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

For the period July 2007 to June 2008

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Executive Summary

The SOL/Div Topical Group held a meeting during the reporting period, in Avila, Spain, January 7-10, 2008. The meeting and location were arranged just before the abstract selection meeting by Program Committee of the Plasma Surface Interactions Conference. As always the focus of our meetings is split between high priority tasks, near-term ITER requests, and new subjects, which need to be investigated for their potential importance to burning plasmas such as ITER.

The primary foci of this meeting were discussions and presentations on predictions of ITER heat and first-wall loadings, as well as ITER T retention in the case with all-W plasma facing components. There were also sessions on ITER diagnostics and what can be accomplished during the ITER H phase. Finally, there were general sessions on D/T retention, dust and divertor physics.

(1) Estimations of particle flux and fuel retention on ITER for current design and W-divertor:

Particle load on the ITER first wall was predicted to be in the range of 3-7x10^{23}/s, various calculations from experiment results in the divertor tokamaks agreed within the factor of 2. Issues for further discussion were the division of fluxes over the various first-wall surfaces (e.g. to the upper divertor vs inner wall vs outer wall PFCs), the heat load distribution for ELMs, and the transport into limiter shadow.

Predictions of T retention were presented both for the current mix of materials in ITER (Be, W and C), W-divertor ITER and all W ITER. The dominant retention for carbon PFCs was confirmed to be co-deposition: the T retention for the current mix of materials leads to the 350g limit being reached after ~ 300 discharges.

The new T retention estimations for W-divertor ITER and all W ITER show large reduction in T retention: the 350 gram in-vessel T limit was an increase of ~ x10 over the current C/Be/W case (2000-3000 discharges) for the former case, and more than 10000 for the latter case, respectively. Plasma irradiation was the primary source of T retention in W with the added effect of neutron damage, leading to a significant increase over ion-impact alone. This area should be emphasized to collect data under ITER to Demo like conditions (within limited numbers of facilities). He^+ impacting surfaces leads to nanostructure growth (up to ~ microns in depth) on W. We have to evaluate their importance for ITER T retention.

(2) New experiments and impact on fuel retentions of W and C:

The level of hydrogenic retention in tungsten generally is lower than for carbon based on the factor of 10 reduction in D retention (post-campaign analysis) in ASDEX-Upgrade following conversion to fully-W and from high-flux/fluence Pilot-PSI showed low fractions of incident D^+ retained (~10^{-6}).

The non-saturation of hydrogenic retention in carbon (linear in ion fluence) was demonstrated in the new Tore-Supra dedicated long pulses over 2 weeks. The co-deposition of D with Be appears to be more complicated, depending on the incident energy of deuterium, the Be deposition rate and the dependence on surface temperature. Use of scavenger gases for C layer inhibition as well as

Thermo-oxiadation for D removal were encouraging in being able to reduce and remove D retention respectively.

(3) Discussion summary of ITER H-phase and ITER divertor diagnostics:

A draft plan for the ITER H-phase was presented by A. Loarte (for the ITER Office). A number of assumptions of availability of the machine as well as diagnostic and heating systems was assumed together with a phased plan for bringing all of those systems up to full levels by staging the powers and plasma current levels at reduced pulse length (20-25s). It will be impossible to properly test the PFCs up to full power-handling capability nor the ability to handle transients such as ELMs or disruptions at full levels. Nevertheless close attention should be paid to monitoring the various surfaces during the startup phase to determine if they fail at those lower power levels.

A second aim during the H phase is to monitor the H retention. While ITER currently has no plan for this it is clear that one is needed and the SOL/divertor group will work closely with the ITER diagnostics group to agree on the proper diagnostics as well as a plan of measurment.

All of the above H-phase PFC information will go into the decision of whether to replace the C divertor PFCs with W before the D or DT phases.

The ITER plans for diagnostic coverage of the main chamber, SOL and divertor were reviewed. It was clear that several measurements were either not available or not well-specified (e.g. wall fluxes, injected gas, dust…). It was agreed that those diagnostics will be reviewed by the SOL/divertor group between meetings for spatial/measurement coverage and specifications. The gas injected during, and gas remaining, after a discharge were discussed at length and are the subject of discussion following the meeting. The group suggested that some method of H retention must be made during the H phase either through a small fraction of D injected, or through lowering the ambient levels of H in the chamber. Diagnostics for surface temperature measurements were reviewed and several laboratory developments of in-situ surface analysis (ion beam analysis and laser beam erosion/desorption) were presented.

(4) Influence of ELM heat load and RF-induced heat load on the first wall:

Wall and divertor loadings during transients continues to be a difficult subject to understand. The ELM characteristics when they are incident on the outer limiter in ASDEX-Upgrade were inferred to be in the range 100 eV per electron-ion pair. New measurements of ELM erosion at the outer divertor (ASDEX Upgrade) show that as the local temperature drops the between-ELM erosion also drops relative to that occurring during ELMs. Increasing the impurity level through impurity puffing may increase divertor radiation but also increases erosion during ELMs. The current model of ICRF-enhanced sheaths, so-called sheath rectification, were contradicted (C-Mod) in that the plasma potential increases proportional to P_{RF} and potentials generated even when the current loop is broken by insulating tiles. The ICRF-enhanced plasma potential has been observed to occur on flux tubes connected to the antenna as well as passing in front of the antenna.

(5) Dust issue:

Dust is of course a very important issue for ITER and was discussed as part of the diagnostics session. Based on a lack of experimental data the ITER organization is assuming that dust is generated at a rate equivalent to the erosion rate of all surfaces in the machine. The SOL/divertor group felt that the specification should be lower as evidenced by JT-60U results, less than 10% of the net surface erosion rate. We hope to also form a working group to obtain a better specification for the dust generation.

(6) Proposals for 2008 IEA/ITPA DSOL:

Fourteen proposals for 2008 IEA/ITPA DSOL inter-machine experiments were presented, and DSOL-11 was combined to MDC-1. Title and spokes-person for each proposal are followings;

DSOL-1 Scaling of Type-1 ELM energy loss and pedestal gradients through dimensionless variables (Loarte)
DSOL-2 Injection to quantify chemical erosion (Brezinsek)
DSOL-3 Scaling of radial transport (Lipschultz)
DSOL-4 Comparison of disruption energy balance in similar discharges and disruption heat flux(A. Loarte, D. Humphreys, G.Pautasso)
DSOL-5 Role of Lyman absorption in the divertor (Lisgo)
DSOL-8 ICRF Conditioning for hydrogen removal (Ashikawa)
DSOL-9 Tracer injection experiments to understand material migration (Philipps)
DSOL-11 Disruption mitigation experiments (Whyte) this 2008 proposal is combined to MDC-1
DSOL-12 Reactive gas wall cleaning (Stangeby)
DSOL-13 Deuterium codeposition with carbon in gaps of plasma facing components (Krieger)
DSOL-14 Multi-code, multi-machine edge modelling and code benchmarking (Coster)
DSOL-15 Inter-machine comparison of blob characteristics (Terry)
DSOL-16: Determination of the poloidal fueling profile (Groth)
DSOL-17: Cross-machine Comparisons of Pulse-by-Pulse Deposition (Skinner)
DSOL-19: Impurity generation mechanism & transport during ELMs for comparable ELMs across devices (Loarte)

Additional proposals will be discussed among contact persons in appropriate devises and will be proposed in next IEA/ITPA CC meeting Nov./Dec. Tentative titles and spokes persons are followings;

(1) Comparison of isotope effect of H/D edge and divertor plasmas for extrapolate to ITER H-operation (Fundamenski)
(2) Dust movement study by dust injection (West)
(3) Dust dynamics study (Krasheninnikov)
(4) Evaluation of carbon and methane generation fluxes based on particle balance study review work (Tabares)

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1. Meetings and reports

A full summary of the 10th meetings of the ITPA Div/SOL Topical Group, and viewgraphs presented, are available at the website (http://efdasql.ipp.mpg.de/divsol/, and only the executive summaries repeated here. Also a summary of results on IEA/ITPA co-ordinated experiments was presented at the November 2007 planning meeting for these experiments.

1.1 Report on the 10th Meeting of the ITPA SOL and divertor physics Topical Group, Avila, Spain

The 10th meeting was held in 7-10, January 2008 at Avila, Spain, and the meeting was hosted by CIEMAT. There were 40 participants: 22 from EU, 8 from US (incl. Canada), 3 from Japan, 4 from ITER, 1 from Russia, 1 from China. The meeting was arranged just before the abstract selection meeting by Program Committee of the Plasma Surface Interactions Conference. Meeting was well organized.

The primary foci of this meeting were discussions and presentations on predictions of ITER heat and first-wall loadings, as well as ITER T retention in the case with all-W plasma facing components. There were also sessions on ITER diagnostics and what can be accomplished during the ITER H phase. Finally, there were general sessions on D/T retention, dust and divertor physics.

The ITER predictions session (4) was quite productive. The speakers had spent significant time projecting current data to ITER for a number of areas. The various calculations on a given subject (e.g. wall fluxes) agreed to factors of 2. ITER wall fluxes were predicted to be in the range 3-$7\text{x}10^{23}$/s. Issues for further discussion were the division of fluxes over the various first-wall surfaces (e.g. how much went to the upper divertor vs inner wall vs outer wall PFCs), the heat load distribution for ELMs, and the transport into limiter shadows (needed for limiter design).

Predictions of T retention were presented both for the current mix of materials in ITER (Be, W and C) as well as for an all-W ITER. The dominant retention for carbon PFCs was confirmed to be co-deposition as expected. Echoing the ‘Progress in the ITER Physics Basis’ the T retention for the current mix of materials, Be, C and W, leads to the 350g limit being reached after ~ 300 discharges.

The new T retention estimates presented at this meeting were on the reduction obtained by going to an all-W ITER. The estimates presented by the EU and US representatives were close. Plasma irradiation was the primary source of T retention in W with the added effect of neutron damage (based on very little published data) leading to a significant increase over ion-impact alone. The end result was that the 350 gram in-vessel T limit was reached later than for C/Be/W, at between 2000-3000 discharges, an increase of ~ x10 over the current mix of materials. The differences between laboratory ion-beam data, which these estimates are based on, and C-Mod results, are large and need to be understood to know whether the predictions for ITER are correct.

The level of hydrogenic retention in tungsten generally is lower than for carbon based on the factor of 10 reduction in D retention (post-campaign analysis) in ASDEX-Upgrade following conversion to fully-W and from high-flux/fluence Pilot-PSI showed very low fractions of incident D^+ retained (~ 10^{-6}). While He^+ impacting surfaces leads to nanostructure growth (up to ~ microns in depth) on W in a narrow range of surface temperatures we have yet evaluate to evaluate their importance for ITER T retention.

The non-saturation of hydrogenic retention in carbon (linear in ion fluence) was demonstrated in the new Tore-Supra dedicated long pulses over 2 weeks. The co-deposition of D with Be appears to be more complicated, depending on the incident energy of deuterium, the Be deposition rate and the dependence on surface temperature. Use of scavenger gases for C layer inhibition as well as Thermo-oxiadation for D removal were encouraging in being able to reduce and remove D retention respectively.

A draft plan for the ITER H-phase was presented by A. Loarte (for the ITER Office) for comment. A number of assumptions of availability of the machine as well as diagnostic and heating systems was assumed together with a phased plan for bringing all of those systems up to full levels by staging the powers and plasma current levels at reduced pulse length (20-25s). It will be impossible to properly test the PFCs up to full power-handling capability nor the ability to handle transients such as ELMs or disruptions at full levels. Nevertheless close attention should, and will be, paid to monitoring the various surfaces during the startup phase to determine if they fail at those lower power levels.

A second aim during the H phase is to monitor the H retention. While ITER currently has no plan for this it is clear that one is needed and the SOL/divertor group will work closely with the ITER diagnostics group to agree on the proper diagnostics as well as a plan of measurment.

All of the above H-phase PFC information will go into the decision of whether to replace the C divertor PFCs with W before the D or DT phases.

Wall and divertor loadings during transients continues to be a difficult subject to understand. The ELM characteristics when they are incident on the outer limiter in ASDEX-Upgrade were inferred to be in the range 100 eV per electron-ion pair. New measurements of ELM erosion at the outer divertor (ASDEX Upgrade, session 3) show that as the local temperature drops the between-ELM erosion also drops relative to that occurring during ELMs (which is concentrated near the strike points). Increasing the impurity level through impurity puffing may increase divertor radiation but also increases erosion during ELMs. The current model of ICRF-enhanced sheaths, so-called sheath rectification, were contradicted (C-Mod) in that the plasma potential increases proportional to P_{RF} and potentials generated even when the current loop is broken by insulating tiles. The ICRF-enhanced plasma potential has been observed to occur on flux tubes connected to the antenna as well as passing in front of the antenna.

The ITER plans for diagnostic coverage of the main chamber, SOL and divertor were reviewed. During the meeting itself it was clear that several measurements were either not available or not well-specified (e.g. wall fluxes, injected gas, dust…). It was agreed that those diagnostics will be reviewed by the SOL/divertor group between meetings for spatial/measurement coverage and specifications. The gas injected during, and gas remaining, after a discharge were discussed at length and are the subject of discussion following the meeting. The group suggested that some method of H retention must be made during the H phase either through a small fraction of D injected, or through lowering the ambient levels of H in the chamber. Diagnostics for surface temperature measurements were reviewed and several laboratory developments of in-situ surface analysis (ion beam analysis and laser beam erosion/desorption) were presented.

Dust is of course a very important issue for ITER and was discussed as part of the diagnostics session. Based on a lack of experimental data the IO is assuming that dust is generated at a rate equivalent to the erosion rate of all surfaces in the machine. The SOL/divertor group felt that the specification should be lower as evidenced by JT-60U results, more like 10% of the net surface erosion rate. We hope to also form a working group to obtain a better specification for the dust generation.

IEA/ITPA multi-machine collaborations

Reports and proposals of the IEA/ITPA multi-machine experiments were presented by spokes-persons (some missing numbers were previously closed). The following 14 multi-machine experiments were proposed in IEA/ITPA CC meeting on Nov. 30, 2007 in Oxford.

14 proposals for 2008 IEA/ITPA DSOL inter-machine experiments were presented (DSOL-11 was combined to MDC-1). Devices and spokes-persons for each proposals are followings;

DSOL-1 Scaling of Type-1 ELM energy loss and pedestal gradients through dimensionless variables (Loarte)
Devise: JET, DIII-D, ASDEX Upgrade, JT-60U

DSOL-2 Injection to quantify chemical erosion (Brezinsek)
Devise: TEXTOR, JET, AUG, JT-60U, DIII-D + MAST, Tore Supra

DSOL-3 Scaling of radial transport (Lipschultz)
Devise: C-mod, MAST, DIII-D +JET

DSOL-4 Comparison of disruption energy balance in similar discharges and disruption heat flux(A. Loarte, D. Humphreys, G.Pautasso)

DSOL-5 Role of Lyman absorption in the divertor (Lisgo)
Devise: C-Mod, JET

DSOL-8 ICRF Conditioning for hydrogen removal (Ashikawa)
Devise: LHD, HT-7, EAST, AUG, TEXTOR

DSOL-9 Tracer injection experiments to understand material migration (Philipps)
Devise: JET, DIII-D, TEXTOR, ASDEX-Upgrade, JT-60U

DSOL-11 Disruption mitigation experiments (Whyte) this 2008 proposal is combined to MDC-1

DSOL-12 Reactive gas wall cleaning (Stangeby)
Devise: TEXTOR, HT-7, EAST, DIII-D

DSOL-13 Deuterium codeposition with carbon in gaps of plasma facing components (Krieger)
Devise: data from AUG, TEXTOR, MAST, DIII-D, ToreSupra, C-MOD, JET, FTU

DSOL-14 Multi-code, multi-machine edge modelling and code benchmarking (Coster)
Devise: Codes only (Database in AUG, JET, DIII-D, JT-60U

DSOL-15 Inter-machine comparison of blob characteristics (Terry)
Devise: C-Mod, PISCES, DIII-D, JT-60U, VTF, JET, AUG, TJ-II, NSTX, TEXTOR+TCV,TS, HT-7

DSOL-16: Determination of the poloidal fueling profile (Groth)
Devise: DIII-D, AUG, JET, MAST, C-MOD, JT-60U

DSOL-17: Cross-machine Comparisons of Pulse-by-Pulse Deposition (Skinner)
Devise: NSTX, AUG, JET

DSOL-19: Impurity generation mechanism & transport during ELMs for comparable ELMs across devices (Loarte)
Devise: JET, DIII-D, ASDEX-Upgrade, Alcator C-mod, JT-60U, MAST

In the discussion, additional proposals will be proposed to ITPA TG after discussion of contents among contact persons in applopriate devises.Tentative titles and spokes persons are followings;

(5) Comparison of isotope effect of H/D edge and divertor plasmas for extrapolate to ITER H-operation (by Fundamenski)
(6) Dust movement study by dust injection (by West)
(7) Dust dynamics study (by Krasheninnikov)
He will start to discuss with dust research community in EU, and will make a proposal based on existing database. At the same time, another proposal for (4) dust growth on the material will considered to be proposed.
(5) Evaluation of carbon and methane generation fluxes based on particle balance study review work was done by Tsitrone, and some presentations in last PSI) (by Tabares)

2. High Priority Research Areas

Area	Progress Reported at ITPA Meeting
1.0 Improve understanding of Tritium retention and development of efficient T removal methods.	
1.1 Comparison of post-campaign and shot-integrated studies of retention w/respect to ion fluence (new)	Thierry Loarer (reported at Toledo PSI conference) and Volker Philipps (reported at Avila ITPA meeting) have shown that post-campaign measurements of the amount of gas injected that is retained for carbon PFC tokamaks is generally in the 1-5% range. The values for single discharge retention fractions, made using gas balance measurements, are typically higher, more in the range 7-50%. The most recent and accurate JET gas balance measurements ('static gas balance' with valves to pumps closed) give retention fractions in the range of 7-15%. It is the general agreement that there are several plausible reasons for the difference between post-campaign and gas balance measurements. These include gas evolving out of tiles during the campaign due to disruptions, conditioning, and general outgassing. **This task will now be closed.**
1.2 Lab & tokamak experiments and modelling of deep D retention in high- and low-Z materials (new)	Analysis of high-Z material tiles by the a number of laboratories (e.g. IPP, U. Wisconsin, U. Toronto) have shown that hydrogen can diffuse deep within such materials, long 'tails' deep within the material. This is consistent with the diffusivities of H or D in tungsten and molybdenum. One would not expect such behavior in carbon due to its much lower diffusivity. However, analysis of Tore Supra tiles has shown H and D are defused into microns from the surface. Analysis of laboratory experiments with CFC graphite show that the H and D diffuse deep into the material along pores due to the porosity of the graphite. This process then scales as the square root of time and should be less of a concern than co-deposition of C with D on the surface which is linear

Area	Progress Reported at ITPA Meeting
	in time. **This task will now be closed.**
2.0 Understand the effect of ELMs/disruptions on divertor and first wall structures	
2.1 Exploration of the effect on the SOL and power loadings of ELM mitigation (new)	A series of experiments have shown that the power carried by an ELM decreases as it crosses the SOL (basically all tokamaks). This is due to the draining of energy and particles along the field to material surfaces. Experimental measurements form ASDEX-Upgrade with a probe array show e-folding lengths of ~ 2.5 cm. Models by Kirk and by Fundamenski have been used to fit existing data and, projected to ITER, give values in the range 2 (Kirk) to 5-10 (Fundamenski) cm. The analysis of changes in ELM characteristics, including power delivered to PFCs, is ongoing as the study of such plasmas is still developing. **This task is ongoing as we continue to investigate the effect of ELM-mitigation on first-wall and divertor heat loads.**
2.2 Scaling of disruption mitigation to ITER (new)	Analysis of data from a range of tokamak sizes (Whyte, EPS 2007) indicates the total time from gas injection to the end of thermal quench is scalable to ~ 20 ms in ITER. This would present no issues for C and W but might for Be. The concern is that there will be a short burst of most of the energy during that 20 ms and it will lead to Be wall damage. Additional issues with massive gas injection for disruption mitigation to be addressed are what the implications are for the gas handling system, the generation and mitigation of runaways, and dust. **This task will now be closed.**
3.0 Improve measurements & understanding of plasma transport to targets and walls to better predict heat loads and effects on the core	
3.1 Code-code comparisons including impurities - specifically carbon (new)	Code-code comparisons have slowed down in the last year due to lack of funding. The main result is the start of the effect if impurities in comparing EDGE2D and B2 by Gulejova. **This task is ongoing.**
3.2 ITER neutral density benchmarking of model physics to	S. Lisgo finished the analysis of C-Mod data for a range in detached conditions and divertor pressure (and thus Lyman alpha trapping). It was found that the trapping effect is

Area	Progress Reported at ITPA Meeting
current experiments (underway)	significant for the divertor conditions. It was hoped that JET would run an experiment this year to make additional measurements for modelling ~~but that did not happen~~. **This task will now be closed.**
4.0 Understand how conditioning & operational techniques can be scaled (or not) to future devices	
4.1 Implications of a metal wall (no coating) for startup, fuel retention, density control and core impurity levels (underway but 1st step done)	ASDEX-Upgrade operated for one full campaign with no boronization and fully-W-coated surfaces. In cases without ICRF the W level was sufficiently low (< 10^{-4}) that the effect of W radiation on confinement was minimal. Results with ICRF from C-Mod and ASDEX-Upgrade are consistent in showing that ICRF enhances sheaths which then leads to enhanced surface sputtering and core plasma performance degradation. Initial results for hydrogenic retention from ASDEX-Upgrade indicate a drop in retention of a factor of 5-10 in going from carbon-dominated to W-dominated PFCs. **We will close this task.**

3. Proposed high priority research areas for 2008-9

The following are currently under discussion as the revised high-priority research areas for 2008-2009

1.0 Improve understanding of Tritium retention and development of efficient T removal methods.

1.1 Compile high-Z experience regarding hydrogenic retention in tokamaks and laboratory studies (new)

1.2 Initiate studies on neutron damage and how that leads to T retention (new)

2.0 Understand the effect of ELMs/disruptions on divertor and first wall structures

2.1 Exploration of the effect on the SOL and power loadings of ELM mitigation (continuing)

2.2 Study runaway effects in disruptions and how to nullify them (new)

3.0 Improve measurements & understanding of plasma transport to targets and walls to better predict heat loads and effects on the core

3.1 Code-code comparisons including impurities - specifically carbon (continuing)

3.2 Identify discrepancies between codes and experiment for SOL and divertor (new)

4.0 High-Z operational experience

4.1 Exposure of tungsten to He fluence and effects on surface properties (new)

4.2 Characterize the level and processes involved in RF enhancement of erosion (new)

4. Future meetings

The next (11th) meeting of the Div/SOL Topical Group tentatively will be planned in 15-18 (3.5 days) September, 2008, Nagasaki, Japan, which is coordinated by Profs. Tanabe and Sakamoto (Kyushu Univ.) with the timing just after the ICPP international conference (International Congress on Plasma Physics 2008, September 8-12, Fukuoka, Japan).