Tokamak edge physics and plasma-surface interactions

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thanks to
Andre Kukuskin, Philip Andrew, ITER Organisation
“The interaction of plasma with first wall surfaces will have a considerable impact on the performance of fusion plasmas, the lifetime of plasma-facing components and the retention of tritium in next step burning plasma experiments”

Outline

• The scrape-off layer (SOL) and divertor
  - SOL power width
  - Divertor detachment
• Plasma-surface interactions
  - Material lifetime – erosion and migration
  - Tritium retention
  - Dust
  - Mixed materials
• What to diagnose ....

CAVEAT: Edge plasma physics and PSI is a vast domain. Can only scratch the surface in this talk. Work referenced throughout the talk is listed at the end.
Divertor and SOL physics
Terminology: limiters and divertors

Scrape-off layer (SOL) plasma: region of open field lines

- Core plasma
- LCFS
- X-point
- LC
- "Upstream"
- Outer midplane
- Inner
- Outer
- Private plasma
- Divertor targets
- Vessel walls
- Limiter
Limiter and divertor phases in most JET shots

Ramp-up and ramp-down phases in ITER will be in limited phase, ~30 s long [5]. Full burn divertor phase of ~400 s for the $Q_{DT} = 10$ inductive scenario
Basics – SOL width, $\lambda_n$ [1]

Any solid surface inserted into a plasma constitutes a very strong particle sink.

In the high tokamak B-field:

$\Gamma_\perp \ll \Gamma_\parallel$

Thin Debye sheath ($\lambda_D \text{ few } 10\text{’s } \mu\text{m thick}$) forms at the surface controls flow of particles and energy $||B$

Quick and dirty estimate of $\lambda_n$ with diffusive approx. for cross-field particle transport (all ionisation inside LCFS):

$\Gamma_\perp \equiv n v_\perp = -D_\perp dn/dr \sim D_\perp n / \lambda_n$

$\Rightarrow v_\perp \approx D_\perp / \lambda_n$, $\lambda_n = \tau_\perp v_\perp \Rightarrow \tau_\perp = \lambda_n^2 / D_\perp$

$v_\parallel \approx c_s \sim (kT/m_i)^{1/2} \Rightarrow \tau_\parallel = L / c_s$

Then, if $\tau_\perp = \tau_\parallel$, $\lambda_n = (D_\perp L / c_s)^{1/2}$

- e.g. $L \sim 30 \text{ m}$,
- $T_{\text{LCFS}} \sim 100 \text{ eV}$, $c_s \sim 10^5 \text{ ms}^{-1}$,
- $D_\perp \sim 1 \text{ m}^2\text{s}^{-1}$ (near SOL)
- $\Rightarrow \lambda_n \sim 1.7 \text{ cm}$!!

cf. $a = 2.0 \text{ m}$ for ITER

Even worse for energy – see next ……
The problem with $\lambda_q$

- SOL width for power, $\lambda_q$, is also small and is an important parameter of the edge plasma.
- As for particles, $\lambda_q$ is determined by the ratio of $\perp$ to $||$ transport (e.g. cross-field ion conduction and parallel electron conduction: $\propto (\chi_{\perp}/\chi_{||})^{1/2}$), where $\chi_{\perp}$ is anomalous.
- Scalings for $\lambda_q$ can be derived from models and experiments, e.g.:
  - "2-point" analytic modelling: $\lambda_q \propto P_{SOL}^{-5/9}$, $P_{SOL}$ = power into SOL [1]
  - Scaling from H-mode experiments on JET [6]: $\lambda_q \propto P_{SOL}^{-0.5} B_\phi^{-0.9} q_{95}^{0.4} n_u^{0.15}$
  - ITER modelling [7] assumes $\lambda_q = 5$ mm, JET scaling gives $\lambda_q = 3.7$ mm (cf. a=2.0 m)
  - Very recent multi-machine scaling [8] gives $\lambda_q/R \sim$ constant
- Note also that the parallel power flux, $q_{||} \propto P_{SOL}/\lambda_q \sim$ as much as 1 GWm$^{-2}$ in ITER.

Stored energy scales strongly with tokamak major radius, $W \propto R^4$ [9]
But power deposition area in the divertor $\propto R\lambda_q$ only (~6 m$^2$ in ITER)

Bottom line is that despite its increased physical size, ITER will concentrate more power into a narrower channel at the plasma edge than today’s devices. The use of divertor detachment, radiation and geometry will be used to reduce the surface power flux densities to manageable levels, but careful monitoring will be critical $\rightarrow$ see talk by Albrecht Herrmann.
Power handling – ITER case (approx)

Max. steady-state power flux density permitted at ITER divertor targets: $q_\perp \leq 10 \text{ MWm}^{-2}$

Magnetic and divertor geometry alone cannot reduce the power to tolerable levels

Most of the parallel power flux must be prevented from reaching the plates $\rightarrow$ divertor detachment and high radiative loss

- Magnetic flux expansion
  $(B_\theta/B)_u/(B_\theta/B)_t \sim 4$ for ITER outer divertor $\rightarrow$ low field line angles at strike points ($\sim 3^\circ$)
  +
  Target tilting in poloidal plane ($\alpha \sim 25^\circ$ for ITER outer target)

$$\text{Area} \sim 4\pi R \lambda_q (B_\theta / B)_u \sim 0.2m^2$$

$\lambda_q = 5 \text{ mm}$

$$q_{||,u} \sim 500 \text{ MWm}^{-2}$$

$$q_\perp \sim 16 \text{ MWm}^{-2}$$

(adapted from [10])
The route to detachment (1)

Mean free paths for particle collisions are long:
\[ \lambda_{\text{coll}} \propto \frac{T_u^2}{n_u}, T_u \sim T_i, \lambda_{ee} \sim \lambda_{ei} \sim \lambda_{ii} \]

SOL collisionality: \[ \nu^* = L/\lambda_{\text{coll}} \] low

Power flow to surface largely controlled by target sheath:
\[ q_{\parallel,t} = \gamma n_{t,c} T_t + n_{t,c} \epsilon_{\text{pot}} \]

\[ \epsilon_{\text{pot}} = \text{potential energy per incident ion} \]

\( \nu^* \) rises as \( n_u \) rises, finite electron heat conductivity:
\[ q_{\parallel,\text{cond}} = -K dT/ds, K_{\parallel} = \kappa_0 T_t^{5/2} \] (note: \( \kappa_{0,e} \gg \kappa_{0,i} \)) allows parallel \( T \) gradients to develop \( \rightarrow T_t \) decreases, but pressure balance maintained (\( \nabla p_{\parallel} \sim 0 \)) so that \( n_t \) rises strongly (\( \Gamma_t \propto n_u^2 \)) \( \lambda_{\text{ion}} (\propto 1/n_t) \) decreases so that target recycling increases strongly \( \rightarrow \) flux amplification

As \( T_t \downarrow \), radiation loss increases \( \rightarrow T_t \downarrow \) further

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The route to detachment (2)

At sufficiently low $T_t$, (< 5 eV), neutral ionisation rate < ion-neutral friction processes (CX, elastic scattering). Momentum transferred from ions to dense cloud of neutrals in front of the plate (recycle region) begins to reduce $n_t$, $\nabla p_{\parallel} \neq 0$ and plasma pressure falls across recycle region. Once $T_t \sim 1-2$ eV (and if $n_t$ high enough), volume recombination locally “extinguishes” plasma, reducing target power flux.

Detachment seen experimentally in many devices, but complex “volumetric” process and relative importance of ion-momentum friction vs. recombination still unclear. X-point geometry → long connection lengths → high residence times in low $T_e$ plasma → efficient radiative loss favouring power reductions where $q_{\parallel}$ is highest (i.e. on flux surfaces near separatrix).
Full detachment is a problem

- Detachment which is too "strong" (particle flux reduced across the whole target) is often associated with zones of high radiation in the X-point region and confined plasma (MARFE)
- MARFE formation can drive a transition from H to L-mode (H-mode density limit) or disruption
- MARFE physics still not well modelled

Limit detachment to regions of highest power flux (where it is needed most). Maintain remainder of SOL in high recycling (attached)
A few ways to arrange that this happens more readily:

- Divertor closure
- Target orientation
- Impurity seeding
Divertor closure

- Increased closure significantly improves divertor neutral pressure $\rightarrow$ increased neutral density ($n_n$), promoting earlier detachment
- Closing “bypass” leaks important for increasing $n_n$
- Divertor closure also promotes helium compression and exhaust – very important for ITER and reactors

*JET, R. D. Monk, et al. [13]*
Target orientation

- Parallel heat fluxes significantly reduced for vertical cf. horizontal targets
- Underlying effect is preferential reflection of recycled deuterium neutrals towards the separatrix

AUG, A. Kallenbach, et al. [14]

- Cooler, less dense plasma
- Hotter plasma near separatrix
- Increased ionisation near sep.
- Higher $n_t$, lower $T_t$
- Higher CX losses
- Pressure loss, $q_{||}$ ↓
Impurity seeding

DIII-D, C. J. Lasnier, et al. [15]

- D₂ puff: 92 torr/s⁻¹ for 1.8 s
  - Heat flux: 7 MW/m²
  - Major radius: 1.1 to 1.7 m

JET, G. F. Matthews et al. [16]

- Ne puff: 12 torr/s⁻¹ for 0.1 s
  - Heat flux: 3 MW/m²
  - Major radius: 1.1 to 1.7 m

Strong impurity seeding reduces ELM size but price is paid in confinement
ITER divertor achieves partial detachment

Deep V-shaped divertor, vertical, inclined targets
Dome separating inner and outer targets – also helpful for diagnostics, neutron shielding and reducing neutral reflux to the core

Kirschner et al. [17]

ITER Divertor DDD 17, Case 489 (SOLPS5 runs by A. Kukushkin)
Divertor exhaust

Apart from power handling, primary function of divertor is to deal with He from fusion reactions → compress D, T, and He exhaust as much as possible for efficient pumping (and therefore also good density control).

Critical criterion for an ITER burning plasma is that He is removed fast enough such that: $\tau_{p,He}^* / \tau_E \leq 5 - 10$ is satisfied. $\tau_{p,He}^*$ is the global helium particle residence time – a function of $\tau_p$, the He neutral density in the divertor and the pumping speed (conductance) [18].

Helium enrichment: $\eta_{He} = \frac{2n_{He}^{pump}}{C_{pump}} = \frac{C_{plasma}}{n_{e}}$ is the ratio of He concentration in the divertor compared to the main plasma.

e.g. ITER: He prod. rate $\sim 2 \times 10^{20} \text{s}^{-1}$
Max. divertor pumping speed
$\sim 200 \text{ Pa m}^3\text{s}^{-1} \sim 1 \times 10^{23} \text{ He atom s}^{-1}$
$\rightarrow C_{pump} \sim 2 \times 10^{-3} = 0.2\%$

Typical acceptable He conc. in the core: $\sim 4\%$ $\rightarrow \eta_{He} = 0.2/4 = 0.05$ is minimum required. The values of $\eta_{He}$ and $\tau_{p,He}^*$ required for ITER have been achieved experimentally.
Plasma-surface interaction
ITER materials choices

- **Be for the first wall**
  - Low T-retention
  - Low Z
  - Good oxygen getter

- **C for the targets**
  - Low Z
  - Does not melt
  - Excellent radiator

- **W for the dome/baffles**
  - High physical sputtering threshold

Driven by the need for operational flexibility

- **Possible alternative:**
  - Be wall, all-W divertor

To avoid problem of T-retention

What are the issues associated with plasma-surface interactions?
Critical issues

Long term tritium retention

Material lifetime

- Short and long range material migration
- Material mixing
- Steady state erosion
- Transient erosion (ELMs, disruptions)
- Redeposition

All strongly interlinked
Impurity migration

Migration = Erosion $\rightarrow$ Transport $\rightarrow$ Deposition $\rightarrow$ Re-erosion
Erosion: Physical and chemical sputtering

**Physical**

- Energy threshold → higher for higher Z substrate
- Much higher yields for high Z projectiles – important if using impurity seed gases

**Chemical (carbon)**

- No threshold
- Dependent on bombarding energy, flux and surface temperature

Current steady state divertor target erosion rates (ERO modelling) due to \( Y_{\text{phys}} \) and \( Y_{\text{chem}} \) estimated at ~0.4 - 2 nms\(^{-1}\) for ITER [17]
Erosion: transients, e.g. ELMs on the divertor

Important factor is max. $\Delta T_{surf}$ due to arrival of short heat pulse (duration, $t$):

$$T_{surf}^{max} \approx \frac{2fW_{th}}{A_{div}} \left( \frac{t}{\pi \rho Ck} \right)^{1/2}$$

$T_{surf}^{max}$ Important to measure $T_{surf}^{max}$

$f =$ fraction of $W_{th}$ lost during transient $A_{div}$, divertor wetted area ($\sim$6m$^2$)
$p, k, C =$ density, conductivity, heat capacity
ELM energy losses must stay below melting/sublimation/evaporation limit to avoid fast erosion (e.g. melt later loss)

This is the very lower limit for Type I ELMs observed today $\rightarrow$ need to mitigate ELMs or find small ELM regimes and provide best possible monitoring of target erosion $\rightarrow$ see talk by E. Gauthier (Thurs. morning)

$\Delta_{transient} \sim$ few $\mu$m and target thickness $\sim$ cm $\rightarrow$ lifetime $\sim$10$^4$ events $\sim$10$^3$ Type I ELMs/discharge $\rightarrow$ lifetime $\sim$ 10 ITER pulses!!

Tests on ITER target mock-ups with realistic energy fluxes show that damage threshold $\sim$2x lower than for ideal materials (crack formation) [23,24]
$\rightarrow$ ELM energy flux $\leq$ 0.7 MJm$^{-2}$ for W and CFC ($\sim$1.5% of $W_{th}$ @ $Q_{DT} = 10$)

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**Ions:**

Cross-field transport – turbulent driven ion fluxes can extend into far SOL
→ recycled neutrals
→ direct impurity release
ELMs can also reach first walls

Eroded Impurity ions “leak” out of the divertor (\(\nabla T_i\) forces)

SOL and divertor ion fluid flows can entrain impurities

**Neutrals:**

- From divertor plasma leakage, gas puffs, bypass leaks → low energy CX fluxes → wall sputtering
- Lower fluxes of energetic \(D^0\) from deeper in the core plasma
- A problem for first mirrors → see talk by Vladimir Voitsenya (Thurs. morning)

**Image Diagram:**

- D\(^0\) from wall ion flux or gas puff
- CX event
- Ionisation
- Bypass leaks
- D\(^0\) from wall ion flux or gas puff
- Edge2D/Nimbus
- Courtesy G.F. Matthews
Migration balance – example from JET

- Make balance for period 1999-2001 with MarkIIIGB divertor: 14 hours plasma in diverted phase (50400 s, 5748 shots)
- Use spectroscopy and modelling to estimate main chamber sources

- Post mortem surface analysis
  - Deposition almost all at inner divertor
  - Surface layers are Be rich → C chemically eroded and migrates, Be stays put
  - Outer divertor – region of net erosion or balanced erosion/redeposition – BUT mostly attached conditions (not like ITER)

~250 kg/year if JET operated full time! Carbon migrates to remote locations forming D-rich soft layers (high T-retention)

Likonen et al. [26]
Coad et al. [27]
Strachan et al. [25]
Tritium retention (1)

- One of the most challenging operational issues for burning plasmas
- If carbon present, complex interplay between erosion $\rightarrow$ hydrocarbons $\rightarrow$ dissociation/ionisation $\rightarrow$ transport $\rightarrow$ re-deposition $\rightarrow$ migration to remote areas with high sticking coefficients and retention in co-deposits
  - Carbon traps D, T very efficiently
  - D/C ratio can be in the range $\sim 0.4 \rightarrow > 1$ depending on the type of re-deposited layer
- Retention very hard to characterise in today’s mostly carbon dominated devices
- Dependent on materials, $T_{\text{surf}}$, geometry (limiter/divertor), operating scenarios (H-mode, L-mode, low/high dens.)

Reported measurements range from 3-50% retention [28]! e.g. on JET, $\sim 3\%$ obtained from long term, post mortem surface analysis, $\sim 10$-20% from gas balance
Tritium retention (2)

- A 400 s $Q_{DT} = 10$ ITER discharge will require $\sim50$ g of T fuelling (cf. 0.01-0.2 g in today’s tokamaks)
- Working guideline for max. in-vess. mobilisable T in ITER $\sim1$ kg [29,30]
- World supply of T is also limited
  - Must avoid build-up in inaccessible locations
- Predicting the expected retention in ITER is notoriously difficult

ITER target [29] is a retention level of $\sim0.05$ g/discharge $\rightarrow \sim7000$ shots before major shutdown for T-removal

Accurate measurement of T-retention and the development of efficient T-removal methods will be critical for the success of ITER

- Very recent estimates (ERO code 2007 including Be main chamber influx) show that the in-vessel limit could be reached after only $\sim140$ shots [17]
  - Modelling does not yet contain effects of transients (ELMs disruptions)!
  - No account taken for trapping in tile gaps
Dust

- Dust is seen in all tokamaks, especially with C walls
- Not generally a concern in today’s devices ...
- But is potentially very important in ITER — see talk by Sandrine Rosanvallon (Wednesday morning)
  - As an inventory for trapped tritium in areas difficult to access
  - As an explosive safety hazard – water leak → hot surfaces → steam → hydrogen (by oxidation) → possible explosion if enough air also present [31]
  - As a radiological or toxic hazard (activation products of W, tritium contained in Be, C dust, toxicity of Be dust)

Carbon dust collected from tokamaks after operation periods is usually micron sized. Formed from flaking and degradation of deposited films, unipolar arcing, brittle fracture (e.g. due to transients) etc.

No real idea yet how much dust ITER will generate, where it will go or how to get it out → a big effort needed to improve this situation

TCV: floor viewing IR camera during disruption, #33448

No fusion device operating today contains the material mix currently planned for the ITER first wall and divertor: Be, W, C. Cross contamination of the material surfaces will be unavoidable. This is likely to have several consequences [29]:

- **Material property changes due to mixing**
  - Formation of metallic carbides → diffusion of C into bulk material at high temperatures
  - Formation of Be-W alloys → melting point can be reduced by as much as ~2000°C

- **Effect on H-isotope retention**
  - Retention of H in BeO can be as high as in C
  - Retention in W can be increased by C or oxide layers but is very low in pure W or Be
  - Very complex – difficult to predict yet for ITER

- **Effect on material erosion**
  - Can both increase and decrease erosion!
  - Heavy ions (e.g. Be\(^{Z+}\), C\(^{Z+}\)) on C, W → increased phys. sputt. but surface coverage (e.g. Be on C) reduces chemical sputtering.

Preparations underway at JET to test a Be/W & Be/W/C wall mix from ~2010 [33]
So, what needs to be diagnosed [34]?

- Target plate heat and particle fluxes, $T_e$
- Surface temperature
- Erosion rate
- Neutral gas pressure
- Cryopump inlet composition ($H/D/T/He, C_xH_y$)
- Dust accumulation?
- Wall temperature and visible image
- Main chamber gas pressure and gas composition
- SOL neutral density ($D/T$)
- Impurity influxes
- SOL $n_e$, $T_e$ profiles (but challenging)

See next talk by Philip Andrew!
A 45 min. talk can only hope to scratch the surface of such a vast field. Some good reference sources covering aspects of the material in this talk are the following:


A number of additional papers have been used to prepare the slides in this presentation. They are listed below in order of appearance in the talk.

References (2)