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** See annex of J. Pamela et al, "Overview of Recent JET Results and Future Perspectives", Fusion Energy 2000 (Proc. 18th Int. Conf. Sorrento, 2000), IAEA, Vienna (2001).*

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ABSTRACT.
Results on material erosion/deposition and fuel retention from recent MKIIGB operation in JET and from TEXTOR are summarised. Recent results from shot-resolved material deposition measurements near the inner JET divertor louver using a quartz deposition monitor are presented. Some modelling results for carbon material transport in the ITER divertor and the associated fuel retention are shown.

1. INTRODUCTION.

Control of long-term tritium retention in future burning plasma devices is among the most challenging tasks for fusion development. Safety requirements limit the amount of tritium retained in the PFC and other components of ITER to 350g and, if reached, no tritium operation is possible and cleaning must be done. Data on fuel retention (hydrogen, deuterium or tritium) in existing devices suggest that this limit in ITER may be reached very soon (after less than 100 pulses). However, these data are extrapolated from full carbon devices operating for pulse lengths of seconds while ITER operates with a majority of Be and W as PFC surfaces, hotter PFC components and quasi steady state long term pulses (400s). This alters the long-term tritium retention significantly and a simple extrapolation from present data to ITER (e.g. scaling with integrated particle flux) is thus not possible. More sophisticated modelling validated against dedicated experiments is needed. Present devices show that the majority of the long term fuel retention (we do not discuss the transient retention by plasma implantation and subsequent release from PFC which dominates the retention on a short time scale) is in hydrogen-rich carbon deposits built up on various locations on the plasma facing sides and remote areas, which are not in direct contact with the plasma. From the JET DTE1 tritium campaign it is concluded that the majority of these codeposits is on the water cooled louvers in the inner divertor while the divertor tile base temperature was at 220^{o}C. More data have been obtained since then from the gas box divertor from surface analysis and other new techniques, which will be briefly summarised together with a comparison with TEXTOR results in this field.

2. JET

It is important to note that during all the MKIIA operation time (including the tritium DTE1) phase and most of MKIIGB time the divertor surface temperature was about 220^{o}C. At the end of MKIIGB operation the temperature was lowered by about 100^{o}C for about 2 months. This is now the routine operation temperature for the new MKIISRP divertor (gas box without septum). Spectroscopy of hydrocarbons and CIII showed a clear decrease of carbon erosion with reduced temperature in the inner divertor by about a factor of two for all operating regimes, while the outer divertor showed no significant change [1]. At the end of the MKIIGB period (corresponding to a total divertor plasma operation time of about 57000 sec with an integrated ion flux towards the inner divertor of about 1.3x10^{27} ions) erosion/deposition was measured bydepth profiling with a stylometer [2] and SIMS depth profiling [3]. The overall inner divertor is a deposition-dominated area for which no comparable net-erosion in the outer divertor exists, which could account for the deposition in the inner (figure 1). The outer divertor shows no clear erosion/deposition pattern (resolution of the method about 10μm) on most of the plasma facing area, except from a narrow deposition stripe near the entrance of the pumping duct. The expected gross erosion in the outer JET divertor (zero redeposition) during the MKIIGB operation can be estimated from the integrated ion flux assuming a strike zone area of $1.3m^2$ and an erosion yield of 3% to about 400μm which would clearly show up in the measurements. This shows that the outer divertor is a region with a local redeposition probability near unity which, in connection with routine variations of the strike point position, leads to a region with roughly balanced erosion/deposition. Therefore the primary source of carbon found in the inner divertor must be the main chamber as indicated by several previous spectroscopic investigations. Identification of the erosion mechanism (ion/versus neutral impact) and quantification of the main chamber erosion is one of the important questions to be clarified in the future. Ion beam analysis of the layers from MKIIA and previous MKIIGB operation show large amounts of Be (which is evaporated in JET only in the main chamber) in the inner but little in the outer divertor. This confirms that the main chamber is the main source of material which is deposited on the inner divertor and that SOL flows drives the Be and C impurities exclusively to the inner divertor [2, 4]. Another important finding is that the layers are Be-rich on the plasma facing sides with a typical Be/C ratio of 2 or larger, although Be is a minority in the downstreaming impurity flux, as estimated from spectroscopic data (≈0.1 Be/C). Obviously the carbon co-deposited together with the Be has been re-eroded, most probably by chemical erosion (physical sputtering is of similar strength for Be) and transported towards the cold areas of the louvers. These layers on the cold louvers are essentially free of Be demonstrating that the long range material transport is specific to carbon and not expected for any metals, like Be or W.

^{13}C marked methane has been injected from the top of the machine into an ohmic target plasma at the end of the MKIIGB with already reduced divertor temperature. SIMS analysis show a strong enrichment of ^{13}C on top of the surface of the inner divertor tiles whereas no corresponding ^{13}C was found in the outer divertor proving the drag of impurities towards the inner divertor. However, the near top surface shows much more C and D than measured in deeper layers and on tiles from previous MKIIGB operation and no ^{13}C could be identified on the shadowed regions of the divertor tiles. This points towards a largely reduced transport towards the louver areas. One suggestion is that this behaviour might be connected with reduced chemical erosion due to reduced wall temperature but the ohmic target plasma chosen might also not be representative to produce the long-range carbon transport. This explanation is also consistent with data from the quartz microbalance (QMB) monitor mounted on the inner louvers.

From the thickness of the layers on the inner divertor and the measured D content, the overall amount of D retained on the plasma facing side of the inner divertor Fig 1 Layer thickness in the divertor after MKIIGB operation [2]) Inner Divertor Outer Divertor directly hitted by ion flux for the MKIIGB operation can be estimated about (10±3) gD and about 15g on the shadowed areas on tile 4 and 3. This leads to a retention rate of 2.2×10^{-3} D/ion (1.6×10^{-4} gD/s) and 3.3×10^{-3} D/ion (2.4×10^{-4} gD/s) on the ion flux and shadowed areas respectively. About 2.4×10^{26} Deuterium atoms have been injected through gas injection and beams (pellet injection is neglected here) resulting in a fuel retention on those areas of 3.1%. One should notice that these numbers do not include the deposition on other areas like the inner divertor louver area, which has been identified as the major carbon and deuterium sink during the DTE1 operation. This topic has been addressed with the new QMB and sticking monitors mounted in front of the inner divertor louver entrance [5].

The QMB has been in operation since the beginning of the new MKIISRP divertor and operates at the same temperature ($\approx100^{o}$C) as the divertor tiles while the louvers are always water-cooled. Figure 2 shows the measured deposition rate for a selection of H-mode shots depending on the strike point position. L-mode shots show in general a significant lower deposition. Large deposition on the QMB is only seen for shots with the strike point down on tile 3. The mean deposition rate (at this strike point position) of 4×10^{15} C/cm^{2}s results in a total deposition of 1.8×10^{19} C/s on the whole louver area (4500cm^2). This leads to an overall D retention of 5.8×10^{-4} D/ion or 4.2×10^{-5} gD/s (assuming D/C=0.7, mean ion flux 2.2×10^{22}/s). This is significbltly smaller than the retention rate at the plasma facing sides from the overall MKIIGB period estimated above and much smaller than the estimated mean D co-deposition rate on the louvers of about 7×10^{-3} gD/s for the DTE1 tritium experiences. The reasons for this behaviour are presently under strong investigation and possible reasons are:

- The QMB monitor operates at 100^{o}C where already significant re-erosion can occur while the louvers are water-cooled. This behaviour is in agreement with recent measurements in the Berlin plasma simulator [6].
- The divertor base temperature during the QMB operation was reduced to 100ºC leading to reduced chemical erosion and material transport
- DTE1 used mainly hot ion ELMy H modes with large ELMs and the strike point at the base plate. This configuration has not been used so far with the QMB but is expected to be especially effective to transport carbon towards the louver gap.

Of similar importance is to identify the underlying reasons for the large variations in the carbon deposition at the lower strike point position. This analysis is not completed but the present analysis show that the largest deposition is for shots at small, natural density resulting in few ELMs with large size. Figure 3 shows the deposition rate of all H-mode shots at lower strike point position against the mean ELM frequency during the opening of the QMB shutter. Shots with fewer ELMs and larger energy drop per ELM show more deposition than gas fuelled shots with more and smaller ELMs. Moreover, a pair of similar H-mode shots with NBI and ICR heating shows about a factor of 3 more deposition for NBI than for ICRH. Simultaneously the edge density of the RF shot is lower which suggests that the important parameter for the variation in deposition is the ELM size rather than the divertor density. More work is needed to confirm this present conclusion.

4. TEXTOR

During the TEXTOR opening a systematic mapping of carbon and D-retention pattern on the limiters, obstacles in the SOL, neutraliser plates on the pump limiter, first wall (liner) and remote areas in the pump ducts was done [7, 8]. Net erosion has been quantified using specially prepared limiter tiles. The results are as follows [9]. The net erosion area is about 2/3 of the surface of the main toroidal limiter ($3m^2$) with an averaged rate of about 6×10^{-3} g/s. About half of this is directly redeposited on the limiter itself (2.7×10^{-3} g/s). Another large part is found on various obstacles in the far SOL of TEXTOR like protection limiters, antenna screens etc (1.6×10^{-3} g/s). About 5×10^{-4} g/s is redeposited on the neutraliser plates of the 8 pump limiters and $2.5-5\ 10^{-4}$ g/s is pumped out in form of volatile hydrocarbons through the pumps. Only about 5×10^{-6} g/s is found on remote ares in the pump duct of the pumping system, very close to the neutraliser plates. The remaining part could not be identified clearly. The deuterium content in these layers has been determined showing large variations with relative D fractions $<10^{-3}$ on special obstacle in the SOL and $<10^{-2}$ on the neutraliser plates up to about 0.7 D/C on the soft layers found in the pump duct. This ends up in an estimated D retention rate of about 3.8×10^{-5} gD/s. Thus TEXTOR would reach the T-safety limit (350g) in about 100 full operation days. It is interesting to note that an overwhelming majority of the eroded carbon in TEXTOR is redeposited on plasma facing areas inside the device, while only a small minority leaves the system and is deposited on remote areas. This is opposite to what is observed after the tritium campaign at JET where 90% of the remaining T was found on remote areas. This isprobably due to the different plasma parameters in the vicinity of the high recycling areas which lead to different local transport processes. In view of the new QMB JET results, other reasons like the absence of ELMs in TEXTOR must be also considered.

5. ITER

To predict the tritium retention of ITER Monte Carlo simulations with the ERO-ITER code using the plasma background (temperature, density from B2-Eirene) for a reference scenario (410MW, Q = 10) were carried out. In first attempts, the material deposition from the main plasma has been omitted and the local carbon transport in the divertor volume has been addressed. The sticking for hydrocarbons has been set to zero, according to assumptions needed in TEXTOR to explain local redeposition of hydrocarbon injection [10]. The redeposition of eroded methane on the targets is about 88% with no significant difference between inner and outer divertor. With no background impurity content, net-erosion of carbon occurs all along the divertor plates. The 12% of carbon non- redeposited on the plates are lost towards the dome region or the base plates in the private flux region and assumed to lead to net deposition. With a D+T/C ratio of 0.7 a tritium retention of about 3mg T/s for a chemical erosion of 1% is obtained. WBC code calculations with somewhat different plasma parameters, a molecular dynamics code sticking model and larger chemical erosion (changing between 1−2 %) result in a higher value of 18 mg T/s [11,12]. None of these values include deposition of Be from the main chamber on the targets and this needs more detailed modelling which is underway.

SUMMARY

- JET as a full carbon device shows that the majority of the fuel trapping is in codeposited layers in the inner divertor. The majority of the carbon deposited there is from erosion in the main chamber
- Carbon and beryllium (evaporated in the main chamber) shows completely different long range transport behaviour with Be staying on the plasma facing sides and carbon transported towards remote areas.
- New measurements from MKIIGB indicate a reduced carbon deposition on the inner louver area when compared with the JET tritium phase, the reason for this is not fully clarified.
- Table 1 shows the different experimental estimates for the fuel retention rate in TEXTOR, JET campaigns and the estimates for ITER.
- From our present understanding we estimate a significant reduction of the fuel retention in a device with a metallic first wall compared to a full carbon device (as for ITER).

Campaign	Base tile temperature (°C)	Integrated particle flux (ions/s)	Retention on PFC (gD/s)	Retention on inner divertor louver (gD/s)
Tritium in DTE 1	220	1.3×10^{22} (inner)	1.5×10^{-2} on all PFC (all limiter fluxes)	7×10^{-3} (after cleaning)
D in MKIIGB	220	2×10^{22} (inner)	4×10^{-4} (on inner divertors tiles)	
D in MKSRP (C5)	120	1.9×10^{22} (inner)		4.2×10^{-5} (from QMB)
D in TEXTOR	250	5×10^{21} (all limiter fluxes)	3.8×10^{-5}	
ITER	220	1.6×10^{21} (both divertors)	3×10^{-3} (*) 18×10^{-3} (**)	

(*) ERO-ITER, no background impurity flux included, 1% chemical erosion, only methane (**) WBC, Brooks et al [11], no background impurities, 1-2% erosion, with higher hydrocarbons

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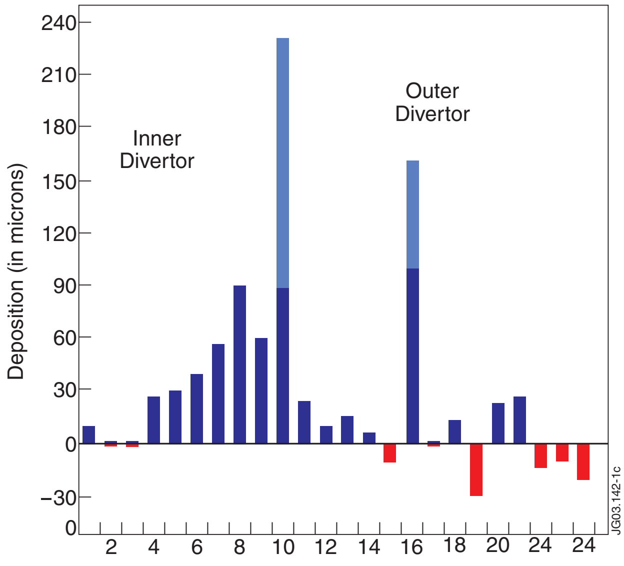

Figure 1: Layer thickness in the divertor after MKIIGB operation [2])

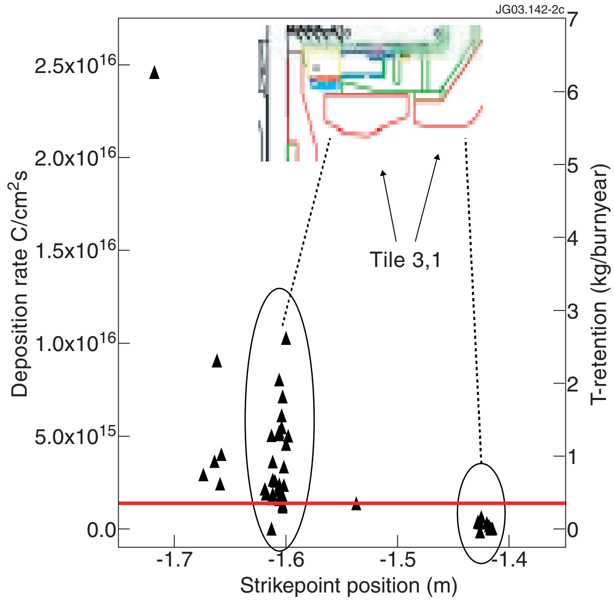

Figure 2: Carbon deposition on the QMB in inner divertor louver versus strike point position. H-mode shot conditions, P_{in}>8MW. The right axis shows the corresponding T-retention in JET assuming a full burn year. The red line is the T-saftey limit (350g).

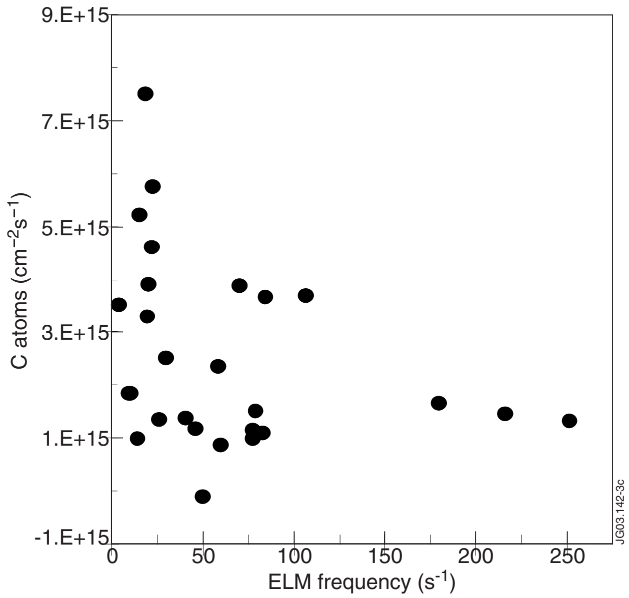

Figure 3: Carbon deposition rate at QMB versus ELM frequency (P_{in}>8MW) point position (m)