma conditions (operation at higher density in AUG, see figure 8 in [13]). In the present study, wall fluxes are used for both divertor machines, JET and AUG, but those data need to be consolidated for a sound extrapolation.

Details are given below, and the resulting retention rates are shown in Figure 1.

2.1 Tore Supra

TS is a circular limiter tokamak with actively cooled components operating at 120°C (15 m^2 of carbon PFCs, out of which 7.5 m^2 for the toroidal pump limiter (TPL)). Data are taken from a dedicated particle balance campaign, where long pulses were repeated with no conditioning in between (P_{LH} = 2 MW, L mode, 18 157 s of cumulated plasma time) [3]. Particle balance integrated over the campaign (taking into account fuel recovery after the discharge, long term outgassing etc) yields a retention rate of 1.7×10^{20} D/s, corresponding to ~50% of the gas injection. The accuracy of particle balance is estimated to be ±10% [4], giving a final range of $1.5\text{-}1.9\times10^{20}$ D/s. From a first step of post mortem analyses [5], retention rates of $\sim8\times10^{19}$ D/s have been found (accuracy ±20%). From 0D modelling of particle balance and experimental measurements of SOL profiles, the incident particle flux on the TPL is estimated to be 1.5×10^{22} D/s.

2.2 ASDEX Upgrade

AUG is a divertor tokamak which has switched progressively from a carbon to a tungsten configuration (~ 40 m^2 of PFC out of which ~ 6 m^2 of divertor), running at room temperature with no active cooling. Data are taken from the 2002-2003 campaign, when AUG was still a carbon dominated machine and performed regular boronisations as well Helium Glow

Discharge Cleaning after the shot (see [6] [7] [8] for a detailed description of the machine configuration and operation). From [9] and [10], where particle balance is shown on a standard H mode discharge, a retention rate of 1.25-3.5×10^{21} D/s can be deduced for a high density scenario (0.5 - 1.4×10^{22} D retained at the end of the 4s discharge respectively from [10] and [9]). Integrating over the campaign, and taking into account a retention corresponding to 10-20 % of the gas injection, as found in [8], a retention rate of 5×10^{20}-10^{21} D/s is derived from the total gas injected (79.1 g of D injected in 2002-2003 [6]). From post mortem analyses [6] [7] , a retention rate of 1.4-1.9×10^{20} D/s is derived for the 2002-2003 campaign (4856 s). The main chamber flux is estimated to range between 2×10^{22} (low flux) and 5×10^{22} (high flux) D/s for H mode discharges in AUG [16], and more generally from 2×10^{21} to 8×10^{22}.depending on plasma conditions (see figure 8 in [13]). The divertor flux is estimated around 5×10^{22} – 10^{23} D/s.

2.3 TEXTOR

TEXTOR is a limiter tokamak, running at high temperature (from 150 to 350 °C) with no active cooling of the PFCs (9.5 m^2 of carbon PFCs, out of which ~3.5 m^2 of main limiter). Gas balance integrated over the campaign shows that ~10% of the gas injection is retained, giving a retention rate of 10^{20} D/s, while post mortem analyses yields a retention rate of 3.6×10^{19} D/s [11]. The particle flux on the limiter is estimated to be in the range 3×10^{21} D/s 10^{22} D/s [12]. The upper value of 10^{22} D/s is taken here.

2.4 JET

JET is a divertor tokamak (~ 20 m^2 of divertor, out of which ~10 m^2 of vertical targets), operating with a first wall at 200°C (divertor at 50°C) and no active cooling of PFCs. Gas balance data are taken for 3 plasma scenario, for L mode (2MW), type III ELMy H mode (6 MW) and type I ELMy H mode (13 MW of heating power) (see [2] for detailed plasma conditions), covering a range of D retention from 8×10^{20} to 2.1×10^{21} D/s if averaged over the divertor phase. Retention rate for an "average" JET discharge is estimated to be lower, in the range 4-8 10^{20} D/s [2]. Post mortem data are taken for the MkII-SRP divertor during the 2001-2004 campaign [14], giving a retention rate of $2.2\text{-}4.1\times10^{20}$ D/s depending if averaged over divertor (94 000 s) or heating time (50 400s) [2]. The particle flux on the divertor is estimated between $5\times10^{22}\text{-}10^{23}$ D/s, while main chamber fluxes are between $5\times10^{21}\text{-}5\times10^{22}$ D/s [13]. For the specific shots listed above, wall fluxes between 1-3 $\times10^{22}$ D/s are estimated from [13].

2.5 Results

Results are shown in Figure 1 as a function of wall fluxes [1], in terms of g D/h for comparison with the ITER T in vessel inventory limit of 700 g [1]. Retention rates from post mortem analysis are compared to campaign averaged gas balance in Figure 1a) while in Figure 1b), gas balance data for specific conditions are shown with corresponding values calculated from the simple scaling described in section 3, as well as predictions for ITER. Indeed, the scaling estimates the instant fuel retention rate from codeposition of D with C, therefore corresponding better with gas balance than post mortem analysis, as it does not integrate fuel recovery processes.

Although the retention data shown here represent a wide variety of plasma conditions (L mode, H mode, campaign averaged) as well as operating conditions (wall temperature, conditioning …), they are seen to increase roughly linearly with wall fluxes, rather than machine size or PFC surface,. For instance, the retention rate is comparable in carbon dominated AUG and JET, although JET is larger than AUG, showing that the wall flux is probably the most relevant parameter as expected. However, due to the large uncertainties on the wall fluxes data used here, this scaling should be taken with caution. More work is needed on specific shots where retention rates as well as wall fluxes are carefully assessed.

[1] For wall fluxes in JET and AUG in Figure 1a), an arithmetic average is taken on data from figure 8 in [13], but lower values would probably be more realistic, as the "average" shot is generally found to correspond to a low performance ohmic discharge at low density.

From particle balance in Figure 1a), the campaign averaged retention rate increases from ~1 g D/h in TEXTOR (L mode, limiter machine) up to ~ 6-12 g D/h in AUG (H mode, divertor machine). For specific high performance shots, it can reach ~ 10-25 g D/h in JET and up to ~ 15-40 g D/h in AUG as seen in Figure 1b). The same trend is seen from post mortem analysis in Figure 1b), with campaign integrated retention rates from ~0.4 g D/h in TEXTOR up to ~2.5-5 g D/h in JET.

This is to be compared with the predicted range of retention due to carbon in ITER full performance conditions, mainly based on simulations with the ERO code. Predictions range from ~30 g T/h taking into account a carbon divertor [1], up to ~115 g T/h for a hypothetical full carbon configuration [15] [2].

3. Simple scaling of retention for ITER

Besides the simulations with the ERO code, simple estimates of retention for a full carbon ITER have been attempted, based on the assumption that C deposition in the divertor is mainly due to the C erosion source from the first wall, as was found in AUG. The following approach is used (see [16] for details) :

- Step 1 : estimate of wall fluxes using different models/scalings, allowing for a low flux/high flux range
- Step 2 : estimate of the gross carbon erosion rate Γ_{gross}, using a fixed erosion yield of 2%
- Step 3 : estimate of the net erosion/redeposition rate Γ_{net} assuming a redeposition fraction ε_{redep} , which corresponds to the fraction contributing to the building up of the deposited layers ($\Gamma_{net} = \varepsilon_{redep} \Gamma_{gross}$ with ε_{redep} = 100% for the low flux case and ε_{redep} = 50% for the high flux case , as was arbitrarily chosen in [16])

[2] When taking into account the beryllium (Be) first wall, retention due to carbon is reduced in the simulations, in the range 2-18 g T/h, depending mainly on assumptions on the Be fraction in the incident flux in the divertor (0.1-1 %), while codeposition of T with Be becomes significant (~20-60 g T/h in total for both C and Be, for 0.1 to 1% of Be in the incident flux respectively [15]).

- Step 4 : estimate of the D retention rate using D/C ratio scalings as a function of incident energy and wall temperature [17], which can play a prominent role in the D content of the deposited layers.

This yields a range of T retention for a full C ITER of 2-30 g T/h [16], depending on the assumption chosen (low/high flux, surface temperature of the targets), below the value computed with ERO (115 g T/h [15]), which takes into account both main chamber and divertor erosion. The same procedure is applied to the 4 tokamaks involved for comparison with experimental retention rates, except that a fixed value is used for the D/C ratio instead of the scaling of [17]. Two options were considered : D/C=0.1 corresponding to redeposited layers found in areas exposed to the plasma (TEXTOR, TS), or D/C=1, corresponding to soft layers found in remote areas (JET, AUG) [2]. Results are summarised in Table 1, where estimates are also given for ITER (5-50 g T/h are found for the fixed D/C ratio options considered here instead of 2-30 g T/h for the D/C ratio as a function of target temperature used in [16]). As shown in Figure 1b), this simple approach yields the right order of magnitude for retention rates, but tends to underestimate the experimental values unless a high wall flux, a low local redeposition fraction and/or a high D/C ratio are used, as described in the assumptions of steps 1, 3 and 4 above. The discrepancy between the scaling and the experimental data seems larger for divertor machines (AUG, JET), where main chamber flux is used, than for limiter machines (TEXTOR, TS), where limiter flux is used. However, as already mentioned, more work is needed to consolidate the wall flux data before interpreting further these discrepancies.

4. Discussion

The uncertainties on the wall fluxes and redeposition fractions (step 1 and 3 described above) are large but will not be discussed further here. A refined calculation of the carbon erosion source has been carried out for TS and will allow discussing step 2, while step 4 will be discussed in the light of post mortem analysis.

4.1 Refined carbon erosion source

Experimental SOL profiles ($n_e(a) = 2\times10^{18}$ m^{-3}, $T_e(a) = 30$ eV, $T_i(a)=100$ eV) and surface temperature measurements have been used as an input to estimate the C erosion source from the TPL of TS in the scenario used for the dedicated particle balance campaign. Physical, chemical and self sputtering are calculated using [18] [3], assuming a fraction Λ_C of carbon in the incident D ion flux of 4% (assumed as C^{4+}), consistent with experimental findings (see figure 3a in [5]). The gross C erosion is calculated to be 7.3×10^{20} C/s, in agreement with experimental measurements [20], corresponding to an equivalent C erosion yield of 4.8 % (in terms of C/D^+), higher than the 2% assumed in section 3, and ~80% of local redeposition. In the TS conditions, self sputtering contributes for half of the gross C erosion source while chemical erosion is not significant. Neutrals could also add an additional contribution (~30 % of the ion flux impinges on the limiter as neutrals as calculated with the Eirene code), as well as enhanced re-erosion of the deposited C layers, not taken into account here. This refined increased C erosion source would allow to match the experimental D retention rate in TS with a D/C ratio closer to findings from post mortem analysis.

More generally, as an illustration, the total C erosion yield from physical, chemical and self sputtering (assuming Λ_C = 2% and T_e=T_i here) is calculated as a function of surface temperature for conditions roughly typical of today's limiter (T_e=100 eV, particle flux 10^{22} m^{-2}s^{-1}) or divertor tokamaks (T_e=10 eV, particle flux 10^{22} m^{-2}s^{-1}) as well as for ITER divertor (T_e=10 eV, particle flux 10^{24} m^{-2}s^{-1}). Please note that only one value was changed at a time between the 3 sets of parameters proposed, which leads to somewhat underestimate the particle flux for today's divertors (rather 5×10^{22} m^{-2}s^{-1} than 10^{22} m^{-2}s^{-1} in AUG for instance)

[3] The reader is referred to [19] for the correct Roth formula for chemical erosion, where the truncated Maxwellian correction should be implemented with care.

or overestimate the electron temperature for the ITER divertor case (present SOLPS simulations predict T_e = 2 eV rather than 10 eV at the strike point). Results are presented in Figure 2 a), showing that, when $T_i=T_e$ is assumed, the 2% erosion yield assumed in section 3 is a rather conservative assumption for typical low plasma temperature/high density divertor cases, while it underestimates the erosion yield in high temperature limiter conditions. According to the Roth formula, the relative contribution of chemical erosion to the total erosion yield is seen to be higher for today's divertor conditions than for limiter (due to high T_e) or ITER (due to high particle flux) conditions. The importance of the T_i/T_e ratio, shown to be >1 in many SOL conditions [21], is illustrated in Figure 2 b), where the total erosion yield is calculated for the same conditions as in Figure 2a), with the additional assumption of 100°C for the limiter/divertor and 800°C for ITER. It shows that for large T_i/T_e (which could be expected for ITER on the first wall, but not necessarily on the divertor due to collisionality), the assumption of 2% might underestimate the erosion yield.

4.2 Discussion of the D/C ratio

The scaling described in section 3 uses the D/C ratio estimated from [17] with the local surface temperature calculated for the PFC substrate. However, codeposition will lead to C layers deposited either in areas in view of the plasma conductive or radiative heat loads, in which case they will be hotter than the surrounding PFC substrate due to their bad thermal conductivity, or in remote areas hidden from heat fluxes, in which case they will be colder (typically the local cooling temperature of the PFC). This is why an approach corresponding to the 2 extreme cases described above (D/C = 0.1 for exposed layers and D/C = 1 for remote layers), in agreement with values from post mortem analysis [2], has been preferred here.

5 Conclusion

A survey of fuel retention in 4 present day carbon dominated tokamaks has been performed, showing that it scales roughly linearly with wall fluxes as expected. Campaign averaged particle balance results in retention rates from ~1 g D/h in TEXTOR (L mode, limiter machine) up to ~ 6-12 g D/h in AUG (H mode, divertor machine), while in specific high performance shots, it can reach up to ~ 15-40 g D/h (AUG). A simple scaling used for predictions of in vessel T inventory in ITER was applied for benchmarking with these experimental values : 1) estimate of wall fluxes, 2) estimate of the gross carbon erosion, 3) estimate of the net erosion/redeposition assuming a redeposition fraction and 4) estimate of the retention rate using D/C ratio scalings. This simple approach yields the right order of magnitude, but tends to underestimate the experimental values unless a high wall flux, a low local redeposition fraction and/or a high D/C ratio are used. The present study on carbon dominated devices is intended only as a first step, as data on tungsten and beryllium are necessary for estimates of retention for the activated phase of ITER. While data on tungsten are becoming available both from laboratory and tokamak experiments with the full tungsten AUG configuration [10], they are urgently needed for beryllium. Indeed, codeposition with beryllium is identified as the major contributor to the retention rate in the activated phase of ITER [1] [15]. The JET ITER like wall should provide the necessary input from 2011 on.

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References

[1] : J. Roth, E. Tsitrone, A. Loarte, et al., Journal Nucl. Mat. 390-391 (2009), 1-9

[2] : T. Loarer, Journal Nucl. Mat. 390-391 (2009), 20-28

[3] : B. Pégourié, C. Brosset, E. Tsitrone, et al., Journal Nucl. Mat. 390-391 (2009), 550-555

[4] : J. Bucalossi, C. Brosset, B. Pégourié et al., Journal Nucl. Mat. 363-365 (2007), 759-763

[5] : E. Tsitrone, C. Brosset, B. Pégourié et al., Nucl. Fusion 49 (2009)

[6] : M. Mayer, V. Rohde, G. Ramos et al., Nucl. Fusion 47 (2007) 1607-1617

[7] : M. Mayer, V. Rohde, K. Sugiyama et al., Journal Nucl. Mat. 390-391 (2009), 538-543

[8] : V. Mertens, G. Haas, V. Rhode, ECA 27A, P-1 128, 2003

[9] : V. Rhode, V. Mertens, A. Scarabosio et al., Journal Nucl. Mat. 390-391 (2009), 474-477

[10] : V. Rohde, M. Mayer, V. Mertens et al., Nucl. Fusion 49 (2009)

[11] : M. Mayer, V. Philipps, P. Wienhold et al., Journal Nucl. Mat. 290-293 (2001), 381-388

[12] : A. Kreter, A. Kirschner, P. Wienhold et al., Phys. Scr. T128 (2007) 35-39.

[13] : G. Matthews, Journal Nucl. Mat. 337-339 (2005), 1-9

[14] : J. Likonen, JP Coad, DE Hole et al., Journal Nucl. Mat. 390-391 (2009), 631-634

[15] : A. Kirschner, D. Borodin, S. Droste et al., Journal Nucl. Mat. 363-365 (2007), 91-95

[16] : B. Lipschultz, J. Roth, A. Kallenbach et al., ITPA SOL divertor report, MIT internal report PSFC/RR-10-4

[17] : R. P. Doerner, M. J. Baldwin, G. De Temmerman, Nucl. Fusion 49 (2009)

[18] : J. Roth et al., Journal Nucl. Mat. 337-339 (2005), 970-974

[19] : D. Naujoks, in Plasma material interaction in controlled fusion, Springer-Verlag Berlin and Heidelberg GmbH & Co. K, illustrated edition (2006), Springer Series on Atomic, Optical, and Plasma Physics

[20] : Y. Marandet et al., these proceedings

[21] : M. Kocan et al, these proceedings

Tables

	TEXTOR	TS	AUG	JET	ITER
Main PFC particle flux (10^{22} D or D+T/s)	1 (limiter)	1.5 (limiter)	2 - 5 (main cham)	1 - 3 (main cham)	10 - 100 (main cham)
Gross carbon erosion source (10^{20}C/s)	2	3	4 - 10	2 - 6	20 - 200
Retention rate (10^{20} D or T/s), D/C = 0.1	0.1 - 0.2	0.15-0.3	0.4-0.5	0.2-0.3	1 - 5
Retention rate (10^{20} D or T/s), D/C = 1	1 - 2	1.5 - 3	4 - 5	2-3	10 - 50
Experimental retention rate (10^{20} D/s)	1	1.5 - 2	15 - 40	8 - 20	

Table 1 : Main PFC particle flux (limiter for TEXTOR and TS, first wall for AUG, JET and ITER as explained in section 2), calculated gross carbon erosion rate assuming a 2% erosion yield and fuel retention rate for 2 values of D/C ratio according to the scaling described in section 3. Experimental retention rates from section 2 are also given. Data are in D/s except for ITER (in D+T/s for the wall flux, in T/s for the retention rate).

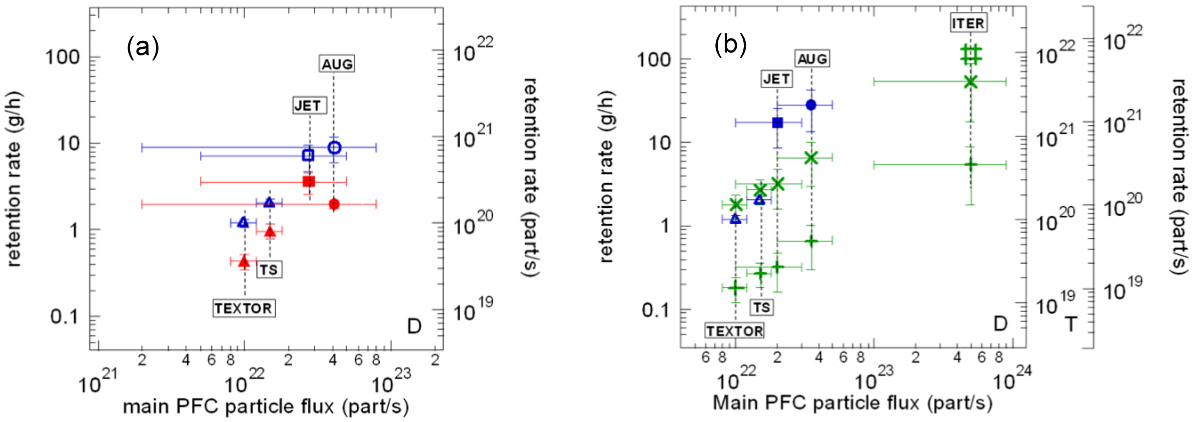
Figure captions

Figure 1 a) : Experimental retention rates for carbon dominated devices (TEXTOR, TS, AUG and JET) as a function of wall fluxes, both from campaign averaged gas balance (blue open symbols) and post mortem analysis (red closed symbols). A detailed description of the experimental data is given in section 2.

Figure 1 b) Experimental retention rates for carbon dominated devices (TEXTOR, TS, AUG and JET) as a function of wall fluxes from gas balance for specific conditions (blue symbols) and results from the scaling of section 3 (green symbols) for 2 values of D/C ratio. Predictions for a full carbon ITER are indicated, both from the scaling and from the ERO code simulations. Corresponding data can be found in Table 1. The left scale corresponds to g D/h for TEXTOR, TS, AUG and JET, and to g T/h for ITER. The associated right scale is double and gives the corresponding D/s for TEXTOR, TS, AUG and JET, and T/s for ITER.

Figure 2 : a) Total carbon erosion yield from physical, chemical and self sputtering (assuming Λ_C = 2% and T_e=T_i) is calculated as a function of surface temperature for conditions roughly typical of today's limiter (T_e=100 eV, particle flux 10^{22} $m^{-2}s^{-1}$) or divertor tokamaks (T_e=10 eV, particle flux 10^{22} $m^{-2}s^{-1}$) as well as for ITER divertor (T_e=10 eV, particle flux 10^{24} $m^{-2}s^{-1}$). The chemical erosion yield alone is also shown (dashed line). The 2% erosion yield used in section 3 is shown for reference (black line). **b)** Total erosion yield as a function of the T_i/T_e ratio for the same conditions as in Figure 2a) in terms of Te and particle flux, with a surface temperature of 100°C assumed for the limiter/divertor and 800°C for ITER.

Figure 1



(a)
retention rate (g/h)
retention rate (part/s)
main PFC particle flux (part/s)
AUG
JET
TS
TEXTOR
D
(b)
retention rate (g/h)
retention rate (part/s)
Main PFC particle flux (part/s)
ITER
AUG
JET
TS
TEXTOR
D
T

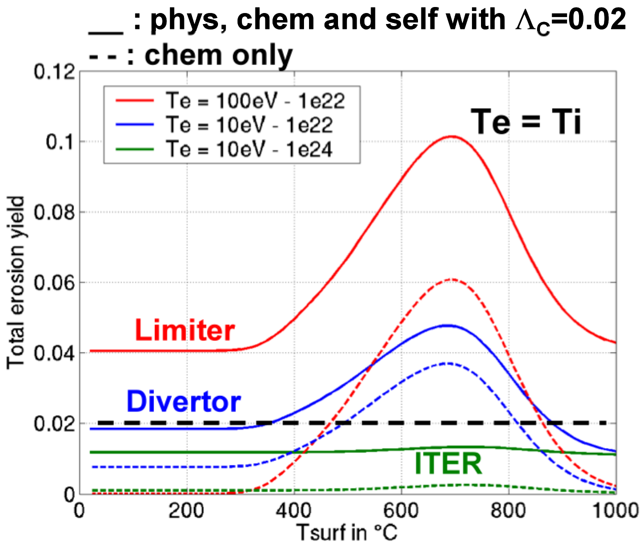


Figure 2 a

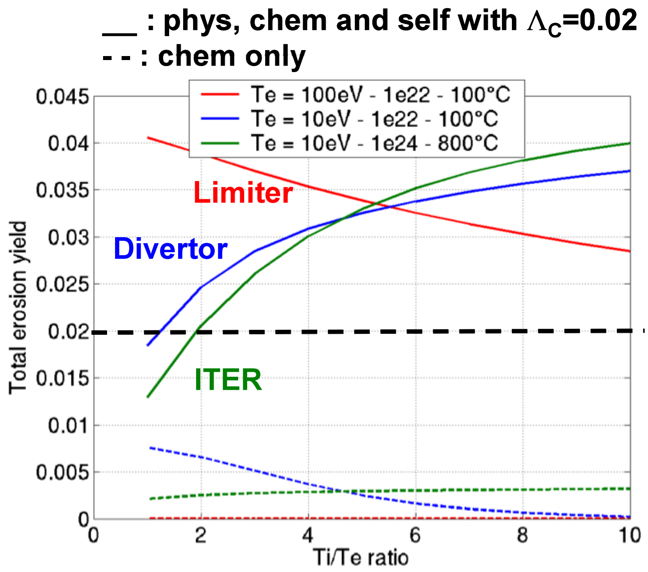


Figure 2b