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Is carbon a realistic choice for ITER's divertor ?

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Abstract

Tritium retention by codeposition with carbon on the divertor target plate is predicted to limit ITER's DT burning plasma operations (e.g. to about 100 pulses for the worst conditions) before the in-vessel tritium inventory limit, currently set at 350 g, is reached. At this point ITER will only be able to continue its burning plasma program if technology is available that is capable of rapidly removing large quantities of tritium from the vessel with over 90% efficiency. The removal rate required is four orders of magnitude faster than that demonstrated in current tokamaks. Eighteen years after the observation of codeposition on JET and TFTR, such technology is nowhere in sight. The inexorable conclusion is that either a major initiative in tritium removal should be funded or that research priorities for ITER should focus on metal alternatives.

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1. Introduction

Serious plasma material interaction issues need to be addressed and resolved in order for ITER to be able to study burning plasmas[1,2]. This article focuses on the choice of material for the most challenging plasma facing region – the divertor. ITER's physics goals with fusion power gain, Q=10, are projected to be met in high confinement regimes that entail a steep pedestal region with a transport barrier that periodically relaxes in an edge localized mode (ELM), sending heat and particle flux from the confined plasma primarily to a narrow region on the divertor target plate. Here the heat flux from ELMs and disruptions could exceed 1 MJ/m^2 and cause material damage[3]. Tungsten is one candidate material, however this would melt under high heat flux and in the worst case, loss of the melt layer could drastically shorten the divertor lifetime. For this reason carbon-fiber-composite (CFC) is the currently favoured material as it does not melt, (though it is subject to brittle destruction[4]). The ITER physics base is founded mostly on tokamaks with carbon plasma facing components (PFCs). However, a major obstacle to the use of carbon in ITER is tritium retention. An excessive tritium inventory in the torus would present a safety hazard in the form of a potential tritium release to the atmosphere in case of a loss of vacuum accident. Due to the ease of mobilisation of tritium trapped in codeposited layers (co-deposited films in tokamaks start to release tritium when exposed to air at temperatures >520 K, [5]), a limit of ~350 g is currently set for the in-vessel codeposition inventory[6]. This limit is set to avoid the need for public evacuation in event of the full accidental release of the tritium inventory under the worst weather conditions. Independent of safety considerations, tritium is expensive and the supply is limited so it is important to avoid inventory build up in inaccessible locations. Though predictions are uncertain, the inventory limit could be reached in as little as 100 pulses under the worst conditions. At this point ITER will only be able to continue its planned burning plasma program if the tritium can be removed from the vessel. An analysis of the tritium removal rate required for ITER to meet its physics mission and a review of removal techniques is presented in [7]. Public sensitivities to tritium issues have previously led to closure of one fission reactor[8] and it is especially surprising that a proven means to remove tritium from ITER is nowhere is sight.

Management of risk is an intrinsic part of any successful large scale project, for example in construction 'megaprojects', in the introduction of new commercial products such as pharmaceutical drugs and in space missions to the distant solar system. Operations research has been applied to optimally managing risks and there is an extensive literature, for example [9, 10, 11] that has applied mathematical modelling to minimise risk. Risk management in a large scale research project poses additional challenges but also opportunities. While in a space mission, the risk, for example, of an electronic component failure can be characterized as the mean time between failure (MTBF) and input to a quantitative model, in fusion research the risks are harder to quantify. At the same time, a worldwide ongoing research and development program is dedicated to better understanding and mitigating risks. However, there are tight constraints on R&D resources. Conscious consideration of mitigating the risk of project failure in decisions on the level and allocation of funding resources is clearly essential. This article discusses the risks in the selection of ITER's divertor material and strategies to minimise the risks.

2. Tritium retention.

In 1987 the operation of the JET[12] and TFTR[13] tokamaks demonstrated that hydrogen isotopes could be codeposited with carbon and the amount retained could increase indefinitely without saturation. In the mid 1990's both TFTR and JET experienced codeposition with tritium fuel. In TFTR, a total of 5 g of tritium was injected into limiter plasmas over a 3.5 year period, mostly by neutral beam injection[14]. The TFTR inner wall limiter provided a large source of eroded carbon and 51% of the injected tritium was co-deposited on the limiter and vessel wall during plasma operations. This was in line with the prior experience with deuterium[15] and consistent with first principles calculations[14,16]. In JET, 35 g of tritium was injected into diverted plasmas over a 6 month campaign, mostly by gas puffing[17]. Tritium behaviour in JET DTE1 experiments was surprisingly different from that in the earlier PTE experiments. The tritium inventory increased a factor-of-two faster than expected peaking at 11.5 g T with more than half of the tritium on site trapped in the vessel. Films were formed with high (~ 0.8) D/C ratio on the divertor louvers leading to subsequent flaking and accumulation in the sub-divertor[18]. The retention rate during the DTE1 campaign was 40%. In both TFTR and JET, tritium retention was initially high following a change from deuterium to tritium gas puffing, due to isotope exchange with deuterium on plasma facing surfaces (dynamic inventory). The contribution of codeposition is lower but cumulative, and is revealed by including periods of D fuelling that reversed the T/D isotope exchange. Tile analysis showed long term retention of deuterium by codeposition in JET at a rate of 16% of the deuterium input[19]. The relatively inaccessible location of tritiated flakes in the sub-divertor hampered efforts at tritium removal. An additional concern was the discovery of tritium trapped in the bulk of CFC tiles[20].

Based on current planning deuterium-tritium operations are scheduled for ITER after a 3-year hydrogen phase, and 1 year of deuterium plasmas. During the full DT phase ITER will be fuelled by approximately 50 g T per 400 s pulse. Of that, ~3 g T per pulse has been predicted to be trapped in the vessel, principally from chemical erosion of carbon from a 20 m^2 area in the divertor followed by codeposition of the eroded carbon with tritium[21]. In the worst case, the ITER in-vessel inventory limit could be approached after a few days of DT operation (Fig. 1). Once the tritium inventory limit is reached, DT plasma operation will be terminated and cannot restart until substantial amounts of tritium are removed from the vessel.

3. Risk Assessment.

3.1 What is the potential impact of the problem?

The cost of unforeseen delays to the ITER operations is estimated to be of order $1million / day. The prospect of ITER operations being stopped to avoid an unsafe condition, with unknown but major costs and delays before it can resume, together with the public sensitivity to tritium issues makes it clear that the potential range of impact includes the risk of project failure and by inference the failure of the fusion energy program.

3.2 How well is the underlying physics understood ?

State of the art modelling codes applied to JET underestimate tritium retention by a factor of x40 possibly because the chemical sputtering yield was higher than expected[21]. On the other hand these codes cannot reproduce detached plasmas in DIII-D where chemical erosion appears to have been suppressed[22] (we should add that the codes have been successfully benchmarked against experiment in attached regimes where physical sputtering dominates). Since ITER depends on detached plasmas to reduce the heat flux to the divertor, tritium retention rate estimated by this code is therefore highly uncertain. Retention could be higher if the ITER outer wall is a carbon deposition area[23] or if there is significant carbon erosion by type 1 ELMS[3]. On the other hand retention could be lower if beryllium impurities impede chemical erosion of carbon near the divertor strike points and hence reduce codeposition[24]. Recently an evaluation of data from ion beams, plasma simulators and fusion devices has revealed a flux dependence of chemical sputtering that might lead to a reduction of tritium inventory by a factor five[25]. While helpful, this would not resolve the problem and retention predictions still need to be validated against experiments in detached plasmas. The likely formation of new mixed materials on plasma facing surfaces adds further uncertainties[1,26,27]. To minimise risk ITER should have the capability for rapidly and efficiently removing the worst credible level of tritium retention (a capability to remove just the most likely, median estimate of retention implies a 50% risk it will be insufficient). For purpose of this paper we take the worst case to be 5 g of tritium retained per ITER pulse.

3.3 What technology is needed to resolve the problem ?

The scale-up in the ITER duty cycle places huge demands on any detritiation technique. To reach the design goal of ≈ 2,000 pulses/year ITER plans to operate with approximately 20 pulses / day, each pulse will be 400 s duration and occur every 33 minutes. A tritium codeposition rate of 5 g T per pulse would necessitate removal of 100 g T every overnight shift to stay below the inventory limit. This tritium removal rate is four orders of magnitude higher than the 1-2 g / month achieved on TFTR and JET.

ITER's high duty cycle also puts unprecedented demands on the efficiency of tritium removal. Residual tritium remaining in ITER after incomplete removal will build up in the vessel and plasma operations will be stopped once this approaches 350 g. Divertor exchange will then be necessary to remove residual tritium (assuming the residual tritium is in fact located on the divertor). In TFTR and JET only half of the tritium inventory was removed by active techniques. In ITER almost complete removal is required (Fig. 2). It can be seen that for a tritium retention rate of 2-5 g/pulse, a removal efficiency above 95% is needed to stay below the inventory limit throughout the divertor lifetime of 3,000 pulses.

Several methods to remove tritium were investigated during the ITER engineering design activity and are reviewed in [1,7]. The methods may be grouped in two classes. One option is to break the a-C:T chemical bond either by heating to high temperatures with a scanning laser[28,29] or radiative plasma termination[30], or by UV or chemical means such as isotope exchange. The tritium is then desorbed as T_2 or DT gas and pumped out. The other option is to remove the whole codeposited layer by oxidation [31] or by ablation with a pulsed laser or flashlamp[32]. Potential techniques should be compatible with the 5 Tesla ambient toroidal field, to avoid delays while the field is cycled off and on. They should also be compatible with the gamma field from activated components. Additional obstacles include the difficulty of accessing codeposition of tritium in grooves and gaps of divertor and limiter structures[33].

While exploratory experiments in laboratories are important, demonstrations in current tokamaks are essential to demonstrate that the released tritium is recoverable and the removal process does not produce reactive radicals that could be reabsorbed before exiting the torus. Debris produced by ablative removal methods will need to be efficiently collected [34]. The restoration of good wall conditions for subsequent high performance plasmas needs to be demonstrated. For oxidative removal schemes the ITER tokamak exhaust processing system will need to be redesigned to cope with the removal exhaust products including up to 700 g of tritiated water (DTO). Recovery of this amount of tritium on a daily basis would be technically challenging, expensive and the technology difficult to license[35,36].

It is crucial to minimise the risk that ITER is a protracted experiment in tritium removal. The peak codeposition rate for the ITER outer divertor is 5 nm/s [37] which would result in a peak codeposited layer thickness growth of 40 μm/day. Since the scale of codeposits in current tokamaks is similar to that in ITER after one day of operations, a convincing demonstration of hydrogen isotope removal technology that is relevant to ITER's needs would be if >95% of the deuterium was removed from a contemporary tokamak in one day with high performance plasmas continuing uninterrupted the following day.

3.4 Are we on a path to develop the technology required ?

We may contrast the status of research on tritium removal to that on ELMs, both key factors in the choice between carbon or tungsten materials for the divertor. In the recent International Conference on Plasma Surface Interactions in Portland, Me., May 24-28th 2004, there were 14 oral talks (including 2 review talks) highlighting advances in ELM physics and control with results from Asdex Upgrade, DIII-D, JET, JT60-U and NSTX[38]. ELM issues are well characterized and modelled and progress is being made with a clear goal of limiting ELM energy density in ITER to 0.5 MJ/m^2 or less. In contrast, there were just two talks on tritium removal. These presented valuable concepts and initial results but it is clear that the level of worldwide effort devoted to tritium removal is far less extensive and mature than that devoted to ELMs.

3.5 What strategies are available to mitigate the risk ?

The present plan is for ITER to use carbon-fiber-composite as divertor material and assess progress in mitigating ELMs and hydrogen isotope retention in the hydrogen and deuterium phases before selecting the divertor material for the DT phase. However in the hydrogen phase this assessment will be compromised by the pre-existing hydrogen in the tiles (from absorbed water). There are still no firm plans for in-vessel deposition diagnostics[39]. After machine activation in the deuterium phase, deposition can only be characterized after a divertor cassette has been removed from the ITER vessel. The time period between definitive information on deposition and the beginning of the DT phase is much shorter than the 3-5 year time to procure a new divertor. Switching the divertor material from carbon to tungsten just before the DT phase would require the development of new plasma scenarios (e.g. no carbon impurity radiation to aid detachment, more high Z core impurity contamination, lower tolerance to disruptions and ELMs etc…) and it is questionable how much of the experience gained in the carbon phase would still be relevant. The long manufacturing lead time (3-5 y) make it imperative to procure a tungsten divertor early in ITER construction, otherwise a switch to tungsten would cause long delays. An additional burden would be the need to develop techniques to remove carbon deposits formed during the hydrogen and deuterium phases to minimise adverse effects from mixed materials formed from residual carbon after the switch to tungsten. The risks of the present strategy appear high.

A clear strategy that would lead to a tokamak demonstration of hydrogen isotope removal at an appropriate rate and efficiency is urgently needed. This would be an integrated plan with the ramifications for the ITER plasma operations, wall conditions and exhaust processing fully worked out. If such a step is not in place when the ITER site is selected, lacks funding, or is deemed unlikely to succeed, then it is clear that carbon should be abandoned as a divertor material for DT plasmas and resources should focus on metals. In any case carbon will be unacceptable in a fusion power reactor.

Eighteen years after tritium retention by codeposition was identified as a serious problem by JET and TFTR, and after repeated calls for action on this issue, a development path leading to tokamak demonstrations of ITER-scale tritium removal is nowhere is sight. It is important to recognize and understand the root cause for this situation. Traditionally ITER R&D is undertaken by the ITER parties on a voluntary basis and major progress has been made in many areas. However there are some technically challenging areas for ITER that fall outside the traditional concerns of contemporary tokamaks (dust diagnosis and removal is another such area). With technical solutions uncertain and cost and schedule overuns likely there is currently little incentive for the ITER parties to take 'ownership' of these thorny issues. As ITER moves toward a construction phase, resources are being diverted from research that could mitigate risk to fabrication of major items of equipment. ITER is an integration experiment of fusion physics and technology, but is fragmented between the six parties, with the risk that the responsibility for overall project success will be overshadowed by the contractual responsibility for each party to deliver its assigned hardware. To address this we recommend that clear 'ownership' of areas with high risk and high consequence for ITER be assigned to a central body that then solicits, peer reviews and funds research proposals that offer the best chance of resolving the issues. An alignment of responsibility and authority together with funding resources contributed by the parties and the time-proven method of competitive proposal solicitations is essential to resolve these long standing but presently 'orphan' issues.

As implied in its name 'International Thermonuclear <u>Experimental</u> Reactor', ITER will necessarily be an experiment and success is not guaranteed. The concentration of the heat flux from the plasma in space - on the divertor strike point, and in time - during ELMs and disruptions, exacerbates the engineering challenge and risks. Success in other challenging large scale scientific projects, viz., the recent mission to Titan, gives grounds for cautious optimism, provided the risks are clearly recognized and funding resources to mitigate them are made available in time.

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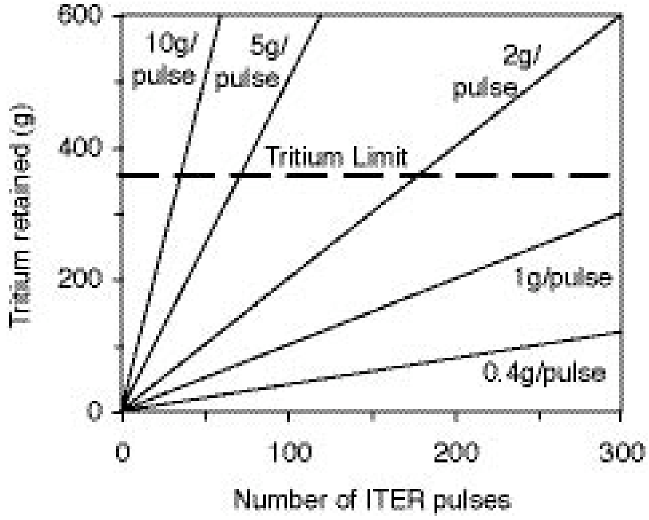
Figure Captions

Figure 1. Tritium accumulation in ITER at different retention rates. Ref. [21] predicts ≈ 3 g T retained per ITER pulse and that the inventory limit at 350 g will be reached in approximately 100 pulses. Considerable uncertainties remain (see text).

Figure 2. Situation with active tritium removal at less than 100% efficiency. The number of ITER pulses before the tritium inventory limit is reached is plotted as a function of retention rate (y-axis) and removal efficiency (labeled in percent). The vertical dashed line represents the divertor erosion lifetime of 3,000 pulses. With inefficient tritium removal the divertor will need to be removed prematurely, in an attempt to remove residual tritium and permit continued plasma operations.



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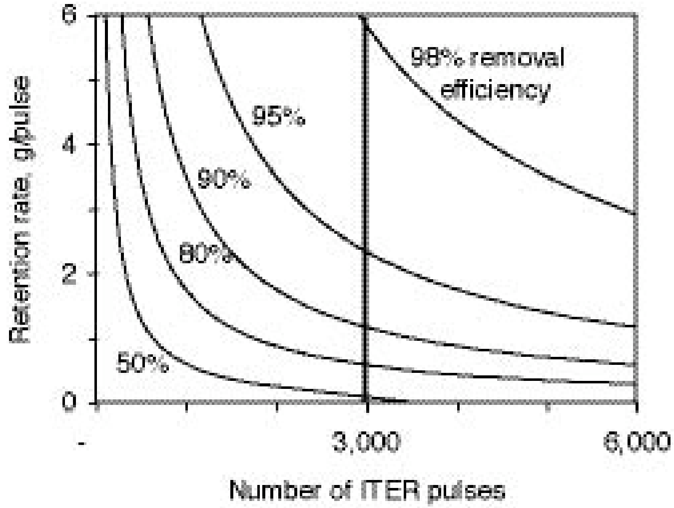

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