Power exhaust and plasma-wall interaction in tokamaks

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Outline

- (Neutron and particle) environment for plasma-facing materials
  - Need for power exhaust (due to particle bombardment)
  - Material options
  - Impact of neutrons on materials
  - Material migration and tritium retention

- Physics models to describe scrape-off layer plasma and plasma-material interactions ⇒ required for extrapolation toward future fusion power plants
The plasma-surrounding (material) walls (vessel) provides a containment and vacuum conditions

- Fusion requires a base pressure of about \( 10^{-8} \) mbar ⇒ pumping system

- Power in \( \alpha \)-particles and auxiliary must be (eventually) extracted through walls
  - Power in neutrons converted to heat in blanket wall
  - Tritium breeding

- Helium removal via in-vessel pumps
The choice of materials in fusion reactors is driven by plasma/neutron-wall interactions

- **Primary issues are**: material lifetime, tritium inventory, and debris formation

  ⇒ Economical/practical aspect, but tritium and debris formation also an additional safety aspect

- **D-T fusion reaction**: \( D + T \rightarrow \alpha \ (3.5 \text{ MeV}) + n \ (14.1 \text{ MeV}) \)

  - \( \alpha \)-particle for plasma self-heating, neutron for blanket heating ⇒ 1 g D-T produces 67.6 GJ in \( \alpha \)-particles and 271.8 GJ in neutrons

  - Following thermalization, \( \alpha \)-particles become helium ash ⇒ need to be removed from system

⇒ In future fusion power plants, both power and particle exhaust and activation of surrounding wall are top issues
Direct contact of the plasma with the vessel wall must be limited to certain (controlled) areas

- Power in the plasma is predominately radiated $\Rightarrow$ (isotropically) spread over wall
- Remaining power in escaping particles
  - Ions following field lines $\Rightarrow$ limiter and divertor plates
  - Charge exchange neutrals $\Rightarrow$ main chamber and divertor plates
In tokamaks, plasma radiation is concentrated in the divertor region (desired situation!)

- Main radiation in Ly emission from hydrogen, and impurities
No uniform engineering boundary conditions since escaping particles have a wide range of fluxes and energies.

\[ \Gamma = O(10^{20} \text{ m}^{-2} \text{ s}^{-1}) \]

\[ \Gamma = O(10^{23} \text{ m}^{-2} \text{ s}^{-1}) \]

\[ T \leq 10 \text{ eV} \]

CX processes
Neutral Atoms
Ions
plasma facing wall elements

divertor plates or limiters
Steady-state heat removal may be computed using standard (finite element) techniques

For given material of density ($\rho$) and thickness ($d$), and with a thermal heat conductivity ($\lambda$) and heat capacity ($c$)

⇒ Steady state conditions:

$$\Delta T = T_{surf} - T_{water} = Q \times d / \lambda$$

- $Q \approx 10 \text{ MW/m}^2$, $\lambda \approx 2000 \text{ W/mK}$, $d \approx 2 \text{ cm}$ ⇒
  $$\Delta T \approx 1000 \text{ °C}$$
While carbon-based plasma facing components sublimate, metals melt above a certain temperature.

**FOR CARBON:**
Above a certain power load (threshold) emission of debris  → BRITTLE DESTRUCTION

**FOR METALS:**
- Splashing
- Formation of droplets
- Formation of dust
While carbon-based plasma facing components sublimate, metals melt

Alcator C-mod, MIT

TEXTOR, Germany
Transient event in the plasmas lead to sudden excursions in heat load and surface temperature

- ELMs: 2 MJ/m$^2$ in 0.5 ms
- Disruptions: > 2 MJ/m$^2$ in ms
The energy in transient events must be absorbed by the target material ⇒ inertial cooling

• **Surface temperature follows:**
  \[ T(t) \propto P \times \sqrt{t} \]

• **ITER ELMs \( \approx 15 \text{ MJ} \), deposition time \( \approx 0.1 – 0.5 \text{ ms} \), deposition area \( \approx 6 \text{ m}^2 \) ⇒ power density \( \approx 10 \text{ GW/m}^2 \)

  \[ \Rightarrow T_{\text{max}} \approx 6000 \, ^\circ\text{C}, \text{ penetration depth} \approx 0.15 \text{ mm} \]

• **Sublimation temperature for graphite \( \approx 2200 \, ^\circ\text{C} \)**
• **Melting and boiling temperature of W = 3410 \, ^\circ\text{C} / 5560 \, ^\circ\text{C}**

  \[ \Rightarrow \text{Graphite will sublimate rapidly, metals will melt} \]

  \[ \Rightarrow \text{No immediate material solution, need to mitigate plasma events!} \]
Edge localized modes can raise surface temperature transiently by more than 50%.

- JET test: elevated tile (lamella) for intentional W melting $\Rightarrow$ ITER decision protungsten divertor.

Coenen, Arnoux et al. JET August 2013.

![Graph showing temperature changes for different stacks.](image-url)
In tokamaks, low divertor temperatures, and thus heat fluxes, can be achieved in detachment

- **Standard picture**: power and particle flow from the confined plasma via the scrape-off layer onto the divertor targets $\Rightarrow$ transfer of kinetic energy to surface

- **Detachment**: recombine plasma to neutrals in front of targets $\Rightarrow$ power loss in radiation and recombination
Detachment can be achieved by operating at high density or by intentionally injecting impurities.

- Line radiation of nitrogen is strongest at temperatures characteristic of the scrape-off layer.
- Low-dose “seeding” leads to reduction of $T_e$ to 5 eV $\Rightarrow$ negligible surface heating.
Neutrons significantly change the thermo-mechanical properties of materials

<table>
<thead>
<tr>
<th>Affected global parameter</th>
<th>Microscopic change</th>
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<tbody>
<tr>
<td>Heat conductivity</td>
<td>Lattice defects</td>
</tr>
<tr>
<td>Swelling</td>
<td>Void formation, gas bubbles (e.g., n → Be)</td>
</tr>
<tr>
<td>Ductility (i.e., ability to stretch</td>
<td>Neutron and helium induced hardening and embrittlement</td>
</tr>
<tr>
<td>material into a wire)</td>
<td></td>
</tr>
<tr>
<td>Composition</td>
<td>Transmutation products</td>
</tr>
<tr>
<td>Trap sites for tritium (retention)</td>
<td>Blister formation</td>
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</table>

- **Investigations into neutron damage of materials requires dedicated facilities (e.g., IFMIF for fusion neutrons, heavy ions) ⇒ need for up to 100 dpa**
Irradiation of tungsten with heavy ions and He reduces the thermal conductivity by factors of 200

- Irradiation by heavy ion beam source (up to 18 MeV)
- Simultaneous He implantation

- Other effects include increase of Ductile Brittle Transition Temperature (DBTT), void swelling, increase of tritium retention
Material migration leads to long-term modification of plasma-facing components and alloy formation

**Plasma**

Fuel ions + atoms (charge exchange) + impurity ions bombard 1st wall

**Wall materials**

Erosion → Transport → Deposition

Re-erosion
Both globally and locally the various wall elements are in dynamic equilibrium.

- **Be erosion** ⇒ layer formation ⇒ T retention
- **Be transport** into remote areas ⇒ T retention in plasma-shadowed areas
- **W erosion**, prompt re-deposition
- **Be/W dust formation**
Both hydrogen and helium can be trapped deeply in tungsten leading to bubbles and blisters.

- Low solubility of H and He ($E_I^S$: 3.5 eV, 5.5 eV)
- Fast interstitial migration into grain boundaries ($E_I^M$: 0.35 eV, 0.24 eV)
- Deep trapping in vacancies (1.4 eV, 4.7 eV)
Formation of Be-W alloys on tungsten surfaces reduce the melting temperature from 3695 to ~ 1570 K
Implantation and co-deposition of tritium on plasma-facing surfaces administratively limits ITER operation

- Tritium is radioactive, most hazardous to the public in $T_2O \Rightarrow$ tritium management

- Metals are significantly less susceptible of absorbing tritium than carbon $\Rightarrow$ preferred (and decided!) for ITER

Roth et al. J. Nucl. Mat. 2009
Changing the JET wall from carbon to beryllium and tungsten reduced the hydrogen retention by 10x

- Hydrogen retention likely due to co-deposition with beryllium

⇒ Another 10x expected for going to full tungsten device
Thick deposition layer can also delaminate and thereby forming radioactive and chemically reactive dust
Physics models

(to predict power exhaust and plasma-material interaction in future reactors)
Physics models are needed to extrapolate and mitigate plasma-material issues

<table>
<thead>
<tr>
<th>Parameter/s</th>
<th>Issue/s</th>
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<tbody>
<tr>
<td>Plasma radiation, power flux and total energy to surface</td>
<td>Power exhaust</td>
</tr>
<tr>
<td>Particle flux and fluence</td>
<td>Erosion and impurity influxes ⇒ plasma impurity content</td>
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<tr>
<td>Plasma temperature</td>
<td>Power exhaust, sputtering yield, total erosion</td>
</tr>
<tr>
<td>Plasma density (impurity seeding)</td>
<td>Detachment/power exhaust (fuel dilution, density limit)</td>
</tr>
<tr>
<td>Helium</td>
<td>Fuel dilution</td>
</tr>
<tr>
<td>Dust</td>
<td>Fuel dilution, explosion hazard</td>
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Diverting the magnetic field lines to dedicated regions inside the vessel reduces the core impurity content.

- Isolation of divertor from main chamber by adding coils at the bottom of device to produce magnetic null.
In diverted configurations, the separatrix divides the core and the SOL, and defines a private plasma region.
Plasma electrons and ions can stick to material surface, and recycle as neutral (‘natural’ fuelling)

- Impinging ions recombine at surface
- Particles can remain at surface, diffuse into material, or are released back into plasmas as atoms (backscattering) or molecules (thermal release)
- Walls acts both as particle sink and source: strongest fuelling process in tokamaks!
Plasma-wall interaction leads to sputtering and macroscopic erosion of material

- Phys. and chem. sputtering processes due to hydrogen ions and neutrals, and impurities (self-sputtering!)

$\text{imp}$ sputtering}

$\text{phys}$ sputtering

$\text{chem}$ sputtering

$Y_{\text{phys}}$

$Y_{\text{chem}}$

Plasma

$H^+$

$Z^0$

$H^+$

$C^0$

$C_xH_y$

$H^0$

Solid negative
The total sputtering yield depends on the impact energy and substrate material.

Ion impact energy:

$$E_i = 3Z T_e + 2T_i$$
Sputtering due to ion and neutral impact on the material surface leads to release of impurities

- **Physical sputtering**: momentum transfer of incoming particle to lattice
  - Threshold energy: $Y_{\text{phys}} \rightarrow 0$ for $E_0 \rightarrow E_{\text{thresh}}$
  - Peak yield correlates with maximum ion/neutral-substrate momentum transfer
  - Yields are strong function of material $\Rightarrow$ future reactors favor high-Z materials
  - Self-sputtering of same-mass impurities can lead to $Y_{\text{phys,eff}} > 1$ $\Rightarrow$ run-away process
Trace amounts of impurities in the plasmas can significantly diminish benefits of high-Z materials.

- For neon and argon impinging on tungsten, already less than 0.5% is sufficient to drop $E_{\text{thresh}}$ from 35 eV to 5 eV.
Chemical sputtering also occurs on metals (more common feature for carbon)

- Yields are strong functions of substrate temperature, alloy composition, and magnitude of fluxes
Impurities are generated at both the main chamber walls and divertor plates

- Impurities can enter the main plasma as neutrals or ions
- Principal pathways include
  - Source (distribution)
  - Edge transport
  - Core transport
- Impurity migration

“If we understand the impurity source distribution, we can mitigate the impurity issue almost entirely!”
Plasma ions crossing the separatrix into SOL experience ‘attractive’ force of limiter and divertor plate

- Upon plasma initiation, negative charged sheath forms in front of limiter/plate (while SOL remains neutral!)
- SOL width is determined by completion between parallel-B and perpendicular-B transport ⇒ order of cms
The divertor target conditions are given by the upstream conditions for power and density:

\[ 2 n_t T_t = n_u T_u \]

\[ T_u^{7/2} = T_t^{7/2} + 7/2 \frac{q_{||} L}{\kappa_0 e} \]

\[ q_{||} = \gamma n_t k T_t c_{st} \]

- Conservation of particles, momentum, and energy ⇒ SOL 2-point model (1-D)
- Eqs. can be manipulated to obtain \( n_t, T_t, \) and \( T_u \) for given \( q_{||} \) and \( n_u \)
In detached conditions, momentum and power losses occur in the SOL in front of target plate

- Momentum losses due to \((\text{CX})\) friction of plasma with neutrals (recycling and volumetric)

- Surface heat load is dispersed by line radiation (line radiation, recombination)

\[
2 \, n_t \, T_t = f_{\text{mom}} \, n_u \, T_u
\]
\[
T_u^{7/2} = T_t^{7/2} + 7/2 \, f_{\text{cond}} \, q_{\|} \, L / \kappa_{0e}
\]
\[
q_{\|} = 1 / (1 - f_{\text{power}}) \, \gamma \, n_t \, kT_t c_{\text{st}}
\]
The most attractive regime for fusion reactors is the detached regime at high upstream density.

- High upstream densities required for high core plasma density (‘natural by-product’)
- Plasma temperature in front of plate 1 eV, or below $\Rightarrow$ low sputtering
- Plasma ionization moves off plate
Summary

• A container (vessel) is required to provide the vacuum conditions for fusion

• Materials are exposed to extreme neutron and particle fluxes ⇒ currently, limited solution to materials issue

• Carbon and metals (beryllium, molybdenum, tungsten) have been tested in tokamaks and linear devices

⇒ Deterioration of thermo-mechanical properties under neutron irradiation and tritium retention swayed ITER to opt for metals (Be and W) only

⇒ Plasma physics (e.g., achieving low plasma temperatures at material surfaces and mitigation of transient events) needs to solve materials issue
Because of tritium retention issue with carbon, and the good experience with the JET-ILW, ITER opted for a full-W divertor from day-one.

Oct 22, 2013
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~10 years