hydrocarbons in the erosion products is essential to obtain a valid erosion yield in these conditions.

New experiments on W migration have been performed with a localized set of W tiles in JT60U. W is found to be redeposited locally toroidally (probably due to prompt re-ionization and short range migration), while C is transported further toroidally (from $^{13}C$ injection experiments). The W core concentration seems to depend more sensitively on the particle transport properties (increasing with counter toroidal velocity) than the W source at the edge. In contrast with JT60U, a complete ring of W divertor tiles installed in C-Mod has led to no observable level in the core. New

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the modeling of retention due to nuclear damage (W) and writing a summary of the work. We also intend to expand the group work in this area with the addition of experts outside of the US and EU (e.g. Japan).

Two issues have led to a study by ITPA members of H levels when operating in D plasmas: 1) The ‘additional’ gas load on cryopumps which, if too high, could lead to reaching the deflagration limit in cryopumps; and 2) the interest in measuring fuel retention during both the ITER H and D phases. Large levels of wall outgassing (H remains in surfaces following a vacuum break) could add an additional unknown fueling source. The data reported for both carbon PFC (EAST, TS, DIII-D) and hand such deformation of the surface obviously damages the surface and probably degrades the material properties in terms of heat load handling and shock resistance. We are just starting to address the effect of neutron damage which creates traps for T retention throughout the tile. The talks at this meeting and at the MIT meeting indicate that such damage can lead to a maximum of ~ 1% [D+T]/W. Then, it is a matter of when the D+ T implanted at the surface can diffuse to those traps and thus be retained. Will it be slow or fast? Lastly it was pointed out that simultaneous implantation of He along with the D,T fuel can affect the diffusion of trapping of D,T. There are differing reports on whether the effect increases or decreases D,T retention and this will be a subject of review in upcoming meetings.

The present ITER dust safety strategy relies on measurements of gross erosion, and assumes as a conservative upper limit a conversion factor $f_d$ between gross erosion and mobilisable dust equal to 1. The previous and current tokamak studies for carbon PFC machines reported values of $f_d$

and circulated prior to the IEA/ITPA committee meeting (Dec. 2008). Five areas were selected which parallel the ITER high priority R&D areas:

1) T retention and removal;
2) development of experience and understanding of tungsten as a Plasma Facing Component (PFC) material;
3) heat fluxes to all surfaces (transient and steady state);
4) dust;
5) material migration.

Two co-leaders were asked to lead each of the above tasks. This meant developing a set of new sub-tasks, deciding the level of priority, soliciting input from the experts in the field, and lastly leading the discussions at the Amsterdam meeting. Summaries of those sessions are found in Section II-2 For each R&D task, there are plans in place varying from paper studies (e.g. to evaluate the effect of higher bakeout temperatures in ITER) to initial data collection (e.g. the timescales and physics of re-attachment), and more directed se2038spove removal, the role of ICWC for impurity removal and whether we can properly predict the requirements for ITER. Future experiments are planned on a number of machines.

Planned mitigated disruptions have been suggested as a mechanism for fuel removal in ITER through uniformly flash heating all surfaces. The aim of the session was to assess the amounts of fuel removed and the dependence on operational/physics parameters from existing data mining. The reported post disruption recovery spans over a large range, from 3-4 $10^{21}$ D to a few $10^{22}$ D (TS, C-Mod, JT60U, JET on average) and up to $10^{23}$ D at high stored energy in JET and high current in C-Mod. It appears to scale roughly linearly with plasma thermal energy (C-Mod, JET, TS), but also stronger than linearly with plasma current and magnetic stored energy (C-Mod, TS). The energy available in ITER disruptions is generally thought to be sufficient for a significant fuel release, but these planned disruptions should be carefully tailored for fuel recovery while avoiding PFC damage and allowing an easy plasma start up for the next discharge. Further work will address fuel recovery after mitigated disruptions versus un-mitigated disruptions, and a more detailed analysis of the thermal behaviour of the PFCs during the disruptions,

Finally, issues related to divertor detachment were addressed, with two sessions on divertor re-attachement and modeling of divertor detachment.

Divertor reattachment can be due to a number of processes: failure of impurity seeding, failure of fueling system, a rapid change in plasma-wall interaction (for example wall outgassing), confinement changes (e.g. L-H, H-L transitions) and plasma state “bifurcations”. An important question to answer is whether or not the currently foreseen gas injection systems on ITER will be sufficient to protect the divertor should divertor reattachment lead to a fast increase in divertor heat loads. This meeting represented an initial survey of existing tokamak experience, in particular on timescales of divertor re-attachment. While the H-L transition can be fast (~milliseconds) it appears that if impurity gas feedback is being utilized at the time, the impurity gas mitigates the rapid increase in SOL power flow to the divertor through radiation – effectively slowing down the transition as far as the divertor heat loads are concerned. Timescales quoted were in the range of 100s of ms. More data is needed beyond this initial study. The recommendation is that dedicated

### 1.3 IEA/ITPA multi-machine collaborations

The status of the DSOL experiments is summarized below (red : closed DSOL, blue : ongoing DSOL, green : new DSOL).

DSOL-1 Scaling of Type-1 ELM energy loss and pedestal gradients through dimensionless variables (A. Loarte) Closed

DSOL-2 Chemical erosion under ITER-like divertor conditions (semi-detached) (S. Brezinsek)

Proposal: TEXTOR, JET, AUG, JT-60U, DIII-D

DSOL-3 Scaling of radial transport (B. Lipschultz) Closed

DSOL-4 Comparison of disruption energy balance in similar discharges and disruption heat flux (A. Loarte) Closed

DSOL-5 Role of Lyman absorption in the divertor (S. Lisgo ) Closed

DSOL-8 ICRF Conditioning for hydrogen removal (N. Ashikawa)

Proposal: LHD, HT-7, EAST, AUG, TEXTOR, TORE SUPRA, JET

DSOL-9 Tracer injection experiments to understand material migration (V. Philipps)

Proposal: JET, DIII-D, TEXTOR, AUG

DSOL-11 Disruption mitigation experiments (D. Whyte) Moved to MDC-11

DSOL-12 Reactive gas wall cleaning (P. Stangeby)

Proposal: TEXTOR, HT-7, EAST, DIII-D

DSOL-13 Deuterium codeposition with carbon in gaps of plasma facing components (K. Krieger)

Proposal: data from AUG, TEXTOR, MAST, DIII-D, TORE SUPRA, C-MOD

DSOL-14 Multi-code, multi-machine edge modelling and code benchmarking (Coster)

Proposal: Codes only (Database in AUG, JET, DIII-D, C-MOD, JT-60U is required)

DSOL-15 Inter-machine comparison of blob characteristics (J. Terry)

Proposal: C-Mod, PISCES, DIII-D, JT-60U, VTF, JET, AUG, TJ-II, VINETA, NSTX, TEXTOR

DSOL-16: Determination of the poloidal fueling profile (M. Groth)

Proposal: DIII-D, AUG, JET, MAST, C-MOD, JT-60U

DSOL-17: Cross-m x DSOL-17:

Concerning DSOL14 on code-code benchmarking, the scope should be re-assessed in view of activities already going on in other structures (EU ITM TF, JET, ITER). It is proposed to focus it on detachment modelling issues.

Ongoing DSOL, with new experiments planned, include : DSOL8 on ICWC (TS, TEXTOR, AUG, JET), DSOL9 on material migration ($^{13}C$ tracer experiments in TEXTOR, AUG, JET and associated modelling), DSOL12 on O cleaning (lab experiments + TEXTORa21)t

## 2. High Priority Research Areas

As mentioned in the executive summary and also evident in the Amsterdam meeting report of Section 1.2, the strategy adopted by the SOL/Divertor TG to address urgent ITER R&D needs in the plasma-wall interaction area, has been to establish a set of high priority R&D areas which parallel those identified by ITER in 2008 and presented to the STAC-5 meeting. Leaders have been identified from within the TG membership to drive the overall research activity in each topical area. They have selected a number of subtasks, for which further coordinators have been appointed or are being sought.

The table below compiles the five targeted areas, summarising the subtasks which have been identified to constitute a work plan in each area and labelling them as high or medium priority. The "R&D Type" descriptors in parentheses after each subtask title indicate by which methodology we expect the subtask work to be achieved or presented.

It is our goal to complete high priority tasks in the next few years at least to the level at which

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|---|---|---|---|
| | coordinated) | | |

Fuel retention in gaps

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|---|---|---|---|
| **Heat fluxes to plasma-facing surfaces**<br>*Leaders:*<br>A. Leonard<br>M. Lehnen | Disruption heat loads (ongoing discussion within ITPA) | 1-2 years | High |
| | Cross-machine characterisation of ELM statistics (DSOL-15, PEP-10, 21) | 2-4 years | High |
| | Heat loads during ramp-up/ramp-down (tokamak experiments, report) | 1-2 years | High |
| | Transient divertor reattachment (DSOL-20) | 2-4 years | High |
| | Modelling detachment (new coordinated effort within the TG, re-focus of DSOL-14) | 2-4 years | Medium |
| | Far SOL heat and particle fluxes (US comparison, extend to ITPA) | 1-2 years | Medium |
| | Divertor and SOL ELM heat fluxes – characterisation of power footprints and parallel heat flux widths (ongoing ITPA meeting, reports) | 1-2 years | Medium |
| **R&D Topic Area** | | | |