Edge physics and plasma-wall interactions

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Importance of plasma-surface interaction

“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”


CAVEAT: Edge plasma physics and PSI is a vast domain. Can only scratch the surface in a single tutorial. Work referenced throughout the talk is listed at the end.

These slides in this tutorial are rather dense and are to be used more as reference material. More important to listen to the speaker!
Outline

• **Part I:** The scrape-off layer (SOL) and divertor
  – SOL particle and power widths
  – Divertor detachment
  – Turbulent transport and SOL flows

• **Part II:** Plasma-surface interactions
  – Material lifetime – erosion and migration
  – Transients (ELMs and disruptions)
  – Tritium retention
  – Dust
Part I

Divertor and SOL physics
Terminology: limiters and divertors

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

Core plasma

LCFS

C_L

Limiter

Vessel walls

Private plasma

“Upstream” Outer midplane

X-point

Core plasma

Separatrix

Inner ↑ Outer

Divertor targets
Part of the ITER ramp-up and ramp-down will be in limited phase – but quite short → ~10 s. Full burn divertor phase of ~400 s for the $Q_{DT} = 10$ inductive scenario.
Basics – SOL density width, $\lambda_n$ [1]

- Any solid surface inserted into a plasma constitutes a very strong particle sink
- In the high tokamak B-field: $\Gamma_\perp << \Gamma_\parallel$
- Thin Debye sheath ($\lambda_D$ few 10’s μm thick) forms at the surface → controls flow of particles and energy $\parallel B$

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

$\Gamma_\perp \equiv n v_\perp = -D_\perp dn/dr \sim D_\perp n/\lambda_n \Rightarrow v_\perp \approx D_\perp/\lambda_n$, $\lambda_n = \tau_\perp v_\perp \Rightarrow \tau_\perp = \frac{\lambda_n^2}{D_\perp}$

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

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

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

cf. ITER minor radius = 2.0 m

Could be even worse for energy – see next ……
SOL power width, $\lambda_q$

- SOL width for power, $\lambda_q$, is also small and is an important parameter of the edge plasma.
- As for particle transport, the physics determining $\lambda_q$ is an extremely active area of research and the experts still do not agree ....
- Scalings for $\lambda_q$ can be derived from models and experiments, e.g.:
  - “2-point” analytic modelling [1]: $\lambda_q \propto P_{SOL}^{-5/9}$ $P_{SOL}$ = power into SOL
  - Scaling from H-mode experiments on JET [7,8]:
    \[
    \lambda_q \propto P_{SOL}^{-0.5} B_{\phi}^{-1.0} n_u^{0.25} q_{95} R^2 \\
    \lambda_q \propto P_{SOL}^{-0.5} B_{\phi}^{-1.0} n_u^{0.25} q_{95}^{0.5} R^{1.5}
    \]
    if parallel conduction dominates
    if parallel convection dominates
  - ITER modelling [9] yields $\lambda_q = 5$ mm, JET scaling gives $\lambda_q = 3.5 - 4.5$ mm mm (cf. $a = 2.0$ m for ITER)
  - Fairly recent multi-machine scaling [10] gives $\lambda_q/R \sim$ constant
  - VERY recent high resolution measurements on several tokamaks are finding $\lambda_q \propto 1/l_p$ (like JET $\Rightarrow \lambda_q \propto q_{95}/B_{\phi}$)
\( \lambda_q \) still the subject of much attention

- Very new high precision near-SOL H-mode (inter-ELM) \( \lambda_q \) measurements reported from NSTX, DIII-D, C-Mod and JET
  - All machines find \( \lambda_q \propto I_p^\alpha \) in H-mode with \( \alpha \sim -1.0 \to -1.6 \) and little dependence on \( P_{SOL}, B_\phi, n_e, \ldots \).
  - Potentially serious implications for ITER target power handling (\( I_p = 15 \text{ MA!} \))
  - What sets the value of \( \lambda_q \)? Parallel SOL transport? H-mode pedestal stability?

C-Mod, B. LaBombard et al. [11]

DIII-D, C. J. Lasnier et al. [12]

NSTX, T. K. Gray et al. [13]
Example power handling – ITER case

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 → divertor detachment and high radiative loss

$q_{\perp} \sim 40 \text{ MWm}^{-2}$ per target if no radiative (or other) dissipation → cannot be tolerated for more than ~2-3 s on actively cooled surfaces
The route to detachment (1)

Low n, high T (high $P_{\text{SOL}}$)

"Sheath limited"

\[ T_t \propto T_u \]

\[ n_t \propto n_u / 2 \]

Target

Moderate n, T

"High recycling"

\[ T_t \propto 1 / n_u^2 \]

Target

Region of strong radiation losses

Mean free paths for particle collisions are long:

\[ \lambda_{\text{coll}} \propto T_u^2 / n_u, T_u \sim T_e \sim T_i, \lambda_{ee} \sim \lambda_{ei} \sim \lambda_{ii} \]

SOL collisionality:

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

Power flow to surface largely controlled by target sheath:

\[ q_{\parallel,t} = m_t c_{st} T_t + n_t c_{st} \varepsilon_{\text{pot}} \]

\[ \gamma = \text{sheath heat transmission coefficient} \]

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

\[ \nu^* \text{ rises as } n_u \text{ rises, finite electron heat conductivity:} \]

\( (\text{note: } \kappa_{0,e} \gg \kappa_{0,i}) \)

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

\[ (\Gamma_t \propto n_u^2) \]

\[ \lambda_{\text{ion}} \propto 1/n_t \text{ decreases so that target recycling increases strongly } \rightarrow \text{ flux amplification} \]

As $T_t \downarrow$, radiation loss increases $\rightarrow T_t \downarrow$ further

Adapted from [2]
The route to detachment (2)

At sufficiently low $T_t$, (< 5 eV), neutral ionisation rate < ion-neutral friction processes (charge exchange, elastic scattering). Momentum transferred from ions to dense cloud of neutrals in front of the plate (recycle region) $\rightarrow$ begins to reduce $n_t$, $\nabla p_{||} \neq 0$ 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 – modelling still has problems to reproduce. X-point geometry $\rightarrow$ long connection lengths $\rightarrow$ high residence times in low $T_e$ plasma $\rightarrow$ efficient radiative loss favouring power reductions where $q_{||}$ 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 → 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. [16]
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. [17]

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

7.5 MW NI; ELM averaged
normalized poloidal flux

Separatrix
Impurity seeding

DIII-D, C. J. Lasnier, et al. [18]

**D$_2$ puff**
92 torr/s$^{-1}$ for 1.8 s

- 2 s
- 3 s

**Ne puff**
12 torr/s$^{-1}$ for 0.1 s

- 2 s
- 3 s

Strong impurity seeding also reduces ELM size but a price may have to be paid in confinement.

JET, G. F. Matthews et al. [19]

- Unfuelled
- Strong D$_2$ puff
- Strong D$_2$+N$_2$ puff

Heat flux (MW/m$^2$)

Electron pressure (10$^9$ eV/m$^2$)

Distance mapped to midplane (cm)
ITER divertor achieves partial detachment

Deep V-shaped divertor, vertical, inclined targets
Dome separating inner and outer targets – reduces neutral reflux to the core.
Also helpful for diagnostics, neutron shielding
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:  \[ \frac{\tau_{p,He}^*}{\tau_E} \leq 5 - 10 \]  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) [21].

Helium enrichment:  \[ \eta_{He} = \frac{n_{He}^{\text{pump}} / 2n_{D2}^{\text{pump}}}{n_{He}^{\text{plasma}} / n_e} = \frac{C_{\text{pump}}}{C_{\text{plasma}}} \]  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 \( \sim 200 \text{ Pa m}^3\text{s}^{-1} \sim 1 \times 10^{23} \text{ He atom s}^{-1} \)  
\[ \Rightarrow C_{\text{pump}} \sim 2 \times 10^{-3} = 0.2\% \]  
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.

To cryopumps
Perpendicular SOL transport

For many years in early tokamak research, measured density profiles in the SOL plasma often seem to obey an exponential fall off, implying that a Fick’s Law type diffusive ansatz is an appropriate description, e.g.:\[ \Gamma_\perp = -D_\perp \frac{dn}{dr} \sim D_\perp n/\lambda_n \] (see slide 7)

But in fact, the SOL density profile, when looked at more closely, often has more structure, itself dependent on discharge density/ SOL collisionality.

Example from the TCV tokamak

Note how broadening occurs mostly in the “far SOL”

O. E. Garcia, R. A. Pitts et al. [22]
What causes the broad $n_e$ profiles?

The particle transport is intermittent, mostly convective, not diffusive. Particle flux time series are bursty. Most of the transport occurs during the bursts.

- Relative amplitude of the bursts very high: $n_{rms}/<n> \to 1$ at high density and in the far SOL at all densities
- These bursts take the form of magnetic field aligned “filaments” as they propagate through the SOL

At high densities, bursts more frequent
Probes provide quantitative data on the real particle flux, but only at fixed toroidal and poloidal locations. Fast visible imaging allows the 3D picture to be seen but analysis more challenging.

These L-mode filaments are nothing more than small amplitude versions those seen during Edge Localised Mode (ELM) events.
Origin of the bursts?

- Thought now to be due to electrostatic interchange turbulence produced in the near SOL region
- Local relaxations in the edge pressure profile → ejection of bursts of excess particles and heat into SOL → radial motion due to electric drift ($B \times \nabla B$ charge separation → $E \times B$ drift), damping by parallel losses on open field lines
- All the basic physics captured by recent 2D interchange turbulence simulations

- ESEL code (Risø [23]) simulates 2D region centred on outboard midplane
- Exhaustively tested against TCV high density case [22, 24-25]
Bursty transport now in ITER baseline!

Stationary, inter-ELM power fluxes to ITER main wall now assumed to be dominated by convective, intermittent transport

Multi-machine study shows far SOL cross-field convective velocity weakly dependent on device size

$$\lambda_{\text{SOL}} = \frac{L c_s}{v_{\text{SOL}}} = 4 - 17 \text{ cm}, \; L \sim 60 \text{ m for ITER}$$

Power to ITER first wall <20 MW (20% $P_{\text{SOL}}$)
Part. flux to ITER first wall < $1 \times 10^{24} \text{s}^{-1}$ (10% $\Gamma_{\text{div}}$)

Region of connected SOL from inner to outer strike points
Parallel SOL ion flows

Determine transport of impurities from source to destination in a tokamak – material migration – T-retention (Slides 41-46)

**Poloidal**

- $E_z \times B$, $\nabla \rho \times B$
- $E_\theta \times B$

**Parallel**

- $B \times \nabla B$

Field direction dependent

FWD-$B_\phi$

$B \times \nabla B$

$B \times \nabla B$

REV-$B_\phi$
Flows can be very strong

Have been measured on several tokamaks – TCV is a good example [27,28]

Main parallel flows are field direction dependent, density dependent and in the same direction as the plasma current

→ Consistent with Pfirsch-Schlüter (neoclassical) flow

$M || = 0.5 \rightarrow v || \sim 30 \text{ km s}^{-1}$!
Plasma-surface interactions
The plasma-wall interaction challenge

Perhaps the most critical area for ITER and for the long term feasibility of fusion energy production

Must guarantee:

- Management of stationary heat flux densities at the limit of cooling technology
- High throughput fuel cycle (low burn-up fractions) with low Tritium retention
- Material erosion rates compatible with adequate lifetime and burning plasma purity
Upscale to ITER is a very big step

Comparison with JET (World’s largest operating tokamak) for illustration

<table>
<thead>
<tr>
<th>Parameter</th>
<th>JET MkII/GB (1999-2001)</th>
<th>ITER</th>
</tr>
</thead>
<tbody>
<tr>
<td>Integral time in diverted phase</td>
<td>14 hours</td>
<td>0.1 hours</td>
</tr>
<tr>
<td>Number of pulses</td>
<td>5748</td>
<td>1</td>
</tr>
<tr>
<td>Energy Input</td>
<td>220 GJ</td>
<td>60 GJ</td>
</tr>
<tr>
<td>Average power</td>
<td>4.5 MW</td>
<td>150 MW</td>
</tr>
<tr>
<td>Divertor ion fluence</td>
<td>1.8x10^{27}</td>
<td>*6x10^{27}</td>
</tr>
</tbody>
</table>

1 ITER pulse ~ 0.5 JET years energy input
1 ITER pulse ~ 6 JET years divertor fluence

*Code calculation

- Stored energy $\propto R^5 \rightarrow \sim 35\times$ higher on ITER than JET [30]
- But deposition area for power in the divertor $\propto R \lambda_p \rightarrow \lambda_{p,ITER} \sim \lambda_{p,JET} \rightarrow \sim 2.5 \text{ m}^2$ ITER cf. $\sim 1.0 \text{ m}^2$ JET $\rightarrow$ ITER will project $\sim 35x$ the energy into only $\sim 3x$ the area
ITER materials choices

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

- For low-active phase: C for the targets
  - Low Z
  - Does not melt
  - Excellent radiator

- W for the dome/baffles
  - High Y_{phys} threshold

Driven by the need for operational flexibility

- For D and DT phases:
  - Be wall, all-W divertor

To avoid problem of T-retention

Surface areas:
Be: ~700 m², W: ~120 m²
CFC: ~35m²
Critical PWI issues

Long term tritium retention

Material lifetime

Dust production

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

All strongly interlinked

T-retention and dust production are safety critical items and form part of the ITER Nuclear Licensing process
Impurity migration

Migration = Erosion $\rightarrow$ Transport $\rightarrow$ Deposition $\rightarrow$ Re-erosion
Impurity migration

Migration = Erosion → Transport → Deposition → Re-erosion
Steady state erosion: sputtering

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

- No threshold $\rightarrow$ dependent on bombarding energy, flux and surface temperature

Steady state divertor target erosion rates (ERO modelling) due to $Y_{\text{phys}}$ and $Y_{\text{chem}}$ estimated at $\sim 0.4 - 2$ nms$^{-1}$ for ITER [32]
Transient erosion

Transients are the biggest threat for large scale erosion in ITER. Burning plasma stored energy (~350 MJ) >> in the largest operating devices but surface areas for energy deposition only factor ~3x larger (slide 29).

ELMs

“Natural” ELMs expected to expel ~6% of $W_{\text{plasma}}$ at 1-2 Hz [18] $\rightarrow$ peak energy densities on ITER divertor of 5-10 MJm$^{-2}$ on timescales of 250-500 $\mu$s.

Disruptions/VDEs

Worst case full energy disruptions $\rightarrow$ peak energy densities on the divertor of 5.0-20 MJm$^{-2}$ on timescales of 1.5-3 ms (thermal quench).

Current quench: runaway electron currents up to 12 MA with 10-20 MeV in localised areas on timescales of a few tens or hundreds of ms $\rightarrow$ energy densities of 35-70 MJm$^{-2}$.
Runaway electrons on Tore Supra

Deliberate creation of runaway electron plateau by injection of Neon impurity during $I_p$ ramp up.

Thermal quench occurs between $-9.552 < t < 0.948$ ms in the video.

Subsequent attempts to control the runaway beam using tokamak PF control circuits → one possible solution being sought for ITER.

Courtesy J. Bucalossi, CEA Cadarache
Disruption induced erosion

Vapour shielding reduces CFC erosion
Loss of melt layer on W occurs if layer deep enough and force (evaporated layer plasma pressure, eddy currents) sufficient to trigger liquid instabilities (Kelvin-Helmholtz, Rayleigh-Taylor) → droplet ejection

Assuming 10% melt-layer loss, W divertor lifetime (0.3 cm PFC end of life thickness) exceeded in ~300 disruptions → efficient disruption avoidance or mitigation techniques required in ITER → major research priority for the ITER team
e.g. W exposed to 100 pulses of 1.5 MJm$^{-2}$

**Significant melting, bridging of castellation gaps, lifetime reduction**

**Issues of operability on damaged targets, dust production**
Real material limits are much lower
\(\rightarrow\) Results from Russian plasma simulators [36]:
Erosion limit for CFC reached due to PAN fibre erosion (fibres parallel to surface)
Erosion limit for W reached due to melting of tile edges
Crack formation on W observed at energy densities ≥ 0.7 MJm\(^{-2}\)

Recommended damage threshold
\(\sim 0.5\) MJm\(^{-2}\) now adopted by ITER
\(\rightarrow\) Will require ELM mitigation strategies to keep \(E_{\text{ELM}} < 1\) MJ
Could even be worse for W ....

Crack formation and melting under ITER-like pulsed and steady loads on water cooled, W monoblock targets
Pulsed loads ≤0.5 MJm⁻² → controlled ITER ELMs

Previous observations of surface crack saturation might be optimistic
Repetitive edge melting and crack propagation a worry for power handling and dust formation
Monoblock design issues → insufficient confidence yet to build all-W divertor for ITER
Impurity migration

Migration = Erosion → Transport → Deposition → Re-erosion
Transport creates and moves impurities

Ions:
- Cross-field transport – turbulent driven far SOL ion fluxes (slides 19-23)
  → 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 (slides 24-25)

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
Impurity migration

Migration = Erosion → Transport → Deposition

Re-erosion
Migration balance – example from JET

- Make balance for period 1999-2001 with MarkIIGB 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 doesn’t move
  - 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! C migrates to remote areas forming D-rich soft layers (high T-retention)
Tritium retention (1)

- A 400 s $Q_{DT} = 10$ ITER discharge will require $\sim 50$ g of T fuelling (cf. 0.01-0.2 g in today’s tokamaks)
- Maximum in-vessel mobilisable T in ITER limited to 1 kg [41]
  - This is a safety issue
- In practice, administrative limit of $\sim 700$ g
  - 120 g in cryopumps
  - 180 g uncertainty
- Predicting the expected retention in ITER is fraught with uncertainty but progress is being made
Tritium retention (2)

- For C, complex interplay between erosion → hydrocarbons → dissociation/ionisation → transport → re-deposition → 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 ~0.4 → > 1
- For Be, co-deposition of T also possible - large potential source of Be from first wall
- For W, most of retention will be from implantation → not thought to constitute a large reservoir
- BUT effects of increased trapping due to neutron irradiation of metals – does not look like an issue from recent results

Co-deposition with C and Be depends sensitively on deposition rate, incoming particle energy, surface temperature [42]

J. Roth et al., [41]
Tritium retention (3)

- EU-PWI Task Force and ITPA DIVSOL group have recently tried to estimate ITER T-retention
- Assume erosion determines co-deposition:
  - $T$-retention = erosion rate x total co-deposition concentration
- Add T implantation in W
- Compare materials options

Main driver of current ITER baseline strategy to begin D-T operations with full W divertor – only ~few 100 full performance DT shots predicted before T-inventory limit exceeded if CFC divertor used in tritium phase
Tritium retention (4)

- Even though Be can readily trap tritium, fuel is released from Be co-deposits at much lower temperature than for C
  - The main ITER fuel recovery strategy → divertor bakeout capability to 350°C is part of the Baseline design → most of the Be co-deposits expected in the divertor
  - NB: main wall bakeable only to 240°C

K. Sugiyama et al., [43]
Dust – why worry?

• Expectation is that increase in duty cycle and erosion in ITER will lead to large scale-up in quantity of dust particles produced

• Like T-retention, dust is a safety issue [44,45]
  - dust particles radioactive (tritium + activated metals)
  - potentially toxic (Be)
  - potentially responsible for a large fraction of in-VV mobilisable tritium
  - chemically reactive with steam or air

• Radiological or toxic hazard depends on how well dust is contained in accident scenarios and whether it is small enough to remain airborne and be respirable
  - size needs to be $<\sim 100 \, \mu m$
  - depends on how dust is produced, e.g. crumbling of co-deposited layers or destruction (thermal overload) of tritiated layers during off-normal events
  - tritiated dust can levitate in electric fields as a result of self-charging due to emission of beta electrons
Dust – seen in all tokamaks

- Dust is seen in all tokamaks, especially with C walls, but most often in first plasmas after long vent, or after disruptions when plasma touches surfaces not normally in contact with high heat/particle flux – not usually an operational issue.
- First papers to recognize the potential importance more than 10 years ago [46]

**TCV**: floor viewing IR camera during disruption, #33448 (J. Marki, R. A. Pitts)

**DIII-D**: floor viewing DiMES TV with near IR filter. 2nd shot in 2007 after “dirty vent”, #127331. Courtesy of D. L. Rudakov [47]
Significant quantities collected in JET

- Very recent dust collection from JET after ~6 years → dominated by C but Be rich due to Be wall evaporation
  - NB: 1 ITER pulse ~ 6 years JET operation in terms of divertor fluence (based on 1999-2001 campaigns)
  - We need to better understand dust formation, transport, fuel retention and decide how to measure it

- In a Be/W environment, is dust formation all due to disintegration of deposited layers?
- When does a layer of given thickness detach? What drives this process?
ITER Dust – safety inventory limits

- Global quantity in the vacuum vessel (VV) – 1 tonne during D-D and D-T operation
- On hot surfaces (corresponds to amount of dust that could produce up to 2.5 kg of H\(_2\) during accident in case of full reaction with steam – requires air ingress):
  - Be + H\(_2\)O \(\rightarrow\) BeO + H\(_2\), C + H\(_2\)O \(\rightarrow\) CO + H\(_2\), W + 3H\(_2\)O \(\rightarrow\) WO\(_3\) + 3H\(_2\)
  - Complete reaction: \(T_{\text{surf}} > 400^\circ\text{C}\)
  - 6 kg of C, 6 kg of W and 6 kg of Be (for CFC/W/Be mix)
  - up to 11 kg of Be (for Be alone) and up to 77 kg W (for W alone)
- When this dust inventory limit is reached (or if T-inventory reaches 1 kg) ITER operation must be stopped and dust removed
- The real dust inventory will be reduced by measurement uncertainties (estimated to be about 30%)
References (1)

A 1.5 hour lecture can only hope to scratch the surface of such a vast field. Some good reference sources covering aspects of the material shown in this presentation 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)


References (3)


[34] “Assessment of the erosion of the ITER divertor targets during Type I ELMs”, G. Federici et al., Plasma Phys. Control. Fusion **45** (2003) 1523


References (4)

[47] “Observations of Dust in DIII-D Divertor and SOL”, D. L. Rudakov et al., 1st Workshop on “Dust in Fusion Plasmas”, 8-10 July 2007, Warsaw, Poland