Unique Opportunities for US Collaboration in Plasma-Wall Interaction Research in JET and EAST

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Summary

Material migration of eroded material from wall and target armour has the potential to cause serious operational problems for high-duty cycle tokamaks starting with ITER. Net erosion and deposition of PFC material is estimated to be of order ~10^2 (~10^4) kg/year for ITER (reactors). The deposition of such massive quantities of PFC slag material, much of which will inevitably occur where not wanted, has the potential to seriously interfere with tokamak operation, including disruptions due to 'UFO's, unacceptably high dust levels, etc. The deposition also traps and retains tritium by co-deposition at a very high rate for low-Z materials such as Be and in-vessel limits on tritium inventory can be rapidly exceeded. It is important therefore to understand and to learn how to manage material migration. Unfortunately present understanding is seriously deficient, primarily because of the dearth of controlled experiments. Studies of material migration have been performed for decades in tokamaks; however, they have virtually all involved campaign-integrated net erosion and deposition which is difficult or impossible to

interpret because of the great variation in plasma conditions involved. There are now unique opportunities in JET and EAST to perform controlled experiments involving material migration in single-condition, well-characterized plasmas, where the test surfaces can be inserted and removed after as many or as few shots as wanted. These experiments will make it possible for the first time to properly benchmark the material migration computer codes such as DIVIMP which is being used by ITER to predict the likely impact of material migration on its operation. The first steps have been taken to establish US collaborative work on JET and EAST on these benchmarking studies, and the US participation has been welcomed. These tokamaks have longer (JET) and much longer (EAST) pulse lengths than any US tokamak and they use (JET) or will use (EAST) full W and/or Be PFC systems, which ITER will use but which US tokamaks do not. Therefore US collaboration with these foreign devices affords unique opportunities for US fusion researchers to contribute to, and potentially lead in, aspects of this critically important field of material migration. The US fusion program will need to assign sufficient resources to these initiatives to be able to realize them.

Content

Introduction

The Appendix “Managing material deposits in high duty cycle tokamaks to avoid fouling plasma operation”, discusses the magnitude of material migration expected to occur in high duty cycle devices starting with ITER, the problems this will cause and the need to understand, control and manage material migration. Section I describes a specific proposal for US collaboration in JET, “Proposed fabrication, installation and use of a Be Probe-Limiter in JET to measure Be sputtering yields and net erosion/deposition under well defined, controlled and measured plasma conditions “, by way of illustrating the type of unique opportunities that exist for the US fusion program to collaborate in plasma-wall interaction research on JET. Section II similarly describes a specific proposal for US collaboration on EAST “Using the unique MAPES, "Materials and Plasma Evaluation System", on EAST to benchmark the DIVIMP impurity code being used to predict the ITER Be wall lifetime”.

I. An opporunity on JET: a proposal for installation in JET of a Be Probe-Limiter and measurements of Be gross and net erosion rates.

I.1 Background and motivation

The ITER Organization is presently using the LIM-DIVIMP code to assess the tritium retention and erosion-wear of the ITER Be wall; first results were presented at the 2010 PSI Conference [1]. From [1]:

"Thus, when the uncertainties due to plasma specification and effective sputtering yield are combined, the calculated net peak erosion is in the range [0.0025 - 0.055] mmBe/h corresponding, in terms of PFC lifetime, to between ~1640 and 36000 full Q_{DT} = 10 discharges.

..... considering the error bars on the empirical scaling of Eqn. 1, leads to a T-inventory in the range 0.48 → 5.4 gT/h for the high flux case (for fixed Y_{eff} ~7%), corresponding to an operational limit between 1070 and 12000 Q_{DT} = 10 reference discharges.

…… As well, as noted, the present analysis assumes ΔR_{sep} ~ 10 cm. For the "worst case" ΔR_{sep} ~ 5 cm, in which case the erosion and tritium retention rates will be roughly 3 times larger, a more accurate assessment, in progress, will take into account the different magnetic configuration at the top of the vessel."

These estimates, while covering a considerable span, include T-retention rates and wear-erosion rates that would pose serious problems for ITER which plans on 3000 Q_{DT} = 10 reference discharges per year when fully operational. From [1]:

"The net peak erosion assessed for the highest and lowest density conditions expected during the Q_{DT} = 10 reference plasma scenario ranges between 2.5 10^{-3} and 0.055 mm/h Be (accounting for uncertainties in the ITER plasma conditions and Be sputtering yields), showing that PFC lifetime

(evaluated between 1640 and 36000 full performance discharges) could be potentially an issue for ITER."

As indicated by the above quotations, the uncertainties involved are very large. This is in significant part due to very large discrepancies in the database for the sputtering yields of Be. From [1]:

"When combined, these three sets of measurements (ion beam, PISCES-B and JET) lead to an extremely large uncertainty on Y_{eff}, with values in the range 0.01 → 0.50."

Quite recently experiments on PISCES-B have found further unexpected and unexplained behavior of Be erosion and deposition in plasmas:

"A simple set of experiments was conducted in PISCES-B in order to reassess the balance between erosion and deposition of Be in Be containing edge plasmas. Solid Be targets were exposed to pure and Be seeded D discharges. Ion energies ranging between 15eV and 150eV, ion fluxes ranged from 1 to 5 10^22 m^-2s^-1 and Be concentrations of up to 3% were applied. D flux was measured with a double probe. Be concentrations were determined spectroscopically with the BeII emission line at 313.1 nm as well as by measuring the mass gain of a floating Be target. Present day transport models assume reflection coefficients of Be and D from Be as well as sputtering of Be by D and Be according to the binary collision approximation (e.g. by TRIM.SP). Given these reflection coefficients and sputter yields erosion should be balanced by the incoming Be flux if the Be seeding rate is equal to the sputter yield of Be. The experiment revealed that the measured net erosion rate is not changed at all when the Be concentration is equal to the sputter yield. Only when the seeding rate is an order of magnitude higher than the expected one net erosion rates are significantly reduced. Experiments with Be seeded He

discharges reveal the same behaviour excluding that the formation of BeD might explain the unexpected behaviour.” Thomas Schwarz-Selinger, reported at May 2011 ITPA DIVSOL meeting in Helsinki.

A further major uncertainty is involved in *all* code calculations of net erosion and deposition of *any* PFC material: almost no well defined experiments have ever been carried out in a tokamak of net erosion and deposition for measured, controlled plasma conditions. Many studies have been carried out for campaign-integrated conditions but, in truth, such experiments are scarcely interpretable. It has not been possible to get the dedicated tokamak time to carry out a large number of repeat discharges for a limiter or divertor target which was carefully measured (thickness of the cladding) before installation, was then exposed only to measured and controlled repeat discharges, and was then removed for post mortem measurements of the cladding thickness. Such experiments have been considered too expensive of tokamak run time.

ITER, however, has now asked EAST to carry out such a dedicated test in order to benchmark the LIM-DIVIMP code that it is using for ITER predictions, as described further in Section II. EAST has agreed to install a special carbon-coated Mo limiter through a large port which will only be exposed to a number of repeat plasma shots, then removed for analysis by Bill Wampler, Sandia. Unfortunately EAST cannot install a Be limiter. In deuterium discharges the graphite limiter will experience the complexity of chemical erosion, which is not Be-relevant. In order to achieve a better approximation to a Be limiter, discharges in He are planned. The EAST experiment has the potential to provide uniquely valuable data for benchmarking DIVIMP, etc; however, since it will not involve actual Be, there is the need to obtain similar data from experiments involving Be. As noted above, there are a number of major questions affecting erosion, deposition and co-deposition that are ***Be-specific***, calling for definitive, tokamak measurements using Be in controlled, well-measured conditions. Anything less will leave too much doubt about how Be will behave in ITER.

In June 2010 I visited JET and discussed with Guy Matthews, leader of the JET ILW Project, the possibility of carrying out controlled net erosion/deposition studies of some Be component in JET, ideally some of the Be tiles located near the top of the vessel which would be carefully kept out of plasma contact except for the controlled, repeat discharges, then removed for measurements. Matthews said this couldn't be done since there are so many different operating conditions that are planned for the campaigns.

Therefore the challenge remains of devising an experiment where:

(a) Be sputtering yields can be measured in an actual tokamak environment under conditions as close as possible to those expected in ITER,

(b) a well-defined and controlled test can be carried out of net erosion/deposition of a Be edge component, that can be used to directly benchmark the LIM-DIVIMP code that the IO is using to predict the tritium co-deposition retention and erosion-wear of the ITER Be wall.

I.2 Proposed fabrication, installation and use of a Be Probe-Limiter in JET to measure Be sputtering yields and net erosion/deposition under well defined, controlled and measured plasma conditions

The proposal is to use the JET RCP (Reciprocating Probe) to insert a Be cylinder of ~ 5 cm diameter into the JET edge plasma - a **RCP Probe-Limiter –** and using a set of dedicated, repeat, controlled, well-diagnosed discharges to measure Be sputtering yields and net erosion/deposition of Be. The local plasma conditons will be measured by Langmuir probes included in the RCP Probe-Limiter.

In order for an inserted object to act as a limiter, in the sense of it experiencing *local* net erosion/deposition, it must be large enough that a substantial fraction of the particles sputtered from the object returns to the object (as neutrals and ions), where they can re-deposit. <u>If Be migration is largely *local* then we have essentially the same situation as with the ITER Be wall, despite the orders of magnitude difference in scale size, in that it presents the same test of the modeling tools, e.g. LIM-DIVIMP, to model a *local* net erosion/deposition scenario</u> - in contrast to the more complex scenario where the inserted object is so small that the source of Be depositing on the object is due to remote sources and the particles sputtered from the object end up depositing in remote sinks.

David Elder has run a number of LIM-DIVIMP cases that assume a solid Be object, 4cm X 4cm (poloidally X toroidally) and extending out to "infinity" radially, i.e. roughly the shape of the JET RCP. It was assumed that $T_e = T_i = 10$ eV at the Probe-Limiter tip with T_e decaying radially with e-folding length = 2cm, and T_i constant radially, which corresponds to typical JET edge plasmas. Various values were assumed for plasma density, n_{e0}, at the Probe-Limiter tip and an e-folding length of 2 cm. The results of the code analysis are given in Table 1.

n_{e0} [m^{-3}]	fraction of sputtered neutrals deposited on RCP Probe-Limiter as neutrals or ions
$2x10^{19}$	68%
$1x10^{19}$	50%

0.5×10^{19}	32%
0.1×10^{19}	8%

Table 1. Fraction of sputtered Be neutrals deposited on RCP Probe-Limiter calculated using LIM-DIVIMP.

Thus, if n_{e0} > ~ 10^{19} the RCP head will act approximately as a limiter, so far as *local* net erosion/deposition is concerned. There may also be a significant rate of deposition due to remote sources but that can be measured using Si sample inserts on the RCP Probe-Limiter and then taken into account.

For approximately the same plasma conditions as above the ITER analysis [1] calculated a net erosion rate of ~ 0.055 mmBe/h = ~ 15 nm/s. Wampler reports that the detection limit for a change in Be thickness (using an implanted Si marker) is ~ 1nm. Thus it would require just a brief exposure of the RCP Probe-Limiter to obtain measurable results to compare with the LIM-DIVIMP code modeling. This is important because the inertially-cooled RCP Probe-Limiter surface will heat up quickly for these plasma conditions. For the $n_{e0} = 2 \times 10^{19}$ case the parallel power flux density will be $q_{||} = 6.9$ MW/m^2, causing the surface temperature of the Be to increase as $T_{Be}(t) \sim T_{Be}(0) + 280t^2$, with t[s]. (If the Probe-Limiter is made of thin Be-cladding on Cu, then $T_{Be}(t) \sim T_{Be}(0) + 210t^2$, only a modest improvement.) Thus after 4 sec of exposure the Be surface will have increased by 560C, which should be tolerable. In fact, the normal operation of the JET RCP involves very rapid motion in/out, with speeds of ~ 0.5 m/s and up to 4 reciprocations per shot. Since the RCP drive cannot be altered to give slower reciprocation to give longer dwell times then more shots will be required to achieve measurable erosion/deposition. Alternatively, the RCP Probe-Limiter could employ its standard reciprocation speed and be briefly inserted more deeply into the plasma, resulting in higher erosion/deposition rates, with the added advantage of achieving a higher fraction of the locally sputtered Be returning to the Probe-Limiter.

The probe can be viewed spectroscopically using cameras equipped with BeI filters which make it possible to measure the sputtering yield based on ADAS values of S/XB. The cameras will only be able to resolve the probe head to some degree (~5 pixels across the head), thus they will tend to give a value which represents a weighted average of the sputtering yield over the probe-limiter head. We really need more localized measurements. The gross erosion rate, i.e. the sputtering yield, can be measured by including Si samples with a thin Be deposited layer in the form of a long, narrow strip, e.g. 1 mm wide and 50 nm thick: when the width of the Be strip is small compared with the mfp for the sputtered Be neutrals, then to a good approximation a measurement of the Be re-deposited on the Si surface adjacent to the Be strip gives the total Be sputtered from the strip, thus the local gross erosion rate [Be/m^2/s] and thus the local sputtering yield, using the LP measurements (I_{sat}) of ion fluxes incident on the Probe-Limiter. Also, the reduction in thickness of the Be strip can be directly measured by NRA; changes down to ~1 nm have been measured by Wampler for DIII-D DiMES Be samples [JNM **233-237** (1996) 791]. A 2nd Si sample without any Be strip would be used to measure the background Be deposition due to the Be originating from the rest of the RCP Probe-Limiter and from the general Be sources in the JET vessel.

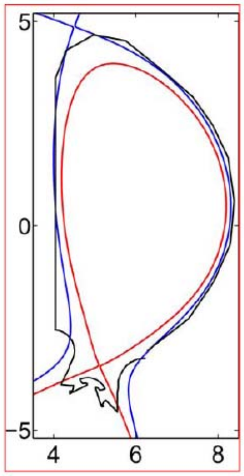
An important advantage of the Probe-Limiter approach is that it will be rather easy to repeat the experiment, for whatever reason, e.g. if something doesn't work first time, etc. By contrast the turn-around time for the EAST erosion/deposition test limiter will be quite considerable.

Such an experiment would constitute a nearly ideal benchmark test of the LIM-DIVIMP code presently being used by the IO to assess the projected performence of the ITER Be wall.

I.3 The proposal has been accepted by JET and assigned experiment number NP-16 in the 2011/2012 campaign

The proposal was formally presented at the JET Task Force E1/E2 meeting on Thursday 28 July 2011 where it was accepted and assigned experiment number NP-16 in the 2011/2012 campaign. The following 4 vgs are from a presentation given a few days earlier by Jari Likonen, VTT Finland and a Member of the JET Team, in support of the proposal.

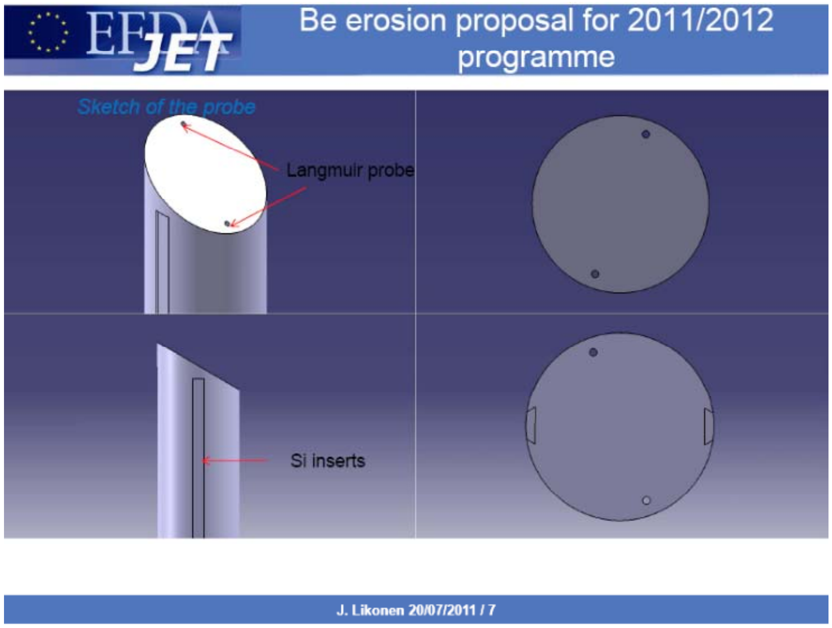
Background



- Large uncertainties in Be sputtering yields Y_{eff} (0.01-0.5) in ITER
- Large uncertainties in plasma parameters and Y_{eff} lead to Be erosion rates of 2.5-55 µmBe/h and T-inventory of 0.48-5.4 gT/h
- P. Stangeby has proposed an experiment aiming at:
  - Determination of Be sputtering yield
  - Investigation of Be erosion/deposition and benchmarking of LIM-DIVIMP code

Be erosion proposal for 2011/2012 programme

- *Aim: To use JET RCP probe to measure Be sputtering yields and net erosion/deposition of Be using a set of dedicated and well-diagnosed pulses*
- Two probe heads required (1st coated with Be and 2nd uncoated)
- Si inserts (with a ~1um thick Be coating, both on ion and electron side), will be analysed before and after exposure
- Background measurements with uncoated probe head and clean Si inserts, identical plasma conditions
- Plasma parameters will be measured with Langmuir probes
- Probe to be inserted as DEEPLY as possible (~10mm from separatrix) in order to have short mfp for sputtered Be atoms to be ionized
- One day most likely enough (for successful experiment)



I.4 US resources required to support this JET collaborative proposal

The Be Probe-Limiter head is being fabricated by JET, which is covering all the costs involved, except for the Be-coating, which is being done at UCSD using the General Atomics Be coating facility. The small costs involved in the latter are covered by existing contracts. It is not necessary that the US send experimentalists or diagnosticians to JET for the experiment since that work can be entirely done by EU researchers; however, it would be advantaegous for the US to exploit this opportunity to participate directly in such potentially seminal materials migration work. It would also likely lead to further US collaborations with JET in plasma-wall interaction research. New US resources are required to support the DIVIMP modeling to interpret the experiment.

Reference

"Modelling of Beryllium Erosion-Redeposition on ITER First Wall Panels", S. Carpentier[a], R. A. Pitts[a], P. C. Stangeby[b], J. D. Elder[b], A. S. Kukushkin[a], S. Lisgo[a], W. Fundamenski[c] and D. Moulton[c], J Nucl Mater **415** (2011) S165. [a]ITER, [b]University of Toronto, [c]JET.

II. An opportunity on EAST: using the unique MAPES, "Materials and Plasma Evaluation System", on EAST to benchmark the DIVIMP impurity code being used to predict the ITER Be wall lifetime

II.1 The preliminary experiments

Richard Pitts, ITER Organization, has arranged for EAST to carry out a controlled erosion/deposition experiment to benchmark the DIVIMP code that his group at ITER is using to estimate the erosion rate of the Be wall and the rate of tritium retention by Be co-deposition. See report on the first stage of this EAST project in the BPO Newsletter of 21 June 2011, which is reproduced here:

EAST erosion/deposition experiments

W. R. Wampler *(Sandia National Laboratories, Albuquerque);* ***G. Xianzu*** *(Academy of Science, Institute of Plasma Physics, Hefei);* ***R. Pitts, S. Carpentier-Chouchana*** *(ITER Organization, Caderache);* ***P. Stangeby*** *(University of Toronto, Toronto)*

Erosion and co-deposition of tritium onto the first wall and divertor in ITER are concerns that have prompted extensive analysis and evaluation. A recent initial experiment in EAST, the superconducting tokamak at the Chinese Academy of Sciences Institute of Plasma Physics in Hefei, is providing benchmark data on erosion for the ITER Project as well as practical experience that will help in the plans for subsequent experiments. The activity is a collaboration between Richard Pitts and Sophie Carpentier-Chouchana at ITER, Bill Wampler at Sandia, Gong Xianzu at the Chinese Academy of Sciences, Institute of Plasma Physics (ASIPP) in Hefei, and Peter Stangeby at the University of Toronto and a frequent collaborator on the DIII-D tokamak at General Atomics. The goal of the experiments is to obtain data from plasma exposure to compare with results from LIM-DIVIMP and also ERO code simulations being performed for ITER with realistic configurations for the ITER first wall panels. This work is motivated by the concern that erosion and re-deposition of beryllium near the secondary magnetic X-point at the top of the main chamber might increase tritium inventory in ITER.

EAST has moveable start-up limiters made of graphite tiles coated with silicon carbide (SiC). An initial test of the concept for the erosion experiment used these limiters with their existing tile geometry but with four of the tiles modified by depositing a carbon film over a tungsten coating that acts as a depth marker. Figure 1 shows the interior of EAST and the four tiles on the start-up (SL) limiter. Existing diagnostics measure the plasma edge conditions. The resulting data on erosion represent an average over the exposure to plasma discharges totaling 37,100 s in the 2010 campaign. This eNews article reports as yet unpublished data that has been provided to ITER and discussed among the collaborators.

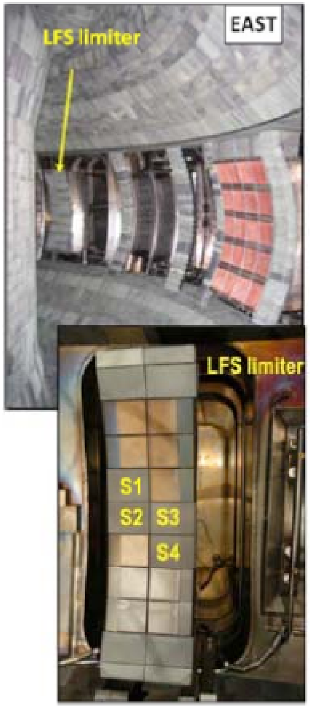


Figure 1. Interior of EAST (top) with location of startup-limiter; closeup of limiter (bottom) with coated tiles (S1 ... S4) indicated.

The critical measurement of erosion is the change in thickness of the thin carbon film. ASIPP sends the tiles to Sandia for deposition of the carbon film and the tungsten layer, and for measurements of the thickness of the carbon layer before and after the exposure in EAST. Deposition of a C over-layer ~1 μm thick follows initial vapor deposition of a tungsten layer ~1 nm thick. The thickness of the deposited C layer is measured using Rutherford backscattering (RBS) at the 3 sets of 9 points shown in Fig. 2. The points lay along a centerline down on the face of the tile and two other lines offset by 23 mm.

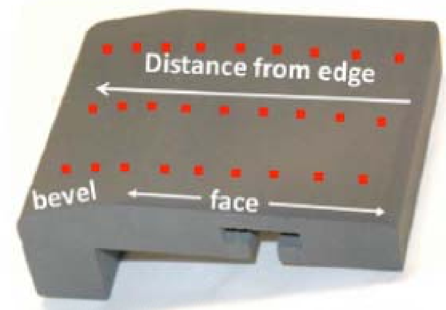


Figure 2. Pattern for Rutherford backscattering scans on EAST tile.

Rutherford backscattering provides measurements of energy loss by He^4 ions, initially at 2 MeV, that penetrate the surface and scatter back into a sensor that analyzes particle energy. The amount of the energy loss depends on the thickness (and stopping power) of the material from the front surface to the depth from which the He^4 scatters.

The upper plot in Fig. 3 shows RBS spectra for tile S2 before exposure in EAST. Also shown are the spectra from the tile before coating (heavy red curve) and from a reference sample of graphite with a thin tungsten coating on the surface (heavy blue curve). The step in the signal near 1200 keV corresponds to beam particles scattered from silicon at the surface. The step near 500 keV is due to scattering from carbon at the surface. The carbon over-layer causes an energy loss ΔE for particles scattered from tungsten and silicon on the coated tile. The heavy black curve is a simulation using SIMNRA [1]. The thickness of the carbon over-layer is determined by matching the energy shift ΔE in the simulated backscatter spectra. The variations among spectra from the individual spots (indicated in the legend) are mainly due to small variations in thickness of the deposited carbon layer.

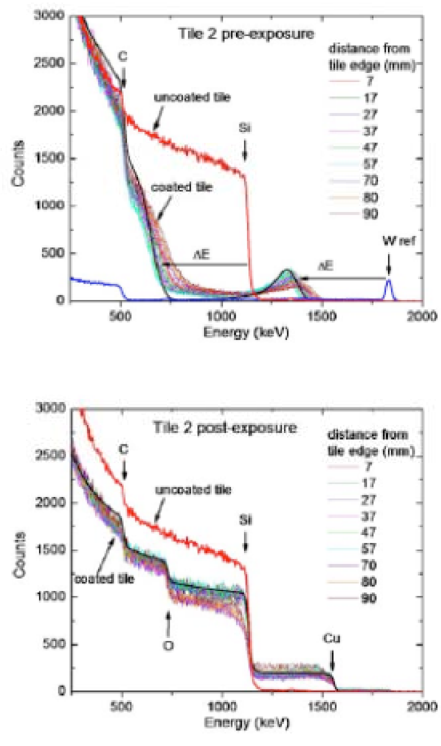


Figure 3. Rutherford backscattering data from pre- (top) and post-erosion (bottom) measurements at the same positions on tile S2.

The bottom plot in Fig. 3 shows post-exposure measurements for the same tile and locations. The second set of measurements were done after Bill Wampler and co-workers relocated equipment from their previous lab into Sandia's new Ion Beam Laboratory [2]. The long exposure time in EAST eroded much of the carbon layer. On a fine scale, due to the peaks and valleys associated with surface roughness, simultaneous erosion and deposition occurs [3].

The height of the Si edge signal gives the fractional area within the beam spot that is bare SiC. The Si feature in Fig. 3 shows that the carbon over-layer was completely removed from about 50% of the area. The additional RBS edges that appear in the post-exposure plot indicate that deposits containing C, O, and transition metals (Cr-Cu) cover the remaining half of the area. The deposited material has an average metal concentration of 0.6-3% of the atoms. The erosion and re-deposition process is fairly uniform over the four tiles.

The large amount of erosion means that only a minimum erosion rate can be deduced from the initial experiment. A planned follow-on experiment will take advantage of a new materials evaluation system on EAST that has a retractable probe similar in concept to the MiMES mid-plane materials probe on the DIII-D tokamak. The ASIPP staff expects to have this probe operational for the next campaign in October. The size of the probe (300×200 mm) permits exposure of fairly large targets to the plasma, and with a retractable probe, surface experiments of short duration will permit better control of the plasma conditions during the exposure.

References

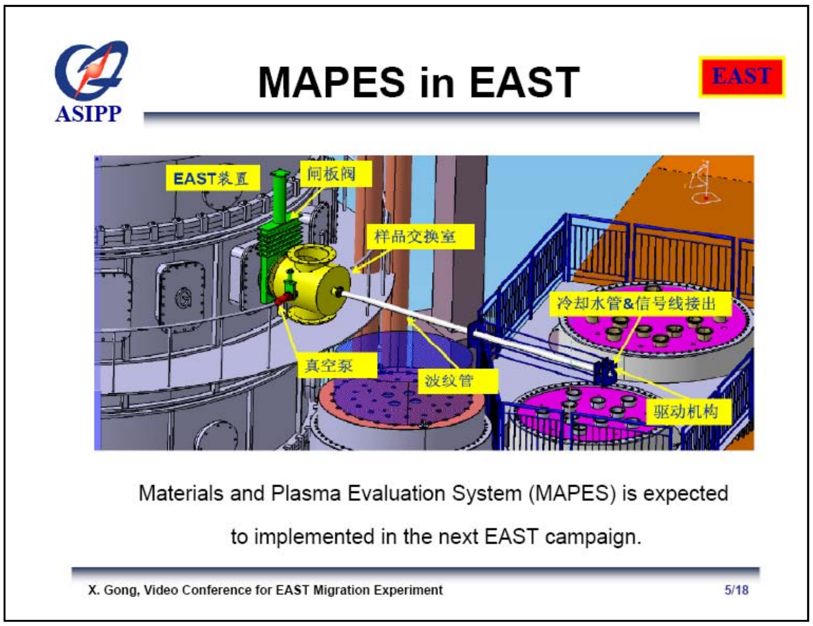
[1] SIMNRA: http://home.rzg.mpg.de/~mam/

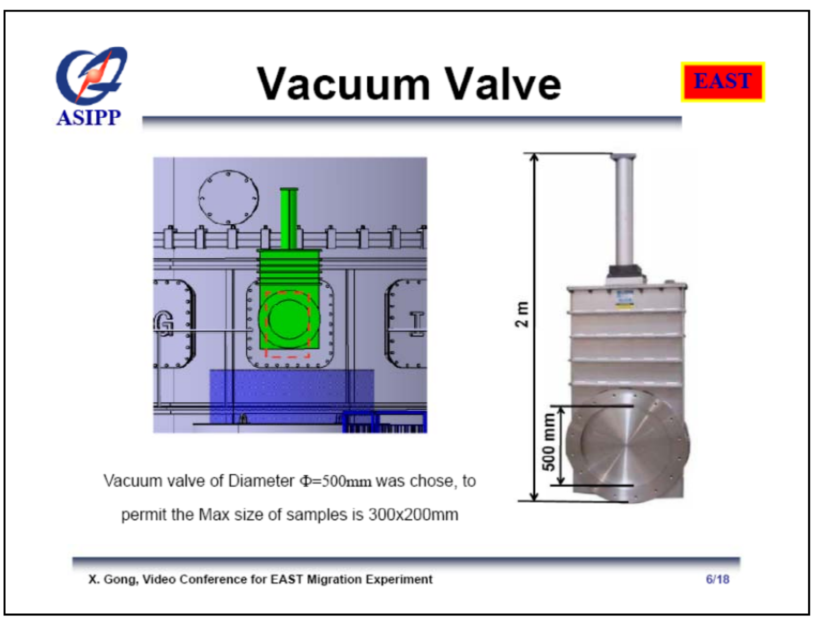
[2] For capabilities of Sandia's new Ion Beam Lab, see: http://www.sandia.gov/pcnsc/departments/iba/ibatable.html

[3] K. Schmid et al., Nuclear Fusion **50** (2010) 105004.

II.2 **The MAPES, "Materials and Plasma Evaluation System", on EAST will provide uniquely valuable opportunities for collaborative US-China research on Plasma-Wall Interactions**

The next and principal stage of the project will use the new plasma-wall interaction facility on EAST called MAPES, "Materials and Plasma Evaluation System":



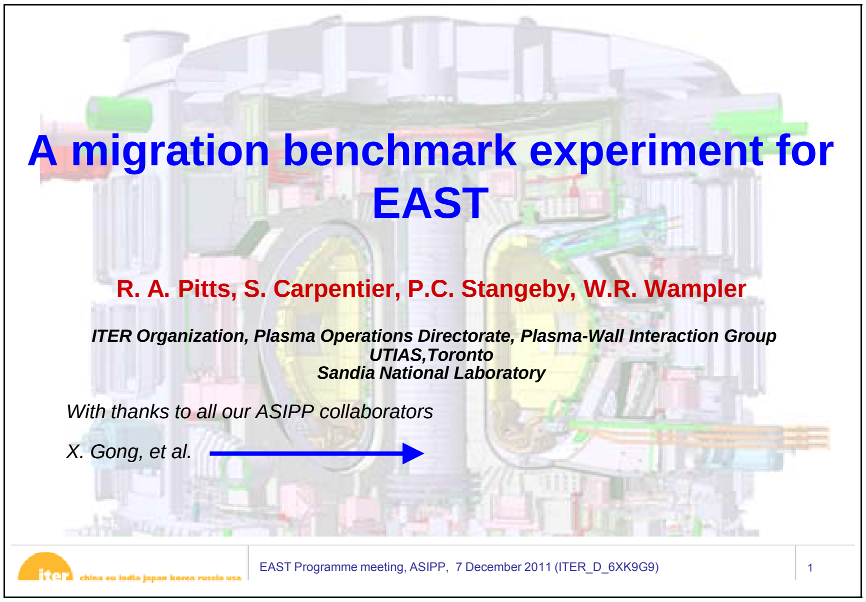


MAPES is a major step forward in boundary research capability. MAPES uses a very large gate valve, 500 mm, and so can insert very large test limiters. This will allow for controlled erosion/deposition experiments relevant to wall locations. For the wall, prompt, local deposition is not effective and impurity transport over larger distances is controlling. To benchmark DIVIMP regarding its use for analysis of the ITER wall requires test limiters that are as large as possible. Carbon will be used on the MAPES limiter with He-plasmas to avoid chemical sputtering, making for a close simulation of Be with DT-plasmas.

MAPES will make EAST a unique facility for boundary research, generally. It will be water-cooled and can be re-positioned during the shot, allowing it to be kept behind the fixed limiters for startup, rampdown, and only exposed during steady plasma conditions. It will be a nearly ideal facility for wall erosion and deposition research, generally.

II.3 **The formal request by ITER to EAST for dedicated run time**

Richard Pitts is now making the formal request to EAST for dedicated time to carry out the actual benchmarking experiment:



Who & Email	Action
Gong Xianzu Qian Jinping (jpqian@ipp.ac.cn)	Coordinator Plasma operation (He , OH) Start-up, Ramp up/down, Configuration, EFIT
*Xia Shibo (xiasb@ipp.ac.cn)	ITER-like shaping tiles : Design, Process and Assembly, Connected to the support of MAPES
Wang Fuming (wangfm@ipp.ac.cn) Yang Jianhu (yangjh@ipp.ac.cn)	IR-Camera Visible CCD
Xu Guosheng (gsxu@ipp.ac.cn) *Jiang Ming (mjiang@ipp.ac.cn)	Plasma performance ： Profiles of Te and ne (edge and core) … embedded Langmuir Probes
*Xia Shibo (xiasb@ipp.ac.cn) Ding Fang (Fding@ipp.ac.cn) Luo Guangnan (gnluo@ipp.ac.cn)	the new exchangeable sample capability (MAPES)

Motivation

ITER is concerned that steady state plasma erosion of beryllium on certain first wall panels may be too high

We have embarked on a 2 year modelling study to try and quantify this

We need to try and check if our baseline simulation methodology is sound

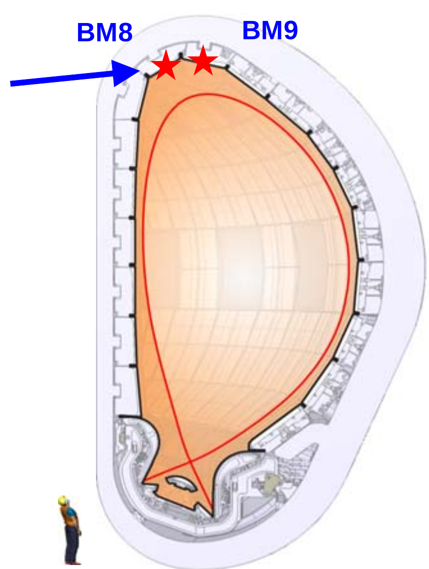
Two steps:

- Cross-check the simulations in a simple, accessible and well controlled experimental benchmark → EAST
- Compare with results of main chamber migration experiment on JET → more relevant (Be, upper divertor), but less flexible and ex-situ analysis will require ~ 1 year after experiment performed

Background

ITER

- Will operate near double null, with a close-fitting, conformal wall
- Will use poloidally and toroidally shaped wall panels
- Will use beryllium as first wall armour → low Z → low physical sputtering threshold → potentially high erosion → T-retention by co-deposition
- Will be long pulse, high performance, high fluence.



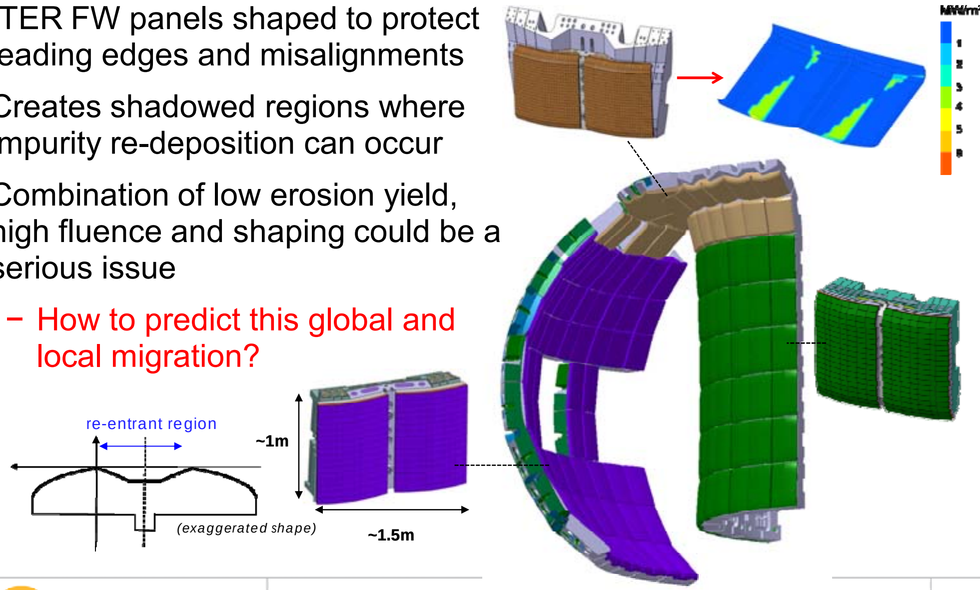
ITER first wall shaping

ITER FW panels shaped to protect leading edges and misalignments

Creates shadowed regions where impurity re-deposition can occur

Combination of low erosion yield, high fluence and shaping could be a serious issue

- How to predict this global and local migration?



Approach

Dedicated modeling effort in the IO

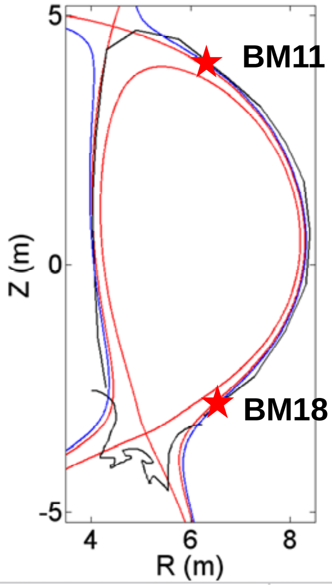
Benchmark the modeling with independent calculations

Make the issue a priority R&D topic within the ITPA DivSOL

Pursue dedicated experimental benchmark on EAST

- After a great deal of preparatory work, we are close to being able to execute these important experiments on EAST
- Need dedicated experimental time

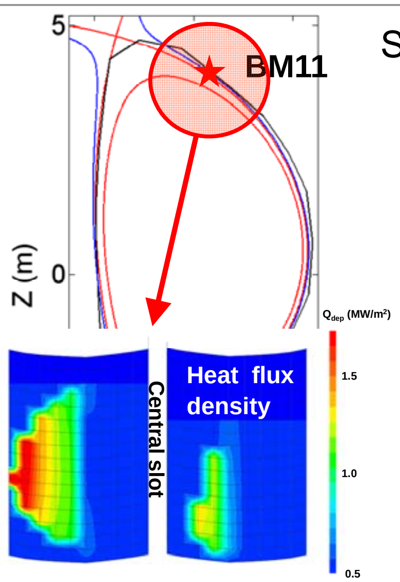
Modeling steps (1)



Start with the simplest situation

- Isolated blanket module, “limiter-like” plasma contact near 2nd X-pt region
- Reference burning plasma equilibrium
- Can be treated with existing 2D impurity transport code “LIM” (installed at IO)

Modeling steps (1)

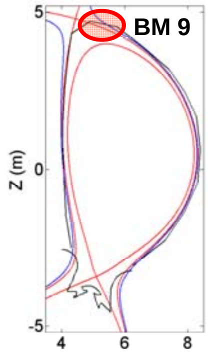


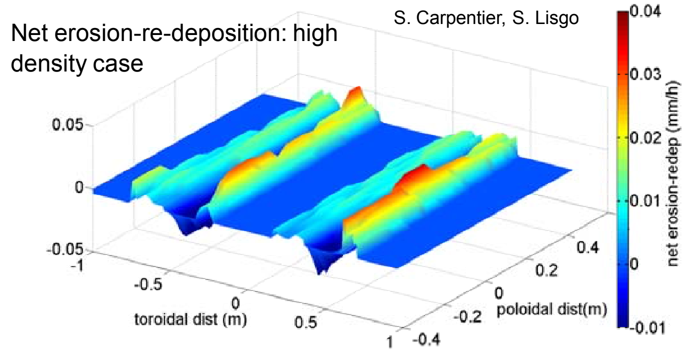
Start with the simplest situation

- Isolated blanket module, “limiter-like” plasma contact near 2nd X-pt region
- Reference burning plasma equilibrium
- Can be treated with existing 2D impurity transport code “LIM” (installed at IO)

Density, temperature and their profiles for particles arriving arriving at the FW panel taken from ITER Heat and Nuclear Load Specifications

Modeling steps (2)





Diverted configuration (secondary X-point region) - ribbon grid:

Y_{eff} ~6-7%, ~50-90% particles locally re-deposited (low – high fluxes)

Max. net peak erosion ~ $2.5.10^{-3}$ → 0.035 mm/h (low – high fluxes)

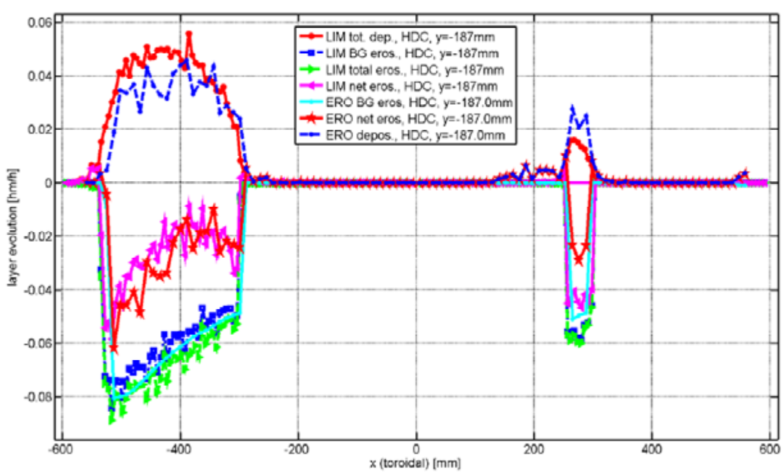
→ PFC lifetime ~ 1540 - 21 600 (assuming an allowed eroded thickness of ~ 6 mm)

T-retention (1 row = 18 BM9) ~ 2.10^{-3} → 4 gT/h (36 BM11-18)

→ Limit ~ 1440 – 3.10^{6} shots (assuming: 50:50 D:T plasma, maximum safety limit ~640 g)

Main goal

Our main aim on EAST is to provide an experimental benchmark for the limiter-like erosion-redeposition simulations



Essential requirements

Simple as possible experimental situation

- No complicating features like ELMs or LH heating which make the SOL plasma more difficult to understand
- Fixed plasma, high enough fluence, no disruptions
- Avoid carbon chemistry as much as possible (we seek to simulate the case for Be → no chemical sputtering) → **He plasmas**
- Best possible diagnosis of the local SOL plasma

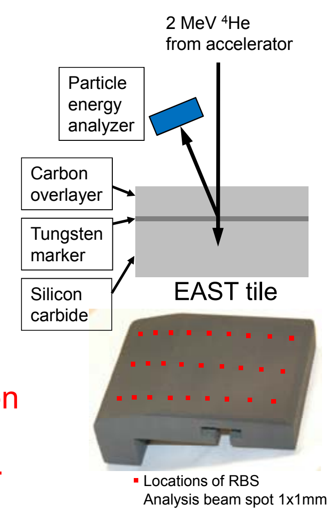
As close as possible to the first wall panel geometry we wish to investigate

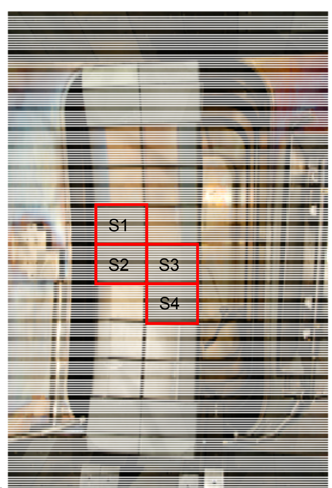
- Dedicated “shaped” limiter → new MAPES head on EAST

Technique for erosion-deposition

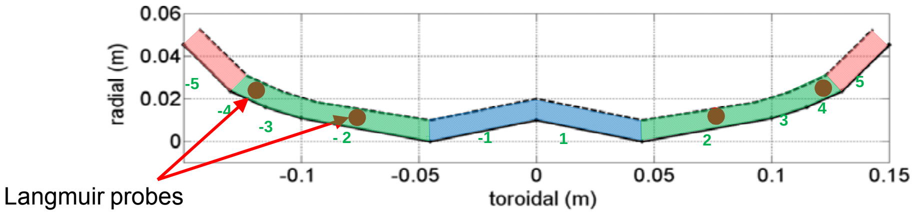
We use thin (~1 μm) carbon coatings

- Sputter deposition in Sandia
- Pre- and post characteriszation (RBS) of the layer in Sandia
- First tests of the technique were run on 4 tiles of one of the EAST outboard start-up limiters





New EAST migration tile proposal



Proxy for ITER first wall panel toroidal profile

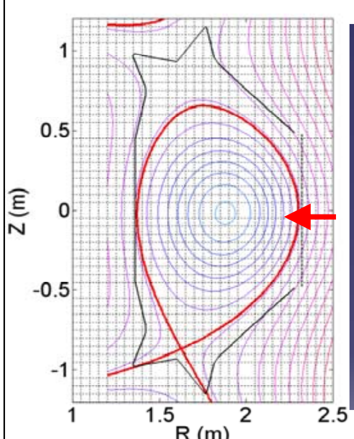
- Embedded Langmuir probes for local plasma parameters (n_e, T_e, fluid flow) and thermocouples for surface temperature
- Needs to be made in several "flats" to allow coating and surface analysis (dimensions and mass limits in Sandia coating device)

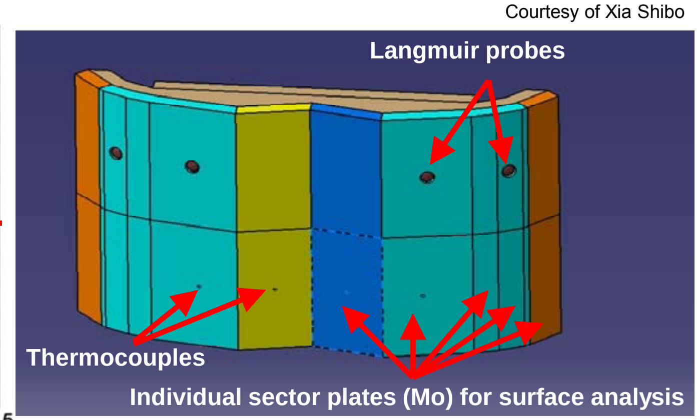
Propose metal (Mo or SS) → C coating directly on substrate

Limiter must be made in separate flats for coating equipment

First outline design for "blank test"

Courtesy of Xia Shibo





Toroidal profile according to generic logarithmic parametric function used to define ITER FW panel shape

Strategy

Execute a "blank test" with prototype migration tile on the MAPES head

- Proof of principle → will it work?
- Establish optimum location in the SOL
- Establish optimum experiment protocol
- Determine the discharge which will be repeated on the real migration tile (i.e. with C coating)

If the first MAPES head we expose is not the "final" version, this blank test must be performed early enough to allow a new head to be built and components to be coated and pre-characterized in Sandia before the final experiment

Final proposal

Blank test to establish conditions and test concept

- Difficult to assess how much machine time → developments required in discharge configuration, proof of principle with MAPES head, He plasma parameter scans to map out SOL profiles (requires radial scans with MAPES also) → R. A. Pitts proposes to visit ASIPP for some of these experiments
- Very important to get this done EARLY in the campaign → timing is key in this experiment

The migration test itself:

- Assuming all is properly scoped in the blank test, experiment itself requires ~100 – 200 secs of stationary plasma time with migration tile in fixed position
- Ohmic, He plasma. E.g. ~10 s flattop with MAPES in place → ~10 – 20 identical discharges, more if flattop duration restricted

II.4 **US resources required to support this EAST collaborative proposal**

As for the JET project described in Sec. I, almost all of the cost of fabrication as well as the experimental costs are being covered by the host lab; however, the coating and surface measurement work at Sandia has to be paid for by the US and new resources are required if this work is to continue. Also, as for the JET project, while it is not necessary for the US to send experimentalists and diagnosticians to EAST for the experiment itself, it would be quite advantageous to do so since this would almost certainly open up further possibilities for US-EAST collaboration on PWI using the unique MAPES facility, also EAST's very long pulse length and its planned all-W PFC system. The DIVIMP modeling required to iterpret the experiment would have been done by Dr Sophie Carpentier, the post-doc on ITER who has been doing the in-house DIVIMP modeling of the ITER wall, with direct support by the Toronto fusion group; however, Dr Carpentier's position has not been renewed (she will be employed elsewhere at ITER). The DIVIMP interpretation of the EAST benchmarking experiment can be done by the Toronto group but this will require new resources.

Appendix. Managing material deposits in high duty cycle tokamaks to avoid fouling plasma operation

A.1 Estimating the magnitude of the material deposition problem

In going from present devices to ITER/reactors, the annual energy load $E_{load}^{year} = P_{heat}\tau_{annual}$ will increase by ~10^3/~10^5 times, see Table 1. Very roughly, total gross erosion can be expected to increase similarly.

It appears likely that high duty cycle tokamaks starting with ITER will experience rates of net erosion and deposition of PFC material in the range of 10^2 – 10^4 kg/year or more, see estimates in Table 1, which it should be noted is conservative, since it is just for erosion of the main walls, not the divertor targets. Lackner, for example, has estimated for a tungsten PFC reactor that the net erosion due just to charge exchange neutral particle impact on the wall alone (neglecting erosion by plasma contact), would be ~ 5000 kgW/yr [1].

Device	P_{heat} [MW]	τ_{annual} run time [s/year]	E_{load}^{year} [TJ/yr] or (kilo-tons of TNT/yr)	beryllium net wall erosion rate [kg/yr]	boron net wall erosion rate [kg/yr]	carbon net wall erosion rate [kg/yr]	tungsten net wall erosion rate [kg/yr]
DIII-D	20	10^4	0.2 (0.048)	0.13	0.11	0.08	0.16
JT-60SA	34	10^4	0.34 (0.081)	0.22	0.19	0.15	0.27
EAST	24	10^5	2.4 (0.57)	1.6	1.2	0.82	1.8
ITER	100	10^6	100 (24)	77 [29*] {60***}	64	44 [53*] {54***}	80 [41*] {46***}
Vulcan [5]	20	10^7	200 (48)	120	100	70	150
FDF [4]	100	10^7	1000 (240)	610	500	340	740
Reactor	400	$2.5x10^7$	10000 (2400)	6500	5300	3700	7900 [5000**]

Table 1. Rough estimate of net erosion rate of main walls based on assumptions in Appendix A. Assumes 100% wall coverage by Be, B, C or W. Other estimates: *Kukushkin [2], **Lackner [1] and ***Behrisch et al [3]. P_{heat} [MW] is the heating power and τ_{annual} [s/yr] is the annual run time. 1 kilo-ton of TNT = 4100 GJ.

Such rates of net erosion – and the resulting deposition – are far beyond anything experienced to date in magnetic fusion devices. Even if the net erosion (wear) problem can be solved by periodic in situ refurbishment, the deposition of such massive quantities of material has the potential to interfere with tokamak operation, including:

(a) high rates of tritium retention due to co-deposition, a particular problem for low-Z like Be where D(T)/Be ratios in co-deposits can approach unity [6],
(b) high levels of dust due to exfoliated and spalled deposits,
(c) disruptions caused by exfoliated/spalled material entering the plasma, so-called "UFO's,"
(d) disruptions caused by the melting of metal deposits which are proud of power-loaded surfaces,
(e) other adverse effects due to buildup of unwanted eroded material at critical locations e.g. metal-bridging of tile gaps which could result in cracking of the coolant channels due to eddy currents and thermal stresses.

It will therefore be essential to learn how to manage material deposits to prevent their fouling tokamak operation by PFC slag.

A.2 Disruptions in Tore Supra caused by 'UFO's' from just ~1 kg of net deposition

In 2007-8 Tore Supra, operating with graphite PFCs, ran 18,000 s of discharges in 2 weeks dedicated to generating substantial net erosion and deposition with the objective of better assessing deuterium retention. An unanticipated operational problem was encountered: the deposits significantly interfered with operation by causing frequent disruptions due to 'UFOs':

> *"The main operational issue was linked to the appearance of UFOs (i.e. large particles with a high impurity content, detached from the wall, PFCs or antennas, and penetrating into the plasma), whose frequency increased dramatically during the campaign, triggering a phase of plasma detachment followed by a disruption in a number of cases…. In order to overcome this limitation it became necessary to modify the LH power waveform, from its initially constant value to a continuous ramp, from 1.2 to 1.6–1.8MW, and also to decrease the plasma duration down to ~80 s ….The continuous increase of the UFO frequency is believed to be related to the build-up of thick deposits in the shadowed regions of the toroidal surface…"* [7]

Following the campaign an extensive cleaning of the Tore Supra interior recovered 0.79 kg of carbon deposits [8]. While by present tokamak experience this is a very large amount of deposition, for ITER such amounts of deposition can be expected to occur every week of full power operation, Table 1. The operating problems encountered in Tore Supra were specific to carbon debris but it seems unlikely that serious operational problems could be avoided, regardless of the PFC material used, when deposition rates of order 10^2 kg/yr are involved. The situation for devices beyond ITER is considerably more serious.

It will be essential to learn how to regularly remove deposits produced from the net erosion of PFCs in order to prevent operational problems caused by PFC slag.

A.3 Net erosion from the divertor targets

As noted the rate of *gross* erosion at the divertor strike points of a reactor could be roughly 5 orders of magnitude higher than in present devices. The measured rate of *net* erosion in present devices is 0.1 – 10 nm/s [9], thus for a typical 10^4 s/year operation the rate of target surface recession is only $\sim 10^{-6} - 10^{-4}$ m/year. However, if net erosion were to also scale up by 5 orders of magnitude then for reactors the rate of recession of the target surface would be 0.1 – 10 m/year which is not acceptable. Fortunately the plasma conditions foreseen at the divertor targets of devices like ITER and FDF, $T_e \sim 5$ eV and $n_e \sim 10^{21} m^{-3}$, are such as to strongly suppress net erosion relative to gross erosion due to prompt local deposition of sputtered particles: when the ionization mean free path for the sputtered impurity neutral L_{ioniz} is less than the fuel ion larmor radius ρ_{DT}, then the strong E-field and frictional forces in the magnetic pre-sheath, of thickness L_{MPS} = 3 - 10 ρ_{DT}, promptly return the ionized impurity to the target. For $T_e \sim 5$ eV and $n_e \sim 10^{21} m^{-3}$, $L_{ioniz} < L_{MPS}$ for both high-Z elements like W and low-Z ones like C, see [10] and references therein. While the process of prompt local deposition should greatly reduce net erosion relative to gross erosion of the divertor targets, it cannot completely suppress it. In any case, it will be challenging to achieve divertor conditions like $T_e \sim 5$ eV and $n_e \sim 10^{21} m^{-3}$ in AT tokamaks, particularly at the experimental stage, and it may be that significant levels of *divertor* net erosion and deposition will have to be accommodated.

A.4 Net erosion from the main walls

We consider next the *walls* where substantial net erosion appears to be entirely unavoidable as a major source of PFC debris, Table 1. Plasma contact with the main chamber walls in divertor tokamaks is now known to be significant, see example of DIII-D shown in Fig. 1: as plasma density is raised and detached divertor operation is approached, the total ion loss rate to the main chamber walls becomes comparable to the total ion loss rate in the divertor [11]. The same behaviour is observed in C-mod [12].

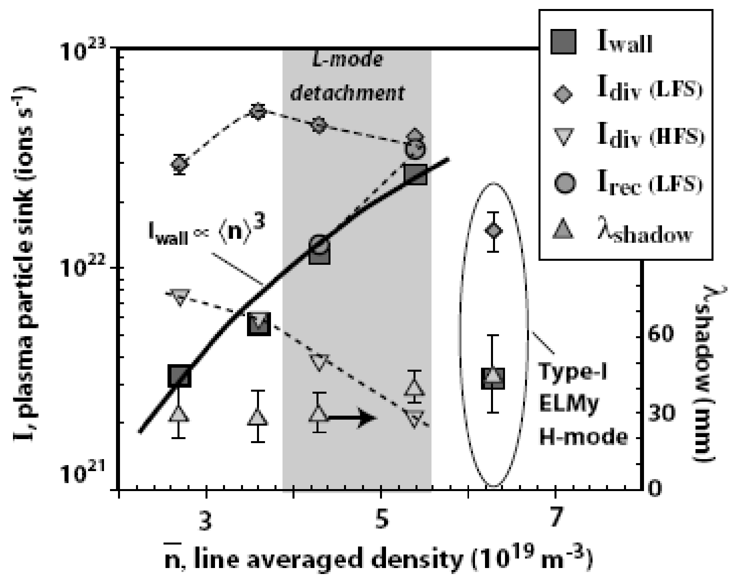


Figure 7. Strength of various ion sinks at the main-wall, I_{wall}, and divertor are shown versus line-averaged density for plasma geometry/SOL profiles as shown in figures 1 and 4. Ion losses to both divertor surfaces (on the LFS and HFS), and by divertor volume recombination are shown. The data are from discharges with L-mode energy confinement except as noted (attached ELMy H-mode). I_{wall} increases proportional to $\bar{n}^3$ in L-mode, while fitted λ_{shadow} (figure 4) is relatively constant versus $\bar{n}$.

Fig. 1. DIII-D. As detached divertor operation is approached the total ion loss rate to the main chamber walls becomes comparable to the total ion loss rate in the divertor [11].

Unfortunately, the processes of prompt, local (re-)deposition of impurities cannot be expected to occur at the main walls where the plasma density is low. Therefore on poloidal average at the main wall, net erosion ~ gross erosion, the latter being due to charge exchange neutrals and (relatively) dilute plasma contact. This situation poses a major challenge but also has the potential to provide a solution to the overall PFC problem in high duty cycle devices, as discussed below.

The plasma-wall contact (ionic) will directly result in substantial annual net erosion of material for ~10^7 s/yr operation. The indirect effect of the plasma-wall contact is also serious since it can result in very energetic cx neutrals, D^o and T^o, impacting the wall, causing strong sputter erosion. Although there are considerable variations depending on exact assumptions about the plasma-wall contact, some EIRENE code calculations for ITER conditions report average cx energies at the outer midplane (the worst location) of ~ 0.6 keV, with 20% of the cx atoms having > 1 keV [13]. There appears to be rather little experimental information on total P_{cx}/P_{heat}; however, for AUG a value of 0.4 MW/5MW = 0.08 has been reported [14].

In current tokamaks the main wall generally tends to be in a (spatially-averaged) state of net erosion with the lost material being transported to the divertor where it accumulates [15]. In detached conditions in DIII-D the entire divertor – both outer as well as inner targets – are in a state of net deposition due to PFC material migration from the walls [9]. A long standing concern in MFE research has been net erosion (wear) at the strike points of reactors. Indeed, as discussed above, even allowing for prompt local deposition of sputtered material, net erosion will still occur near divertor strike points, particularly the outer one, and since the absolute rates will be so high for high duty cycle devices, the target armour at the strike points could be lost at an unacceptably fast rate. However, in light of the copious net erosion from the main walls, divertor target wear may not be the critical issue. For reactor-relevant, high power, high density plasmas, it may be that the entire divertor will be in a state of net deposition due to the copious migration of eroded wall material. Such migration of wall material could compensate – indeed over-compensate – for the net erosion that would otherwise occur at the outer strike point. Because of the large wall area, the wall erosion itself may be tolerable providing it is not highly localized. The problem, however, will be to clear the PFC deposits out of the divertor rapidly enough to avoid disrupting plasma operation, i.e. to avoid deposition fouling of operation by PFC slag.

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Sub-Appendix. Basis of the rough estimates for the wall erosion rate in Table 1.

The rough estimates in Table 1 are based on the assumptions:

(i) physical sputtering by cx neutral tritons only; $\langle E^{cx} \rangle = 0.3$ keV T_0 assumed;

(ii) normal incidence sputtering yields [16] doubled to account for surface roughness: for (Be, B, C, W), Y^{cx} = (0.083, 0.056, 0.035, 0.0048);

(iii) for a conservative estimate, no sputtering included for D_0, He_0 or any plasma-wall contact (ionic);

(iv) no chemical sputtering or radiation enhanced sublimation of C at reactor-relevant temperatures, of ~ 700C < T_{wall} < 1100C [see references in [17], Chap 3];

(v) assumed $P_{cx} = 0.05\, P_{heat}$ [14, 18] thus $0.025 P_{heat} = \langle E^{cx}_{T_0} \rangle \varphi^{cx}_{T_0}$, where $\varphi^{cx}_{T_0}$ is T_0 particle flux to walls, thus gross erosion rate = $Y^{cx} \varphi^{cx}_{T_0} \approx$ net erosion rate for the main wall.

For the low-Z elements the results are not very sensitive to the value assumed for the triton's energy, but they are for W. Relative to the results for $\langle E^{cx} \rangle = 0.3$ keV, the multiplication factor for 0.1, 0.2, 0.3, 0.4, 0.5, 1 keV for C are: 0.54, 0.88, 1, 1.03, 1.07, 0.96, while for W they are: ~0, 0.13, 1, 2.0, 3.1, 6.2.