Annual Report of ITPA Topical Group on Scrape off Layer and Divertor

For the period January 2012 to December 2012

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Executive Summary

The SOL/Div Topical Group held two meetings during the reporting period, in Juelich, Germany, January 16-19, 2012 and in San Diego, USA, 15-17 October 2012. The first meeting was held in connection with the selection committee for the International Conference on Plasma Surface Interactions (PSI), while the second was coordinated with the timing and location of the IAEA Fusion Energy Conference. The San Diego meeting was organized simultaneously with other ITPA TG meetings, allowing joint working sessions, focussed on issues of common interest (SOL width with Pedestal TG, disruptions with MHD TG).

The DivSOL TG organization has evolved in 2012, as the TG activities are now structured around 3 main topics (instead of 5 previously), gathering fuel retention, dust and material migration under a single topic, as the 3 issues are closely related. The new structure and the associated topical leaders are the following :

- Heat fluxes : A.W. Leonard (US), M. Lehnen (EU)
- Tungsten R&D : Y. Ueda (JA), G de Temmerman (EU)
- Material migration, fuel retention, dust : R. Doerner (US), K. Schmid (EU)

Selected scientific highlights over the reporting period are given below in the 3 research areas. More details can be found in the summary of the meetings in Section 1, as well as on the ITER IDM ITPA Sharepoint site. Specific ITER IO activities, in particular regarding the status of the PFC design or modelling activities, were regularly presented, providing up to date information to the TG experts, as well as generating fruitful exchanges of views.

- Heat fluxes (leaders A. Leonard, M. Lehnen)
    - Predicting H mode SOL width for ITER (joint session with pedestal TG) : the large effort undertaken in EU and US to establish a multi-machine scaling of SOL width was pursued, leading to a consolidated database from 6 tokamaks for attached low recycling regimes. The SOL width scaling derived from this study predicts a narrow SOL for ITER (λ_q~ 1 mm). The consistency with pedestal stability was discussed, showing similar scaling for the ballooning critical pressure scale length just inside separatrix,, but predicting a larger critical width. A DSOL is now proposed to extend this work to high recycling/partially detached regimes and progress on the physics understanding.
    - Limiter SOL width for ITER start up : a multi-machine database has been assembled, containing probe data from 7 tokamaks to provide input for the ITER limiter heat load specifications. Extrapolation from the present scaling indicates a 50-60 mm SOL width for a 7.5 MA ITER configuration (larger than previous assumptions from diverted L mode data). However, compared to probe data, IR data, when available, show enhanced heat flux at grazing angle incidence near the tangency point. This effect is significant enough that when Be melting was observed at JET, it was at the near tangency point. This issue of enhanced near SOL heat flux on limiters should be further investigated

- Disruptions in metallic environment (joint session with MHD TG): data from metallic devices were discussed and compared with carbon references. For unmitigated disruptions, tokamaks equipped with metallic walls exhibit a significantly lower radiated fraction compared to carbon walls. This leads to higher heat loads on the plasma-facing components, and Be melting has been observed on the upper dump plates and protection limiters in JET. The disruption timescales are also impacted, with slower current quench for the metallic walls. This in turn influences the runaway e- generation : no runaway e- have been observed so far with the JET ILW. Massive Gas Injection (MGI) allows significant radiation fractions to be recovered during disruptions. MGI is also shown to shorten the current quench and lower the halo currents.

- Tungsten R&D (leaders Y. Ueda, G. de Temmerman)
  - Impact of tungsten melting on plasma operation : tungsten melting experiments were reported from various tokamaks, showing possible concern for subsequent plasma operation depending on the experimental conditions. In AUG, melt ejection events were evidenced but melt droplets in the divertor do not affect the confined plasma. However, detachment of melt debris from resolidified and brittle tungsten can cause disruptions. In C-mod, operation was prevented when the strike point was located in the area of a previously heavily damaged W protruding tile. The preparation for the JET melting experiment, targeted at shallow melting from transients, was also discussed.
  - Tungsten damage under combined loads: combined steady state and pulsed heat loads showed that the W damage threshold is much lower than the melting threshold (~0.2 MJ/m^2). A large number of transient cycles also lead to rather low threshold for W cracking (0.1 MJ/m^2 for high repetition rates (~10^6)). For ITER grade tungsten with grain boundaries perpendicular to the surface, cracks develop along these grain boundaries and do not affect heat removal capability as long as the grain structure is maintained. However, at elevated temperatures, recrystallization takes place, altering the microstructure of tungsten. The effect of W recrystallization and repetitive pulsed heat loads, as well as combined effects with exposure to He plasmas, will be further investigated (DSOL proposal).
  - Tungsten morphology changes under plasma exposure: dedicated studies were performed to study the behaviour of W fuzz under transients. It was shown to survive exposure to VDE loads in DIII-D, as well as moderate transients in PISCES-B. However, melting of the fuzz top surface is observed for larger heat loads. W fuzz was also shown to be prone to arcing, which removes the fuzz efficiently.

- Material migration, fuel retention, dust (leaders R. Doerner, K. Schmid)
  - First results from the JET ITER-Like Wall (ILW): the JET ILW provided very important preliminary information on Be migration and fuel retention. Significant Be re-erosion flux in the divertor and the corresponding impact on W erosion were seen right from the beginning of the experiment. Fuel retention is strongly reduced with the ILW, compared to the full carbon configuration, as expected (factor ~10 decrease). The material migration code DIVIMP-WALLDYN is being benchmarked against JET data, to gain confidence for extrapolation to ITER. Plasma restart after

shutdown is also seen to be faster with the ILW, while less conditioning is required during operation, probably due to the good gettering properties of Be.

- Control of W accumulation required: it was shown in JET ILW as well as in AUG that W control is required to avoid W accumulation, through central heating, ELM flushing and/or minimal fuelling. Enhanced core W concentration is observed with ICRH, which currently cannot be explained by a corresponding rise in the W source.
- Dust monitoring during plasma operation: dust has been monitored during plasma operation, showing a strong correlation between dust production and disruptions.
- Fuel removal from Be: it was shown that efficient removal by baking needs higher temperature than the temperature at which co-deposition occurs. The efficiency is also sensitive to the composition of deposited layers. Simulations with the TMAP code indicate that thick co-deposits are less efficiently desorbed. The analysis of the Be deposits produced in JET will be important for comparison with lab experiment and code benchmarking

In addition, the San Diego meeting was also used to initiate work on an ITPA DivSOL report regarding the proposed ITER strategy to start operation with a full W divertor. This was requested by the ITPA CC chair, following a recommendation from the ITER STAC (ITER STAC 12 meeting, May 2012). Preliminary discussions were organized on the main PWI issues impacting the divertor material choice: W melting/damage, W material issues and fuel retention for the C/W/Be versus W/Be material configurations. The structure of the report, associated deadlines as well as contributing authors were agreed. Furthermore, the DivSOL TG chair was asked to coordinate the work amongst the various ITPA TG involved, and the outline of all TG reports were discussed and agreed in a dedicated meeting in San Diego.

Finally, the status of collaborative DSOL experiments is reviewed in Section 2. Five DSOL proposals were running in 2012, two news proposals are foreseen for 2013.

The next meeting of the Div/SOL Topical Group is planned in Hefei, China, March 2013. A large part will be devoted to discussing the DivSOL report on the ITER divertor strategy requested by the ITPA CC.

Content

1. Meetings and reports

The summary report and all presentations given at the 16th and 17th meeting of the ITPA Div/SOL TG, can be found under the ITER IDM web site and only the executive summary is repeated here.

1.1 Report on the 16th Meeting of the ITPA SOL and divertor physics Topical Group, Juelich, Germany

The 16th meeting of the ITPA Divertor and Scrape-off Layer (DivSOL) Topical Group (TG) was held from January 16 – 19 2012 in Jülich, Germany, hosted by the FZJ plasma edge physics team. The meeting was largely attended as usual (more than 50 participants). The 3.5 days DivSOL TG meeting was followed by the paper selection committee meeting of the International Conference on Plasma Surface Interactions, as is usually the case before this major conference in our field, which will be organized by FZJ in Aachen in May 2012.

During the meeting, an evolution of the DivSOL TG organization was proposed by the new leadership, in place since end 2011. It is proposed to structure the activities of the TG around 3 main topics (instead of 5 previously), gathering fuel retention, dust and material migration under a single topic, as the 3 issues are closely related. The new structure and the associated topical leaders is thus the following :

- Heat fluxes : A.W. Leonard (US), M. Lehnen (EU)
- Tungsten R&D : Y. Ueda (JA), G de Temmerman (EU)
- Material migration, fuel retention, dust : R. Doerner (US), K. Schmid (EU)

Papers for the next IAEA were also discussed, and the joint effort on SOL widths recently performed in EU and US (see highlights below) was flagged as a good candidate for an oral contribution.
The scientific highlights of the meeting included :

- the first results of the JET ITER Like Wall, in particular in the field of beryllium migration and fuel retention
- the output from the large effort dedicated to multi device SOL widths scaling, pointing at potentially narrow SOL width for ITER.
- the first discussion on in situ measurements of dust production during tokamak operation
- first evidence of W fuzz formation in the CMod tokamak, following studies in linear devices

The summary below is divided into the 3 topical areas, summarizing the main topics discussed at the meeting.

Heat fluxes

A large session was devoted to discussing the progress in the area of SOL widths multi machine studies, as a follow up to the previous TG meeting. The same fitting procedure has now been used to derive inter-ELM widths, measured at the outer target in strongly attached, high power regimes, on 5 tokamaks. Scalings drawn for the US and EU devices indicate in both cases $\lambda_q \propto 1/B_{pol}$, with little or no P_{SOL}, size (major radius) or density dependence. Both datasets are consistent in absolute magnitude with the Goldston drift model, which predicts that λ_q should be proportional to the poloidal gyroradius ($\lambda_q \propto 1/B_{pol}$). Extrapolated to ITER, these scalings suggest $\lambda_q \cong 1$ mm at 15 MA, about a factor of 3 shorter than the lowest values predicted on the basis of earlier scalings. The consequences for ITER divertor target heat loads was addressed through preliminary simulations with the SOLPS package with a short λ_q performed by IO, showing that provided the divertor neutral density is not too low, the volumetric power dissipation (mostly radiation) is strong enough to reduce the peak target heat fluxes to or below the power handling capability of the monoblocks. However, more runs where transport coefficients are varied independently are needed to consolidate this result. Next issues to investigate are the consistency of this short SOL width prediction with pedestal stability and the extension of the database to higher density partially detached plasmas.

Good progress was also reported on the issue of limiter heat flux widths, with a rigourous effort from Tore Supra to improve the derivation of SOL widths from probe measurements, eliminating large uncertainties in the procedure. After an exhaustive analysis, the conclusion is that the measured λ_q are about half the previously reported values but that the overall scaling from T-S (going like $\lambda_q \propto P_\Omega^{-0.6}$) is preserved. New data have been presented from JET both from reciprocating probe measurements as in Tore Supra, but also with an IR camera viewing the inboard limiter. An effort to combine the data available (TS, DIIID, JET) once they are fully analyzed will be organized soon, since the information is required for the ITER heat load specifications.

Finally, a session was devoted to disruption heat loads mitigation by massive gas injection (MGI). New results were presented from DIII-D on VDE heat loads, both with and without Neon MGI. Sufficiently early MGI can partially mitigate heat loads (up to a factor 2 has been observed). Full suppression has not been seen, even at almost 100% radiated energy, which is tentatively attributed to heat fluxes from radiation. C-Mod reports on significant broadening of the power footprint extending beyond divertor surfaces during both triggered VDEs and in-place disruptions. In all cases, the heat loads were so much broadened that assigning a width was not possible. JET data with the ITER-like wall shows high heat loads during upward VDEs due to a lack of radiating impurities (in contrast to the case with a carbon wall), which results in a significant fraction of magnetic energy being conducted to PFCs. High radiation has been re-gained using MGI. The situation in MAST is similar: in non-MGI disruptions, most of the magnetic energy is conducted to the divertor. A reduction of the energy load in the divertor of down to 40% of the total stored energy can be achieved with MGI. About 40% is radiated and the rest is unaccounted for. TEXTOR reported on broadening of limiter heat fluxes of a factor 2-3, being consistent with earlier publications, but less than reported from divertor machines. JET and TEXTOR reported on strong background emission polluting the temperature measurements during MGI, whereas C-Mod has seen no significant background for non-MGI disruptions. Runaway heat loads observed on a test limiter in TEXTOR show a very small wetted area with radial deposition width less than 1 mm. Tore Supra has tested the impact of ultra-fast gas injection at high pressure using rupture discs to mitigate runaways. This technique aims to destabilise the CQ plasma. However, runaway suppression has not yet been achieved in the preliminary experiments performed. Concerning disruptions, the Div/SOL TG will continue collaborations with the MHD TG to make sure that the appropriate surface flux measurements are made during more general disruption experiments. A joint session with the appropriate experts will likely take place at the next meeting in the fall after the IAEA meeting.

Tungsten R&D

The formation of tungsten nano-structure, usually called fuzz, was observed so far in linear plasma devices. For the first time, the formation of tungsten fuzz was also evidenced in a tokamak, loading a W probe in CMod under the conditions identified as leading to fuzz formation from linear devices studies (high heat flux in He plasma). However, results from the linear device Pilot PSI seem to indicate that ELMs might eliminate the fuzz structure without significant release of W into the plasma. More integrated studies in relevant conditions (plasma conditions, He concentration, surface temperature, ELMs, impurity seeding etc) are needed to elucidate whether W fuzz might be a concern for ITER, in particular in terms of arcing and enhanced erosion.

Tungsten melting experiments were reported from AUG, CMod and TEXTOR, showing possible concern for subsequent plasma operation depending on the experimental condistions. In AUG, a protruding W edge showed small core penetration of initially released W droplets. However, the resolidified W appeared very brittle and caused significant pollution and disruption in the consecutive discharge. W melt experiments in TEXTOR showed improved behavior of ultra-high purity tungsten compared to Ta-doped tungsten. However, the ultra-high purity W appears more prone to crack formation due to the large grain size after recrystallization. C-mod showed toroidal and poloidal redistribution of W after tile melting, which prevented operation when the strike point was located on the heavily damaged W area. Modelling of melt layer transport with the MEMOS code showed good agreement with experiment.

A session was devoted to testing W component behaviour under relevant ITER conditions (steady sate and transient heat flux, He plasmas …). Development of tungsten divertor monoblocks technology to withstand the ITER heat load specifications is in progress. High heat flux studies show the design is now mature for baffle conditions (up to ~15 MW/m^2) but still needs improvement for strike point conditions (up to ~20 MW/m^2 for slow transients, where damage can be observed). He ions bombardment from DT fusion plasmas or He plasmas in a non-active phase would further cause surface damage, which was shown by ion beam irradiation in GLADIS. This effect was also seen by D and He mixed ion irradiation. Combined effects of recrystallization at elevated temperatures and He ion exposure are another important issue. Micrometer size He bubbles are formed on the surface of tungsten at around 2000 ˚C or more by high density He plasma exposure. Simultaneously, some He atoms diffuse into bulk and accumulate along grain boundaries to form He bubbles, leading to weakening of adhesion between adjacent grains. This combined effect on surface morphology change and grain ejection has to be investigated.

Response of tungsten to ELM pulsed heat loads is also an issue for full W divertor. Combined steady state and pulsed heat loads in Pilot-PSI showed that the damage threshold was about 0.2 MJ/m^2, much lower than the melting threshold (~1 MJ/m^2). Cracking thresholds for various tungsten grades were investigated by the electron beam facility JUDITH, showing that a large number of cycles lead to rather low threshold for cracking (0.13 MJ/m^2 for high repetition rates (~10^6)). For ITER grade tungsten with grain boundaries perpendicular to the surface, cracks are developed along these grain boundaries and do not affect heat removal capability as long as the grain structure is maintained. However, at elevated temperatures, under ~20 MW/m^2, recrystallization takes place, altering the microstructure of tungsten. The effect of recrystallization and repetitive pulsed heat loads, as well as combined effects with exposure to He plasmas, are important issues to be further investigated to assess the potential impact on plasma performance.

Concerning W migration, first results from W sputtering in JET were presented. Tungsten sputtering exhibits the expected parameter dependence with plasma density and heating power, showing a strong plasma temperature dependence. The tungsten concentration depends strongly on ELM frequency, pointing towards the importance of W flushing. A strong rise of the core W concentration with ICRH is observed, which currently cannot be explained by a corresponding rise of the W source. The quantitative analysis of the W influx from atomic data is still a complex issue. COREDIV modeling of the W behavior in N seeded AUG discharges showed good agreement with most global discharge parameters, but unresolved discrepancies on the core tungsten concentration and the corresponding radiative losses

Material migration, fuel retention, dust

Recent results on fuel retention from different devices were presented. In JET with the new ITER Like Wall (Be wall, W divertor), fuel retention is shown to be strongly reduced in comparison with the carbon configuration, as expected (by a factor 10 at least). However, with the ITER Like wall (ILW), the dynamic retention during the limiter phase is higher that with the C wall, but this is compensated by a stronger outgassing after the discharge. Further understanding of mechanisms responsible for dynamic retention is still needed. Tore Supra reported that the issue of the large discrepancy between measurements of fuel retention by gas balance and post mortem analysis has been reconciled by taking into account long term outgassing between discharges (over nights, week ends etc). In EAST and HT7, the retention after coating the PFC's with various light elements (B, Li, Si) was investigated, showing all of them increased retention; in particular Li coatings. In JT60U, significant retention due to co-deposition was measured in the gaps. The total retention rate including both plasma facing and shadowed areas was found to be 10^{21} D/s, in line with JET or AUG.

Recent results on isotope exchange, proposed as a method for fuel removal in ITER, were presented. In particular, questions remain on how efficient the process is and which depths can be reached. To that end dedicated experiments were performed on the isotope exchange in Be and W both in ion beam (High Current Ion Source in Garching) and linear plasma devices (PISCES). In the ion beam experiments it was found that already a very small H fluence ($2x10^{-4}$ of the previously implanted D fluence) was required to remove 30% of the previously implanted D from a W sample. However during a subsequent D implantation the re-uptake of D into the W is close to 100% for low fluences. This suggests that the uptake of D in an already H saturated sample is much faster than into a pristine W sample. Similar results were also obtained for Be in ion beam experiments. The results from PISCES are quite different, as it takes essentially the same H fluence to exchange the previously implanted D. A possible reason for this difference is that the D removed via isotope exchange is recycled and subsequently re-implanted, thus making isotope exchange in a plasma environment less effective than in ion beam experiments. Whether or not isotope exchange will be applied to ITER depends on how efficient it is compared to a 350°C bake of the ITER divertor or a 200°C bake of the ITER main wall.

Concerning material migration, the JET ILW provided very important preliminary information on the Be erosion and migration behaviour. Significant Be re-erosion flux in the divertor and the corresponding impact on W erosion were seen right from the beginning of the experiment. The W erosion yield at the divertor target reached steady state after approximately 50s while the Be re-erosion yield at the outer target was seen to still decrease on a longer time scale (100 s) which could be due to local Be redistribution into shadowed tile castellations. In contrast to the Be main chamber erosion flux, which became constant after the first few discharges, residual carbon sources were continuously decreasing even after 1000s accumulated discharge time, in line with observations from AUG. In PISCES, the Be erosion yield in Be seeded plasma is higher than expected : this is now identified as being due to an enhanced re-erosion of redeposited Be, similarly to what was observed for deposited carbon. This behaviour may influence the local and global migration of Be in ITER and needs further clarification.

In addition, it should be noted that due to the good gettering properties of Be, almost all plasma breakdowns were successful with the ILW and no glow discharge cleaning was required during operation. However, the clean plasma conditions resulted in high plasma temperatures in the divertor

(~50eV) due to low impurity radiation, and required fuelling or seeding to minimise W sputtering and to reduce the power load to the W PFCs.
Plans for future experiments (MAPES set up in the EAST tokamak) were presented for progressing in benchmarking the codes used to describe material migration in castellations of PFC, an important issue for ITER Be first wall in particular.
Finally, measurements of dust production during tokamak operation were presented, showing that in most devices, dust production is strongly correlated to abnormal events such as disruptions. Ignoring these abnormal events, the long time trend seen in KSTAR is a decrease of the dust events detected with operation time after a restart, as PFC become conditioned. IPP has developed an automatic statistical dust analysis procedure, allowing to get information on the chemical composition and shape of thousands of individual particles. This was successfully applied at LHD. A dust database, based on a statistical dust analysis from various devices, has been started as part of a CRP IAEA effort : a preliminary version is now operational.

1.2 Report on the 17th Meeting of the ITPA SOL and divertor physics Topical Group, San Diego, USA

The 17th meeting of the ITPA Divertor and Scrape-off Layer (DivSOL) Topical Group (TG) was held from October 15 – 17 2012 in San Diego, USA, hosted by GA. The meeting was largely attended as usual (~ 50 participants). The 3.5 days DivSOL TG meeting followed the 24th IAEA Fusion Energy Conference, organized in San Diego from 8-13 October 2012. It was organized simultaneously with other ITPA TG meetings, which was used to set up joint working sessions, focussed on issues of common interest (SOL width with Pedestal TG, disruptions with MHD TG).

The meeting was structured along the 3 topical areas listed below, under the responsibility of the associated topical leaders :

- Heat fluxes : A.W. Leonard (US), M. Lehnen (EU)
- Tungsten R&D : Y. Ueda (JA), G de Temmerman (EU)
- Material migration, fuel retention, dust : R. Doerner (US), K. Schmid (EU)

The scientific highlights of the meeting included :

- Discussion on the compatibility between the experimental SOL widths scaling, pointing at potentially narrow SOL width for ITER, and the pedestal stability (joint session with the Pedestal TG)
- Comparison of the main disruptions features in a carbon versus a metallic environment (joint session with the MHD TG)
- First attempts to benchmark the material migration codes (DIVIMP/WallDYN package) to model Be migration and associated fuel retention
- Update on controlled tungsten melting experiments, with evidence of droplets ejection during tokamak operation
- Preliminary discussion of the DivSOL report on the ITER divertor strategy, as requested by the ITPA CC

The summary below is divided into the 3 topical areas, summarizing the main topics discussed at the meeting.

Heat fluxes

A first session was dedicated to limiter SOL width scaling, as needed for designing ITER start up limiters. A multi machine database has now been assembled, containing probe data from 7 tokamaks. Extrapolation from the present scaling indicates a 50-60 mm SOL width for a 7.5 MA ITER configuration (larger than previous assumptions from diverted L mode data). However, compared to probe data, IR data, when available, show enhanced heat flux at grazing angle incidence near the tangency point. This effect is significant enough so that when Be melting was observed at JET, it was at the near tangency point. This issue of enhanced near SOL heat flux on limiters should be further investigated.

An update on plasma detachment and associated plasma edge modeling was given. In JET, the changeover from carbon to the ILW has significantly reduced radiated power in the divertor, as anticipated. An unexpected impact of the changeover is the decrease of the pedestal confinement with the ILW, which can be partly recovered with the injection of nitrogen. Concerning modeling activities, dedicated impurity seeding experiments were performed in L mode in AUG and JET to allow benchmarking of the SOLPS N radiation package. Qualitative agreement is found, showing the importance of asymmetries between the inner and outer divertor plasma conditions to reproduce the radiation pattern (cross field drifts are needed in these low density cases).

A joint session was organized with the MHD TG to assess the impact of a metallic environment on disruptions characteristics. Data from metallic devices (JET, AUG, C-Mod) were discussed and compared with carbon references. For unmitigated disruptions, tokamaks equipped with metallic walls exhibits a significantly lower radiated fraction compared to carbon walls. This leads to higher heat loads on the plasma facing components, and Be melting has been observed on the upper dump plates and protection limiters in JET. The disruption timescales are also impacted, with slower current quench for the metallic walls. This in turn influence the runaway e- generation : no runaway e- have been observed so far with the JET ILW. Massive Gas Injection (MGI) allows to recover significant radiation fraction during disruptions : it is now mandatory for operation above 2.5 MA with the JET ILW. MGI is also shown to shorten the current quench and lower the halo currents.

Finally, a joint session was organized with the Pedestal TG to discuss the compatibility between the narrow SOL width currently predicted for ITER from experimental scalings, and pedestal stability issues. The multi machine experimental scaling based on 6 tokamaks data is now consolidated. However, it should now be extended from low density attached conditions to higher density cases, where divertor dissipation is expected to play a strong role (DSOL proposed). EPED analysis of ballooning critical pressure scale length just inside separatrix shows similar scaling as experimental near-SOL λ_q but predicts larger critical width for ITER. More physics based investigations are needed to gain confidence on the empirical scaling derived from present day devices data.

Tungsten R&D

Results from recent controlled tungsten melting experiments were discussed. In AUG, melt ejection events were evidenced but melt droplets in the divertor do not affect the confined plasma. However, detachment of melt debris can cause disruptions. The melt damage area is slowly propagating under repetitive plasma exposure, but results are not yet conclusive at this stage : further experiments are planned. Preparation for the JET melting experiment was also presented, which will allow to gain insight on shallow melting from transients, in addition to experience gained on massive melting from excessive steady state heat fluxes.

A large session was also dedicated to tungsten morphology changes under plasma exposure. The importance of taking into account synergistic effects (combined steady state and transient heat loads, large number of transient cycles, mixed material formation, particle and neutron irradiation) was underlined to determine the damage behaviour of the tungsten components. W fuzz was exposed to plasma guns in PISCES and VDE loads in DIID. It survived exposure to DIIID VDE, as well as moderate loads from plasma guns (up to 0.3 MJ/m^2, equivalent to ITER mitigated ELMs). However, melting of the fuzz top surface is observed for larger heat loads in PISCES. W fuzz was also shown to be prone to arcing, which removes efficiently the W fuzz. The formation of W blisters under very high particle flux was also studied, showing enhanced surface modification by low energy high particle flux D plasma, as expected in the ITER divertor. Finally, a proposal for a multi machine test programme (Gladis, Judith, Magnum) was presented to study the behaviour of recrystallized W under transient thermal shock (DSOL proposal).

Material migration, fuel retention, dust

A first session was devoted to Be migration studies and associated modelling. A self consistent model, including both local erosion / redeposition balance at the wall surface and global impurity migration through plasma transport, is being developed (DIVIMP/WallDYN). It has been applied to model the Be migration experiment at JET, performed in the initial phase of exploitation of the ILW by repeating identical discharges to follow the evolution of the Be sources until steady state is reached. Experimental trends can be qualitatively reproduced by the code only if initial Be coverage of the divertor is assumed. This is thought to be due to the limiter phase. Next steps include therefore modelling of the limiter phase in addition to the divertor plasmas. The code was also used to perform a comparison of impurity production and fuel retention for different material configuration in ITER. 3 configurations were studied : 1) full W divertor / Be wall, 2) full W divertor, Be wall and W upper dump plate, 3) CFC/W divertor, Be wall. Preliminary results indicate that fuel retention is increased by a factor 2 for the configuration 3 (CFC/W divertor) compared to configurations 1 and 2 (full W divertor). Modelling also predicts C migration from the divertor towards the main chamber for case 3. The impact of the W upper dump plate (case 2) is negligible for fuel retention, but could bring an additional W source, although DIVIMP calculations predict small W penetration in the core plasma. Given the large uncertainties in the complex processes to take into account, in particular to assess fuel retention, it was advised to further benchmark the code against experimental data by comparing JET full carbon and ILW configurations, to gain confidence in predictions for ITER.

Simulations of Be droplet ejection in ITER was also carried out using coupled DUSTT/UEDGE calculations, to assess the droplet transport and the plasma background evolution respectively. Results show that the Be droplets are mainly ablated in the SOL, while only a small fraction can penetrate in the core plasma (for large droplets of ~100 μm). According to the simulations, roughly half of the ablated material gets redeposited on the walls, the other half in the divertor. It was proposed to use the recent W melting experiments in AUG to benchmark the code, although W droplet modelling requires additional development, due to the large number of charge states of the impurity.

Concerning material migration in gaps between tiles, the status of the DSOL27 experiment was summarized. An optimized gap shape was identified through modelling, in order to minimize gap deposition. Test targets were manufactured and are now being exposed to plasma in different devices. Preliminary results indicate that deposition is indeed reduced with the optimized gap shape, but further work is needed to assess power handling of the gaps.

Conditioning was also discussed, with a comparison between the JET full C and ILW configuration. After baking and ~200 hours of D_2 GDC, it was shown that the plasma restart is faster in the ILW configuration (5 shots instead of 45 shots to reach the 2 MA plasma target). O and C impurity levels are lower with the ILW, which is attributed to the gettering effect of the Be walls. First results on isotopic exchange with the ILW (GDC and ICWC) were shown, showing a similar accessible reservoir (~10^{22} atoms) for the C and the ILW configuration, while the cumulated wall inventory from fuel retention during plasma operation has been significantly reduced with the ILW.

Further work was presented on fuel retention in irradiated W and fuel removal from Be codeposits. Fuel retention in neutron irradiated W was studied and compared to results where high energy ion beams were used to simulate neutron damage. Neutron irradiation was performed in a fission reactor up to 0.025 and 0.3 dpa. Preliminary results indicate that D penetration in bulk W is more pronounced in the case of neutron irradiated W than for ion beam damaged W. Further work is planned to identify the trapping mechanisms at play. Fuel removal from Be codeposited layers was studied for the baking temperature foreseen in ITER (513 K for the first wall, 623 K for the divertor), as well as for mixed material layers (Be-W, Be-C, Be-O). The desorption behaviour differs depending on the impurity, but in all cases, it is shown that efficient removal requires higher baking temperature than implantation temperature. Modelling of the desorption behaviour was performed using the TMAP code, indicating that thick codeposits could be less efficiently desorbed. The analysis of the Be deposits produced in JET will be important for comparison with lab experiment and code benchmarking.

2. IEA/ITPA multi-machine collaborations

A detailed DSOL report for 2012 and DSOL proposals for 2013 have been provided for the Coordinating Committee meeting in December 2011.

The status of the DSOL experiments is summarized below, in the 3 topical areas of the DivSOL TG. The color coding is the following : red : closed DSOL, blue : ongoing DSOL, green : new DSOL since last report.

Heat fluxes

DSOL-24 Disruption heat loads (E. Hollmann), ongoing

Proposal : AUG, DIII-D, JET, MAST

DSOL-28 Narrow heat flux widths and divertor power dissipation (T. Eich/R. Goldston), new

Proposal : AUG, DIII-D, JET

Tungsten

DSOL-25 Melt layer motion and disintegration, droplet propagation and resulting impact on plasma performance (J. Coenen), ongoing

Proposal : AUG, TEXTOR, JET, LHD (tbc)

DSOL-29 Behaviour of recrystallized tungsten under transient thermal shock (R. Pitts), new

Proposal : JUDITH, GLADIS, MAGNUM-PSI

Material migration, fuel retention, dust

DSOL-23 Efficiency of ICRF Conditioning (D. Douai), ongoing

Proposal : JET, AUG, TEXTOR, KSTAR, EAST

DSOL-26 Marker experiments to study material migration (S. Brezinsek), ongoing

Proposal : JET, AUG, DIII-D, TEXTOR

DSOL-27 Mitigation of fuel accumulation and impurity deposition in the gaps of castellated structures (A. Litnovsky), ongoing

Proposal : TEXTOR, AUG, EAST, LHD, KSTAR and Pilot PSI/Magnum PSI.

Ongoing DSOL, with new experiments planned in 2013, include : DSOL23 on ICRF conditionning, DSOL24 on disruption heat loads, DSOL25 on melt layer motion and impact on plasma operation, DSOL26 on marker experiments for material migration studies, and DSOL27 on fuel accumulation in gaps of castellated PFC structures.

New proposals concern : DSOL28 on narrow heat flux widths and divertor power dissipation, and DSOL29 on behaviour of recrystallized tungsten under transient thermal shock.

3. Future meetings

The 17th DivSOL meeting took place in San Diego, USA, in October 2012. The next meeting of the Div/SOL Topical Group will take place in Hefei, China, in March 2013. A large part will be devoted to discussing the DivSOL report on the ITER divertor strategy requested by the ITPA CC.