7h ITPA meeting on SOL/divertor physics

The meeting was held over the period January 9-11. 2006 at the Donghua University 1882, West Yan'an Road, Shanghai,China

The local coordinator was Professor Jiangang Li of the ASIPP in Hefei. The meeting lasted 3 days and was dwelt primarily on first-wall issues (D retention on sides of tiles, long-pulse fuel retention, surface melting dynamics) and transport (radial and parallel). The agenda is given in Appendix A. There were ~ 46 participants (Appendix B).

There is an executive summary (section 1) followed by a more detailed summary of the talks themselves (section II).

I. Executive Summary

1. D/T retention: T retention in gaps continues to be a strong focus of committee work (e.g. DSOL-xx), results being reported from both dedicated discharges and long-term averaged data. In present machines, fuel retention in gaps is exclusively attributed to co-deposition with carbon and boron. Generally, both carbon/boron and deuterium amounts decrease along the tile sides with distance from the front surface with e-folding lengths of 1-8mm generally corresponding to ~2x the gap width; very small gaps (< 0.5 mm) may violate this relation (JET Be tiles). Although the working hypothesis is that molecules dominate the deposition there appears evidence for ions playing a role based on comparison of toroidal vs poloidal edge deposition profiles; we find deeper deposition, or longer e-folding lengths, for toroidally-running gaps. Initial results of models based on hydrocarbon molecule/carbon atom transport and sticking indicate that the observed deposition patterns must result from high sticking species. Retention in tiles placed in the Private Flux Zone or the outer strike point in detached conditions went down significantly by increasing the surface temperature from RT to about 160C. But of course, this positive result is tempered by the knowledge that higher temperatures also brings higher chemical erosion.

The other part of the session was devoted to long pulse operation. In both Tore Supra and HT-7 the fraction of injected gas that is retained for long periods after a shot can be of order 50% of that injected. This is not true for shorter pulses. In contrast, the D retention in long pulse JT-60 discharges tends to saturate within few pulses. The retention in Tore Supra depends mainly on the LH power injected, and very little on the plasma density, the fuelling method and the ICRH power. Potential explanations of the high retention are codeposition with C (either high D/C fraction or low local redeposition) or long range D diffusion into the bulk. Laboratory experiments aimed at isolating the process show deeper penetration of D in CFC carbon up to 15 μm and amounts of 10^{22}D/m^2 at high fluences. The retention increases like (fluence)$^{0.5}$ but does not increase with temperature indicating that this retention is not due to normal diffusion. Deep retention studies are being pursued as part of a new DSOL proposal (see session 8 summary) involving studies of both C and Mo.

In the fully Mo-cladded long pulse TRIAM device, codeposition of hydrogen with eroded Mo can also contribute to fuel retention but on a much smaller scale compared with C devices. Oxygen strongly affects the structure formation of the deposits which show a larger capability of H retention compared with bulk Mo.

2. Wall conditioning and coating

Conditioning before restart after venting

This basic procedure appears to be used in most machines to reduce impurities and remove H from surfaces. It is performed after baking (temperature ranging from 120°C to 350°C). Conditioning is done by He/D_2 GDC (or ECR DC with swept B for C-Mod). Ne GDC successfully reduced the conditioning time needed before a restart in LHD.

Conditioning during the campaign

Most present day machines rely on systematic conditioning overnight or at the beginning of the run day to lower H/D levels in PFC surfaces. The techniques used are He GDC (AUG, TS on request, JET, JT60U, NSTX) or EC discharge cleaning (C-Mod) with D or He.

Conditioning is also performed between shots to ensure similar wall conditions every shot and lower H/D in the walls (AUG, Textor, NSTX, DIIID – He GDC). The He GDC duration is adapted to the program: a longer duration is required for a better density control (NSTX) while a shorter duration is required to lower the He content of the plasma (AUG). Superconductive devices with permanent B field (TS, LHD) and JET generally do not bother. HT7 uses He pulsed RF DC between shots.

Recovery from disruptions is done with a variety of conditioning methods: He TDC (repetitive plasma breakdowns at low current) in TS, HT7 and JT60U; He GDC is used in AUG; and finally just running natural density shots without gas injection is also used (TS, JT60U). In JET, it is claimed that a fresh Be evaporation also decreases risks of non-sustained plasma breakdowns after disruptions (see wall coating section).

Avoidance of disruptions in TS is accomplished by monitoring the wall saturation and performing He TDC when the wall is nearly saturated.

Conditioning techniques have also been assessed in terms of detritiation efficiency. ICRF cleaning is seen to be more efficient than GDC (Textor, HT7, TS, AUG) except on LHD.

Wall coating

Boronisation is the most widely used coating technique (AUG, C-Mod, TS, HT7, LHD, JT60U, NSTX). Other techniques include siliconisation (HT7, AUG), Li coating (HT7) and Be evaporation (JET).

Boronisation is in general performed with 10-20% B_2D_6 in He GDC or ECR DC (AUG, TS, C-Mod, LHD), building up layers from 30 nm to 200 nm typically. Other compounds are also used ($B_{10}D_{14}$ in JT60, $B(CD_3)_3$ in NSTX). The boron deposition can be non uniform toroidally and localized near the glow electrodes and the injection point (JT6OU, LHD, NSTX) but a toroidal gas plenum helps (C-Mod…). It seems more efficient with hotter walls (TS, NSTX). It is generally performed a few times per experimental campaign. Procedures for boronisation between shots (C-Mod) or during shots (C-Mod, NSTX) are also being developed.

The main effect is a decrease of the O level (AUG, TS, LHD, JT60U, NSTX) and the overall impurity and radiation level (specially in C-Mod where Mo is replaced by B). In AUG, it increases the amount of gas needed for start up (also seen in TS, NSTX) and is essential for low-density operation (advanced scenarios, H mode in natural density). In JT60U, recycling and ELM frequency are affected. Flux consumption and MHD activity are also seen to decrease after boronisation in NSTX. In contrast with C Plasma Facing Component machines, C-Mod shows an enhanced wall recycling after boronisation, and improved confinement attributed to lower radiation while B replaces Mo.

In JET, Be evaporation is performed in general at the beginning of the campaign, and then periodically upon request (1 Be evaporation every 80 shots on average, ~weekly basis) when breakdown is more difficult It is often associated with running advanced scenarios, but because it provides an easier recovery from disruptions, which are more likely for these programs, not a gain in performance. The effect on O is not clear during normal operation, but significant after venting.

3. Start-up and shutdown on limiters

It was found that all experiments showed a similar relation between the edge plasma parameters during ramp-up and those during the flat-top limiter phases of the discharge (Z_{eff}, radiative fractions, separatrix temperatures and densities, scrape-off layer thicknesses, impurity content). Present experimental evidence indicates that the level of input power during ramp-up phases is typically up to a factor of 2 larger than during the flat top, with the absolute value of the resistive loop voltage decreasing with device size. The fraction of radiated power can be kept to very low values (~10-20%) for discharges limited on low Z plasma facing components (or in some cases, also for high Z boronised components). The higher range of the radiated power during ramp-up is determined by the development of MARFEs, which appear at a radiated power fraction of ~ 60% and densities somewhat lower than the corresponding flat top values. Values of the density and temperature at the separatrix, parallel power fluxes and their decay lengths during the limiter ramp-up phases are comparable to the flat top phases of the discharge. Impurity production and plasma contamination during limiter ramp-up phases are comparable to limiter flat top values, which lead to larger impurity concentrations and core plasma radiation than comparable divertor discharges. The achievable density range seems to be the major obvious restriction in operations during the ramp-up limiter phase. This is limited on the high-density side by the radiative collapse of the plasma and on the low-density side by the generation of runaway electrons. Ramp-down phases, on the other hand, seem to pose less critical plasma-wall interaction problems, as the plasma is able to sustain very large radiative fractions in a stable way.

4. Flows: Flows are central to determining impurity transport (determining the core impurity level) and also play a role in material migration and concomitant co-deposition of D/T with impurities (principally C). There are also suggestions that SOL flows may be linked to the L-H power threshold for the L- to H-mode transition. With better understanding of what drives the flows and their dependence on plasma and magnetic topology we may be better prepared to predict core impurity levels, T retention, and the L-H threshold.

In both circular limited plasmas and shaped, diverted configurations the SOL poloidal flow is observed to be from the low field side (LFS) to the high field side (HFS). The flow is over the top of the plasma for lower single null plasmas and down around the bottom for upper single nulls. For circular plasmas the path of the flow depends on the location of the limiter structure but it is still always from LFS to HFS. The LFS to HFS nature of the flows did not change with direction of the toroidal field. However, field direction did modulate the flows at the LFS significantly. In the diverted configurations, the flows between the X-point and targets were directed toward the targets.

Modelling of these flows has only been partly successful. Standard Pfirsch-Schluter and ExB drift effects explain the effect of changing the toroidal magnetic field. However, at the moment, the only mechanism found to increase the magnitude of model flows to the levels measured is to force the radial transport across the separatrix to be poloidally non-uniform, much higher at the plasma LFS (factors of 10 variation poloidally). This creates a pressure imbalance from HFS to LFS and flows to give pressure balance.

These observations suggest two primary foci for future research toward the understanding of SOL flows. First, we must develop the ability to predict the relative importance of the various terms outlined above which drive the SOL flows. This will require more flow measurement capability (multiple poloidal locations) as well as more experiments aimed at delineating the importance of various terms (e.g. by varying toroidal field direction and magnetic topology). The results of such new measurements and experiments can then be used for further verification that the SOL simulations codes contain the correct physics models of these flow drives. Data is of course needed for a variety of conditions. Ohmic and L-mode, such that proper dependence on dimensionless quantities (e.g. collisionality) can be determined giving confidence that model prediction of ITER performance.

5. Melting: Melting of metal surfaces has been observed in several tokamak experiments either as a result of mechanical damage of tiles or in planned high power plasma exposure. Generally, plasma operation in subsequent discharges was not affected. Castellation, or gaps, in the high-Z surface structure has been shown to minimize the flow of the melted material; the melted material did not bridge TEXTOR castellation gaps > 0.5 mm. The positive effect of the castellation is attributed to the lower temperatures along the gap side compared to the surface. This forces the molten metal re-solidify at the gap.

The major processes which drive the molten metal along the surface are jxB forces due to thermo-emission currents (TEXTOR), thermo-electric currents (DIII-D divertor) and the effects caused by the plasma pressure. For relatively thin molten layers the surface tension seems to dominate and keeps the material on the surface. In contrast, a deeper reservoir of molten Li leads to a high loss fraction (DIII-D DiMES, CDX-U) when J x B forces are sufficiently large.

During the session discussion it was pointed out that a simple analytic model could be used to estimate the relative importance of the various forces (see session summary). The existence of a large reservoir of cold electrons near the surface repels thermally emitted electrons back to the surface. This would indicate that plasma pressure, linked to the plasma flux, is the more important mechanism driving molten metal flow.

No clear experimental evidence for vapour shielding in front of a molten metal surface has been reported. A non-linear rise of the radiation with energy is observed in JET above 0.7 MJ/ELM for a carbon target, which indicates onset of enhanced surface material loss. In fact it is even unclear that vapour shielding would have a positive effect on the surface layer losses, since it may increase the plasma pressure in front of the surface which may lead to an increased melt layer loss fraction.

6. Cross-field transport: One of the issues facing the ITER team at the moment is whether to modify the design requirement to limit the ITER divertor re-circulating flux to 10% of the fuelling/pumping rate (the term 're-circulating flux' refers to neutrals escaping the divertor either though leaks in the structure, or through the plasma). The divertor design and projected pumping rate could be considerably improved if the requirement could be relaxed. Estimates of main chamber recycling particle flux are expected to be much larger than this 10% flux; it was generally agreed that this restrictive requirement should be relaxed.

Significant plasma has been found in the shadowed regions of the ASDEX-Upgrade divertor (e.g. under the dome) where it is not expected. Such plasmas could potentially result in excessive dust generation and tritium retention in ITER. One potential explanation for the existence of such plasmas is cross-field plasma being pushed by a 'neutral wind'. It was agreed that further experimental measurements on a variety of tokamaks is needed as well as modelling.

New results from Tore Supra give further support to the model of ballooning like transport at the high-field side of the plasma is the dominant path for plasma crossing the separatrix and filling in the SOL. By varying the limiter configuration and measuring the varying SOL plasma characteristics it was estimated that radial plasma transport was limited to ~30 degrees around the outer midplane. An additional aspect of the Tore Supra study was that, as found in diverted plasmas, the regions near and far from the separatrix are characterised by significant changes in the distribution of turbulence size vs direction: In the far SOL there are significantly more turbulent filaments going away from the separatrix then headed in (skewness of the probability distribution functions) – thus driving outward ion fluxes.

Results from C-Mod pointed towards a theoretical basis for transport in the SOL near the separatrix as well as a potential connection to the gradients in the pedestal just inside the separatrix. Near SOL gradients can be normalized in a dimensionless sense with parameters derived from analytic and 3D modelling of Electromagnetic Fluid Drift Turbulence (EMFDT). Near SOL probe measurements show that local gradients scale like I_p^2 and have a strong dependence on the local collisionality. Initial studies of the pedestal characteristics also show a similar dependence on the EMFDT parameters and gives hope to being able to being able to characterize the entire pedestal through the near SOL; comparisons of gradients across machines requiring some data transference was proposed. This would also facilitate further testing of the ASDEX-Upgrade model showing that the

JET ELM filament structures were successfully modelled with parallel losses and polarization drift. The result is that a high ion temperature in the filaments is expected to strike the main chamber wall in ITER: Ti = 350 eV having implications for sputtering on outer limiters and the upper divertor. Roughly 8% of the initial energy is expected to reach the second separatrix and thus, vessel structures. A lower level of ELM energy, ~2%, would reach the limiter radiaus, r – rsep = 15 cm.

Several presentations on the current state of turbulence modelling were presented. There are two new US initiatives to simulate the edge pedestal and SOL. These are ambitious projects aiming to simulate edge transport in fusion plasmas from first principles, including turbulence and kinetic effects. Finally a theoretical perspective on edge transport suggested that it is now time to start including more physics in edge plasma models, including flows, particle-wall interactions and particle kinetics.

II. Session summaries

Session 1: D/T retention (sides of tiles, long pulse, saturation); Volker Philipps, chair

1.1: 'Deposition in gaps and modelling'; K. Krieger, Contributions from: TEXTOR (A. Litnovsky, A. Kirschner, V. Philipps), DIII-D (D.L. Rudakov (UCSD), W.P. West , C.P.C. Wong), ITER Team (G. Federici, G. Strohmayer), IPP materials research (W. Jacob, T. Schwarz-Selinger), ASDEX Upgrade (V. Rohde)

Formation of carbon deposits and fuel inventories in gap structures of plasma facing components have been studied in dedicated discharge scenarios in TEXTOR, ASDEX Upgrade and DIII-D. Generally one observes a decrease of both carbon and deuterium amount at the side faces from the gap entry into the gap with scale lengths of 1-2mm for a typical gap width of 0.5mm. Models of the underlying processes based on hydrocarbon molecule/carbon atom transport and sticking have been developed and benchmarked against the experimental data. First results show that the observed deposition patterns must result from high sticking species. However, the particular production mechanism of these molecule radicals and the mechanism responsible for the strongly decreased D inventory growth rate at elevated surface temperature have not yet been conclusively identified.

1.2: 'Analysis of C-Mod tiles for B/D co-deposition on tile sides'; D. Whyte

Ion beam surface analysis at UW-Madison was used to measure tile gap deposition on molybdenum tiles from Alcator C-Mod. Four poloidal locations were assessed: the upper divertor horizontal plate, the inner wall above the midplane, the nose at the entrance to the inner divertor and the lower divertor vertical target plate. Deuterium codeposition with boron was found in tile gaps with typical e-folding distance of film thickness ~ 1-2 mm. Boron films ~ 1 micron were found near the plasma-facing surface. Deuterium/boron ratios were found to increase from 1-2% near the surface to ~5-10% several mm down the gap, qualitatively indicating softer film deposits further down the gaps. Boron is introduced in C-Mod by boronizations, which is carried out in a EC discharge with a weak toroidal field.

Since the boron gryoradius is ~ 10-20 mm << 0.5 mm tile gap size, gap deposition must be through neutral deposition rather than ions during boronization. A simple geometric model was used to test this hypothesis by predicting the shape and magnitude of boron gap deposition. The magnitude of deposition is roughly consistent with the relative area of the gaps to the total plasma-facing area of the tiles: ~ 2-4%. This indicates that boronizations themselves are playing an important role in setting the tile gap inventory. The profile of film thickness down the gaps matched the neutral deposition model well at the upper divertor, in the lower inner divertor and in poloidal-running gaps at the inner wall. In contrast, toroidal-running gaps at the outer divertor and inner wall did not agree well with the expected profile. This result indicates that ionic deposition during plasma discharges, when the ion gryoradius << tile gap, is also playing a role in gap deposition. In general it is difficult to correlate plasma fluence with tile gap deposition, since two very different processes (boronizations and plasma discharges) are competing in C-Mod to set the tile gap inventory.   A positive result is that a large majority of film growth and D retention is within the first few mm of the plasma-facing surface, raising the hope that optical cleaning techniques such a flash-lamps and disruption cleaning can access and de-tritiate such deposits in ITER.

1.3: 'DiMES Results on the temperature dependence of re-deposition/co-deposition'; *M.E. Fenstermacher, D. Rudakov*

Comparison of carbon re-deposition and deuterium co-deposition on heated vs room temperature DiMES samples shows that the re-deposition down gaps between surfaces can be reduced by factors of 2 – 4 and co-deposition reduced by factors of order 10. However, the heated samples also show increased net erosion from the plasma facing surface. Identical samples containing a radially oriented gap were exposed to identical LSN L-mode detached plasma near the OSP. One sample was at room temperature and the other was heated to 200 C by a heater built into the DiMES sample. The carbon redeposition profile down the gap had similar fall-off length in the two cases but the magnitude of the re-deposition was down a factor of up to 4 in the heated sample. The deuterium co-deposition in the layers along the gap sides also showed similar fall-off length in both samples with the magnitude reduced a factor of ten for the sample at 200 C. This would imply that the co-deposition in the gaps between ITER tiles would be significantly less than can be extrapolated from room temperature results since the ITER tiles will be at least 200 C during plasma operation. However, the experiments also showed a 3 nm/s erosion rate on the plasma facing side of the sample for conditions (L-mode, detached divertor) that give no net erosion with room temperature samples. This would extrapolate to 1.2 um/ITER shot, an unacceptably high erosion rate. Further experiments with heated vs. room temperature samples exposed to attached ELMing H-mode plasmas are needed to improve the prediction for ITER. In addition, a separate set of experiments showed that the re-deposition rate on diagnostic mirrors exposed to the private flux plasma was substantially reduced when the mirrors were at elevated temperature (between 80 and 140 C in the experiments). This is a very positive result for ITER since it must use diagnostic mirrors without shutters for many of the diagnostics, and the mirrors can be easily maintained at elevated temperatures if needed.

1.4: 'Deposition and Fuel Inventory in Castellated Structures at JET and TEXTOR'; M. Rubel

Until recently, the only large-scale castellated structure used in fusion experiments was the Mk-I divertor of the JET tokamak. It was operated first for ~60,000 s with carbon fibre composite tiles (CFC) and then for ~20,000 s with castellated (6x6 mm with 0.6 mm deep groove) beryllium blocks. The essential recent results may be summarised by the following: (i) significant co-deposition of carbon and deuterium has been observed up to a few cm deep in the gaps between the tiles, both in the CFC and Be divertors; (ii) in the CFC gaps the fuel inventory has been found to exceed that on plasma facing surfaces by up to a factor of 2; (iii) in the gaps between the Be tiles the inventory was reaching 30% of that found on plasma facing surfaces; (iv) in the narrow castellated grooves of Be tiles the co-deposited fuel was around 2 % of that found on top surfaces indicating that the gap width may be decisive for material transport into such structures; (v) fuel retention on Be tiles was associated with the co-deposition of carbon originally eroded from the main chamber wall. Based on the comparison of co-deposition on side surfaces of Be and CFC tiles implications for next-step fusion devices is discussed. One may suggest that in a machine with non-carbon walls in the main chamber the material transport and consequential fuel inventory in the castellation would be reduced.

A tungsten macro-brush limiter brased to a copper base was exposed to a number of high-power shots at the TEXTOR tokamak (up to 35 MW m^{-2}, *see talk by A. Kreter*). Studies of the composition in the castellated gaps are summarized by the following: (i) no fuel inventory has been detected; (ii) formation of copper droplets occurred at a distance of 2-3 cm from the overheated limiter surface; (iii) the presence of W-O containing deposit has been detected. The formation of W-O "leaves" in the gaps needs further analysis and clarifications.

1.5: 'Recent results on D retention in Tore Supra long discharge'; E. Tsitrone

In 2005, the Tore Supra long pulse database has been extended to higher density – higher power (Ip = 0.6 MA, nl up to 5 10^{19} m^{-2}, P_{LH} = 3 MW, P_{ICRH} up to 4 MW, duration up to 1 minute). The retention rate is still constant during the discharge as in lower density-lower power shots, with no sign of saturation. In the range explored, it is shown to depend mainly on the LH power coupled to the plasma, and very little on the plasma density, the fuelling method and the ICRH power. After short pulses (< 30 s), the D recovery after the shots and during 1 night of He glow discharge compensates roughly the D trapped during the discharges, so that the integrated D inventory remains small. For long pulses (> 30 s), the D recovery after the shot is only a small fraction of the D trapped during the discharge, and the integrated D inventory becomes significant. Since the main mechanism responsible for retention was thought to be codeposition of D with eroded C, deposited layers have been analysed. However, the D inventory found in the machine (D/C ratio from 1 to 10%) is still lower than the D inventory deduced from particle balance. Progress has also been made in closing the C balance, showing that the net C erosion source deduced from visible spectroscopy and modelling is coherent with the net erosion measured on the toroidal pump limiter (TPL) as well as with the net deposition observed in the machine. However, if codeposition dominates, the experimental D retention rate is only compatible with a high C erosion source (not coherent with the TPL erosion measurements) and/or a high D/C ratio (not seen in post mortem analysis, with D/C ratio from 1 to 10%).

This might indicate that another mechanism than codeposition could play a role in the retention process. One candidate is bulk diffusion of D into the porous CFC material of the TPL. Indeed, recent lab experiments have confirmed that after long plasma exposure to high D fluence, significant D quantities can be found much deeper than the implantation range in the CFC material. In particular, retention is seen to increase as $(\text{fluence})^{0.5}$. Preliminary calculations, applying this result to the recycling flux on the TPL, show that this effect could be significant. However, more lab data are needed to asses this mechanism (effect of exposure time versus fluence, incident ion energy, sample temperature etc).

1.7: 'Long term retention in Triam', Sakamoto

Codeposition of hydrogen with eroded metal makes a substantial wall pumping. Oxygen strongly affects the structure formation of the deposits which has higher capability of H retention than the bulk Mo.

1.8: 'Deuterium Retention in Carbon Fibre Composite Materials'; J. Roth, V.Kh. Alimov, A.V. Golubeva, E. Tsitrone, Ch. Brosset, R. Doerner

Deuterium retention in carbon fibre composites (CFC) NB31 and N11 was studied by irradiation with 30 to 200 eV D ions, both from ion beams (IPP Garching) and plasma devices (PISCES-A, Inst. Phys. Chemistry Moscow) using thermal desorption spectrometry and $D(^3He,p)^4He$ nuclear reaction analysis in a resonance-like technique. It is found that at irradiation temperatures in the range from 323 to 723 K the amount of deuterium trapped in the CFC materials increases with the ion fluence, Φ. No saturation was reached as observed in pyrolytic graphite. At room temperature, the deuterium retention increases proportionally to $\Phi^{0.5}$. Depth profiles show that saturation in the retention occurs only within a near surface layer equivalent to the ion range. The increase in total retention is accompanied by an increasingly long profile tail extending beyond 14 μm with D concentrations above 10^{-1} at.% at fluences above 10^{24} D/m^2. The diffusivity of deuterium in the bulk of the carbon fibre composites as derived from the depth profiles depends only weakly on the irradiation temperature and lies between 10^{-16} and 5_10^{-15} m^2 s^{-1} for irradiation temperatures in the range of 323 to 723 K.

First estimates show that this deep retention in CFC can account for an important fraction of the observed D inventory in Tore Supra after long pulse discharges.

1.9: 'Carbon deposition and tritium retention at plasma shadowed area - comparison of TFTR, JET and JT-60U', T. Tanabe –

Tritium retained in carbon deposited layers in gaps of plasma facing tiles are measured by tritium imaging pate technique and compared for TFTR, JET and JT-60U. The amount of carbon deposited layers and their tritium retention are well correlated and two different origins of carbon deposition are clearly distinguished. One is deposition of carbon and hydrocarbon ions caused by gyration of eroded carbon species at the plasma facing surface, showing short penetration length of less than 10 mm from the top surface. The deposition of the ionic species is larger for tiles of which plasma facing surface were eroded, and the penetration depth is independent on the gap width. This could be significantly reduced by decreasing the tile gap width below gyro-radius of the ionic species.

The other is deposition of molecular or radical species with longer penetration length. The deposition of the molecular species is small for erosion dominated tiles and the deposited amount is quite dependent on the position of the gap. The gas flow to the direction of pumping duct strongly enhances the deposition, and tile sides which are plasma shadowed but facing to a pumping duct have significant deposition, such as the inboard side pumping slot of JET Mark-IIA divertor and the outboard pumping slot of the W-shaped divertor of JT-60U. The deposition of molecular species could be suppressed by appropriate divertor design.

1.10: 'Thermal instability of neutral-saturated walls, (20th anniversary of the INTOR report)'; S. Krasheninnikov

Interactions of plasma with saturated wall can cause thermal instability (see also [S. I. Krasheninnikov, INTOR, USSR Contributions to the Phase IIa, IAEA, Vienna, December 1985]) of wall temperature resulting in massive desorption of gas from the wall. In can lead to the formation of MARFE or even disruption and can be crucial for a long pulse ITER operation. Somewhat similar phenomena are seen on many dev ices including JT-60 and HT-7. Rather rudimentary theoretical model accounting for plasma-wall interactions and the heat balance of the wall predicts the time scale of such instability below 1 sec for JT-60 like tokamak. More experimental data and more refined theory are needed.

1.11a: 'Deuterium and tritium deposition on gaps between TFTR tiles'; *C. H. Skinner, C A Gentile, (PPPL) W R Wampler (SNL), K Sugiyama, T Tanabe (Kyushu-U) and the PPPL tritium group*

For TFTR there is general consistency between measurements of deuterium retention, tritium retention and tritium gas balance and modeling. Of the 45% total deuterium fuel retained, 9% was deposited in tile gaps especially in erosion areas. The deuterium on the sides of TFTR tiles extended to a depth 6 mm e-folding length as measured by NRA. Imaging plate measurements of tiles exposed during the DT campaign showed tritium on tile sides to extend with a short (2mm) and long (>10 mm) decay pattern.

1.11b: 'Time resolved measurements of deposition in NSTX'; *C.H. Skinner, H. Kugel, L. Roquemore (PPPL)*

In 2005 pulse-by-pulse deposition was recorded at three locations on the NSTX vessel wall during 17 weeks of plasma operations and during boronization and showed a complex pattern of erosion and deposition. Preliminary analysis of some of the data show strong deposition on the first discharge of a run day with erosion or deposition on subsequent discharges. The erosion/deposition at the upper divertor increased during double-null plasmas (as compared to lower single null). A comparison of high performance lower single null plasmas and ohmic plasmas on the same day showed a changeover from deposition in short duration low energy plasmas to erosion after long high energy plasmas. The results suggest that deposition during the startup/rampdown phases of the discharge may be significant and should be modeled.

1.12: 'Wall retention and saturation experiments in HT-7', Y. Yang

It's observed on HT-7 that both high and low Z impurity concentration could impair LHCD. Another important limit is the uncontrollable density rise caused by either intended gas puffing or enhanced recycling. At lower plasma current and electron density, with careful displacement control, plasma could sustain all these disadvantageous factors for as long as 300 seconds. These low parameter plasma provides a high energy fluence condition for PSI studies. Particle balance shows that about 60% of the fuelled gas is retained relatively permanently. More retention happens in longer pulses. Recycled H ranges from 10% to 80% of the released gas after plasma termination, depending on the wall condition. Release of the retained D molecules could be the major limit for the longest pulses. Efforts are made for higher parameters of long pulses by improved position control and heat load handling.

1.13: 'Discussion summary'; V. Philipps

There is common opinion that deposition in gaps is largely specific to PFC materials that can form molecules (e.g. C and B), although details of the deposition mechanism, e.g. what is due to direct ion deposition (so does not require molecules) and what to molecular related neutrals are not known so far. Differences between poloidal and toroidal gaps indicate that ions are at least non-negligible. The JET data show a surprisingly small retention in small grooves compared to data from other experiments; more work is needed with small (fractions of a mm grooves). The JET Be tile divertor results indicate the importance of local C production (through erosion) for gap deposition. Increased temperature can mitigate or prevent carbon deposition, but can also lead to larger net erosion of C (due to chemical sputtering) on the front surfaces, as seen in DIII-D.

The mechanism is of D retention in long pulses compared with short pulses with similar accumulated fluences was also discussed, with no clear answer. The physics of deep D penetration in CFC is not compatible with classical assumptions of diffusion. It was agreed that long-range bulk retention in CFC should be measured more intensively under tokamak conditions. This will be the subject of the the new IEA/ITPA collaboration DSOL-19 (see session 8) which will address the deep retention both in lab and tokamak for both graphite and Mo.

Session 2: 'Wall conditioning and coating'; E. Tsitrone, chair

2.1: 'Wall conditioning in ASDEX Upgrade'; V.Rohde

Typical oxygen concentrations in AUG are 0.4%, so O is not an outstanding impurity. The dominating influence of boronisation is the modification of wall pumping. A fresh coating pumps 5e20 particles at during plasma start up. Due to the high gas puff during normal scenarios (typically 2e22 at/s), the wall coating affects mostly scenarios with low densities: plasma start-up, advanced scenarios and natural density. Whereas for the advanced scenarios the wall pumping during limiter start-up is important, the natural density is influenced by modification of the ELMs.

Between shots HeGD is needed for plasma start up after high density shots, disruptions and after disruption mitigation gas puffs.

ICRF-DC is very effective to clean the surface in the presence of magnetic fields; a sweep of the plasma position is needed to clean PFC´s at different positions.

2.2: 'Conditioning in Tore Supra'; P. Ghendrih

The Tore Supra wall is mainly the bottom toroidal limiter made of CFC (typically 15% of the total wall surface), and stainless steel. All components are actively cooled with a base temperature that is usually set between 120 °C and 150 °C.

Restart after shutdown

D_2 and He Glow discharge conditioning at 200°C until the CO production measured by mass spectrometers drops below 4 10^{16} molecules/s/A (at 200°C the CO production is observed to be 6 times larger than at 120°C).

Plasma operation

As for start-up a significant effort has been devoted to run plasmas with as little modification of the wall as possible in order to avoid both uncontrolled changes of particle content as well as well as messing up the wall structure with He glow discharges between shots and aborted current ramp-ups.

Controlled plasma breakdown : Premagnetisation plasmas have been achieved just prior to the actual shots. They consist of plasma breakdown at given loop voltage and have led to very different plasmas. These are characterised by their density, plasma current and duration. Very short plasmas characterise unsaturated walls when the density is small, as well as fully saturated wall when the density is large. The prefill and breakdown loop voltage were determined automatically according to these properties.

Operation with controlled wall particle content : During ergodic divertor operation large particle fluxes were required to sustain the core densities that led to large wall particle loading. The wall status was monitored by the plasma current peaking evaluated by the internal inductance l_i early in the discharge, typically for edge an edge safety factor of 9. If the l_i value exceeded 1.2, the following couple of shots would be performed at natural density, i.e. with no gas injection. This would ensure a non-disruptive operation of Tore Supra at high density.

Post disruption plasma breakdown :

* IC conditioning was successfully experienced on Tore Supra in terms of conditioning. However, it led to a major failure in the ICRH system due to arcing in the capacitors related to the high neutral pressure required for the conditioning procedure. No technological solution has been implemented to overcome this problem so that in practice, IC conditioning is not used.

* Plasma breakdown conditioning is the routine conditioning procedure after a strong disruption. It consists of a series of about 50 breakdowns at low plasma current, typically 50 kA (less than for a Taylor discharge), lasting a couple of seconds followed by about 8 seconds of pumping. This procedure is maintained for 3 minutes and, for a reasonable oxygen content of the plasma, ensures a successful breakdown.

Boronisation

Boronisation in Tore Supra is infrequent and only performed in case of a too large oxygen content. After boronisation, the boron line radiation vanishes in a few shots while the reduction of the oxygen content, and the associated reduction in carbon, appear to last for much longer times. A coating effect monitored by the reduction in Fe radiation is found to last for some 1500 s of plasma and it is claimed that the aging of this property is governed by plasma ramp-up and plasma ramp-down.

It has been well documented that boronisation strongly modifies the deuterium recycling by allowing for larger postdischarge outgasing and consequently much larger wall particle uptake with delayed saturated wall behaviour.

2.3: ‘Roles of wall-conditioning and coating in Alcator C-Mod’; Bruce Lipschultz

The type of wall-conditioning used in C-Mod utilizes an ECR discharge to clean the walls. It is primarily used after a vacuum break to remove adsorbed H when the walls are hot (120C) but is also used prior to each runday. Wall-coating is done using the same ECR discharge but with a 90%He, 10% diborane mixture to deposit a coating of 150-200 nm. A single wall coating per campaign seems to suffice to minimize most impurities (O, C, F, Mo) from almost all surfaces. In certain areas (parts of the outer divertor and outer limiters) the B coating is quickly worn off in 10s of shots (~ 50 MJ of injected energy) and Mo becomes the dominant impurity (n_{Mo}/n_e approaching 10^{-3}) and reducing confinement essentially to L-mode. Boronization can be required for optimal core performance. Boronization also has an added benefit of reducing impurities during the initial plasma formation and reducing runaway production and damage to outer limiters.

2.4: 'Summary on JET Be-evaporation wall conditioning'; V. Philipps

JET does not perform any intershot wall conditioning like GDC during daily operation. After plasma disruptions, plasma break down becomes difficult and start scenarios have been developed for break down after disruptions by adopting a proper gas fuel programme. Overnight GDC in D_2 followed by short He GDC is regular. This procedure leads to highly reproducible plasma conditions, in particular during the X-point divertor phase. "Bad" wall conditions appear mainly as breakdown problems in the current ramp up phase. Non-sustained break downs after disruptions increase unless Be evaporation is done in a regular manner and this is the main rationale for Be evaporation. The role of Be evaporation after major vacuum breaks, leaks etc. is not fully explored presently but under investigation. After most of the accidental leaks or vessel ventings, plasma restart is done by conventional conditioning, such as baking at 300C (earlier times) and at 200C (presently), followed by GDC in D_2 and He , typically 24-48 h. In recent events plasma restart Be-evaporation was done directly after baking and GDC cleaning without plasma operation first, with good success. In the average (C6-C14 campaigns) Be- evaporation is done each of 80 plasma discharges. The spectroscopically measured ratio of Be/C fluxes shows a clear increase after fresh Be-evaporation but falls off in typically 3-10 discharges.

2.5: 'Wall Conditioning Study in LHD'; N. Ashikawa NIFS

A) Comparison of glow discharges using different gas

- Ne-GDC was effective to reduce the conditioning time on the initial phase of the experimental campaign
- He-GDC seems to be the origin of heavy damage and large adsorptions of gas on the wall

- Hydrogen removal rate of He-GDC is 10 times larger than He-ICC, therefore most effective to control gas retention and recycling gas from the wall in LHD

B) Effective coating method of boronization

- The coating area depends on the distance to the GDC anode
- Top surface region on boron film is also effective to getter oxygen, therefore re-deposition layer prevents boron films gettering.
- Coating area of boronized wall is needed as the trapping area of re-sputtered impurities from the first wall by GDC
  - ✓ Case of only a small fraction of partial coverage of full torus area (= not full cover) with boronized wall effectively contributes to an oxygen gettering

2.6: 'Boronization and conditioning in JT-60U'; N. Asakura

(1) Boronization (3-6 in past years, recently 2-3 times in a year): during ~50 shots, boron was observed in core plasma, and suppression of sputtering at carbon plates resulted in low core carbon content. Then, boron content decreased to ~ 0.5%, and the carbon content increased to constant level of ~ 2.5%. Boronization (with 70g $B_{10}D_{14}$) suppressed the core oxygen content $\leq$ ~ 1.0% in 400 - 500 shots. Frequent boronization every 200 shots (with 10 - 20g $B_{10}D_{14}$) successfully kept the core oxygen content $\leq$ ~ 1.0%. Confinement improvement for ELMy H-mode was caused by (i) recycling reduction (*more than two*), (ii) smaller particle exhaust by ELM, (iii) ELM became more regular, (iv) reduction of radiation (lower n_e, Z_{eff}). For ITB plasmas, confinement was not sensitive, but some boronization effects on impurity and density control were observed.

(2) He-GDC after ~15 shots (1day) : 3-9h He-GDC can exhaust 0.5-1.6x10^{23}D(T/H). Removed D number is comparable to the level of the first wall saturation. Wall retention is generally compensated by the He GDC D recovery, except from large gas puff series.

(3) He-TDC between shots: 7min He-TDC can exhaust ~2.5x10^{20}D(+T+H). D removal from the first wall is small and local. It is used to remove deuterium and impurities on the wall after disruption for plasma build-up. When wall saturation appeared in the middle of discharge, confinement factor was reduced and plasma build-up was difficult in the following shot. Main TMP and NBI cryo-pump exhaust during about 60 min. was enough so far for plasma breakdown and build-up. Large particle exhaust between shots will be required.

2.8: 'NSTX Wall Conditioning Experience – Implications for ITER'; H. Kugel, **C.H. Skinner**, L. Roquemore, D. Mueller

In NSTX He GDC is necessary to reduce recycling for density control, and boronization is necessary for peak performance and easy H-mode access. The implications for ITER are uncertain as the lack of diagnosis of wall composition in present experiments impedes fundamental understanding. The experience with JET ITER-like wall project, and long pulse superconducting tokamaks should help however no long pulse (>100s) tokamak exists or is planned with ITER PFC materials. A comprehensive review of ITER's risks and consequences is needed to establish the priority of wall conditioning R&D.

2.9 Discussion of conditioning issues - E. Tsitrone

There was a general agreement that conditioning is needed for plasma operation, in particular for the start up phase (recovery after disruptions, reproducible wall conditions), but not on whether it is needed to improve plasma performance (specially confinement) or not.

In particular, results from present day machines seem difficult to extrapolate as recycling neutrals are not expected to penetrate into the main plasma in ITER, while they can reach the pedestal region and affect confinement in most present devices. Moreover, some present devices run at room temperature, which could also affect recycling.

It was underlined that conditioning was not recognized as an important topic judging from the length of the corresponding chapter in the new ITER physics basis to appear. The reply was that this reflects the fact that there was no significant progress in this area since the last ITER physics basis, not the fact that it is not important.

It was recognized that there is a lack of fundamental physics understanding in the way conditioning affects plasma operation (for instance which surfaces do we need to clean / cover to start up after disruptions or high density operation ?), due in particular to a lack of dedicated diagnostics and operational time. The knowledge on these issues is rather empirical.

It was recognized that conditioning is likely to be needed for start up after disruptions or accidental events. Cleaning techniques similar to present TDC but compatible with ITER should be developed. Recovery after mitigated disruptions involving large impurity gas flow should also be assessed.

The following ITER specific issues were raised :

- Non saturation of CFC : new evidence from long pulse actively cooled devices (TS, HT7) and lab samples exposures to high D fluence suggests that CFC can accumulate D inventories significantly higher than graphite (bulk diffusion). Should this process be taken into account when estimating T inventory in ITER (see also session 1 on retention) ?
- Saturation of the O gettering capability of the Be wall : should a Be evaporator be contemplated for ITER ? Results are expected from the JET ITER like wall project, but too late for ITER schedule ?
- Compatibility of a Be wall with oxidation techniques ?
- Do we need to cover specific impurity sources at given locations ? Results are expected from AUG W wall and JET ITER like walls.
- Mixed layers properties ?

The following actions were suggested :

- Make up a table gathering information about wall conditioning and coating techniques as well as operational conditions (temperature, PFCs material …) in present day devices to derive similar trends and opposite behaviours.
- Check with the designers of the RF heating system if the design also allows RF conditioning to be performed : J. Li
- Check with the diagnostic ITPA group if diagnostics are compatible with conditioning (shutters …) : C. Skinner (first answer : the diagnostic group seems to be aware of the problem

Session 3 - 'Limiter startup and shutdown'; A. Loarte, chair

3.1: 'Simulations of ITER limiter start-up and preliminary power load assessment'; G. Federici.

The analysis of the current ramp-up phase in ITER is an important part of the design of the Be start-up limiter (maximum steady-state power flux capability of ~8 MW/m^2), in particular of its performance and engineering margins. In particular, it is necessary to estimate for a range of start-up scenarios the SOL power flow during ramp-up and the fraction of this flow that reaches the limiter(s) and the spatial structure of the power flux onto the limiter. An initial study of the limiter power loads has been carried out by the IT supported by experts from NIFS, IPP-Garching and EFDA. From this validation exercise, physics modelling assumptions for ITER, in particular on energy confinement and impurity radiation, have been derived. Modelling for ITER shows that, for the reference low density ramp-up (n/n_G~0.2, i.e., <$1x10^{19}$ m^{-3}), the power to the limiter is moderate (<3 MW). Further work remains to be done to understand the feasibility of such low-density ramp-up scenarios in ITER, in particular with respect to generation of slide-away/runaway electrons. High density (n/n_G ~ 0.4-0.5) ohmic ramp-up scenarios end up in a radiative collapse and additional heating is required during the limiter ramp-up phase in this density range. The power to the limiter for the cases with additional heating analysed could increase up to ~ 4 MW. Given the remaining uncertainties, a ±50% uncertainty factor in the present estimates might be assumed, leading to an upper boundary for the limiter power of ~ 6 MW.

The assessment of ITER limiter power fluxes is in progress and has required the development of tools to provide a detailed mapping of the magnetic field in the ITER limiter SOL. A significant proportion of SOL field lines can miss the limiters in their first poloidal turn causing the SOL flux tubes to become very long ($L_c >> 2\pi$ Rq). These leads to the existence of poloidal structures in the SOL with large connection lengths, which can cause local power loads on the limiter edges and/or the first wall. The 3D edge transport code, EMC3 is being applied for ITER start-up limiter configuration. Also a "conventional" limiter model has been developed and applied in support

3.2: 'Analysis of JET ramp-up/down limiter phases'; A. Loarte

Experimental measurements of the SOL power flow and plasma parameters have been analysed for the ramp-up phases of JET ohmic discharges with plasma current in the range 1-7 MA limited by carbon discrete limiters, carbon belt limiters and beryllium belt limiters. The SOL plasma parameters and decay lengths during ramp-up are comparable to those during the steady-state phases, with similar dependencies on global plasma parameters such as I_p and n_e. The SOL power flux is larger during the ramp-up phases than in steady state, with typical resistive loop voltages of 1.5 V instead of the typical 0.8 V for steady-state. Radiated power levels during ramp-up phases are typically lower that in high density steady-state phases and limited by the development of MARFEs to ~ 60% of the ohmic heating. Plasma impurity contamination for ramp-up phases is comparable to that of steady-state phases and correlated with the total level of radiated power and density in a similar way than for steady-state conditions (scaling by Behringer, IAEA 1986). Direct extrapolation of the JET results to ITER indicate that an ohmic ramp-up scenario at low density (same n_e/n_G in both devices ~ 0.2) will be viable in ITER with low power flow to the limiter (~ 3MW) and low radiated power fractions (20% of the ohmic heating), but not at higher densities (n_e/n_G ~ 0.4) due to the radiative collapse of the plasma. Runaway production during ramp-up in JET is typically observed at densities of ~ 0.5-0.7 10^{19} m^{-3} (n_e = 0.11-0.18 n_G) but the physics basis required to extrapolate this evidence to ITER is still under development.

3.3: 'ASDEX-Upgrade ramp-up limiter studies'; A. Kallenbach

Dedicated pulses have been run in ASDEX Upgrade to compare plasma startup on HFS and LFS limiters. No pronounced differences have been observed in performance and flux consumption. Measurements of the radial power width on the low field side limiters by means of thermography showed values around 5 mm, comparable to those obtained in high power H-modes. Quite high tungsten concentrations (several 10^{-5} to 10^{-4}) are found in the core plasma during the early limiter phase. This is in particular also the case for start-up on LFS limiters, which are predominantly carbon (10 C, 2 W). Since the plasma-wall interaction with the tungsten-coated HFS limiter is much smaller during LFS ramp-up, the high W concentration in the plasma is probably explained by a higher relative W penetration probability with larger distance between plasma and inner wall.

3.4: 'Alcator C-mod ramp-up/down limiter studies'; B. Lipschultz

There are a large range of conditions found during the limiter start-up phase on C-Mod. When the walls are cleaned of boron the Z_{eff} is high (~ 3) dominated by Mo, Fe and a combination of low-Z impurities (C, O, F). This is accompanied by a high radiated fraction (~ 50%) which apparently forces the initial breakdown to low densities resulting in a significant population of runaway electrons. The latter hit the outer limiter during the shot leading to melting of those surfaces. When boronisation is applied (see other summary) the impurity levels drop (Z_{eff} approaching 1) and the radiated fraction drops to ~ 5-10%. The discharge breakdown and initial start-up can then be optimized to rid the discharge of runaway electrons and concomitant melting of the outer limiters.

3.5: 'TEXTOR ramp-up limiter studies'; A. Kreter

Typical I_p ramp-up duration in TEXTOR is 0.5 – 0.8 s. Plasma starts at the inner limiter and is moved to the central position during ramp-up. There is a moderate increase of the power load of the main ALT-II limiter due to increased ohmic heating power in the ramp -up phase. The load of the inner limiter does not increased significantly. Pre-pressure value (~10^{-4} mbar) is crucial for the successful start-up. If it is too low, plasma experiences MHD instabilities, if it is too high, the breakdown cannot be achieved. The radiated power fraction is 20-30% during ramp up (60-80% in flat-top). There is a slight increase of impurity content by 10-20% during the inner wall phase. No enhanced production of runaways during high loop voltage phase is observed in TEXTOR. In the ramp down phase usually detachment occurs. Saturated wall condition results in the hard landing of the plasma.

3.6: 'Plasma start-up under low loop voltage conditions in HT-7'; J. Luo

Start-up experiments in HT-7 have shown that boronisation can decrease the electric field in this phase by a factor of almost 2, contributing to a longer flat top length. Similarly, LHCD and ICRH have been used effectively to reduce the loop voltage during ramp-up compared to ohmic heating but, in particular for LHCD, its reliable use requires careful adjustment of the density evolution during ramp-up to avoid runaway generation.

3.7: 'JT-60U ramp-up limiter studies'; N. Asakura

The plasma in JT-60U was limited at the inner wall during ramp-up (typically 1-3s). In wall cleaning case, limiter period extended to ~ 7s: inner wall was swept. During the limiter period, V_l ~ 2V for dI_p/dt ~ 0.4MA/s, and V_l~0.8-1.4V for dI_p/dt~0.06MA/s. Larger n_e (ne/nGW~0.25) for the limiter (strong pumping by divertor plates and cryo-pumping was not affected). Total recycling flux, impurity and radiation power during the limiter period were larger than those for the divertor period, i.e. P_{rad}~0.6MW (~50%, in limiter) compared to P_{rad} = 0.3-0.4MW (divertor)

3.8 Discussions for limiter startup session'; A. Loarte

The evidence of all experiments was compared and the following preliminary conclusions were extracted on various topics of importance for the ITER limiter ramp-up phase:

a) The separatrix values and SOL decay lengths for temperature, density and power for during the limiter ramp-up are similar to comparable steady-state limiter (and divertor) plasmas, once details of the limiter configuration are taken into account (based on JET and ASDEX Upgrade evidence). The extrapolation of this to the ITER limiter plasmas requires an adequate edge modelling effort, similar to that performed for the reference diverted regime, possibly guided by basic turbulence SOL plasma simulations.

b) The ohmic input power during ramp-up phases can be up to a factor of 2 higher than during steady state phases and there is a correlation of the ohmic heating level with current ramp rate, as expected from simple arguments, with the largest rates leading to the larger input powers (based on the JET, JT-60U and TEXTOR evidence). The radiated power level during ramp up is usually higher than for the flat top, particularly for devices with high Z plasma facing components. Radiated power fractions of up to ~ 50% can be sustained during plasma ramp-up (and larger during ramp-down) while maintaining a stable plasma (based on the JET, JT-60U, TEXTOR ASDEX Upgrade and Alcator C-mod evidence). On the basis of this, the level of radiated power during start-up will be restricted to ~ 50% of the ohmic input power. The ohmic heating power magnitude is seen to scale inversely with device size (for the same level of plasma current) as expected from simple physics arguments, but a precise scaling to ITER remains to be done.

c) Impurity production and/or screening is poorer during start-up limiter discharges than in diverted phases (ASDEX Upgrade, Alcator C-mod) but similar to the flat top of limiter discharges (JET and TEXTOR). Both Alcator and ASDEX Upgrade show that the outer limiter plays a major role on the interaction of the plasma with plasma facing components, even when the plasma itself is limited on the inner wall. This could be associated with enhanced transport at the outer midplane of the tokamak or with poorer screening of impurities generated at the outer midplane. The levels of high Z impurities during the start-up phase are affected by boronisation, being lower for a boronised device. On the basis of the evidence presented it is not possible to predict the value of Z_{eff} during the start-up limiter phase beyond that it will be lower than 4. Dedicated studies of impurity generation and screening in limiter discharges are required to progress further in this area.

d) The range of operating density during ramp-up is limited on the high side by the radiative collapse (around ~60% radiative fraction from JET, Alcator C-mod, JT-60U and TEXTOR) and on the low side by runaway generation (JET, Alcator C-mod and HT-7). In general, poor conditioning and presence of high Z impurities in the initial phases of the discharge are counter-productive for runaway generation, as it forces the ramp-up to be carried out at low densities (to avoid radiative collapse) and leads to significant runaway generation. Operation with low Z plasma facing components (and/or low Z coatings) allows operation at high densities and lower radiative fractions during ramp-up (10-20%) as shown in JET Be operation and Alcator with boronised PFCs, which is favourable for the use of Be at the ITER limiter. The minimum densities required to avoid runaway generation are in the range of n_e = 0.11-0.18 n_G for JET and n_e = 0.25-0.35 n_G for Alactor C-mod. How to extrapolate these results to ITER requires the development of a physics-based scaling on which to carry out this extrapolation.

In general, the ramp-down limiter phase of discharges seems to be much more robust than the ramp-up phase in all devices, as expected from the current profile stability. In particular, the levels of radiated power, plasma purity, plasma density, limiter power load, etc., that can be achieved in present experiments indicate that this phase in ITER will require a much less challenging plasma-wall interaction control than for the ramp-up phase.

Despite this encouraging initial evaluation, it is clear that, in order to provide a proper quantitative evaluation of the plasma edge characteristics and power loads to the limiters in ITER, dedicated/coordinated experiments across the present tokamaks and data analysis are required. Such coordinated experiments are being planned within the ITPA activities for 2006/2007.

Session 4 – Flow Characterization and Modeling —- ***Chair - M.E. Fenstermacher***

4.1: 'Multi-fluid modeling of large parallel plasma flows in the tokamak SOL' - *A.Yu. Pigarov,* ***S.I. Krasheninnikov****, and B. LaBombard*

Based on UEDGE simulations of edge plasma transport in the single-null magnetic configuration of C-Mod, DIII-D, NSTX tokamaks, we obtain the following "big picture" of the origin of nearsonic flows in the SOL. The strong ballooning-like transport causes large cross-field plasma fluxes at the outer side that results in HFS/LFS asymmetry of plasma parameters. A key component is intermittent convective transport (e.g. blobs), which brings plasma density, energy, and momentum into the far SOL region. Since plasma in divertor regions connected to the far SOL is weak, it does not build up a high pressure due to recycling processes (as plasma near separatrix does) and does not cause stagnation of plasma flow. Therefore, plasma ejected into the far SOL on the LFS, flows almost freely into the inner and outer divertors with Mach about unity. The tokamak magnetic configuration also affected the parallel plasma flow. The HFS/LFS asymmetry in magnetic field causes variation of the cross-section of effective magnetic tube in the SOL and the corresponding change in the parallel flow velocity. Combined effects of cross-field transport asymmetry, configuration, and classical drifts should be studied.

4.2: 'Modelling of the Bx∇B independent component of parallel SOL flow using the ESEL turbulence code' - *W. Fundamenski*

Edge-SOL plasma turbulence modelling using the 2-D reduced fluid, electrostatic code ESEL has been presented for TCV and JET. The code is described in O.E.Garcia et al, PPCF 48 (2006) L1. It was suggested that sub-sonic advection to increased pressure associated with plasma blobs, then parallel Mach numbers may be estimated as 0.5 x fraction of the time that local pressure exceeds twice the time-averaged value. This estimates ballooning like parallel flows with $M_{||}$ up to 0.1 in TCV and JET, which can partly account for the measured flows.

4.3: 'SOL flows in Tore Supra' - *J. P. Gunn*

We present original measurements that demonstrate the universality of many phenomena that are observed in X-point divertor tokamaks, especially concerning the ion flows. In Tore Supra, as in the JET tokamak, surprisingly large values of parallel Mach number are measured on top of the torus midway between the two strike zones. These flows, when interpreted using 1D or 2D fluid models, imply that a significant fraction of the particle source is poloidally localized near the outboard midplane. The poloidal variation of the Mach number of the parallel flow was studied by moving the contact point of a small circular plasma onto limiters at different poloidal angles. By moving the plasma contact point around the poloidal section, it is possible to modify the edge flows at the probe location. This is somewhat like a poloidal exploration of the parallel flow profile. Asymmetric flows are observed in cases that should be symmetric from simple geometrical considerations, in perfect agreement with the prediction of our simple 1D model, and confirmed after the fact by 2D modelling with the TECXY multifluid code. We show that the measurements can only be explained if the effective source includes a region of incredibly enhanced radial convection with poloidal extent ~30° centered on the outboard midplane. If no object obstructs the parallel motion of the plasma that gets ejected onto open magnetic flux surfaces, which is the case when the contact point lies on the inboard midplane, the SOL expands to fill all the available volume between the LCFS and the wall (the density decay length varies between 10 and 20 cm). The mechanism that causes this spectacular expansion appears to be favoured by the existence of long field lines that pass unobstructed across the outboard midplane. This is demonstrated by moving the plasma from HFS to LFS contact : the SOL becomes very thin and a thick vacuum region separates the plasma from the wall. In general, the necessary condition for the observation of large plasma density and flow far from the LCFS is that the flux tube that intersects the Mach probe pass unobstructed across the outboard midplane.

4.4: 'Flow characterisation in ASDEX Upgrade' - *A. Kallenbach*

Two manipulator systems equipped with Langmuir or Mach probes are occasionally available in ASDEX Upgrade, one situated about 0.2 m above the midplane, the other in the lower divertor about 0.1 m below the X-point. Plasma flow measurements have been done for Ohmic and low power H-mode pulses. Generally, quite strong flows with peak values M=0.5 are observed in the main SOL directed towards the inner divertor for lower SN plasmas with ion grad-B drift towards the X-point. The peak of the flow is close to the separatrix and the Mach number decreases outward towards the limiter. With the divertor probe, flows with M~1 directed towards the targets are seen in both inner and outer divertor leg. During ELMs, the flow reverses in the outer divertor.

4.5: 'Characterization and physics of flows' - *Bruce Lipschultz*

It is clear both on C-Mod and other tokamaks that there is a strong flow from the outside edge to the inner divertor. The inboard SOL flow results from C-Mod show that the flow accelerates continuously moving poloidally around the plasma reaching Mach 1 levels at the inner midplane. The lack of inner SOL plasma during double-null indicates that plasma arrives there by flowing from the outer midplane region. The implied ballooning source crossing the separatrix at the outer midplane region is narrow poloidally and peaked there. We believe this mechanism dominates the flows there (thus independent of B) while ExB and Pfirsch-Schlutter are much more important at the outer midplane (where the ballooning-transport driven flows are small). The same ballooning-transport driven flows occur for plasmas limited on the inner divertor – clearly consistent with the mechanism and the absence of the x-point. With an x-point or inner-wall limited location in the location favorable for flows in the co-current direction the H-mode threshold is lower than for the opposite case/ The argument is made that the ballooning-transport driven flows are ultimately controlling the H-mode threshold.

4.6: 'Indications of SOL Flows in DIII-D' – ***M.E. Fenstermacher****, M. Groth*

Data from the tangentially viewing visible emission cameras show evidence for SOL flows in DIII-D. The technique for detecting SOL flows is to monitor the locations of emission from multiple carbon lines during CH4 injection and look for poloidal shifts of the emission between different charge states. Three experiments that used CH4 injection from the toroidally continuous outer baffle plenum at the top of the vessel have been analyzed: 1) a low power L-mode with the ion Bx∇B drift toward the lower divertor, 2) the same plasma except with the Bx∇B drift up, and 3) a high density ELMing H-mode plasma with the ion drift down. Poloidal shifts of the higher charge state emission toward the inner SOL and divertor compared with the lower charge state emission are observed in both L-mode cases. The shifts do not qualitatively change with reversal of the toroidal field direction leading to the immediate conclusion that ExB forces are not the dominant drive for the flows leading to the shifts. The direction of the shifts is consistent with the flow being driven by pressure asymmetry along the field lines from the outer midplane toward the inner SOL over the top of the LSN plasma crown. This is consistent with asymmetry of cross-field radial transport at the outer midplane due to ballooning instabilities. However, in the H-mode case the shifts between the emission profiles of the different charge states are not seen. A possible explanation is that for the steeper gradients in the near SOL in H-mode and the reduced outer midplane cross-field transport, the ExB and pressure drives could be comparable in the SOL at the top of the plasma in H-mode. Further experiments in H-mode with CH4 injection and both directions of BT are needed to verify this hypothesis.

4.7: 'Summary of the Discussion' – ***M.E. Fenstermacher***

The discussion of this session focused on 1) observations of SOL and divertor flow that were common to several of the talks presented in the session, 2) questions that remain at this stage of research into SOL flows, and 3) possible future experimental and theoretical work that could contribute to answering some of the questions. The talks in this session described flows in the SOL in both circular limited plasmas and shaped, diverted configurations. In all cases presented, the SOL poloidal flow is observed to be from the low field side to the high field side. The flow is over the top of the plasma for lower single null plasmas and down around the bottom for upper single nulls. For circular plasmas the path of the flow depends on the location of the limiter structure but it is still always from LFS to HFS. Almost all results presented were from ohmic or L-mode confinement plasmas. The LFS to HFS nature of the flows did not change with direction of the toroidal field. In the diverted configurations, the flows between the X-point and targets were directed toward the targets. This data is consistent with outer midplane dominated cross field transport and the resulting pressure asymmetry in the SOL being the dominant force driving the flows. This is also consistent with neoclassical ballooning theory and the observations that for limited plasmas when the limiter is on the HFS a thick SOL forms on the LFS but when the limiter is on the LFS the SOL formed on the HFS is much thinner.

These observations suggest two primary foci for future research toward the understanding of SOL flows. First, we must develop the ability to predict the relative importance of the various terms driving the SOL flows, including the pressure drive due to outer midplane radial ballooning transport, guiding center drift terms such as the ExB and Bx∇B drifts, and diamagnetic drift forces. This involves both verifying that the SOL simulations codes contain the correct physics models of these flow drives, and validating those models against experimental data that can be obtained in ohmic and L-mode plasmas. Second, we must develop understanding of how the flows will be different in ELMing H-mode operation with a high density, low collisionality SOL as in ITER. The discussion at the close of this session concluded that several experimental techniques should be used to address these development needs. First all experiments investigating flows should try if possible to do identical repeat discharges with both directions of the toroidal field so that the B-field dependent guiding center drift effects can be isolated from drive terms that do not depend on field direction. Second, flow experiments should be done in both L- and H-mode with density scans to isolate the effects of temperature gradients and collisionality. Finally, development of diagnostics that can measure SOL flows in a hot H-mode SOL is needed for complete understanding and validation of the predictive codes for flows.

Session 5 - Melting

Summary of Session 5: Melt layer stability and vapour shielding

Melting of metal surfaces has been observed in several tokamak experiments either as a result of mechanical damage of tiles or in planned high power plasma exposure. Generally, plasma operation in subsequent discharges was not affected. Castellation of the high-Z surface structure succeeded in keeping the melt layer loss fraction small and has demonstrated a very positive effect on the performance. No bridgeing of a castellation gap was observed in TEXTOR for gaps > 0.5 mm. The positive effect of the castellation is attributed to the lower temperatures at its sides, where the molten metal re-solidifies.

The major processes which drive the molten metal along the surface were jxB forces due to thermo-emission currents (TEXTOR) and thermo-electric currents (DIII-D divertor), in addition to effects caused by the plasma pressure. For relatively thin molten layers the surface tension seems to dominate and keeps the material at the tiles. In contrast, a deeper reservoir of molten Li seems to be lost to a high fraction (DIII-D DiMES, CDX-U) when radial J x B forces are sufficiently large.

The ratio of the forces due to plasma flow, F_p, and jxB force, F_jxB, have been estimated by S.Krasheninnikov as follows: F_p/F_jxB = (J_p/J_c)(rho_s/Delta) where J_p is the plasma flux to the surface, J_c=j/e is the net flux of charged particle carrying the current j, and rho_s is the effective ion gyro-radius, and Delta is the depth of molten layer. In practice J_c can be much smaller than it would follow from the Richardson expression. The reason for this is a large charge of cold electrons accumulated near the surface and repelling thermally emitted electrons back to the surface.

No clear experimental evidence for vapour shielding in front of a molten metal surface has been reported. A non-linear rise of the radiation with energy is observed in JET above 0.7 MJ/ELM for a carbon target, which indicates onset of enhanced surface material loss. In fact, as pointed out by A. Loarte, it is even unclear that vapour shielding would have a positive effect on the surface layer losses, since it may increase the plasma pressure in front of the surface which may lead to an increased melt layer loss fraction.

5.1 A.Kreter "Tungsten macrobrush limiter under high heat load in TEXTOR"

ITER-like macrobrush tungsten test limiter was exposed to high power loads of up to 35MW/m^2 until melting. FEM modelling of temperature distribution shows a good agreement with pyrometry measurements. Both experiment and modelling indicate the importance of the shaping of castellation cells (e.g. roof-like geometry) for the surface temperature distribution. Thermo-emission current was identified as the driving force (jxB) of the melt layer motion. Significant material re-distribution within single cell was observed. No bridge formation occurred for gaps >0.5 mm. No integral material loss occurred in the single melting shot. However, the re-solidified layer has cracks on its surface, which might lead to the loss of loosely bounded structures in subsequent shots. To clarify this point, a further exposure of the limiter in TEXTOR is planned.

5.2 DiMES experiments with Li samples in PF and OSP regions in DIII-D – *M.E. Fenstermacher, D.G Whyte*

Experience with molten lithium and aluminum on DiMES samples in DIII-D has shown that the surface stability or integrity of the liquid is the 0th order issue determining the viability of the melt-layer and of the nearby plasma. The experiments with lithium used a 0.5 mm thick layer in a stainless steel cup imbedded in the DiMES sample. This type of sample was exposed to three different conditions in separate experiments: 1) the outer strike-point region of L-mode plasmas with 0.45 MW/m^2 heat flux, 2) an accidental locked mode from a low density L-mode producing 5 MW/m^2 heat flux on the sample, and 3) ELMs that produced transient heat fluxes of 10 MW/m^2 on the sample during H-mode . In each of these cases the Li was melted and then ejected from the sample holder on DiMES. In each case the evidence pointed strongly to JxB forces on the liquid, due to currents flowing through the melted Li from the SOL to the vacuum vessel, as the driving force for the ejection. Molten movement of the Li across the tile face can lead to non-conforming surface structure as the melt layer “runs into” solid surfaces, such as the lip of the DiMES cup. Such distortion, particularly with a highly electrically conducting material like lithium in a graphite environment, can concentrate the path of intercepted parallel currents through the lithium, leading to molten ejection away from the tiles. For these thick samples the JxB forces dominated over any surface tension that might hold the sample together. In contrast, the Al sample showed no signs of JxB motion even after exposure to a disruption producing 200 MW/m^2 heat flux to the sample. There was clear evidence of melting but most of the Al remained on the sample after exposure in the form of large droplets. In this case the Al sample was much thinner (10 μm) and it appeared that surface tension forces dominated over the JxB forces. The combination of these results leads to several conclusions: 1) if JxB forces dominate on melt layers then large ejection of the layer into the surrounding core plasma can result, 2) since the ITER core can not withstand any appreciable contamination by tungsten, the most serious melt layer loss could be from melting of the W surfaces just outside the divertors and ejection of droplets of W toward the separatrix during ELMs. Further experiments with castellated tungsten samples on DiMES could provide valuable data on the likelihood of this scenario for ITER.

5.3: ‘EU-RF collaboration on material damage under transient loads in ITER: melt layer stability and vapour shielding’; A. Loarte

A collaboration has been established between the EU and the Russian Federation to study ITER-relevant material damage under transient loads (ELMs and disruptions) by experiments (in Russian plasma-guns and EU e-beam facilities) and modelling. Experiments have shown that a macro-brush structure is effective in preventing large melt layer erosion following ELMs by limiting the mobility of the molten layer. On the other hand, the loading of the macro-brush edges causes the on-set of melting at lower energy densities than those on the flat surface, because of the large impact angle of the plasma on these exposed edges. Optimisation modelling studies show that an inclination angle of ~ 1.5-2^{o} (macro-brush surface with respect to a flat surface) is optimum for macro-brush edge shadowing and for maintaining a large effective surface area for the energy deposition. With this design, the degree of surface roughening after repetitive ELM loads of the macrobrush target (caused by melting and melt layer displacement) can be decreased by more than an order of magnitude with respect to a flat macrobrush design. Experiments in plasma guns on flat W-macrobrush targets at power levels (on the flat surface) of 0.5 → 1.0 MJm^{-2} and Δt = 500 μs are in reasonable agreement with the modelling results.

5.4: ‘Experimental measurements of C/Be damage under ELMs in JET’; A. Loarte

Experiments at JET on a carbon divertor (MkIIA and MkIIGB-SRP) show indications of enhanced material damage under ELMs when the main plasma Type I ELM energy loss is ~ 1MJ. Repetitive Type I ELMs of this size can be achieved at JET when operating at medium/high plasma current (I_p > 3 MA). The estimated ELM energy flux to the divertor and its timescale in these conditions approaches the range in which significant carbon ablation by the ELM pulse is expected (> 20 $MJm^{-2}s^{-1/2}$). The onset of enhanced ELM carbon target damage has been quantified at JET by the integral of the radiation spike caused by the carbon influx during and after the ELM. For ΔW_{ELM} < 0.7 MJ, the ELM-caused radiated energy spike (ΔW_{ELM}^{rad}) is proportional to ΔW_{ELM} and amounts to $\Delta W_{ELM}^{rad} \sim 0.25\ \Delta W_{ELM}$, as one would expect from carbon erosion by sputtering with the carbon production being proportional to the plasma flux. For ΔW_{ELM} > 0.7 MJ, this proportionality factor changes abruptly to $\Delta W_{ELM}^{rad} > 0.5\ \Delta W_{ELM}$, indicating the onset of an additional mechanisms for carbon impurity production by the ELM. This onset is correlated with the divertor target surface reaching 2500 ^{o}C but a clear correlation between the enhanced radiation and the spectroscopic carbon influxes remains to be evaluated. Occasional ~ 1 MJ ELMs in the MkI Be target caused significant melting of the Beryllium target, which is expected to occur for ELMs with ΔW_{ELM} ~ 0.5 MJ. Despite the melting, the castellated structure of the target was effective in preventing significant melt layer displacement, in agreement with experiments on W macro-brushes in plasma-guns.

Discussion of Session 5: Melt layer stability and vapour shielding

A. Kallenbach – IPP Garching

In the discussion of session 5, a number of participants reported about melting tile occurrences in their machines. Partly, this was recognized not until a vent after the experimental campaign. A disturbance of subsequent discharges after the melting event rarely occurs.

An overall positive rating of tile castellation was made for tungsten and molybdenum tiles. A sufficiently wide gap width (5 mm) turned out to be important to avoid bridging of gaps by molten metal.

Various physics mechanisms were discussed which lead to melt layer loss. These include a direct effect caused by the plasma pressure in front of the surface and jxB forces due to electric currents into the tile. The latter comprise thermo-emission currents and thermo-electric currents driven by temperature differences along the SOL. Variations in the strengths of such forces on the molten layer is made responsible to the very different observations (no melt layer loss up to almost complete loss) in different experiments.

Except a non-linear radiation rise with ELM energy observed in high power JET ELMs, no evidence was reported for vapour shielding. It is not clear whether this is a desirable effect at all, since the vapour cloud leads to increased plasma pressure in front of the tile which promotes the loss of molten layer.

Discussion - Forces to the plate: **j**x**B** versus plasma flows: After some algebra one can show that the ratio of the force on the molten layer of the depth Δ due to the plasma flow, F_p, to the $\mathbf{j}\times\mathbf{B}$ force, $F_{\mathbf{j}\times\mathbf{B}}$, can be written as follows: $F_p / F_{\mathbf{j}\times\mathbf{B}} \sim (J_p / J_c)(\rho_s / \Delta)$, where $C_s = \sqrt{T/M}$, J_p is the plasma flux to the surface; $J_c = j/e$ is the net flux of charged particle caring the current, and $\rho_s = C_s / \Omega_{Bi}$ is the effective ion gyro-radius. Both J_p and J_c can be found from experimental data and used for the estimation of the ratio $F_p / F_{\mathbf{j}\times\mathbf{B}}$. We just notice that in practice J_c can be much smaller than it would follows from the Richardson expression. The reason for this is a large charge of cold electrons accumulated near the surface and repelling thermally emitted electrons back to the surface.

Session 6 - Radial Transport - Chair, A. Leonard

6.1: 'Recirculation Requirements for the ITER Divertor'; C.Lowry (ITER)

The reduction of bye-pass leaks from divertors was been linked to improved confinement, and reduced H-mode threshold in Asdex and PDX. There is also good reason to believe that it could lead to increased wall sputtering, and reduced He compression along the SOL. However, dedicated experiments on JET and C-mod have not supported these observations, and in C-mod closure of the bye-pass leak had negligible effect on the main chamber pressure. For these reasons the present ITER physics basis specifies an upper limit on the recirculating flux of 10% of the fuelling/pumping rate.

The present ITER design does not meet this requirement. Therefore it has been proposed to include a duct between the divertor and the pump. Such a solution also needs sealing between cassettes to avoid leakage through this path. With this technically messy solution it should be possible to reduce the leakage to 20% of the fuelling, but only at the cost of a factor of 3 reduction in the pumping. A similar level could be achieved by judicious design of the clearance between the cassette and vessel, without a reduction of the pumping. However, when the pump is throttled the leakage remains the same, and therefore its fraction of the fuelling increases, and one has 3 times the leakage at the same fuelling. It is believed that the criteria as specified is not correct, but that the recirculation flux should be compared to the total main chamber fluxes, including that due to the plasma interaction with the wall.

6.2: 'Plasma Transport Under the Dome' Sergei Krasheninnikov (UCSD)

The effects of the „neutral wind" (see Krasheninnikov and Smolykov, PoP **10** (2003) 2030) can result in a strong plasma cross-field transport in the shadows of divertor region including beneath the dome. As a result, plasma density under the dome can be high and cause parasitic effects including wall erosion/deposition, dust generation, pumping alteration, etc…. Both experimental data and physics/computer models of plasma in shadows are needed.

6.3: 'Summary of talk on edge turbulence and transport in Tore Supra'; J. P. Gunn, CEA Cadarache

Starting in 2005 we measured regularly the radial profile of ion saturation fluctuations with probes in the SOL of Tore Supra (a large limiter tokamak with circular cross section) over a wide range of discharge conditions (density scans up to and above (with pellet injection) the Greenwald density limit, q scans, B field scans, heating power scans, and limiter geometry scans). At this meeting we present some preliminary results. In general we observe similar behaviour as compared with X-point machines such as TCV. Near the last closed flux surface, the probability distribution function (PDF) of the ion saturation current is nearly symmetric, but in the far SOL it can become positively skewed. According to 2D fluid modelling by various groups, skewed PDFs are the signature of intermittent avalanche-type transport. The fact that we observe them in divertor and limiter tokamaks indicates that we are dealing with a univeral phenomenon. Our measurements of parallel flows during the moving plasma experiment, which were discussed in the session on SOL flows, led us to conclude that enhanced radial transport, concentrated near the ouboard midplane, is responsible for the broad density and temperature profiles that are observed if a large gap exists between the outboard LCFS and the first wall. If this gap is closed, the parallel flow becomes nearly stagnant and the SOL becomes very thin (density decay length ~2 cm). If this enhanced radial transport is caused by a turbulent phenomenon, one might expect to see an effect of the PDF. We performed an experiment to investigate this question in a standard bottom-limited plasma. Additional outboard modular limiters were placed in contact with the LCFS and then gradually moved outward on a shot-by-shot basis. Radial profiles of density, temperature, parallel Mach number, and ion current fluctuations were measured on each shot. As the modular limiters were removed, the SOL width and the parallel flow increased, indicating that the main source is concentrated near the outboard midplane, in agreement with the moving plasma experiment. At the same time, the radial profiles of the PDFs, which were only moderately skewed at all locations on the first shot, became more and more non-uniform, with the strongest skewness far from the LCFS. The triple correlation between wide SOL, large parallel flows, and bursty ion current signals far from the LCFS, lead us to postulate that the main SOL source is due to intermittent transport events concentrated near the outboard midplane. The width of the SOL is mainly determined by the amount of space that is available between the LCFS and the wall on the outboard side of the tokamak.

6.4: 'Correspondence of edge and SOL transport to turbulence models'; Bruce Lipschultz, B. LaBombard

Studies of the near SOL region in C-Mod show a connection of local gradients to plasma current, collisionality, and q. A connection is made to Electromagnetic Fluid Drift Turbulence (EMFDT) and turbulence models through the parameters (α_{MHD}, α_D). A third controlling parameter appears to be flows as higher co-current flows at the outer SOL lead to higher pressure gradients (and have lower H-mode thresholds). Initial studies of the pedestal characteristics also show similar dependence on the EMFDT parameters and gives hope to being able to being able to characterize the entire pedestal through the near SOL.

6.5: 'ELM limiter interaction on JET and ITER'; W. Fundamenski

Evidence for ELM filament contact with main chamber wall on JET have been presented, including wide angle D alpha views, missing ELM power from divertor, outer limiter probes, turbulence transport probe, retarding field analyser, divertor IR camera in outer gap scans, vaccum vessel displacement measured using mechanical sensors. Radial velocities, densities and temperatures of ELM filaments have been reported. A model of ELM filament evolution based on parallel losses has been developed and shown to successfully reproduce JET experimental data. The same recipe was used to predict the ELM filament evolution in ITER. At r – rsep = 5 cm (location of secondary separatrix, and hence Be wall contact), the peak ELM filament quantise were predicted as Ti = 350 eV, Te = 140 eV and ne = 1.2e19 m-3. Roughly 8 % of the initial energy is thus expected to be deposited on the vessel wall (with a roughly 2% reaching the limiter at r – rsep = 15 cm).

6.6: 'U.S. Edge Simulation Initiatives'; A. Leonard

Two new major edge simulation programs have been initiated in the U.S. The first of these is a U.S. DOE SciDAC program, The Center for Plasma Edge Simulation (CPES) lead by CS Chang of New York University. The aim is to develop a predictive, first principles, integrated edge simulation framework applicable to existing and burning plasmas. The simulation will be based upon a 5-d kinetic PIC code, XGC-NT, which will embody self-consistent neoclassical, turbulence and neutral physics with a realistic wall shape and flux geometry. The eventual goal will be a predictive time dependent simulation, including ELMs, of the pedestal and SOL plasma. The other major initiative is the Edge Simulation Laboratory (ESL), led by LLNL with contributions from GA, UCSD, LBNL, Lodestar, PPPL, and CompX. This effort will be based upon a 5-d gyro-kinetic simulation of the pedestal and SOL plasma using continuum methods. Neoclassical and neutral physics will also be included. This effort should be considered complimentary to the CPES program

6.7: 'Theoretical Perspectives on Edge Radial Transport'; P. Ghendrih

The physics of the edge pedestal and SOL presents uncertainties in several areas for predicting ITER performance. These include the input power necessary for achieving H-mode, the pedestal transport barrier width, scaling of the SOL plasma parameters and neutral behavior. Turbulence issues include the level of SOL transport in relation to Bohm gyroradius scaling, the parallel wavelength of the turbulence, the degree of ballooning characteristics and X-point effects as well as spreading of the SOL turbulence both towards the far SOL and towards the core plasma through the pedestal region. The nonlinear aspects of transport can lead to bifurcations in a number of areas including radiation, divertor detachment, density limits and plasma flow. Tracking these rapid changes of plasma state in actual experiments or even simulations is difficult due to the numerous non-linear aspects involved Models for predictive capability need to include all the relevant physics. The inability of current codes to predict, or model the observed plasma in the private flux region is a clear example that not all of the important physics is yet included. It is now time to start including other known physics into the codes, including flows, impurities and particle-wall interaction, cross field transport and kinetics. The new challenge for the edge plasma modeling requires a renewed effort in this activity which can be as demanding as going to a new generation of codes.

6.8: 'Radial transport discussion summary'; A. Leonard

The discussions following these presentations concentrated on the path forward to better characterize the physics of edge radial transport. A number of devices have reported similar features of convective transport in the far SOL from propagating filaments. Models for these filaments are improving and matching data. The near SOL and pedestal plasma is now a very important region for characterizing the flux into the far SOL. The fluctuations in the near SOL appear ballooning in nature as evidenced by the Tore Supra data where transport was very localized to the outboard midplane. The fluctuation and transport levels appear consistent with fluid-drift models. Further additions needed for edge transport simulations include the H-mode transport barrier and neoclassical effects. It was pointed out that the dimensionless scaling approach is very useful in identifying the important physics in edge models. It was pointed out that a study of JET data indicated the heat flux width was consistent with neoclassical ion transport. An issue remaining is the fraction of power carried across the separatrix separately by the ions and electrons and how this division will scale to ITER.

A second item for discussion was the wall flux due to ELMs. The ELM filaments that propagate to the wall appear to be well fit by a model of polarization drift with parallel losses. The filaments and the associated transport appear very similar to that observed in L-mode and between ELMs, though they are larger in size. It was pointed out that in ITER, operation with large ELMs will not be allowed. This implies a need to characterize the wall fluxes due to small ELMs. SOL and divertor measurements of ELMs also play an important role in developing models of ELM evolution within the pedestal. Important measurements include fast measurements of electron temperature simultaneously in the SOL and divertor and fast pedestal profile evolution during an ELM.

Session 7: 'Next meeting and high priority tasks'; Chair - B. Lipschultz

7.1 Next meeting and topics

A number of potential dates for the 8th meeting were discussed. Unfortunately, because of PSI, it does not make sense to have an ITPA meeting near in time. The general thought was to put it off till the Fall. Since the meeting was just in Asia, the hope was to have it in the USA.

However, visa issues caused a number of people to not support this. The backup was to have it in Toronto, Canada sometime around APS (October 30 - November 3, 2006, Philadelphia, PA), probably afterwards. A second possibility would be in Europe after the PFCM11 meeting – October 10-12. It appears that the meeting will be in Toronto after APS.

We discussed dates for the 9th meeting, following the Toronto meeting. The obvious time is around EPS in June.

We also discussed what topics should be emphasized for the next meeting. The following are what were discussed. As we get nearer to the meeting this list will have to be revisited as it is dependent on what can be accomplished as well as who can attend.

D/T issues
    Removal rates and applicability to various materials and hidden areas
    Long-range D retention in solids?
    Measurement of surface erosion and redeposition
Modelling (code-code comparisons, benchmarking codes)
Spectroscopy of hydrocarbons in low-T plasmas
C^{13} experiments and modelling
Density limits (H-L…) and physics thereof
Disruption & ELMs – mitigation, wall loads….
Properties of mixed materials
Properties of the SOL during ELM-mitigated and small-ELM regimes
High-Z experience
ICRF conditioning evaluation

7.2 High Priority tasks

The ITPA coordinating committee has requested that the list of high priority tasks be modified to reflect goals that can be accomplished in 1-2 years. The Pedestal group has successfully modified their list by taking the more general headings (as we have now) and adding sub-bullets that can be accomplished on a more specific time scale - see bottom of this message. Below are a draft set of high priority tasks that are being discussed among our group.

These need to be shortened to a subset that will fit on a viewgraph and that the ITPA Coordinating committee can understand.

1. Improve measurements & understanding of plasma transport to targets and walls to better predict heat loads and effects on the core plasma
    - Code-code comparisons with no drifts
    - High neutral density (ITER limit) benchmarking of interpretive code physics to current experiments
    - New measurements of near and far SOL transport with connection to pedestal transport & stability limits
    - Exploration of the role of radial transport in driving SOL flows
    - Investigation of the effect of wall main chamber recycling and core impurity levels

- Effect of connection length variation and radial transport on heat deposition

2. Understand the effect of ELMs/disruptions on divertor and first wall structures
   - Development of more precise measurements of where power goes in disruptions/ELMs (new IR measurements, better time resolution)
   - Update current specification to ITER team of the levels of power levels and areas of deposition during ELMs and disruptions

3) Improve understanding of Tritium retention and development of efficient T removal methods.
   - Comparison of tile-side D retention level physics across tokamaks
   - Characterize macroscopic D retention in tokamaks and laboratory expts –
   - Comparison of various D/T removal techniques for single and mixed materials as well as variation of geometry
   - Develop localized measurements of material erosion/redeposition and D/T retention

4) Understand how conditioning and operational techniques can be scaled (or not) to future fusion devices
   - Compare startup/rampdown experience across experiments and evaluate the influence of limiter configuration on SOL properties
   - Investigate the effect of conditioning methods on plasma start-up and overall discharge development, including operational limits
   - Implications of a metal wall (no coating) for startup, fuel retention, density control and core impurity levels

Session 8: 'IEA/ITPA collaborations'; Chair - N. Asakura

The following reports and proposals of the IEA/ITPA multi-machine experiments were reported by spokes-persons (some missing numbers were previously completed):

DSOL-1	Scaling of Type I ELM energy loss — report and proposal were presented by Loarte.
DSOL-2	Injection to quantify chemical erosion -- Report and proposal were presented by Philipps on behalf of Brooks.
DSOL-3	Scaling of radial transport -- report and proposal were presented by Lipschultz.
DSOL-4	Comparison of disruption energy balance in similar discharges and disruption heat flux profile characterisation -- Report and proposal were presented by Loarte.
DSOL-5	Role of Lyman absorption in the divertor -- Report and proposal were presented by Lipschultz on behalf of Lisgo.
DSOL-7	Multi-machine study on separatrix density and edge density profiles -- This work was completed in 2004.
DSOL-8	ICRF Conditioning -- Report and proposal were presented by Ashikawa.
DSOL-9	Carbon-13 injection experiments to understand C migration – Report and proposal were presented by Philipps on behalf of Stangeby.
DSOL-11	Disruption mitigation experiments – Report and proposal were presented by Lipschultz on behalf of Whyte.

DSOL-12 Oxygen Wall Cleaning
Report and proposal were presented by Philipps on behalf of Stangeby.

DSOL-13 Hydrogen/Deuterium codeposition in gaps of plasma facing components
Report and proposal were presented by Krieger,

DSOL-14 Multi-code, multi-machine edge modelling and code benchmarking.
This is not a joint experiment. Since we had a session of this benchmark work in last meeting (Tarragona, 2005, 7), this will be discussed when new results are obtained.

DSOL-15 Inter-machine comparison of blob characteristics
Report and proposal were presented by Lipschultz on behalf of Terry.

DSOL-16 Determination of the poloidal fueling profile –
This new plan was proposed in last IEA/ITPA committee in Nov. 2005. This work (spokesperson is M.Groth) will start between DIII-D and AUG. We did not discuss this time, and will be discussed in next meeting.

Additional 3 new proposals were presented and discussed,

DSOL-(17) Cross-machine Comparisons of Pulse-by-Pulse Deposition –
This proposal was presented by C. Skinner. It will start between NSTX and ASDEX-U so far.

DSOL-(18) Limiter start-up experiment:
We agree that coordinated experiment similar to ITER start-up (current and volume are increasing using outer limiter) is planned in order to solve questions discussed in Session 3. Lowry will start to plan this proposal.

DSOL-(19) D retention at high fluence in fusion materials --
This proposal was presented by Tsitrone, Coordinated measurements will be planned in tokamaks : TS, AUG (carbon), C-Mod (Mo), and PSI devices : PISCES, IPP ion beam, MEPHI, IPC magnetron,

Appendix A - Agenda

Appendix B - Meeting attendees

Total 46: EU (13)+JP (6)+Korea (1)+US (6)+CN (16)+ITER (4)

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