The first ITER tungsten divertor: operating space and lifetime

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The views and opinions expressed herein do not necessarily reflect those of the ITER Organization.
• Introduction to the ITER W divertor
  ▪ Basic physics/design features and expected lifetime
• Stationary power loading – the design simulation database
  ▪ Overall characteristics
  ▪ Focus on factors influencing the peak power loading and definition of acceptable loads
  ▪ Are scalings from simple models applicable?
• Summary
• Note 1:
  ▪ 3-D fields aspects (ELM control) not considered here.
  ▪ See talk (72) by H. Frerichs in 3-D fields session for ITER divertor modelling (Tuesday morning)

• Note 2:
  ▪ Transients (ELMs) not discussed here (unless there is time)

• Note 2:
  ▪ Much of the material in this talk can be found in the paper just published which accompanied the PSI-2018 review talk: R. A. Pitts et al., https://doi.org/10.1016/j.nme.2019.100696
The ITER tungsten divertor

• The most sophisticated tokamak divertor ever built
  ▪ 54 individual cassettes, fully water cooled, designed to handle up to ~100 MW in steady state
  ▪ Now entering the procurement phase → design complete
Monoblocks in HHF areas will be toroidally bevelled to protect inter-PFU misalignments, vertical targets tilted to protect inter-cassette misalignments.

- Compromise between poloidal gap edge overheating and increased surface stationary loading.
Revised ITER schedule and divertor lifetime

- Divertor replacement currently foreseen in the ITER Research Plan at the end of the first D-T phases (3 campaigns, FPO-1,2,3) Plan
  - ~13 years after installation
Updated ITER Research Plan

- Available publicly as ITER Technical Report (ITR-18-003)

- IRP informs the fusion community on details of experimental plans to achieve the Project goals and defines the required supporting R&D

- Expect ~900 days of D-T operation over ~5 years in FPO-1,2,3
  - ~12,000 pulses
  - ~8x10^6 s plasma time (~2200 hrs)
Burning plasma operating window

• Focus on “burning plasma” conditions → the most challenging for the ITER divertor
  ▪ $Q_{DT} = 10$, $P_{IN} \sim 100$ MW
  ▪ Ne and N seeding (emphasis on Ne where database currently largest)
  ▪ No discussion of “integrated modelling” here
  ▪ Divertor simulation database largely constructed with SOLPS-4.3, with more recent analysis using SOLPS-ITER

• An important fact to bear in mind: ITER will operate always quite close to the H-mode power transition threshold
  ▪ Cannot afford (too) much edge/core radiation (i.e. not “DEMO-like”)
Main simulation database parameters

- Steady state – no ELMs
- No fluid drifts, “L-mode” edge
  - Neutral-neutral collisions included
- Fixed equilibrium
  - $q_{95} = 3$, $B_T/I_p = 1.8/5, 2.65/7.5, 5.3/15$
- Fixed cross-field transport
  - $D_\perp = 0.3 \, \text{m}^2\text{s}^{-1}$, $\chi_\perp = 1.0 \, \text{m}^2\text{s}^{-1}$
- Scans in fueling, seed impurity, power into numerical grid ($P_{\text{IN}}$)
- All-metal walls
  - Assume Be everywhere, but no sputtering

<\!c_Z\!>_\text{sep} = n_Z/n_e
averaged around first ring outside separatrix

60 SOLPS-4.3 case numbers used in this talk:
2250 2251 2252 2253 2257 2258 2264 2265 2266 2269 2316 2317 2332 2333 2396 2397 2398 2399 2400 2401 2402 2403 2404 2407 2408 2409 2410 2411 2412 2413 2414 2415 2416 2439 2463 2467 2468 2469 2470 2471 2472 2476 2477 2478 2481 2483 2484 2485 2496 2497 2498 2508 2509 (Ne) 2480 2493 2494 2502 2503 2533 2534 (N)
Main simulation database parameters

• Seek to map out operational space in two key parameters:
  - Divertor neutral pressure
  - Peak divertor target power loading

\[
\alpha = 3.2^\circ \text{ (w/o shaping)} \\
\alpha = 4.7^\circ \text{ (with shaping)}
\]

\[
\alpha = 2.7^\circ \\
\alpha = 4.2^\circ
\]
SOL heat flux width

- Divertor conditions across database do not strongly influence upstream $\lambda_q$
- See talks by C.-S. Chang (66) and X. Xu (16) for more on $\lambda_q$ scaling

$q_{\parallel}(r)$ outer midplane
$q_{\perp}(r)$ X-pt.
$q_{\parallel}(r)$ OT

$q_{\parallel}(r)$ X-pt.

$\lambda_q = 3.4 \pm 0.5$ mm

$P_{IN} = 100$ MW Ne seeded

$r - r_{sep}$omp (mm)
$q_{target, projected}$ (MWm$^{-2}$)

OT Conc.%
- 0.3 Ne
- 0.4 Ne
- 0.6 Ne
- 0.8 Ne
- 1.2 Ne
- 1.8 Ne
Sensitivity to material?

- Target material decides ratio of reflected atoms/molecules
- More molecules from Be target, higher fraction of fast reflected atoms from W
- The overall effect of the two populations is to produce almost the same momentum and power losses

J. S. Park et al, APS (2019)
### Operating window in peak power flux density

- **q\text{peak,target} (MWm}^{-2}\)**
- **Divertor neutral pressure (Pa)\)**

**Note:** \(q_{\text{peak,target}}\) is insensitive to \(c_Z\) especially at low \(p_n\)

Approximate "Detachment limit"

"Historical" stationary power handling limit – will be higher in reality

- Out-in peak power asymmetry reduces in the code at high \(p_n\)
- \(N_2\) and \(Ne\) behave similarly
- Need 3-5x \(N\) than \(Ne\) in the code for given fueling to get same \(<c_Z>_{\text{sep}}\)

OT and IT, Ne & \(N_2\), no shaping
Detachment evolution

- “Classic” evolution from high recycling to partially detached state
- He pumping improves with increased $p_n$ but not if far-SOL also detached

Avoid “complete” detachment $\rightarrow$ keep finite ion flux in outer part of the SOL to maintain sufficient neutral plugging
Now add shaping

- Effects less marked at high $p_n$ where thermal plasma contributions lower
- 20 MWm$^{-2}$: CHF limit with factor 1.4 margin
  - Recently updated* after review of MB CHF test protocol and new dedicated tests now that final thickness decided

$q_{\text{peak,target}}$ (MWm$^{-2}$)

Divertor neutral pressure (Pa)

F. Escourbiac et al, FED 146 (2019) 2036
Impact of drifts

- "H-mode" SOLPS-ITER drift modelling*
- Strong impact on OT loading at low $p_n$ but effect reduced as detachment deepens
  - Drifts increase characteristic pressure at which OT reattachment occurs due to increasing Ne leakage
  - Drifts increase need for good detachment control

*E. Kaveeva et al., submitted to NF
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$q_{\text{peak,target}}$ (MWm$^{-2}$)

Shaping and drifts push operating point to higher $p_n$. Worse still if $\lambda_q$ lower than assumed

Divertor neutral pressure (Pa)

*E. Kaveeva et al., submitted to NF
Radiated fractions

- Radiation largely confined to the divertor region
  - $f_{\text{RAD,DIV}} \sim 0.8-0.9$ across operating window for Ne
  - $f_{\text{RAD,TOT}} \sim 0.3 - 0.7$
  - N radiates more efficiently in the divertor than Ne for same $c_Z$
  - Lower core radiation with N
Radiated fractions

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Divertor radiation distribution

Total radiated power (Wm⁻³)

- Ne radiation more extended than N
  - Expected from differences in ionization potential
  - But still mostly confined to divertor volume

\[ P_{RAD,DIV} = 56.6 \text{ MW} \]
\[ 41.3 \text{ (Ne)} + 15.3 \text{ (D)} \]

\[ c_{Ne} = 0.8\% \]
\[ p_n = 10.3 \text{ Pa} \]

\[ c_N = 0.8\% \]
\[ p_n = 10.1 \text{ Pa} \]

\[ P_{RAD,DIV} = 54.0 \text{ MW} \]
\[ 38.6 \text{ (N)} + 15.4 \text{ (D)} \]
Negligible drift impact at high $p_n$

- SOLPS-ITER with drifts activated
  - $P_{IN} = 100$ MW
  - Matched Ne, N cases
  - H-mode pedestal

Total radiated power (Wm$^{-3}$)

<table>
<thead>
<tr>
<th>Ne</th>
<th>N</th>
</tr>
</thead>
<tbody>
<tr>
<td>$P_{RAD,DIV} = 53.4$ MW</td>
<td>$P_{RAD,DIV} = 54.0$ MW</td>
</tr>
<tr>
<td>38.7 (Ne) + 14.2 (D)</td>
<td>44.7 (Ne) + 9.2 (D)</td>
</tr>
</tbody>
</table>

$c_{Ne} = 2.1\%$ $p_n = 11.6$ Pa

$c_{N} = 2.3\%$ $p_n = 11.4$ Pa

E. Sytova et al, NME 19 (2019) 72
E. Kaveeva et al., submitted to NF
Impact of scale size?

- Now have a set of SOLPS-ITER detached “H-mode” drift simulations with Ne-seeding across a factor 3 in machine size → we are in position to analyse the impact of scale
  - There appears to be a gradual evolution from stronger to weaker drift effect and weaker to stronger impurity retention with increasing size → Ne leakage still occurs in ITER, but cannot radiate in the edge/pedestal because too high $T_e$. 

I. Senichenkov et al, PPCF 61 (2019) 045013

L. Kaveeva et al., unpublished

L. Kaveeva et al., subm to NF

$\Gamma_D = 2 \times 10^{22} \text{ el/s}, \Gamma_{Ne} = 5 \times 10^{20} \text{ el/s}$

$\Gamma_D = 3.4 \times 10^{22} \text{ el/s}, \Gamma_{Ne} = 2 \times 10^{20} \text{ el/s}$

$\Gamma_D = 2 \times 10^{23} \text{ el/s}, \Gamma_{Ne} = 2 \times 10^{21} \text{ el/s}$

$I_{Ne} (10^{19} \text{ m}^{-3})$

$<c_{Ne}>_{sep} = 2.2\%$

$<c_{Ne}>_{sep} = 0.9\%$

$<c_{Ne}>_{sep} = 2.1\%$
Upstream density dependence

- Little or no dependence of \( n_{e,sep} \) on \( p_n \).
- Strong dependence on \( c_{Ne} \) at fixed \( p_n \).
  - \( c_{Ne} \uparrow \rightarrow P_{rad,div} \uparrow \rightarrow \) power available for dissociation/ionization/excitation of fuel molecules/atoms decreases \( \rightarrow n_{e,sep} \downarrow \)
- Low \( c_{Ne} \) favourable for \( Q_{DT} = 10 \), but higher \( c_{Ne} \) for higher \( P_{rad,div} \), so compromise required (especially if \( \lambda_q \) low)
Density dependence of $c_Z$ (l)

- Dependence ranges from $n_{e,sep}^{-4} \rightarrow n_{e,sep}^{-2}$ for points at low (high) $P_{rad,div}$
- Stronger than predicted by simple models (e.g. Lengyel) at lower $P_{rad,div}$

$$<c_{Ne}>_{sep} \%$$

$$P_{rad,div} (MW) \quad \alpha$$

- 26.3 $\rightarrow$ 33.5  -3.9
- 33.5 $\rightarrow$ 40.7  -2.7
- 40.7 $\rightarrow$ 48.0  -2.7
- 48.0 $\rightarrow$ 55.2  -2.8
- 55.2 $\rightarrow$ 62.5  -2.0

$$<c_{Ne}>_{div} \propto n_{e,sep}^{-\alpha}$$

$n_{e,sep} (10^{19} \text{ m}^{-3})$
Density dependence of $c_Z$ (II)

- $<c_{Ne}>_{div}$ averaged over first SOL ring outside the separatrix below the X-point at the outer target

- Overall dependence
  
  $<c_{Ne}>_{div} \propto n_{e,sep}^{-2.8}$

- Similar to that found by S. Henderson on AUG with N-seeding close to detachment (talk 43)
Comparison with Lengyel

- Focus on region around rollover at each cZ, and 3rd SOL ring outside separatrix (r-rsep)OMP \sim \lambda_q
- Remarkably good agreement with trend
- x4 higher cZ predicted by LM due to:
  - Additional heat losses (radial transport and neutrals) and heat flux channels (convection, ion conduction) in SOLPS

Simplified LM (T_{e,t} = q_{||,t} = 0)

\[ c_z = \frac{q_{||}^2}{2 \kappa_{e||0} n_{eu} T_{eu} T_z^{ADAS} \sqrt{T_e dT_e}} \]

\[ T_{eu} = T_{eu}^{2PM} = \frac{7}{2} \left( \frac{q_{||} L_{||}}{\kappa_{e||0}} \right)^{\frac{2}{7}} \]
Avenues for reducing $q_{\text{peak,target}}$ (I)

- SOLPS-ITER to push the detachment level beyond the SOLPS-4.3 database

Work just starting → towards X-point radiator

J. Lore, ORNL
Avenues for reducing \(q_{\text{peak,target}}\) (II)

- SOLPS-ITER to study expanded (poloidal) configurations

- Magnitude of \(q_{\text{peak,target}}\) reduction is roughly consistent with reduced angle of incidence

J. Canik et al., APS 2019
“Lifetime” power density limit?

- Define “Operational budget” for $q_{\text{peak,target}}$ in terms of time required for W hardness to drop by 50% at 2 mm depth below MB surface
  - ~2 mm recrystallization depth consistent with recent FEM modelling for crack onset due to low cycle fatigue
  - Two new curves from dedicated studies under ITER contract added since PSI-2018
"Lifetime" power density limit?

- Define "Operational budget" for $q_{\text{peak,target}}$ in terms of time required for W hardness to drop by 50% at 2 mm depth below MB surface.
  - ~2 mm recrystallization depth consistent with recent FEM modelling for crack onset due to low cycle fatigue.
  - Two new curves from dedicated studies under ITER contract added since PSI-2018.

Gives $q_{\text{peak,target}} \approx 16$ MWm$^{-2}$ for first ITER divertor to end of first FPO phases.

G. De Temmerman et al, PPCF, 60 (2018), ICFRM 2019
P1-P5, from S. Panayotis, NME 12 (2017)
Summary

• The ITER W divertor design is now complete
  ▪ Prototyping of all major components at an advanced stage

• Divertor burning plasma operating window well established
  ▪ Impurity seeded, partially detached operation with radiation well confined to the divertor - N or Ne seem acceptable
  ▪ But target shaping, drifts, possible narrow $\lambda_q$ all push window to higher divertor neutral pressure
  ▪ Increased peak power handling capacity (based on W recrystallization threshold) adds some margin

• ELM suppression remains the objective
  ▪ If ELMs are to be allowed, they have to be extremely small
Still much to do for ITER

• The ITER divertor is being procured and the design cannot now change.
• But a lot of R&D still required in the years before the divertor is first used
  ▪ See Page 23 of https://doi.org/10.1016/j.nme.2019.100696
Thank you
Reserve slides
W divertor: key physics characteristics

- Deep vertical targets and baffle regions promoting detachment and reducing neutral escape to the core.
- Dome to improve pumping → lower pumping speed required for given upstream He conc or fuel throughput.
- Reflectors plates protect against downward strike point excursions.
- Transparency between targets for neutral recirculation – lower power asymmetries.
Vertical targets and component shaping

16 PFUs
138 monoblock/PFU
Total: 119,232

22 PFUs
143-146 monoblock/PFU
Total: 172,962

Monoblock CuCrZr tube
Reduces impact of downward VDEs (limiter config.)
Progress in manufacture/prototyping

Both JA-DA and EU-DA have met tolerances on vertical target MB alignments
Impurity charge state distribution

- ~87% of the divertor radiation from \( \text{Ne}^{+3} \rightarrow \text{Ne}^{+6} \)
  - Well confined in the divertor region \( \rightarrow T_e \) high enough, far enough
  - Ne fully stripped in pedestal region and cannot radiate
Integrated target ion fluxes

Integral ion current ($x10^{24}$ Ds$^{-1}$)

- Turnover in total plate current generally rather gentle
  - Loose criterion for “tolerable detachment” fixed as point at which integral flux reaches ~80% of peak value after rollover (based historically on discussions with JET) → happens typically near $p_n$ ~10 Pa
Total pressure-momentum losses

\[ p_{tot} = k (n_e T_e + n_i T_i) (1 + M^2) \]

- Pressure loss downstream as \( p_n \) increases
  - Upstream \( p_{tot} \) unaffected by downstream conditions (as for \( \lambda_q \))
  - Beyond region of pressure loss, upstream and downstream profiles overlap
• Momentum and power losses in the ITER simulation database strongly correlated with $T_{e,t}$
  ▪ Functions proposed by Stangeby* work well

\[ p_{\text{tot}, t}/p_{\text{tot}, u} = 1 - f_{\text{mom-loss}} \]
\[ q_{\parallel, t}/q_{\parallel, u} = 1 - f_{\text{cooling}} \]

\[ 1 - f_{\text{tot}} = A (1 - e^{-T_{et}/T^*})^n \]

\[ T_{e,t} = 4.97 \times 10^{13} n_{D2,t}^\beta \]

$R^2 = 0.93$

*P. C. Stangeby, PPCF 60 (2018) 044022
What will be the true $\lambda_q$ on ITER?

- XGC1 electrostatic global gyrokinetic simulations match $\lambda_q \propto 1/B_{pol}$ scaling
- Data out to ITER $B_{pol}$ on C-Mod continue to follow scaling

C. S. Chang et al. NF 57 (2017) 116023

D. Brunner et al., NF 58 (2018) 094002
Impact of shaping

- Need to apply angle corrections for global target tilting and monoblock toroidal shaping only to thermal plasma components
  - Kinetic plasma plus potential energy of recombination at the plate: $\gamma n_{et}c_{st}T_{et} + n_{et}c_{st}E_{pot}$
Impact of shaping

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  - Kinetic plasma plus potential energy of recombination at the plate:
    \[ \gamma n_\text{et} c_\text{st} T_\text{et} + n_\text{et} c_\text{st} E_{\text{pot}} \]
Brand new results from SOLPS-ITER

- Be/W walls, same SOL transport as SOLPS-4.3 database
- Ne seeding
- \( P_{IN} = 100 \) MW
- H-mode pedestal now included
- Sophisticated code speed-up schemes required just to make drift runs possible\(^1\)

\(^1\)E. Kaveeva et al, NF 58 (2018) 126018
• Use FE simulations to transform $q_{\text{peak}}$ to $T_{\text{surf}}$

- Take value at monoblock centre → where cracking seen to start under high heat flux testing

"Lifetime" power density limit?

$T_{\text{surf,MBcentre}}$ ($^\circ\text{C}$)

Divertor neutral pressure (Pa)

0 500 1000 1500 2000 2500

OT and IT, Ne and N, incl. shaping

S. Panayotis et al., NME 12 (2017) 200

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3rd IAEA TM on Divertor Concepts, Vienna, Austria, 4 - 7 Nov. 2019

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3rd IAEA TM on Divertor Concepts, Vienna, Austria, 4 - 7 Nov. 2019

IDM UID: 2FGYSX
ELMs – what if ELM suppression not possible?

- Encouraging new scaling for target parallel ELM energy $\varepsilon_{||,\text{targ}} \approx 6\pi p_\text{eRq}_{\text{edge}}$

- For ITER targets with shaping, scaling gives:
  - 7.5 MA: $\varepsilon_{\bot,\text{targ}} \approx 0.36 \pm 0.18 \text{ MJ m}^{-2}$
  - 15 MA: $\varepsilon_{\bot,\text{targ}} \approx 1.2 \pm 0.6 \text{ MJ m}^{-2}$
Problem of toroidal gaps

Toroidal bevel protects poloidal leading edges
BUT long toroidal edges are still exposed

• ELM ions problematic due to large Larmor radii of particles arriving from pedestal region

• Toroidal gap (TG) loading really does occur\(^1\)

\(^1\)R. Dejarnac et al, NF 58 (2018) 066003
Constraints on ELM energy loss

- To avoid toroidal gap edge melting, $\frac{\Delta W_{\text{ELM}}}{W_{\text{plasma}}} \lesssim 0.2\%$
  - Very similar to limits found for no surface damage from large cycle number, ELM-like e-beam testing of ITER-grade W$^*$
  - Such ELMs not found naturally on tokamaks
  - ELM buffering in detached divertor?

Adapted from J. P. Gunn et al. NF 57 (2017) 046025

*M. Wirtz et al., NME 12 (2017) 148