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In-vessel tritium retention and removal in ITER

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Abstract

Tritium retention inside the vacuum vessel has emerged as a potentially serious constraint in the operation of the International Thermonuclear Experimental Reactor (ITER). In this paper we review recent tokamak and laboratory data on hydrogen, deuterium and tritium retention for materials and conditions which are of direct relevance to the design of ITER. These data, together with significant advances in understanding the underlying physics, provide the basis for modelling predictions of the tritium inventory in ITER. We present the derivation, and discuss the results, of current predictions both in terms of implantation and codeposition rates, and critically discuss their uncertainties and sensitivity to important design and operation parameters such as the plasma edge conditions, the surface temperature, the presence of mixed-materials, etc. These analyses are consistent with recent tokamak findings and show that codeposition of tritium occurs on the divertor surfaces primarily with carbon eroded from a limited area of the divertor near the strike zones. This issue remains an area of serious concern for ITER. The calculated codeposition rates for ITER are relatively high and the in-vessel tritium inventory limit could be reached, under worst assumptions, in approximately a week of continuous operation. We discuss the implications of these estimates on the design, operation and safety of ITER and present a strategy for resolving the issues. We conclude that as long as carbon is used in ITER – and more generically in any other next-step experimental fusion facility fuelled with tritium – the efficient control and removal of the codeposited tritium is essential. There is a critical need to develop and test in situ cleaning techniques and procedures that are beyond the current experience of present-day tokamaks. We review some of the principal methods that are being investigated and tested, in conjunction with the R&D work still required to extrapolate their applicability to ITER. Finally, unresolved issues are identified and recommendations are made on potential R&D avenues for their resolution.

Keywords: ITER; Tritium inventory; Divertor erosion; Plasma facing materials; Hydrogen retention

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1. Introduction

The International Thermonuclear Experimental Reactor (ITER) is envisioned to be the next major step in the world's tokamak fusion program, designed to study fusion plasmas with a reactor-relevant range of plasma parameters. During normal operation, it is expected that some of the unburned tritium (T) that is used to routinely fuel the discharge will be retained, together with deuterium (D), on the surfaces and in the bulk of the plasma-facing materials (PFMs) surrounding the core and divertor plasma. To date, there are still large uncertainties in quantifying the in-vessel tritium inventory of ITER [1]. They arise mainly from the uncertainties of plasma edge physics parameters, which are anticipated to strongly affect the erosion, deposition and codeposition patterns and rates. Moreover, mixed-materials effects, arising from the simultaneous use of different PFMs to protect the in-vessel components, introduce significant uncertainties for the operation of a tokamak like ITER.

As part of the ITER R&D program, new relevant data are becoming available in areas where no data existed or where data were largely scattered. Based on these data and on the advances in the related modelling, it is now possible to conclude that for ITER, the dominant mechanism for hydrogenic retention is codeposition with carbon. Retention by other mechanisms such as implantation and surface absorption, which may be significant for small short-pulse machines, is expected to rapidly reach saturation in ITER.

The organisation of this paper is as follows. Section 2 provides an overview of the design and operating conditions of the main components which define the plasma boundary of ITER. Section 3 reviews the database and the results of recent relevant experiments conducted both in laboratory facilities and in tokamaks. These data provide the experimental basis and serve as an important benchmark for both model development (discussed in Section 4) and calculations (discussed in Section 5) that are required to predict tritium inventory build-up in ITER. Section 6 emphasises the need to develop and test methods to remove the tritium from the codeposited C-based films and reviews the status and the prospects of the most attractive techniques. Section 7 identifies the remaining R&D priority needs. Finally, a summary is provided in Section 8.

Although, this review touches on most aspects of experimental research related to the subject of tritium retention and removal, there are a number of important areas which have not been covered. In particular, the subject of tokamak dust, and the related aspects of its production in tokamaks, and methods of diagnostics and removal, although important, have not been treated here because they are considered to be outside the scope of this review.

2. ITER PFC design and operation overview

A detailed description of the design features and operational aspects of the ITER plasma-facing components (PFCs) is given in Ref. [2] and references therein.

A cross-sectional view of ITER, showing the various PFCs, is portrayed in Fig. 1. The divertor PFCs consist of: (1) a *vertical target* outboard of the separatrix clad with CFC in the lower part, near the strike-points and elsewhere, (2) a *target/liner* assembly clad with CFC on a small area of the dump target and with W on the majority of the surface, and (3) a W-clad *centre post*. Although the utilisation of CFC is known to exacerbate the problem of tritium codeposition and mixing of materials, its use at the divertor target, near the strike points, is viewed as indispensable to withstand, with a reasonable erosion lifetime, the high thermal loads expected during attached-plasma transients, ELMs and disruptions.

The majority of the *first-wall* is clad with Be, primarily because of its plasma-compatible low atomic number, its good oxygen gettering properties, and its ability to protect the underlying first-wall structures from the off-normal thermal loads. It also offers the prospect of repair by plasma spray. Direct contact with the plasma is expected only during off-normal events. The *baffles*, whose main function is to retain the recycling neutrals inside the divertor, are not normally in contact with the plasma, but are subject to moderate

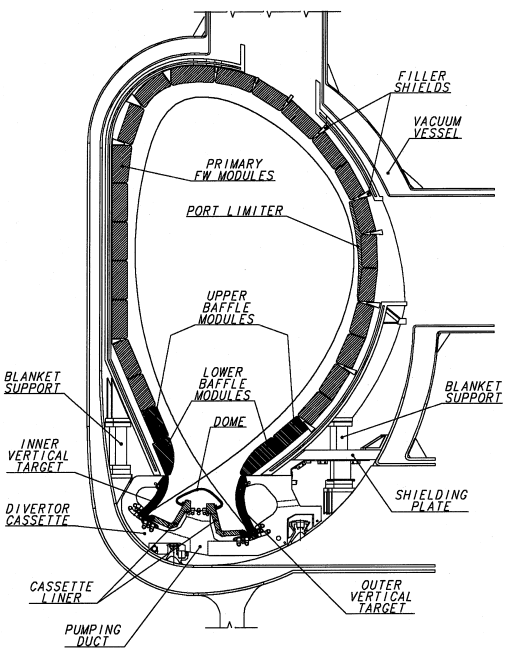

Fig. 1. Cross section of ITER showing the major plasma-facing components.

heat loads caused by MARFEs and the charge–exchange flux arising from neutral–plasma interactions near the entrance to the divertor channel. They are protected with Be in the upper part, while W is used in the lower part near the divertor throat to minimise the sputtering erosion from charge–exchange neutrals due to tungsten's higher sputtering threshold energy compared to that of C or Be. Finally, the Be *port limiters* are located in two of the equatorial ports for improved maintainability.

The set of nominal design parameters and operation conditions during the basic performance phase (BPP) adopted for our analysis is shown in Table 1.

Erosion/redeposition is one of the crucial physical issues to be faced in the design of PFCs for ITER. The erosion of PFMs produces impurity influxes which degrade plasma performance and redistribute the wall material due to poloidally asymmetric transport. Results are now available from several divertor tokamaks, indicating that the first-wall is an area that experiences net erosion, while the divertor has areas of erosion (outer leg) and heavy deposition (inner leg). Dust is also an important consequence of erosion. It represents a potential safety hazard for ITER because it can cause steam-induced hydrogen explosions, and it can increase the spread of radioactivity during an accident involving a sudden vent, if mobilised [3]. The study of dust production has just begun and is not included in this review.

The amount of material eroded from the PFMs by sputtering, chemical reaction, ablation and melt-layer loss will be much greater (at least by a factor 10^3) for ITER compared to present-day tokamaks because of the longer pulse length and cumulative run-time, together with the high heat loads and densities and more intense disruptions. For the most severe design assumptions, the erosion lifetime of the divertor PFCs will be relatively short, requiring about three replacements during the BPP. In present experiments, by contrast, the net erosion is barely measurable. Thus, for the first time in fusion research, erosion and its consequences (e.g., codeposition) will determine the operational schedule of a fusion device. As laboratory measurements have shown that these codeposited films, when exposed to air, completely decompose releasing tritium at a temperature $\geqslant 250°C$, they are considered a risk under ITER accident conditions, and their accumulation will be strictly controlled on safety grounds. Based on the release from postulated accident sequences, 1000 g-T is the assumed in-vessel

Table 1
Design parameters and operating conditions of the PFCs in ITER during the BPP [a]

Component/armour material	Surface area (m^2)	Armour thickness (mm)	Heat flux (MW/m^2)	Armour temperature (front/back) (°C)	Particle flux (10^{20}DT/m^2 s)	Energy (eV)	Fast neutron flux[k] (10^{18} n/m^2 s)
First-wall							
•First-wall: Be	1020 [b]	10	0.25 [c] − 0.5 [d]	285/245 [c]	0.1 − 1	100 − 500	1.9 − 2.3
•Start-up limiter: Be	∼9	4 − 5	∼8 [e]	790/360 [h] 190/170 [i]	10 − 100 [h]	100 −500	2.3
•Upper baffle: Be	80	10	1 [c] −2 [d]	485/295 [d]	0.1 − 1	100 − 500	2
•Lower baffle: W	110	10	1.5 [c] − 3 [d]	640/360 [d]	1 − 10	100	1.1
Divertor							
•Lower target: CFC	75	<20	<10 [f] −20 [g]	1280/340 [f]	<10000	<5	0.4 − 0.6
•Sidewall: W	80	15	2 [c] − 5 [d]	460/240 [c]	1 − 10	5 − 100	0.6 −1.1
•Dome: W	85	10	1 [c] − 5 [d]	315/210 [c]	1 − 10	5 − 100	0.6 − 1.1
•Cassette liner: W	90/400 [b]	Louvre	0.1 − 0.7	1000 − 150	<1000 [j]	<1	0.35 − 0.4

The first-wall and baffle are assumed not to be changed during the BPP, so they will have a lifetime of about 8×10^6 s. The divertor is assumed to be changed at the end of $\sim 3 \times 10^6$ s. The nominal pulse duration is ∼1000 s.

[a] At present ITER operation is envisaged in two phases: the basic performance phase (BPP) and the enhanced performance phase (EPP), each planned to last about 10 yr.

[b] Surface area of the louvers.

[c] Design average heat flux (at 1.5 GW).

[d] Design peak value at any condition.

[e] During start-up; during flat-top burn the limiter will experience the same thermal load as the primary first-wall.

[f] Peak load expected during normal semi-detached conditions.

[g] Peak load expected during off-normal transients (10% frequency, 10 s duration).

[h] During limiter operation.

[i] During divertor operation.

[j] Dominant molecular hydrogen species at very low energy: $\Gamma_{H_2} \gg \Gamma_H$.

[k] $E > 0.1$ MeV.

tritium mobilisable limit. When the accumulated torus inventory reaches this level, operation must be discontinued and dedicated cleaning techniques/procedures applied until the tritium inventory is reduced to acceptable levels. The frequency of cleaning intervention will be simply determined by the rate of codeposition. Independent of safety limits, control of the in-vessel tritium inventory is also necessary to avoid exhausting the available tritium supply. However, here we assume that the most stringent requirements will come from the safety considerations.

3. Tritium retention database with direct relevance to the design of ITER

In the case of Be and W, physical sputtering is expected to be the controlling erosion mechanism during normal operation, while for carbon, under certain plasma conditions, radiation-enhanced sublimation (RES) and chemical erosion could dominate [4–6]. Here, we provide a synopsis of the database of the chemical erosion of carbon for which some unresolved critical issues still remain.

3.1. Chemical erosion

At present, the mechanisms associated with chemical erosion of carbon due to hydrogen impact are not fully understood. Recent modelling advances [7–10], however, have provided new insights into the complex physical/chemical interactions. Work on this front is continuing. While the chemical erosion database generated by controlled laboratory experiments is extensive for incident H energies above ~50 eV [4], it is only recently that the range has been extended down to detached divertor-relevant energies, $\leqslant 10$ eV [11,12]. A major constraint of controlled ion beam experiments is the low available flux at low energies. Alternatively, laboratory linear plasma devices, capable of delivering high fluxes, are constrained by other difficulties, such as reliable plasma characterisation and the effect of redeposition of eroded carbon on the derived erosion yields. The complexity of fusion devices makes it even more difficult to measure erosion yields in tokamaks. Indirect methods are used to determine the flux densities and energies of the impacting and eroded particles. The most advanced method is emission spectroscopy in the plasma edge as discussed in Ref. [6].

In brief, the current chemical erosion database at divertor-relevant fluxes is insufficient to draw conclusions, with confidence, on the possible nature of the flux dependence of the chemical erosion yield of carbon. The scatter in the available data, spanning the flux range $10^{18} - 10^{23}$ ions/m^2 s, is too large to discern a significant flux dependence [5,6]. Recent studies, dedicated to investigating the flux range between 10^{21} and 10^{23} ions/m^{2}s, show the possibility of a reduced yield with increasing flux [13–15]. However, due to the potential importance of a flux dependence effect for ITER, it is necessary to expand the available experimental database at high fluxes (i.e., $>3 \times 10^{22}$ D/m^2 s) in linear plasma simulator devices and tokamaks.

Notwithstanding the above limitations, a careful assessment of the available low-energy chemical erosion database was undertaken in an attempt to provide divertor relevant yields for erosion/codeposition calculations being performed by Brooks [16] for ITER. As most of the controlled laboratory yield data were obtained at fluxes $<10^{20}$ ions/m^2 s, a key requirement is to project the low-flux results to divertor fluxes. In addition, the incident D energy needs to be extrapolated to ~5 eV, which is below the lowest energy (10 eV), for which measured yields are available. Also, H and D data need to be projected to tritium. Table 2 comprises the 'mean values' of projections arrived at by two different procedures. For the most part, the tabled values are within a factor of two of the two sets of projections. The two procedures are briefly described: (i) The 'Garching' procedure is based on an erosion model [9], which includes terms for energy and mass dependence through a sputtering-like function added to the thermal model of Horn et al. [7]. A flux-dependence correction term was also added to get agreement with experimentally measured yields. (ii) The second approach, the 'Toronto' procedure, is based on a set of chemical erosion measurements at low energies (down to 10 eV) [12] and an erosion model [10], incorporating kinetic and thermal processes in a self-consistent way. Thus, projections down to 5 eV and to tritium were obtained using the model. Projections to high fluxes (~10^{23} ions/m^2 s) were made on the basis of an experimentally observed trend that T_m (temperature where maximum chemical erosion is observed) shifts to larger values with increasing flux density [17,18]

Other chemical erosion issues relevant for ITER include the effect of dopants (e.g., Si) on the erosion yield of doped graphite, and the erosion behaviour of rede-

Table 2
Total chemical erosion yields (C atoms/ion) based on the 'mean values' resulting from the Garching and Toronto procedures 'Engineering' tests of T retention and permeation in PFC duplex structures

D^+	E(eV)	500 K	600 K	700 K	800 K
	5	0.008	0.010	0.012	0.007
	10	0.012	0.016	0.018	0.014
	15	0.014	0.019	0.022	0.018
T^+					
	5	0.009	0.014	0.014	0.009
	10	0.015	0.019	0.021	0.016
	15	0.017	0.022	0.025	0.020

posited carbon surfaces. Based on the extensive experimental database on the chemical erosion of doped graphites (e.g., [19,20]), yield reductions are observed for relatively high (>100 eV) impact energies and temperatures >430°C, i.e. at the maximum of the chemical sputtering yield. At energies <100 eV and temperatures <430°C, the yield reductions are minor and are comparable to the reduction observed in the C concentration at the surface [19].

As the projected high 'gross chemical erosion yield' for the ITER divertor is balanced by local redeposition [16], the question of stability and enhanced erosion of the promptly redeposited material arises. Vietzke et al. [4] observed for thermal atomic H impact on various forms of hydrogenated materials significantly enhanced rates for the hydrogenated materials in comparison to pyrolytic graphite. This is in agreement with erosion experiments using low-pressure plasmas showing that, in particular at low ion energies (10–20 eV), the erosion rates depend critically on the microscopic film structure [21].

3.2. Tritium codeposition experience

3.2.1. Tokamaks

D retention has been measured in many tokamaks including studies of trace tritium produced by D-D nuclear reactions (see Ref. [1] and references therein). However, the most extensive data on D and T codeposition has come from TFTR and JET. In this review we concentrate on the experience of these tokamaks. Tritium issues in tokamaks were also the subject of a recent IEA workshop [22]

The Tokamak Fusion Test Reactor (TFTR) was the first fusion facility with extensive experience with tritium fuelling and removal. The fraction of tritium retained in the vessel was found to vary with discharge type, clean-up history and the period studied ([23] and references therein). The modelling of tritium retention in TFTR is reported in Ref. [24]. Overall, during the three periods of DT plasma operations (excluding periods of active tritium removal), over half of the tritium fuel injected was retained in the torus. Active tritium removal by glow discharge cleaning (GDC) and air ventilation and other techniques [23] were successful in removing substantial amounts of tritium in periods between plasma operations. The resulting in-vessel inventory was less than 1 g. The tritium fraction retained was similar to the deuterium retention fraction estimated earlier from in-vessel surveys and analysis of components, including wall coupons and a standard set of limiter tiles removed after each major run campaign [25] as well as from the fuel balance analysis [26]. These studies showed that (i) the D retention fraction compared to the injected amount averaged over the 5 year period was 44 ± 17%, and (ii) the dominant mechanism for retention was codeposition of C with D. Of the retained D, 44% was on the plasma-facing surface of the limiter, 15% was in the gaps between the limiter tiles, and 41% was on the remainder of the stainless steel vacuum vessel wall.

An extended phase of D–T operations (referred to as DTE1) was also carried out in the Joint European Torus (JET) from June to November 1997 as part of the Mk-II-A divertor campaign [27,28]. Of the 35 g of tritium that entered the torus, 34.4 g was via gas fuelling and 0.6 g via neutral beams [27,28]. The T was introduced in a mixture with D, with the T content varying from 1% to 100%. Immediately after the last T-fuelled discharge, the amount of T remaining in the vessel was ~11.5 g. A series of discharges was performed with D or H fuelling, resulting in a reduction of the fraction of tritium in the plasma, similar to that following the JET Preliminary Tritium Experiment (PTE1) in 1991 [29], before the pumped divertor was installed. However, the T remaining in the vessel was only reduced by about 50%, to ~5.5 g, whereas after PTE1 it was reduced by 87%.

During the Mk-II campaign, but prior to the DTE1 experiment, heavy deposits were discovered on water-cooled copper louvres through the gap at the inner corner of the divertor (see Fig. 2). These deposits were tending to flake off, probably on exposure to air. They represent an extra, and larger, contribution to the retained D inventory than in previous JET operational campaigns [30]. Therefore, it is presently assumed that much of the tritium remaining in the vessel after DTE1 is trapped in such deposits. Since the regions of the deposits cannot be contacted by the plasma, they are unaffected by plasma discharge clean-up operations. However, anomalously high T outgassing rates were found upon venting in comparison to that after PTE1. Efforts are underway to model the codeposition patterns observed in JET and satisfactory agreement between predicted and observed features is necessary before extrapolation to ITER could be made with confidence. Further important information is expected to be gained from ongoing post-mortem analyses of tiles and samples of tokamak dust and flakes from the heavy deposition at the inner corner of the divertor, and future operation of JET with different divertor configurations.

It is worth noting at this point that the retention fractions in terms of throughput experienced in existing machines cannot be simply applied to ITER, because codeposition is mainly related to the net-erosion rates as a function of spatial position on the wall and divertor surfaces together with the impurity transport in the SOL. In general, the fraction of tritium retained in a tokamak is a complex function of the machine and plasma geometry, material coverage and discharge history. TFTR, for example, operated at low wall temperatures which enhances the amount of deuterium stored in the codeposits and decreases the efficiency of clean-up removal techniques. In addition, low density plasma scenarios with high edge temperatures, as used routinely

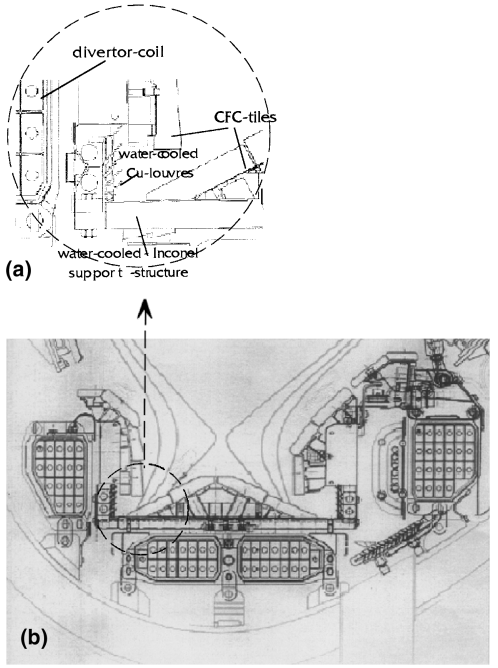

Fig. 2. (a) In-vessel configuration of the divertor region (Mk-II-A) for the first JET Divertor Tritium Experiment (DTE1). The divertor floor and side walls are covered with CFC tiles (each $\approx 200 \times 400$ mm), with gaps about 50 mm wide at the inner and outer corners which allow pumping through the divertor by the cryopump mounted below. The main chamber walls are protected from confined plasma by graphite poloidal limiters, which under normal conditions are only contacted by the plasma during the start-up and ramp-down phases. The plasma is run in the divertor configuration for more than half of the discharge period. Conditioning of the vessel is maintained by occasional evaporations of Be from four sources near the outer midplane. (b) Detail of the water-cooled louvres behind the slot at the inner-corner of the divertor where heavy deposits and flakes have been observed.

in TFTR, result in large carbon impurity release rates which in turn results in thicker carbon deposits. In JET, erosion/redeposition and tritium uptake in the codeposited layers was found to be dependent on the divertor geometry. Also, results from present-day machines (with several second pulse length) may be affected disproportionately by transient effects such as impurity generation and transport during the start-up and shutdown periods. First-wall erosion/redeposition, for instance, can be markedly different during start-up and shutdown than during the flat-top period, where an equilibrium divertor plasma is established. In contrast, the 1000 s ITER pulse length makes the start-up/shutdown effects much less significant. In addition, the ITER edge density is some ten times higher than that in JET or TFTR, the walls are hotter than TFTR, and the amount of oxygen contamination could be lower. Also, there are still uncertainties concerning the effect that thermal transients, like disruptions or ELMs, might have on production and flaking of films and on their hydrogenic species content. This is being investigated in tokamaks.

3.2.2. Laboratory studies

In addition to the information available from tokamaks, the codeposition of hydrogenic species with ITER candidate PFMs is also being extensively studied in laboratory experiments, where individual processes can be investigated and understood in isolation and where conditions are better controlled and diagnosed than in tokamaks.

While there is a very large database and many publications on the subject of codeposition of carbon with tritium, the data on the codeposition properties of beryllium are much more limited. New results on the codeposition or coimplantation of beryllium and deuterium are now available from Causey et al. [31] and Mayer et al. [32]. The results of these investigations indicate that H-isotope retention in re-depositing Be is very dependent on the relative magnitudes of the hydrogen and beryllium fluxes and the fluxes of oxygen and carbon that are incident on the collecting surface, as well as on the energy of the incident hydrogen. If the oxygen and carbon impurity fluxes are comparable to the hydrogen and beryllium fluxes, the re-depositing Be layer will most likely grow as a BeO layer with incorporated carbon. If the hydrogen is energetic enough to be implanted in the BeO layer and causes displacement damage, it will be retained at the damage sites. If the oxygen and carbon impurity fluxes are significantly less than the hydrogen and beryllium fluxes, the re-depositing layer will probably be relatively pure Be with low oxygen and carbon impurities. In this case, hydrogen retention will be low and dependent on the formation of near-surface bubbles and blisters. Temperatures above 500°C are needed to remove retained hydrogen from codeposited BeO/C layers formed with energetic (hundreds of eV) hydrogen [32]. Retention of hydrogen in redeposited Be with low oxygen and carbon impurities approaches zero for temperatures above 300°C [31].

In conclusion, the estimation of tritium retention in re-deposited Be for ITER is highly dependent on the knowledge of the oxygen and impurity fluxes incident on the collecting surfaces and this is accounted for in the calculations.

3.3. Implantation experience

New important implantation data are now available from laboratory experiments for Be and for W, for conditions representative of ITER where previous data were either missing or largely scattered. The majority of studies conducted to date have sought to understand the processes of implantation-driven diffusion, trapping and release at the surface.

Beryllium: Retention data for Be have been, at least in part, presented elsewhere [1,33]. A key finding observed in high ion-flux experiments is the saturation of the beryllium surface such that inventories of implanted particles become almost insensitive to continued implantation fluence. This phenomenon has a strong impact on the design of ITER because it implies a small inventory in the beryllium components as discussed in Section 5. D(T) retention in beryllium was found to vary strongly with temperature and is reduced by about one order of magnitude as the temperature varies from 27°C to 700°C. The maximum D(T) retention is about 1×10^{21} D(T)/m^2 or less for ITER-relevant plasma-exposure conditions. For further experimental details see review [33] and references therein.

Studies have also been initiated to investigate hydrogen isotope retention in beryllium whose surface is contaminated with carbon and oxygen. This condition may result in ITER because of the erosion-redeposition phenomena that are anticipated to generate complex mixed-material layers whose behaviour in terms of tritium retention and removal is not well known. While mixed-material layers are expected to play a major role only in the ITER divertor, during disruptions there might be a transient outflux of C from the divertor that could lead to deposition onto first-wall surfaces. Most of the present effort is dedicated to the investigation of conditions necessary for the formation of mixed-material layer involving carbon, beryllium and tungsten [34,35]. The available results, although preliminary, indicate that C-containing mixed-materials layers are expected to form on beryllium for conditions representative of the ITER baffle and first-wall [33]. In this case, the total D retention tends to increase, but is not significantly greater than the D uptake and retention in Be without an impurity layer [33]. For temperatures >400°C, Be diffuses into C overlayers, and at temperatures between 500–600°C, Be_2C is formed. Hydrogen retention is found to be very low in Be_2C at these temperatures [36].

The effect of beryllium deposition onto carbon also needs to be examined. Beryllium incident in the bombarding plasma may act to reduce the chemical erosion of a graphite plate, in much the same manner as the addition of dopants to graphite have been shown to reduce chemical erosion under some conditions. This could potentially reduce the erosion of the CFC divertor plates in ITER.

Tungsten: ITER-relevant experiments on hydrogen isotope retention in various forms of tungsten (with energies below the displacement threshold for hydrogen isotopes in tungsten) have been performed at several laboratories during the last few years ([37–39]). Preliminary interpretation of the data, aided by modelling, shows that hydrogen release from tungsten is not recombination limited. In addition, it appears that intense implantation, such as that expected in the divertor region of ITER (and duplicated in plasma simulators such as PISCES-B and the Tritium Plasma Experiment (TPE), results in enhanced reemission from the implant zone. This phenomenon results in very low hydrogen retention (i.e., less than 10^{21} T/m^2 [38]).

Although modelling has shown the release of hydrogen from tungsten to be rapid, based on these data, there is still a build-up of trapped atoms at moderate temperatures (130–530°C). NRA profiling of these deuterium plasma exposed samples [38] shows the near-surface D concentration as ~0.1 at.%. Similarly, measurements by Haasz et al. [39] show possible D trapping occurring in the bulk of tungsten at a level of ~0.1 at. %. Saturation of a 10 mm thick tungsten layer to this level with a 50/50 mixture of deuterium and tritium would result in less than 0.5 g-T/m^2. The contribution to trapping at damage sites created by neutrons will also need to be considered. However, due to the residual radiation in tungsten after exposure to neutrons, it is not expected that experiments will be available on the tritium retention in n-irradiated tungsten.

Microscopic pits of sizes ranging from 0.1 to 5 μm were observed in 1% lanthanum oxide-doped W samples exposed to extended deuterium plasma bombardment in PISCES. These pits may result from the formation, and subsequent rupturing, of deuterium bubbles formed at defect sites and grain boundaries below the sample surface. The erosion rate associated with this phenomenon, although enhanced, is still small (yield ~10^{-4}) at ITER temperatures [37]. The damaged surface structure appears to be beneficial by allowing enhanced reemission of deuterium from the sample.

Measurements are also underway to investigate C-containing layer formation on tungsten and details can be found in Ref. [37]. In general, it is observed that C films grow on W surfaces at much higher C impurity concentration in the plasma than for Be and similar to that expected in ITER and when a C-containing mixed-material layer forms, the retention increases, primarily due to the high retention capacity in the carbon film.

4. Modelling of D–T retention

Here we describe the results of several different modelling efforts aimed at providing understanding of the mechanisms expected to govern the tritium uptake in ITER. Emphasis of modelling is on implantation in beryllium and tungsten and codeposition in carbon and beryllium.

4.1. Modelling of implantation in beryllium and tungsten

One process not built into most of the models which is required to study H-transport and retention in mate-

rials such as beryllium is saturation. This became evident in experiments with high particle fluences and high fluxes, particularly for beryllium, but also more recently for tungsten at ambient temperatures. After reaching fluences of about 10^{22} atom/m^2, it has been observed that further implantation results in a negligibly increased inventory [33]. Details of mechanisms responsible for this effect are only partially understood [40], but two factors seem to be important – low capacity for hydrogen in the material and a high implantation flux which generate surface-connected porosity that provides a rapid return path back to the plasma. Evidence of such damage has been reported by several authors (see Ref. [33] and references therein).

The TMAP4 code [41] has been used with saturation and erosion effects to successfully model a number of experiments discussed in Ref. [40]. The model gave good results in each of the cases mentioned. Comparing the inventory results for the conventional *recombination-limited* model with calculations implementing *saturation* and erosion, as well as tritium breeding from the neutrons, the recombination model estimates for ITER an inventory of ~800 g at the end of the BPP as compared with ~105 g when saturation and erosion were considered.

A further consideration, at least for beryllium in tokamak applications, is the production of helium and tritium by neutronic transmutations in the beryllium itself. These processes are incorporated into the FISPACT code [42] which was used by Forty et al. [43]. Based on results of measurements of tritium release from n-irradiated beryllium conducted by Baldwin and Billone [44], it is assumed that this tritium is not released at the temperature reached during normal operation and accident conditions in ITER and is distributed throughout the bulk material, retained primarily in microscopic defect sites and voids containing helium

The TMAP4 code has also been used to simulate experiments performed on pure tungsten and on tungsten doped with about 1% La_2O_3 that were exposed to deuterium and tritium plasmas [37,38] and to ion beam implantation-driven permeation experiments [39,45]. In each case, there was a break point in temperature above which classical recombination-limited re-emission of implanted ions from the surface seemed to apply using literature values for trapping and transport parameters [45] and below which classical concepts proved inadequate to model the measured retention. Analyses [46] showed that these retention profiles, together with deuterium concentration profiles measured by NRA [47], could be reproduced by simply postulating enhanced transport of implanted atoms from the implantation depth back to the surface, a process that appears to be characteristic of saturation. Further work remains to establish the mechanism(s) for this enhanced transport.

4.2. Erosion/codeposition modelling

A combination of kinetic impurity transport codes, sheath code, and charge–exchange code was used, all coupled to plasma fluid type solutions for the divertor scrape-off layer to assess the wall and divertor component lifetimes due to sputtering, the tritium codeposition rates in redeposited material, and the likely effect on core plasma contamination. Briefly, the central technique is to use a combination of the WBC [48] Monte Carlo code and the REDEP [49] deterministic code. To compute divertor or first-wall erosion, these codes are coupled, with WBC supplying very detailed space-localised impurity parameters (e.g. redeposited charge state distribution) to REDEP which is then used to generate a self-consistent solution (erosion, redeposition, self-sputtering fluxes, etc.) for the entire surface.

Both codes incorporate numerous models and subcodes for sputtering and reflection coefficients, atomic and molecular processes, plasma-impurity collisions, surface temperature and temperature-dependent hydrogen isotope trapping ratios, and sheath parameters. Critical models for the plasma regime and design analysed here include carbon chemical sputtering coefficients at low energies (~5–15 eV), rate coefficients for hydrocarbon molecular impact processes with plasma ions and electrons, hydrocarbon sticking coefficients, and the effects of mixed materials, namely beryllium and carbon, on sputtering and tritium trapping. Some details of these critical models are given in Ref. [16].

Clearly, the reliability of the code output for erosion/codeposition can be no better than the input plasma solution. For this work the plasma parameters are taken from the B2 EIRENE code solutions of Kukushkin et al. [50] with low-variance charge–exchange fluxes additionally supplied by a DEGAS$^+$ code solution [16,51].

In a recent code validation effort on DIII-D [52] using the DiMES probe and other measured near-surface plasma data as input to the WBC and REDEP erosion codes, the code predictions were found to match erosion data well, for the overall carbon divertor and for several metal test spots. This was done for an *attached* plasma regime ($T_e \sim 70$ eV) and provides some confidence that given a plasma solution, the erosion/codeposition estimates would be reasonably valid for the attached portion of the divertor targets, as well as the first-wall. Most tritium codeposition for the ITER reference design is currently predicted to arise from chemical sputtering and subsequent transport in the *detached* portion of the plasma along the divertor plates. As calculations for this regime are not yet validated, and in any event, involve numerous model uncertainties, one needs to await considerable additional data and validation work to claim to have a predictive solution. However, the results shown here should be nevertheless reasonably valid to illustrate trends. Code validation

work with carbon surfaces exposed to detached plasmas has begun for several devices, namely DIII-D, JET, and PISCES. In general, however, this effort is substantially too small in comparison to the experimental data available and to the different machines and geometries used.

5. Analysis and estimates of tritium inventory

5.1. Implantation analysis (including breeding for beryllium)

The retention and permeation of tritium in beryllium- and tungsten-clad PFCs of ITER have been studied using the TMAP4 code [41], including saturation and erosion, for the set of nominal design parameters and operation conditions summarised in Table 1. By far the largest contributor of tritium in the first-wall will be tritium bred from neutron transmutations in the beryllium. Table 3 shows the estimated inventory and integrated permeation through the various PFCs.

5.2. Erosion/codeposition analysis

Using the WBC/REDEP/DEGAS$^+$ analysis described above and in Ref. [16], erosion and redeposition of the inner and outer carbon-coated ITER divertor target plates were analysed. The analysis computes physical sputtering by plasma D–T ions and atoms, helium, carbon self-sputtering, and chemical sputtering by D–T ions, atoms, and non-thermal molecules. Chemical yields used for the critical detached plasma region are given in Table 2. Transport and redeposition of carbon back to the plates themselves, and to the plenum region is computed. Tritium codeposition in net carbon growth areas is computed using the redeposited carbon fluxes and surface-temperature-dependent H/C trapping ratios.

Ref. [16] describes detailed results of these calculations. Regarding tritium codeposition, the critical results are: (1) carbon chemical sputtering in the detached region along each plate is the main source of net carbon erosion and tritium codeposition, (2) physical sputtering of the entire plate contributes about 20% to the codeposition, (3) chemical sputtering in the attached regions has essentially no contribution to net erosion (due to ~100% redeposition), (4) ~75% of growth/codeposition occurs on the bottom of the divertor and/or dump plates, and ~25% occurs in the private-flux region.

Various plasma parameter sensitivity studies were performed for the codeposition calculations, as discussed further in Ref. [16]. Roughly speaking, the analysis gives a likely range of codeposition rates of 10–20 g-T/1000 s pulse. As explained previously, this analysis is subject to uncertainties due to the lack of relevant data and code validation for the detached plasma zone.

In addition to the divertor analysis, the tritium codeposition in wall-sputtered redeposited beryllium has been computed. Using worst-case (coldest) surface temperatures for the critical beryllium deposition zones of ~230°C, we estimate approximate worst-case codeposition rates as ~0.1 g-T/1000s using low-oxygen-content TPE trapping data [31] or ~0.6 g-T/1000s using 'carbon-corrected' Mayer data [32]. In either case the wall-derived T/Be codeposition is much less than codeposition from carbon sputtering in the divertor. The main reason for this is simply the low overall wall beryllium erosion, predicted for the semi-detached regime, with its low overall charge-exchange flux to the walls.

Table 3
Summary of TMAP4 calculated tritium inventory and permeation in ITER plasma-facing components for the basic performance phase

Component	Tritium inventory (g)	Tritium permeation rate (g/day) [a]
First-wall (impl.+breeding)	106 [b]	7.0×10^{-4}
First-wall (incl. n-effect)	194 [c]	3.4×10^{-11}
Startup limiter	9.2×10^{-3}	4.7×10^{-7}
Upper baffle	0.78	4.4×10^{-5}
Lower baffle	0.12	6.5×10^{-7}
Upper vertical target	31	8.1×10^{-9}
Dome	2.0	1.5×10^{-6}
Total [d]	140 [b]/228 [c]	7.5×10^{-4}

[a] It is assumed that ~10 000 pulses are distributed uniformely over 10 yr.

[b] Breeding is assumed on the basis of 14-MeV neutron current of 1.15 MW/m^2, a peak value, as compared with 0.94 MW/m^2 poloidal average, and production rates calculated by Forty et al. [43] from FISPACT for the first wall; trapping characteristics were those of unirradiated beryllium (0.0005 atom fraction, 0.8 eV trap energy).

[c] In addition to breeding, this calculation included neutron-induced traps taken to be 2% atom fraction and 1.4 eV trap energy. The values give excellent replication of thermal desorption data of tritium from fully dense beryllium irradiated to ITER-like conditions in ATR reported by Baldwin and Billone [44].

[d] Inventory was in first 10 μm of beryllium. Inventory in the bulk of carbon was estimated using the DIFFUSE code and was found to be small ($\leqslant 1$ g/m^2).

While the amount of tritium retained is high, the predicted tritium retention fraction in ITER, based on fuel throughput, is lower than that observed in present-day tokamaks (TFTR, JET – see for example Refs. [23,27,28]). Retention models need to be benchmarked against the experience of existing machines to identify the contribution of various effects. Specific recommendations will be made in Section 7 on how to reduce these uncertainties in order to improve the confidence of predictions for ITER.

6. Tritium inventory control and removal from the codeposited layers

6.1. Clean-up needs and likely methods in ITER

The prospect of ITER reaching its in-vessel tritium inventory limit in the order of one hundred pulses highlights the critical need to develop and test in situ efficient tritium recovery and cleaning techniques.

Thus, the mechanisms of tritium removal from the codeposited layers are as important to the operation of ITER as the mechanisms of retention. If in situ codeposit removal techniques are rapid and effective, with minimum impact on machine availability, and plasma performance is recovered promptly after the cleaning, then the limitations associated with tritium retention become less important for the long-range inventory problem.

6.2. Status and perspectives of techniques for tritium removal from codeposits

It is worth noting that tritium removal from existing tokamaks proved to be successful to achieve the required limits; however, it took place over several weeks as compared to the total time of about 1000 s of high power DT discharges [23,28]. The tritium removal rate in ITER will need to be orders of magnitude higher when considering its operational schedule.

There are in principle two mechanisms by which H-isotopes can be removed from materials: thermal desorption and ion-induced desorption. However, past studies have shown that thermal desorption of tritium from codeposited layers requires in-vacuum temperatures above 700°C [53] and T removal by ion bombardment is limited to depths corresponding to the ion range, typically a few nm for plasma discharges. Thus, removal of T from μm thick codeposits requires removal of the codeposits themselves.

Tritium can be removed from thick codeposits by oxidising the C layers to produce volatile DTO, CO, and CO_2, leading to the release of hydrogen.

In Ref. [1] possible methods of tritium removal are summarised with emphasis on those that do not require vessel opening and remote handling access which would have an adverse impact on machine availability. A significant constraint for tritium removal techniques for ITER is the presence of the toroidal field. The superconducting coils generating the field are rated for 1000 on/off cycles in their lifetime, so cleaning techniques performed, e.g. weekly, will need to be done with the toroidal field on.

6.2.1. Codeposit removal with high temperature oxidation: laboratory and tokamak studies

Several laboratory measurements of H/D/T release and removal rates from codeposited films and D-implanted graphite during exposure to air or oxygen are now available [54–61]. Here we focus on two key findings: (i) the release of D in an oxygen-containing atmosphere occurs mainly in conjunction with C erosion; (ii) the D-removal and C-erosion rates strongly depend on the film structure. Annealing of C:D films in air or oxygen causes surface oxidation in contrast to vacuum annealing which leads to thermal desorption of D_2 and hydrocarbons at temperatures between 600°C and 730°C. The D-release rate during oxidation is a critical function of the annealing temperature and the film structure. Hard films (H/C $\sim$ 0.4) require temperatures of at least 380°C [60] while soft films (H/C $>$ 1) start to decompose already at $\approx$250°C [61].

From an analysis of the reaction products formed by exposing an a-C:D film to $^{18}O_2$ at 200°C, Haasz et al. [57] concluded that essentially all of the D is removed via D_2O formation, and C is removed by the formation of $C^{18}O$ and $C^{18}O_2$. No D_2 and no methane was observed.

D removal from hydrogenated films was also studied under exposure to other atmospheric gases. Nitrogen exposure and heating in vacuum at or below 300°C was found to have no effect on the release of deuterium from D-implanted layers [56]. On the other hand, exposure of D-implanted layers [56], as well as TFTR codeposits [58], to moist air did result in D removal, but with no evidence of C erosion. It is suggested that D is removed via isotope exchange between the impacting H_2O and the trapped D in the film. Comparing the effectiveness of oxygen and water, oxygen was seen to be considerably more effective in removing the trapped D [56,58].

Based on the results discussed above, tritium can be removed from thick codeposits of carbon and tritium by heating the deposits in air or oxygen. However, it is evident that the D-removal and C-erosion rates depend on the film structure, and differ greatly for laboratory-produced films and codeposits produced in tokamaks. C-erosion rates from hard a-C:D films and D-implanted layers are very similar and are of the order of a few to 10s of nm per hour at temperatures $<$350°C [54–61]. In contrast, measured C-erosion rates for tokamak codeposits (see for TFTR Refs. [55,58,59], JET [58], DIII-D [58], ASDEX-U [60,61]) are 2–3 orders of magnitude

higher under similar conditions. Available results from tokamak codeposits indicate that C-erosion rates in oxygen or air range from ~1–10 μm/h at 250–350°C temperature [58–61]. Further studies are needed to extrapolate these results to ITER (see Section 7).

Three critical issues need further consideration in the application of the oxidation technique for T-removal to ITER. (1) post-oxidation plasma performance recovery, where only limited tokamak experience is available [62]; (2) the effect of oxygen on hot surfaces (e.g., beryllium, tungsten, copper, joints, etc.) of other components in the vacuum vessel; and (3) implications for the pumping and tritium plant when large quantities of DTO are present in the torus exhaust.

First experiments in a tokamak using oxygen with hot walls ($\leqslant$350°C) to remove codeposits have been started in TEXTOR; preliminary results are presented in [62]. In the initial tests, the external pumps were closed and the vessel was filled with $^{16}O_2$ or with $^{18}O_2$ isotopes to pressures ranging between 0.005 Torr up to about ~0.2 Torr. At a filling pressure of <0.01 Torr, most of the oxygen was absorbed on the wall and the remaining 10–20% oxidised the deposits to form CO and CO_2 which were then released. At higher filling pressures (~0.2 Torr), the fraction of oxygen adsorbed decreased by 20–30%, whereas the fraction of CO formation was about constant and the formation of CO_2 increased. Encouragingly, TEXTOR did not experience any long term adverse consequences after the use of oxygen to remove deuterium, and high performance plasma operation could be recovered after several minutes of GDC in helium and deuterium. Further experiments are planned (see Ref. [62]).

6.2.2. Codeposit removal with GDC, ECR and ICR plasma discharges

Glow discharge cleaning: In the absence of magnetic fields, He–O glow discharges can produce rapid, controlled codeposit removal with minimal O-contamination. Erosion rates of 0.4 μm/h have been observed for graphite erosion and about 3 times greater for TFTR codeposits [63]. Laboratory studies [63] have determined this erosion to be a two-step process: oxidation, followed by particle-induced desorption, with the maximum erosion rate limited by the latter. Temperatures in excess of 250°C are required for rapid thermal desorption of CO from carbon surfaces [63]. Helium ions of a few hundred electron volts have a high CO desorption yield, typically much higher than for electrons or photons. Rapid, efficient evacuation of the desorbed impurities from the system is also important. Dissociation and redeposition are primarily caused by energetic electrons; thus, electron densities and energies should be kept as low as possible. This conclusion is supported by conditioning studies with Taylor discharges and ECR. In both cases desorbed impurities were efficiently evacuated only after the discharge was terminated.

The erosion rate for He–O GDC was found to decrease with time due to surface texturing. This texturing has been observed for several materials and appears to be a characteristic of bombardment at a single well-defined angle, in this case, normal incidence. Accordingly, it is argued that this texturing will be eliminated if impact is random, as produced by discharges generating neutral particles. It is concluded that energetic neutral atom bombardment from a low pressure, low temperature plasma should produce the maximal CO desorption with minimal redeposition or texturing.

Ion cyclotron resonance (ICR) discharge cleaning (DC): In a strong magnetic field, modelling shows that large fluxes of energetic neutral He can be produced by direct He ICR. This is due to helium's relatively small cross-section for scattering compared to charge exchange. Wall fluxes of neutral He, greater than the ion fluxes with He GDC, can be produced by an RF geometry which directly heats the He ions. In addition, at low pressures, energy transfer from He to the electrons is slow and should result in minimal electron heating. For the recently reported ICRF experiments in TEXTOR [64], an RF geometry and frequency were used, which energised He ions through the directly energised electrons. This resulted in a small net H desorption rate from an H-loaded surface. It appears likely that rapid ionisation of desorbed species by the hot electrons prevented their evacuation. Similar conclusions can be reached for the recent TORE-Supra studies [65]. Clearly, electrons should be heated only to the level necessary to sustain the plasma.

Laboratory experiments are in progress at Sardia National Laboratories (SNL) to assess the potential of using ICR methods to remove codeposits in ITER.

Electron cyclotron resonance (ECR) discharge cleaning (DC): The carbon removal rates observed in laboratory hydrogen ECR, for unbiased samples at about 80°C, are about 20 nm/h [66–68]. Amorphous carbon film thicknesses were monitored in situ during removal, either by monitoring the electrical resistance of the film or by ellipsometry, and ex situ by ion beam analysis. The erosion was found to depend critically on the actual film structure at the surface. While the microscopic processes active during interaction of carbonaceous surfaces with hydrogen discharges are relatively well understood, the situation is less satisfying for the interaction with other reactive gases, in particular with oxygen. Therefore, an experimental study was started at IPP-Garching to investigate the erosion of a-C:H films in low-pressure discharges using a variety of gases with particular emphasis on oxygen [68]. Among the investigated gas compositions, oxygen shows always the highest erosion rates; the relative rates behave as O_2:D_2:H_2 $\approx$ 10:2:1. The erosion rates

increase with substrate temperature, ion energy, and ion flux. These correlations are true for soft and hard C:H films, although the absolute rates differ significantly. Erosion rates of 1.7 and 3.6 μm/h were measured for hard a-C:H films and oxygen plasmas with surfaces maintained at floating potential at 25°C and 350°C, respectively. The rates for soft C:H films are roughly a factor of two higher [68]. For discharges in pure oxygen, no large oxygen inventories were found in ex situ analyses of a-C:H samples. The dominant erosion products as measured by mass spectrometry are H_2, CO, CO_2, and H_2O. Etching by neutral species (most probably atomic oxygen) was demonstrated to occur, but the underlying processes remain unclear [68].

Electron cyclotron discharge cleaning is being tested as a method for removal of a-C:D films in the Alcator C-Mod tokamak [69] and removal rates in excess of 3 nm/h were measured using a deuterium ECDC plasma.

Application of this technique to mitigate tritium accumulation in ITER depends on several unresolved questions. They include: (a) how much CH is removed by ions and neutral species; (b) dependence of the removal rate on the absorbed RF power density, plasma temperature, plasma density, gas pressure, gas throughput, vessel geometry, magnetic field configuration, angle of field onto surface, etc.; (c) whether ECDC needs to be localised in a tokamak (e.g. in the divertor) by localising the RF power absorption close to the surface to be cleaned; (d) whether ECDC/ICDC in a tokamak can remove CH from the vessel, or just redistribute it from one region to another, i.e. how much CH is removed by ECDC and how much is redeposited elsewhere (via chemically reactive radicals) before being pumped out; (e) active conditioning requirements to remove residual O; (f) detrimental effects of an O plasma in tokamaks.

6.2.3. Codeposit removal with other methods

Alternative 'oxygen-free' techniques have been proposed [1] but need further development. They include CO_2 pellet blast cleaning [70], and in situ rapid heating of codeposited layers in vacuum or in an inert gas using an impinging laser beam surface heating [71]. Although CO_2 pellet blast cleaning is widely used in the semiconductor and nuclear industries [1], the effectiveness of this method in removing codeposited tritium layers from carbon tiles is not yet known. Preliminary tests using existing equipment at SNL [70] did remove the surface layer from a graphite tile from the DIII-D tokamak but also severely eroded the tile. Better control of the removal rate by using softer pellets is being explored. Experimental tests are also needed to evaluate the in situ applicability and efficiency of rapid heating of codeposited layers using an impinging laser beam or other methods of surface heating [71].

7. Further R&D needs

An urgent co-ordinated effort, involving extensive participation by all parts of the fusion community, is needed to undertake the proposed (see Table 4) R&D activities related to tritium retention and removal.

In addition, further development and optimisation of the divertor design, based on information provided by future dedicated operation experience in existing tokamaks will be useful to control and to mitigate codeposition. This can be achieved either by: (1) enhancing the formation of T-rich films in specific areas (e.g., by means of 'cold catchers' which could be periodically heated to recover the T), or (2) by appropriate design, whereby regions of probable deposition are kept 'hot' during operation, leading to reduced T retention. Prudently, new design solutions without using carbon should also be explored. The primary candidate for high heat flux regions is tungsten. However, efforts to reduce transients and mitigate disruptions in existing tokamaks must continue with high priority.

8. Summary

The design, operation and safety of ITER is affected by tritium retained in the redeposited films of eroded armour materials and implanted directly in the surfaces of components surrounding the plasma. These issues are being investigated both in laboratory experiments designed to probe the fundamental retention processes, and in tokamaks, as well as by modelling.

Some of the most recent tokamak and laboratory data on hydrogen, deuterium and tritium retention in beryllium, tungsten and carbon materials which are candidates for cladding various surfaces of the ITER PFCs have been reviewed. The issue of mixed-materials which is still poorly understood and requires continuing investigations has also been addressed. The data presented and discussed here have been gathered by the Home Teams as part of the R&D program supporting the engineering design phase of ITER.

Codeposition of tritium with eroded carbon that originates from limited regions of the divertor plate near the separatrix strike points is expected to be the dominant T-retention mechanism in ITER. The in-vessel tritium inventory in the codeposited layers is predicted to accumulate at a rate of 10–20 g-T/1000 s pulse. This rate strongly depends on the assumptions made for the plasma edge conditions, erosion yields, and other factors. Tritium accumulating in the codeposited layers in ITER has safety implications as it can be mobilised and released to the environment during a severe accident. Therefore, a safety limit on the accumulation of tritium will be imposed. This limit could be reached, under worst conditions, in a few days or weeks of operation.

Table 4
Tritium retention and removal R&D priorities

(A) Modelling of tokamak data	(B) T removal from codeposits	(C) Laboratory experiments	(D) Tokamak experiments	(E) In-vessel PMI diagnostics
•Using tokamak data provide a physical explanation of: (1) the erosion/redeposition patterns and rates; (2) the amount of $D(T)$ retained in codeposits; (3) the mechanisms and parameters controlling the ormation of films/flakes and the retention of D(T) for divertor configurations tested in present-day tokamaks	•Develop methods for T removal (both with and without oxygen) from plasma-facing surfaces with minimum impact on machine availability •Although tests in tokamaks are needed, laboratory experiments are necessary to quantify and understand underlying mechanisms.	•Experiments under well controlled and diagnosed conditions simulating the tokamak edge, supported by modelling, are needed to investigate: (1) chemical erosion of C-based materials in plasma simulator and tokamak experiments to determine erosion yields and their dependence on plasma parameters for ranges of conditions where data are missing or scattered (i.e., high fluxes, low temperature, high fluences);	•Obtain information on the relative importance of wall vs. divertor erosion sources by surveying and interpreting existing tokamak results and conduct new experiments •Erosion of the main chamber wall by C-X neutrals •Conduct tokamak global gas balance measurements for a series of well defined discharge conditions to investigate the effect of plasma and operation parameters on $D(T)$ retention	•Improved real-time diagnostics of the plasma wall interaction coupled with better integration of wall and plasma edge modelling codes [b] •Real-time measurements of erosion rates by installing, for example, erosion and film thickness, diagnostics, and a means of measuring H-retention under different operational conditions are needed

•Improve quantitative understanding of retention and develop models that can be validated by comparing the predicted retention with that experienced in existing machines •It is worth noting that the complex and varied discharge history in tokamaks makes often archaeological studies of limited utility to challenge the models (see E)	•Continue studies of tokamak codeposited films including analysis of film stoichiometry, microstructure and impurity concentrations •Tokamak studies of recovery of plasma performance subsequent to cleaning and assessment of collateral damage which may result from oxygen exposure •Investigate the effect of oxygen on the release of D(T) from mixed Be/C layers	(2) stability and enhanced erosion of promptly redeposited materials; (3) mixed-materials effects; (4) T retained in n-irrad. Be; (5) 'Engineering' tests of *T* retention and permeation in PFC duplex structures	•*D*(*T*) removal experiments (see B) •Tests are required in tokamaks which would closely mirror the material situation proposed for ITER. Such experiments would help address synergistic effects not easily determined in laboratory simulation experiments and one-material lined tokamaks [a]	•Instrumentation and access similar to that available for the DiMES probe in DIII-D are very useful to further progress in understanding the complex environment of the tokamak edge •New methods to measure erosion/redeposition, T accumulation, dust on plasma facing surfaces of ITER need to be further developed and tested in today's tokamaks

[a] These include: (i) the nature, magnitude and location of T codeposition and dust to be expected in ITER; (ii) the origin of carbon impurities in the plasma and the importance of chemical sputtering; (iii) the development and testing of engineering solutions to mitigate codeposition and/or ease the removal of codeposited films; (iv) testing and implementation of T clean-up procedures, if required, and subsequent conditioning methods; (v) the effect on hydrogen (D/T) retention in the wall; (vi) mixed-materials effects; (vii) operational aspects of using beryllium on the first-wall.

[b] The recycling of neutral atoms, radicals and molecules, including impurities between the surface and the plasma edge needs closer scrutiny in tokamaks and laboratory experiments. This is true especially for high density, low temperature detached plasma regimes where neutrals play an important role and conventional plasma diagnostics like Langmuir probes and spectroscopy do not work very well.

At this point, plasma operation in ITER could only be resumed after tritium recovery methods will have removed the majority of tritium.

Efficient techniques of tritium removal from the codeposited layers will be essential in the operation of ITER and more generically of any other next-step fusion device using carbon and fuelled with tritium. These techniques are beyond the experience of current tokamaks. Some of the principal methods that are being investigated and tested have been critically discussed in this paper; the required R&D has also been identified. Based on the encouraging results available from tests in laboratory experiments, and anticipating further positive developments in this area, it is conceivable that high-temperature baking of the divertor system in an oxygen atmosphere in-conjunction with low-pressure plasma discharges with oxygen (e.g., ECR, ICR) could represent viable methods to efficiently control the build-up of tritium codeposition inventory in the ITER vessel. Further developments and optimisation of the divertor design, based on information provided by future dedicated operation experience in existing tokamaks, will also be very important to control the occurrence, and mitigate the extent, of the codeposition problem.

Recommendations are made for further urgent R&D aimed at narrowing the remaining uncertainties associated with modelling work and experiments in both laboratory plasma simulators and tokamaks. In particular, there is a need to conduct experiments in existing tokamaks whose edge environment and PFC configuration closely mirror the material wall situation proposed for ITER. Such experiments can offer opportunities for complex and non-linear plasma surface interactions beyond those accessible in laboratory investigations. Tests of this type have usually received low priority in present tokamaks, mainly due to the competition for machine time and resources with more conventional plasma physics experiments. There is, in particular, a need for dedicated operation time, diagnostics/instrumentation, and in-vessel reconfigurations in tokamaks dedicated to addressing specific T issues of high relevance for ITER. To be able to successfully compare the results to predictions, it will be necessary to improve the modelling and do bench-marking tests, and improve communication/collaboration among plasma experimentalists, modellers and materials specialists.

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