Spherical Torus Fusion Contributions and Game-Changing Issues

Spherical Torus (ST) research contributes to advancing fusion, and leverages on several game-changing issues

1) What is ST?
2) How does research in ST contribute?
3) Game-changing issues

Martin Peng
GCEP Fusion Energy Workshop
Opportunities for Fundamental Breakthrough and Research in Fusion
May 1-2, 2006, Princeton, NJ
Minimizing Tokamak Aspect Ratio Maximizes Field Line Length in Good Curvature

Inboard similar to Tokamak; outboard closer to CT.
World Spherical Torus Research Has Been Expanding – 22 Experiments Operating or Being Built
Record High $\beta_T$ (~40%) was Achieved by START (U.K.) in 1998 and Confirmed Recently by NSTX

(Courtesy of A. Sykes & START Team, U.K.)
New Coils Improved Plasma Shaping, Stability, and Duration

⇒ Enabled record pulse length and normalized performance projected for Component Test Facility (CTF)

![Diagram showing previous and new coil configurations with a graph illustrating normalized performance vs. energy replacement times.](image)
NSTX Operates, and ITER Will Operate with a Large, Super-Alfvénic, Fast Ion Population

- ITER in new, small $\rho^*$ regime for fast ion transport
  - $k_{\perp}\rho \approx 1$, "short" wavelength Alfvén modes
  - fast ion transport from interaction of many modes

- NSTX also routinely operates with super-Alfvénic fast ions;
  - Although $\rho^*$ is large, can study multi-mode transport
  - Only machine capable of measuring q profile at large $v_{\text{fast}} / v_{\text{Alfvén}}$

DOE Plan: Complete First Round of Testing in a Component Test Facility … (2025)

Dr. Ray Orbach, DOE Director of Science: After 10 years of operation (2014 to 2024), and, in parallel, operation of materials test facility(ies) we will have the confidence, as well as the physics and technical basis to design a demonstration power plant based on fusion.
CTF Bridges Gaps between ITER and Demo in T Self-Sufficiency, Neutron Fluence & Divertor Heat Flux

[Abdou et al, Fusion Technology, 29 (1999) 1.]

<table>
<thead>
<tr>
<th>Fusion Power Goals and Conditions?</th>
<th>ITER</th>
<th>CTF Goals</th>
<th>Demo</th>
</tr>
</thead>
<tbody>
<tr>
<td>Tritium self-sufficiency goal (%)</td>
<td></td>
<td>&gt;80</td>
<td>&gt;100</td>
</tr>
<tr>
<td>Sustained fusion burn duration (s)</td>
<td>~10^3</td>
<td>&gt;10^{6-7}</td>
<td>~10^{7-8}</td>
</tr>
<tr>
<td>Total 14-MeV neutron fluence (MW-yr/m^2)</td>
<td>~0.3</td>
<td>&gt;6</td>
<td>6–20</td>
</tr>
<tr>
<td>14-MeV neutron flux on wall (MW/m^2)</td>
<td>~0.8</td>
<td>2</td>
<td>3–4</td>
</tr>
<tr>
<td>Divertor heat flux challenge, P/R (MW/m)</td>
<td>24</td>
<td>67</td>
<td>100</td>
</tr>
<tr>
<td>Expected fusion power (MW)</td>
<td>~500</td>
<td>150</td>
<td>2500</td>
</tr>
<tr>
<td>Total area of (test) blankets (m^2)</td>
<td>~12</td>
<td>~65</td>
<td>~670</td>
</tr>
</tbody>
</table>

It is attractive to provide “prototypical fusion power conditions” in reduced size, power and tritium use.
Projected Limit in Tritium Supply Dictates Use by CTF before Demo Starts (How Many?)

- **ITER** uses ~11 kg T to provide 0.3 MW-yr/m²
- Assuming 80% tritium recovery,
  - Demo needs 3 kg/month to produce 2500 MW fusion power.
  - One CTF needs 5 kg to accumulate 6 MW-yr/m²
The ST Configuration Naturally Fits the Demands of CTF Performance

- Natural elongation at low $l_i \rightarrow$ simple shaping coils
- $I_{TF} \sim I_p$; moderate $B_T \rightarrow$ slender, single-turn, demountable TF center leg
- No central solenoid $\rightarrow$ no inboard nuclear shielding
- No inboard blanket $\rightarrow$ smaller aspect ratio & size
- $\sim 6\text{-}7\%$ fusion neutrons lost to center leg $\rightarrow$ adequate tritium recovery

Testing on CTF Limited by Capabilities in P/R and $W_L$

[Assuming example parameters based on the PPCF, 47 (2005) B263 study]

$R_0/a = 1.2/0.8 \text{ m}, \kappa \sim 3, B_T = 2.55 \text{ T}, I_{\text{TFC}} \sim 15 \text{ MA}, P_{\text{TFC}} \sim 200 \text{ MW}$

- Can provide testing for $W_L = 0.5 - 4 \text{ MW/m}^2$ within the physics margins

<table>
<thead>
<tr>
<th>14-MeV neut. flux, $W_L$, MW/m$^2$</th>
<th>1.0</th>
<th>2.0</th>
<th>4.0</th>
</tr>
</thead>
<tbody>
<tr>
<td>P/R, MW/m</td>
<td>42</td>
<td>67</td>
<td>102</td>
</tr>
<tr>
<td>$I_p$, MA</td>
<td>8.7</td>
<td>11.2</td>
<td>14.1</td>
</tr>
<tr>
<td>Plasma magnetic flux, $\mu_0\ell_iR_I_p$, Wb</td>
<td>3.9</td>
<td>5.1</td>
<td>6.4</td>
</tr>
<tr>
<td>Global $H_{98pby}$ factor</td>
<td>1.3</td>
<td>1.3</td>
<td>1.3</td>
</tr>
<tr>
<td>$\beta_T$, %</td>
<td>13</td>
<td>21</td>
<td>35</td>
</tr>
<tr>
<td>Safety factor, $q_{\text{cyl}}$</td>
<td>4.4</td>
<td>3.4</td>
<td>2.7</td>
</tr>
<tr>
<td>$n/n_{\text{GW}}$</td>
<td>0.27</td>
<td>0.28</td>
<td>0.34</td>
</tr>
<tr>
<td>$\langle T_i \rangle/\langle T_e \rangle$</td>
<td>1.6</td>
<td>1.5</td>
<td>1.4</td>
</tr>
<tr>
<td>$I_{\text{NB+RF}}/I_p; I_{\text{BS}}/I_p$</td>
<td>0.46; 0.54</td>
<td>0.48; 0.52</td>
<td>0.45; 0.55</td>
</tr>
<tr>
<td>$P_{\text{fusion}}$, MW</td>
<td>72</td>
<td>144</td>
<td>288</td>
</tr>
<tr>
<td>$P_{\text{NBI+RF}}$, MW</td>
<td>41</td>
<td>52</td>
<td>65</td>
</tr>
<tr>
<td>Neutral beam energy, kV</td>
<td>180</td>
<td>238</td>
<td>358</td>
</tr>
</tbody>
</table>
Key R&D Needs for CTF – Candidates for “Step-out” Ideas with High Leverage

• Have no-wall $\beta$, high $q_{\text{cyl}}$, $I_{\text{BS}}/I_p \sim 0.5$, NB & RF powers – but,
• Very high $P/R$ at low to moderate plasma densities
  • Will ITER divertor be adequate (>ITER, ~0.5xDemo)?
  • Advanced divertor - liquid Li?
• Nearly solenoid-free operation
  • Initiation: CHI, plasma gun, ECW/EBW, LHW to CTF scale?
  • Partial induction: small iron core, vertical field swing, external solenoids?
  • Ramp-up & sustainment: ECW/EBW, LHW, NBI (& *AEs), bootstrap?
• Effects of large rotation and Hot-Ion H-Mode on confinement
  • What determines $\chi_i$, $\chi_e$, $\chi_\phi$?
  • How to allow large rotation & reduce MHD modes and field errors?
• Disruptions and ELMs near no-wall limit, and loads on divertor
• Very long pulse: minutes $\rightarrow$ hours $\rightarrow$ beyond
• Radiation resistant insulators for possibility of insulated center
• Large, high-current, low-voltage power supply for single-turn TF coil
Candidates for Scientific Research to Enable Game-Changing Technology for CTF and Fusion in General

- Super Insulator?
- Plasma Gun?
- Liquid Lithium Divertor?
- Neutral Beam Source?
Scientific Basis for Cesiated Sources That Produce Negative-Hydrogen (D,T) Ions Is Still Under Active R&D
Some Basic Research Issues to Facilitate Negative Ion Neutral Beams  [Larry Grisham, lgrisham@pppl.gov]

- What is the origin of the negative hydrogen ions which are actually extracted from cesiated ion sources?
- Are there other ways to make usable current densities of D- that don’t require alkali metals in the source?
- How much does the oxygen in cesiated sources lower the extractable current density? Is it stored in the cesium? Where does it originate (old leaks, the insulators) and through what processes (breakdowns, hydrogen leaching)?
- What is the origin of the large halo on the beamlets from cesiated negative ion sources? Is this intrinsic to the process, or an accident of design?
Some Basic Development Issues to Facilitate Negative Ion Neutral Beams

• Need to understand genesis of plasma discharges producing unipolar arcs in accelerator support structures, and find ways to suppress them.

• Need to test insulators under beam environment and find ways to hold voltage much better.

• Need novel plasma discharge suppression schemes, such as magnetic insulation (produced by large low voltage currents through accelerator electrodes).

• Need usable aperture offset beamlet steering to compress space charge in beam envelop (presently implemented version shown not to work).
Scientific Feasibility of Current Start-up in ST Was Recently Shown on Pegasus

Can current be scaled up 20 ($I_p \sim MA$) → 200 times ($I_p \sim 10MA$)?

[garstka@engr.wisc.edu]
**Multi-Pinch Experiment**

**Aims:**
Stable Screw Pinch plasma with the full dimension and mushroom shape, but reduced current $I_e$ (2.7 Vs. 8.5 kA)

**Phylosophy:**
Almost all parts should be reutilized in PROTO-SPHERA

START vacuum vessel
(already in Frascati)

- Provisional simplified PF coils system with constant current (partially recoverable for PROTO-SPHERA)
- Fed with 0.6 kA, 120 V (no water cooling)
  (1.9 kA, 350 V in PROTO-SPHERA)

*ICC2004 Madison, Wisconsin*
Can Such Filaments Be Adapted for Use by Sustainable High Current Plasma Guns?

• Each coil designed for 150A
  Present test aims for 2.7 kA total
  Future test aims for 60 kA total

[Paolo Micozzi, micozzi@efr406.frascati.enea.it]
Bursts of Plasma Pulse on ITER Divertor from Edge Localized Modes (ELMs) Set Severe Lifetime Limits

- ELMs result from instabilities at edge plasma pedestal
- Substantial fraction of pedestal energy escapes in frequent plasma pulses
- Divertor plate suffers substantial erosion
- Could liquid Li surface be better?

[Mike Ulrickson, maulric@sandia.gov]
[Dick Majeski, rmajeski@pppl.gov]

OAK RIDGE NATIONAL LABORATORY
U. S. DEPARTMENT OF ENERGY
With Single-Turn Toroidal Field Coil (TFC), CTF Requires Expensive Power Supply for Magnets

- TFC power supplies for ITER vs. CTF = 68 kA, 900 V vs. 15 MA, 20 V dc. Single-turn TFC avoids electrical insulators, which do not survive radiation; can radiation-resilient insulator be developed? [Steve Zinkle, zinklesj@ornl.gov]
ST Research Contributes to Advancing Fusion, and Leverages on Several Game-Changing Issues

- ST research worldwide is growing rapidly
- ST research is making critical contributions to fusion
- CTF using ST configuration will potentially make strategic contributions in component testing
- Several game-changing issues are identified potentially making CTF and fusion energy more attractive
  - Cesiated neutral beam sources
  - Sustainable high current plasma gun
  - Liquid Li surface divertor
  - Radiation resilient electrical insulator