Heat and particle exhaust in tokamak reactors

ITER tungsten divertor

challenges and potential solutions

Brian LaBombard

ITER tungsten divertor
figures courtesy of Richard Pitts

PSFC Seminar, Dec. 20, 2013
Heat and particle exhaust in tokamak reactors: challenges and potential solutions

Outline:

“The tail is wagging the dog.”

Challenges for ITER
Disruptions
ELMs
Power exhaust

Challenges for DEMO
Extreme power + steady state

Potential solutions
Advanced divertors
Double null

Critical R&D needed
Advanced divertor test facility
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ITER tungsten divertor
Choice: ‘vertical target’ design
54 cassettes, 486 tonnes
~ €340M, ~20 years to DT
One chance.

Did we choose correctly?
What about DEMO?
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MFE development has entered a new era: **boundary & PMI rules** – “the tail is wagging the dog”

Tremendous progress has been made in optimizing core plasma performance (with boundary & PMI a relatively minor concern) ...
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... while surface heat removal technologies are limited to ~5 MW/m$^2$ in nuclear environment; they cannot be enhanced.

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-- P. Stangeby, NF 51 (2011) 063001.

Boundary & PMI now set reactor design & performance

ARIES-ACT study (2012)\(^1\): “Device size is set by the divertor heat flux”-- F. Najmabadi

\[1\] http://aries.ucsd.edu/LIB/REPORT/CONF/ANS12/1208-TOFE-Najmabadi-presentation.pptx
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ITER is confronting boundary & PMI challenges
-- identifying critical research needs

ITER $Q_{DT} = 10$; $t \sim 400$ s
Type I ELMing H-mode

- Disruptions
- ELMs
- Power exhaust

$P_{FUS} = 500$ MW
$P_{\alpha} = 100$ MW
$P_{RAD} = 50$ MW
$W_p = 210-350$ MJ
$\Delta W_{ELM} = 10-20$ MJ
$P_{SOL} \sim 100$ MW
Energy-damage limits of materials are well characterized. Plasma operation must respect them.

ELM simulation results from QSPA facility in Troitsk\(^1\)

Test: ELM duration ~ 0.5 ms
ELM size ~ 1/10th ITER

Exposed samples (preheated to 500\(^0\)C)

- **W (>99.96%)**, 
  \(Q_{\text{max}} = 1\text{MJ/m}^2\) 
  (100 pulses) 
  Negligible mass loss

- **W-1%La\(_2\)O\(_3\)**, 
  \(Q_{\text{max}} = 1\text{MJ/m}^2\) 
  (100 pulses) 
  \(\Delta h = 0.04 \mu\text{m/pulse}\)

- **W (>99.96%)**, 
  \(Q_{\text{max}} = 1.5\text{MJ/m}^2\) 
  (100 pulses) 
  \(\Delta h = 0.06 \mu\text{m/pulse}\)

\(^1\)Klimov, N., et al., "Experimental study of PFCs erosion under ITER-like transient loads at plasma gun facility QSPA," 18\(^{th}\) PSI Conference, Toledo, Spain, 2008.

Melt threshold for W: \(F_{\text{HF,melt}} \sim 50 \text{ MJm}^{-2}\text{s}^{-1/2}\)
ITER Q_{DT} = 10; t \sim 400 \text{ s}
Type I ELMing H-mode

Disruptions –must be mitigated!

ELMs
Power exhaust

Melt threshold for W: F_{HF,melt} \sim 50 \text{ MJ m}^{-2} \text{s}^{-1/2}

[1] Richard Pitts, APS 2013, Denver
ITER Q_{DT} = 10; t \sim 400 \text{ s} 
Type I ELMing H-mode

Disruptions –must be mitigated!
ELMs –must be mitigated x1/20!

Power exhaust

For Q_{DT} = 10, maximum ELM size is \Delta W_{ELM} \sim 0.7 \text{ MJ.}
ITER $Q_{DT} = 10; t \sim 400$ s
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Disruptions –must be mitigated!
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For $Q_{DT} = 10$, maximum ELM size is $\Delta W_{ELM} \sim 0.7$ MJ.
ITER is confronting boundary & PMI challenges -- identifying critical research needs

\[ P_{\text{FUS}} = 500 \text{ MW} \]
\[ P_{\alpha} = 100 \text{ MW} \]
\[ P_{\text{RAD}} = 50 \text{ MW} \]

\[ P_{\text{DT}} = 10; \ t \sim 400 \text{ s} \]
Type I ELMing H-mode

Disruptions –must be mitigated!
ELMs –must be mitigated x1/20!

Power exhaust?

For \( \lambda_q = 5 \text{ mm} \),
\[ q_{\perp,\text{max}} \sim 40 \text{ MWm}^{-2} \]
ITER Q_{DT} = 10; t \sim 400 \text{ s} 
Type I ELMing H-mode

Disruptions –must be mitigated!
ELMs –must be mitigated x1/20!

Power exhaust?

Without divertor dissipation, surface will be damaged in only \sim 1\text{s} of ITER operation!

For \lambda_q = 5 \text{ mm},
q_{\perp,max} \sim 40 \text{ MWm}^{-2}

\lambda_q \sim 10x \lambda_q
Unmitigated steady-state heat fluxes in C-Mod results in divertor surface damage

- Axisymmetric melt damage to divertor tiles \( \text{in } \sim 0.2 \text{ s of 1 discharge} \)
- \( \sim 4 \text{ MW into SOL (I-Mode)} \) -- P/S \( \sim 0.6 \text{ MW/m}^2 \)
- \( q_{\parallel} \sim 0.7 \text{ GW/m}^2 + \text{operator mistake, strike-point on pumping slot tiles} \)

Message: C-Mod simulates ITER divertor conditions \( (q_{\parallel}, n_{Te}, n_0, \ldots) \)
ITER is confronting boundary & PMI challenges -- identifying critical research needs

ITER $Q_{DT} = 10$; $t \sim 400$ s
Type I ELMing H-mode

- $P_{FUS} = 500$ MW
- $P_\alpha = 100$ MW
- $P_{RAD} = 50$ MW

$P_{IN} = 50$ MW
$P_{SOL} \sim 100$ MW

Disruptions –must be mitigated!
ELMs –must be mitigated x1/20!

Power exhaust?

Without divertor dissipation, surface will be damaged in only $\sim 1$ s of ITER operation!

For $\lambda_q = 5$ mm,
$q_{\perp,\text{max}} \sim 40 \text{ MWm}^{-2}$
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- $P_{\text{FUS}} = 500$ MW
- $P_{\alpha} = 100$ MW
- $P_{\text{RAD}} = 50$ MW

$P_{\text{IN}} = 50$ MW

Disruptions – must be mitigated!
ELMs – must be mitigated $\times 1/20$!

Power exhaust?

For $\lambda_q = 5$ mm, $q_{\perp,\text{max}} \sim 40$ MWm$^{-2}$

Require $f_{\text{rad,div}} \sim 0.7 \rightarrow \text{partial divertor detachment}$ to reduce $q_{\perp,\text{max}}$ to $\sim 10$ MWm$^{-2}$
ITER Q\textsubscript{DT} = 10; t \sim 400 \text{s}
Type I ELMing H-mode

Disruptions –must be mitigated!
ELMs –must be mitigated x1/20!

Power exhaust?

\textbf{BUT}--New results from multi-machine database (2010):
\(\lambda_q \) appears to be independent of machine size –
depends only on \(B_{pol} \)

\( P_{FUS} = 500 \text{ MW} \)
\( P_\alpha = 100 \text{ MW} \)
\( P_{RAD} = 50 \text{ MW} \)

\( \lambda_q \sim 1 \text{ mm?} \)
1/5 of ‘planned’ value

(caveat: low divertor recycling, H-mode conditions)
ITER is confronting boundary & PMI challenges
-- identifying critical research needs

ITER $Q_{DT} = 10$; $t \sim 400$ s
Type I ELMing H-mode

Disruptions –must be mitigated!
ELMs –must be mitigated $\times 1/20$!

Power exhaust?

$\lambda_q$ appears to be independent of machine size –
depends only on $B_{pol}$

BUT--New results from multi-machine database (2010):

(caveat: low divertor recycling, H-mode conditions)

Increase $P_{rad,core}$?
Increase $P_{rad,div}$?

Yet, must avoid full detachment...

ITER

$\lambda_q \sim 1$ mm?
1/5 of ‘planned’ value

$P_{FUS} = 500$ MW
$P_{\alpha} = 100$ MW
$P_{RAD} = 50$ MW
ITER is confronting boundary & PMI challenges -- identifying critical research needs

ITER $Q_{DT} = 10$; $t \sim 400$ s
Type I ELMing H-mode

Disruptions –must be mitigated!
ELMs –must be mitigated $x1/20$!
Power exhaust?

Sensitivity study\(^1\) shows strong impact of $\lambda_q$ on ITER’s operational window to access to $Q=10$.

Widow of $Q_{DT}$ vs. $P_\alpha$ at fixed $q_{\perp,max} = 10$ MWm\(^{-2}\)

$\lambda_q = 3.6$ mm

Tolerable
$P_{\text{FUS}} = ?$ MW
$P_\alpha = ?$ MW

ITER is confronting boundary & PMI challenges -- identifying critical research needs

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Widow of $Q_{DT}$ vs. $P_\alpha$ at fixed $q_{\perp,\text{max}} = 10$ MWm\(^{-2}\)

$\lambda_q = 3.6$ mm

$\lambda_q = 1.2$ mm

Note: Intrinsic carbon only; impurity seeding not optimized ...
ITER is confronting boundary & PMI challenges -- identifying critical research needs

ASDEX-Upgrade

$P_{\text{sep}}/R \sim 7 \text{ MW/m}^2$ experiment\(^1\)

Disruptions –must be mitigated!
ELMs –must be mitigated $x1/20$!
Power exhaust?

Experiments are trying to find mix of core & divertor radiation to demonstrate a PMI solution for ITER.

Feedback controlled N-seeding; $P_{\text{sep}} \sim 12 \text{ MW}$, $q_{\text{div}} < 5 \text{ MW/m}^2$, $H \sim 1$, $\beta_N \sim 2.8$

$P_{\text{sep}}/R \sim 7$ is $\sim 1/2$ value required for ITER

(But if $\lambda_q \sim 1/B_{\text{pol}}$, $P_{\text{sol}}B/R$ is a better metric => $1/4$ ITER.)

ITER is confronting boundary & PMI challenges -- identifying critical research needs

Alcator C-Mod
High $P_{\text{rad}}$ seeding experiments\(^1,2\) $P_{\text{sep}}/R \sim 3 \text{ MW/m}$

Disruptions –must be mitigated!
ELMs –must be mitigated x1/20!
Power exhaust?

Experiments are trying to find mix of core & divertor radiation to demonstrate a PMI solution for ITER.

The game is to trade-off some core performance ($H_{98}$, $Z_{\text{eff}}$) for reduced $P_{\text{sol}}$ and $P_{\text{div}}$.

It is not yet known what performance will be attained in ITER, compatible with PMI.

[1] Loarte, PoP 18 (2011) 056105
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Challenges for DEMO
- Extreme power + steady state
  1 – Power exhaust
  2 – Material erosion
  3 – Divertor/PMI solution compatible with burning plasma core

+ fully mitigated disruptions and ELMs (no ELMs)!
Challenge #1: Power exhaust

‘ITER solution’ for power handling -- does not apply for DEMO

ACT1, ACT2 have 4 to 10 times higher $P_\alpha B/R$ than ITER

<table>
<thead>
<tr>
<th></th>
<th>ITER</th>
<th>ARIES-ACT1</th>
<th>ARIES-ACT2</th>
</tr>
</thead>
<tbody>
<tr>
<td>R(m)</td>
<td>6.2</td>
<td>6.25</td>
<td>9.75</td>
</tr>
<tr>
<td>B(T)</td>
<td>5.3</td>
<td>6.0</td>
<td>8.75</td>
</tr>
<tr>
<td>$P_\alpha$ (MW)</td>
<td>100</td>
<td>360</td>
<td>520</td>
</tr>
<tr>
<td>$P_{\text{fusion}}$ (MW)</td>
<td>500</td>
<td>1800</td>
<td>2600</td>
</tr>
<tr>
<td>$P_\alpha B/R$</td>
<td>85</td>
<td>350</td>
<td>810</td>
</tr>
<tr>
<td>$f_{\text{rad,core}}$</td>
<td>66%?</td>
<td>~85%</td>
<td>~90%</td>
</tr>
</tbody>
</table>

Conventional divertor requires extreme levels of core radiation to survive.

Noted by Kotschenreuther\(^1\) -- discussed at PSFC Seminar, Dec. 16, 2005.

http://aries.ucsd.edu/ARIES/DOCS/bib.shtml

Challenge #2: Material erosion

Even if the power handling challenge could be met with a *conventional divertor*, requirements for SS material erosion/redeposition control in FNSF/DEMO are nearly impossible to meet.

~10 MW/m² SS heat removal requires < ~5 mm thick armor plate.
This restricts net tungsten erosion < 1 mm/year ($\Gamma_W \sim 2 \times 10^{18}$/m²/s).
But divertor ion flux is high ($\Gamma_{i,\perp} \sim 1-2 \times 10^{24}$/m²/s) requiring $\Gamma_W / \Gamma_{i,\perp} < 10^{-6}$.

D.G. Whyte, APS 2012; Stangeby and Leonard, NF 2011

He-cooled tungsten divertor plate with integrated finger units

90,000 to 500,000 ‘finger units’ in a reactor
Even if the power handling challenge could be met with a conventional divertor, requirements for SS material erosion/redeposition control in FNSF/DEMO are nearly impossible to meet.

\[
\text{~10 MW/m}^2 \text{ SS heat removal requires } < \sim 5 \text{ mm thick armor plate.}
\]

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But divertor ion flux is high \((\Gamma_i, \sim 1-2x10^{24}/\text{m}^2/\text{s})\) requiring \(\Gamma_W/\Gamma_i, \sim < 10^{-6}.\)

\[\text{Challenge #2: Material erosion}\]

\[\text{Factor of } < 10^{-6} \text{ net yield requires:}\]

- Efficient prompt redeposition \(> \sim 99\% \text{ (redep. material is mixed } \sim \text{ poor quality)}\)

or

- Fully detached divertor with ion energies below sputtering threshold, \(i. \text{ e., } T_e < \sim 5 \text{ eV (with impurity ions)}\)

or

- Liquid metal targets?

K. Krieger, JNM 266-269 (1999) 207

\[\Gamma_w/\Gamma_i, \sim vs T_{e,\text{div}} \text{ measured in AUG}\]

D.G. Whyte, APS 2012; Stangeby and Leonard, NF 2011

Doerner NF 2012: redep. sputters at x10 rate of bulk (Be)
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  or

- *Fully detached divertor with ion energies below sputtering threshold,*
  
  i.e., $T_e < \sim 5$ eV (with impurity ions)

  or

- Liquid metal targets?
**Challenge #2: Material erosion**

**The divertor “slag” problem**

If $T_e = \sim 10\text{eV}$ at all plasma-wall contact areas, gross material erosion rates would exceed $\sim 30,000 \text{ kg/yr}$ for tungsten in DEMO.

<table>
<thead>
<tr>
<th>Device</th>
<th>$P_{\text{heat}}$ (MW)</th>
<th>Annual run time (s/year)</th>
<th>Fluence of D/T ions to surfaces (TC/yr)$^a$</th>
<th>Beryllium circulation rate (kg/yr)</th>
<th>Boron circulation rate (kg/yr)</th>
<th>Carbon circulation rate (kg/yr)</th>
<th>Tungsten circulation rate (kg/yr)*</th>
</tr>
</thead>
<tbody>
<tr>
<td>DIII-D</td>
<td>20</td>
<td>$10^4$</td>
<td>0.0007</td>
<td>2.8</td>
<td>1</td>
<td>0.5</td>
<td>0.7</td>
</tr>
<tr>
<td>JT-60SA</td>
<td>34</td>
<td>$10^4$</td>
<td>0.0017</td>
<td>4.8</td>
<td>1.7</td>
<td>0.9</td>
<td>1.2</td>
</tr>
<tr>
<td>EAST</td>
<td>24</td>
<td>$10^5$</td>
<td>0.01</td>
<td>34</td>
<td>11</td>
<td>5</td>
<td>8</td>
</tr>
<tr>
<td>ITER</td>
<td>100</td>
<td>$10^6$</td>
<td>0.4</td>
<td>1680</td>
<td>580</td>
<td>270</td>
<td>410</td>
</tr>
<tr>
<td>FDF</td>
<td>100</td>
<td>$10^7$</td>
<td>3</td>
<td>13400</td>
<td>4600</td>
<td>2100</td>
<td>3300</td>
</tr>
<tr>
<td>Reactor</td>
<td>400</td>
<td>$2.5 \times 10^7$</td>
<td>40</td>
<td>141000</td>
<td>50000</td>
<td>22000</td>
<td>31000</td>
</tr>
</tbody>
</table>

$^a$(tera-coulombs/year).

* Tungsten yield based on 0.5% N seeding.

A fully-detached divertor would largely eliminate this problem. But, erosion from first wall surfaces will remain, perhaps at $\sim 1/1000$ the rate.

Wall-eroded material will end up as ‘slag’ in the divertor. A means to continuously remove it must be devised.
Q: Why not just operate with a *conventional divertor* in a fully detached regime?

A: The ‘thermal front’ of divertor detachment is unstable; it ‘jumps’ to the X-point region leading to:

- reduced radiation in divertor volume
- reduced screening of impurities
- increased radiation in X-point region
- cooling of LCFS and pedestal region
- reduced plasma confinement
- near thermal collapse (H-L transition, density limit disruption)

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"...It thus appears that it is not possible to have [fully] detached discharges with good H-mode confinement..."
- analysis of JET H-modes: G. McCracken et al., JNM 266 (1999) 37

...2013  -- Fully detached divertor + high performance core plasma has not yet been achieved.
Q: Why not just operate with a *conventional divertor* in a fully detached regime?

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Approach for ITER:

*Fully-detached conventional divertor operation must be avoided.*

- Produce a controlled, *partial detachment* using feedback on impurity seeding and gas puffing to reduce divertor heat fluxes to tolerable levels
- **Take a hit** on core $P_{rad}$ increase, confinement degradation and $Z_{eff}$ increase *for the sake of the divertor.*
Q: Why not just operate with a *conventional divertor* in a fully detached regime?

A: The ‘thermal front’ of divertor detachment is unstable; it ‘jumps’ to the X-point region leading to

- reduced radiation in divertor volume
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Approach for DEMO?

Challenge #3: Develop robust, advanced divertor solutions that achieve full detachment with minimal impurity seeding

Open up access to enhanced core/divertor plasma regimes, inaccessible in present devices

Alternative solutions for the divertor are necessary.

A solution for the heat exhaust in the fusion power plant is needed. A reliable solution to the problem of heat exhaust is probably the main challenge towards the realisation of magnetic confinement fusion. The risk exists that the baseline strategy pursued in ITER cannot be extrapolated to a fusion power plant. Hence, in parallel to the programme in support of the baseline strategy, an aggressive programme on alternative solutions for the divertor is necessary. Some concepts are already being tested at proof-of-principle level and their technical feasibility in a fusion power plant is being assessed. Since the extrapolation from proof-of-principle devices to ITER/DEMO based on modelling alone is considered too large, a dedicated test on specifically upgraded existing facilities or on a dedicated Divertor Tokamak Test (DTT) facility will be necessary.

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Potential solutions
Advanced divertors
- increase field line length
- increase divertor radiation
- increase target ‘wetted area’
- ...

$\lambda \sim 10 \times \lambda_q$
Some advanced magnetic divertor concepts are well developed. Potential high-performers have not yet been tested.

**Snowflake (SF)**
- Operated on TCV, NSTX, DIII-D

**Cloverleaf (CL)**
- Scenario developed for TCV

**X Divertor (XD)**
- SF with XD target operated on NSTX, DIII-D

**Super X (SXD)**
- Not yet tested (MAST 2015)
- W. Vijvers APS 2013, D. Ryutov, 2013

**Triple-X (XXX)**
- Produced on TCV
- Similar to old JT-60
- J. Kesner NF 1990, W. Vijvers APS 2013

**X-point Target (XPT)**
- Not yet tested (TCV 2015?)
Snowflake divertor (and Cloverleaf)

**TCV**  F. Piras, PPCF 51 (2009) 055009.

**NSTX**  V.A. Soukhanovskii, PoP 19 (2012) 082504.

**DIII-D**  S. Allen, IAEA, 2012.
Snowflake divertor (and Cloverleaf)

**TCV**  
F. Piras, PPCF 51 (2009) 055009.

**NSTX**  

**DIII-D**  

---

**Advantages:**

*Poloidal flux expansion near x-point; increased L_||; multiple strike points; increased x-pt shear (stabilizes BMs?); lower detachment threshold (~XD target); ELM resilient? Reduces ELMs? Implement in existing tokamaks*

**Disadvantages:**

*PMI near core plasma; full detachment not demonstrated (X-pt MARFE)*
Increase flux expansion at divertor target by locating an x-point just behind it – an “X divertor”.

X divertor

Originally, the idea was proposed with modular coils...

Increase flux expansion at divertor target by locating an x-point just behind it — an “X divertor”.

... did not help to sell idea.
X divertor

Advantages:
- Poloidal flux expansion at divertor target; increased $L_{||}$;
- lower detachment threshold

Disadvantages:
- PMI near core plasma; full detachment not demonstrated (X-pt MARFE)

‘XD target’ has been produced as part of Snowflake experiments on NSTX and DIII-D

Increase flux expansion at divertor target by locating an x-point just behind it – an “X divertor”.

XD also produced on TCV

W. Vijvers APS 2013

Note:
Poloidal flux expansion using PF coils is similar to that obtained by a tilted target plate (but with no $L_{\parallel}$ increase).

For attached plasmas, field line angles must be $> \sim 1$ degree, otherwise shadowing will occur. This sets maximum poloidal flux expansion.

=> a problem with all poloidal flux-expansion ideas

Advantages:
* Poloidal flux expansion at divertor target; increased $L_{\parallel}$;
* lower detachment threshold

Disadvantages:
* PMI near core plasma; full detachment not demonstrated (X-pt MARFE)
Super X divertor – a potential solution


**Key concept:** Major radius of target plate is increased; B at target plate is decreased; heat flux density at target decreases. 2-point model yields

$$n_t \propto \left( \frac{R_t}{R_{OMP}} \right)^2 \quad T_t \propto \left( \frac{R_{OMP}}{R_t} \right)^2$$

for attached plasma. Increased $L_{\parallel}$ also helps.

Idea of accessing a fully detached, radiative divertor with SXD was not considered. => field line angle ($>1^0$) not a requirement?
Super X divertor – a potential solution

Potential Benefits:
Increased $L_{\parallel}$; reduce target $q_{\parallel}$ & $T$; increased target $n$; flux expansion at divertor target; lower detachment threshold; PMI away from core; operation with fully detached divertor plasma without core X-pt MARFE

Divertor heat flux & erosion problem solved?

Key concept: Major radius of target plate is increased; $B$ at target plate is decreased; heat flux density at target decreases. 2-point model yields

$$n_t \propto \left( \frac{R_l}{R_{OMP}} \right)^2$$
$$T_t \propto \left( \frac{R_{OMP}}{R_l} \right)^2$$

for attached plasma. Increased $L_{\parallel}$ also helps.

Idea of accessing a fully detached, radiative divertor with SXD was not considered. => field line angle ($>1^0$) not a requirement?

X-point target divertor – a potentially better solution

Concept: Use a remote X-point to produce a fully detached, radiating plasma (X-point MARFE) as a \textit{virtual target}.

Build on and improve the SXD idea:
- Increase major radius of target plate
  \[ n_t \propto \left( \frac{R_t}{R_{OMP}} \right)^2 \]
  \[ T_t \propto \left( \frac{R_{OMP}}{R_t} \right)^2 \]
- \( L_{\parallel} \) set to infinity (via target X-point adjust) on flux tube carrying peak heat flux.
- No field line shadowing at target plate; attack angles always > 1 deg
- Tight baffling for high neutral pressures
- Employ feedback control for ‘divertor MARFE’

Concept: Use a remote X-point to produce a fully detached, radiating plasma (X-point MARFE) as a virtual target.

Potential Benefits:
- Fully detached divertor over large operational space;
- Core x-point MARFE avoided; PMI away from core plasma;
- Divertor heat flux & erosion problem solved?

Build on and improve the SXD idea:
- Increase major radius of target plate:
  \[ n_t \propto \left( \frac{R_t}{R_{OMP}} \right)^2 \]
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- Employ feedback control for ‘divertor MARFE’
Advanced divertors have the potential to solve another critical problem for fusion – tritium breeding.

$^6\text{Li} + \text{n} \rightarrow \text{He} + \text{T} + 4.8 \text{ MeV}$

Tritium breeding requirements severely restrict the first wall surface area available for high heat-flux removal.

Divertor high heat-flux surface (4 cm steel)
Advanced divertors have the potential to solve another critical problem for fusion – tritium breeding requirements severely restrict the first wall surface area available for high heat-flux removal.

D.G. Whyte, APS 2012

Tritium breeding requirements severely restrict the first wall surface area available for high heat-flux removal.

Send parallel heat flux to remote chamber:
- Increase first-wall area for tritium breeding materials
- Decrease neutron load on high heat flux handling surfaces
Goal: X-pt MARFE in divertor volume, not in core plasma

Put radiating, *cold* plasma where it belongs – in the divertor!
Key to success: get control of thermal fronts (MARFEs)
Thermal fronts (a.k.a. MARFEs) accommodate a \(~\)maximum value of incident $q_\parallel$ that depends on plasma pressure, $nT_e$ and impurity fraction, $f_I$.

Example Carbon$^1$:

\[
q_\parallel \approx 30 f_I^{1/2} \frac{n T_e}{10^{20} m^{-3} eV} \left[ MW / m^2 \right] \quad \rightarrow \quad q_\parallel \approx 0.6 \ GW / m^2
\]

$^1$ I. Hutchinson, NF 34 (1994) 1337.
Key to success: get control of thermal fronts (MARFEs)

Large major radius of X-pt target provides stability/control

Upstream $q_{//}$ decreases as $\sim 1/R$ along divertor leg.
Key to success: get control of thermal fronts (MARFEs)

Large major radius of X-pt target provides stability/control

Upstream $q_{\parallel}$ decreases as $\sim 1/R$ along divertor leg.
Thermal front position is stable to perturbations $(n, T_e, f_i)$.
Key to success: get control of thermal fronts (MARFEs)

Large major radius of X-pt target provides stability/control

Upstream $q_{\parallel}$ decreases as $\sim 1/R$ along divertor leg.

Thermal front position is stable to perturbations ($n$, $T_e$, $f_I$).

Caveats: cross-field transport, neutrals also play role.

Need to optimize geometry & poloidal vs. total flux expansion.
What if advanced magnetic divertors can’t do it? Do we need to go to liquid metals?

Lithium in Capillary Porous Structure

LiMIT Concept

Advantages of Liquid Metals:
Self annealing, good transient thermal response, ...

But, liquid metals introduce a host of new issues.
Extra complexity, pumping, temperature limits, corrosion, core impurities, ...

Fig. 1. Principal scheme of the sheet (a) and jet-drop curtain (b) divertor plates; (d) MHD pump.

Mirnov JNM 1992

C. Lao, MIT 1992

Mirnov PPCF 2006

Ruzic NF 2011

Ono FED 2012
Spread heat exhaust over a large surface area in a divertor chamber by tailoring magnetic geometry and radiation/neutral interaction zone.

Message: Magnetic geometry is a powerful tool that has not yet been fully explored to tackle the divertor problem.
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Spread heat exhaust over a large surface area in a divertor chamber by tailoring magnetic geometry and radiation/neutral interaction zone

Tame the plasma-material interface with plasma physics

=> use physics knowledge to optimize the unconfined plasma
- Cold, fully detached divertor = ~ zero erosion
- Hot separatrix and pedestal regions = good core performance
Heat and particle exhaust in tokamak reactors: challenges and potential solutions

Outline:

“The tail is wagging the dog.”

Challenges for ITER
- Disruptions
- ELMs
- Power exhaust

Challenges for DEMO
- Extreme power + steady state

Potential solutions
- Advanced divertors
- Double null

Critical R&D needed
- Advanced divertor test facility
Experiments clearly show that heat and particles enter the scrape-off layer primarily on the low-field side.

**Double Null**

**ELMs do not appear in high-field SOL in double null.**

DIII-D -- Petrie NF 2003
Experiments clearly show that heat and particles enter the scrape-off layer primarily on the low-field side.

Plasma on high field SOL ‘disappears’ when magnetically disconnected from low-field side.

Alcator C-Mod -- LaBombard NF 2004
Experiments clearly show that heat and particles enter the scrape-off layer primarily on the low-field side.

Double Null Plasma on high field SOL ‘disappears’ when magnetically disconnected from low-field side.

Alcator C-Mod -- LaBombard NF 2004

High-field side SOL profiles can be controlled by tuning double-null flux balance

Energetic ion drift orbits enter low-field side scrape-off layer and can reach wall/antenna structures.

Can be a serious problem for outboard limiters and ICRF antennas.

Alcator C-Mod G-H Limiter
High-field side SOL is quiescent. Radial transport is \(~\text{zero}\). No ‘blobs’, ELMs or energetic ions.

**Fluctuation-induced radial transport is essentially zero on high field side.**

Smick NF 2013
High-field side SOL is quiescent. Radial transport is ~zero. No ‘blobs’, ELMs or energetic ions.

Fluctuation-induced radial transport is essentially zero on high field side.

Power-starved inner divertor legs are naturally detached at moderate core densities (with no x-point MARFE)

Smick NF 2013
Double null topology is an essential tool for power handling and plasma-material interaction control.

Double null operation is a ‘no-brainer.’
- Solves heat exhaust for inner divertor legs
- Creates low-PMI high-field SOL for RF actuators
- Sends heat to the 2 divertor legs that can take it

Fluctuation-induced radial transport is essentially zero on high field side.

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Advanced divertors
Double null

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Advanced divertor test facility
Four critical milestones must be achieved to demonstrate readiness for FNSF/DEMO.

1. Demonstrate robust divertor power handling solutions at DEMO boundary plasma parameters
2. Demonstrate sufficient suppression of divertor erosion at DEMO parameters, scaling to SS ($10^7$ s)
3. Achieve goals 1 and 2 while attaining reactor-relevant core plasma performance
4. Demonstrate low PMI, reactor-compatible current drive and heating technologies

Innovation and new R&D facilities are needed to reach these goals.
ADX -- a concept for a high power density, advanced divertor test facility*

**Key Elements:**

- **Demountable**, LN₂ cooled, copper TF magnet
- Vertically-elongated VV
- Advanced divertor poloidal field coil sets (top and bottom)
- High power ICRF, 8MW
- Reactor-level P/S, SOL $q_\parallel$ and plasma pressures
  => same and higher than Alcator C-Mod
- Development platform for low PMI RF actuators:
  - Inner-wall LHCD
  - Inner-wall ICRF

**Using Alcator Technology:**

- **extremely strong super-structure**
- Sliding TF joints
- Coaxial OH/PF coil feeds
- Electro-formed terminals
- PF and OH coils supported by rigid vacuum chamber
- Reactor-relevant RF heating and current drive systems

**ADX**

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Major/Minor Radius</td>
<td>0.73 / 0.2 m</td>
</tr>
<tr>
<td>Elongation</td>
<td>1.7</td>
</tr>
<tr>
<td>Magnetic Field</td>
<td>6.5 Tesla (8 Tesla)</td>
</tr>
<tr>
<td>Plasma Current</td>
<td>1.5 MA</td>
</tr>
<tr>
<td>$P_{AUX}$ (net)</td>
<td>8 MW ICRF, 2 MW LHCD</td>
</tr>
<tr>
<td>Surface Power Density</td>
<td>~ 1.5 MW/m²</td>
</tr>
<tr>
<td>SOL Parallel heat flux</td>
<td>$q_\parallel$ ~ 2 GW/m²</td>
</tr>
<tr>
<td>Advanced Divertor Concepts</td>
<td>Vertical target; Snowflake; Super-X; X-point target; Liquid metal target</td>
</tr>
<tr>
<td>Divertor and first-wall material</td>
<td>Tungsten/Molybdenum</td>
</tr>
<tr>
<td>Pulse Length</td>
<td>3s, with 1s flat-top</td>
</tr>
</tbody>
</table>

*http://burningplasma.org/web/fesac-fsff2013/whitepapers/LaBombard_B.pdf
Edge plasma pressure is set by critical values of $\beta_p$, as demonstrated by the success of the EPED1.6 model\(^1\) in simulating H-mode pedestal heights.

$$P_{\text{ped}} \sim B^2$$

Reactor-level SOL plasma pressures can be obtained only by operating at the same magnetic fields as a reactor (5 to 8 tesla).

Alcator C-Mod is the only diverted tokamak in the world that can do this.
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Reactor-level SOL pressures facilitates reactor level divertor densities...

$$P = n(T_i + T_e) ; \ T_e \sim 10 \text{ eV} \ ; \ T_i \sim T_e$$

$$\Rightarrow n \sim 10^{21} \text{ m}^{-3}$$

...which is required to access the relevant divertor and PMI regimes\textsuperscript{1}.

\textsuperscript{1} Snyder, P.B., et al., NF 51 (2011) 103016.

\textsuperscript{1} Stangeby and Leonard, NF 51 (2011) 063001.
**ADX -- high field and power density uniquely access reactor-level SOL conditions.**

<table>
<thead>
<tr>
<th></th>
<th>MAST Upgrade</th>
<th>NSTX Upgrade</th>
<th>DIII-D Upgrade</th>
<th>EAST Upgrade</th>
<th>ADX</th>
<th>ITER</th>
<th>ARIES-ACT1</th>
</tr>
</thead>
<tbody>
<tr>
<td>$B_T/I_p$</td>
<td>0.84/2</td>
<td>1/2</td>
<td>2.2/1.5</td>
<td>3.5/1.5</td>
<td>6.5/1.5</td>
<td>5.3/15</td>
<td>6/11</td>
</tr>
<tr>
<td>$a/R$</td>
<td>0.65/0.85</td>
<td>0.62/0.93</td>
<td>0.6/1.75</td>
<td>0.45/1.85</td>
<td>0.2/0.73</td>
<td>2/6.2</td>
<td>1.6/6.25</td>
</tr>
<tr>
<td>$P_{tot}^*$</td>
<td>7.5</td>
<td>16</td>
<td>39</td>
<td>36</td>
<td>14</td>
<td>150</td>
<td>405</td>
</tr>
<tr>
<td>$P_{tot}/S$</td>
<td>0.18</td>
<td>0.37</td>
<td>0.60</td>
<td>0.82</td>
<td>1.7</td>
<td>0.22</td>
<td>0.64</td>
</tr>
<tr>
<td>$P_{tot}B/R$</td>
<td>8</td>
<td>18</td>
<td>50</td>
<td>69</td>
<td>126</td>
<td>131</td>
<td>390</td>
</tr>
<tr>
<td>$\lambda_q/\lambda_q^{DX}$ (Eich)</td>
<td>4.7</td>
<td>4.2</td>
<td>3.5</td>
<td>2.0</td>
<td>1</td>
<td>1.2</td>
<td>1.3</td>
</tr>
<tr>
<td>$q_{\parallel}/q_{\parallel}^{DX}$ (Eich)</td>
<td>0.02</td>
<td>0.06</td>
<td>0.34</td>
<td>0.62</td>
<td>1</td>
<td>0.82</td>
<td>3.1</td>
</tr>
<tr>
<td>$q_{\parallel}/q_{\parallel}^{DX}$ (ped.)</td>
<td>0.03</td>
<td>0.09</td>
<td>0.40</td>
<td>0.55</td>
<td>1</td>
<td>0.10</td>
<td>0.52</td>
</tr>
</tbody>
</table>

ADX’s high field and high power density, $P_{tot}/S$ & $P_{tot}B/R$ will produce SOL $q_{\parallel}$ that exceeds all other tokamak experiments (including planned upgrades).

SOL parallel heat flux, $q_{\parallel}$ (normalized to ADX)

$q_{\parallel}$ same as ITER (and DEMO, depending on scaling)

* $P_{tot}$ is the total of all auxiliary source power. Net power depends on transmission efficiencies and operational beta limits.
With $q_\parallel$ and $B$ being the same as in a reactor, complete divertor similarity with a reactor may be obtained.\footnote{Hutchinson and Vlases, NF 36 (1996) 783.}

$q_\parallel = \text{reactor}$

$B = \text{reactor}$

=> plasma and atomic physics dimensionless parameters in the divertor ...

$$T_e, \nu^*, \rho^*, \beta, \lambda_0 / \Delta_d$$

...may be made identical by adjusting poloidal flux expansion and divertor leg length.
ADX -- a facility to develop and test advanced divertor concepts at reactor conditions

Advanced Divertor Experiment

Idea: configure internal PF coils to test the most promising magnetic geometries and divertor targets.

“Tame the plasma-material interface with plasma physics.”

- Double-null geometry
- Advanced divertors -- low-field side SOL
- Quiescent, low heat flux -- high-field SOL

Liquid metals may be considered also.
Grad-Shafranov equilibria obtained using ACCOME (Selene)\textsuperscript{1}

\begin{itemize}
  \item Idea: configure internal PF coils to test the most promising magnetic geometries and divertor targets.
  \item ‘ASDEX’
\end{itemize}

\textsuperscript{1} Tani et al., J. Comp. Phys. 98 (1992) 332.
Grad-Shafranov equilibria obtained using ACCOME (Selene)\textsuperscript{1}


Idea: configure internal PF coils to **test the most promising magnetic geometries and divertor targets.**

Vertical Target
Grad-Shafranov equilibria obtained using ACCOME (Selene)\(^1\)

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Splitter and multi-junction fabrication techniques produce compact LHCD launchers that can fit on the inside wall.

- **High B-field side**
  - $\Rightarrow$ lower $n_{//}$
  - $\Rightarrow$ penetrating rays$^1$
  - $\Rightarrow$ higher CD efficiency

- **Quiescent SOL**
  - $\Rightarrow$ Low PMI
  - $\Rightarrow$ Excellent impurity screening$^2$

High field side launch is highly favorable for LHCD, as noted in VULCAN study$^3$.

---

[1] GENRAY modeling by Syun'ichi Shiraiwa: $n_{//}=1.6$, Alcator C-Mod I-mode


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Critical Milestone: Develop robust, reactor-compatible current drive & heating techniques

(for SS burning plasma)


FW mode-conversion to IBW with efficient flow drive and heating

**Integrated vacuum vessel, designed for tests of inside-launch ICRF**

- **High B-field side**
  - Reduced energetic ion impact on antenna structures

- **Quiescent SOL**
  - Low PMI
  - Low neutral pressures
  - Excellent impurity screening

**Critical Milestone:**

- Develop robust, reactor-compatible current drive & heating techniques

---

**ADX** -- an innovation platform for low PMI, reactor-compatible RF actuators

**Critical Milestone:**

(for SS burning plasma)

---

TORIC simulation: $B = 5.4$ tesla, $f = 80$ MHz, 15% H in D, $n_\phi = -10$, 40% to electrons, 30% to H $1^{st}$ harmonic and 30% to D $2^{nd}$ harmonic

[1] TORIC modeling by Yijun Lin
Summary

MFE has entered a new era: plasma boundary & PMI now set reactor design & performance -- "the tail is wagging the dog."

ITER is confronting significant risks to its mission, identifying critical research needs in boundary & PMI.

The jump to DEMO will be extreme; the success of MFE depends on developing new plasma physics solutions for extreme power handing, material erosion and RF actuators that scale to SS conditions.

Advanced magnetic divertors, clever geometries and advanced RF actuators have potential to solve these problems – we have not yet been given the experimental freedom to explore such possibilities.
Goal: inform the conceptual development and accelerate the readiness for deployment of next step devices (FNSF/DEMO)

**Summary**

ITER

**ADX**

**Advanced Divertor Experiment**

**ADX** - uniquely able to test advanced divertor concepts at reactor-level conditions in an affordable university-scale experiment.