Boundary Conditions for a Solid State Divertor in a Fusion Power Plant

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Outstanding Technical Challenges with Gaps beyond ITER

For any further fusion step, safety, T-breeding, power exhaust, RH, component lifetime and plant availability, are important design drivers and CANNOT be compromised.

**Tritium breeding blanket**
- most novel part of DEMO
- TBR >1 marginally achievable but with thin PFCs/few penetrations
- Feasibility concerns/ performance uncertainties with all concepts -> R&D
- Selection now is premature
- ITER TBM is important

**Power Exhaust**
- Peak heat fluxes near technological limits (>10 MW/m²)
- ITER solution may be marginal for DEMO
- Advanced divertor solutions may be needed but integration is very challenging

**Remote Maintenance**
- Strong impact on IVC design
- Significant differences with ITER RM approach for blanket
- RH schemes affects plant design and layout
- Large size Hot Cell required
- Service Joining Technology R&D is urgently needed.

**Structural and HHF Materials**
- Progressive blanket operation strategy (1st blanket 20 dpa; 2nd blanket 50 dpa)
- Embrittlement of RAFM steels and Cu-alloys at low temp. and loss of mechanical strength at ~ high temp.
- Need of structural design criteria and design codes
- Technical down selection and development of an Early Neutron Source (IFMIF-DONES)
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- Need of structural design criteria and design codes
- Technical down selection and development of an Early Neutron Source (IFMIF-DONES)
Readiness of underlying physics assumptions makes the difference.

The systems code PROCESS is being used to underpin EU DEMO design studies, and another code (SYCOMORE), is under development.
Divertor and H-mode Operation as “size-drivers”

- One crucial point is the size of the device and the amount of power that can be reliably produced and controlled in it.
- This is the subject of research and depends upon the assumptions that are made on the readiness of required advances in physics, technology and materials developments.

Main objectives:
* **Protect divertor**
  \[ P_{\text{sep}} = P_\alpha + P_{\text{add}} - P_{\text{rad,core}} \]
  - Physics/ Material limits
  \[ P_{\text{sep}} / R \leq 17 \text{MW/m} \]

* **H-mode operation** \((P_{\text{LH}} \sim R)\):
  - \( f_{\text{LH}} = P_{\text{sep}} / P_{\text{LH,scal}} \rightarrow \text{confinement quality and controllability} \)
  - \( P_{\text{sep}} \geq P_{\text{LH}} \)

\[ P_{\text{LH}} = k n^\alpha B^\beta \omega^\gamma R^\gamma \]

R. Kemp (CCFE) - PROCESS
Fix \( P_{\text{el,net}} = 500 \text{ MW} \) \( \tau_{\text{pulse}} = 2 \text{ h} \) - Scan \( Z_{\text{eff}} \)
Preliminary DEMO Design Choices under Evaluation

<table>
<thead>
<tr>
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<tbody>
<tr>
<td>R₀ / a (m)</td>
<td>6.2</td>
<td>9.1 / 2.9</td>
<td>7.5 / 2.9</td>
</tr>
<tr>
<td>K₉₅ / δ₉₅</td>
<td>1.7</td>
<td>1.6 / 0.33</td>
<td>1.8 / 0.33</td>
</tr>
<tr>
<td>A (m²) / Vol (m³)</td>
<td>683 / 831</td>
<td>1428 / 2502</td>
<td>1253 / 2217</td>
</tr>
<tr>
<td>H non-rad-corr / βᴺ (%)</td>
<td>1.0 / 2.0</td>
<td>1.0 / 2.6</td>
<td>1.2 / 3.8</td>
</tr>
<tr>
<td>Pₜₚₑₚ (MW)</td>
<td>104</td>
<td>154</td>
<td>150</td>
</tr>
<tr>
<td>Pₑ / Pₙₑₜ (MW)</td>
<td>500</td>
<td>2037 / 500</td>
<td>3255 / 953</td>
</tr>
<tr>
<td>Iₚ (MA) / fₜₛ</td>
<td>15</td>
<td>20 / 0.35</td>
<td>22 / 0.61</td>
</tr>
<tr>
<td>B at R₀ (T)</td>
<td>5.3</td>
<td>5.7</td>
<td>5.6</td>
</tr>
<tr>
<td>Bₘₙₙₖ × conductor (T)</td>
<td>11.8</td>
<td>12.3</td>
<td>15.6</td>
</tr>
<tr>
<td>BB i/b / o/b (m)</td>
<td>0.45 / 0.45</td>
<td>1.1 / 2.1</td>
<td>1.0 / 1.9</td>
</tr>
<tr>
<td>Av NWL MW/m²</td>
<td>0.5</td>
<td>1.1</td>
<td>1.9</td>
</tr>
</tbody>
</table>

Under revision
Boundary Conditions for Divertor Plasma Facing Components

- **Plasma Compatibility:**
  radiation, dilution, stability, ...
- **Tritium Compatibility:**
  retention, co-deposition, penetration, ...
- **Erosion Behaviour:**
  lifetime, dust production, ...
- **‘Corrosion‘ Issues:**
  reactions with seeding impurities, air, water, ...
- **Thermo-Mechanical Behaviour:**
  thermal conductivity, strength, DBTT, ...
- **Joining Issues:**
  joining to / compatibility with heat sink materials, ...
- **Behaviour Under n-Irradiation:**
  activation, transmutation, change of thermo-mechanical properties, ...
- **Industrial production routes:**
  availability, scalability, reliability of processes, ...

R-2: A. Leonard    R-6: M. Wischmeier
R-8: W. Morris     I-3: B. Lipschultz

I-10: J.W. Coenen

I-10: J.W. Coenen    I-5: S. Hong
P4: S. Khirwadkar   P5: Th. Loewenhoff

I-10: J.W. Coenen

I-11: Rieth    I-10: Coenen    I-6: Firdaouss
O-6: Riesch     O-11: You       P-8: Nikolic

I-11: M. Rieth    O-11: J.-H.You

I-11: M. Rieth    I-10: J.W. Coenen

O-10: N. Wang / G. Luo
Basic Plasma - Wall Interaction Processes

PWI & PFM determine
- component lifetime
- T retention
- dust production
- plasma compatibility
## P/R as a figure of merit

| Device | \( \frac{P_{\text{heat}}}{R} \) (MW/m) | \( \text{upstream } q_{||} \) (GW/m²) | unmitig. \( q_{\bot} \) \( (\lambda_{\text{int}} = 2.6 \text{ mm}) \)* (MW/m²) |
|--------|---------------------------------|---------------------------------|---------------------------------|
| JET    | 7-12                            | 2                               | 8                               |
| AUG    | 14                              | 3.5                             | 13                              |
| ITER   | 20                              | 5                               | 20                              |
| DEMO   | 80-100                          | \( \geq 30 \)                    | 115                             |

*based on the scaling of upstream SOL width (no size scaling and no radiation losses) by T. Eich et al. PRL 2011, see also A. Scarabosio JNM 2013

⇒ strong mitigation (> factor 7) of heat flux necessary
⇒ radiative cooling (bulk, SOL & divertor)

M. Wischmeier, JNM 2015
Steady state and transient thermal loads (in ITER)

Disruptions
- fast current quench

VDEs
- Vertical Displacement Events
  (loss of position control)

ELMs:
- Edge Localized Modes
  (periodical ejection of particles and energy in high confinement (H)-Mode)

(⇒ all transients need mitigation!)
Operational domain of high power H-mode in AUG

\[ P_{\text{sep}}/R \] is divertor identity parameter, provided similar density and power width

\[ \Rightarrow \] applying the ITER divertor solution to DEMO, high \( f_{\text{rad}} \) is needed!
Bulk radiation will strongly narrow operational range

0-D Calculations for Ignition

He Concentrations are typically 10-20% (for $\tau_{He}/\tau_E=5$)

T. Pütterich, EFPW Split 2014
Hydrogen Retention in Irradiated Tungsten

- damage by 20 MeV W ions, 0.9 dpa
- decoration of defects by gentle D implantation

⇒ typical D concentration ~ 1.5% due to damage (saturation @ ~ 1 dpa)
⇒ some annealing of defects already at 550 K, 30% of initial defects still present after annealing at 1150 K
⇒ substantial decrease of defects requires annealing temperatures above 1150 K

Amount of D trapped in W after damaging and thermal annealing of defects for 60 min

Hydrogen Isotope Exchange in Self-Damaged Tungsten

- damage by 20 MeV W ions, 0.9 dpa
- decoration of defects by gentle D implantation

Changeover to other H species:
- exchange throughout the whole damaged layer (≈ 2 µm)
- dynamics: a “filled” trap behaves like an “empty” trap in contrast to previous model assumptions
- quantitative comparison with new TESSIM code can be achieved using new trapping model

\[ T = 450 \text{ K}, \ E < 5 \text{ eV/atom}, \ j = 5 \times 10^{19} \text{ m}^{-2}\text{s}^{-1} \]

D removal as function of H fluence for D saturated and
D uptake as function of D fluence for
a) virgin and b) H saturated self-damaged W

1.4 at.-% D in damage peak

T Diffusion Barrier towards Cooling Channel

Hydrogen isotopes diffuse easily in metals

- Radioactive inventory and material embrittlement
- Permeation of T₂ into coolant (safety, corrosion!)

⇒ Reduction of permeation by a factor 50 - 100 necessary

• Activation properties of films have to be considered (erbia and alumina possibly out!)

J.W. Coenen
PFMC2015
Impact of power flux limit: Limit for particle flux

Power on target:

\[ P = (8T + 13.6 + 2.2) \times 1.602 \times 10^{-19} \Gamma \ [W] ; \ T_e = T_i = T [eV] \]

power across sheath surface recombination

(neglecting power loads on PFCs from radiation)

⇒ for \( T_e < 2.5 \) eV:
heat flux similar to power deposited by surface recombination processes!

with power load via radiation to \( \sim 2 \) MW/m\(^2\) (for ITER A. Loarte et al. PoP 2011)

⇒ 5 MW/m\(^2\) with \( T = 1.5 \) eV and \( \Gamma < 5 \times 10^{23} \) m\(^{-2}\)s\(^{-1}\)

⇒ Reduction of \( T < 2 \) eV not meaningful (without volume recombination)

M. Wischmeier, PSI 2014, See also:
“ITER Physics basis: Chapter 4, power & particle control”, Nucl. Fusion 39 (1999) 2391
Divertor temperature constraints from W PFC lifetime arguments

- For 'typical' impurity mix \( T < 5 \text{ eV} \) to stay below 5 mm / 2 y W erosion (80% prompt re-deposition included)
- ELMs contribute significantly

A. Kallenbach et al., PPCF 2013
Dust generation: Potential Safety & Operational Concern

Safety concerns:

Potential release to environment  ⇒ 1000 kg limit
  W is the major radioactive source
  Dust contains trapped Tritium

Hydrogen production when hot dust reacts with air/steam
  Be major contributor  ⇒ 11 kg Be, 230 kg W limit
  (with carbon:  ⇒ 6 kg C, 6 Be, 6 kg W limit)

Possible pure Dust or Hydrogen/Dust explosion
  Be, (C,) W involved

Operational concerns:

  Dust particles may cause disruptions
    UFOs seen in many machines, e.g. Tore Supra, JET, …
  Possible influence on diagnostics: mirrors, optics, …

B. Pegourié et al., PSI 2008
Assumption:
- Dust generation dominated by erosion, deposition, layer disintegration
- Conversion from erosion to dust about 10 % in Tore Supra and JT-60U
- For safety reasons: 100 %

Total dust limit not reached before scheduled maintenance and exchange of divertor cassettes

Fraction of dust resides in hot (>600°C) areas?

Caution:
Possibly additional dust sources:
- ‘brittle destruction’, melt droplets

Roth et al. PSI 2008
Influence of transient heat loads on PFM integrity

- transient heat loads lead to **surface modifications** at energy/power densities much lower than those necessary for surface melting (< 0.5 MJ/m²)
- possible strong impact on PFC
  - power handling capability
  - lifetime
  - dust production
  - hydrogen retention
- role of e-beam vs. plasma loading to be clarified

W surfaces exposed in JUDITH-2 to ~10⁵ transient pulses of 0.48 ms @ 10 Hz (w/o pre-loading in GLADIS)

Th. Loewenhoff, PSI 2014, JNM accepted
Armour layer thickness: Compromise power handling erosion life-time

<table>
<thead>
<tr>
<th>Material</th>
<th>Thermal Conductivity (W/(m K))</th>
</tr>
</thead>
<tbody>
<tr>
<td>Tungsten</td>
<td>$\kappa \approx 178$</td>
</tr>
<tr>
<td>Copper</td>
<td>$\kappa \approx 390$</td>
</tr>
<tr>
<td>Stainless Steel</td>
<td>$\kappa \approx 17$</td>
</tr>
</tbody>
</table>

\[ T_{surf} = T_{cool} + \frac{q \cdot d_2}{\kappa_2} + \frac{q \cdot d_1}{\kappa_1} \]

\[ q = \frac{T_{sur} - T_{cool}}{d_1/\kappa_1 + d_2/\kappa_2} \]

**1mm W-Armor & 3mm Cu/Steel-pipe**

(T\(_{cool}\)=300°C) \( T_{max}(W) = 1100°C \quad T_{max}(Cu, SS) = 300°C / 600°C \)

\( q_{max}(W/Cu) \approx 40\text{MW/m}^2 \)

\( q_{max}(W/SS) \approx 10\text{MW/m}^2 \)

**20mm W-Armor & 3mm Cu/Steel-pipe**

(T\(_{cool}\)=300°C) \( T_{max}(W) = 1100°C \quad T_{max}(Cu, SS) = 300°C / 600°C \)

\( q_{max}(W/Cu) \approx 7\text{MW/m}^2 \)

\( q_{max}(W/SS) \approx 3\text{MW/m}^2 \)

J.W. Coenen PFMC 2015
Deep cracking of PFCs in steady state HHF tests

Deep cracking of ITER W monoblock targets
In HHF tests at 20 MW/m²

- assessment of **low cycle fatigue lifetime** to **crack initiation** on the armor surface and
  **FEM (J-integral)** calculations for the crack tip load of **brittle fracture**

- good agreement with experiments: brittleness of W leads to failure during cooling

\[ \Rightarrow \text{need to increase toughness of W!} \]
Improvement of Materials Properties:
for example $W_f / W$ Composites

Increase the toughness of tungsten through $W$ fibres embedded in $W$ matrix

**Results:**

- K-Doped $W$ wires show high strength and ductility up to annealing temperatures of 2200 K
- Very high toughness at room temperature due to ductility of fibres
- Toughness after high temperature embrittlement
- $W_f / W$ prospects for use in future fusion reactors:
  - Enhanced of temperature window
  - Solution for cracking problem

$W_f / W$ sample: 10 layers a 220 fibres, fibre volume fraction $\approx 0.3$

$A \approx 25 \text{ cm}^2$, $V=10.7 \text{ cm}^3$, density 93-96%

J. Riesch, ICFRM 2015
Increasing operational temperature window of materials

- (pure) W poorly matched to heat sink / structural materials
- $W_f / W$ could significantly increase operational window
- $Cu-W_f / Cu - W$ laminates / $Cu-W$ composites could increase temperature window for heat sink

Suitable heat sink materials

Solid elements at RT with thermal conductivity > 50 W/mK

J.H. You, ISFNT 2015
Temperature Profiles in Cooling Channels

Predicted temperature profiles in the cooling tube (coolant: 150 °C)

<table>
<thead>
<tr>
<th></th>
<th>10 MW/m²</th>
<th>15 MW/m²</th>
<th>18 MW/m²</th>
</tr>
</thead>
<tbody>
<tr>
<td>Max. temp. at top</td>
<td>263 °C</td>
<td>316 °C</td>
<td>348 °C</td>
</tr>
<tr>
<td>Mid-temp. at side</td>
<td>172 °C</td>
<td>181 °C</td>
<td>187 °C</td>
</tr>
<tr>
<td>Min. temp. at bottom</td>
<td>150 °C</td>
<td>150 °C</td>
<td>150 °C</td>
</tr>
</tbody>
</table>

Critical material issues for the heat sink

- irradiation creep
  ⇒ high-temperature strength
- neutron embrittlement
  ⇒ toughness / non-ductile structural design

J.H. You, ISFNT 2015
CuCrZr Material Limits

**Design stress limits over temperature**

- Irradiated CuCrZr
- $S_d$ (TF=1)
- $S_d$ (TF=2)
- $S_e$
- $S_m$

<table>
<thead>
<tr>
<th>Limit</th>
<th>Expression</th>
</tr>
</thead>
<tbody>
<tr>
<td>Local fracture due to</td>
<td>$S_d$</td>
</tr>
<tr>
<td>exhausted ductility</td>
<td></td>
</tr>
<tr>
<td>Plastic flow localisation</td>
<td>$S_e$</td>
</tr>
<tr>
<td>Ratchetting</td>
<td>$3S_m$</td>
</tr>
</tbody>
</table>

Allowable operation temp. range according to *elastic* design rules:
- 250 °C – 300 °C
- Impracticable for DEMO divertor

Coolant temp. → Severe corrosion

Heat sink temp. range (10-20 MW/m²)

J.H. You, ISFNT 2015
Neutron irradiation effect on thermal conductivity

J. Linke, Phys. Scr. 2006
HHF performance of neutron irradiated divertor modules

Dunlop Concept 1 (12 mm) / CuCrZr $T_{irr} = 350^\circ C / 0.3 \text{ dpa}$

J. Linke, Phys. Scr. 2006
Thermal fatigue testing of a W macrobrush module irradiated in the HFR-Petten

irradiation condition: 200°C – 0.1 dpa (in W)
loading condition: 1000 cycles at 10 MW/m²

J. Linke, Phys. Scr. 2006
Thermal fatigue testing of W monoblock mock-ups

- Results -

<table>
<thead>
<tr>
<th></th>
<th>W-monoblock</th>
<th>W-monoblock</th>
<th>W-lamellae design</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>ENEA</td>
<td>CEA</td>
<td>Plansee AG</td>
</tr>
<tr>
<td>unirradiated</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1000 x 14.5 MW/m²</td>
<td>1000 x 9.6 MW/m²</td>
<td>1000 x 7.5 MW/m²</td>
<td></td>
</tr>
<tr>
<td>1000 x 18.0 MW/m²</td>
<td>1000 x 14.4 MW/m²</td>
<td></td>
<td></td>
</tr>
<tr>
<td>0.1 dpa</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>$T_{irr} = 200^\circ$C</td>
<td>1000 x 10.0 MW/m²</td>
<td>1000 x 10.0 MW/m²</td>
<td></td>
</tr>
<tr>
<td>100 x 13.7 MW/m²</td>
<td>1000 x 13.7 MW/m²</td>
<td></td>
<td></td>
</tr>
<tr>
<td>1000 x 17.9 MW/m²</td>
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<td></td>
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<tr>
<td>0.6 dpa</td>
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<td></td>
<td></td>
</tr>
<tr>
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<td>1000 x 10.0 MW/m²</td>
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</tr>
<tr>
<td>1000 x 18.0 MW/m²</td>
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<td></td>
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</tr>
</tbody>
</table>

J. Linke, Phys. Scr. 2006

no failure observed!
Activation: Careful selection of constituents

**DEMO: Activation of first wall materials**
Contact dose rates after 5 years of operation (PPCS model B)

![Graph showing activation rates of different materials](image)

- **W/W materials** (fibre volume = 30 %, \( d_{\text{fibre}} = 150 \, \mu m \), thickn.\( \text{coating} = 1 \, \mu m \))
- 18 ppm K

*Source: R.A. Forrest et al., Handbook of Activation Data, 2009*
Actively water-cooled targets
- 900 components with ~18,000 CFC tiles
- only HHF loading generates thermo-mechanical stress similar to the expected loading

Development of a statistical assessment method based on the surface temperature increase ΔT after 100 cycles at 10 MW/m²

Results of HHF tests
- 8% of delivery tested in GLADIS
- no failures for the ~900 tested CFC tiles
- add. HHF tests confirm high quality of process:
  - 1000 x 16 MW/m², 30 s at 30 MW/m²
- individual samples match well to the expected Gaussian distribution
- expected number of undiscovered tiles exceeding the specification limit: <5 × 10⁻⁶ tile ➔ low risk

Histogram of ΔT for all tested CFC tiles

H. Greuner PFMC 2015
Summary and Conclusions

- divertor PFCs face a multitude of partly contradicting requirements
- there is a strong interdependency of PFMs and plasma solutions
- DEMO will most probably require advanced materials and design rules (almost independently of detailed solution)
- 'new' armour materials must also qualify under relevant PWI conditions
- properties changes under high energy neutron irradiation must be taken into account at an early stage
- solutions must be scalable to reliable industrial production routes
R-2: A. Leonard
Divertor constraints for core & performance
I-3: B. Lipschultz
Key issues and goals for divertor detachment performance and control
I-4: N. Krasheninnikov
On the physics of divertor detachment and detachment stability
I-8: M. Bernert
High radiation scenarios in pronounced detached divertor conditions at ASDEX Upgrade
R-6: M. Wischmeier
A review on the current status of power and particle exhaust physics: modeling, experiment and open issues
R-8: G. Morris
Integrated exhaust for DEMO class devices
PFM Relevant Talks and Posters at Technical Meeting

O-6 : F. Maviglia
Development of DEMO wall heat load specification

O-7: R. Neu on behalf of J. Riesch
Contributions of WfW composites to divertor concepts of future fusion reactors

I-5: S.H. Hong
Safety Issues of Dust

I-6: M. Firdaouss
Technological drivers & operational window of a water cooled divertor

O-10: N. Wang / G. Luo
ITER-like tungsten divertor development and experiments on EAST

O-11: H. You
Critical issues & challenges in the engineering of DEMO divertor target

I-9: M. Rieth
The European R&D Programme on Divertor Armor materials and Technology – Status and Strategy

I-10: J. Coenen
Materials for DEMO and Reactor Applications – Boundary Conditions and New Concepts

P4: S. Khirwadkar
Performance of ITER-like divertor targets under non-uniform and transient thermal loads

P5: Th. Loewenhoff
Thermal shock simulation by electron beam and laser devices

P-8 V. Nikolic
How to obtain ductile tungsten