Optimizations of Plasma Configuration and Performance of CFETR

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OUTLINE

• Introduction
• Snowflake divertor configuration
• Optimizations for the size of PF coils
• Selections of the plasma profiles
CFETR missions and major parameters

- Missions of **Chinese Fusion Engineering Test Reactor (CFETR)**
  - Complementary with ITER
  - Demonstration of fusion energy production
  - Demonstration of tritium self-sufficiency
  - Exploring options for DEMO blanket & divertor solution
  - Solution for easy remote maintenance of in-vessel modules

- Major parameters

<table>
<thead>
<tr>
<th>Parameter</th>
<th>CFETR</th>
<th>ITER</th>
</tr>
</thead>
<tbody>
<tr>
<td>Plasma major radius (R0)</td>
<td>5.7 m</td>
<td>6.2 m</td>
</tr>
<tr>
<td>Plasma minor radius (a)</td>
<td>1.6 m</td>
<td>2.0 m</td>
</tr>
<tr>
<td>Plasma elongation (Sepratrix/95% flux surface)</td>
<td>2.0/1.87</td>
<td>1.85/1.70</td>
</tr>
<tr>
<td>Toroidal field at R0</td>
<td>5.0 T</td>
<td>5.3 T</td>
</tr>
<tr>
<td>Plasma current (Ip)</td>
<td>8~10 MA</td>
<td>9 ~ 15 MA</td>
</tr>
<tr>
<td>Safety factor at 95% flux surface (q95)</td>
<td>3.3 ~ 4.3</td>
<td>3.0 ~ 5.0</td>
</tr>
<tr>
<td>Fusion power</td>
<td>~200 MW</td>
<td>500 MW</td>
</tr>
</tbody>
</table>
Plasma operation mode (by 0D analysis)

<table>
<thead>
<tr>
<th>Operation mode</th>
<th>A</th>
<th>B</th>
<th>C</th>
<th>D</th>
<th>E</th>
<th>ITER-SS</th>
<th>Upgraded</th>
</tr>
</thead>
<tbody>
<tr>
<td>$I_p$(MA)</td>
<td>10</td>
<td>10</td>
<td>10</td>
<td>8</td>
<td>8</td>
<td>9</td>
<td>15</td>
</tr>
<tr>
<td>$P_{aux}$(MW)</td>
<td>65</td>
<td>65</td>
<td>65</td>
<td>65~70</td>
<td>65</td>
<td>59</td>
<td>65</td>
</tr>
<tr>
<td>$q_{95}$</td>
<td>3.9</td>
<td>3.9</td>
<td>3.9</td>
<td>4.9</td>
<td>4.9</td>
<td>5.2</td>
<td>3.9</td>
</tr>
<tr>
<td>$W$(MJ)</td>
<td>171~174</td>
<td>193</td>
<td>270~278</td>
<td>171</td>
<td>255</td>
<td>287</td>
<td>540</td>
</tr>
<tr>
<td>$P_{Fus}$(MW)</td>
<td>197~230</td>
<td>209</td>
<td>468~553</td>
<td>187~210</td>
<td>409</td>
<td>356</td>
<td>1000</td>
</tr>
<tr>
<td>$Q_{pl}$</td>
<td>3.0~3.5</td>
<td>3.2</td>
<td>7.2~8.5</td>
<td>2.7~3.2</td>
<td>6.3</td>
<td>6.0</td>
<td>15</td>
</tr>
<tr>
<td>$T_{i0}$(keV)</td>
<td>17.8~18.5</td>
<td>29</td>
<td>19.8~20.8</td>
<td>20.6~21</td>
<td>21</td>
<td>19</td>
<td>25</td>
</tr>
<tr>
<td>$N_{el}(10^{20}/m^3)$</td>
<td>0.75</td>
<td>0.52</td>
<td>1.06</td>
<td>0.65</td>
<td>0.94</td>
<td>1</td>
<td></td>
</tr>
<tr>
<td>$n_{GR}$</td>
<td>0.6</td>
<td>0.42</td>
<td>0.85</td>
<td>0.65</td>
<td>0.95</td>
<td>0.82</td>
<td>0.85</td>
</tr>
<tr>
<td>$\beta_N$</td>
<td>1.59~1.62</td>
<td>1.8</td>
<td>2.51~2.59</td>
<td>2</td>
<td>2.97</td>
<td>3.0</td>
<td>2.7</td>
</tr>
<tr>
<td>$\beta_T$(%)</td>
<td>~2.0</td>
<td>2.3</td>
<td>3.1~3.25</td>
<td>2</td>
<td>2.97</td>
<td>2.8</td>
<td>4.2</td>
</tr>
<tr>
<td>$f_{bs}$ (%)</td>
<td>31.7~32.3</td>
<td>35.8</td>
<td>50~51.5</td>
<td>50</td>
<td>73.9</td>
<td>48</td>
<td>47</td>
</tr>
<tr>
<td>$\tau_{98Y2}$(s)</td>
<td>1.82~1.74</td>
<td>1.55</td>
<td>1.57~1.47</td>
<td>1.37</td>
<td>1.29</td>
<td>1.94</td>
<td>1.88</td>
</tr>
<tr>
<td>$P_{N}/A$(MW/m^2)</td>
<td>0.35~0.41</td>
<td>0.37</td>
<td>0.98</td>
<td>0.33~0.37</td>
<td>0.73</td>
<td>0.5</td>
<td>1.38</td>
</tr>
<tr>
<td>$I_{CD}$(MA)</td>
<td>3.0~3.1</td>
<td>7.0</td>
<td>2.45</td>
<td>4.0</td>
<td>2.76</td>
<td>3.0</td>
<td></td>
</tr>
<tr>
<td>$H_{98}$</td>
<td>1</td>
<td>1.3</td>
<td>1.2</td>
<td>1.3</td>
<td>1.5</td>
<td>1.57</td>
<td>1.2</td>
</tr>
<tr>
<td>$T_{burning}$(S)</td>
<td>1250</td>
<td>SS</td>
<td>2200</td>
<td>M/SS</td>
<td>SS</td>
<td>??</td>
<td></td>
</tr>
</tbody>
</table>
Plasma configurations

- Main differences in configuration between CFETR and ITER:
  - \( \kappa = 2.0 \)
    - Higher bootstrap current fraction
    - Higher beta limit
    - Need higher power source to control the vertically stability, but it is still acceptable (Liu et al, JFE, 2015)

- Divertor
  - Two dedicated divertor coils for snowflake (SF) divertor
  - SF divertor is compatible with ITER-like divertor
OUTLINE

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Snowflake divertor configuration

- With the two dedicated divertor coils, all three kinds of snowflake divertor configurations can be obtained.
- The current in the PF coils does not exceed the current limit.
Snowflake is good to flux expansion

- Snowflake divertor increases the flux expansion with a factor $\sim 2$

 Flux expansion is defined as $\Delta / \xi$
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Flux state history for a discharge

- By empirical law, the flux consumption for the 10MA discharge ramp-up could be estimated:

<table>
<thead>
<tr>
<th>Flux Type</th>
<th>Formula</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Resistive flux loss</td>
<td>$\mu_0 I_p R_0 C_{Ejima}$</td>
<td>~ 35 Wb</td>
</tr>
<tr>
<td>Inductive flux loss</td>
<td>$\mu_0 I_p R_0 l_i/2$</td>
<td>~ 39 Wb</td>
</tr>
<tr>
<td>External flux loss</td>
<td>$L_{ext} I_p = \mu_0 I_p R_0 [\ln (8R_0/a) - 2]$</td>
<td>~ 87 Wb</td>
</tr>
<tr>
<td>Breakdown</td>
<td></td>
<td>~ 10 Wb</td>
</tr>
<tr>
<td>Heating/CD save</td>
<td></td>
<td>-20 ~ -30 Wb</td>
</tr>
</tbody>
</table>

- Fiducial point
  
  SOP: start of plasma
  
  PIR: point in ramp
  
  SOD: start of divertor
  
  SOF: start of flattop
  
  MOF: middle of flattop
  
  EOF: end of flattop

- If we leave ~30 Vs for the flattop, then the 10 MA discharge require about 175 Vs

- Then at each fiducial point, the corresponding flux state is:

<table>
<thead>
<tr>
<th>Fiducial point</th>
<th>SOP</th>
<th>SOD</th>
<th>SOF</th>
<th>MOF</th>
<th>EOF</th>
</tr>
</thead>
<tbody>
<tr>
<td>Flux state (Wb)</td>
<td>65</td>
<td>-7</td>
<td>-80</td>
<td>-95</td>
<td>-110</td>
</tr>
</tbody>
</table>
Histories of current in PF coils by static equilibrium analysis

- For each fiducial point, the equilibrium is calculated to match the flux state, then the correspond current in PF coils is obtained.
- Connect the points to form the time histories of current in PF coils.
- For ITER-like divertor 10 MA case, if the PF coil material is same as ITER, the size of PF coils are big enough.
- For SF 10 MA inductive case, the current in CS1 exceeded the limit.
Optimization for the size of PF coils

- For 10 MA operation, some PF coils are oversized
  - CS3U, CS3L, CS2L, PF3, PF4, PF5
- For SF 10 MA inductive case, currents in CS1U and CS1L exceed the limit
  - We may lower the plasma current or operate with the fully non-inductive scenario

<table>
<thead>
<tr>
<th></th>
<th>Max current (MA)</th>
<th>Limit (MA)</th>
<th>Max/limit %</th>
</tr>
</thead>
<tbody>
<tr>
<td>CS1U</td>
<td>-31.77</td>
<td>26.5</td>
<td>-119.89%</td>
</tr>
<tr>
<td>CS2U</td>
<td>-20.47</td>
<td>26.5</td>
<td>-77.25%</td>
</tr>
<tr>
<td>CS3U</td>
<td>18.13</td>
<td>26.5</td>
<td>68.42%</td>
</tr>
<tr>
<td>CS3L</td>
<td>18.06</td>
<td>26.5</td>
<td>68.15%</td>
</tr>
<tr>
<td>CS2L</td>
<td>-16.18</td>
<td>26.5</td>
<td>-61.06%</td>
</tr>
<tr>
<td>CS1L</td>
<td>-31.5</td>
<td>26.5</td>
<td>-118.87%</td>
</tr>
<tr>
<td>PF1</td>
<td>22.56</td>
<td>27.8</td>
<td>81.15%</td>
</tr>
<tr>
<td>PF2</td>
<td>-11.02</td>
<td>12.3</td>
<td>-89.59%</td>
</tr>
<tr>
<td>PF3</td>
<td>-6.55</td>
<td>14.6</td>
<td>-44.86%</td>
</tr>
<tr>
<td>PF4</td>
<td>-10.27</td>
<td>14.6</td>
<td>-70.34%</td>
</tr>
<tr>
<td>PF5</td>
<td>-6.19</td>
<td>12.3</td>
<td>-50.33%</td>
</tr>
<tr>
<td>DC1</td>
<td>11</td>
<td>12.3</td>
<td>89.43%</td>
</tr>
<tr>
<td>DC2</td>
<td>12</td>
<td>12.3</td>
<td>97.56%</td>
</tr>
<tr>
<td>PF6</td>
<td>-25</td>
<td>27.8</td>
<td>-89.93%</td>
</tr>
</tbody>
</table>

Summary of the max current over current limit
OUTLINE

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About Plasma profiles

- Profiles are key aspects of the operation scenario
  - Electron density, impurity density, temperature, current/safety factor, rotation, pedestal structure
- Profiles are extremely complicate, coupled each other
- No detailed self-consistent discharge simulation has been performed yet
- Here we discuss some aspects (pedestal structure and density profile) we are considering
- The profile selection is not an optimization, it is just the consequence of the scenario, constrained by the global plasma parameters
Density profile

- Two scaling laws for density
  - Density peaking
  - Greenwald density limit

- The density profile is specified with the form:

\[
n_e(\psi) = n_{sep} + a_{n0}\left\{\tanh\left[\frac{2(1 - \psi_{mid})}{\Delta}\right] - \tanh\left[\frac{2(1 - \psi_{mid})}{\Delta}\right]\right\} + a_{n1}H\left(1 - \frac{\psi}{\psi_{ped}}\right)\left[1 - \left(\frac{\psi}{\psi_{ped}}\right)^{\alpha_{n1}}\right]^{\alpha_{n2}}
\]

\[
\nu_{\text{eff}} = 0.1 Z_{\text{eff}} \langle n_e \rangle R / \langle T_e \rangle^2
\]

For CFETR baseline case, the density peaking is about 1.8

\[
\text{ne for baseline case with } ne=0.52, \text{ peaking}=1.8
\]
Pedestal structure

- Presently EPED is the unique theory model to predict the pedestal height and width
- We use EPED1 model and Sauter bootstrap current model to determine the pedestal structure (pressure height and width, current profile)
- For the baseline scenario, the pedestal has the height of ~42 kPa and the width of ~0.033ψ
- The edge current is self-consistently constructed by the Sauter bootstrap current model
Summary

- All kinds of SF configuration (exact SF, SF+, SF-) could be obtained, with the two additional dedicated divertor coils.
- SF configuration could reduce the heat flux on the divertor plate by magnetic flux expansion.
- Size of PF coils is optimized, by using a static time slice equilibrium analysis method.
- EPED1 model and Sauter bootstrap current model are used to construct the self-consistent pedestal structure. For the baseline scenario, the pedestal has the height of ~42 kPa and the width of ~0.033ψ.
- Density profile is constructed by the constrains of line average density and density peaking.
Discussion

• It is still very preliminary 1D and 2D analysis
• Further detailed, self-consistent simulations should be carried out to optimize the plasma configuration and performance.
• We are considering the phase-II of CFETR, with higher magnetic field and larger plasma size. The design of phase-I should be compatible with phase-II
ISFNT-12国际会议见闻

李国强

2015年9月30日

四室系列学术活动
ISFNT-12在济州国际会议中心（ICC）举行
What is ISFNT

• International Symposium on Fusion Nuclear Technology

• Technical Topics
  1. Plasma-Facing High Heat Flux Components
  2. Blanket/First Wall Technology
  3. Fuel Cycle and Tritium Processing
  4. Material Engineering for FNT
  5. Vacuum Vessel (incl. FNT of Ex-vessel Components)
  6. Nuclear System Design
  7. Safety Issues and Waste Management
  8. Models and Experiments for FNT
  9. Repair and Maintenance
  10. Burning Plasma Control and Operation
  11. Inertial Confinement Fusion Studies and Technologies
  12. Fission-Fusion Synergy and Cross Cutting Technologies
会议报告分布

• 7 keynote talks
  – ITER, Bernard Bigot, ITER status
  – Korea, Keeman Kim, Korean progress on fusion research and plans
  – US, Mohamed Abdou, Challenges and pathways towards DEMO
  – EU, Gianfranco Federici, EU DEMO
  – Japan, Hiroshi Yamada, Japanese endeavor for DEMO technology
  – India, Shishir Deshpande, India DEMO
  – China, Lu Delong, Chinese contributions to ITER

• 9 Plenary talks
  – Japan 1, China 1 (Yican Wu), ITER 3, EU 4

• 72 Oral talks
  – China 12 (ASIPP 3, CAEP 3, SWIP 2, INEST 4)

• ~500 posters
参会人员情况

- 中日韩欧人较多，美印人相对较少，俄国人最少。可能因为美国、俄国没有TBM
- 中国人很多，主要来自
  - 等离子体所7人
  - ITER-CN
  - 核安所，10多人，多人为Program committee成员，三人做Chairman
  - 科大核学院，10多人，学生为主
  - 公司，西部超导体
  - SWIP，2人
  - 中国原子能研究院 2人
  - 九院，2人，氚相关
  - 大连理工大学
  - 西安交大、西南交大、清华、杭州电子科大、华中科大...
热点: ITER and DEMO related, superconducting tokamak

- Blanket
  - TBM on ITER
  - Blanket on DEMO
- Tritium technology
- Divertor technology
- Remote handling
- Material
Challenges on technology

Background

Outstanding Technical Challenges with Gaps beyond ITER

For any further fusion step, safety, T-breeding, power exhaust, RH, component lifetime and plant availability, are important design drivers and CANNOT be compromised.

Tritium breeding blanket
- most novel part of DEMO
- TBR >1 marginally achievable but with thin PFCs/few penetrations
- Feasibility concerns/performance uncertainties with all concepts → R&D
- Selection now is premature
- ITER TBM is important

Power Exhaust
- Peak heat fluxes near technological limits (>10 MW/m²)
- ITER solution may be marginal for DEMO
- Advanced divertor solutions may be needed but integration is very challenging

Remote Maintenance
- Strong impact on IVC design
- Significant differences with ITER
- RM approach for blanket
- RH schemes affects plant design and layout
- Large size Hot Cell required
- Service Joining Technology R&D is urgently needed.

Structural and HHF Materials
- Progressive blanket operation strategy (1st blanket 20 dpa; 2nd blanket 50 dpa)
- Embrittlement of RAFM steels and Cu-alloys at low temp. and loss of mechanical strength at high temp.
- Need of structural design criteria and design codes
- Technical down selection and development of an Early Neutron Source (IFMIF-DONES)

→ O. Croft (CCFE) - O1C.1

→ ICFRM17 - Aachen
其它一些点滴

- CFETR, FNST 介于 ITER 和 DEMO 之间，其它国家一般是 DEMO，尺寸大于 ITER。没有看见俄罗斯的 DEMO 计划
- 来自中国的报告和 poster 中，涉及 CFETR 的挺多
- CFETR 所用的氚从哪里来？
- 从物理设计的角度来看，各国 DEMO 设计的进度差不多，主要是零维设计 + 一些简单的一维计算，还没有做像 ITER 那样较为完整的 scenario 模拟
- Abdou award，一个日本年轻人获得
- JET DT 实验，2017 T 实验，2018 氘氚实验
Thank you