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# Tritium Retention in Next Step Devices and the Requirements for Mitigation and Removal Techniques

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
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## **ABSTRACT**

Mechanisms underlying the retention of fuel species in tokamaks with carbon plasma facing components are presented, together with estimates for the corresponding retention of tritium in ITER. The consequential requirement for new and improved schemes to reduce the tritium inventory is highlighted and the results of on-going studies into a range of techniques are presented, together with estimates of the tritium removal rate in ITER in each case. Finally, an approach involving the integration of many tritium removal techniques into the ITER operational schedule is proposed as a means to extend the period of operations before major intervention is required.

## **1. INTRODUCTION**

Carbon Fibre Composite (CFC) remains the current material of choice for high heat flux regions in the ITER divertor. Experience from existing fusion devices operating with carbon-based plasma facing components suggests that this choice could lead to significant retention of tritium in the ITER vessel during operations with a deuterium/tritium mix [1][2]. Since ITER will be required to operate with strict safety limits for the maximum allowable inventory of tritium in the vessel, the mechanisms underlying this retention are of great interest. One such mechanism is predicted to be the co-deposition of tritium and eroded carbon in the form of tritiated hydrocarbon films.

In JET, co-deposited films have been demonstrated to be responsible for the bulk of long term tritium retention [3][4]. Co-deposition has therefore become the focus of efforts to minimise tritium retention and to provide efficient methods and technologies for removing tritium from the ITER vessel, with minimum impact on operations. Section 2 of this paper briefly discusses the current understanding of how and where co-deposition takes place in tokamaks, presents estimates for the resulting retention of tritium in the ITER vessel and draws attention to the consequences for ITER operations. Section 3 highlights the potential impact of fine scale structure in plasma facing components, such as tile gaps; castellations and CFC micro-structure, which could significantly increase tritium retention in ITER. Section 4 presents the status of efforts to develop and study techniques which might be applicable in ITER for removing tritium from co-deposited films. Lastly, section 5 demonstrates how a combination of these techniques might be deployed on ITER to slow the progress during operations towards the in-vessel tritium inventory limit.

## **2. CO-DEPOSITION AND THE CONSEQUENCES FOR ITER**

### ***2.1 CARBON MIGRATION AND CO-DEPOSIT FORMATION***

Understanding of the mechanisms underlying co-deposition has advanced significantly over the last few years. This advance has resulted partly from improved diagnosis of the distribution of carbon erosion and deposition in tokamaks and partly from the development of models that handle the erosion of carbon and the transport hydrocarbon species in the plasma-surface interaction region. In turn, these models have benefited from detailed laboratory and tokamak measurements of chemical erosion yields and surface sticking coefficients for hydrocarbon radicals.

Whilst the details of co-deposition in any particular device will be strongly dependent on the specific plasma conditions, plasma facing component structure and materials present, it is nevertheless now possible to highlight generic characteristics: Chemical erosion of carbon surfaces by hydrogenic ions and atoms results in the release of volatile hydrocarbon species such as methane. Where a sufficiently dense and hot local plasma exists, these molecules are quickly dissociated creating hydrocarbon radicals and ions. These fragments return to the surface but are displaced from the site of the initial erosion by plasma forces (e.g. ion gyromotion) and cross-field transport (especially for neutrals). They arrive together with carbon ions created by complete dissociation of hydrocarbon molecules released from erosion sites further away. The carbon ions stick to the surface with high probability temporarily forming regions of re-deposition whilst the hydrocarbon fragments, which have a much lower sticking probability, are mostly ejected back into the plasma. The re-deposited carbon, however, has a far higher coefficient of chemical erosion than the bulk carbon and is quickly re-eroded. There is thus a gradual migration of carbon and hydrocarbon species away from sites of erosion. Eventually they arrive at areas protected from the flux of hydrogenic ions and atoms from the plasma, where the rate of re-deposition exceeds that of re-erosion. In these areas, the carbon and hydrocarbon species build up to form co-deposited, amorphous hydrocarbon layers. Such a process of migration is unique to carbon as it depends on the nature of chemical erosion and hydrocarbon chemistry.

The extent of co-deposition is strongly dependent on the flux of carbon ions into the neighbouring regions. In JET, the outer divertor is normally a region of net carbon erosion and the inner divertor is where co-deposition occurs. Until recently, a quantitative understanding of how such long range carbon transport occurs was not available. This was partly because net erosion and deposition could only be determined by surface analysis techniques following a campaign that could include many different plasma conditions and equilibrium configurations. Recent experiments, however, have helped to resolve this issue [5][6].  $C^{13}$  methane was injected into the outer divertor of a well defined JET plasma. Subsequent analysis of a complete poloidal set of divertor tiles was able to distinguish between the injected  $C^{13}$  and intrinsic  $C^{12}$  and thus determine, quantitatively, where it had been re-deposited. These plasmas were then modelled using the EDGE2D plasma fluid code and the NIMBUS Monte-Carlo code to follow the injected carbon. Reasonable agreement is obtained for the percentage of  $C^{13}$  redeposited at the inner divertor target -  $\sim 7\%$  split between tiles in the SOL and private flux region - and the modelling demonstrates the importance of flows in the main chamber Scrape-Off Layer (SOL) and E $\times$ B drifts across the private flux region in transporting the carbon from the outer to inner divertor.

## ***2.2 EROSION AND RE-DEPOSITION IN ITER***

Improvements in the understanding of carbon sources and long range transport have been complemented by advances in codes that model carbon migration, through chemical erosion and hydrocarbon transport, into the areas where co-deposits grow. One such code, ERO has been used with considerable success to model carbon migration in TEXTOR during localised  $C^{13}$  methane injection experiments. It has

now been employed to make a preliminary study of carbon migration local to the strike points in ITER and to scope the likely range of tritium retention in co-deposited layers [7]

ITER has a quite different plasma-facing material mix to JET or TEXTOR, which use predominantly graphite and CFC armour. In particular, the first wall and start-up limiters in ITER will be protected by beryllium tiles with a total surface area of  $700\text{m}^2$ , much greater than the  $50\text{m}^2$  of CFC in the high heat flux regions. It has been suggested that transport of beryllium eroded from these tiles along the SOL to the divertor may suppress carbon erosion and thus reduce co-deposition. The ERO studies are a first attempt to model the impact of beryllium fluxes on co-deposition in ITER. A background plasma for ITER was taken from modelling using the coupled B2 (fluid) and EIRENE (neutral, Monte Carlo) code. The most up to date estimates available for the dependence of chemical and physical erosion yields on plasma parameters were used in ERO. Key parameters for the effective sticking coefficient of hydrocarbon fragments ( $S_{\text{eff}} = 0$ ) and the chemical erosion yield of re-deposited carbon (a factor 10 higher than for bulk carbon) were taken from TEXTOR modelling of  $\text{C}^{13}$  methane injection experiments. An example of the net particle profile across the inner and outer divertor targets of ITER, from ERO modelling, is shown in figure 1.

Regions of deposition and erosion are seen on both targets, determined by a balance between beryllium coverage of the target (which reduces the area of carbon exposed to the plasma) and enhanced physical sputtering of carbon by beryllium ions. The flux of beryllium ions into the divertor therefore play a key role and a range from 0.1% to 1% of the background hydrogenic ion flux was explored.

### **2.3 TRITIUM RETENTION IN ITER AND THE IMPACT ON OPERATION**

Tritium retention was estimated from the ERO results [7] by assuming a tritium fraction in the regions of carbon deposition varying from  $\text{T/C} = 5\%$  in the high heat flux regions to  $\text{T/C} = 50\%$  in more remote areas. These values are consistent with laboratory analysis of hydrogenic retention in tokamak co-deposited films. In addition, the impact on retention of assuming that tritium is also trapped in redeposited beryllium was explored, assuming  $\text{T/Be} = 5\%$ . With these assumptions a range of tritium retention per 400s D-T (50%.50%) ITER shot is estimated as  $f_T \sim 0.5\text{g} - 2\text{g}$  for carbon co-deposition only and  $f_T \sim 2.6\text{g} - 6.4\text{g}$  if beryllium trapping is also included. These results are presented in figure 2, which shows the tritium inventory with ITER shot number for a range of  $f_T$ . Also shown is the inventory limit imposed on ITER operations, which currently stands at 350g.  $f_T$  would need to be as low as 0.14g for a reasonable number of ITER shots, say 2,500, before the safety limit is reached necessitating a shutdown phase for maintenance. It is clear from figure 2 that strategies for tritium removal in ITER are essential if CFC targets are used in the D-T phase. These strategies must ensure a removal efficiency of up to 98%.

### **3. THE EFFECT OF LOCAL STRUCTURE ON TRITIUM RETENTION**

Tritium removal strategies employed on ITER must be capable of dealing with trapping of tritium in all the areas it can occur. Until recently attention was focused on the large area, shadowed regions of

the divertor, where co-deposition was known to be a significant effect. However, it has now become clear that co-deposition can occur in locally shadowed regions, such as those between plasma facing components ('tile gaps') and in the narrow slots separating castellated elements on tiles ('castellation gaps'). There will be over 2 million such gaps on ITER, largely due to the use of castellated tiles to reduce thermal stresses, and the plasma exposed area will be increased by up to a factor 5 in some regions by the presence of gaps.

Dedicated experiments have been conducted on a number of tokamaks to determine the likely extent of co-deposition in gaps. For example, TZM mono-blocks arranged in a castellated structure with 0.5mm gaps, 10mm deep were exposed to the TEXTOR SOL for 200s [8]. A profile into the gaps of deuterium retained in co-deposits was determined by nuclear reaction analysis. The deuterium areal density on the poloidal facing gaps falls away from the surface with scale lengths of between 0.6 and 2mm, depending on the deuterium ion flux from the plasma to the entrance of the gap. The total deuterium retention as a percentage of incident flux is found to be in the range 0.4% – 4% for high and low flux conditions, respectively. Extrapolating to ITER, using a deuterium neutral and ion flux provided by B2-EIRENE modelling, yields an estimated tritium retention in gap co-deposits of 0.5 – 5g per 400s shot, that is of similar order to the retention in large area, shadowed regions discussed in Section 2. A note of caution is sounded, however, by analysis of castellated tiles originating from the JET MkI divertor, which suggests that both the period of exposure and whether the source of carbon is local or remote from the gap may be important in determining the level of co-deposition. Taking these effects into account would act to lower substantially the estimated retention in ITER. In addition, the JET analysis reveals a deuterium areal density in the tile gaps (6mm wide) which is 33 times higher than that in the castellation gaps (0.5mm), suggesting that gap width plays a key role.

Analysis of deuterium retention in Tore Supra, suggests that micro-structure at the surface of CFC tiles may also increase the level of tritium retention [9]. In long pulse (>100s), high fluence discharges on Tore Supra, deuterium retention does not appear to be consistent with the estimated level of carbon erosion and thus simple co-deposition processes. Retention in mm-scale shadowed regions near the convoluted surface of CFC, determined by the fibre arrangement and voids in the carbon filler between fibres, is proposed as one explanation. Laboratory experiments with the grade of CFC used in Tore Supra (NB11) and that proposed for ITER (NB31) indicate that deuterium retention increases as the square root of the deuterium ion fluence to the surface and with a scale length of up to ~10mm (dependent on both fluence and surface temperature). Extrapolating this analysis to the Tore Supra environment confirms that this mechanism of retention may initially dominate conventional co-deposition.

#### **4. STRATEGIES FOR TRITIUM REMOVAL IN ITER**

Techniques presently employed to re-condition tokamaks following shut-down phases and, in the case of JET, to de-tritiate the vessel before opening include running deuterium-only tokamak plasmas; baking in vacuum; glow discharges in helium; isotopic exchange (flushing with deuterium) and out-



gassing (flushing with air). The efficiency of tritium removal using any of these methods alone, however, is well below the 98% required to prolong the operation of ITER for an acceptable period (see section 2) and, with the exception of deuterium-only shots, the removal rate is typically at least a factor 10 too slow to provide a practical solution for ITER. Deuterium only shots in JET following the DTE1 tritium experimental campaign did prove reasonably effective [10], recovering around 45% of the retained tritium over ~1000 shots at a rate equivalent to ~2g of tritium per hour for 150m<sup>2</sup> of co-deposit (gT/h/150m<sup>2</sup>). The area unit for this normalisation will be used throughout this section as it represents an estimate of the main area of co-deposition in ITER (i.e. the high heat flux zones, divertor baffles and dome) and thus allows a ready comparison of retention (est. 0.5g – 6.4g/400s shot, see section 2) and removal rates.

Integrating several of the existing techniques as an essential part of ITER operations and paying attention to optimising all discharges for minimising retention (e.g. by consideration of the fuelling strategy) will certainly help extend the period before the tritium inventory limit is reached but is unlikely to provide a complete solution (i.e. one which ensures operation between the planned maintenance periods). Three main approaches are currently being explored, which can be broadly described as oxidation, scavenging (or alternative chemistry) and ablation.

#### **4.1 OXIDATION**

Oxidation aims to release trapped tritium primarily by eroding co-deposited films through the reaction  $aC:D/T + O/O_2 \rightarrow CO_x + DTO:D_2O:T_2O$ , allowing the volatile products to be pumped from the vessel. Laboratory and tokamak (TEXTOR, Asdex Upgrade) experiments [11][12][13] to investigate oxidation show that plasma generated O/O<sup>+</sup> fluxes are around 7 orders of magnitude more efficient per particle at removing co-deposited films than O<sub>2</sub> fluxes, even for surfaces at elevated temperatures. Typically O/O<sup>+</sup> fluxes are 50 - 100% efficient (i.e. 1 carbon atom released for every 1 – 2 incident oxygen atoms or ions).

Glow discharge (GDC), electron (ECR) and ion (ICR) cyclotron low gas pressure (1 – 10 mbar) plasmas have all been used to generate O/O<sup>+</sup> fluxes, typically forming the plasma in a mixture of helium and oxygen. Laboratory experiments suggest that absolute C-atom removal rates,  $v_E$  of co-deposited films do not increase significantly for O<sub>2</sub>/He ratios above 10% in these kinds of plasma [13] and, indeed, reach 30 - 40% of their peak level at O<sub>2</sub>/He ratios as low as 2%.  $nE$  increases with the surface temperature of the co-deposit, suggesting that chemical reactions do indeed take place, but also increases with the negative bias voltage applied to the substrate, suggesting that collisions play a role. The picture is of a two stage process where helium ions (predominantly) damage bonds at the surface of the co-deposit, which accelerates chemical erosion due to O/O<sup>+</sup> fluxes.  $v_E$  is up to a factor 10 higher for pure, laboratory produced films compared to co-deposited films produced in the tokamak environment. This is thought to be a consequence of impurities (e.g. nickel, iron etc.) in the tokamak films building up at the surface as the hydrocarbon is eroded. This hypothesis is supported by a marked decrease in  $nE$  with O/O<sup>+</sup> fluence (i.e. time of exposure) for the tokamak film. In ITER,

tungsten and beryllium could both act as impurities in the co-deposited films and their impact (and that of their carbides) remains to be fully understood.

Typical values of  $nE$  for tokamak co-deposited films, when exposed to plasma generated O/O+ fluxes in glow discharge and ECR plasmas, are equivalent to a range of tritium removal rates in ITER from 0.075 - 0.3gT/h/150m<sup>2</sup>. There is some evidence that this could be up to a factor 5 higher in ICR plasmas, which are typically higher density and thus have a higher ionisation fraction than glow discharges or ECR plasmas.

During experiments in Asdex Upgrade in which low pressure GDC O<sub>2</sub>/He plasmas were maintained for up to 49 hours, sample coupons coated in laboratory produced films were used to evaluate the variation of  $v_E$  around the vessel. As might be expected,  $v_E$  falls off rapidly in regions shadowed from the plasma, such as deep in the divertor or behind limiters. Similar results were obtained in the TEXTOR experiments. Laboratory studies indicate, however, that reasonable rates may still be possible with co-deposits formed in castellated gaps [14][15], at least for gaps with a sufficiently large width/depth ratio.

$v_E$  in low pressure O<sub>2</sub>/He plasmas is around an order of magnitude higher than that obtained by heating co-deposited surfaces to 620K in an oxygen environment at 300 mbar (O<sub>2</sub> baking) [11][12]. However O<sub>2</sub> fluxes, although much less efficient than O/O<sup>+</sup>, are not limited by the low gas pressures necessary for coupling power into GDC, ECR and ICR plasmas.  $nE$  increases strongly with the O<sup>2</sup> pressure and values equivalent to a tritium removal rate in ITER of 1.6gT/h/150m<sup>2</sup> were obtained during baking of co-deposited TEXTOR tiles at 573K in atmospheric pressure air. It is unlikely, however that such high pressure baking will be possible in ITER (in part due to operational constraints on the active gas handling system). In addition, the maximum bake-out temperature in ITER is currently 513K and  $nE$  falls strongly with temperature (dropping by an order of magnitude between 623K and 523K). It is not yet clear whether a satisfactory window in temperature and pressure exists for oxidation through O<sub>2</sub> baking. Also, although most of the deuterium was removed from the co-deposit in these experiments the film itself remained, becoming flaky and peeling off the tile. It is hypothesised that this is again the result of impurities in the co-deposit, inhibiting O<sub>2</sub> penetration and release of volatiles. High removal rates (0.5mm/h) have been demonstrated, however, at much lower wall temperature (458K) in laboratory experiments with a few % mix of ozone and oxygen [16] but collateral damage to neighbouring carbon surfaces appears to be unacceptably high [17].

A key aspect of all the oxidation techniques being explored is the time taken to recover normal plasma operations after co-deposit removal. Around 10% of oxygen injected during low pressure GDC O<sub>2</sub>/He plasmas in TEXTOR was retained in the vessel [12] after the experiment. Laboratory studies indicate that this is most probably in the form of metal oxides (e.g. on unprotected areas of the vacuum vessel and metallic plasma facing components). Hydrogen ECR discharges in the laboratory were found very effective at removing oxide deposits on tungsten [13]. Glow discharge cleaning in H<sub>2</sub> or He for 60–70 hours after O<sub>2</sub>/He GDC plasmas, followed by boronisation, was sufficient to reduce oxygen impurity levels in both TEXTOR and Asdex Upgrade to normal levels in subsequent

tokamak plasmas [12][13]. Whether this extrapolates to beryllium oxide, as would be formed in ITER, is an open question.

#### **4.2 SCAVENGING/ALTERNATIVE CHEMISTRY**

Oxygen is not, of course, the only element that reacts with hydrocarbons. Some gases, including N<sub>2</sub>; NH<sub>3</sub>; CNH and SH<sub>2</sub>, are known to suppress net deposition of hydrocarbon films in laboratory plasmas (such as those used for diamond-like carbon coatings, e.g. [18]). The origin of this suppression is unclear: gas phase, surface chemistry and ion impact all may be involved. It was proposed that this effect could be used to reduce co-deposit formation without forming oxides.

Laboratory experiments in H<sub>2</sub> glow discharge plasmas with a 5% CH<sub>4</sub> and N<sub>2</sub> mix showed a factor 4 reduction in co-deposit formation for N<sub>2</sub>/CH<sub>4</sub> ~ 0.5 [19], an effect not seen when Ar was substituted for N<sub>2</sub>. Similar results, in fact a factor 5 reduction, were obtained by injecting N<sub>2</sub> into the sub-divertor plasma of Asdex Upgrade [20], an experiment which also demonstrated no significant contamination of the core plasma with N<sub>2</sub> injection, no change in carbon impurity levels and recovery of normal operation in the first shot without N<sub>2</sub>. ‘Scavenging’, in which reactive radical pre-cursors to hydrocarbon deposition are ‘mopped up’ by nitrogen ions and atoms in the plasma to form unreactive molecules, was proposed as an explanation of these results. However, recent laboratory experiments [15][21] suggest an alternative (or possibly complimentary) explanation. The C-atom removal rate,  $v_E$  for a laboratory produced co-deposited film in a 1μbar N<sub>2</sub>/He (20%:80%) GDC plasma is found to be comparable to that in O<sub>2</sub>/He plasmas. In low pressure N<sub>2</sub>/H<sub>2</sub> ECR plasmas,  $nE$  can actually exceed that in comparable oxygen plasmas at high negative bias voltages on the substrate. A strong peak in  $nE$  is seen for a 25%:75% N<sub>2</sub>/H<sub>2</sub> mix, suggesting a synergistic interaction of H and N at the surface of the co-deposit. These results are consistent with nitrogen-enhanced chemical erosion, rather than scavenging, being the mechanism behind the observed reduction in net co-deposition rate in the laboratory and Asdex Upgrade plasmas.

The observation of high co-deposit erosion rates in nitrogen seeded, hydrogen plasmas is itself very promising. It may eventually provide a route to tritium removal without oxide formation and thus offer a more rapid recovery of normal operating conditions.  $nE$  has not yet been measured, however, for tokamak co-deposited films and the impact of impurities has yet to be evaluated.

#### **4.3 ABLATION**

Several technological approaches to in situ co-deposit removal are being developed for use in tokamaks. These rely on delivering a high heat flux of photons, neutrals or ions to the contaminated surface. Depending on the nature and magnitude of this flux, it can thermally desorb tritium from the co-deposit, raise its temperature to the sublimation point or dissociate the co-deposit into gaseous by-products. All of the technologies being considered are derived from systems used in industry for low impact cleaning of surfaces (for example in the aviation and nuclear sectors) and would be most appropriate for use during planned maintenance periods on ITER (although more frequent use is not

totally precluded but would require a vacuum and magnetic field compatible remote handling system). The most developed technology so far for the tokamak environment is photonic cleaning by flash-lamp [22]. The flash-lamps used in these studies are capable of releasing 500J in 140ms over a spectral band-width of ~200nm to 1mm, a peak power output of 3.6MW. Light from the flash-lamp is focused by a semi-elliptical cavity onto the surface being treated, producing a footprint of around 30 cm<sup>2</sup>. Peak power densities up to 375MW/m<sup>2</sup> and energy densities up to 6 J/cm<sup>2</sup> are delivered to the surface. The short pulse length of the flash-lamp limits propagation of the heat pulse into the co-deposit to a few mm at the surface, thereby providing good control of the cleaning process and minimising collateral damage.

In 2004 the engineering feasibility of employing this technique in a large tokamak was successfully demonstrated using the JET remote handling tool to manipulate a powered, water cooled flash-lamp assembly around the JET inner divertor, treating areas on a number of co-deposited tiles [22]. These trials were not designed, however, to yield quantitative data on co-deposit removal rates. Laboratory studies have now been conducted using the flash-lamp to treat ~150mm thick co-deposited films on a CFC tile removed from the inboard, horizontal target of the JET MkII-GB/SRP divertor. The co-deposits are known from earlier analysis to be rich in tritium. Three positions on the tile, with varying co-deposit thickness and tritium content, were treated with up to 2,600 pulses.

A total of ~9mg of tritium was released during the flash-lamp exposure, 40% of all the tritium predicted to be in the treated area [23]. Subsequent analysis of the tile using scanning electron microscopy revealed that between 70 – 90mm of co-deposit had been removed, about half the total co-deposit thickness in the untreated areas. This corresponds to an equivalent, average tritium removal rate in ITER of 0.075gT/h/150m<sup>2</sup> for a flash-lamp pulse repetition rate of 5Hz. In fact the tritium removal rate was initially a factor 2 – 3 higher than the average rate but decreased throughout the treatment period. Secondary Ion Mass Spectrometry (SIMS) of the treated areas indicated a build up of nickel impurities in the residual co-deposit, which may have resulted in this decrease, figure 3. The SIMS analysis also showed that the residual co-deposit had been depleted of deuterium to a depth of ~7mm. Finite element modelling of heat penetration into the co-deposit during the flash-lamp pulse was undertaken, using the thermal and physical properties of diamond-like carbon as a model for the co-deposit. The results suggest that the co-deposit was heated above 700 K to a depth of 7 – 10 mm, consistent with the observed deuterium out-gassing. The modelling also suggested that the surface temperature reached between 1600 K and 2300 K during each 140 ms pulse.

The closely related technique of laser cleaning has also undergone substantial laboratory testing for its application to co-deposit removal. Regions of 1 cm<sup>2</sup> on a TEXTOR tile were treated using a 20W Ytterbium fibre laser (1060nm, 120ns, 20kHz) producing 2J/cm<sup>2</sup> on a  $\phi$ 250 $\mu$ m spot at a distance of 40cm [24]. The spot was scanned over the region being treated using a galvanometric mirror system. 50mm of co-deposit were removed in the treated areas using between 1 and 10, 2s scans. This corresponds to a tritium removal rate in ITER in the range 0.03 – 0.3gT/h/150m<sup>2</sup>. A clear energy density threshold for co-deposit removal was measured at about 0.5 J/cm<sup>2</sup>. This is a factor

5 lower than the measured threshold for ablation of pure graphite, indicating that good selectivity is possible and collateral damage to the tile underlying the co-deposit can be avoided. No difference was observed in the threshold for co-deposit removal in air and in an inert, argon atmosphere.

Finally, there is growing interest regarding the possibility of using mitigated disruptions (already a technique under consideration for ITER as part of their safety systems) as an alternative source of the high heat flux to co-deposited surfaces. There has been extensive experience on devices such as DIII-D using massive injection of noble gases (typically neon or argon, sometimes as a mixture with hydrogen) to mitigate disruptions by stimulating a controlled release of core stored energy [25]. Calculations indicate that in larger devices the thermal load (of radiated photons) on the first wall and divertor during mitigated disruptions would be sufficient to ablate co-deposited films. A massive gas puffing system has been installed on JET during the recent shut-down and experiments to investigate the impact of disruption mitigation on co-deposit build-up are planned for 2006.

## **CONCLUSIONS GOOD HOUSEKEEPING' AND INTEGRATION OF TRITIUM REMOVAL INTO ITER OPERATIONS**

The long term retention of tritium with carbon has been known about for at least 18 years but efforts to diagnose, model and resolve this issue have only recently begun to grow, as the prospect of a tritium burning device comes ever closer. There is still a considerable way to go before models providing reliable estimates for tritium retention with carbon in ITER are available but the extent of the challenge presented is clear. Several tritium removal schemes are now being investigated but all have drawbacks, there are no easy solutions and much more effort is needed to provide ITER with reliable technologies. Even if the D-T phase of ITER does not include CFC targets, schemes for efficient co-deposit removal after the hydrogen and deuterium-only phases will still be required to ensure that the vessel is free of carbon for later operations.

Operating ITER with a CFC divertor during the D-T phase will need a 'deep toolbox' of available techniques for dealing with the issue of tritium retention. All the techniques currently being investigated show promise at some level and need further exploration. There is, however, unlikely ever to be a single solution. Rather 'good housekeeping', in which efforts are made to reduce the tritium inventory on a continual basis, is likely to be the key to successful operation. Table 1 summarises what might be achieved with such a multi-faceted approach, where an appropriate tritium reduction scheme is employed during the shot, inter-shot, daily, weekly, monthly and during annual maintenance periods. A tritium retention rate of 3g per 400s shot is assumed, which is in the middle of the range presented in section 2. Nevertheless the 'good housekeeping' regime is rather rigorous and necessarily assumes an operational schedule of 10 shots per day, 5 days per week, 3 weeks per month and 8 months per year to give the necessary time for tritium removal and avoid meeting the tritium inventory limit in as little as 1 month. With a strategy such as this, however, it appears that there may be a window for operating ITER with CFC targets in the D-T phase for an extended period.

The figures for percentage of tritium removal in table 1 are broadly extrapolated from the estimated tritium removal rates presented in section 4 but with an allowance for up to a 20 - 30% improvement over the coming years as the techniques are optimised. Clearly, therefore, it is necessary to continue at this point with development of as broad a range of tritium removal and mitigation techniques as possible. No door should yet be closed.

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	No Action	‘Good housekeeping’	%T removal /mitigation	Possible technique
<b>During the shot</b>	3g	3g ⇒ 1.8g	40%	N <sub>2</sub> Scavenging Optimisation of fuelling
<b>End of shot &amp;/ or inter-shot</b>	3g	1.8g ⇒ 1.1g	40%	D-only phase (20%) Disruption cleaning D-only discharges D μW-plasma (ECR/ICR)
<b>Overnight (10 hours)</b>	30g	11g ⇒ 9g	20%	D μW-plasma D <sub>2</sub> flush
<b>Weekends (2 days)</b>	150g	45g ⇒ 30g	35%	O <sub>2</sub> /He or N <sub>2</sub> μW-plasma and D-mW plasma recovery
<b>Monthly (9 days)</b>	<b>450g</b>	90g ⇒ 45g	50%	O <sub>2</sub> /He or N <sub>2</sub> μW-plasma O <sub>2</sub> /He GDC (fields off?) and D-μW plasma recovery
<b>Annual (4 months)</b>	<b>3.6kg</b>	350g ⇒ <b>35g</b>	90%	Photonic-cleaning by flash-lamp or laser (entry with remote handling)

*Table 1: An example of how a range of tritium removal schemes might be incorporated, on different time-scales, into ITER operations. The result of taking ‘no action’ is compared to a ‘good housekeeping’ approach in which best efforts are made to reduce the tritium inventory on a continuous basis.*



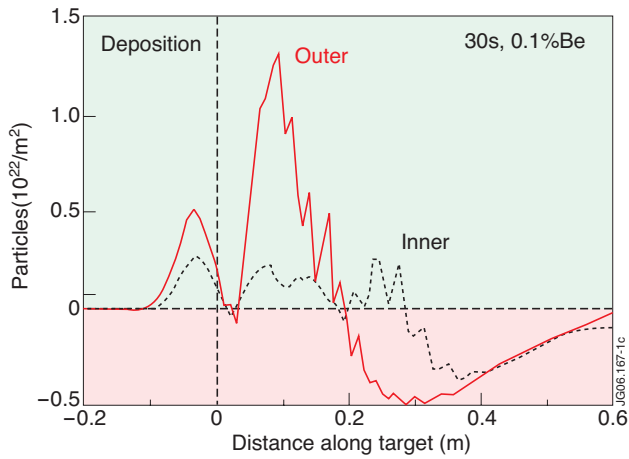


Figure 1: ERO calculations for the change in particle inventory (deposition and erosion) across the ITER inboard and outboard divertor after 30 seconds in a plasma with a 0.1% concentration of Be<sup>2+</sup> ions in the target ion flux.

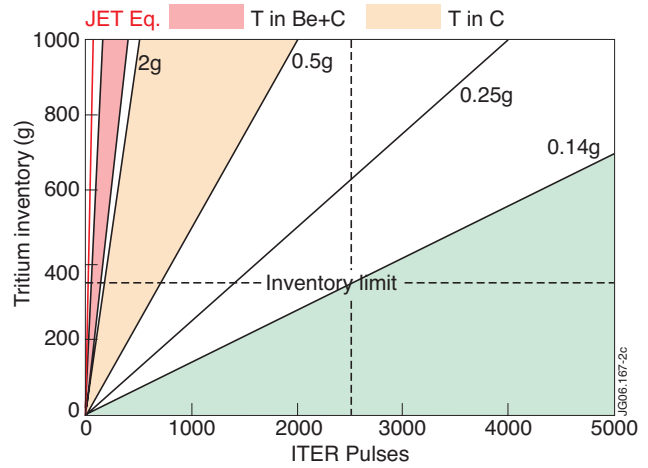


Figure 2: The tritium inventory with increasing number of 400s, 50%:50% D-T shots in ITER, for a range of values of the tritium retention per shot. The amber and red zones show the range of retention estimated by ERO modelling, with and without an assumption of tritium trapping in beryllium dominated deposits. The green zone represent the range of 'acceptable' operation for ITER – i.e. that allowing more than 2,500 shots before a major intervention.

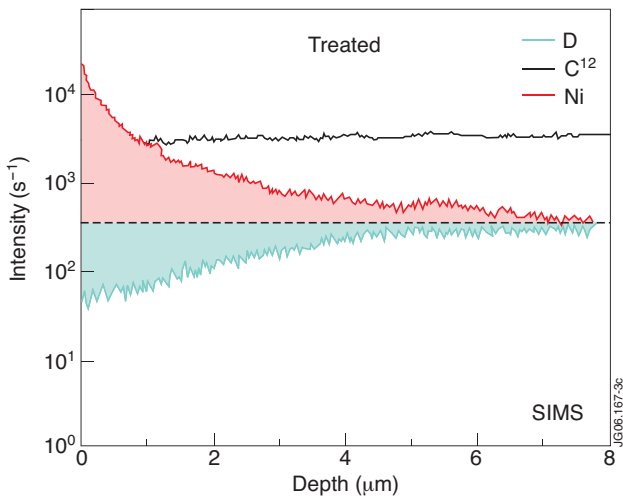


Figure 3: Secondary ion mass spectroscopy in regions of a CFC tile from the inboard, horizontal target of the JET MkII-GB/SRP divertor treated with ~1,300 pulses from a flash-lamp photonic cleaning system. The green region shows depletion of deuterium within ~7mm of the surface – an indication that the co-deposit was heated above 700K to that depth. The red region shows an accumulation of nickel impurities, not removed by the photonic cleaning, at the surface. Impurities in the co-deposited films are likely to affect all the tritium removal techniques investigated to some level.