6th ITPA meeting on SOL/divertor physics

The meeting was held over the period July 4-7, 2005 at the Facultat de Ciències Jurídiques de la Universitat Rovira i Virgili in Tarragona. The local coordinator was Carlos Hidalgo. The meeting lasted 3-1/2 days and concentrated primarily on materials issues (see agenda in Appendix A). There were ~ 42 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

D/T inventories and methods for D/T removal from surfaces:

At the time of the 2004 Naka meeting it was recognized that fuel retention in the gaps between tiles is significant (up to 50% of total retention). At this meeting the results of IEA/ITPA collaborative work investigating this further was reported. The deposition depth down the gap varied with gap type (toroidal vs poloidal). The amount of deposition appeared to correlate with fluence to the front surface. These observations along with the driving physical process, assumed to be hydrocarbon neutrals) will be tested in future work.

D/T removal was an emphasis of this session. Four types of oxygen cleaning have been investigated with glow discharge cleaning appearing to be the most efficient. However, ICRF was almost as good and could be applied with the magnetic field on in ITER. More work is needed on the recovery of good plasmas (time taken and the level of O in surfaces and plasmas) following the cleaning, the efficacy at lower temperatures, the applicability to mixed material layers, whether it properly cleans down tile gaps, and whether the removal rate can be increased to that needed for ITER.

The removal rates for photo-cleaning are approaching that needed for ITER. When the deposits are thick high deposited energies are needed which raises worries about dust creation and how much T would be retained in the dust. There are three potential methods – flashlamps, lasers, and disruptive discharges. Besides the question of dust the group is planning to look into questions of power requirements, the applicability to mixed material surfaces and compatibility to magnetic fields.

Dust:

Dust is a concern for ITER from the point of view of safety. A secondary question is how the dust affects the core plasma in terms of impurity level. The amount of work in this area has rapidly increased over the last year. A range of presentations covered the formation of dust (e.g. agglomeration in the plasma, diffusion-limited aggregation on surfaces, and formation of quasi-homogeneous films). Modelling shows acceleration of dust particles to very high speeds (up to a km/s) which allows them to penetrate through the separatrix. Novel diagnostic techniques were shown which allow measurements of the dust falling on surfaces and in the plasma. The level of dust needed to halt operation was found to be substantially higher than normally found in current tokamaks. More work is needed to better understand the distribution of dust sizes, how the use of mixed materials affects their behavior, and how the dust affects the core plasma.

High-Z experience:

The effects of a variety of materials is being addressed at different levels within the world program; from JET, with the initial mix of materials, to ASDEX-Upgrade with steadily increasing amounts of W mixed with C (and B), to C-Mod (fully high-Z, with B coatings). In all cases we need a better understanding of D retention for individual materials, the effect of mixed materials on retention/removal, and the effect on core plasma operation (impurities and confinement).

All current tokamaks utilize wall-conditioning (e.g. boronization, Be gettering) to achieve the best energy confinement regimes. ITER performance is based on these results while at the same time it is unlikely that ITER will have any wall conditioning similar to current machines. Lastly, since it is currently thought that any reactor will have all high-Z PFCs to reduce neutron damage and radiation levels, it is considered important by the committee that at some point ITER convert to all high-Z PFCs. To address the importance of conditioning C-Mod removed all B from the machine (full coverage with Mo PFCs). Unfortunately, operation without B was much poorer in terms of energy confinement and core high-Z levels. Positive results were reported for minimal adsorption of water after a vacuum break and general D retention.

ASDEX-Upgrade results show clearly that the intermediate case of W, C, and boronized surfaces can work well; low W level H-mode discharges can be achieved with boronization and central heating to keep impurity accumulation down. However, when there is no boronization, as after the initial installation of W surfaces in the main chamber, core plasma performance was degraded. ASDEX-Upgrade plans to finish the conversion to all W PFCs over the next several years.

He bubble and blister formation in W has been shown to be dangerous for W surfaces in plasma simulators. Tokamak experimental results do not reflect this. Further work is needed to clarify the importance of this process.

First wall loadings (joint meeting with MHD group):

During the last several years the committee has worked towards a better understanding of first-wall particle and heat loadings. Much progress has been made in identifying the various channels: a) steady state, fluctuation drive fluxes; b) ELM fluxes; & c) disruptions. Based on a review of disruption power loadings it was apparent that the original assumptions used by the ITER group are incorrect in a number of points; e.g., the fraction of core plasma stored energy arriving at first-wall surfaces, the time scales.

The understanding of first-wall loadings during ELMs has advanced considerably in the past year. The filamentary structure (toroidal/poloidal number, rotation velocities) as well as the transmission through the SOL have been characterized. New measurements of the localized heat loads on first-wall structures have begun to emerge. The ELM energy that goes to the wall is localized toroidally and poloidally leading to localized erosion/melting of limiter structures. The localized nature of the power loading makes the characterization of heat loads (frequency, total heat load) difficult.

Disruption mitigation is being demonstrated to be an important tool for extending divertor lifetime. It was shown that gas could be successfully used to penetrate plasmas with pressures similar to ITER. New measurements (imaging, MHD, Thomson scattering, bolometry) are providing much-needed data for comparison with codes being applied (MHD and plasma evolution).

Given the importance of this session’s subject it was agreed that position papers summarizing recent progress would be written for the ITER group on disruption power loads, ELM power loads and disruption mitigation strategies.

Mixed materials

Utilizing different materials for different surfaces has the advantage of being able to tailor their capabilities for different applications. Recently it has been noted that there are unintended consequences as well; both positive and negative. Be deposited on C tends to reduce C chemical erosion. Be layers will retain less T than C layers and the T can be removed at lower temperatures. However, Be layers on W have negative consequences. Thin layers of $Be_{12}W$ form which, if they remain, have low melting temperatures and poor thermal contact to the bulk W below. What is not clear is whether such layers will form in regions where the deterioration of surface characteristics will reduce divertor lifetime. In other words it is possible that in high heat flux areas the layers will not form because Be will be re-eroded before it can diffuse into the W.

Be operation in ITER

This session was aimed at taking current knowledge of first-wall heat loads, material erosion and migration, and predicting the performance of the Be first wall in ITER. The goal was to examine whether the community could reach agreement of how Be would affect ITER operation. We also hoped to examine the basic assumptions going into these estimates and determine if more work could be done to verify them. Both extrapolations of current experimental results and modelling were presented.

The current estimates of first-wall erosion cover a wide range from low to 1 μm/discharge, thus placing some limit on first-wall lifetime and being a significant impurity source.

The general expectation is that if ELM heat loads on the divertor can be reduced to acceptable levels, the same may be true for ELM erosion of the first-wall surfaces. The same cannot be said about disruptions where, even if the energy is spread over a large fraction of the first-wall during the thermal quench (mitigated or not) there is likely to be significant melting.

It was generally agreed that the majority of the eroded Be goes to the inner divertor region. This is of course based on current C and Be migration measurements. However, the details of how it reaches that location and whether the differences in ITER divertor will affect the prediction are certainly highly uncertain.

The replacement of C with Be as the main eroded material is likely to reduce T retention from that predicted for a full C ITER. The concentration of T in Be is typically in the 1-5% level (can be 40-100% in C). T is released from Be at lower temperatures (600 oK) than for C. Finally, once Be migrates to the inner divertor, it does not appear to migrate to sheltered areas that are difficult to access with T cleaning techniques. That said, it is not clear how well T can be removed from mixed T/W/C layers and whether Be-W alloys will lead to a degradation of the W divertor capability in high heat flux areas.

Once first-wall Be surfaces are damaged (by ELMs of disruptions) we are concerned about how much the heat-handling capability is reduced. This is a particular concern for the upper divertor and the startup limiter. Be (and W) melt-layer dynamics in real tokamak situations need to be better understood as well as the resiliency of the core plasma to absorb large amounts of low-Z ions for short periods.

Discussion of the ITER dome

No consensus was reached on whether removal of the dome would enhance or hurt ITER operation. There exists no experimental evidence or modelling results indicating that the dome reduces neutral backflow to the plasma (part of the original motivation). At first glance this result together with concerns about dust and T collecting under the dome would seem to argue for its removal. However, a detailed engineering design for the dome replacement does not exist. It is thus difficult to assess whether dust removal and shadowed areas would be significantly reduced. Finally, the removal of the dome has very negative implications for diagnostics that will be compromised or removed. If a replacement for the dome is designed and can retain diagnostics the SOL/divertor committee stands willing to help evaluate its performance.

Next ITPA meeting and topics

It was agreed thatsince the meeting had not been held in Asia for at least a year we should have one there. The most convenient time is to combine it with the Plasma Surface Interaction meeting abstract selection meeting which already includes a number of our members. If Jiangang Li agrees to this it will likely occur either in China or Japan in late January or early February. Now that the US is being allowed to host meetings again it is possible that the following meeting would be held somewhere in the US in the Summer of 2006.

A number of topics were suggested for the next meeting. They include D/T retention in surfaces, wall conditioning, startup studies, flow characterization, melt layer stability, vapor shielding, and spectroscopy of hydrocarbons in low-T plasmas.

We are preparing position papers advising ITER or revised assumptions for first-wall and divertor loadings under ELM and disruption conditions. We are also planning to revise the high-priority task list to reflect an ITPA CC request for more near-term goals.

II. Detailed session summaries

Session 1: ‘D/T inventories (surfaces and sides of tiles) and their removal

1.2 ‘Recent results on carbon deposition and fuel retention in gaps of plasma facing structures’, K. Krieger

Plasma facing components for the ITER divertor and baffle regions will likely be manufactured with macrobrush structures or castellated surfaces. Material samples with gaps of similar geometry were exposed to different plasma conditions in ASDEX Upgrade, TEXTOR and DIII-D. In all devices an exponential decrease of both carbon and deuterium inventories at the side faces from the gap entry into the gap is found. The scale length is mainly related to the gap width and the orientation of the B-field. The fraction D^+ incident on the front surface which is retained in the gaps is in the range of a few percent. Extrapolation of growth rates under low flux conditions to ITER dimensions shows no significant contribution to the total tritium inventory. Growth rates observed in gaps exposed to high fluxes such as near the strike point zones are 100 times higher, which might be of concern for ITER operation with carbon PFCs. On the other hand, experiments with samples kept at elevated temperature (200ºC) show a decrease of retained D by a factor of up to 10, although this observation is restricted to detached plasma conditions.

1.3 ‘Boron & Deuterium down tile gaps in C-Mod’, D. Whyte

Ion beam analysis was used to measured boron and deuterium deposition down tile gaps of the C-Mod Mo tiles after ~7 years of exposure. Boron is applied frequently to Mo tiles using ECDC of He + diborane with a weak magnetic field (~900 Gauss) Boron layers up to ~1 micron thick were found on edges near the plasma-facing side with typical e-folding distances of 1-3 mm down the gaps, which is 2-6x the distance between tiles (0.5 mm). Deuterium deposition clearly correlated to B, indicating codeposition with B as the main reason for D retention. The D/B ratio varied from ~2-10% with an average value of 5% in most cases. The D/B ratio is similar to those found on the plasma-facing tiles. It is estimate that 10-20% of the deuterium retention was found in the gaps for the C-Mod tiles. Toroidal running gaps were found to have deeper penetration of boron and deuterium down the gaps than in the poloidal running gaps. Variation was also found in e-folding distance as a function of poloidal location of the tiles. This suggests that B deposition, along with D codeposition, is occurring during plasma discharges.

1.4 - 'Recent results on carbon migration and deposition in ASDEX Upgrade', V. Rohde

During the 2002/03 campaign a divertor marker experiment to get a complete data set on the carbon migration and deposition was performed in ASDEX Upgrade. Even below the divertor structure deposition and erosion are observed. As the erosion is direction sensitive, it is assumed that ions play an important role. Parasitic plasma observed below the roof baffle can produce these ions. The surface loss probability at remote areas is determined by cavity probes: gamma = 0.65-0.75. This value is consistent with the observation that deposition on the qmb's is strongest if the strike point position is in the line of sight of the qmb location. The temperature dependence of the deposition was investigated by heated and cooled probes. As expected for re-erosion by atomic hydrogen the layer thickness can by reduced by a factor of 50 for a probe temperature of 200 C. Probes mounted at the LN2 shield of the in vessel cryo pump measure a factor of 100 enhanced deposition. The deposition on the target plates shows significant differences for short term ($^{13}CH_4$ puffing) and long term deposition. A peak at the inner baffle, caused by direct ionisation of the injected $^{13}CH_4$ is not observed on long term data. Whereas the $^{13}CH_4$ data shows a stronger deposition at the outer divertor, the long term data shows the dominant deposition at the inner divertor. Obviously redistribution of the deposited carbon has to be taken into account to understand the long term deposition pattern. QMB data show constant layer growth at the inner divertor structure. At the outer divertor erosion and deposition phases are observed.

1.5 - '^{13}C injection at the outer divertor in JET', V. Philipps

$^{13}CH_4$ injection into the outer SOL of JET under identical H-mode discharges show about 10% of injected ^{13}C deposited on the inner tiles, preferentially on the horizontal tile, while the strike point was on the vertical tiles. The transport towards the inner divertor seems to originate from ^{13}C that has entered the confined plasma , diffused to the SOL and driven by SOL flows to the inner divertor. Interestingly the ^{13}C has undergone further transport in the inner divertor from the vertical strike point tile to the horizontal tile which was in the PFR which is probably due to Elms. The amount found on the vertical tiles near the outer injection is comparably small (about 15%), but 2 horizontal outer divertor tiles have not been analysed so far.

1.6 - ‘Recent activities on photocleaning in the EU’, C. Grisolia

In the EU, photon detritiation using flash lamp or laser ablation techniques is under strong development. Flash lamp detritiation has been tested in JET and films have been removed with a rate of 2.5 to 10 m^2 of 10μm co-deposited films per hour with the flash lamp in the ablation regime at 250 Joules per pulse. Laser detritiation technique has been proven at a laboratory scale with a rate of 1 m^2 of 50μm deposited per hour using a 100 W laser. However, more R&D is necessary to meet the constraints of ITER.

1.7 - "Oxygen cleaning activities in TEXTOR", A. Kreter

Oxygen cleaning techniques were applied in TEXTOR to remove co-deposited carbon/hydrogen layers. The following table gives an overview of achieved C removal rates, main advantages and drawbacks of the techniques as well as possibilities to increase the C removal rates. C removal rates have to be compared with an integral TEXTOR carbon redeposition rate of $2.7 \cdot 10^{20}$ C/s

Technique	C removal rate	Advantages	Drawbacks	Possible improvements
Oxygen venting	$2.5 \cdot 10^{18}$ C/s for 0.3 mbar, Twall=620 K	Simplicity Access to all wall areas Selective removal of redeposited layers	Low removal rates T_{wall} > 600 K needed	Higher O pressure
Glow discharge conditioning	$2\text{-}3 \cdot 10^{19}$ C/s	Higher removal rates Applicable for low T_{wall}	Incompatible with steady state B non-selective carbon removal Limited wall area access	Higher GD current Higher pumping rate
ICRF conditioning	$1.8 \cdot 10^{19}$ C/s for 1:10 duty cycle for pump out	Higher removal rates Applicable for low T_{wall}	O injection limited by pressure limit at antenna box	higher pumping rate (also for steady

				Compatible with steady state magnetic field	Non-selective carbon removal Limited wall area access	state ICRF)

1.8 - ‘Oxidation experiment in HT-7’, J.S Hu

The oxidation experiments, including O-ICR conditioning, O-GDC and O-baking have been done in the HT-7 superconducting tokamak. The temperature of the limiter is 402-425K and that of the liners is 435-470K. The higher pressure and conditioning power are favorable for removal of hydrogen and carbon. Pure oxygen and gas mix of He/O were used. The highest removal rates of H, D and C-atoms in pure O-ICR experiment up to 2.64×10^{22}, 7.76×10^{21} and 1.49×10^{22} atoms/hour respectively were obtained in 40kW 9×10^{-2}Pa O-ICR cleaning, corresponding to the removal rate of co-deposits of about 317nm/day (7.2g/day for carbon). The highest removal rates of H and C atoms in gas mix He/O-ICR experiment up to 5.4×10^{21} and 7.2×10^{21} atoms/hour, respectively, were obtained in 4:1 40kW 9×10^{-2}Pa He/O-ICR cleanings. Average removal rates, 5.2_10^{22} H-atoms/hour, 5.65×10^{21} D-atoms/hour and 5.53×10^{22} C-atoms/hour, respectively, were obtained in 145min O-GDC experiment in a pressure range of 0.5~1.5 Pa.. Only O-ventilation has very little effect to remove co-deposited layers and hydrogen. By correct cleaning, most oxygen retained on the wall was sufficiently removed before plasma discharge. Plasma discharges could be recovered after a few tens of shots with large disruptions. Even though the impurities, such as C, O increased, the hydrogen recycling was largely reduced by He/O_2-ICR experiment. No apparent damage in the torus after oxidation experiment was observed.

1.9 - ‘ICRF H/D removal in LHD’, N. Ashikawa,

ICRF wall conditioning under high magnetic fields has been started in LHD. Operation was achieved with P_{icrf} from 8 to 149 kW and helium pressure from 10^{-2} to 10^{-1} Pa.

In these initial experiments sufficient power was absorbed by hydrogen as determined from electron density and FNA data. An increase in hydrogen pressure with RF power was observed. LHD did not have RF breakdown in the range 10^{-2} to 10^{-1} Pa in He pressure. The hydrogen removal rate by GDC is larger than with ICC. Optimization of the ICC with RF phasing is an important part of future work.

1.10 - ‘Summary of nitrogen injection experiments in AUG’, P. Tabares

Nitrogen injection experiments at the subdivertor region of Asdex Upgrade have shown for the first time that a decrease of up to a factor of 5 can be achieved in the carbon deposition rate at positions not accessible directly by the plasma. Furthermore, the required nitrogen flows are fully compatible with the unperturbed operation in H mode discharges. No change in the deduced carbon concentration in the divertor plasma has been detected so far, in line with the expected scavenger effect of the injected species.

1.11 - ‘Effects of wall saturation on plasma operation’, E. Tsitrone

The main effects of saturation of carbon walls on plasma operation are : 1) loss of the density control during a discharge; 2) difficulties to start up the plasma (no sustained breakdown).

The first effect has been clearly observed in all tokamaks going to long pulse operation with a non or partly actively cooled machine (Tore Supra before the CIEL upgrade, JT60U, HT7 …), resulting in an uncontrolled density rise (typically < 1 minute); temperatures of the plasma facing components rise during the entire discharge.

The CIEL upgrade to a completely actively cooled machine dramatically improved the density control in Tore Supra, allowing discharges up to 6 minutes/3 MW/1 GJ and 1 minute/7 MW/0.4 GJ. Conditioning is needed before reaching reproducible conditions for the high power/high density shots. This is attributed to plasma facing components not in direct interaction with the plasma, with long thermal time constants.

Wall saturation experiments running repetitive discharges have been performed on JT60U and Tore Supra. The wall pumping capacity is seen to decrease from shot to shot, and the series of shots end up with a disruption, and subsequent not sustained breakdowns.

To overcome this problem, techniques to monitor the wall saturation status have been successfully developed in Tore Supra. Cleaning techniques to restore the wall pumping capacity use cycles of very low current (< 40 kA) successive breakdowns. This process is compatible with the toroidal field, but is not directly applicable to ITER as the magnetic coils system is different.

In ITER, which will run under very high D fluence within one discharge, the consequences of wall saturation on plasma operation should be better assessed. Operation of most present day machines relies heavily on conditioning techniques (boronisation, Be evaporation, He glow discharges …), and that the best performances in terms of confinement, on which the extrapolation for ITER is based, have generally been obtained with pumping walls. ITER relevant techniques to monitor the wall saturation status and optimise the start up should therefore be developed.

1.12 - 'Summary of the Session 1 discussion', V. Philipps

Material and fuel retention in gaps: all the present data show that material and fuel deposition in gaps of ITER wall components will seriously contribute to the tritium retention. In order to extrapolate more quantitatively from present data to ITER, the mechanism of how the material and fuel penetrate into the gaps must be better clarified. An important question is whether this migration is mainly restricted to carbon, due to the special carbon chemistry, or is also expected e.g for Be. Presently, it is believed that the hydrocarbon chemistry is important for the gap deposition, but more studies are necessary. The other important question is how to extrapolate from present data to ITER. The fraction of fuel deposited on top of the tile surfaces compared to that in gaps is sometimes used. A perhaps more appropriate quantity to use is the fraction of material and/or plasma fuel arriving on the surface compared to that trapped in gaps. Some data presented showed 0.2 % of the surface D fluence being retained in the gaps of an ITER-like wall component (TEXTOR) but the field line inclination was steeper (20°) than that expected in ITER More experiments elucidating the transport mechanism are necessary.

Material migration: while the overall tendency for the inner divertor to be a deposition dominated area is common to all divertor tokamaks, the behaviour of the outer divertor is still difficult to predict. DIID and ASDEX-Upgrade data show that erosion and deposition dominated phases are both observed at the outer divertor. Recent data show consistently in DIID and AUG a strong influence of the substrate temperature on deposition rates, with a decrease by about one order of magnitude from ambient to 200°C. This calls for hot surfaces at the deposition area in ITER, mainly the dome and PFR region. The transport of material from the outer to the inner divertor is still an open issue and needs further clarification. In particular, particle flow through the PFR may play a role. Analysis of recent 13C injection into the outer divertor region of JET is not yet available.

Fuel removal: development of fuel removal methods that are applicable in ITER environment is most important. Photo-cleaning has been demonstrated on larger areas in JET but concerns remain about the concurrent production of dust (both for T retention and for safety issues of large quantities of C dust). Dust collection in parallel with photo-cleaning is absolutely mandatory but may be difficult and not 100% efficient. Oxygen cleaning can only be used for carbon layers. It is assumed presently that GDC and ICRH will clean only the plasma facing sides. The removal action of these standard methods in hidden areas should be investigated. Oxygen molecule oxidation of C layers needs temperatures above 250-350°C, as shown again in the HT-7 tokamak. Such temperatures are extremely difficult for ITER to achieve. Plasma recovery after oxygen treatment needs to be documented and scaled to ITER. Scavenger techniques like the use of N may be of use but are far away from being a proven technique for ITER.

Density control and wall saturation becomes more and more important with increasing pulse length as seen in Tore Supra and other machines. However, based on our present understanding, the importance of such effects for ITER which has a larger particle throughput (needed for He exhaust) is not clear. More experiments are needed to develop a proper understanding of the processes involved, how to control them, and how to prepare the walls for such long pulses.

Session 2: 'Dust'

2.2 'Results of Dust in JT-60U & LHD' , N. Ashikawa

Based on dust measurements we have extrapolated the amount of dust in the whole vessel as 7 g in JT-60U and 3-10g in LHD. The production rates normalized to discharge time are 0.2 mg/s in both devices. The total amount and production rate are similar in both devices. But the particle size distribution was different. In JT-60U, dust deposition was found in remote areas of the outer divertor region. Small dust with nm order size can not be bypassed to the total amount by large dust with glowing process. Moving particles were observed by high speed IR camera and this transport time in the camera FOV is about 40ms and estimated transporting speeds are about 20 m/s at the outside edge.

2.3 Development of dust diagnostics', R. Maingi, C. Skinner: ITER safety will depend on knowing and controlling the inventory of dust on interior surfaces, but diagnostics to estimate the dust inventory need to be developed. A novel electrostatic dust detector has been demonstrated in both air and vacuum environments in the laboratory [Rev. Sci. Instrum., 75 (2004) 370]. A fine grid of interlocking traces with spacing down to 25 μm is biased with 30-50 v DC. Impinging dust produces a short circuit and the resulting current pulse both vaporises the dust and produces a pulse that is recorded with standard nuclear counting electronics. Particle size information can be obtained from the electrical waveform. A detector has been installed at the NSTX at a port under divertor. Such detectors could be adapted for specific areas in ITER. (Maingi for Skinner et al.)

2.5: "Modelling of Dust Formation in Plasmas" - Xavier Bonnin.

A new edge plasma physics group has been formed at Université Paris XIII, to work on issues at the confluence of industrial cold plasmas and edge fusion plasmas. One such common research area is dust. We have built a model for dust formation and nucleation, via carbon cluster chemistry, and compared it to a DC Argon discharge experiment in Marseilles where Ar sputters a Carbon cathode and dust is collected at the anode. The dust is cauliflower-like and reaches cluster sizes of order 40 nm. The model can reproduce such a formation by agglomeration of C2 and C3 sputtered clusters onto larger ones and its experimental time evolution to within a factor of two. More accurate chemistry rates (in particular for negatively charged clusters) is expected to improve the match to experiment. Additionally, a new plasma Chemical Ablation, Sputtering, Ionization, Multi-wall Interaction and Redeposition (CASIMIR) device is currently being built to address issues related to hydrocarbon erosion products chemistry, transport and redeposition in parasitic plasma environments as those expected under the ITER divertor dome.

2.6: ‘Dust in DIII-D’, M. Fentermacher, D. Rudakov, P. West

Intrinsic dust in DIII-D has been characterized using signals from the Rayleigh scattering channel of the core and divertor Thomson scattering systems. A small particle (about 60 nm) distribution and a large particle distribution (> 300 nm) are indicated by the data. Average dust particle density ranges from 0.005 particles/cc in the upper part of the vessel to 10 times that level in the lower part of the vessel. Dust injection experiments have also been done using 1-10 mm particles on a DiMES sample. Tracks of dust particle motion when the dust encounters the OSP show a preferential direction that is about 15-45 degrees inboard of the magnetic field direction. Drag force caused by plasma flows, ExB drift in the pre-sheath electric field can explain this direction. Some curvature is also observed in the particle trajectories. These measurements provide a test dataset to validate dust particle transport codes.

2.7: 'Operational recovery from enormous amounts of titanium dust', B. Lipschultz

Because of an unfortunate degradation of the font end of the new C-Mod LH coupler (made of Ti) a very large amount of Ti dust was created in C-Mod., After removal of the waveguide ~ 300 g remained in the vessel affecting startup and operation. This is equivalent to something in the range of 24 kg (scaling by area) to 220 kg (scaling by volume) in an ITER-size device. The length of plasma discharges increased and reached full length over ~ 200 shots. The Ti level and the gas retained by the dust dropped over a similar amount of time. Given the comparison to Mo levels in the plasma it appears that normal amounts of dust are not directly affecting the discharge impurity levels. Only when the dust levels get very large is it a problem.

2.8 - 'Dust dynamics in MAST', GF Counsell (UKAEA Fusion, EU)

A model has been developed in association with Imperial College, London which allows the motion and thermal properties of dust to be evaluated in the fusion plasma environment. The code takes 2D data of plasma properties (temperatures, densities, plasma flows etc) and the magnetic equilibrium from standard output of the B2SOLPS5.0 fluid code. Monte Carlo test particles can then the launched with an arbitrary mass and velocity distribution from any poloidal location and followed until they either evaporate in the plasma or strike a wall. Both tungsten and carbon dust has so far been modelled. The dust charge at each time step is calculated from OML theory, allowing for the impact of secondary electron emission, using the local plasma parameters and the acceleration is then evaluated using an equation of motion including the Lorentz, pressure gradient, flow pressure and gravitational forces. Temperature evolution of the dust is evaluated taking into account all significant heating and cooling mechanisms (e.g ion, electron and neutral bombardment, radiative and ablative cooling etc.). For a typical dust trajectory, flow pressure is the dominant force when the grain is large, with the Lorentz force dominating as the dust moves into regions where evaporation occurs. For MAST plasmas, dust launched from a toroidally and poloidally localised source is widely redistributed around the vessel by the plasma and some dust trajectories can lead to evaporation occurring inside the separatrix. Modelling of dust transport in ITER plasmas is on-going, with the eventual aim of predicting regions where dust is likely to accumulate in the vessel.

2.9 - ‘Summary of discussion’ - S. Krasheninnikov

In situ experiments with dust: Intrinsic dust in DIII-D has been characterized using signals from Thomson scattering systems. Average dust particle density ranges from 0.005 particles/cc in the upper part of the vessel to 10 times that level in the lower part of the vessel. Dust injection experiments have also been done using 1-10 micron particles on a DiMES sample. Tracks of dust particle motion show a preferential direction that is about 15-45% inboard of the magnetic field direction and the estimates of dust particle speed give ~10-100m/s. A similar magnitude for dust speed was also observed in JT-60. In C-Mod a huge generation of dust due to Ti structures of antenna resulted in a very short, resistive plasmas with no density control etc. After removing Ti structures and boronization, and following a few hundred shots, standard core performance was recovered. ~ 300g of dust still remains in vessel which may indicate that the remaining dust is localized in the regions with rather small exposure by plasma. Since the Mo dust level is much lower and Mo is affecting the core plasma more, the inference is that dust is normally not a contributor to steady state impurity levels. Dust is likely to be more important for ‘injections’.

Analysis of dust collected in fusion devices: Dust production rate ~ 0.2 g/s was found in both JT-60 and LHD. Taking into account the erosion rate of divertor tiles in JT-60 outer divertor it gives ~ 7% transformation of eroded carbon into dust;

Development of dust diagnostics: A biased fine grid with the spacing down to 25 microns biased to ~50 V has been developed for measurement of dust landing on it. It works fine in a test chamber. Particle size information can be obtained from the electrical wavefront. The detector is installed in NSTX at a port under the divertor. An SEM measurement, allowing in situ monitoring of the first wall structure, will be installed at JET;

Dust modeling: To model the dynamics of dust in tokamaks 3D codes (DUSTT at UCSD and another one at Culham/Imperial Collage) are being developed. Codes allow tracking of a test dust particle in toroidal geometry with realistic plasma background. Preliminary results of dust transport modeling in NSTX, DIII-D, ITER and MAST tokamaks have been reported showing reasonable agreement with available experimental data. It was shown that dust obviously penetrates deeper into edge plasma than individual neutral impurity atom causing a significantly bigger impact on both edge plasma parameters and detachment processes. A new edge plasma physics group, at Université Paris XIII is to build a model for dust formation and nucleation, via carbon cluster chemistry, and compare to experiments in Marseilles. A new plasma device CASIMIR is currently being built to address issues related to the chemistry, transport, and redeposition in parasitic plasma environments such as those expected under the ITER divertor dome.

Session 3: 'High-Z experience'

3.1 - 'W operation in ASDEX Upgrade', A. Kallenbach – IPP Garching

ASDEX Upgrade is currently coated with 70 % tungsten PFC coverage, it is planned to come to a complete coverage within the next 2 years. The dominant sputtering source is caused by low-Z impurity ions of C, B and O. Strong variations of the peripheral W concentration (measured around Te= 1 keV) and the central W concentration are observed. Divertor operation is affected under 2 experimental conditions due to strong central W radiation: in the low power H-mode with low ELM frequency, impurities are driven from the SOL to the pedestal by a strong inward drift. The resulting strong radiative losses may cause a H-L transition. This behaviour can be cured by pellet ELM pace-making. For discharge conditions with good confinement and central electron density peaking inside $r_p= 0.4$, strong central accumulation of W occurs. This is driven by neoclassical effects and can lead to central radiation approaching the heating power density. Central heating above a power threshold flattens the central fuel density profile and consequently leads to flat tungsten profiles with W concentrations around 10^{-5}.

3.2 - 'High temperature erosion and melting of tungsten in TEXTOR', A. Kreter

Experiments on high temperature erosion and melting of tungsten were performed in TEXTOR. Observed tungsten erosion was attributed to the sputtering by carbon and to the sublimation. No indication of an enhanced high temperature erosion of tungsten was found. A local tungsten flux for T_{surf}<3000K of up to $4x10^{21}$ atoms/m^2/s was measured, corresponding to 4% of the background flux, in agreement with previous observations in TEXTOR. No increase of the tungsten concentration in the core was observed. Power deposition onto the tungsten plate resulted in a melt layer formation and a fast melt layer motion with a velocity of ~1.5 m/s. The motion of molten tungsten can be attributed to the thermo-emission current and the resulting JxB force. Surface studies indicated a recrystallization of tungsten with the maximum grain sizes in the zone of the highest temperatures. Carbon dust particles were found incorporated into the melting zone. No blistering in the melting zone and other parts of the tungsten plate was observed.

3.3 - ‘Removal of boron from Alcator C-Mod and the effect of boronization’, B. Lipschultz

All currently operating tokamaks use some kind of PFC coating - whether it be boronization (e.g. ASDEX-Upgrade, C-Mod, DIII-D, JT-60U…..). ITER performance is based on this while at the same time ITER presently will not be able to use such wall coatings. Information is needed on what the wall coating provides. Unfortunately, coatings in current C PFC machines cannot be readily removed to make this comparison. C-Mod has provided unique results in this sense by removing the B coating and examining changes in performance. In addition this provides data for an all high-Z first-wall, under consideration for ITER. The results indicate that the B coating is necessary to reduce the effect of Mo on core performance. At the beginning of H-mode the impurity confinement is essentially infinite and any Mo source is seen in the plasma, often leading to an H-L transition. The B layers also play a role in adsorbed D, lengthening the pumpdown. On the other hand the B layer does not affect the absorbed D compared to a bare Mo surface. Ongoing experiments are being made to investigate inter- and intra-shot boronization in support of ITER. The implication for ITER is that all high-Z PFCs might lead to significant core Mo concentrations during H-mode. Whether this will have an adverse effect on confinement and the exact level of W requires further study.

3.4 - ‘W-tile experience from JT-60U’, Y. Ueda/T. Nakano

Tungsten divertor tile experiments were started in JT-60U in the experimental campaign 2003-2004. 13 tungsten coated CFC tiles were installed in the P-8 section (12 tiles) and the P-17 section (1 tile) just above the CFC tiles where the outer strike points normally intersect. Plasma heat load up to 6 MW did not cause any notable damage on W-tile surfaces. On the CFC tile toroidally adjacent to the W-tiles, localized thick W deposition near the W tile edge was observed. Except for this area, almost uniform thin W deposition was observed. W migration to the surface of the dome top tile in the P-8 section was observed, while no observable amount of

W was found on the inner baffle tiles. In high NBI power shots (~15 MW), W accumulation could take place. This accumulation could be mitigated by edge gas puffing (preliminary results).

3.5 - 'He ion irradiation effects on W', Y. Ueda/N. Yoshida

Tungsten samples pre-irradiated by 8 keV He at RT to the fluence of up to 10^{22} m^{-2} were exposed to 13 MW/m^2 heat load by the electron beam. It was found that He pre-irradiation strongly deteriorated thermal shock resistance of W. In the case of 10^{22} m^{-2}, erosion up to the depth of 0.8 mm (10 times larger than the He ion range) took place due to reduction of thermal conductivity and hardening by He bubble and blister formation. He bubbles were also formed by low energy He irradiation under the conditions of T > 1300 K, fluence > (10^{25} ~10^{26}) m^{-2}, and He energy > 6 eV. Since edge plasma parameters on the ITER outer divertor will satisfy these conditions, He bubble formation on the outer tungsten diverter plates will likely occur at elevated temperatures and affect the operating conditions (surface temperature limit, allowable ELM heat pulse energy, etc.). Intensive works are needed to evaluate the impact of He irradiation to W for the material selection of ITER divertor.

3.6 - Discussion and summary of Session 3 "High-Z experience", A. Kallenbach

During the sessions talks, several new results were presented on tokamak operation with high-Z plasma facing components: Operational restrictions imposed by tungsten radiation have been observed in AUG with 70 % tungsten coverage. These can be cured by central heating for density profile flattening and ELM pace-making to avoid long ELM-free phases. JT-60U observed preferential W migration from tiles installed in the outer divertor in toroidal direction as well as to the dome. The behaviour of W erosion under high heat fluxes was investigated in TEXTOR. Tungsten erosion could be attributed to sputtering by carbon and sublimation, and no anomalously high yields were observed at high surface temperatures. These experiments were driven up to W melting, and the motion of the molten W could be attributed to thermo-emission currents and the resulting jxB force. No W blistering was found. Tungsten samples exposed to high He fluences in NAGDIS-II showed bubble and blister formation. During the discussion, different possibilities were brought up why these negative effects occur in plasma simulators, but not under tokamak conditions. One reason could be the fact that monoenergetic He beams produce a different deposition characteristic in the material. Further work is needed to clarify this topic. The importance of wall conditioning in a full-metal device is a topic which is highlighted by recent results from Alcator C-Mod, where degraded plasma performance was observed during non-boronized operation with Mo walls cleaned from B.

In the discussion session, it was emphasised that a strategy needs to be developed for the possible implementation of additional high-Z PFCs in a later ITER phase, which takes into account technical boundary conditions like the possibility of a change of the first wall. While a much lower fuel retention by tungsten is expected compared to carbon, the reduction of divertor radiation will require additional impurity seeding. Two topics are foreseen to be put on the agenda for the next meeting: 1) possible impact of a high-Z startup limiter (erosion, flux consumption) and 2) wall conditioning of an all-metal device.

Session 4: 'First wall loadings', joint meeting with the MHD group

4.1 'EU-PWI Task Force Studies on Disruption Energy Fluxes', A. Loarte

Analysis of the energy balance and timescales for energy fluxes during similar disruptions in JET, ASDEX Upgrade, MAST, DIII-D, TEXTOR and FTU was presented. On the basis of the experimental data analysed, the following conclusions have been reached :

a) The thermal energy of the plasma at the thermal quench is significantly lower than that during the full performance phase of the discharge for most disruptions, typically by a factor of 2 to 4.

b) There is a favourable size scaling of the timescale for energy flux to PFCs during the thermal quench, but there is a significant spread in this timescale (by factors of ~ 6) within each experiment and a large variation disruption-to-disruption for nominally identical discharges.

c) There is a significant broadening of the heat flux during the thermal quench (by a factor of 5-10 compared to the full performance plasma) in diverted discharges. This seems to be absent in limiter disruptions.

These new measurements call for a re-evaluation of the conditions expected for the average disruptions to be encountered when operating ITER in its reference Q_{DT} = 10 scenario based on the ELMy H-mode. In the first place, the average plasma energy at the thermal quench in ITER will be ~ 88 MJ to 175 MJ, if no disruption amelioration is applied, i.e., ~ 25 -50 % of the full performance plasma energy. This energy will flow to the divertor over a very large area, which will be typically in the range of 5-10 times the divertor wetted area for power flux in steady-state conditions ($A_{s.s}$), i.e. 18–35 m^2 in ITER. The typical timescale for the thermal quench power pulse will be in the range 1.5–3 ms. As a consequence of these factors, the expected divertor energy flux in the average ITER disruption, if amelioration actions are undertaken by the systems, will be ~ 3.3 MJm^{-2}, with a timescale of ~ 2.3 ms. Using the standard deviations in the distribution functions, the lowest and highest thermal quench energy fluxes will be ~ 7.5 MJm^{-2} in ~ 1.5 ms and ~ 1.3 MJm^{-2} in ~ 3.0 ms, respectively. These values have to be about a factor of ~ 2 higher if only the natural degradation of plasma confinement in advance of the disruption is considered for ITER. The characteristic surface temperature rise, which determines the material damage caused by the power flux, can be estimated by the so-called “ablation-melting” parameter $f \sim \Gamma_{Et.q.}/\sqrt{t_{t.q.}}$, where $\Gamma_{Et.q.}$ is the energy flux during the thermal quench and $t_{t.q.}$ is its timescale. For the average ITER disruption, the expected “ablation-melting” parameter will, thus, be in the range 46–129 MJ $m^{-2}s^{-1/2}$, which is about a factor of 3-4 higher than that required to cause carbon ablation (35 MJ $m^{-2}s^{-1/2}$) and tungsten melting (40 MJ $m^{-2}s^{-1/2}$) for ameliorated disruption and 64–322 MJ $m^{-2}s^{-1/2}$ for disruptions without amelioration actions. While these values correspond by no means to a low energy flux, they are typically about an order of magnitude lower than that expected under the previous ITER assumptions ($W_{plasma,t.q.}$ = 350 MJ, $A_{t.q.}$ = 3 $A_{s.s.}$, $t_{t.q.}$ = 1 ms, f = 1054 MJ $m^{-2}s^{-1/2}$). As a consequence, the expected divertor target lifetime under these “revised” most frequent disruptions in ITER is significantly longer than previously estimated.

While the new estimates alleviate the problem of energy deposition by disruption at the ITER divertor and its lifetime, they lead to significant energy flux reaching components (Be-clad blanket modules) on which no significant disruptive energy flux was expected previously.

More details on the measurements and analysis can be found in “Expected energy fluxes onto ITER Plasma Facing Components during disruption thermal quenches from multi-machine data comparisons”, A. Loarte, et al., Paper IT/P3-34, Proc. 20th IAEA conference, Villamoura, Portugal, 2004.

4.2 – ‘Disruption mitigation studies in JT-60U’, Kawano

It is well known that relativistic runaway electrons can be generated during a tokamak disruption, and the plasma facing components would be damaged if the localized and intense irradiation of runaway electrons occurs. In order to mitigate the post-disruption runaway electrons with good controllability, their characteristics with impurity pellet injection were investigated for the first time using the JT-60U tokamak device.   A clear deposition of impurity neon ice pellets was observed in the post-disruption runaway plasma. The pellet ablation was attributed to the energy deposition of relativistic runaway electrons in the pellet.   A high normalized electron density was stably obtained with n_e^{bar}/n^{GW} ~2.2. Prompt exhaust of runaway electrons and reduction of runaway plasma current without large amplitude MHD activities were observed. One possible explanation for the basic behavior of runaway plasma current is that it follows the balance of avalanche generation of runaway electrons and slowing down predicted by the Andersson-Helander model, including the combined effect of collisional pitch angle scattering and synchrotron radiation. Our results suggested that the impurity pellet injection reduced the energy of runaway electrons in a stepwise manner.

On the other hand, it has been pointed out that the current quench time is extended by the appearance of runaway electrons.   This fact suggests that runaway electrons can be used to mitigate or avoid the current quench when the safe and reliable control of runaway electrons is available. Taking such a new point of view, an experiment for avoiding a rapid current quench by allowing runaway electrons has been carried out. An impurity neon pellet was injected as a killer pellet into an Ohmic discharge during the period with the ECRF injection. Here, the current quench was avoided presumably due to relatively high electron temperature just after the pellet injection. While the electron temperature decreased afterwards, runaway electrons were generated. These runaway electrons reinforced the discharge to survive against the low electron temperature of less than several tens eV and additional impurity neon pellet injection. Thus the robust discharge was obtained, and the plasma current was maintained and terminated as programmed. This result indicates that the new concept for mitigation and avoidance of current quench by runaway electrons is attractive.

4.3 – 'DIII-D & Alcator C-Mod disruption mitigation', D. Whyte

Experiments show that the gas delivery rate is important for runaway suppression and overall mitigation effectiveness. The gas must deliver electrons (free + bound) in a time less than the current quench time. The multi-device / multi-jet experiments will be needed to separate this effect from any gas pressure scaling for edge penetration. Complementary to the experimental work, realistic modelling of gas shock, including friction, for gas delivery down the tubes is in progress.

Fast imaging shows that the gas/impurity interaction with the plasma is complex. The toroidal and poloidal distribution of impurities and their associated radiation will need to be better measured to extrapolate radiation uniformity. Both experiments and models suggest that MHD plays a role in final particle "penetration" to the center, without producing large heat fluxes to the walls. Robust low-order MHD mixing lessens gas jet requirements for ITER, but requires carefully benchmarked MHD modelling, which has now started using the 3-D NIMROD code.

First experiments showed that gas jet delivery "worked" in C-Mod with its high absolute plasma and magnetic pressure. Tailoring of the gas species mixture can optimize thermal- and current-quench times.

4.4 – 'Plasma shut-down with fast impurity puff on ASDEX Upgrade', S. Günter

Neon gas puffing is routinely used to mitigate disruptions on ASDEX Upgrade, triggered by a locked mode signal. Around 4.5×10^{21} atoms (50-100% of the fuel inventory) are injected in 2ms. The neon does not penetrate into the core plasma on the time-scale of the thermal quench, and builds up a cooling mantle at the edge ($\rho > 0.8$). Subsequent (1-2ms later) narrowing of the core temperature profile and an order of magnitude density increase triggers large MHD modes and disrupts the plasma. Halo current forces are robustly mitigated with neon puffing, decreasing with increasing neon puff pressure, but the impact on power loads is still under investigation.

4.5 – 'Disruption Mitigation at JET', P. deVries

Over the past years various disruption mitigation experiments have been carried out. Gas injection has been applied to mitigate runaway generation and pre-empt the current quench. However, it was found that more gas and faster injection was required. In the 2004/5 shutdown a new gas injection valve has been installed at JET, dedicated to disruption mitigation. This fast gas valve, has a operating pressure between 5-35Bar and a response time of 0.5ms. Presently, the valve is connected to the main chamber via a 4m long pipe, reducing its injection speed to 10Bar-litres in 100ms. Fast mitigation experiments require it to be mounted closer to the vessel. For the coming campaigns several experiments are planned to test the properties of the valve and the effect of the gas injection onto the plasma and vessel conditions.

4.6 – ‘Far SOL ELM ion energies in Far SOL ELM ion energies in JET’, R. Pitts

Using a retarding field analyser (RFA) probe reciprocating into the JET SOL at the top LFS of the poloidal cross-section, ion currents due to Type I ELMs have been detected in a limited set of pure hydrogen discharges at low I_p, B_φ. The low electrical bandwidth of the probe and its associated electronics prevents conventional RFA voltage scanning on the ELM timescale and so the energy of the detected ions is inferred by fixing the internal ion retarding bias at the highest permitted value and combining a simple model of RFA function with predicted values for T_e, T_i and n_e in the ELM filament. These values are obtained using a new transient model for parallel energy loss from the filament which requires as input the ELM plasma parameters at the point of filament formation and the radial speed of filament propagation. The latter has been characterised experimentally in the past at JET, but the point of ELM filament origin is still a matter of speculation. To account for this, experimentally observed “mid-pedestal” values of T_i, T_e and n_e are used and a range of expected values at the RFA location computed for a ELM propagation distances corresponding to filament creation at the separatrix or at the top of the pedestal. The result is a predicted RFA ion current, representing the average value carrried by the ELM, which is in very good agreement with the observed values, bounded by the upper and lower limits corresponding to the spatial uncertainty in the ELM filament origin. A sensitivity test to radial propagation speed shows clearly that whilst a somewhat higher value to that assumed on the basis of previous measurements (v_{ELM} = 0.6 kms^{-1}) must also be considered compatible with the data, much lower values cannot. The transient model predicts that T_e falls much faster than T_i as the ELM filament propagates, an observation compatible both with the RFA data and earlier direct measurements of T_e on the ELM timescale in the JET far SOL. The fact that the model is based on parallel loss rates determined by sheath boundary conditions requires that it be seen as a description of the filament as propagating in the SOL (as in the simple blob or plasmoid theory) but connected to the divertor targets. The good agreement between model and experiment lends considerable support to this picture. The model has also been applied to estimate ELM ion impact energies on the ITER first wall at the outboard midplane for the Q_{DT} = 10 Type I ELM reference scenario, finding E_{ion} ~ 1.1 keV and thus considerably above even the tungsten sputtering threshold. It also predicts ~8% of the ELM expelled energy reaching the outboard limiters.

The presentation was based on two recently submitted JET papers, both currently under review: R.A. Pitts, W. Fundamenski et al, “Far SOL ELM ion energies in JET”, submitted to Nucl. Fusion and W. Fundamenski and R. A. Pitts, “A parallel transport model of tokamak power exhaust transients”, submitted to PPCF.

4.7 – ‘Heat and particle flux in the SOL and to the wall’, A. Herrmann

Experimental data from MAST, DIII-D and AUG on filaments and ELM extension in the SOL were presented. The observed behaviour of the filaments can be described phenomenologically as field aligned structures fed in the outer mid-plane with energy and/or particles. Most of the ELM energy is deposited continuously in the strike point region and only a fraction of the midplane loss is transported to the inner wall. Fast Thomson scattering and Beam emission spectroscopy reveals the origin of the filaments in the separatrix region. The radial velocity of the filaments as measured by reflectometry, turbulence measurements (Langmuir probes) and measurement of the delay between an ELM marker (H_) and the onset of the ion saturation current at reciprocating probes is below 1km/s, decelerated with radius at DIII-D and accelerated at MAST and AUG. The filaments are rotating toroidally and/or poloidally. Applying the velocity of toroidal rotation from CXRS-measurements in the pedestal a toroidal mode number of about 10-15 can be deduced for AUG and MAST.

The extension of the filaments into the SOL is measured with reciprocating probes, combined with thermography at AUG. The resulting e-folding length for the heat flux and the ion saturation current is a few centimetres. The inter ELM decay length tends to be shorter compared to the ELM decay length which increases with decreasing density (DIII-D). This results in an increasing contribution of ELMs to the first wall ion flux (80% at low densities, DIII-D). The magnetic field strength effects the variation (scatter) of the ELM size in the SOL rather than the e-folding length (AUG, MAST). The interaction of filaments with outboard limiters at AUG is very local with a few centimetres poloidal width. Its contribution to the ELM power balance is about 1%. The convective energy transport by filaments lasting about 100 μs is comparable to the mid-plane ELM energy loss.

4.8 – ‘ELM flux to the first walls on JT-60U’, N. Asakura

ELM plasma transport in the SOL and divertor of JT-60U was analyzed using an extensive array of reciprocating and target Langmuir probes. Large peaks in particle flux propagate radially to the first wall at 1.3 – 2.5 km/s (faster than in MAST, DIII-D and JET), arriving in less than the parallel transit time to the outer target, and deposit their associated heat load locally (over a few cm).

4.9 – 'Intermittent Transport of SOL/Divertor Plasmas in JT-60U', N. Ohno

The role of fluctuations in causing particle transport in the far SOL ($\Delta r \sim$60-70mm) in JT-60U was explored. Fluctuations are found to be 5-10 times larger during ELMy H-mode compared to L-mode and 4-5 times higher at the outboard mid-plane compared to the X-point.

4.10 - 'Summary of first wall loadings discussion', G. Counsell

Our knowledge of disruption power loads to the divertor has advanced sufficiently that an update to the ITER physics design guidelines is sensible. Data from most divertor tokamaks now show that the target heat flux width broadens by a factor of 5-10 during the thermal quench, larger than previous estimates, and that the plasma energy at the thermal quench is a factor 2-4 lower than that during the full performance phase of the discharge for most disruptions. There is however still a need to quantify the impact of vapour shielding on target heat loads during disruptions and the extent of heat loads on the upper divertor. Our understanding of first wall loads in unmitigated disruptions is much less developed, with data being exceptionally sparse and ill conditioned. All that was generally known is what fraction of disruption energy does not arrive at the divertor but much more work is needed before any useful physics guidelines for the distribution of this missing energy on other areas can be developed. Until this data becomes available, estimates for the lifetime of first wall components in ITER will be subject to large uncertainties.

On disruption mitigation, impurity injection (either as cryogenic pellets or through gas puffing) had been demonstrated to provide a reliable means of mitigating the impact of the current quench, both runaways and halo currents, on several devices. The impurity injection rate is important for mitigation, it being necessary to deliver sufficient electrons (bound and free) to the plasma in a time less than the current quench time. With gas puff injection, mixing light and heavy species gases can help by increasing the effective flow velocity of the heavy species and tailoring the mix could help optimise the thermal and current quench times. The interaction of impurity puffing with the plasma is complex with low order MHD appearing to play a role in penetration of impurities to the core, which might impact on the toroidal and poloidal uniformity of radiation on the first wall during the mitigation. It was noted that impurity gas puffing, which dominated the session, is only one possible technique and it is likely that a range of systems might need to be employed (including for example killer pellets) to handle different types of disruption. It was recommended that a working group be established to develop a list of high priority research items in this area over the next 2 years, which would include further assessment of key issues such as the impact of MHD activity in the plasma during the mitigation and the possibility that radiation flux alone to the first wall could cause melting for some classes of mitigated disruption.

On ELMs, it was agreed that there had been significant advances in our understanding of first wall loads, including a deeper understanding of their origin and structure provided by observations of filamentary structures (and associated theories) on most machines. Desposition of ELM energy onto radially protruding first wall components (such as poloidal limiters) has now been observed on many machines (at least 25% of W_{ELM} for large type I ELMs), although the poloidal and toroidal localisation of this energy and it's temporal behaviour are not well understood. Extrapolations for the limiter power load during type I ELMs in ITER indicate potentially large melting and erosion rates for some credible deposition scenarios. However, it was recognised that divertor power loads may provide the 0^{th} order limit on allowable ELM size and might actually necessitate operation in 'small' ELMs regimes (either naturally small type I ELMs, type II/III, QH etc., or actively mitigated, such as by rapid pellets or magnetic perturbation). It was likely that limiter erosion would not represent a significant problem in small ELM regimes, however even less was known about energy deposition outside the divertor in these regimes and study was required, both for the first wall and upper divertor.

As a result of the discussion on first wall loads, it was agreed that the Divertor and SOL ITPA Group would aim to produce separate position papers for ITER on the three main topics in this session (Disruption power loads, ELM power loads and disruption mitigation strategies).

Session 5: ‘Next ITPA SOL/divertor meeting’

Location: Several locations are being considered for the next 2 meetings. The primary choice for the meeting following this (so #7) is to have it before the abstract selection meeting for PSI17. This will likely be at the end of January or beginning of February in China. It is possible it would be elsewhere but that will be up to the PSI chairman – Jiangang Li. Having the meeting in conjunction with the abstract selection meeting makes it easier for the 5-6 people who are members of both committees to attend.

Assuming that the 7th meeting is in China, then the time of the 8th meeting would be best in July. The choice from the point of balance would be to have the meeting in Japan as we have not been there in a while. However, that would mean 2 meetings in Asia in a row, not to mention that there will be a PSI meeting in May in China – so 3 trips to Asia! Alternatively we could try and have the 8th meeting back in the US or Europe. A third possibility is to wait till the fall and have the 8th meeting to coincide with IAEA but this will be back in China!

Potential Topics for the next meeting – A number were discussed and found support:

1) D/T retention in surfaces (Long-pulse, saturation….., processes involved)

2) Wall conditioning (characterizing the effect on confinement, methods of steady-state replenishment compatible with ITER conditions)

3) Startup studies (flux consumption, operation on outside limiter)

4) Flow characterization

5) melt layer stability and vapor shielding

6) spectroscopy of hydrocarbons in low-T plasmas

Session 6: ‘Mixed material effects’

6.1 - ‘Results on alloy formation of Be and W’**,** J. Roth

The interaction of Be with W to form low melting point W-beryllides was performed in a collaboration of IPP Garching and PISCES, UCSD in three procedures: The interaction of a 235 nm W layers deposited onto polycrystalline Be, the slow evaporation of 5 nm Be onto W, and the exposure of W to a Be-seeded D-plasma.

The results show that W-beryllide phases are real, and bear the potential of a major malfunction: At 1070 K W-beryllide phases can form which melt below 1700 K. For W on Be this phase has clearly been identified as $Be_{12}W$ through ion beam and XPS analysis. The alloy formation appears to be limited by the diffusion of Be in the alloy.

However, under typical ITER conditions (0.1% Be in incident D flux) only a thin Be_2W layer will form, excess Be will sputter/reflect/evaporate and deposit elsewhere. Studies using higher Be plasma concentrations (up to 1%) are underway in PISCES-B. The stability of the beryllide layer during ELM-like transient heat loads will be investigated in PISCES-B

6.2 – 'PISCES Be/C mixed materials studies'**,** R. Doerner
Mixed material investigations of the Be-C system in the PISCES-B facility were described. Be-rich surface layers form on plasma-exposed C samples and lead to co-deposits that are composed almost entirely of Be and D. Although the level of co-deposition of Be and D is on the same order as that expected from co-deposition of C and D, the D retained in Be is released thermally at much lower temperature. When C is included in the co-deposits an amount of the D retained in the co-deposits is trapped much more strongly and cannot be easily removed by baking.

6.4 - 'The IPP dual beam experiment – first results on simultaneous irradiation of tungsten by deuterium and carbon'**,** K. Krieger
Plasma facing surfaces in tokamaks are generally subject to impact of a mixture of fuel ions and impurities. To study the basic processes involved in multi-species bombardment, a dual beam experiment has been commissioned at IPP Garching. First results are presented on erosion rates and fuel retention under simultaneous irradiation of tungsten at room temperature with deuterium and carbon. In contract to results obtained for a 25% fraction of C, irradiation of C with D and a 5% fraction of C yields continuous erosion of W enhanced significantly over pure D bombardment. Carbon is implanted in the surface and evolving to a mixed surface layer with stationary thickness and composition. Although the principal processes have been identified, the new results are a challenge for the existing models.

6.4 'Mixed materials discussion and conclusions'. J. Roth:

The following discussion of Mixed Material Session was structured according to the three main expected material interactions and their potential specific problems:

Be-W system: This system bears the potential for major malfunction. Layers may form with low melting point (< 1700°C) and may have bad thermal contact to substrate. The process must be discussed under the most likely ITER scenario, i.e. the exposure of W to a Be-seeded plasma, and all processes such as deposition, reflection, re-erosion and evaporation have to be taken into account to assess the failure potential. Disruptions may deposit up to 10 kg of evaporated Be on W divertor structures. Be-W system needs extensive investigation, appears most critical.

Be-C system: Experiments in PISCES show, that deposited Be reduces the chemical erosion of C. Still, for interpretation of results in JET, the simultaneous Be and C deposition from the D-plasma needs to be investigated.
Further needed investigations are: What is influence on T retention? The Be/C mixture exhibits similar D co-deposition at 300 K, but D release occurs at much lower temperatures than for carbon.

W-C system: W erosion in present devices (ASDEX Upgrade) is enhanced by C impurity ions in the plasma
As W is eroded by C ions, the C is removed by D ions leaving the clean W surface exposed. W-carbide formation occurs at temperatures above 700°C and is well understood. Deviations from thermo-dynamical phase diagrams occur due to the highly non-equilibrium conditions.

In the discussion it was pointed out that a much larger variety of mixing compositions may occur during the history of different fusion experiments which lead to hardly reversible wall conditions. Much larger varieties of mixing conditions should be studied in ion beam experiments, plasma simulators and fusion experiments, such as JET, ASDEX Upgrade etc.
Under clean conditions, still seed impurities for radiation are present and may contribute to material modifications.
It was concluded that it seems impossible to cover all possible combinations and simulate identical ITER divertor conditions. This requires that test experiments need to be extensively

simulated in computer codes and the conditions be extrapolated using validated data. The main pathway needs to be finally explored in the fusion experiments, while laboratory experiments remain necessary to avoid major pitfalls.

Session 7 : Estimates of the effects of Be operation in ITER. Leader - A. Loarte

This session reviewed the implications of the use of a Be wall in ITER, including the lifetime of the wall, Be migration, implications of the Be-wall for T-retention and removal in ITER and general considerations on the operability of ITER associated with the main chamber being made of Be.

7.1. Introduction. A. Loarte

This talk reviewed the physics basis which lead to the decision of a Be wall in ITER and discussed the needs for a re-evaluation of this decision in view of the new findings in tokamaks since 1998. Be was chosen for first wall material in ITER because its low Z compatibility with a wide operating range and low T-retention under the following assumptions:

a) The flux of ions on the wall is determined by diffusive anomalous transport and C–X as modelled by B2-Eirene. The typical ratio of divertor ion flux to main wall ion flux is larger than 100. This is associated with a low Be erosion of the wall during normal operation of the device and small associated T-retention.

b) There are no significant power loads (compared to divertor) on the wall during ELMs, as the distance from the separatrix to the first Be-PFC elements is more than 7 e-folding lengths for the power energy flux during ELMs (no significant broadening of the ELM power flux was assumed).

c) There are only small power loads (compared to divertor) on the wall during disruptions, as the distance from the separatrix to the first Be-PFC elements is more than 2 e-folding lengths for the power energy flux during disruptions (only a factor of 3 broadening of the disruption power flux was assumed).

d) Issues related to material-mixing, its influence on T-retention and cleaning techniques were not considered in detail.

7.2. Be wall erosion in steady-state (Kallenbach, Philipps, Federici, Roth).

Estimates of the average Be wall erosion presented were based on physical sputtering of Be by ion impact and C-X neutrals from modelling results with B2-Eirene and various experimental extrapolations with the following results :

- B2-Eirene extrapolations. Depending on modelling assumption on the plasma in the far SOL the expected Be erosion is 4.5 10^{21} s^{-1} (Federici) to 8.0 10^{21} s^{-1} (Roth).
- Experimental extrapolations range from 2.0 10^{21} s^{-1} (Philipps) to 1.2 10^{23} s^{-1} (Kallenbach) depending on whether the presently measured wall fluxes are assumed to be similar to those in JET in absolute amount or whether an empirical relation which fits the JET and ASDEX Upgrade density and wall fluxes is used.

Typical rates taking into account that the main chamber Be will probably form BeO are a factor of 2 lower. The resolution of this issue relies on the experimental scaling of the ion fluxes on the wall and whether there is a favourable or unfavourable size scaling of the anomalous particle flux density or not. If the total flux on the wall in ITER is similar to today's experiment, the expected erosion rate of ~ 40 nm/discharge allows ITER operation with a Be wall for its expected operating life (Federici). On the contrary if the anomalous ion flux density in ITER is similar to today's experiment, Be erosion will be in the range of ~1 μm/discharge and the Be wall lifetime limited to several thousand full performance discharges (Kallenbach). The above values represent average values of Be erosion of main wall components, the local erosion of components which are closer to the plasma such as the upper x-point modules and limiters could be significantly higher than these average values.

7.3. Be wall erosion under transients (Herrmann, Whyte).

Estimates of the Be erosion under ELMs based on extrapolations from measurements in ASDEX Upgrade indicate that for ELM loads that do not cause significant target erosion (sublimation of C or melting of W), they will not cause Be melting of main wall elements, even of the upper X-point modules (Herrmann). These estimates are in agreement with estimates done by A. Loarte and presented at the PSI 2004 in Portland. However, significant melting of the Be limiter could occur in these conditions. The situation is much less clear for disruption loads in which significant melting of the Be wall and of the Be limiter may occur even if the energy is spread over a large area of the wall during the thermal quench (Herrmann).

This will be indeed the case for mitigated disruption by impurity injection and VDEs (Whyte). Ideal rapid, quasi-uniform radiation flash desired for disruption mitigation can lead to melting of the ~500 m^2 beryllium wall. Q=10 plasma termination produces a large-area thin melt layer (~10's microns) that remains molten during thermal quench ~ 0.2 - 2 ms. Because the layers melt simultaneously with the dissipation of the plasma thermal energy, the layer is exposed to a radial inward, mobilizing JxB force resulting from induced toroidal eddy currents caused by the expulsion of the diamagnetic toroidal flux through the highly electrically conducting Be wall. The body acceleration is ~ 100 g and the layer displacement is found to be > mm with terminal velocity ~ m/s. Surface tension acts as a stabilizing force, but for nominal tile size is insufficient to balance JxB (~ 1mm surface "dimples" may work). Since little plasma is present to stop the melt layer, it is expected that 10-50 kg of Be will likely "splash" onto other main-wall surfaces and down into the divertor by gravity. Similar quantities of Be loss were found for an unmitigated VDE with thicker melt layers (0.5 mm) spread over a smaller area (~30 m^2). The mobilized Be will likely be in the form of droplet due to the surface tension and these will present non-uniform surfaces, both at the location of melting and where it splatters, for heat loading in subsequent discharges.

Finally, it was highlighted that concentrated energy deposition from fast particles could be a problem for the Be wall in ITER, but no estimates of these loads in ITER were considered (Herrmann). In view of the results, developing regimes which are acceptable for the divertor lifetime will probably lead to regimes which are acceptable for Be erosion of the main wall with respect to ELMs. The open issues are the ELM loads on the ITER Be limiter and the dynamics of the molten layer during disruption, mitigated disruptions and VDEs. Experiments in simulation facilities and tokamaks are necessary to address the issues besides more refined estimates of the expected erosion.

7.4. Main wall eroded Be migration to the divertor and C/W migration onto the Be wall in ITER (Philipps, Roth, Kukushkin, Coster).

Two approaches were presented to do these estimates: a) one based on the experimental evidence, which indicates that most eroded material from the main wall ends up at the inner divertor and that the main wall is a net erosion zone; and b) modelling by B2-Eierene of the erosion deposition balance. In the first estimate the main conclusion is that most of the eroded Be will be transported to the divertor (mostly to the inner one) and will deposit there in areas with direct plasma contact (Philipps, Roth). Following this empirical picture, it is expected that C and/or W escaping the divertor will only be found at the wall transitorily and will finally migrate to the divertor (predominantly the inner) were it will deposit permanently. Some deposition of C and W could be expected in shadowed areas of the main wall but this is not expected to be a major contributor to overall deposition. Modelling with B2-Eirene provides a much more complicated picture for main wall/divertor erosion and material migration following the studies by Kukushkin and Coster. The divertors are found to be deposition zones for the eroded material both from the divertor and the wall, except close to the strike zones where significant carbon erosion can take place. The main wall has areas which can be deposition dominated and erosion dominated. Both modelling studies show material deposition from the divertor (Carbon) on the LFS of the main wall and erosion (of Be) in the HFS. The rate of carbon deposition is obviously affected by the rate of re-erosion of the deposits. If this is large enough, the balance from deposition to erosion and its location can change significantly in the calculations. To clarify which features of this modelling are ITER specific and which ones are associated with inaccuracies in the modelling of SOL flows, which are seen to be present in present experiments, and re-erosion of deposits requires a detailed comparison of similar modelling results for actual experiments. Using markers to determine with high precision the migration of materials in the experiments is highly recommended as it facilitates the interpretation of experiments and validation of models.

7.5. Implications of a Be wall on T retention and removal in ITER (Philipps, Roth, Federici, Counsell).

The typical concentration of T in Be is very low, typically in the 1-5 % level, which amounts to a retention in the range of 0.1-0.6 g-T/discharge depending on modelling assumptions (Philipps, Roth). This amount is similar to that expected for carbon chemical erosion if the target is assumed to be covered by Be following ERO modelling and simple estimates (Roth, Counsell). However significant T release is expected if these Be-co deposits are brought up to temperatures higher than 600 K (Federici, Roth). This is likely to be the case at the ITER divertor, as Be deposits only occurs in plasma wetted areas of PFCs (Federici, Roth). If Be is expected to be in the form of BeO and co-deposit at the divertor the retention could be larger. Estimates of the T retention on BeO based on the incoming of oxygen due to the reference leak rate of the ITER vessel or by purity of the initial material show that the amount of T retained is similar or smaller than that associated with pure Be for ITER operation excluding transients (Counsell). It is expected that the T retained in the Be deposits will be easily removed by energy deposition of the plasma itself which keeps the divertor surface temperature high. If this is not the case, the present foreseen techniques for T removal from C, such as photonic cleaning or oxidation will be difficult to apply, if not impossible. In summary, for Be erosion rates which are acceptable with the lifetime of the Be wall the amount of T retained in ITER will be in the 0.1-0.6 g-T/discharge at most. The deposits will be localised in areas where plasma exposure should be effective to remove the deposited T. Oxygen and C content in the layers makes the retention larger but still acceptable by control of the divertor target temperature by plasma exposure within standard ranges of ITER operation. Despite this positive picture, it is important to highlight that none of the foreseen schemes to remove T from hydrocarbon deposits seem applicable for Be and some T-removal technique from Be deposits should be developed in case it is necessary. Detailed analysis of Be-containing samples exposed to plasmas in tokamaks or divertor simulators and dedicated experiments to remove the retained fuel from such samples are required to progress further in this area.

7.6. Operability of ITER with a Be main wall/limiter including effect of PFCs damage due to transients (Kallenbach, Counsell, Loarte, Federici, Whyte).

The major issues in this area refer to the damage of the Be wall and the effect of a damaged wall on ITER operations. Main Be wall erosion between ELMs, although potentially a lifetime issue for the wall does not seem to be a potential threat to the purity of ITER plasmas (Counsell, Kallenbach). Damage by ELMs is likely to cause irregularities on Be-wall components in the range of 10-100 μm which do not appear to pose a danger for the integrity of the components (Counsell, Loarte, Federici). These irregularities could, however, compromise the power handling capabilities of the Be-PFCs, in particular of the limiter, which is required to have a good power handling and will be exposed to direct plasma contact during the ramp up and ramp down phases of the discharge (Counsell, Federici). The expected effect of such occasional Be melting of the first wall on ITER plasmas is the occurrence of transient increases of the Be impurity concentration and decrease of the fusion performance. Causing a radiative collapse of an ITER plasma by Be influx requires about a 200 g influx of Be (Loarte). Depending on the frequency of such influxes and the amount of Be that gets into the plasma, the real consequences for the fusion performance of ITER are obviously different. Disruptions (even if mitigated) and VDEs, on the contrary, have the potential to cause serious damage to the Be wall (Counsell, Whyte, Federici) and can cause several kgs of Be to be eroded from the wall after every one of these events. In order to progress further in this field and provide better estimates for ITER it is, therefore, necessary to: a) collect experimental evidence of plasma response to well know amounts of low Z solid impurity influxes (to determine plasma resilience to the ELM Be-influxes in ITER, b) carry out modelling and experiments of the Be-melt layer dynamics under ELM and disruption loads and c) develop scenarios to ameliorate the effects of disruptions and VDEs on the Be wall.

7.7. Status of the JET ITER-like wall project. V. Philipps.

Be blanks for the main wall, Be cladding, and new CFC tiles to be coated with W for the ITER-like wall project have been tendered. The total amount of Be tiles in the main chamber will be somewhat reduced due to budget problems with ordering of the Be blanks. The possible use of Be plasma-sprayed Inconel tiles between the inner wall guard limiters will be decided soon. With respect to W coating of the CFC divertor tiles, prototype coatings with 5 and 10μm and "thick" (>200μm) W-layers on JET Dunlop CFC are underway, to be tested in the fall under high heat load conditions. A bulk W-tile is under construction for the horizontal divertor tile (number 5, LBSRP tile). The project also includes upgrades of diagnostics (spectroscopy, IR and deposition probes).

Session 8: 'The ITER dome'

8.1 - 'Original strategy for the dome and its physical functions', M. Shimada

The present strategy of the dome is to investigate the physical functions of the dome and pros and cons of its removal and to investigate the consequences of possible dome removal on diagnostics and other in-vessel components. Originally the dome was intended to reduce the neutral backflow from the divertor to the main plasma to facilitate the partial detachment for the enhancement of radiative cooling and particle exhaust and to protect the liner and the pumping slot from the plasma. However, much deposition of tritium, carbon and dust is expected at the backside of the dome and the liner, where measurement and removal are difficult. Furthermore, experimental and modeling studies suggest that the shielding effect including the neutral-neutral collisions could reduce the neutral backflow. Therefore a proposal of removing the dome has been made to possibly facilitate the measurement and removal of tritium, carbon and dust. For the improved argument of the benefit of the dome removal, it is necessary to estimate the deposition rate on the backside of the dome with consideration of the effect of parasitic plasma and to estimate the deposition rate of dust below the pumping slot.

8.3 - 'ITER calculations of divertor performance with & without the dome', A. Kukushkin

In order to study the effect that the dome removal would have on the plasma performance in ITER, a number of modeling runs was done with the B2-Eirene code including the recent upgrade of the neutral particle transport model (neutral-neutral collisions, improved molecular reactions, Lyman radiation transport). The results indicate that the most significant effect would be an increase of the required pumping speed at the pump duct entrance by a factor 3, needed to compensate for the enhanced neutral particle absorption by the plasma. Other aspects of the dome removal, such as placing the divertor diagnostics, reduction of neutron shielding, design of the pumping slots, and so on, must still be thoroughly considered before the removal of the dome could be recommended for the ITER design.

8.4 - 'Interpretive analysis of the C-Mod divertor for partially detached plasmas: possible implications for the ITER dome', Lisgo

Tangential camera images in D_γ light have shown significant emission from the C-Mod private flux region (PFR) for plasmas that are weakly attached in the outer SOL (the inner SOL is almost always partially detached), with the emitting region extending from the separatrix to near the divertor floor. The transport processes generating this extended plasma structure in the PFR are poorly understood, and may result from the collisional nature of the C-Mod divertor. OSM-EIRENE modeling suggests that the divertor neutral pressure is primarily controlled by the PFR plasma in this instance, a result to consider when discussing the physics implications of including a dome in the ITER PFR (since the collisionality of the ITER divertor is also expected to be high). The C-Mod study indicates that the neutral processes included in the EIRENE neutral code are sufficient for first order estimates of divertor neutral pressures in ITER, provided that a reasonably accurate description of the ITER divertor plasma is available.

8.5 - ‘Summaries of talks on the ITER dome’, D. Whyte

Consideration is being given to removing the dome from the ITER divertor design. This is motivated by the desire to inhibit possible tritium codeposition and dust accumulation underneath the dome, and to ease geometric access to the divertor for tritium removal by optical cleaning techniques.

Originally the dome was primarily intended to reduce the neutral backflow from the divertor to the main plasma, to facilitate partial detachment, and to enhance particle exhaust. However both experimental data (e.g. septum removal in JET) and the recent improvement of neutral-neutral collisions in the B2-EIRENE modeling appears to have removed these motivations. No significant change was found in divertor parameters (peak heat load, radiative cooling) or upstream plasma (density limit, impurity content) as a function of the absence or presence of dome. The most significant effect would be an increase of the pumping speed at the pump duct entrance by a factor 3, needed to compensate for the enhanced neutral particle absorption by the plasma. This pumping speed appears to be within the present capability in ITER. The issue of increasing neutral leakage out of the divertor by removing the dome was raised. It was pointed out that modeled neutral leakage from the private-flux region (PFR) was very small, due to the much larger ratio of size to neutral mean free path in ITER as compared to present devices. Interpretative modeling of the C-Mod divertor, which most closely approaches ITER divertor collisionality, showed that the inclusion of neutral-neutral collisions was required to reproduce the measured baffle pressure, generally validating the new B2-EIRENE results for ITER. An additional insight from the C-Mod modeling was that the presence of dense cold plasma in the PFR plasma largely determined the baffle neutral pressure. Since the reasons for this PFR plasma are poorly understood this raises questions about the implication of a dome in the PFR of ITER.

Another important purpose of the dome, which evolved from its long-term presence in the ITER design, is the protection of components associated with pumping (liner, pump slot) and optical components of divertor diagnostics. The most worrying implications of dome removal appear to be divertor diagnostic losses: IR thermography of the target plates, spatially resolved visible impurity spectroscopy, and the laser system for measuring divertor plate erosion are all lost. The absence of these diagnostics, particularly the IR required to measured target plate temperature, could have a strong impact on divertor operation. A full evaluation of the impact on diagnostics, including determining the consequences to the machine operation and physics studies arising from the reduced measurement capability, should be completed before a decision to remove the dome is taken.

Furthermore, no detailed engineering design is yet available as to what would replace the dome structure in ITER. For instance, the detailed technical aspects of protecting the pumping slots, liner and lower floor from plasma and neutrons were not yet available. Such a design is required for a realistic assessment of the diagnostic implications, as well as the implications for tritium removal,.

A further concern raised during the session was the level of certainty about the presence of tritium codeposits and dust under the dome in ITER. It was pointed out that present tokamak experience can be misleading. Due to the fact that neutrals are in the fluid regime in the ITER divertor, line-of-sight carbon deposition, which seems to dominate deposition in present devices, will not occur in ITER. Dust formation and transport is even less certain. It was also unclear if the simple removal of the dome would ameliorate tritium retention or recovery, since the exposed pump liner may serve the same purpose of trapping and hiding tritium and dust. Again a more detailed alternative divertor design seems required.

Conclusions

No consensus was reached on the balance of benefits vs. losses with regard to the dome removal. While tritium and dust recovery are clearly important issues to the operational viability of ITER, it remains unclear if there is yet enough evidence to remove the dome based on present data or modeling. The dome removal clearly has important consequences to divertor diagnosis, so this must be balanced against the tritium inventory considerations.

Suggestions for further work on this topic include:

- Further studies on the origin and effect of the PFR plasma.
- Initiating an engineering design as to what would replace the dome structure in ITER considering protection of the vessel and pumping slots.
- Assessment of the neutron shielding issues with the dome removal.

- A full evaluation of the impact on diagnostics, including determining the consequences to the machine operation and physics studies arising from the reduced measurement capability.
- The effect of ITER neutral collisionality on carbon atom and molecule deposition patterns with and without the dome (likely coupled to PRF plasma studies).
- Empirical assessments on carbon/deuterium/dust accumulation behind “dome-like” structures in present devices.

Appendix A - Agenda

1st Day: July 4th (Mon.)

Day	Session	Start time	Duration	Speaker	Topic/title
Monday	0	9:15	0:15	Hidalgo/Lipschultz	Introduction
	1 D/T inventories (surfaces and sides of tiles				Chair - V. Philipps
		9:30	0:05	V. Philipps	Introduction
		9:35	0:25	K. Krieger	Recent results on carbon deposition and fuel retention in gaps
		10:00	0:20	D. Whyte	Analysis of C-Mod tiles for B/D co-deposition on tile sides
		10:20	0:20	V. Rohde	Recent results on carbon migration and deposition in AUG
		10:40	0:15	Philipps/Rubel	13C injection at the outer divertor in JET
		10:55	0:20	C. Grisolia	Recent activities on photocleaning in EU
		11:15	0:20	Kreter/Philipps	Oxygen cleaning activities in TEXTOR
		11:35	0:20	coffee break	
		11:55	0:20	J. Hu	Oxidation wall conditioning in HT-7
		12:15	0:20	Ashikawa	ICRF H/D removal in LHD
		12:35	0:20	Tabares	N2 seeding experiments in AUG
		12:55	0:20	Tsitrone	Long pulse operation and related wall saturation effects
		13:15	0:20	Philipps	Discussion
		13:35	1:15	Lunch	
		14:50	1:00		Discussion continued
	2 Dust				Chair - S. Krasheninnikov
		15:50	0:15	Krasheninnikov	US dust meeting
		16:05	0:20	Ashikawa/Ohno	JT-60 & LHD results
		16:25	0:15	Maingi	Detection of dust particles on remote surfaces
		16:40	0:15	Kirnev	T-10 results
		16:55	0:15	Bonnin	Modelling of dust formation in plasmas
		17:10	0:15	Fenstermacher	DIII-D results
		17:25	0:20	Coffee	
		17:45	0:15	Lipschultz	Enormous quantities of dust in C-Mod and the effects
		18:00	0:15	Counsell	Modelling of dust dynamics
		18:15	0:15	Krasheninnikov	Modeling of dust dynamics and transport in ITER with the code DUSTT
		18:30	1:00	Krasheninnikov	Discussion

2nd Day: July 5th (Thu.)

Day	Session	Start time	Duration	Speaker	Topic/title
Tuesday					
	3 High-Z experience				Chair - A. Kallenbach
		9:30	0:15	Kallenbach	ASDEX Upgrade W operation
		9:45	0:15	Kreter	High temperature erosion and melting of W in TEXTOR
		10:00	0:25	Lipschultz	Removal of B from C-Mod and effect of boronization
		10:25	0:20	Ueda/Nakano	W-tile experience from JT-60U and laboratories
		10:45	0:15	Ueda/Yoshida	Conclusions about the use of high-Z materials
		11:00	0:20	Coffee	
		11:20	1:15	Kallenbach	Discussion
	4 First wall loadings & joint meeting with MH				Chair - G. Counsell
		12:35	0:20	Loarte	EU-PWI disruption wall loadings
		12:55	0:20	Kawano	JT-60U disruption mitigation
		13:15	1:15	Lunch	
		14:30	0:20	Whyte	C-Mod & DIII-D status of disruption mitigation work
		14:50	0:15	Guenter	AUG disruption mitigation
		15:05	0:10	deVries	JET disruption mitigation
		15:15	0:30	Wesley	Discussion of first wall loading database
		15:45	0:20	Pitts	JET ELM measurements/modeling
		16:05	0:30	Herrmann	AUG, DIII-D & MAST measurements of wall fluxes
		16:35	0:20	Coffee break	
		16:55	0:20	Asakura	JT-60U ELM fluxes to walls
		17:15	0:15	Ohno	Analysis of intermittent transport in JT-60U
		17:30	1:15	Counsell	Discussion of ELM and disruption wall fluxes
	5 Next meeting				Chair - Lipschultz
		18:45	0:45	Lipschultz	toics and location for next meeting

3rd Day: July 6th (Wed.)

Day	Session	Start time	Duration	Speaker	Topic/title
Wednesday					
	6 Mxed materials effects				Chair - J. Roth
		9:30	0:20	J. Roth	IPP results on alloy formation of Be and W
		9:50	0:20	R. Doerner	PISCES Be/C mixed materials studies
		10:10	0:20	K. Krieger	Studies of mixed material erosion/deposition at the Garching dual beam experiment
		10:30	1:00	Roth	Discussion
	7 Estimate of the effects of Be operation in IT				Chair - A. Loarte (5 minute showing of calculation, 10 minutes of discussion each)
		11:30	0:10	Loarte	Overview
		11:40	0:20	Coffee break	
		12:00	0:35	several	Be-wall erosion during steady-state and transient plasma conditions including its spatial distribution
				A. Kallenbach	Be wall erosion during steady-state
				D. Whyte	Be wall erosion during steady-state and transients (ELMs and disruptions
				A. Herrmann	Be wall erosion during transients (ELMs and disruptions)
				G. Federici	(ITER-IT estimates of Be wall erosion during steady-state and transient
		12:35	0:35	several	Main wall eroded Be migration to the divertor and C/W migration onto the Be wall in ITER
				V. Philipps	Experimental estimates of Be and C/W migration in ITER, including information from M. Rubel)
				D. Coster	Modelling estimates of Be and C/W migration in ITER
				D. Whyte	Estimates of Be and C/W migration in ITER
				A. Kukushkin	ITER-IT estimates of material migration in ITER
		13:10	0:35	several	Implications of a Be wall on T retention and removal in ITER (including mixed material formation
				V. Philipps	Experimental estimates of T retention in ITER with Be/C/W wall and implications for removal methods
				J. Roth	Experimental estimates of T retention in ITER with Be/C/W wall and implications for removal methods
				G. Counsell	Experimental estimates of T retention in ITER with Be/C/W wall and implications for removal methods
				G. Federici	ITER-IT estimates of T retention in ITER with Be/C/W wall and implications for removal methods
		13:45	1:15	Lunch	
		15:00	0:35	several	Operability of ITER with a Be main wall/limiter including effect of PFCs damage due to tranients
				D. Whyte	Evaluation of the implications of Be use for limiter and main wall components in ITER
				A. Loarte	Evaluation of the implications of Be use for limiter and main wall components in ITER
				G. Counsell	Evaluation of the implications of Be use for limiter and main wall components in ITER
				G. Federici	ITER-IT view of the implications of Be use for limiter and main wall components in ITER
		15:35	0:25	V. Philipps	Plans and status of Be wall experiment in JET.
		16:00	0:45	Loarte	Discussion
	8 ITER dome				Chair - D. Whyte
		16:45	0:20	Shimada	ITER presentation on original strategy for the dome and why to remove it
		17:05	0:20	Kukushkin	ITER calculations of divertor performance with & without dome
		17:25	0:15	Itami	Impact of dome removal on ITER divertor diagnostics
		17:40	0:20	Lisgo	Extrapolation of a diffusive neutral solution to ITER
		18:00	1:15	Whyte	Discussion of how to make progress

4th Day: July 7th (Thu.)

Day	Session	Start time	Duration	Speaker	Topic/title
Thursday					
	9 IEA/ITPA				Chair - Lipschultz
		9:30	2:00		1 vg updates by current IEA/ITPA collaboration leaders
		11:30	0:20	coffee break	
		11:30	0:30		discussion of new collaborations
	10 TPB (1 hour?)				Chair - Asakura
		11:50	1:00	discussion of current status	

Appendix B - Meeting attendees

Total 45 : EU(18)+JP (6)+US(11)+CN(2)+RF(2)+ITER(6)

Attendee	**Institute**	**e-mail**
EU(18):		
C. Hidalgo	(CIEMAT)	carlos.hidalgo@ciemat.es
F. Tabares	(CIEMAT)	tabares@ciemat.es
V. Philipps	(FZJ)	v.philipps@fz-juelich.de
A. Kreter	(FZJ)	a.kreter@fz-juelich.de
O. Ogorodnikova	(FZJ)	o.ogorodnikova@fz-juelich.de
K. Krieger	(IPP-Garching)	krieger@ipp.mpg.de
V. Rohde	(IPP-Garching)	rohde@ipp.mpg.de
A. Kallenbach	(IPP-Garching)	Arne.Kallenbach@ipp.mpg.de
J. Roth	(IPP-Garching)	roth@ipp.mpg.de
A. Herrmann	(IPP-Garching)	albrecht.herrmann@ipp.mpg.de
D. Coster	(IPP-Garching)	dpc@ipp.mpg.de
C. Grisolia	(CEA)	christian.grisolia@cea.fr
E. Tsitrone	(CEA)	emmanuelle.tsitrone@cea.fr
X. Bonnin	(CEA)	bonnin@limhp.univ-paris13.fr
G. Counsell	(UKAEA)	glenn.counsell@ukaea.org.uk
R. Pitts	(CRPP)	Richard.Pitts@epfl.ch
A. Kleyn	(FOM)	kleyn@rijnh.nl
A. Loarte	(EFDA-Garching)	loartea@ipp.mpg.de
JP (6):		
Nobuyuki Asakura	(JAERI)	asakuran@fusion.naka.jaeri.go.jp
Naoko Ashikawa	(NIFS)	ashikawa@lhd.nifs.ac.jp
Tomohide Nakano	(JAERI)	nakanot@fusion.naka.jaeri.go.jp
Noriyasu Ohno	(Nagoya Univ.)	ohno@ees.nagoya-u.ac.jp
Seio Sengoku	(JAERI)	sengokus@fusion.naka.jaeri.go.jp
Yoshio Ueda	(Osaka Univ.)	yueda@eie.eng.osaka-u.ac.jp
US(11):		
Bruce Lipschultz	(MIT)	blip@psfc.mit.edu
Peter Stangeby	(Univ. Toronto/GA)	stangeby@fusion.gat.com
Sergei Krashnenikov	(UCSD)	skrash@mechanics.ucsd.edu
Dennis Whyte,	(Univ. Wisconsin)	whyte@engr.wisc.edu
Max Festenmacher	(GA)	Max.Fenstermacher@gat.com
George Tynan	(UCSD)	gtynan@ucsd.edu
Russ Doerner	(UCSD)	rdoerner@ucsd.edu
Steve Lisgo	(Univ. Toronto)	steven@starfire.utias.utoronto.ca
Rajesh Maingi	(PPPL)	rmaingi@pppl.gov
Mathias Groth	(GA)	groth@fusion.gat.com
Vlad Soukhanovskii	(LLNL)	vlad@pppl.gov
CN (2):		
Yu Yang	(IPP)	yangyu@ipp.ac.cn
Jiansheng Hu	(IPP)	hujs@ipp.ac.cn
RF(2)		
Valery Kurnaev	(MEPhI, State Univ.)	kurnaev@plasma.mephi.ru
Gennady Kirnev	(Kurchatov Inst.)	kirnev@nfi.kiae.ru
ITER IT (6):		
Gianfranco Federici	(ITER IT)	federig@itereu.de

Kukushkin Andrei	(ITER IT)	kukusha@itereu.de
Michiya Shimada	(ITER IT)	shimadm@itergps.naka.jaeri.go.jp
Kiyoshi Itami	(ITER IT)	itamik@fusion.naka.jaeri.go.jp
Alan Costley	(ITER IT)	costlea@itergps.naka.jaeri.go.jp
Horst D. Pacher	(ITER IT/INRS-EMT)	pacher@emt.inrs.ca