Divertor issues and magnetic geometry on FNSF

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Why consider novel magnetic divertor geometries

- To allow operation of FNSF and DEMO (both with high P/R) accounting for
  - SOL width uncertainties
  - Enable operation with higher plasma current (both driven and bootstrap)
  - Attaining higher burn fraction
  - Enhanced disruption resilience
  - Allow more conservative operation given huge uncertainties in PMI
  - Allow better helium exhaust

- A single PF coil set can create divertor geometries for X-divertor, snowflake and Super-X
  - Downsides of unsatisfactory divertor operation are so large that all geometries should be considered and tested
Recent SOL width scaling experiments

- Strong inverse dependence on plasma current $\sim 1 / I_p$
- FNSF (CTF) will be about the same size as present experiments, but will have current many times higher
- This is not accounted for in codes like SOLPS, UEDGE, where the $\chi$ does not reflect this current scaling
- SOL width could potentially be several times less than present experiments, while SOL power is several times more
- Must allow for advanced divertor geometries to handle this challenge
- In addition, we’d also like the capability operate at a low edge density-an additional challenge to divertors- because…
DEMO- (and FSNF) - more current at low edge density

Due to better current drive at low density, the 2007 ITER Physics basis says:

“The fusion gain in steady state maximizes at low density for constant $\beta_N$. The limitation on reducing the density in next generation tokamaks is set by the impact on the divertor.”

- Beta optimization studies (e.g. ARIES)-bootstrap current increases with density peaking-hence, hence so does $\beta$

Beta vs. density peaking for ARIES-AT like profile at constant $\beta_N$

In steady state, lower edge density leads to higher current- and higher fusion performance- constrained by the divertor
What are the geometries?

- Historical order:
  
  **X-Divertor**
  
  **Snowflake**
  
  **Super-X**
  
  Easier coils, but plate next to plasma
  
  Most complex, but the largest advantages
Properties of novel geometries

• X-divertor and snowflake:
  – Flux expansion to increase wetted area and reduce heat flux
  – Increased line length and radiating volume
  – Does not reduce $q_{\parallel}$, so no direct reduction in plasma plate temperature due to plasma physics - but 2D codes find some reduction from atomic physics

• Super-X adds has as much of these elements (flux expansion), **PLUS**
  – Further increases wetted area and radiating volume
  – Flux tube exhausting heat expands, geometrically reducing $q_{\parallel}$ - **unique to super-X**
  – Reduced $q_{\parallel}$ lowers plasma temperature at plate due to plasma sheath physics, which
    • Increases radiation
    • Reduces sputtering, erosion
    • Allows access to partially detached regime at higher power into SOL
  – No gas puffing is not needed to reduce temperature, so burn fraction is increased
  – Increases plasma neutral density, which increases helium exhaust
My comparison a nutshell

• If divertor problem is only moderately worse than can be handled with a standard divertor, additional flux expansion alone (X-divertor or snowflake) may suffice

• The Super-X adds an additional level of advantage, as well as qualitatively new advantages of physical divertor isolation (the long divertor neck)
  – Strongly reduced neutron damage - a factor of ~6 or more in both dpa and He production
    • This might be crucial in enabling divertor materials at high heat flux to survive with fusion neutrons
  – Potential to isolate full detachment front from the main plasma
  – Potential to isolate high temperature liquid metal evaporation from main plasma
    • Advantages of liquid metals - transient resistance, replaceable surface, liquid condensates (no dust) might become more possible for high temperature, high heat flux components
ST-CTF: Super-X Divertor (SXD) provides the desired operation - unlike the standard divertor

- ST-CTF case with $n_s = 3 \times 10^{19}$, $P_{SOL} = 50$ MW, no gas puffing
- SXD- exhausted plasma is “partially detached”- what ITER design aims for
  - $T < 10-20$ eV
- Standard divertor - strongly in the sheath limited regime
  - $T \sim 100$-150 eV

Electron Temperatures for SD and SXD

Calculations by John Canik  ORNL
Super-X Divertor (SXD) provides the desired operation- including potential for full detachment

Needed: calculations comparing the SXD, snowflake, XD and standard divertor for the same parameters, oir heat flux, plasma temperature and helium exhaust

Calculations by John Canik  ORNL
SXD-from theory to experiment

• Worldwide plans in motion to test SXD:

  – MAST upgrade now includes SXD, now funded
  – Long-pulse superconducting tokamak SST in India designing SXD
  – NSTX: XD and future SXD- especially interesting with Lithium PFCs (lower edge density and smaller SOL width) and high heating/CD power
It appears that a single coil set can produce all geometries by varying currents.
My conclusions

- Given the enormous uncertainties in
  - SOL physics
  - PMI physics
  - material properties under severe neutron bombardment
  - material properties under severe plasma bombardment
- and considering the cost of an FNSF (failure is not an option),
- and the enormous stresses on the divertor plus the need to ultimately attain very long Mean Time Between Failure
- FNSF designs must include provisions for advanced divertor geometries, including the SXD
Back-ups
UEDGE results for 150 MW into SOL for ITER geometry

- Divertor Radiation nearly saturates with increasing impurities
- Core radiation would continue to increase linearly with $Z_{\text{eff}}$
- No amount of impurities (up to $Z_{\text{eff}}=6$) brings divertor heat load below 10 MW/m$^2$
  - Depending on profiles, core heating is totally radiated at $Z_{\text{eff}} \sim 3-4$

![](image1.png)
Steady state devices beyond ITER - power handling appears more challenging than ITER

<table>
<thead>
<tr>
<th>Device</th>
<th>Heating Power (MW)</th>
<th>$R_{\text{major}}$ (m)</th>
<th>P/R (MW/m)</th>
</tr>
</thead>
<tbody>
<tr>
<td>ITER</td>
<td>120</td>
<td>6.2</td>
<td>19</td>
</tr>
<tr>
<td>NHTX</td>
<td>40</td>
<td>1</td>
<td>~40</td>
</tr>
<tr>
<td>ST-CTF (III)</td>
<td>~ 50</td>
<td>1.2</td>
<td>~40</td>
</tr>
<tr>
<td>FDF</td>
<td>~100</td>
<td>2.5</td>
<td>~40</td>
</tr>
<tr>
<td>ARIES-AT, ARIES-RS</td>
<td>390-510</td>
<td>5.2-5.4</td>
<td>74-93</td>
</tr>
<tr>
<td>PPCS C,D</td>
<td>571-792</td>
<td>6.1-7.5</td>
<td>94-106</td>
</tr>
<tr>
<td>SLIM CS, CREST</td>
<td>645-691</td>
<td>5.4-5.5</td>
<td>117-128</td>
</tr>
</tbody>
</table>
He exhaust

- **ITER simulations:** \( n_{\text{He}} \sim P_{\text{SOL}}^{3.3} n_s^{-5.5} \)
  - He exhaust degrades very rapidly with increasing plate T/ lower plate n
  - We worry that tilt, flux expansion could degrade He exhaust, since recycled neutral He from plate must transport a longer distance through the plasma without being ionized to be pumped out

- **SXD advantages:**
  - Large reduction in \( T_{\text{plate}} \)/increase in \( n_{\text{plate}} \) to conditions which give good He exhaust
  - For a the same wetted area, increasing R by 2-3 decreases the width on the plate by 2-3 and so reduces the distance He must go to get to the private region
Simultaneous divertor requirements

1. Low enough plate heat flux within thermohydraulic limits
2. Low enough plasma temperature at wall
   - Low erosion (dust generation, material re-deposition/tritium retention, long lifetime, etc.)
   - Low impurity contamination of the plasma
   - High divertor radiation to ameliorate heat flux
3. High divertor neutral pressure
   - Acceptable helium pumpeding
   - Acceptable burn fraction for tritium processing
4. All consistent with other crucial operations
   - High core plasma performance
   - Without excessive fueling requirements for tritium processing system

Even if point 1 is satisfactory, the others may well not be
**Divertor: a problem even for ITER**

2007 ITER Physics basis:

“It should be noted that presently developed advanced scenarios have not yet provided fully integrated scenarios and several issues remain to be solved, such as edge compatibility with the divertor”

Also:

“The fusion gain in steady state maximizes at low density for constant $\beta_N$

\[ \frac{\langle p \rangle}{\langle B^2 \rangle} \cdot \frac{I}{aB} \]. The limitation on reducing the density in next generation tokamaks is *set by the impact on the divertor.*”

*Reducing density increases plasma current drive efficiency which increases beta and thus increases fusion power*