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CONFERENCE REPORT

Report on the 11th European Fusion Physics Workshop (Heraklion, Crete, 8–10 December 2003)

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Abstract

The 11th EFPW took place in December 2003 at Heraklion in Crete, hosted by the Association EURATOM-Greece and the FORTH Institute, Heraklion and sponsored by the European Commission. Within the overall theme of 'plasma–wall interactions (PWI) and their implications for impurity generation and transport', four topics of importance to the future development of magnetically confined fusion were discussed in detail. Key PWI issues for ITER were also reviewed, the programmes of the two European physics task forces, on PWI and on integrated tokamak modelling, were discussed, and several topical reviews on key physics R&D issues for ITER were presented. The main issues discussed and the areas identified as requiring further study are summarized here.

1. Introduction

At the 11th European Fusion Physics Workshop four topics of particular importance for the understanding and control of plasma–wall interactions (PWI) in fusion devices and their implications for the choice of first wall materials in ITER were reviewed:

(i) Erosion processes and impurity generation;
(ii) Material transport, redeposition and tritium retention;
(iii) Transport processes in the divertor and scrape-off layer: experiment and modelling;
(iv) Core impurity transport.

In addition, key PWI issues for ITER were reviewed, the programmes of the two European physics task forces, on PWI and on integrated tokamak modelling (ITM), were presented and discussed, and the final workshop session was devoted to topical reviews of several key physics R&D issues for ITER:

(i) Physics of the H-mode pedestal;
(ii) Tolerable edge conditions for fusion plasma operation;
(iii) Active control of MHD instabilities;
(iv) Development of common hybrid/steady-state scenarios.

It is currently foreseen that the plasma facing materials in ITER will consist of beryllium on the first wall of the main plasma chamber, carbon fibre composites (CFCs) on the high power handling surfaces in the divertor and tungsten on all other divertor surfaces exposed to plasma. It is known that the high power and particle fluxes in ITER represent a very demanding environment and that the edge and divertor operating regimes must be carefully controlled to minimize erosion and to ensure an adequate lifetime for the plasma facing components (PFCs). In addition, the amount of tritium retained within the in-vessel structures must be limited to satisfy licensing regulations for ITER operation. Federici opened the workshop by giving an overview of the key issues arising from PWI in ITER which influence the choice of plasma facing materials.

A detailed understanding of the principal erosion processes and rates for the several materials to be used in ITER PFCs under stationary and transient conditions is required to enable accurate predictions of the component lifetime in the ITER environment. Erosion is also the principal source mechanism for impurities and therefore an improved characterization of the underlying physics is essential as a basis for predicting the impurity sources in ITER and, ultimately, the plasma impurity content. Krieger organized and chaired the first topical session of the workshop, which dealt with advances in experimental investigations and modelling of the fundamental physics processes influencing erosion and impurity generation.

The consequences of erosion were addressed in the following session, organized and chaired by Kirschner, which reviewed progress in developing an improved understanding of material transport and redeposition and their impact on in-vessel tritium retention. The last is a key issue for ITER since the limit for the tolerable amount of retained tritium is currently set at 350 g. To predict component lifetime and tritium retention, a detailed knowledge of current devices on erosion processes, locations of erosion/deposition and the transport mechanisms involved is necessary. Experimental evidence from a range of devices was reviewed: long-range transport effects, including asymmetric flows from the main chamber to the divertor, can only be studied in divertor devices, whereas short-range transport can also be studied in limiter configurations and plasma simulators.

Considerable progress has been made in recent years in improving understanding of how the various transport processes at work in the SOL govern a number of macroscopic features that are likely to play an important role in the operation of a next step device such as ITER. In many cases this has been the result of improvements in the quality of critical diagnostics and coordinated efforts to make similar measurements on as many different devices as possible. Advances in this area were discussed in a session on transport processes in the divertor and SOL organized and chaired by Pitts, which considered recent experimental evidence obtained during both stationary plasmas and transient events and the implications for transport of flows and turbulence in the SOL.

Sustained high-Q operation in ITER will rely on achieving a high level of plasma purity, but the relationship between the rate of material erosion and the resultant contamination of the plasma core is not yet sufficiently well characterized to allow a quantitative prediction of core impurity concentrations in ITER based on impurity source rates at the PFCs. The current understanding of core impurity transport processes was reviewed in the final topical session of the workshop, which was organized and chaired by Zastrow. This considered the experimental determination of impurity transport coefficients, the state of understanding of fundamental physics processes influencing experimental measurements of plasma impurities, the status of theoretical descriptions of impurity transport and, finally, our ability to predict the impurity content of ITER plasmas.

In recent years, the increased integration of experimental, modelling and theoretical activities has been a specific feature of the European fusion programme, which has stimulated significant progress in understanding many key areas of ITER-relevant physics. Under the European Fusion Development Agreement (EFDA), the scientific exploitation of JET is organized under task forces with membership drawn from all the EU fusion laboratories. In addition, the scientific research of the medium-sized devices within the programme is increasingly organized around European (and international) collaborations. In order to better focus the activities of the EU programme on certain key issues for ITER, two EU-wide task forces have been established under EFDA, on plasma–wall interactions (PWI-TF) and on integrated tokamak modelling (ITM-TF). Philipps and Bécoulet, the leaders, respectively, of the PWI-TF and the ITM-TF, presented the current activities of the task forces and their plans for the further development of their programmes in the future.

The final session of the workshop was devoted to a review of four key aspects of physics R&D for ITER: 'Physics of the H-mode pedestal', presented by Suttrop; 'Tolerable edge conditions for fusion plasma operation', presented by Counsell; 'Active control of MHD instabilities', presented by Ortolani; and 'Development of common hybrid/steady-state scenarios', presented by Imbeaux. Each of these presentations considered the present status of our understanding, highlighted the implications for ITER and proposed key issues which should be addressed to improve our predictive and operational capabilities for ITER.

The workshop was held at Heraklion in Crete and hosted by the FORTH Institute, Heraklion, on behalf of the Association EURATOM-Greece. It was sponsored by the host organization and the European Commission, and their support is gratefully acknowledged. The local organization of the workshop was carried out under the leadership of Dr P Laloulis.

2. Plasma–wall interactions in ITER

The choice of plasma facing materials is a key decision for the construction and operation of ITER. Currently, the ITER design contemplates the use of beryllium for the whole first wall, CFC near the divertor strike points and tungsten elsewhere. The main rationale for using Be on the first wall is that it is a low-Z material, which minimizes the effects of impurities in the main plasma, and is also effective in gettering oxygen. The use of CFC follows directly from the expected material damage for the projected levels of thermal loads at the strike points during plasma transients (e.g. type I ELMs and disruptions). Tungsten is planned as a material for divertor baffle areas and as a possible alternative material for the divertor high heat flux areas. Technological issues associated with the development of robust PFCs have been the subject of extensive R&D during the ITER design activities, while plasma–material interaction (PMI) issues remain the subject of intensive research within the fusion programmes of the ITER partners.

Current material choices are known to confer certain benefits but may also have potentially undesirable features. For example, erosion and re-deposition of C-based materials during both normal operation and off-normal events will lead to tritium codeposition, and efficient methods of removal are required in ITER to control the in-vessel tritium inventory. Effective tritium removal techniques need to be developed and proven in tokamaks with the relevant material mix and temperatures. The possible melting of metals during thermal transients (i.e. disruptions and ELMs) represents a further issue which must be resolved. The resulting impurity production and plasma contamination could limit the available operational space. In addition, use of different materials in different areas gives rise to the formation of mixed-materials, whose behaviour is uncertain and whose implications on operation are difficult to project. The remaining key R&D issues have been identified and are being vigorously investigated, primarily in Europe, or in appropriate bilateral co-operations with other ITER partners when suitable facilities are not available in the EU (e.g. experiments in PISCES-B in the US and in Troitsk in Russia). These issues include the following: (i) carbon and tritium migration to remote areas and gaps of plasma facing surfaces; (ii) mixing of materials; (iii) tritium removal; (iv) divertor and first wall thermal loads during ELMs and disruptions and ensuing material effects; (v) erosion and transport of W and compatibility with plasma scenarios; and (vi) PMI measurements and diagnostics.

According to current ITER construction plans, several years remain in which further R&D can be conducted in areas required to improve confidence in the final material choices for ITER and to improve/optimize the design of the divertor and first wall. Such research includes experiments necessary to test all relevant effects in existing machines (e.g. Be wall in JET and full W coverage in ASDEX Upgrade) and to minimize the ITER operational time required to resolve PMI issues. Continued development of high confinement scenarios with low ELM power flux to the divertor and experiments in disruption simulators to quantify the impact of transient heat loads on candidate armour materials are important R&D topics.

It is nevertheless likely that some uncertainties will remain, and experiments will have to be conducted in ITER to determine the precise implications of PMI effects on plasma performance and operation. ITER will have the ability to change the divertor or first wall material, albeit with consequences on machine availability and the need to develop operating scenarios compatible with new divertor and wall materials. Replacement of first wall modules in ITER is more complex than replacement of the divertor target cassettes and will lead to longer down times with additional implications for machine availability. Initial operation with H and D plasmas will explore these questions and quantify the resulting effects and the attendant uncertainties.

The built-in flexibility in the ITER design, which allows modifications to in-vessel components and changes in plasma facing materials, is seen as a key aspect of its contribution to the development of the technology required for fusion power plants. In particular, the hydrogen operating phase should be used to identify the operational space and optimal materials for the DT phase. The strategy currently under consideration for the ITER divertor is to use CFC as the power handling material in the divertor and to test all issues related to plasma performance and PMI during operation with H and D plasmas. A fall-back solution is to start with a full tungsten divertor. The first option implies the availability of reliable and accurate H/D retention and erosion diagnostics and validated modelling tools (which is a significant challenge *per se*), which would enable accurate estimates of H/D retention and its location in the machine during H/D-operation, leading to projections of T retention during DT operation. Maintaining C during the DT phase is contingent on the availability of fast and efficient methods for T removal which remain under development.

The use of a full W divertor from the very start of operation will require reliable methods of controlling and mitigating power loads during transient events. Because of the long lead-time (3–5 years) required to manufacture a new tungsten divertor, it may be prudent to procure an alternative full W divertor during the ITER construction phase. Delaying procurement until (if) T retention proves significant (and if no satisfactory removal schemes are devised) would create substantial delays in the operational phase. A change of the divertor material during operation requires the development of new plasma scenarios (e.g. no intrinsic divertor impurity radiation from tungsten, possibly more high-Z core impurity contamination, lower tolerance to disruptions or large ELMs, etc) and much of the experience gained in the operational phase with CFC PFCs might be of little relevance. An additional issue which needs to be better quantified by R&D is the possible need to develop techniques for the removal of C deposits formed in various locations in the device during the H/D and DT phases. This may be necessary to prevent adverse effects after the change to tungsten due to C-mixed materials formed from residual carbon (e.g. Be/C could be codeposited with tritium).

The main challenge associated with the use of Be in the ITER main chamber is associated with energy deposition on the first wall during type I ELMs, disruption thermal quenches, mitigated disruptions, VDEs and, possibly, runaway electrons. Damage to the Be first wall will be determined by the size and spatial distribution of the energy pulses, which is still uncertain for ITER, but is the subject of active research in existing experiments. In addition, there is some concern over possible adverse effects arising from the mixing of Be and W, e.g. the formation of low melting-point compounds, which require further urgent investigations. The outcome of this research should determine whether the present choice of materials is retained, possibly with small modifications of the PFCs in the areas subject to the highest transient fluxes (limiters, upper X-point blanket modules, possibility of poloidal protection limiters, etc) or whether the use of tungsten as a first wall material is pursued, as might anyway be necessary in a fusion reactor. This latter option is supported by promising results in ASDEX Upgrade, but the demonstration of plasma compatibility with a full W wall is an outstanding issue which future research will address. Even if W is not chosen for the reference start-up for the main chamber PFCs, it could be considered for a second, reactor oriented, phase of ITER operation. Experiments to demonstrate whether W is a viable alternative for ITER are therefore of the highest priority in parallel with ITER construction and operation.

3. Erosion processes and impurity generation

The contributions to this session were focused on recent advances in the understanding of carbon erosion processes by chemical reactions and by multi-species bombardment of carbon and on the use of tungsten as a plasma facing material. Roth presented an overview of current knowledge on the parametric dependence of carbon chemical erosion rates on incident particle flux, incident particle energy and target temperature, highlighting its implications for ITER. The experimental data base for the carbon chemical erosion yield consists mainly of data from weight loss measurements, mass-spectroscopic detection of methane and other products of the chemical erosion process and of spectroscopic measurements of the molecular band emission of CD molecules. To achieve a consistent data set for the particularly important decrease of the erosion yield with incident hydrogen flux, the data from different devices were normalized to an incident ion energy of 30 eV and to the substrate temperature at which maximum erosion was observed, typically in the range of 600–800 K. A further improvement in relation to previous measurements involved the *in situ* calibration of spectroscopic measurements. Applying this normalization procedure, one obtains a consistent data set which enables derivation of the parametric dependence on particle flux using a Bayesian fitting procedure. This procedure takes the individual systematic errors of the respective data subsets into account. One obtains finally $Y(T_{\max}) = 0.82/(1 + (\Gamma/3 \times 10^{21}\,\mathrm{m^{-2}\,s^{-1}})^{0.58})$. This yield does not take into account higher hydrocarbon molecules. Experimental results were presented which give correction factors in the range of 1.3–3 to the methane yield. Compared with previous assumptions, one infers from the new formula a greatly reduced importance of chemical erosion for carbon components exposed to high particle fluxes.

Details of the spectroscopic determination of carbon chemical erosion rates from molecular band emission and absorption were discussed by Brezinsek. Since the detected molecular radicals are produced in a chain of dissociation processes from the originally eroded molecule species, with large uncertainties in the corresponding molecular rate coefficients, measurements of erosion rates are generally calibrated using well-characterized gas puffs. Depending on discharge and surface conditions, C_2 molecular band emission is often observed in addition to CD band emission. This indicates that erosion products do not necessarily consist only of methane. The spectral resolution and the modelling of the band structure is, however, not yet adequate to allow conclusions on the origin of the C_2 molecules. Possible candidates are higher hydrocarbon erosion products and/or carbon cluster emission. A correction formula for the inclusion of the ethane family in the evaluation of CD band emission was derived under the former assumption.

In laboratory experiments, several groups have found enhanced carbon erosion by the simultaneous impact of hydrogen isotopes and noble gas ions. The erosion yield due to the combined impact of both species is significantly higher than the sum of the single species yields. Tabares summarized these findings and their relevance for plasma cooling scenarios with noble gas seeding and reported results from DIII-D and JET. At present, results from fusion experiments do not unambiguously confirm the laboratory findings. Codeposition of tritium with carbon has to be strongly reduced in fusion machines using carbon PFCs. A proposed mechanism for this is seeding of scavenger gases such as N_2, which convert hydrocarbon radicals with high sticking coefficients to radicals with low sticking coefficients. Corresponding experiment results, e.g. from JET and PISCES-B, were presented. However, these results do not yet provide a clear confirmation of the scavenger action of N_2.

In ITER, beryllium is expected to be one of the main plasma impurities. Schmid presented recent results on the influence of beryllium impurities in a hydrogenic plasma on the erosion of carbon surfaces. This was studied in a controlled experiment at PISCES-B, where Be was seeded into a linear plasma and carbon erosion was quantified by spectroscopic observation of CD molecular band emission and by target mass change. The methods show independently that carbon erosion vanishes almost completely for plasma concentrations of Be $> 0.1\%$. This is explained as an effect of the immediate coverage of the carbon surface with deposited beryllium. In the discussion it was pointed out that this behaviour should be expected, but is not clearly observed, in the JET experiment, where Be is evaporated between discharges for wall conditioning. Additional erosion mechanisms might prevent deposition of a closed Be layer in this case. For example, strongly increased Be erosion yields have been recently observed in PISCES-B at elevated surface temperatures. This effect was explained as a result of evaporation of weakly bonded surface atoms which are continuously created by the ion bombardment.

Molecular dynamics (MD) simulations provide a means of improving the understanding of the fundamental erosion mechanisms. Nordlund discussed the application of MD to model the energy and flux dependence of carbon chemical erosion by hydrogen. Experimental findings are well described by the simulations. The underlying model shows that the chemical erosion is not a result of physical sputtering or chemical etching processes but rather that hydrogen penetrates the space between adjacent carbon surface atoms, thus increasing their energy relative to their binding energy. A similar model was applied to the formation of amorphous CH layers. This model yields a strong dependence of hydrocarbon sticking coefficients from the local neighbourhood of dangling bonds and from the angle of incidence of hydrocarbon radicals. These effects were advanced in explanation of the wide variety of sticking coefficients observed under otherwise similar experimental conditions. As a further application, the first results were presented showing MD simulation of tungsten cluster sputtering and blister formation. The model predicts the formation of bubbles in a W matrix by agglomeration of He atoms and bubble growth by interstitial loop punching. However, the results indicate that the rupturing of bubbles, at least for those produced by low energy He impact, does not lead to significant W erosion.

For energy producing fusion devices beyond ITER, erosion rates of low-Z elements render the use of such materials in PFCs impractical owing to lifetime limitations, even at the relatively low incident particle fluxes expected at the first wall of the main plasma chamber. On ASDEX Upgrade, tungsten is studied as an alternative, high-Z, material. Neu reported recent results from this device. Since tungsten surfaces are gradually replacing the original carbon-based PFCs, this configuration also provides the possibility of studying the relative importance of carbon sources and the corresponding carbon recycling and migration. Despite a reduction in the carbon surface area by more than 50%, only a marginal decrease in carbon plasma concentration has been observed. This is explained as being a result of carbon recycling at W surfaces. In addition, carbon limiters have been identified as the major C source, so that a significant reduction can only be expected if those components are also replaced by W. On the other hand, with large area tungsten walls, the total tungsten source has the potential to cause a significant degradation of plasma performance. It was shown that, in a divertor machine such as ASDEX Upgrade, the screening action of the boundary plasma is usually sufficient to avoid this problem. Moreover, in cases where plasma transport is strongly decreased, W accumulation can be avoided by central heating and/or (externally triggered) increased ELM frequency. Using these techniques, both impurity seeding and internal transport barrier (ITB) plasma scenarios were shown to be compatible with a tungsten plasma facing first wall.

4. Material transport, redeposition and tritium retention

This session considered carbon transport in divertor and limiter tokamaks and in the plasma simulator PSI-2, results of tungsten transport in ASDEX Upgrade and possible methods for *in situ* measurement of erosion, deposition and tritium retention. Transport and redeposition of carbon in ASDEX Upgrade and JET was reviewed by Mayer. In ASDEX Upgrade, the outboard limiters provide the primary source of carbon erosion, and the carbon resulting is transported and redeposited on the inner heat shield, which acts as a secondary carbon source via re-erosion. Carbon redeposition mainly takes place on the inner strike point region and below the divertor, while redeposition on the walls of the pumping duct is negligible in the overall carbon balance. Layers below the divertor are soft, with D/C ratios between 0.7 and 1.4. They are formed mainly by hydrocarbon radicals with high sticking coefficients. Simulations of deposited profiles inside the pumping duct implicate two radical species: one with a sticking coefficient, β, of 0.3 and a second with $\beta < 10^{-3}$. However, 99.7% of the flux in the duct consists of species with low sticking coefficients. This is also reflected in the fact that only 0.008% of the total deuterium input is trapped in the pumping ducts, whereas the layers below the divertor (produced by species with high sticking coefficients) trap 0.3%.

Total erosion measurements in the main chamber of the JET MkIIGB configuration (1999–2001) yielded values between 160 and 480 g, depending on the method used, while measurements of carbon deposition resulted in values of 570–870 g. The inner divertor of the JET MkIIGB is deposition dominated, whereas no clear tendency is seen for the outer divertor. Deposits in the MkIIGB are rich in C and D at the surface, but below the surface the D content is lowered, C is depleted and large amounts of Be are found. Detailed analysis of deposited layers close to the louvres at the inner module and below the septum reveal a D/C ratio of about unity (soft a-C : D) and almost no Be. In contrast to beryllium, carbon is transported to remote areas, which can be explained by a selective chemical re-erosion of deposited carbon. From deposition profiles inside a sticking monitor mounted at the septum it follows that the layers are mainly ($> 99.8\%$) formed by species with high sticking coefficients. Summarizing, the carbon sources of JET MkIIGB and ASDEX Upgrade are in the main chamber and asymmetric flows drive eroded carbon principally to the inner divertor. Carbon migration to remote areas is determined by multiple deposition/re-erosion steps.

Tsitrone reviewed results on carbon migration and deposition in the limiter devices TEXTOR and Tore Supra. The toroidal pump limiter, ALT, forms the main carbon source in TEXTOR and a significant fraction of the eroded carbon is redeposited on the limiter tiles themselves (45%). A detailed global carbon balance showed that obstacles such as the bumper and the neutralizer plates are areas where most of the remaining carbon is deposited. Only a small amount is observed at remote areas inside the pumping ducts. Deposited layers on the ALT limiter contain the majority of the retained deuterium (85%). The D/C ratio of the layers is linked to the surface temperature and ranges from <0.01 to ~ 0.7, with the higher fraction in the cold pumping duct. About 8% of the total deuterium input into TEXTOR is retained in deposited a-C : H layers inside the vessel. By modelling the local hydrocarbon transport of injected $^{13}CH_4$ through a test limiter, it is concluded that the (extremely low) observed ^{13}C redeposition efficiency is due to an enhanced re-erosion of fresh deposits. This leads to a long-range transport of carbon via repetitive redeposition/re-erosion.

Active cooling of the entire vessel in the CIEL configuration of Tore Supra enables the study of PWI phenomena on long time scales under stationary conditions—a discharge of 378 s duration was presented. The global carbon balance in Tore Supra is rather preliminary, but exhibits substantial similarities to the TEXTOR results. The toroidal limiter is again the main carbon source, and, as in TEXTOR, heavy deposition is measured on the neutralizer. The deposits have a fractal structure and a D/C ratio of less than 0.01. This is in agreement with observations in TEXTOR for films on the neutralizer plates. As in TEXTOR, no deuterium-rich films are found inside the vessel (maximum D/C ratio of 0.1), in contrast to JET, an observation which might be due to the fact that the operation temperature is above 100˚C in TEXTOR and Tore Supra. A complex pattern of carbon deposition with films, flakes and dust is observed on the Tore Supra toroidal pumped limiter. In the gaps between limiter tiles, adherent layers are formed which are also visible in IR imaging as locations of higher surface temperature.

The linear PSI-2 machine allows transport and deposition studies of injected hydrocarbons, C_xH_y, in a well-diagnosed plasma under stationary conditions. The experimental set-up and experimental results were presented by Bohmeyer. Depending on the fuel (H, D or Ar) the electron temperature can be varied between 5 and 15 eV and the density between 10^{18} and 10^{19} m^{-3}. Local deposition of carbon at a collector sample is due to particles with a high sticking coefficient which are produced during the first passage through the plasma, but global deposits appear at the chamber wall due to sticking after many passages through the plasma. Higher plasma densities lead to a shift from global to local deposition. Erosion caused by H atoms is seen only at higher plasma densities. In an Ar discharge with very low H atom concentration, the deposition is independent of surface temperature within a (collector) temperature range of 320–470 K. No difference is seen between the erosion of ‘fresh’ and ‘old’ films. Modelled (ERO code) deposition rates of injected C_2H_4 into an Ar discharge are at least a factor of 4 smaller than measured rates. Moreover, the model predicts a stronger dependence on plasma density than observed. Possible reasons for these discrepancies are still under discussion.

In ASDEX Upgrade, the central column of the main chamber, the divertor baffles and the passive stabilizing loop cover tiles have been covered with tungsten. Krieger reported recent results. Most of the eroded tungsten originates from the central column during limiter configurations (plasma ramp-up/down). Twenty per cent of the eroded tungsten is locally redeposited, 9% flows into the divertor and 4% penetrates to the core plasma. However, about $\frac{2}{3}$ of the eroded tungsten is not balanced by deposition—one possible explanation is the existence of hidden deposits. Modelling of the tungsten transport with DIVIMP results in qualitative agreement with the measured redeposition pattern. For the limiter configuration, 90% of eroded tungsten is locally redeposited at the central column. In the divertor configuration modelling confirms the transport of eroded tungsten to the inner divertor, but the modelled deposition at the outer strike point is smaller than observed.

Schweer addressed proposals for *in situ* diagnostics to localize and quantify erosion/deposition and tritium retention in ITER. From spectroscopic measurements of line intensities one can determine the flux of released particles as long as relevant conversion factors (e.g. light intensity to particle flux) and plasma parameters are known. A quartz microbalance can be used to measure net deposition, or erosion, by means of the change of the resonance frequency associated with the change in thickness of the layer deposited on the quartz. This method is used successfully at JET and TEXTOR. With speckle interferometry, erosion and deposition can be measured in the range of 0.1–1 μm (which is related to the number of fringes). Ellipsometry and colorimetry determine net deposition. Transparent layers must not exceed 500 nm in thickness and their optical properties must be known. *In situ* ellipsometric measurements will be performed at TEXTOR in 2004, while colorimetry is used routinely. Laser ablation can remove particles from the surface during short, high power density, laser pulses: the layer thickness (net deposition) can be determined from the emitted line radiation of the ablated particles. The main limitation is the possibility that bulk material is also removed. Further tests of all methods are necessary to establish their applicability for ITER.

A global measurement of retained tritium can be made via a gas balance, but this gives no information on the fraction codeposited, which can be obtained only by local measurements. *In situ* laser desorption combined with plasma induced spectroscopy is a possibility for which first experiments have been carried out at TEXTOR and JET. One significant problem is the inadequate knowledge of conversion factors and sensitivity (background radiation, dynamic range).

5. Transport processes in the divertor and SOL: experiment and modelling

The aim of this session was to present an overview, both of theory and experiment, of some aspects of SOL transport as they relate to issues of importance for future large devices. One such issue, reviewed by Chankin, is the possible link between SOL flows and the strong asymmetries observed in levels of impurity deposition at the inner and outer tokamak divertors (in particular at JET, where, for the usual B_φ direction with the ion $\boldsymbol{B} \times \nabla B$ drift directed downwards, a series of measurements with different techniques has shown conclusively that the inner target is an area of net deposition and the outer mostly a region of net erosion). Three main processes are known to drive tokamak SOL drift flows: perpendicular ion drift flow resulting from $\boldsymbol{E}_\mathrm{r} \times \boldsymbol{B}$ and $\boldsymbol{B} \times \nabla \mathrm{p_i}$ (diamagnetic) drifts, which drives a parallel field (Pfirsch–Schlüter) return flow, the $\boldsymbol{E}_\mathrm{pol} \times \boldsymbol{B}$ drift, which leads to a net toroidal rotation, and outboard enhanced (ballooning) perpendicular transport which must be at least partially closed by parallel ion flows away from the midplane. Of these three mechanisms, the last is the only basic process which is both insensitive to the reversal of B_φ and which could, in principle, be the cause of impurity migration from the *outer* to the *inner* divertor. However, recent experiments on JET have demonstrated that impurity deposition begins at the *outer* target during sustained reversed B_φ operation, indicating that it is probably the changes in divertor asymmetries caused by (and driving some of) the SOL flows that are responsible for much of the observed difference in impurity migration. While edge codes now include the effects of drifts and, to some extent, ballooning transport, none can yet match the large measured flows in the SOL—further work is required.

The important question of cross-field energy transport in the near-SOL (in the separatrix vicinity where power fluxes are highest) and the physical mechanisms determining the divertor power deposition length, λ_q, were comprehensively reviewed by Fundamenski. Summarizing previously reported λ_q scalings from a variety of tokamaks, he highlighted the absence of an acceptable theory for the perpendicular heat diffusivity, $\chi_\perp^{SOL}$ (for either plasma species) capable of explaining all experimental data. This poses an obstacle to a physics-based prediction of λ_q for ITER, requiring instead extrapolations based on empirical scalings alone.

Concentrating on the outer divertor target (where peak power fluxes are a factor of 5 higher than at the inner target for normal field direction, i.e. $\boldsymbol{B} \times \nabla B$ towards the X-point) and considering only the inter-ELM H-mode phase (during which most of the power arrives), a detailed analysis of recent λ_q measurements on JET was presented. The ion SOL collisionality, ν_i^*, is found to be the main ordering parameter for both the peak deposited heat flux, q_{max}, and the closely correlated narrow structure in the observed heat flux profiles $(\lambda_q \sim 3\rho_i) : q_{max}$ was found to be dominated by ions for $\nu_i^* < 5$ and by electrons for $\nu_i^* > 10$. The inferred H-mode λ_q scaling exhibits a negative dependence on the deposited power in the outer divertor ($\lambda_q \propto P_{div}^{-0.48}$), in contrast to the more favourable, positive, scaling reported in the ITER Physics Basis ($\lambda_q \propto P_{div}^{0.35}$). Comparison of experimental λ_q scalings with power, density, field, current and ion mass (charge) with those predicted by a wide range of $\chi_\perp^{SOL}$ theories (with λ_q related to $\chi_\perp^{SOL}$ using a simple SOL model) suggests that (neo-)classical ion conduction is the dominant perpendicular energy transport mechanism in the near-SOL during the inter-ELM phase. In the collisionless limit ($\nu_i^* \ll 1$), a smooth transition of this mechanism to ion orbit loss may be expected. Anomalous (turbulent) electron conduction, although playing a secondary role in the near-SOL, is likely to dominate further away from the separatrix. Extrapolating the recent JET results to ITER yields a prediction of $\lambda_q \sim 3.7 \pm 1.1$ mm, which should be compared with the ITER design value of 5 mm. This extrapolated value, however, does not account for the higher compression of the ITER divertor, which is expected to significantly broaden the power deposition profile.

Goncalves discussed the role of turbulence in governing the rate and nature of cross-field particle transport in magnetic confinement systems, though again with reference to the L-mode and inter-ELM regimes only. The availability of turbulence measurements has recently increased significantly at a number of devices as recognition grows that the 'bursty', or intermittent, nature of the fluctuations in the tokamak SOL sets the width of the far SOL density profile and, in many cases, carries a significant particle flux onto the first wall. This transport competes with and, at high densities, can exceed that due to parallel losses, so that the divertor no longer controls particle exhaust. An important question is the extent to which this main chamber interaction will be important as a source of wall erosion in ITER. The two-dimensional interchange model was considered as an example of how non-diffusive (ballistic) transport can be described theoretically, demonstrating that this approach can reproduce many features of the experimentally measured turbulence and may indeed be at the origin, for example, of the mean density profile most commonly measured in the SOL. Examples of data from JET, TJ-II and TCV illustrated that the experimentally measured probability distribution functions of density or flux are often empirically similar and may be described by universal statistical parameters or scaling factors. Experimental data also support the strong link between fluctuations in local gradients and $\boldsymbol{E} \times \boldsymbol{B}$ transport, suggesting relaxation back to a marginally stable state in which the size of transport events is minimized. On JET and elsewhere, effective velocities of these larger events can be several $100\,\mathrm{m\,s^{-1}}$, an order of magnitude higher than normal diffusive velocities.

Eich focused on ELM transport in the SOL and divertor, reviewing some of the latest experimental results from tokamaks. Data from fast IR imaging of divertor targets in JET are demonstrating that as much as 75% of the ELM energy can be deposited after the maximum surface temperature, $T_{\mathrm{surf}}^{\mathrm{max}}$, has been attained and that the baseline, inter-ELM power profile width is broadened by, at most, a factor of only 1.5 during the ELM. In/out divertor target ELM power deposition becomes more balanced with increasing I_{p} in JET and progressively more unbalanced (in favour of the inner target) with increasing density in AUG. For very large ELMs, power balance indicates that only $\sim$50% of energy expelled from the pedestal reaches the targets on JET, while new measurements on AUG suggest that 10–40% of the ELM energy is radiated, with $\sim$15% reaching the outboard wall structures. IR thermography at AUG has also identified for the first time a fine structure of the ELM target power deposition, from which approximate mode numbers can be derived and a toroidally asymmetric, outboard midplane energy efflux inferred. Such inferences are qualitatively consistent with images of D_α emission from JET and high-speed video images from MAST, where clear filamentary structures have been seen. Further evidence is being found in the form of multiple bursts during ELMs on fast Langmuir probe measurements of SOL parallel ion fluxes (JET, TCV, DIII-D). Probes are also being used to monitor ELM cross-field propagation speeds, showing that larger ELMs travel faster and can thus lose less energy to parallel transport before arriving at the walls.

In a closely linked presentation, Loarte discussed the extrapolation of type I ELM power fluxes from the current database to ITER and the implications for PFCs. In making such extrapolations, much depends on the nature of the expected ELMs. At high pedestal collisionality, ν_{ped}^*, ELMs become more convective, with only small decreases in pedestal T_{e} owing to the ELM, more ELM energy being deposited on divertor targets after $T_{\mathrm{surf}}^{\mathrm{max}}$ is reached and power fluxes which would be acceptable (from the point of view of divertor lifetime) when extrapolated to ITER. At the lower ν_{ped}^* of ITER, however, it is not clear that pure convective ELMs will be obtained, although recent experiments at JET have identified a regime with high q_{95} and high triangularity in which the low ELM energy losses are possible at low ν_{ped}^*. If no mechanism can be found to avoid conductive ELMs in ITER, extrapolation yields ablation/melting thresholds that are higher than can be tolerated for an acceptable lifetime, but are considerably less threatening than previous estimates had indicated. Extrapolation of main chamber energy deposition leads to the expectation that even relatively low ELM energies arriving at the ITER limiters may lead to unacceptable power loads (though these calculations have significant uncertainties owing to the lack of experimental data and the need to estimate wetted area of the ITER limiters during ELMs). Such limiter power loading is of particular concern in view of the current choice of Be as the limiter material, since melting occurs at lower energy fluxes (a factor of $\sim$2–4) than for the W or C divertor targets. It was also highlighted that although type I ELMs may be triggered at precise values of pedestal parameters, there is large scatter in the experimentally measured energy release per ELM. Since ITER divertor target lifetime will largely be determined by events beyond the ablation threshold, this scatter remains an issue of concern.

Edge and divertor physics issues impacting the operation and understanding of the island divertor stellarator, e.g. W7-X (under construction) and W7-AS, were reviewed by König. Although the plasma boundaries of island divertors and poloidal field divertor tokamaks are governed by the same physics, the much reduced field line pitch angle ($\sim 100\times$ smaller than for tokamaks) and reduced core to target distance in the island divertor lead to a much stronger role for cross-field transport of particles, momentum and energy. The inherent three-dimensionality of the island divertor is also a major difference, causing toroidally inhomogeneous radial transport fluxes and localized recycling zones which cannot be smoothed by parallel heat conduction. Numerical simulations using the EMC3/Eirene code are in good agreement with experiment, predicting the observed absence of a high recycling regime in W7-AS, demonstrating how the strong cross-field transport is responsible for the high upstream densities required for detachment and qualitatively modelling the drift effects on particle deposition seen in B-reversal experiments. Stable partial detachment is found only when target to X-point distances, Δx, are long enough, connection lengths, L_c, short enough and the divertor plasma remains locally attached to some location which is topologically connected to plasma on the low field side. Code simulations successfully reproduce the observed jump in both radiation intensity and position of the radiation zone at detachment (a process essentially controlled by local power balance between cross-field transport and impurity radiation). The evolution of the radiation zones as a function of Δx and L_c has also been modelled with EMC3, demonstrating the sensitivity of the radiation distribution to island geometry and consistent with measurements of strong inboard radiation at high density, similar to the tokamak MARFE, when L_c is low and Δx large.

6. Core impurity transport

Puiatti presented an overview of experiments to determine impurity transport coefficients. Under favourable conditions, these can be derived for fully ionized species in the source free plasma region, using the experimental data directly, by measurement of densities, gradients and fluxes combined with a linear regression. The plasma conditions must be stationary and high time resolution is required, but it is necessary to smooth data in space and time, which introduces errors. The most commonly used technique, therefore, is an interpretative scheme in which a model with free parameters is constructed and compared with experimental data. This allows extension of the analysis to partially ionized species, which nevertheless requires detailed knowledge of atomic data. The main issue with this approach is that the final solution may not be unique, but depends on a certain level of individual interpretation. However it is possible to distinguish the main features of individual experiments, i.e. changes in diffusion and convection between different confinement regimes. A fully predictive scheme would be preferable, but present knowledge of impurity transport does not yet enable satisfactory prediction of existing experiments. Further work is underway and an example of first results from a model for RI-Mode discharges by Tokar were shown.

Impurity accumulation, defined as a change in the impurity density profile shape, rather than a simple increase of the impurity density (which may be due to increased impurity influx), was reviewed by Guirlet. The impurity density profile shape is determined by the ratio of convection to diffusion, where the inward convective terms giving rise to accumulation are believed to be neoclassical in improved confinement regimes. However, it is difficult to calculate these for real plasmas as the theory is complex and requires the determination of gradients of experimental data. Inward convection due to peaked deuterium density profiles is seen to be the norm. Temperature screening has been observed in some advanced scenarios, so that impurity accumulation is not inevitable. It is also observed that anomalous diffusion, including the effects of MHD events such as sawteeth and fishbones, helps to reduce the peaking, although there is currently no definite theoretical prediction for turbulent impurity transport.

The influence of uncertainties in atomic data, specifically how to quantify and incorporate them in the analysis or prediction, was addressed by O'Mullane. The goal of this work is to provide a parallel database of errors to accompany the rate coefficients in the ADAS spectroscopic database. The error sources were explained on the basis of spectral line emission data. The data can be obtained from atomic models of various sophistication, from Born approximation and distorted wave calculations, converged close coupling and, finally, R-Matrix calculations, which represent the current state of the art. However the last is often incomplete, as a simplified description of the atomic species has to be made. The results then vary depending on how the neglected states are treated. A survey of data for all nuclear and ion charges shows that high quality, validated data exist for species up to Neon, although beyond this there are mainly studies along isoelectronic sequences of relatively simple species (He- or Na-like ions). Data along isonuclear sequences for elements higher than Neon are sparse (there are only baseline data for Ar, Kr, Xe and W).

Error propagation from individual cross-sections to predicted line emission under typical fusion plasma conditions was illustrated for He, where the error had a temperature dependence. A further example was shown for state selective charge-exchange cross-sections for collisions with hydrogenic neutral atoms, where low and high energy approximations were shown to differ in the intermediate energy range. However, the exclusion of excited states for hydrogenic neutrals in the plasma model can introduce a larger error in predicted recombination rates at low collision energy.

Weisen presented an overview of theoretical predictions for anomalous particle and impurity transport, specifically convection, together with comparisons with experimental observation. Anomalous pinches are expected from the generic results of turbulent equipartition (TEP) theories, as well as from drift wave (TEM, ITG) turbulence theories. All of these predict a dependence on magnetic curvature (hence on shear and aspect ratio), while the last also predicts a dependence on temperature gradients (anomalous thermodiffusion). The existence of anomalous pinches has been unambiguously demonstrated in fully current driven ($E_{\mathrm{tor}} = 0$) discharges in Tore Supra and TCV, since the potentially important neoclassical Ware pinch is suppressed in such cases. LHCD experiments in JET where the q-profile was varied systematically from positive to negative shear have density profiles which are consistent with the curvature pinch and show no evidence for anomalous thermodiffusion. Results from ASDEX Upgrade show that density peaking decreases with collisionality in ITG-dominated ELMy H-modes. The theory of anomalous impurity transport, specifically pinches, is still in its infancy. Recent simulations by Weiland suggest that the peakedness of impurity ion density profiles should be less than that of the main ions, as expected from the 1/Z scaling of magnetic drifts. This important and favourable prediction calls for a systematic validation by experiments.

Concerning the relationship between energy and impurity transport, it was noted that the majority of published results compare energy confinement times with particle confinement times and that there is so far only one study comparing this on different devices, namely JET and Tore Supra. It is frequently observed that the particle (or impurity) confinement time degrades with power, but not always in the same way as the energy confinement time. Furthermore, the plasma shape has been observed to have a different effect on the impurity and on the energy confinement times.

The ITER requirements for impurity content, specifically considering the removal of He ash in the presence of additional impurities, were reviewed by Dux. For low-Z impurities, the requirements are related to dilution, while for high-Z impurities radiative power loss is the most significant issue. Results of impurity density profiles for He, Be, Ar and W, in addition to the D and T fuel, from calculations of anomalous convection using GLF23 (i.e. ITG and TEM terms are considered) and from neoclassical calculations were illustrated. Using identical ratios of anomalous convection velocity to diffusion coefficient for all species, the dominant drive for impurity accumulation is due to the Ware pinch. Assuming fully non-inductive operation under otherwise identical conditions would lead to a substantial reduction of central impurity densities. Transport across the last closed flux surface was not included in these calculations, and the concentration of the impurities at the plasma edge was instead based on assumed fixed values.

These studies conclude that impurity accumulation is not expected in ITER plasmas if turbulent transport is dominant. If the alpha heating power profile on ITER is not sufficiently peaked that anomalous transport dominates, it may be possible to use the technique demonstrated on AUG of using central electron heating (e.g. ECRH), possibly at the expense of a reduction in Q. Sawteeth with suitable period and mixing radius can also suppress high-Z impurity accumulation.

The final discussion concentrated on two main issues. First it was recognized that there are few studies of impurity behaviour that encompass the whole chain from PWI, through transport in the SOL to transport in the core. Experimentally, studies on methane screening come closest to bridging this gap. Integrated plasma modelling is in its infancy (even for the considerably simpler problem of energy transport), but it is clear that experimental data in the edge and SOL are not readily available, in particular data on impurities with sufficient spatial and temporal resolution. The key implication for ITER is that we cannot predict how much of the eroded impurity influx will penetrate to the plasma edge.

The second key issue was the experimental basis for making predictions. It had become clear from the presentations that the types of studies performed are disjointed. For example, results from different devices are in different plasma regimes and tend to complement the scientific goals pursued at each device. In addition, publications tend not to include the background plasma data in a way that allows comparisons between devices. The current evidence therefore remains fragmentary, and a co-ordinated comparison across devices, probably involving identity experiments on several devices and the establishment of an international database on impurity transport, is required.

7. European physics task force programmes

7.1. Plasma–wall interactions

The work programme of the EU task force on PWI (EU-PWI-TF) concentrates on issues directly associated with the choice of the first wall materials for ITER: a Be first wall (700 m^2), W on the divertor upper baffles (70 m^2) and dome and CFC graphite on the lower part of the divertor (50 m^2). This selection results from tokamak experience worldwide and is focused on achieving the primary goal of ITER to demonstrate burning plasma operation, validating predictions on confinement, MHD stability, ELM and disruption behaviour, exploring the physics of current drive and demonstrating effective divertor power and particle exhaust. A low-Z Be wall avoids the risk of degrading plasma performance by wall impurity influx, while CFC is the optimal material to facilitate plasma operation over a range of operating regimes, though target lifetime and power exhaust after high transient power loading at ELMs and disruptions remain an issue for ITER. The use of W-armoured baffles and dome should ensure an acceptable PFC lifetime while minimizing the use of CFC. However there is a large step (substantially larger than in many other areas of physics) in duty cycle and in power fluxes at transients between present devices and ITER. This has brought the most critical PWI issues for ITER, target erosion/ablation at transients and long-term tritium retention, to the centre of fusion physics research and motivated the establishment of the EU-PWI-TF.

The critical questions associated with the use of Be in the main chamber are particle and power fluxes at transients. Recent data indicate a much stronger interaction of the plasma with first wall structures at ELMs and disruptions than previously assumed, leading potentially to enhanced erosion and/or localized melting of Be. A detailed characterization of perpendicular and parallel power flow during all types of ELMs and disruptions in present devices is required to assess the first wall power loading in ITER in greater detail, in particular the power loading at the top of the vessel (second separatrix effects) and on the main chamber start-up limiters. This will guide further design analysis of the ITER first wall structures. In parallel, the exploration of W as an alternative first wall material is of high importance, and this is largely a question of plasma compatibility.

The use of CFC at the divertor targets must be assessed in the light of potentially high erosion (physical and chemical) at transients and the associated tritium codeposition. Extrapolations from present full carbon devices indicate an unacceptable level of T retention in ITER under such (full carbon) conditions. However, the carbon which codeposits with hydrogenic species in the divertors of present devices originates mainly from main chamber carbon erosion and this will be absent in ITER. The ITER Be wall will therefore change the material deposition characteristics, in both space and composition, and significantly reduce T retention compared with a full carbon device. However, uncertainties remain due to the complexity of material migration processes and a lack of data on mixed Be/C/W layers which are expected to be formed under ITER conditions.

To consolidate predictions for ITER, the investigation and understanding of material migration paths is a major objective of current experiments and of the PWI-TF work. Remarkable progress has been achieved recently, but further R&D is necessary to allow prediction of material migration and T retention in ITER with the desired degree of confidence. During the ITER ‘non-activated’ hydrogen operational phase, assessment of fuel retention will be a major aim and appropriate diagnostics must be prepared. The feasibility of changing to a full W divertor depends in large measure on an improved knowledge of disruption power deposition and on the development of mitigation techniques, which is, and must be, a central element of PWI research. In parallel (and independently of the final choice of the divertor material), development of tritium removal scenarios and techniques compatible with ITER conditions and wall materials is essential and requires substantial further research in present devices.

7.2. Integrated tokamak modelling

The European task force on ITM was established ‘to co-ordinate the development of a coherent set of validated simulation tools for the purpose of benchmarking on existing tokamak experiments, with the ultimate aim of providing a comprehensive simulation package for ITER plasmas’. This long-term activity was initiated in November 2003 and the immediate goal has been to structure the activity and provide a short-term work programme. The TF activity is now organized around four topical areas. The work to be performed in each area during the initial phase of the TF is described below. This work is broadly aimed at reviewing the current status of modelling and also at developing the longer-term strategy of the TF, towards an integrated suite of codes to optimize the European exploitation of the ITER project.

Area #1—identification and models will take an initial census of the existing codes and models available. This concerns a priori *all* domains of the Modelling Activity, and stellarator and RFP groups are encouraged to participate. Thus, while the primary goal of the ITM–TF is ITER integrated scenario activity, the work is not restricted to tokamak geometry. Area #1 should give a clear and realistic assessment of the present situation, provide a first list of codes and models (with documentation) and highlight the areas where no modules are available, or where a need for further development exists.

An expert working group (EWG) has been initiated to provide the following:

- a classification of codes and models, following the relevant domains of physics involved, in view of progressively setting up integrated suite(s) of codes;
- a list of the existing codes to be integrated (EU) and/or imported through collaboration;
- a format for code documentation;
- identification of (short and long-term) code integration tasks for key ITER physics issues (common task with Area #3, and preparation of Area #4 activity);
- recommendations and priorities for code/model development;
- recommendations and priorities for numerical optimization of existing codes;
- an estimate of future resources required to support the activity.

Area #2—interfacing procedure and numerical support must initially provide a detailed procedure and a common interfacing package, enabling cross-coupling between codes and databases. A key goal is to converge at an internationally accepted standard for the interfacing of code modules and access to experimental data. Area #2 must also explore the possibility of providing high-level expertise in the domain of numerical optimization when necessary. A systematic handling of code versions must be rapidly put in place.

An EWG is now charged with providing the following:

- the global structure of integrated modelling;
- the interfacing procedure;
- a code version handling procedure;
- the corresponding recommendations in terms of language, libraries, etc;
- an initial list of tools to be developed by the TF (Interfacing Package);
- an evaluation of the capability inside the EU fusion programme to provide the relevant expertise and hardware;
- a strategy for setting-up a ‘numerical support’ structure;
- an estimate of future resources required to support the activity.

Area #3—code validation and benchmarking must provide a detailed procedure for benchmarking and validating codes and models. This will involve close comparisons with experimental data.

An EWG will provide the following:

- an initial version of the benchmarking procedure;
- an initial version of the ITM-TF database;
- possible recommendations for dedicated experiments necessary for a correct validation;
- a format for the validation exercise documentation;
- a detailed documented example of the validation exercise (physics model testing, code validation, etc);
- a plan for validation of codes (consistent with Area #1);
- an estimate of future resources required to support the activity.

Area #4—ITER integrated scenario activity should progressively undertake a comprehensive assessment of the integrated ITER scenarios, relying on the tools validated by the ITM-TF and supporting the development of ITER-relevant scenarios in current experiments. This work relies on substantial progress in Areas #1–3, and will be initiated later in 2004.