“Thinking outside the box”

New integrated approaches are needed to solve divertor, main chamber and steady state sustainment challenges for fusion.

Brian LaBombard, Earl Marmar, Martin Greenwald, Dennis Whyte
MIT Plasma Science and Fusion Center
Powerful new physics ‘solution’ ideas have emerged in areas of divertor, main chamber and steady state sustainment -- these have potential to solve critical challenges for fusion.

In order to exploit these ideas, new integrated reactor design approaches will be needed; some are now being proposed.

Rapidly evolving technologies (e.g., high-temperature superconductors, additive manufacturing) may make this possible; further technological advances are almost assured on a ~20 year development time frame.

Main message: fusion research should focus on potential game-changing physics-based solutions in the near term, not being overly concerned about technology constraints...
Innovative plasma physics solutions are critically needed:

• Robust divertor solutions for power exhaust handling
  New divertor concepts must be developed, and demonstrated to handle plasma exhaust densities that are an order-of-magnitude higher than current schemes.

• Robust divertor solutions for divertor target lifetime
  Divertor target erosion must be suppressed to essentially zero, while attaining excellent core plasma performance.

• Robust main chamber component lifetime solutions
  First wall components, including RF actuators, must survive the PMI onslaught of a DT reactor for sufficient time to be economically viable.

• Efficient, low PMI, heating and current drive technologies
  Achieving steady-state tokamak + net electricity production requires efficient (wall plug to plasma), low PMI RF actuator technologies that attain effective current profile control in a reactor.

... otherwise the tokamak will not be a viable concept for a steady-state, electricity-producing fusion power plant.
Challenges and Potential Solutions

• Robust divertor solutions for power exhaust handling

• Robust divertor solutions for divertor target lifetime
ITER Q_{DT} = 10;
t \sim 400 \text{ s}
Type I ELMing
H-mode

P_{IN} = 50 \text{ MW}

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

P_{SOL} \sim 100 \text{ MW}

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

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

Challenge #1: Power exhaust

ITER steady state power exhaust challenge...
...more difficult than originally planned

"Thinking outside the box ..." B. LaBombard – IAEA TM on Divertor Concepts
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**Challenge #1: Power exhaust**

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\[ P_{\text{IN}} = 50 \text{ MW} \]

\[ P_{\text{FUS}} = 500 \text{ MW} \]

\[ P_\alpha = 100 \text{ MW} \]

\[ P_{\text{RAD}} = 50 \text{ MW} \]

\[ P_{\text{SOL}} \approx 100 \text{ MW} \]

\[ \lambda \sim 10 \times \lambda_q \]

For \( \lambda_q = 5 \text{ mm} \),

\[ q_{\perp,\text{max}} \approx 40 \text{ MWm}^{-2} \]

 Require \( f_{\text{rad,div}} \approx 0.7 \rightarrow \text{partial divertor detachment} \) to reduce \( q_{\perp,\text{max}} \) to \( \approx 10 \text{ MWm}^{-2} \)
Challenge #1: Power exhaust

ITER steady state power exhaust challenge... ...more difficult than originally planned

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)
Challenge #1: Power exhaust

New Result: \( \sim P_{\text{SOL}} B/R \) is scale parameter for \( q_{\parallel} \) into divertor

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

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

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

Increase \( P_{\text{rad,core}} \) ?
Increase \( P_{\text{rad,div}} \) ?

ITER

Yet, must avoid full detachment...

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

"Thinking outside the box ..." B. LaBombard – IAEA TM on Divertor Concepts
**Challenge #1: Power exhaust**

**New Result:** \( \sim P_{SOL} B/R \) is scale parameter for \( q_{\parallel} \) into divertor

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

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

\[ \lambda_q = 3.6 \text{ mm} \]

Tolerable
\[ P_{FUS} = ? \text{ MW} \]
\[ P_\alpha = ? \text{ MW} \]

\[ Q_{DT} = 10 \]

Contours are normalized div. neutral pressure, \( \mu \)

Challenge #1: Power exhaust

New Result: $\sim P_{SOL} B/R$ is scale parameter for $q_\parallel$ into divertor

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

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

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

$\lambda_q = 3.6 \, mm$
$\lambda_q = 1.2 \, mm$

$Q_{DT} = 10$

$Q_{DT} = 5.5?$

Contours are normalized div. neutral pressure, $\mu$

Note: Intrinsic carbon only; impurity seeding not optimized ...

“Thinking outside the box ...” B. LaBombard – IAEA TM on Divertor Concepts

$q_{\parallel}$ can be reduced by increasing core radiation to lower $P_{SOL}$... but at the penalty of reducing core confinement

Good confinement requires $P_{SOL} > \sim P_{LH}$ (LH power threshold)

Loarte, PoP 18 (2011) 056105
Good confinement requires \( P_{SOL} > \sim P_{LH} \) (LH power threshold)

Experiments are presently seeking a mix of core & divertor radiation to demonstrate a power-handling solution for ITER’s W divertor.

### Table

<table>
<thead>
<tr>
<th></th>
<th>( f_{\text{rad,core}} )</th>
<th>( P_{SOL} B/R )</th>
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</thead>
<tbody>
<tr>
<td><strong>ITER, ( Q_{DT}=10 )</strong></td>
<td>33%</td>
<td>90 MW-T/m</td>
</tr>
<tr>
<td>with ( P_{SOL}=P_{LH} )</td>
<td>66%</td>
<td>45 MW-T/m</td>
</tr>
</tbody>
</table>

**Divertor demonstration experiments** (\( H_{98} > 1 \), \( q_{\text{target}} < 5 \text{ MW/m}^2 \), ITER-like div.)

- **AUG:** \( P_{\text{sol}} B/R = 25 \) \( \text{Kallenbach, NF 55 (2015) 053026} \)
- **C-Mod:** \( P_{\text{sol}} B/R = 25 \) \( \text{Loarte, PoP 18 (2011) 056105} \)

Max demonstrated \( P_{SOL} B/R \) is only \( \sim 1/2 \) of ITER \( P_{LH} B/R \)

Not yet known what performance will be attained in ITER, compatible with div.

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Requirements for a PILOT/DEMO reactor are much more extreme than ITER.

**C-Mod:** $P_{\text{sol}} B/R = 25$

**AUG:** $P_{\text{sol}} B/R = 25$

**ACT1, ACT2 design studies:** [http://aries.ucsd.edu/ARIES/DOCS/bib.shtml](http://aries.ucsd.edu/ARIES/DOCS/bib.shtml)

**ARC design study:** Sorbom, et al., Fusion Eng. Des. (2015), [http://dx.doi.org/10.1016/j.fusengdes.2015.07.008](http://dx.doi.org/10.1016/j.fusengdes.2015.07.008)
Requirements for a PILOT/DEMO reactor are much more extreme than ITER.

**DEMO:** Extreme values of $P_{\text{SOL}} B/R$

ACT1, ACT2 from ARIES design studies  
http://aries.ucsd.edu/ARIES/DOCS/bib.shtml

Power entering SOL should be reduced (but may not be possible)

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<th>$P_{\text{SOL}} B/R$</th>
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<tbody>
<tr>
<td><strong>ACT1</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>with $P_{\text{SOL}} = P_{\text{LH}}$</td>
<td>85%</td>
<td>57</td>
</tr>
<tr>
<td><strong>ACT2</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>with $P_{\text{SOL}} = P_{\text{LH}}$</td>
<td>80%</td>
<td>114</td>
</tr>
</tbody>
</table>

Power handling of divertor must be improved by factor of 4 to 10 for a DEMO.
Challenge #2: Material erosion
~complete suppression of material erosion and damage

Erosion
- Gross sputtering yield on tungsten divertor target must be suppressed to less than $10^{-6}$ to achieve < 1mm/year erosion rate

D.G. Whyte, APS 2012; Stangeby, NF 51 (2011) 063001

![Graph showing W-sputtering yield vs $T_e$]  
Gamma_w/Gamma_i,\perp vs $T_{e,div}$ measured in AUG

Gamma_w/Gamma_i,\perp vs $T_{e,div}$ measured in AUG

suppress sputtering: $T_e < 5$ eV

K. Krieger, JNM 266-269 (1999) 207
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- Prompt redeposition does not help because redep. material is mixed ~ poor quality
  Doerner NF 2012: redep. sputters at x10 rate of bulk (Be)

 suppress sputtering: $T_e < 5$ eV

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$\Gamma_w/\Gamma_{i,\perp}$ vs $T_{e,\text{div}}$
measured in AUG

Gross sputtering yield on tungsten divertor target must be suppressed to less than $10^{-6}$ to achieve < 1mm/year erosion rate.

Prompt redeposition does not help because redeposited material is mixed ~ poor quality.

Doerner NF 2012: redeposited sputters at x10 rate of bulk (Be) compared to bulk material.

Suppress sputtering by ensuring $T_e < 5$ eV to achieve the required erosion rate.
**Erosion**
- Gross sputtering yield on tungsten divertor target must be suppressed to less than $10^{-6}$ to achieve < 1mm/year erosion rate
  
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**Damage**
- Helium implantation ($E_{\text{He}^+} > 20$ eV)

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

- Fraction of net yield $\Gamma_W/\Gamma_{i,\perp} < 5$ eV (with impurity ions)

**G. Wright, NF 52 (2012) 042003**

- Tungsten nanotendrils in C-Mod; formation at $T_{\text{surf}}>1000$ K

- To avoid “fuzz”: $E_{\text{He}^+}<20$ eV, $T_e<7$ eV
Challenge #2: Material erosion

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**Damage**
- Helium implantation ($E_{He^+} > 20$ eV)

**Solutions?**
- Ion impact energies below sputtering and damage thresholds – cold (‘fully detached’) divertor plasma required
  
  or

- Liquid metal targets?

Advanced materials alone will not solve this problem.

“Thinking outside the box ...” B. LaBombard – IAEA TM on Divertor Concepts
The ultimate goal is to find a way to operate with a **fully detached divertor** while maintaining excellent confinement.

**Conventional divertor:**
Detachment front tends to intrude into X-point (‘X-point MARFE’) 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)

... pedestal/core performance degradation from reduction in $P_{SOL}$.

Evolution to a **“pronounced detachment H-mode”** in AUG with $N_2 + Ar$ seeding

$\Rightarrow$ X-point MARFE, $H_{98} \sim 0.8$

Kalenbach, NF 55 (2015) 053026

“Thinking outside the box ...” B. LaBombard – IAEA TM on Divertor Concepts
Advanced magnetic divertors offer potential solutions.

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

Super X\(^1\) and X-point Target\(^2\)

Locate targets at large major radius relative to primary x-point

Exploit **total** flux expansion

\(B\) at target plate decreased \(\sim 1/R\); 
\(q_\parallel\) at target is decreased \(\sim 1/R\); 
Long field line length; detachment at reduced core densities

Detachment front (MARFE) may be robustly ‘stabilized, staying in divertor volume.

\(q_\parallel\) decreases as \(\sim 1/R\) along outer divertor leg: \(q_\parallel > q_\parallel\).

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Thermal fronts (a.k.a. MARFEs) accommodate a \(~\text{maximum value of}\) incident \(q_{//}\) that depends on plasma pressure, \(nT_e\) and impurity fraction, \(f_i\)

Example Carbon\(^1\):

\[q_{//} \approx 30 f_i^{1/2} \frac{nT_e}{10^{20} m^{-3} eV} [\text{MW} / m^2] \implies q_{//} \approx 0.6 \text{ GW} / m^2\]

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X-point Target Divertor
- knock out peak $q_{\parallel}$ with an X-point as a ‘virtual target’
- can be a ‘snowflake target’, promoting ‘churning’ mode.$^3$

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- knock out peak \(q_{\parallel}\) with an X-point as a ‘virtual target’
- can be a ‘snowflake target’, promoting ‘churning’ mode.\(^3\)

- Cold, fully detached divertor = \(\sim\) zero erosion
- Hot separatrix and pedestalal regions = good core performance

X-point Target Divertor concept -- Initial modeling with UEDGE shows promising results

Potential test of X-point Target Divertor concept

- Stable, fully detached outer divertor leg over a wide range of $P_{\text{SOL}}$
- $T_e$ at LCFS (including x-point) $\sim$ 150 eV
- No core plasma X-point MARFE

[1] M. Umansky et al., this workshop

“Thinking outside the box ...” B. LaBombard – IAEA TM on Divertor Concepts
Boundary/divertor understanding is an empirical science at best. IMHO: Reliable first-principles models of boundary/divertor transport & turbulence will not be forthcoming anytime soon.
Q: What is really required to qualify a divertor for DEMO?

Present models for boundary/divertor plasma are extremely crude. Model/code extrapolation to untested regimes is not reliable.

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IMHO: Reliable first-principles models of boundary/divertor transport & turbulence will not be forthcoming anytime soon.

High speed D\textsubscript{\textalpha} camera image of divertor fluctuations on MAST\textsuperscript{1}


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**IMHO:** Reliable first-principles models of boundary/divertor transport & turbulence will not be forthcoming anytime soon.

**Answer: Experimental Qualification**

A1: Divertor conditions ~identical to DEMO in terms of plasma & atomic physics dimensionless parameters

A2: Plasma-surface conditions ~identical to DEMO in terms of plasma & atomic physics dim-less parameters

A3: Surface heat/particle fluxes ~identical to FNSF/DEMO in absolute magnitude of flux densities

ITER will be an important test for the conventional divertor.

A **“Divertor Test Tokamak”** is needed to qualify advanced divertor schemes at relevant plasma conditions.
Desirable attributes of a Divertor Test Tokamak:

Produce DEMO-like $q_\parallel$, $nT$ and $B$ in the actual divertor geometry being considered.

This is the primary motivation for the ADX facility.

World Tokamaks

World Tokamaks in $(q_\parallel, B)$ space

Reactor-like divertor regimes are already being accessed by C-Mod -- high-field, compact, high power density.

A similarly constructed ADX would be cost-effective platform for an divertor test tokamak.

“Thinking outside the box ...” B. LaBombard – IAEA TM on Divertor Concepts
Challenges and Potential Solutions

• Robust main chamber component lifetime solutions

• Efficient, low PMI, heating and current drive technologies
Efficient, low PMI, **RF current drive and heating technologies** must be demonstrated that project to effective current profile control; **otherwise the tokamak is not a viable concept for fusion electricity.**

"The auxiliary systems typically used in current experiments, while extremely useful tools, are not generally suitable for a reactor. **RF schemes are the most likely systems to be used and will require significant research to achieve the levels of reliability and predictability that are required**"

2007 US DoE FESAC Report

Priorities, Gaps and Opportunities:
Towards A Long-Range Strategic Plan For Magnetic Fusion Energy

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**Issues for Neutral Beams**

- Neutron streaming, extension of tritium boundary, reduction in TBR
- Large complex systems, limited lifetime, low system efficiency
- Standard $E_{\text{beam}} \sim 100 \text{ keV} \Rightarrow \sim 0.1\text{m penetration at } 10^{20} \text{ m}^{-3}$

Reactor studies (e.g., the ARIES series) recognize that neutral beams are not suitable for reactors.
Efficient, low PMI, **RF current drive and heating technologies** must be demonstrated that project to effective current profile control; otherwise the tokamak is not a viable concept for fusion electricity.

“The auxiliary systems typically used in current experiments, while extremely useful tools, are not generally suitable for a reactor. **RF schemes are the most likely systems to be used and will require significant research to achieve the levels of reliability and predictability that are required**”

**Requirements and Challenges for RF Systems**
- Low plasma-wall interaction, SS operation (< 1 mm/yr erosion)
- Good wave coupling, high system efficiency (wall-plug to plasma)
- Effective tool for current profile control (ITBs, $H_{98}$ enhancement)
- Applicable to reactor environment (neutrons, T breeding)

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**Thinking outside the box -- physics-based solutions:**
- Exploit ‘quiescent SOL’ on high-field-side (HFS) to control PMI
- Employ near double-null plasmas to control power exhaust
- Employ HFS RF launch: Excellent wave physics, maximum CD efficiency, improved neutronics, TBR ...
Double null topology: essential tool for power handling and plasma-material interaction control.

High-field side (HFS) SOL is quiescent. Radial transport is ~zero. No ‘blobs’ or ELMs [1].

Fluctuation-induced radial transport is essentially zero on HFS [2].

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No energetic ion losses on HFS [3].

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

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

Advantage:
- Mitigates heat exhaust for inner divertor legs
- Sends heat to the 2 divertor legs that can take it
- Creates low-PMI high-field SOL for RF actuators

High-Field Side RF launch structures can be readily integrated into Reactor Designs

Compact passive-active multi-junction LHCD launchers are robust designs that can fit within the blanket structure on the high-field side.

**HFS LHCD on ADX**

High-Field Side physics is extremely favorable for reduced PMI

- Quiescent SOL; no ‘blobs’ (wave scattering)
- No ELMs, runaway $e^-$, energetic ion orbit loss
- Low neutral pressure – increased RF voltage
- RF-generated fast $e^-$ drift away from launcher

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**HFS LHCD on ADX**

- 10 cm

**Advanced Divertor and RF tokamak Experiment**

**High Field Side** $n$, $T_e$ profiles are very sharp – allows small gap; precise control of local conditions; reduction in parasitic losses (PDI, collisional damping)

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HFS launch is a potential game-changer for LHCD wave physics – dramatic improvements in accessibility, efficiency, current profile

Example: GENRAY/CQL3D simulations [1,2] for FDF [3]

- Accessibility & damping determine “access window” for LH wave penetration
- Higher |B| on HFS improves accessibility to lower n∥ waves
- HFS launch produces dramatic improvement in wave penetration, off-axis CD – needed for AT control
- LHCD efficiency increases by 40% -- critical for reactor concepts (ARC)

HFS launch is potential game-changer for RF-generated impurities – a factor of ~10 reduction in core for same impurity source strength.

Impurities injected on high-field side have remarkably low penetration factors than low-field side – even lower than impurities injected into the divertor [1].

Impurity Penetration Factor = \[ \frac{\text{Core Impurity Ions}}{\text{Local Impurity Injection Rate}} \]

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**Order of magnitude reduction in core impurity concentration is possible with HFS launch**, even with similar local impurity source rate.

- Argues for placing all close-fitting wall surface on HFS – *eliminate LFS sources*
- Another game-changer: may eliminate need for low-Z coatings (boronization)

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\begin{equation*}
\text{Impurity Penetration Factor} = \frac{\text{Core Impurity Ions}}{\text{Local Impurity Injection Rate}}
\end{equation*}

Interchange turbulence (‘blobs”) may be mechanism for rapid inward impurity transport on LFS \cite{2}.

Secret for successful boronization on C-Mod: \textit{cover high-Z components on the low-field side} \cite{3}.

Latest results on C-Mod find low HFS penetration factors persist in Balanced Double Null plasmas (< ½ LFS) \cite{4}.

- \textbf{Order of magnitude reduction in core impurity concentration is possible with HFS launch}, even with similar local impurity source rate.
- Argues for placing all close-fitting wall surface on HFS – \textit{eliminate LFS sources}
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\cite{1} McCracken PoP 1997; \cite{2} Krasheninnikov EPS 2002; \cite{3} Lipschultz PoP 200; \cite{4} LaBombard, 2015.

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HFS launch is highly beneficial for ICRF wave physics [1] – **efficient IBW/ICW mode conversion; no ion tails; poloidal flow drive**

HFS ICRF on ADX

- Incident fast wave (FW) power is absorbed nearly 100% via mode conversion (IBW/ICW)
- No formation of energetic ion tails
- No fast-ion loss, destabilization of energetic particle modes (fast alphas)
- IBW mode conversion has been found to produce flow drive in C-Mod[2] and TFTR[3]

- High single pass absorption of FW is also important for suppressing RF-enhanced sputtering on first-wall surfaces – a process involving FW to SW conversion and plasma-sheath potential enhancement [4].


“Thinking outside the box ...” B. LaBombard – IAEA TM on Divertor Concepts
Neutron power load distribution favors placement of RF launchers on the High-Field Side, off-midplane

Simulated ITER neutron flux density [1]

- Neutron flux density is near minimum on HFS, off midplane

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- Neutron flux density is near minimum on HFS, off midplane.
  Corresponds to ideal launch locations for HFS LHCD.
- Off mid plane neutron HFS flux is <75% of mid plane – unlikely to require additional shielding for waveguides and therefore no impact on reactor size.

Tritium Breeding considerations favor placement of RF launchers on the High-Field Side, off-midplane


Rapidly Evolving, ‘Disruptive’ Technologies ...

• High Field, High Temperature Superconductors

• Additive Manufacturing (3D printing) of structural (e.g. Inconel) and refractory (e.g., tungsten) materials

... are enabling exciting new reactor design approaches.

These may allow us to fully exploit plasma physics solutions that are critically needed for fusion.
... A high-field, compact fusion pilot plant may now be possible, due to commercial development of high-field, high-temperature superconductors.

ITER
500 MW fusion
Q ~ 10
No electricity
B = 5.3 T, R = 6.2 m

White papers submitted to FESAC Strategic Planning Panel (2014):
https://www.burningplasma.org/activities/?article=FSSP%20Priorities%20and%20Initiatives%20Whitepapers
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[1] J. Minervini, "Superconducting magnets research for a viable US fusion program"
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---

**ITER**

- 500 MW fusion
- $Q \sim 10$
- No electricity
- $B = 5.3 \, \text{T}, \, R = 6.2 \, \text{m}$

**ARC**

- 500 MW fusion
- $Q \sim 14$
- ~200 MWe
- $B = 9.2 \, \text{T}, \, R = 3.3 \, \text{m}$

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White papers submitted to FESAC Strategic Planning Panel (2014):
https://www.burningplasma.org/activities/?article=FSSP%20Priorities%20and%20Initiatives%20Whitepapers

[2] D. Whyte, “Exploiting high magnetic fields from new superconductors will provide a faster and more attractive fusion development path”
New Vision: a high-field pathway to fusion electricity

Doubling B enhances core performance by order of magnitude

\[ nT\tau_e \sim \frac{\beta_N H}{q_*^2} R^{1.3} B^3 \]

\[ \frac{P_{\text{fusion}}}{S_{\text{wall}}} \sim \frac{\beta_N^2 \varepsilon^2}{q_*^2} R B^4 \]

High field allows operation well within established limits.

[1] D. Whyte, “Exploiting high magnetic fields from new superconductors will provide a faster and more attractive fusion development path”
ARC Design Study [1] ~ Key Enabling Technology: High-field, demountable superconducting magnets

- Small size Pilot Plant (~ 1/10th volume of ITER), reduced cost
- Removable, on-piece vessel, enabling a dual FNSF/PILOT mission - lifetime of in-vessel components (including embedded internal coils) need only be ~1.5 years
- Immersion blanket for heat removal & tritium breeding


“Thinking outside the box ...” B. LaBombard – IAEA TM on Divertor Concepts
Replaceable [non-superconducting] coils “built in” to vacuum vessel could be used for advanced divertor configurations.

<table>
<thead>
<tr>
<th>System</th>
<th>Power Requirement</th>
</tr>
</thead>
<tbody>
<tr>
<td>TF cooling</td>
<td>0.57 MW</td>
</tr>
<tr>
<td>TF joint losses</td>
<td>9.9 kW</td>
</tr>
<tr>
<td>AUX coil cooling</td>
<td>~0</td>
</tr>
<tr>
<td>AUX resistive losses</td>
<td>2.5 MW</td>
</tr>
<tr>
<td><strong>Total</strong></td>
<td><strong>3.07 MW</strong></td>
</tr>
</tbody>
</table>

Single-turn Cu control coils in ARC design

“Thinking outside the box ...” B. LaBombard – IAEA TM on Divertor Concepts
Additive Manufacturing (3D printing) will transform the design and construction of Fusion Power Plants.

- 3D printing of high-strength super alloys (e.g., inconel 625) is now routine [1]

- Custom 3D printing of tungsten is now a commercial product [2]

Unlimited potential for fusion:
- Target plate & first wall with embedded coolant channels
- Tungsten antennas & waveguides
- 3D coil forms
- 3D print the entire vacuum vessel
- ...

Bottom Line – think outside the box.
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- Test and demonstrate *plasma physics based solutions* that can meet engineering specifications for divertor, main chamber and steady state sustainment without worrying so much about other technology constraints.

If we can't find solutions to meet power density, PMI & sustainment challenges, we don't have to worry about other issues.
Advanced divertors are needed. A Divertor Test Tokamak is needed. Example innovation: Super-X and X-point Target divertors

Advanced divertors must be evaluated against five essential metrics:

1. Handle DEMO-level $P_{\text{sol}}$ B/R (4 – 10 x present) while obtaining:
   2. Peak $q_{\text{surface}} < 5 \text{ MW/m}^2$
   3. Acceptable core confinement (e.g., $H_{98} > 1.x$)
   4. Acceptable divertor target lifetime (< 1 mm/yr)
   5. Acceptable helium pumping

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Main-chamber PMI + steady state sustainment solutions needed. 
Example innovation: DN topology and inside-launch RF systems 
Essential Metrics: Acceptable component lifetime ($\sim 1.5 \text{ yrs}$), reliable, efficient current drive and heating
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- Explore innovative reactor designs using the latest technologies

“Thinking outside the box ...” B. LaBombard – IAEA TM on Divertor Concepts
End
New Vision: a high-field pathway to fusion electricity

Goal: Pilot plant based on ARC idea

Achieved core plasma performance is ~good enough!

Focus is now on critical R&D for support systems:

• High-field, demountable superconducting magnets
• Removable vessel, enabling a dual FNSF/PILOT mission
• Immersion blanket for heat removal & tritium breeding
• Plasma power exhaust and material erosion solutions
• RF-based current drive and heating solutions
• Integrated performance: reactor power density, reactor-relevant regimes, reactor-relevant actuators

[1] D. Whyte, "Exploiting high magnetic fields from new superconductors will provide a faster and more attractive fusion development path"

“Thinking outside the box ...” B. LaBombard – IAEA TM on Divertor Concepts
Important consequence of $\lambda_q \sim 1/B_\theta$ scaling:
A small, high-field tokamak can be used to match divertor conditions ~identical to a reactor -- a ‘divertor identity experiment’

$\lambda_q$ independent of machine size

```
<table>
<thead>
<tr>
<th></th>
<th>Reactor</th>
<th>Test Tokamak</th>
</tr>
</thead>
<tbody>
<tr>
<td>$\lambda_q$</td>
<td>$\lambda_q$</td>
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</tr>
<tr>
<td>$q_{\parallel}$</td>
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Major Radius
X-point Target Divertor concept --
Initial modeling with UEDGE shows very promising results

Potential test of X-point Target Divertor concept

[1] M. Umansky et al., this workshop
X-point Target Divertor concept --
Initial modeling with UEDGE shows very promising results\(^1\)

**ADX**

New UEDGE capability -- secondary X-point in divertor leg

**Potential test of X-point Target Divertor concept**

**Preliminary results (ADX test case):**

- Stable, fully detached outer divertor leg over a wide range of \(P_{\text{SOL}}\)
- \(T_e\) at LCFS (including x-point) \(\sim 150\) eV
- No core plasma X-point MARFE

[1] M. Umansky *et al.*, this workshop

"Thinking outside the box ..." B. LaBombard – IAEA TM on Divertor Concepts