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

For the period July 2008 to June 2009

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Executive Summary

The SOL/Div Topical Group held two meetings during the reporting period, in Nagasaki, Japan, September 15-18, 2008, and in Amsterdam, Netherlands, May 7-10, 2009.

The links between the ITER IO and our ITPA TG have been strengthened with the new organization now in place, as illustrated by the fact that during the meetings in 2008-2009, sessions have been suggested and chaired by ITER IO (hydrogen level, outgassing after disruptions, divertor re-attachement), or talks have been given by ITER IO on request from the ITPA TG (ITER Plasma Facing Components status). The presentations from our TG can now be found on the ITER IDM website.

During the reporting period, a strong effort has been devoted by our group to set up a R&D plan addressing ITER high priority issues, based in particular on large discussion sessions organised during the Amsterdam meeting. Five areas have been identified (fuel retention and removal, dust, heat loads, tungsten R&D and material migration), and experts have been appointed to lead the effort for each topic. The resulting High Priority Research Areas are given in section 2.

The main scientific highlights over the reporting period in the five areas identified as high priority are listed below. More details can be found in the summary of the meetings in section 1.

- Fuel retention and removal
    - The tritium (T) inventory predictions for ITER have been refined. In particular, a group of experts met in MIT in June 2008 to get a general agreement on the input parameters used (main wall fluxes etc). Results were presented in an oral paper at the IAEA 2008 conference.
    - Fuel retention in tungsten (W) has been further investigated to evaluate the effect of He and neutron irradiation (see W R&D below).
    - Recent coordinated efforts have been devoted to Ion Cyclotron Wall Conditioning (ICWC). Although the mechanism responsible for fuel removal is still an open question (ions or high energy neutrals impact), cleaning rates are approaching that needed for ITER. The IC plasma has been extended in the vessel, e.g. using vertical fields, adding He in D or H, and proper phasing of the antenna. Current concerns include possible damage of the ICRF antennas if not used in the right parameter range. Future experiments are planned on a number of machines.
    - Planned mitigated disruptions have been suggested for fuel removal, as part of the ITER strategy for T inventory control. First data on fuel release after disruptions were collected, showing that the energy available in ITER disruptions is thought to be sufficient for a significant fuel release, but these planned disruptions should be carefully tailored for fuel recovery while avoiding PFC damage and allowing an easy plasma start up for the next discharge..
- Dust
    - First estimates of conversion factors from gross erosion to dust production were presented, covering a range from 1-15 %. However, this study has shown the

difficulty of a correct assessment, in particular for transients during plasma operation (ELMs, disruptions) or maintenance phases. For carbon PFCs, the main source of dust production seems to be the flaking of the thick deposited layers, while for high Z PFC, it seems to be the transient events (disruptions or strong ELMs) leading to melting.

- The effect of dust on plasma operation has also been evidenced for carbon devices, showing an impact on long pulse operation as well as on high power performance (disruptions due to flakes ejection from thick deposited layers). The performances were restored after an extensive PFC cleaning.
- Coordinated dust injection experiments have been prepared (new DSOL) and the associated effort on dust transport modelling started, with the first results of simulations on dust penetration in the plasma presented.

- Heat loads
  - The characterization of mitigated ELMs is just starting. Pellet induced ELMs in AUG are similar to natural ELMs at the same frequency. Heat loads in RMP mitigated discharges in DIIID seem similar to averaging between ELMs of a non mitigated discharge. Given the importance of this topic for ITER, it is clear that more data are needed (in particular with pellet pacing in JET, RMPs in DIIID).
  - Optimisation of disruption mitigation using massive gas injection is ongoing. Different gas mixtures are under study, but understanding the physics of the penetration of the injected gas is essential to progress further (cross machine comparison needed). New fast bolometry systems have evidenced complex toroidal/poloidal asymmetries (radiation peaked near the injection location), raising concerns for localized heating of the Beryllium (Be) wall in ITER. Mitigation of runaway electrons remains critical (large gas quantity required).
  - Preliminary data on divertor re-attachment (timescales and heat loads) were presented, to document what happens in case of loss of detachment in ITER divertor (failure of gas injection, H-L transition etc). To progress further, dedicated experiments are needed. A coordinated effort is proposed to improve the associated modelling, as deficiencies were identified, including simultaneous detachment of both the inner and outer divertors in simulations (while experimentally the inner divertor is observed to be detached for anything but the lowest densities) and the decrease in peak ion flux after detachment (observed in experiments but not in codes).
- W R&D
  - The (already low) fuel retention in W is reduced (factors of 10-1000) under simultaneous He and H bombardment, which might be due to He nanobubbles in the near surface, acting as a H diffusion barrier. However, the reduction might be smaller (factor of 2 rather than 100) for the case of pre-damaged W (nuclear damage of order .01-.1 dpa). A bigger concern is the development of nanostructure at the surface as W is implanted with H and He, which could lead to enhanced erosion (and dust).
  - The importance of tile shaping to avoid local leading edges and melting for W PFCs has been underlined and is being discussed with the ITER IO team.
- Material migration
  - Chemical erosion is now being investigated for detached plasmas, but a correct assessment of the contribution of neutrals to the erosion yield and heavy

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

- New experiments on W migration have been performed with a localized set of W tiles in JT60U. W is found to be redeposited locally toroidally (probably due to prompt re-ionization and short range migration), while C is transported further toroidally (from ^{13}C injection experiments). The W core concentration seems to depend more sensitively on the particle transport properties (increasing with counter toroidal velocity) than the W source at the edge. In contrast with JT60U, a complete ring of W divertor tiles installed in C-Mod has led to no observable level in the core. New experiments on Be migration have also been performed in JET using strong Be evaporation. In all cases, impurity transport is recognised as the most uncertain point to predict material migration in ITER (and associated fuel retention, dust production …) as well as impurity core concentration.

The status of collaborative DSOL experiments is reviewed in section 1.3. 9 DSOL proposals are running, 2 were proposed to be closed at our last meeting (poloidal fuelling and pulse by pulse deposition), while 2 new proposals (coordinated dust injection and divertor re-attachement) were started.

The next meeting of the Div/SOL Topical Group will be in San Diego, US, coordinated with the timing and location of the abstract selection meeting for the Plasma Surface Interactions Conference. The proposed week would be December 14-18, 2009.

Contents

1. Meetings and reports

A full summary of the 11th and 12th meetings of the ITPA Div/SOL Topical Group, and viewgraphs presented, are available at the ITER IDM website (https://user.iter.org/?uid=2MV2EG), and only the executive summaries repeated here. A summary of results on IEA/ITPA co-ordinated experiments was also presented at the November 2008 planning meeting for these experiments and will not be repeated here.

1.1 Report on the 11th Meeting of the ITPA SOL and divertor physics Topical Group, Nagasaki, Japan

The meeting was held over the period September 15-18, 2008 at the Hotel New Urakami in Nagasaki, Japan. The local coordinators were M. Sakamoto (Advanced Fusion Research Center in the Research Institute for Applied Mechanics) and T. Tanabe (Interdisciplinary Graduate School of Engineering Sciences) of the University of Kyushu. The meeting lasted 3-1/2 days and concentrated primarily on gathering information from current experience on H retention, dust generation, and material migration. There were over 40 participants.

The new organization of ITPA was briefly presented showing, in particular, the strengthening of the links with ITER IO. This was illustrated in the meeting by sessions initiated by ITER IO, such as multi machine comparison of H levels and dust production. The presentations of the meeting can now be found on the ITER IDM website (access (user ID and password) available upon request).

The SOL/divertor group had previously reviewed (Avila meeting, January, 2008) the calculations of T retention in ITER made by the US and EU during the ITER review period last year. It was clear that the underlying assumptions and methods used by the 2 groups were not in agreement. Bruce Lipschultz and Jochen Roth then organized an effort to resolve these differences. 13+ scientists came to an ‘ITPA ITER Tritium Inventory Assessment workshop’, June 23-24, 2008 at MIT. Agreement was reached on a number of underlying parameters such as the flux of ions and atoms to all surfaces (and their energies), the co-deposition rate (T with Be and C) at various surfaces and the neutron damage to W which leads to D retention deep within the W bulk. Most projections to ITER have been redone. However the effect of neutron damage on retention deep within the W is still in the process of being modeled. As expected, the present selection of PFCs for ITER (Be/C/W) lead to large uncertainties in retention (4.5 - 260g at 10^5s of full operation or 250 shots). Only taking into account ion implantation into W (ignoring nuclear damage) leads to a lower projected range of T retention for an all-W ITER (3-30g for the same period of operation). Future work includes finishing the modeling of retention due to nuclear damage (W) and writing a summary of the work. We also intend to expand the group work in this area with the addition of experts outside of the US and EU (e.g. Japan).

Two issues have led to a study by ITPA members of H levels when operating in D plasmas: 1) The ‘additional’ gas load on cryopumps which, if too high, could lead to reaching the deflagration limit in cryopumps; and 2) the interest in measuring fuel retention during both the ITER H and D phases. Large levels of wall outgassing (H remains in surfaces following a vacuum break) could add an additional unknown fueling source. The data reported for both carbon PFC (EAST, TS, DIII-D) and high-Z (C-Mod) machines showed that the H/[H+D] fraction in the plasma started high after a vacuum break as expected, but then dropped fairly rapidly (100s of seconds of operation) to levels in the range of 1-5%, staying fairly constant thereafter (not known why). Even such low levels could make accurate gas balance measurements difficult if the retention rate is low. Normalization of the H source rate by plasma wetted area leads to values between $10^{18}/m^2/s$ and $2x10^{19}/m^2/s$. Projected to ITER (assuming 20 m^2 of wetted area) we estimated ~ $5x10^{20}/m^2/s$ (1 Pa-m^3/s) of pump load. On the other hand assuming 5% H/[H+D] we came up with a pump load of 10 Pa-m^3/s of H. It was pointed out that disruptions are the much more worrisome cryo load - if the T in PFCs is at the limit (700g) then up to 1400g of D+T could come out of surfaces in one disruption. This information was passed back to the pumping group at ITER.

2 sessions were then devoted to fuel retention, on carbon and high Z PFCs respectively.

Concerning carbon, a wide range of retention fraction is observed, from 20-30% (JET) to 50% (TS) and up to 80% (recent results from DIII-D in un-pumped discharges) of the injected gas, while it is close to 0% on JT60U for saturated walls conditions. The effect of plasma type (L-mode to Type I ELMy H-modes) was shown to be small but worth further investigation. The discussion showed that, because of varying pumping conditions, the absolute value of the retention rate, rather than the retention fraction, should be used for comparison between devices; Increased pump rates leads to increased injection rates, but the same amount of retention. Recent work on JET and Tore Supra show that the fuel retention estimated from post mortem analysis and particle balance can come to a reasonable agreement (factor 2 in TS), provided they are carried out on similar plasma conditions (dedicated campaign with repetitive pulses in TS) and that an extensive post mortem analysis is performed (for instance gaps in TS, louvers in JET).

Previous results have pointed to co-deposition as the dominant retention process for carbon PFCs, primarily occurring at the inner divertor for diverted tokamaks. New at this meeting was a report from JT-60U which showed that the erosion areas of the main wall had significant D in the near surface (perhaps implanted by fast particle losses) that, although low per unit area, may rival the divertor retention. A similarly surprising result was that co-deposition on tile sides in high erosion areas of the castellated Tore Supra limiter dominates the co-deposition in remote areas. Deep diffusion of D into CFC graphite (which might be due to co-deposition inside the porosity network of CFC) was evidenced in erosion zones, and shown to be ~ 10% of overall retention in Tore Supra.

It was proposed that the same method used for ITER T retention estimates be benchmarked against present day machines (estimate of wall/diverter fluxes ➔ erosion sources ➔ fuel inventory using D/C scalings) and the results be compared with particle balance (TS, JET, AUG, DIIIID, JT60U …), as well as gap modelling be benchmarked against experimental data (Textor, TS).

Concerning high Z materials, the number of hydrogenic retention processes are quite varied. While implantation of ions is certainly the initial entry point for retention of fuel it is not so clear what all the possible retention mechanisms are within lattice nor their relative importance relative to each other and to release from the surface. From the point of view of retention bubbles and blisters can be good as they often release their trapped gas and inhibit diffusion deep into the bulk. On the other hand such deformation of the surface obviously damages the surface and probably degrades the material properties in terms of heat load handling and shock resistance. We are just starting to address the effect of neutron damage which creates traps for T retention throughout the tile. The talks at this meeting and at the MIT meeting indicate that such damage can lead to a maximum of ~ 1% [D+T]/W. Then, it is a matter of when the D+ T implanted at the surface can diffuse to those traps and thus be retained. Will it be slow or fast? Lastly it was pointed out that simultaneous implantation of He along with the D,T fuel can affect the diffusion of trapping of D,T. There are differing reports on whether the effect increases or decreases D,T retention and this will be a subject of review in upcoming meetings.

The present ITER dust safety strategy relies on measurements of gross erosion, and assumes as a conservative upper limit a conversion factor f_d between gross erosion and mobilisable dust equal to 1. The previous and current tokamak studies for carbon PFC machines reported values of f_d between 1-15%. In this meeting we concentrated on confronting the existing data for gross erosion and how truly it is related to creation of dust. For carbon PFCs, the main source of dust seems to be the peeling off of thick deposited layers (both exposure to air and layer thickness probably play a role in the peeling process), leading to complex flakes structures. Even though the amount of C gross erosion as measured by spectroscopy is often much larger than the collected dust it is not clear whether they are related. In contrast, for high Z machines (C-Mod and AUG), preliminary results indicate that gross erosion estimated from spectroscopy is marginal to explain the dust produced. The meeting participants felt that it was likely that transients are playing a central role for high-Z dust formation. The collected metallic dust is often spherical, which again could be linked to droplets formation during transients (disruptions, strong ELMs, arcing …). In all cases, dust is often evidenced in the plasma (cameras …) after disruptions. Further studies somehow directly linking dust types/quantities (flaking of C, droplets for high-Z) to erosion mechanisms (steady state chemical erosion and/or transients) is needed before projections to ITER can be made.

Dust can have serious effects on plasma operation. As an example Tore Supra compared operational space (and UFOs) with and without thorough extensive cleaning of dust and C flakes from all PFCs. Cleaning led to a dramatic reduction in UFOs and the operational space defined by heating power limits were greatly increased. Expansion of such cleaning/non-cleaned comparisons to diverted machines and Be walls (JET ILW) will be crucial. The study of dust transport is beginning to reach a stage where codes (DTOKS in EU, DUSTT in US) can make predictions of dust trajectories and be used as tools for understanding dust evolution (e.g. is the light from dust due to blackbody emission or line excitation). It is proposed to launch a new DSOL task on coordinated dust injection experiments (with the same pre-characterized dust) and associated modelling for code benchmarking. However, this is limited to carbon (eventually W) dust, as no Be dust is readily available … Experiments in plasma guns (in Russia) and/or plasma simulators (PISCES, Pilot …) could also be used for code benchmarking.

A session was devoted to the mitigation of transients (ELMs and disruptions), which is now included in ITER baseline, as it is recognized that non mitigated events will lead to serious PFCs damage.

The characterization of mitigated ELMs is just starting. Pellet-induced ELMs in AUG are similar to intrinsic ELMs in terms of SOL plasma and heat loads at the same frequency. Results from pellet pacing in JET are expected to provide more data in this area, in particular to explore the scaling with pellet frequency. The measurements of the effect of ELMs mitigation with RMPs (Resonant Magnetic Perturbations) on divertor heat loads is primarily on DIII-D as other tokamaks either do not have the RMP capability operative at the moment or have no ELMs. The experiments are hampered by lack of IR measurements at the outer strike point and matching non-RMP and RMP equivalent discharges. However, the general assessment was that mitigated discharge heat fluxes were similar to averaging over the ELM, between-ELM periods of a non-mitigated discharge. Given the minimal amount of information available and the importance to ITER it is clear much more information from DIII-D and now JET is needed..

Concerning disruptions, new, fast bolometry systems are just coming online at several machines to diagnose the wall and divertor heat loads with mitigated disruptions (AUG, C-Mod, TCV). Complex behaviours are in evidence, with pronounced toroidal /poloidal asymmetries (peaked near the injection location) during the initial gas penetration. While this certainly raises concerns for localized heating of the Be wall in ITER, the measurements show that when the radiation is highest (thermal quench) the asymmetries drop to ~ 1 (C-Mod). Different gas mixtures are under study for optimisation of the mitigation process, but understanding the physics of the impurity injected in these mitigated plasmas is critical to progress further (cross machine comparison essential). It is recommended that the ITER system remains flexible in terms of the nature/amount of gas injected. The results of the ITER workshop on runaway electrons during disruptions were presented to our group to keep them informed on the potential damage to in-vessel components.

Finally, a session on material migration was organised, covering all steps from erosion to transport and redeposition.

Chemical erosion of carbon is now being addressed in detached plasma parameters, but the contribution of atoms is essential in these conditions, and should be correctly diagnosed for valid erosion yield. The effect of seeding gases (Ne, Ar …) needs to be investigated. Erosion products (heavy hydrocarbons) and their sticking properties (strong impact on C migration modelling) should also be addressed.

While C^{13} injection has been used in the past to study the transport of C a new experiment in JT-60U has allowed the study of W material migration: A localized set of W tiles (1/12 of toroidal circumference) served as the W source. W is found to be redeposited locally toroidally (probably due to prompt re-ionization and short range migration), while C is transported further toroidally (from ^{13}C injection experiments). The amount of W in the core was strongly dependent on core rotation and conditions The W core concentration seems to depend even more sensitively on the particle transport properties (increasing with counter toroidal velocity) than the W source at the edge. In contrast, a complete ring of W divertor tiles installed in C-Mod has led to no observable level in the core. W migration seems to indicate a strong role of poloidal drifts in JT60U. Be migration was also studied in JET using Be evaporation.

In all cases, impurity transport is recognised as the most uncertain point to predict material migration in ITER (and associated fuel retention, dust production …) as well as impurity core concentration.

1.2 Report on the 12th Meeting of the ITPA SOL and divertor physics Topical Group, Amsterdam, Netherlands

The meeting was held over the period May 5-8, 2009 at various locations in and around Amsterdam. The local coordinators were J. Rapp and G. van Rooij of the FOM Rijnhuizen laboratory. The meeting lasted 4 days and was split between discussions on the SOL/div R&D plans in support of ITER urgent needs and several research topics. There were over 50 participants.

The meeting included both the usual sessions dedicated to selected physics issues, and specific R&D sessions devoted to building the group work program to address the ITER urgent R&D needs.

Concerning the latter point, the group undertook an effort to organize around high priority ITER R&D needs defined in the ITER Physics PWI research plan [1] . An initial R&D plan was drafted and circulated prior to the IEA/ITPA committee meeting (Dec. 2008). Five areas were selected which parallel the ITER high priority R&D areas:

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

Two co-leaders were asked to lead each of the above tasks. This meant developing a set of new sub-tasks, deciding the level of priority, soliciting input from the experts in the field, and lastly leading the discussions at the Amsterdam meeting. Summaries of those sessions are found in Section II-2 For each R&D task, there are plans in place varying from paper studies (e.g. to evaluate the effect of higher bakeout temperatures in ITER) to initial data collection (e.g. the timescales and physics of re-attachment), and more directed sets of experiments to be performed in a coordinated way (e.g. main chamber erosion and material migration).

Concerning the physics sessions (summarized in section II-1), 2 sessions were devoted to tungsten related issues.

M. Merola (ITER IO) reported on the status of divertor components and the development of tungsten (W) as a divertor Plasma-Facing Component (PFC) material. While the design has reached a level of maturity allowing the procurement to begin, the main issue in the case of tungsten PFCs is the alignment of individual tiles, potentially creating a large number of leading edges exposed to unacceptably high heat fluxes and leading to melting. Shaping individual tiles to provide leading edges shadowing was discussed. This topic requires a substantial effort to optimize the design, as shaping could reduce significantly the wetted surface, and add considerable manufacturing cost, particularly for the all-W divertor currently foreseen for the DT phase of ITER operation.

In a second session, retention in tungsten under simultaneous implantation of He with T was discussed. The hydrogenic retention is reduced (factors of 10-1000) for a range of He fractions in the incident ion flux (0.2-10%). He ion bombardment at low energy creates nanobubbles near the W surface, which might act as a diffusion barrier to hydrogenic species. There is some indication that the reduction in retention due to simultaneous implantation with He is much smaller (factor of 2 rather than a factor of 100) for the case of pre-damaged W (nuclear damage, of order .01-.1 dpa). What appears to be a bigger concern is the development of nanostructure at the surface as the tungsten is implanted with hydrogenic ions and He: the surface becomes rougher (e.g. blisters, bubbles) which could lead to enhanced erosion (and dust).

Fuel removal issues were also treated, with a session on Ion Cyclotron Wall Cleaning (ICWC) and fuel release after disruptions.

Wall conditioning will be required for a number of reasons in ITER – removal of impurities (e.g. after a vacuum break or a major disruption) and removal of tritium. We reviewed the current experience with ICWC which could be performed in ITER with the toroidal field present. It is generally found that the IC plasma can be achieved and with cleaning rates approaching that needed for ITER. There has been significant work to make the plasma more uniform in the vessel, e.g. using vertical fields, adding He in D or H, and proper phasing of the antenna straps. The current concerns are that the use of ICRF antennas for this purpose (as opposed to heating tokamak plasmas) might damage the antenna (as happened at Tore Supra) if not used in the right parameter range and with the proper set of control tools; whether neutrals or ions are the primary particles that lead to fuel removal, the role of ICWC for impurity removal and whether we can properly predict the requirements for ITER. Future experiments are planned on a number of machines.

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

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

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

A session on divertor modelling was organized to better understand where the problems lay with modelling of detachment. A number of deficiencies were identified, including simultaneous detachment of both the inner and outer divertors in simulations (experimentally the inner divertor is always observed to be detached for anything but the lowest densities) and the decrease in peak ion flux after detachment (observed in experiments but not in codes). The modellers performed a number of experiments with variations of the plasma models to try and overcome these deficiencies. Some improvements were found based on these ad hoc assumptions. It was agreed that the modellers will try to join together those within the ITER PWI Section to test the codes against specific experimental data. JET, AUG and C-Mod data were discussed as possible case studies. After the meeting it has been decided that M. Wischmeier (IPP Garching) will take on the role of coordinator of this combined modelling activity establishing a database of experimental data to model and common nodes for the various codes to use in writing results for direct comparison.

[1] : ‘Proposed near term R&D programme for ITER plasma-surface interactions and edge physics’ by R. A. Pitts, A. Kukushkin, A. Loarte, M. Shimada, V.3: 4 October 2008, submitted by the ITER IO to STAC 5

1.3 IEA/ITPA multi-machine collaborations

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

- DSOL-1 Scaling of Type-1 ELM energy loss and pedestal gradients through dimensionless variables (A. Loarte) Closed
- DSOL-2 Chemical erosion under ITER-like divertor conditions (semi-detached) (S. Brezinsek)

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

- DSOL-3 Scaling of radial transport (B. Lipschultz) Closed
- DSOL-4 Comparison of disruption energy balance in similar discharges and disruption heat flux (A. Loarte) Closed
- DSOL-5 Role of Lyman absorption in the divertor (S. Lisgo) Closed
- DSOL-8 ICRF Conditioning for hydrogen removal (N. Ashikawa)

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

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

Proposal: JET, DIII-D, TEXTOR, AUG

- DSOL-11 Disruption mitigation experiments (D. Whyte) Moved to MDC-11
- DSOL-12 Reactive gas wall cleaning (P. Stangeby)

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

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

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

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

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

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

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

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

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

- DSOL-17: Cross-machine Comparisons of Pulse-by-Pulse Deposition (C. Skinner)

Proposal: NSTX, AUG, JET, TEXTOR

- DSOL-19: Impurity generation mechanism & transport during ELMs for comparable ELMs across devices (A. Loarte) Closed
- DSOL-20: Transient divertor reattachment (R. Pitts)

Proposal : DIII-D, ASDEX-Upgrade

- DSOL-21: Introduction of pre-characterized dust for dust transport studies in divertor and SOL (D. Rudakov)

Proposal: DIIID, TEXTOR, MAST, NSTX

It was proposed :

- To close DSOL16 on poloidal fuelling (final report)
- To close DSOL17 on pulse by pulse deposition (final report) and join the work in DSOL on 1st wall migration

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

Ongoing DSOL, with new experiments planned, include : DSOL8 on ICWC (TS, TEXTOR, AUG, JET), DSOL9 on material migration (^{13}C tracer experiments in TEXTOR, AUG, JET and associated modelling), DSOL12 on O cleaning (lab experiments + TEXTOR), DSOL13 on gaps (TEXTOR, AUG, TS).

New DSOL (DSOL20 on divertor reattachment and DSOL21 on dust injection) have started and experiments are planned (AUG, DIIID, JET, C-Mod for DSOL20; TEXTOR, DIIID, LHD for DSOL21)

2. High Priority Research Areas

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

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

It is our goal to complete high priority tasks in the next few years at least to the level at which educated decisions can be made by the IO regarding important hardware issues (e.g. in the case of ICWC). Thus, the label "medium priority" should be taken to imply that meaningful results are expected on the 2-4 year timescale, whilst the high priority assignment implies a higher level of urgency, with closure or significant advancement of the task requested on the 1-2 year timescale, matching that over which conclusions are expected from the ITER side. The reader will, however, note that in the "timescale" column of the table, even some of the designated high priority items are considered more "medium term" given our assessment of how long it might take before substantial results are accrued.

The distribution of subtasks in terms of high and medium priority is still being discussed and may change slightly compared with the attributions in the table below. More details of the participating laboratories or facilities in each subtask can be found in the SOL/Divertor TG presentation given at the July 2009 CC meeting in Cadarache.

R&D Topic Area	Subtask	Timescale	Priority
Tritium retention and removal *Leaders:* R. Doerner J. Roth	Refine predictions for expected retention on ITER (report)	1-2 years	High
	Constitute multi-machine retention database (new DSOL)	2-4 years	High
	Pursue development (multi-machine) of ion cyclotron wall cleaning (ICWC) to establish feasibility for ITER wall conditioning, explore potential for T-removal and determine compatibility with planned ICRH system (DSOL8)	2-4 years	High
	T-removal by outgassing to 350°C – the baseline ITER divertor bakeout temperature (collaboration)	1-2 years	High
	Influence of mixed impacting species on fuel retention (collaboration)	1-2 years	High
	T-removal potential of disruption flash heating (IO	1-2 years	Medium

	coordinated)		
	Fuel retention in gaps (DSOL-13,9 – with material migration R&D Topic Area)	2-4 years	Medium
	Isotope exchange/tailoring using plasma discharges (new DSOL under discussion)	1-2 years	Medium
	Carbon removal capability and associated risks (report)	1-2 years	Medium
R&D Topic Area	**Subtask**	**Timescale**	**Priority**
Tungsten *Leaders:* A. Kallenbach Y. Ueda	Impurity generation due to ICRH operation (modelling, design, report)	2-4 years	High
	Melt layer behaviour and effect of divertor target damage on subsequent operation (tokamak experiments, report)	1-2 years	High
	Balance between ELM driven impurity sources and outflux due to ELM flushing (tokamak experiments, report)	1-2 years	High
	Material mixing, cracks, surface morphology changes, blistering (report)	1-2 years	High
	Tritium permeability and retention in neutron damaged W (lab experiments, report)	2-4 years	Medium
	Power load control by low Z extrinsic seeding (tokamak experiments, report)	1-2 years	Medium
	Edge modelling including W and W/Be (code development)	2-4 years	Medium
	Effect of mixed impacting species on T-retention in W (lab experiments, report)	1-2 years	Medium
R&D Topic Area	**Subtask**	**Timescale**	**Priority**
Dust *Leaders:* N. Ashikawa D. Rudakov	Characterisation of dust production rates, conversion factors from erosion and damage to dust production (collaboration)	2-4 years	High
	Cross-machine studies of dust injection including benchmarking of dust modelling tools (DSOL-21)	1-2 years	High
	Quantification of dust character under high loads: ejection velocities, dust sizes/morphology (IO coordinated work)	1-2 years	High
	Study the role of T-removal techniques in dust generation (laboratory collaboration)	2-4 years	Medium
	Contribution to development of dust measurements (in collaboration with ITPA Diagnostics TG)	1-2 years	Medium
R&D Topic Area	**Subtask**	**Timescale**	**Priority**

Heat fluxes to plasma-facing surfaces *Leaders:* A. Leonard M. Lehnen	Disruption heat loads (ongoing discussion within ITPA)	1-2 years	High
	Cross-machine characterisation of ELM statistics (DSOL-15, PEP-10, 21)	2-4 years	High
	Heat loads during ramp-up/ramp-down (tokamak experiments, report)	1-2 years	High
	Transient divertor reattachment (DSOL-20)	2-4 years	High
	Modelling detachment (new coordinated effort within the TG, re-focus of DSOL-14)	2-4 years	Medium
	Far SOL heat and particle fluxes (US comparison, extend to ITPA)	1-2 years	Medium
	Divertor and SOL ELM heat fluxes – characterisation of power footprints and parallel heat flux widths (ongoing ITPA meeting, reports)	1-2 years	Medium
R&D Topic Area	**Subtask**	**Timescale**	**Priority**
Material migration *Leaders:* P. C. Stangeby V. Philipps	Cross-machine comparisons of main wall erosion and local redeposition (modification of existing DSOL-9)	2-4 years	High
	Development (and benchmarking) of local models accounting for surface shaping to predict erosion and subsequent deposition in shadowed regions during steady state and limiter start-up/ramp-down phases (study, tokamak experiments, code development)	1-2 years	High
	Characterise outer and inner divertor erosion and movement of impurities between divertors and from main chamber to divertor (new DSOL under discussion)	2-4 years	Medium
	Understand the driver of fuel retention in tile gaps - dependence on material, flux, temperature (DSOL-13)	1-2 years	Medium

3. Future meetings

The next meeting of the Div/SOL Topical Group tentatively will be in San Diego-USA, coordinated with the timing and location of the abstract selection meeting for the Plasma Surface Interactions Conference. The proposed week would be December 14-18, 2009.