Road Map of Chinese Fusion Research and the First Chinese Fusion Reactor- CFETR

Yuanxi Wan¹, ²

¹ University of Science and Technology of China, Hefei, China
² Institute of Plasma Physics, CAS Hefei, China
E-mail: wanyx@ipp.ac.cn or wanxy@ustc.edu.cn

30th April – 2nd May 2013
Physikzentrum Bad Honnef, Germany
Content

- Introduction
- background information
- Opinion and consideration
- Progress on the concept design of CFETR
- Summary
China is facing to the serious energy problems

The largest population which is 1.3 billion now and will be 1.5~1.6 billion in 2050 in China;

The capitation energy consumption is only the half of the world level and 1/10 of the developed country;

Coal still is main consumption energy resources (~ 70%) and it only can be used for about 80 years according to current prediction;

The socio-economic development with annual rate of 8~10 % has been kept for more than 20 years and the consumption of energy will be more and more large.
China is facing to the serious shortage and pollution of energy from now and will more serious in near future;

China must develop renewable energy and nuclear energy as fast as possible;

Both development of fission power plant and fusion research in China are getting strong support now.
Government Policies

In near term (2006-2020):

- Renewable energy will be increased from 7% to 16%;
- Closedown small size thermal power plants and promote the power plants with high efficiency and finally the energy consumption per GDP should be decreased about 20%, the CO$_2$ should be decreased ~10%.
- Strong support the development of the Hydro, wind, solar and renewable power plants and it will be ~30% of total power plants at 2020.
National program on fission

In middle long and long term (2006-2050):

- Now: 9 GW (~2% of total capacity in operation) and 25.4GW under construction
  - 2020: 70 -100 GW
  - 2050: 240GW at least (20% of total) or 360 GW 30% of total

Promote the Fast Reactor:
  - Now: The CEFR with ~ 20 MWe has been completed
  - 2020: DEFR with ~ 600 MWe;
  - 2028: Power plant of FR with 1~1.5 GWe

Promote more HTGR:
  - 10MW HTGR has been operated for several years;
  - A new power plant has been designed and will be constructed soon
The fusion research projects

ITER in China:
- in kind contribution
- related domestic projects

- HL-2A tokamak in SWIP;
- EAST superconducting tokamak in ASIPP;
- University Programs: theory, experiments and education.
**ITER in China**

**Contribution by in kind**

**Domestic sub-project for:**

- Improving the basic conditions;
- Theory and education;
- Fusion materials;
- Development for key tech.
- Reactor design.

**Budget for ITER-CN (10 years)**

Total is 0.6 billion US dollars

~ 0.6 billion US dollars

30th April – 2nd May 2013 Physikzentrum Bad Honnef, Germany
HL-2A in SWIP

- **R**: 1.65 m
- **a**: 0.40 m
- **Bt**: 1.2~2.8 T
- **Configuration**: Limiter, LSN divertor
- **Ip**: 150 ~ 480 kA
- **ne**: 1.0 ~ 6.0 x 10^{19} m^{-3}
- **Te**: 1.5 ~ 5.0 keV
- **Ti**: 0.5 ~ 1.5 keV

**Auxiliary heating:**
- ECRH/ECCD: (3+2) MW
  - (6/68 GHz/500 kW/1 s)
  - modulation: 10~30 Hz; 10~100 %
- NBI(tangential): 1.5 MW
- LHCD: 1 MW
  - (2/2.45 GHz/500 kW/1 s)

**Fueling system (H_2/D_2):**
- Gas puffing (LFS, HFS, divertor)
- Pellet injection (LFS, HFS)
- SMBI (LFS, HFS)
  - LFS: f =1~80 Hz, pulse duration > 0.5 ms
  - gas pressure < 3 MPa
EAST Superconducting Tokamak in ASIPP
University programs

- More than 10 universities are involved in 10-15 tasks (40-60M$/per year) with 200 Staff; 200 students in MCF project;
- 3 theoretical research centers (Hefei, Zhejiang, Beijing);
- School of NST in USTC has been created: 100 undergraduate/year 100-150 (MS+Ph.D)/year

SUNIST in Tsinghua Univ.

KT-5 in USTC

J-TEXT in Huazhong Univ.
Support system for fusion research in China

Premier

- NSFC
- CNNC
- MOST
- NDRC
- CAS
- MOE

ITER and Related
National project

Fusion supported by

- National Development and Reform Commission – NDRC
- Ministry of Science and Technology – MOST
- Ministry of Education – MOE
- Chinese Academy of Sciences – CAS
- China National Nuclear Corporation – CNNC
- Natural Science Foundation of China – NSFC

30th April – 2nd May 2013 Physikzentrum Bad Honnef, Germany
Why China joint ITER?

What is the goal of fusion research in China?
What should be the next step for MFR in China?

Under discussion  final option

What is Possible Road map of Fusion to DEMO in China

30th April – 2nd May 2013 Physikzentrum Bad Honnef, Germany
National Integration Design group for Magnetic Confinement Fusion Reactor has been founded.

关于成立磁约束聚变堆总体设计组（筹备）的通知

教育部科技司、中科院基础局，中核集团科技部:

为全面接纳国际热核聚变实验堆（ITER）设计技术，掌握聚变堆相关的物理和工程设计及关键技术，开展我国磁约束聚变堆总体设计研究，经研究，决定成立磁约束聚变堆总体设计组（筹备）。

磁约束聚变堆总体设计组（筹备）由19名成员组成，元光教授任组长，李京研究员、刘永研究员和汪剑研究员任副组长。磁约束聚变堆总体设计组（筹备）成员名单见附件。总体设计组成员可根据需要适当增补或调整。

磁约束聚变堆总体设计组（筹备）的职责是:

1. 全面收集、整理、保存、吸收、消化和利用ITER总体设计及总体管理技术资料;

2. 开展我国聚变堆总体设计研究，编制聚变堆总体设计各项任务的规划、计划和实施方案，根据发展需求对计划实施提出调整建议，为条件成熟时建设中国聚变堆奠定必要的设计基础;

3. 研究提出聚变堆设计人才培养方案建议，并根据需要协助推进实施;

4. 完成科技部交办的其它任务。

中国科学技术大学作为磁约束聚变堆总体设计组（筹备）依托单位，科技部将每两年对依托单位进行评估。

请各部门和单位支持磁约束聚变堆总体设计组（筹备）的工作。

附件：磁约束聚变堆总体设计组（筹备）成员名单

二O一一年三月十六日

主题词：成立 聚变 设计 通知

抄送：中国科学技术大学，总体设计组各成员单位。
Working task and schedule

2012-2014: provide two options of engineering concept design of CFETR which should include in:

- Missions; Type; Size; Main physics basis; Main techniques basis to be taken
- The concept engineering design for all sub-systems
- Budget & Schedule; Location; Management system
  (Budget is ready)

2015: proposal to government to try to get permission for construction in the national next 5Y plan—started at 2016
Content

- Introduction
  - some background information
  - opinion and consideration
- Progress on the concept design of CFETR
- Summary
Workshops & group meetings
Commens understanding:
final goal is to obtain realistic FE (FPP)

Fusion Power Station

D-T burning plasma with the SSO or high duty time

30th April – 2nd May 2013 Physikzentrum Bad Honnef, Germany
The steps for going to fusion energy (FPP)

If the fusion energy is the goal, the necessary steps should be:

1. **achieve the burning plasma**:  
   - high density \( n \)  
   - high temperature \( T_i \)  
   - high energy and particle confinement \( \tau_E \)

\[
P_f \propto n \times T \times \tau_E
\]

2. **sustain the burning plasma to be SSO or long pulse with high duty cycle time**:  
   - CW heating: \( \alpha \) particles or external heating such as NBI, RFH ?  
   - CW CD: bootstrap CD ? or external inductive or no-inductive ?  
   - CW fueling  
   - CW exhausting the ash of burning by divertor  
   - CW extracting the particle’s energy by divertor  
   - CW extracting the fusion neutron energy by blanket via first wall

3. **Tritium must be self-sustainable by blanket**;

4. **The materials of first wall and blanket have suitable lifetime**;

30th April – 2nd May 2013 Physikzentrum Bad Honnef, Germany
The scientific goals of ITER

ITER is the burning plasma device and its scientific goals are:

• to produce a plasma dominated by $\alpha$-particle heating
• produce a significant fusion power amplification factor $(Q \geq 10)$ in long-pulse operation (300 - 500s)
• aim to achieve steady-state operation of a tokamak $(Q=5)$
• retain the possibility of exploring controlled ignition $(Q \geq 30)$
• demonstrate integrated operation of technologies for a fusion power plant
• test components required for a fusion power plant
• test concepts for a tritium breeding module
Gaps between ITER and FPP

Even if ITER can make great contribution to long pulse or SSO burning plasma but it is mainly on physics and not on real fusion energy because of the real burning time during it’s 14 years D-T operation is only about 4 %, which results:

1. There is no enough fusion energy produced for utilization.
2. As the consequent the total neutron flux is not enough to demonstrate the real tritium breeding for self sustainable of tritium by blanket.
3. No enough neutrons to do the material tests in high flux fusion neutron radioactive environments.
Gaps between ITER and FPP

4. Therefore there only are shielding blankets for ITER.

5. Even if adding the TBM with addition budget but it is only concept testing for tritium breeding and not real self sustainable blanket and related material tests.

Conclusion: the engineering test reactor is necessary to be constructed parallel with or after ITER and before the fusion power plant (FPP).
Common understanding (1)

- The CFETR must be built before the fusion power plant in China.
- ITER can be a good basis for CFETR both on SSO burning plasma physics and some technologies.
- The goals of CFETR should be different with ITER and aimed to the problems related with fusion energy.
- Both physic and technical basis of the CFETR should be conservative and realistic but it should have the capability for further upgrade.
Common understanding (2)

- So mission must be realistic and sequence must be right !!
- The cost for fusion energy, the multi application of blanket could be lower priority in compare with T selves- sustainable and heat conversion and extracted;
- The divertor will be another key component for the success of future FPP- it will be the most important components related with the basic requirements for SSO both on physics and technologies.

The goal of our design is to try to build the reactor for fusion energy as early as possible !!
Content

- Introduction

  - some background information

  - opinion and consideration

- Progress on the concept design of CFETR

- Summary
Preliminary design consideration

1. Mission
2. Two options
3. Key parameter investigation
4. Integrated design of the device with RH
5. Blanket
6. Divertor
7. Tritium
8. Diagnostics, etc.
Mission of CFETR

- A good complementary with ITER
- Demonstration the fusion energy with a minim
  \[ P_f = 50 \sim 200 \text{MW}; \]
- Long pulse or steady-state operation with
  duty cycle time \( \geq 0.3 \sim 0.5; \)
- Demonstration of full cycle of T self-sustained with TBR \( \geq 1.2 \)
- Relay on the existing ITER physical (\( k<1.8, q>3, H\sim1 \))
  and technical bases;
- Exploring options for DEMO blanket & divertor with a easy
  changeable core by RH;

The goal of our design is to try to build the reactor for fusion
energy as early as possible!!
Preliminary design consideration

1. Mission
2. Two options
3. Key parameter investigation
4. Integrated design of the device with RH
5. Blanket
6. Divertor
7. Tritium
8. Diagnostics, etc.
Option 1: Superconducting Tokamak

- $B_t = 4.5-5T$;
- $I_p = 8-10$ MA;
- $R = 5.7$ m;
- $a = 1.6$ m;
- $K = a/b = 1.8 \sim 2.0$;
- $\beta_N \sim 2.0$;
- $q_{95} \geq 3$;
- Triangularity $\delta = 0.4-0.8$;
- Single-null divertor;
- Neutron wall loading $\approx 0.5$ MW/m$^2$;
- Duty cycle time $= 0.3-0.5$;
- TBR $\geq 1.2$;
- Possible UG: ($R \sim 5.9$ m, $a \sim 2$ m, $B_t = 5$ T, $I_p \sim 14$ MA)
Option 1: Superconducting Tokamak

- Based on ITER physics and techniques, emphasizing SSO and TBM;
- Physical and engineering conceptual design can basically meet the design requirement;
- The detailed conceptual design of tokamak machine (including RH) is technically feasible;
- Device size is slightly smaller than ITER (~75%);
- The required electric power can be smaller when operated on the parameters of 150-200MW. The net electric power demand is about 150MW;
- The risk is not great based on ITER and the existing technology, except for the tritium blanket;
- The development space (\(R \sim 5.9\)m, \(a \sim 2\)m, \(B_t = 5\)T, \(I_p \sim 14\)MA) has been reserved for future upgrades.
Option 2: Water-Cooling Cu Magnets Tokamak

- $B_t = 3T \sim 4T$ (can be disassembly)
- $I_p = 8-12\text{MA}$;
- $R = 3.7m-3.9m$;
- $a = 1.2m$;
- $K = 2.0-2.2$;
- Triangularity $\delta = 0.6-0.8$;
- Double-null divertor;
- Size is 20% of ITER;
- Neutron wall loading $\approx 0.3\sim0.62\text{MW/m}^2$;
- Duty cycle time $= 0.3-0.5$;
- $TBR \geq 1.2$

16 TF coils, 6 CS coils, 12 PF coils. The coil material is copper alloy conductor.
Option 2: Water-Cooling Cu Magnets Tokamak

- The water-cooling scheme based on ITER physical basis is with high feasibility;
- With good accessibility, relatively low investment, flexibility of burning, operation and shutdown, and many other advantages;
- CS coils lay inside of the TF coils, which can provide larger volt-seconds. The inner side of TF is thin while outer side is thick to reduce electric power and meet cooling requirements for the coils. (Cu coils power is about 450MW);
- Engineering feasible PF coils lay inside of the TF coils to reduce the current of PF coils and can strengthen its control ability. ITER-like, Snowflake, Super-X and mixed divertor configurations can be realized;
- The engineering conceptual design should be completed.
Preliminary design consideration

1. Mission
2. Two options
3. Key parameter investigation
4. Integrated design of the device with RH
5. Blanket
6. Divertor
7. Tritium
8. Diagnostics, etc.
Along the equator:
- CS
- TFC
- TS
- VV
- Blanket
- Plasma blanket
- VV
- TS
- TFC
- Cryostat

Poloidal configuration:
- Cryostat
- TS
- PF (divertor coils)
- TF
- TS
- VV
- Blanket
- Divertor
- Elongated pl;

Integrated configuration of key parameters
Parameter investigation (10 MA)

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value 1</th>
<th>Value 2</th>
<th>Value 3</th>
<th>Value 4</th>
<th>Value 5</th>
<th>Value 6</th>
<th>Value 7</th>
<th>Value 8</th>
</tr>
</thead>
<tbody>
<tr>
<td>$P_{aux}$ (MW)</td>
<td>50</td>
<td>50</td>
<td>80</td>
<td>80</td>
<td>80</td>
<td>80</td>
<td>80</td>
<td>80</td>
</tr>
<tr>
<td>$E$ (MJ)</td>
<td>168</td>
<td>191</td>
<td>1701</td>
<td>209</td>
<td>270</td>
<td>302</td>
<td>237</td>
<td></td>
</tr>
<tr>
<td>$P_{Fus}$ (MW)</td>
<td>215</td>
<td>280</td>
<td>209</td>
<td>332</td>
<td>498</td>
<td>583</td>
<td>333</td>
<td></td>
</tr>
<tr>
<td>$Q$</td>
<td>5.38</td>
<td>7.01</td>
<td>3.26</td>
<td>5.19</td>
<td>7.78</td>
<td>9.1</td>
<td>5.21</td>
<td></td>
</tr>
<tr>
<td>$T_i$</td>
<td>15.3</td>
<td>14.3</td>
<td>18.1</td>
<td>15.7</td>
<td>20.2</td>
<td>22.6</td>
<td>25.1</td>
<td></td>
</tr>
<tr>
<td>ICD (MA)</td>
<td>1.43</td>
<td>1.1</td>
<td>3.16</td>
<td>1.93</td>
<td>2.49</td>
<td>2.78</td>
<td>4.38</td>
<td></td>
</tr>
<tr>
<td>$\beta_N$</td>
<td>1.52</td>
<td>1.73</td>
<td>1.54</td>
<td>1.9</td>
<td>2.44</td>
<td>2.73</td>
<td>2.14</td>
<td></td>
</tr>
<tr>
<td>$\beta_P$</td>
<td>0.79</td>
<td>0.89</td>
<td>0.79</td>
<td>0.98</td>
<td>1.26</td>
<td>1.41</td>
<td>1.1</td>
<td></td>
</tr>
<tr>
<td>$f_{bs}$</td>
<td>31.67</td>
<td>35.9</td>
<td>32.05</td>
<td>39.46</td>
<td>50.8</td>
<td>56.76</td>
<td>44.48</td>
<td></td>
</tr>
<tr>
<td>Res</td>
<td>7.77E-09</td>
<td>8.63E-09</td>
<td>6.06E-09</td>
<td>7.49E-09</td>
<td>5.13E-09</td>
<td>4.34E-09</td>
<td>3.71E-09</td>
<td></td>
</tr>
<tr>
<td>$n_{el}$</td>
<td>0.82</td>
<td>1</td>
<td>0.7</td>
<td>1</td>
<td>1</td>
<td>1</td>
<td>1</td>
<td>0.7</td>
</tr>
<tr>
<td>nGR</td>
<td>0.7</td>
<td>0.85</td>
<td>0.6</td>
<td>0.85</td>
<td>0.85</td>
<td>0.85</td>
<td>0.6</td>
<td></td>
</tr>
<tr>
<td>$\tau_{E98Y2}$</td>
<td>2.25</td>
<td>2.21</td>
<td>1.79</td>
<td>1.79</td>
<td>1.53</td>
<td>1.43</td>
<td>1.55</td>
<td></td>
</tr>
<tr>
<td>$P_{n/Awall}$</td>
<td>0.37</td>
<td>0.48</td>
<td>0.36</td>
<td>0.57</td>
<td>0.86</td>
<td>1</td>
<td>0.58</td>
<td></td>
</tr>
<tr>
<td>Pthr</td>
<td>64.9</td>
<td>74.6</td>
<td>58.1</td>
<td>74.6</td>
<td>74.6</td>
<td>74.6</td>
<td>58.1</td>
<td></td>
</tr>
<tr>
<td>H98</td>
<td>1</td>
<td>1</td>
<td>1</td>
<td>1</td>
<td>1.2</td>
<td>1.3</td>
<td>1.3</td>
<td></td>
</tr>
<tr>
<td>$T_{burn}$ (s)</td>
<td>715</td>
<td>656</td>
<td>1377</td>
<td>972</td>
<td>2500</td>
<td>4500</td>
<td>7093</td>
<td></td>
</tr>
</tbody>
</table>

$\text{Ip} \sim 10\text{MA}, B_t=5\text{T}, q_{95}=4.17, Z_{\text{eff}} \sim 1.76, \quad \gamma_{CD} = 0.16 \sim 0.26$ (ITER target 0.4)

Rampup 120V*s, break 10V*S, burning 30V*s

Higher confinement needs a certain of momentum (toroidal rotation), It is still a problem for the reactor level plasma (high density)!
### Parameter investigation (8MA)

<table>
<thead>
<tr>
<th>Parameter</th>
<th>141</th>
<th>159</th>
<th>178</th>
<th>196</th>
<th>206</th>
<th>183</th>
</tr>
</thead>
<tbody>
<tr>
<td>E(MJ)</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>183</td>
</tr>
<tr>
<td>P_Fus(MW)</td>
<td>155</td>
<td>193</td>
<td>234</td>
<td>276</td>
<td>298</td>
<td>226</td>
</tr>
<tr>
<td>Q</td>
<td>2.4</td>
<td>3.0</td>
<td>3.7</td>
<td>4.1</td>
<td>4.6</td>
<td>3.53</td>
</tr>
<tr>
<td>Ti0</td>
<td>13.2</td>
<td>14.8</td>
<td>16.6</td>
<td>18.4</td>
<td>19.3</td>
<td>20.8</td>
</tr>
<tr>
<td>ICD(MA)</td>
<td>2.03</td>
<td>2.29</td>
<td>2.56</td>
<td>2.83</td>
<td>2.97</td>
<td>3.9</td>
</tr>
<tr>
<td>betaN</td>
<td>1.59</td>
<td>1.79</td>
<td>2.00</td>
<td>2.22</td>
<td>2.33</td>
<td>2.07</td>
</tr>
<tr>
<td>betaP</td>
<td>1.03</td>
<td>1.16</td>
<td>1.29</td>
<td>1.43</td>
<td>1.50</td>
<td>1.33</td>
</tr>
<tr>
<td>fbs</td>
<td>41.4</td>
<td>46.7</td>
<td>52.2</td>
<td>57.8</td>
<td>60.7</td>
<td>54</td>
</tr>
<tr>
<td>Res</td>
<td>9.72E-09</td>
<td>8.13E-09</td>
<td>6.89E-09</td>
<td>5.90E-09</td>
<td>5.49E-09</td>
<td>4.88E-09</td>
</tr>
<tr>
<td>nel</td>
<td>0.79</td>
<td>0.79</td>
<td>0.79</td>
<td>0.79</td>
<td>0.79</td>
<td>0.65</td>
</tr>
<tr>
<td>nGR</td>
<td>0.85</td>
<td>0.85</td>
<td>0.85</td>
<td>0.85</td>
<td>0.85</td>
<td>0.7</td>
</tr>
<tr>
<td>taoE98Y2</td>
<td>1.65</td>
<td>1.56</td>
<td>1.48</td>
<td>1.41</td>
<td>1.38</td>
<td>1.38</td>
</tr>
<tr>
<td>Pn/Awall</td>
<td>0.27</td>
<td>0.33</td>
<td>0.40</td>
<td>0.47</td>
<td>0.51</td>
<td>0.39</td>
</tr>
<tr>
<td>Pthre</td>
<td>63.6</td>
<td>63.6</td>
<td>63.6</td>
<td>63.6</td>
<td>63.6</td>
<td>55.3</td>
</tr>
<tr>
<td>H98</td>
<td>1</td>
<td>1.1</td>
<td>1.2</td>
<td>1.3</td>
<td>1.35</td>
<td>1.35</td>
</tr>
<tr>
<td>T_burn(s)</td>
<td>1933</td>
<td>3075</td>
<td>5714</td>
<td>15693</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

Ip~8MA, Bt=5T, q95=5.2, Zeff~1.76, g_CD = 0.15~0.22 (ITER target 0.4)
Rampup :100V*s, Breakdown:10V*S, burning 50V*s

30th April – 2nd May 2013 Physikzentrum Bad Honnef, Germany
Core Plasma performance

On the basis of current device parameters and employed heating power, under standard H mode operation, the goal of fusion power, and achieve plasma burning above thousands of seconds in the region of $\beta_N = 1.5 \sim 2.0$ which is far away from the limit value of $\beta_N = 3.0$ can be achieved. Means that:

1. Plasma can be operated in steady region, corresponding to the requirements of simplifying engineering technology;
2. Leaving enough space for the device to develop higher performance plasma and other potential applications in the future;
3. If $q_{95} \geq 5$, can make the plasma operated in steady state;
4. If $q_{95} \geq 3.0$, Plasma may be more possible to operate for a short time at larger current levels, may demonstrate higher fusion power, and can conduct burning plasma study which is closely related to fusion energy.
Preliminary design consideration

1. Mission
2. Two options
3. Key parameter investigation
4. Integrated design of the device with RH
5. Blanket
6. Divertor
7. Tritium
8. Diagnostics, etc.
Along the equator:
- CS
- TFC
- TS
- VV
- Blanket
- Plasma Blanket
- VV
- Cryostat
- TFC
- Cryostat

Porotal configuration:
- Cryostat
- TS
- PF
- (divertor coils)
- TF
- TS
- VV
- Blanket
- Divertor
- Elongated pl;

Integrated configuration of key parameters
**Divertor Design Study (Superconducting)**

Refered to the ITER magnets design parameters, TF and CS magnets design for the three divertor configurations at \( R = 5.7 \text{m} \) can meet the requirements of physical parameters, except for the case of Super-X configuration at \( R = 5.9 \text{m} \) in which the CS coils field intensity is about 13T (reached the threshold of Nb\(_3\)Sn materials). It can be improved greatly, by redesigning the magnets and using mixed magnets or Nb\(_3\)Al for CS in the future.
Remote Handling (Superconducting)

Duty time of CFETR is 0.3~0.5. The device maintenance downtime is request to be as short as possible to ensure this parameter.

Cryostat and thermal shield also have three kinds of designs, according to the different design of tokamak machine: Big Windows, Middle Windows, and Up, Middle, Down 3 Windows.
Main Parameters under 3 divertor configurations

<table>
<thead>
<tr>
<th>Parameter</th>
<th>ITER-Like</th>
<th>Super-X</th>
<th>Snowflake</th>
<th>ITER</th>
</tr>
</thead>
<tbody>
<tr>
<td>Total number of TF coil(N)</td>
<td>16</td>
<td>16</td>
<td>16</td>
<td>18</td>
</tr>
<tr>
<td>Plasma current (MA)</td>
<td>10</td>
<td>10</td>
<td>10</td>
<td>15</td>
</tr>
<tr>
<td>Center field(T)</td>
<td>5.0</td>
<td>5.0</td>
<td>5.0</td>
<td>5.3</td>
</tr>
<tr>
<td>Maximum current of TF</td>
<td>67.4/70 kA/turn</td>
<td>67.4/70 kA/turn</td>
<td>67.4/70 kA/turn</td>
<td>68 kA/turn</td>
</tr>
<tr>
<td>Major radius of plasma(m)</td>
<td>5.7/5.9</td>
<td>5.7/5.9</td>
<td>5.7/5.9</td>
<td>6.2</td>
</tr>
<tr>
<td>Minor radius of plasma(m)</td>
<td>1.6</td>
<td>1.6</td>
<td>1.48/1.58</td>
<td>2.0</td>
</tr>
<tr>
<td>The center radius of CS(m)</td>
<td>1.415/1.615</td>
<td>1.415/1.615</td>
<td>1.415/1.615</td>
<td>2.055</td>
</tr>
<tr>
<td>Maximum V.S</td>
<td>160</td>
<td>160</td>
<td>160</td>
<td>240-250</td>
</tr>
<tr>
<td>Elongation</td>
<td>1.8/2.0</td>
<td>1.8/2.0</td>
<td>2.17/2.14</td>
<td>1.70/1.85</td>
</tr>
<tr>
<td>Number of PF coils</td>
<td>6</td>
<td>8</td>
<td>8</td>
<td>6</td>
</tr>
</tbody>
</table>

Consist of 16 TF coils, 6 CS coils, 24 CC and PF coils (for the ITER-like divertor, 6 PF coils, but for the Super-X and Snowflake divertor configuration, 8 PF coils). TF and CS coils are proposed to adopt Nb₃Sn superconductor, while PF and CC coils will adopt NbTi.
The VV design matched TF coils has been completed, which focused on the optimization of RH scheme to ensure the duty time of CFETR. Detailed design and engineering analysis will be done after the conceptual design completed.
Inner and Outer Thermal Shield Conceptual Design

Inner and Outer thermal shields also have three kinds of designs, according to different designs of tokamak machine and cryostat: Big Windows, Middle Windows, and Up, Middle, Down 3 Windows.
Outer Cryostat Conceptual Design

Big Windows

Up, Middle, Down 3 Windows

Middle Windows

Anticipated loads act on Cryostat

<table>
<thead>
<tr>
<th>Load Type</th>
<th>Anticipated loads Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>Gravity</td>
<td>Total Cryostat Weight</td>
</tr>
<tr>
<td></td>
<td>$\text{MPa (Category I)}$</td>
</tr>
<tr>
<td></td>
<td>$0.12 \text{ MPa (Category IV)}$</td>
</tr>
<tr>
<td>External Pressure</td>
<td>vacuum: the incident that air or helium enters into cryostat during normal operation (Category II)</td>
</tr>
<tr>
<td>Internal Pressure</td>
<td>the incident that water or helium enters into cryostat during normal operation (Category IV)</td>
</tr>
<tr>
<td>Seismic Load SL-1</td>
<td>SL-2 * 0.34</td>
</tr>
<tr>
<td>Vertical Displacement Event/Disruption</td>
<td>For the whole structure can be neglected</td>
</tr>
<tr>
<td>Thermal Load</td>
<td>Thermal loads produced by magnets, VV support and cryostat radiation</td>
</tr>
<tr>
<td></td>
<td>Cryostat wall temperature ranges from 200k to 310k under accident condition</td>
</tr>
</tbody>
</table>
On the basis of matching the machine size, i.e., the plasma major radius $R = 5.7\,\text{m}$ and minor radius $r = 1.6\,\text{m}$, considering as well the plasma design scheme, if the device will be further upgraded in the future that VV should be large enough to satisfy of $R = 5.9\,\text{m}$ and $r = 1.6\,\text{m}$.

The magnets and VV systems design scheme should satisfy the three divertor configurations simultaneously.

Can realize both of the balance configurations of $R = 5.7\,\text{m}$ and $R = 5.9\,\text{m}$, under the same magnets system.
The duty cycle time of CFTER will be impacted by the principle of RH significantly.
Remote Handling (Superconducting)

Duty time of CFETR is 0.3~0.5. The device maintenance downtime is request to be as short as possible to ensure this parameter.

Cryostat and thermal shield also have three kinds of designs, according to the different design of tokamak machine: Big Windows, Middle Windows, and Up, Middle, Down 3 Windows.
RH conceptual design for big window style strategy

All of the in-vessel components can be moved out in one time.

Maintenance process:
1. Dismantle the window’s flange
2. Cut the cooling pipe and other connection things by remote handing
3. Inset wheels assemble under the blanket and lifting the blanket
4. Move back the wheels with the blanket, by gear system
5. Use remote handing and guide rail to keep the blanket balance
6. Close the window’s flange and move the CASK to hot cell for repair
RH conceptual design for big window style strategy

CASK system

- The CASK inner space is:
  - high=12.5m,
  - width=5.5m,
  - length=11.8m.
- And the CASK wall thickness is 100mm

- CASK
- Flange

- travelling crane at the top of CASK
- Remote handling
- Electromotor and gear system
- Wheels assemble, will insert under the blanket
- Guide rail and it support

windows

- Top guide groove: keep blanket balance
- Bottom guide rail
- Window's support to support the guide rail
- Groove in the VV, to lay up the wheels assemble

30th April – 2nd May 2013 Physikzentrum Bad Honnef, Germany
RH conceptual design for medium window style strategy

Every equatorial port shall be used as maintenance port for blanket Remote Handling (RH) system. 4 lower port shall be used for divertor RH system. (TBD)
RH conceptual design for medium window style strategy

Maintenance process:

1. Cask01 and VV port connection
2. Open sealed door, drag trolley enter
3. Drag the plug-in module out
4. Close the sealed door, disconnection

1. Cask02 and VV port connection
2. Open sealed door, manipulator and transporter enter
3. Manipulator grab the blanket module
4. Manipulator put the blanket module on the transporter
5. Transporter move the module out
6. Move all blankets modules out then disconnect, move the Cask02 to hot cell

30th April – 2nd May 2013 Physikzentrum Bad Honnef, Germany
RH conceptual design for Upper port strategy

The reactor has 16 sections with 8 equatorial ports, 4 lower ports and 4 upper ports. The shield blanket modules are permanent and the breeding blanket modules are removable.
RH conceptual design for Upper port strategy

Maintenance process:

1. Move cask to the upper port
2. Connection, open the sealed door
3. Manipulator enter VV
4. Manipulator grab the blanket module.
5. Transfer the module to lifting position (under upper port)
6. Lift the module
7. Close the sealed door and disconnect
8. Move to Hot cell
Preliminary design consideration

1. Mission
2. Two options
3. Key parameter investigation
4. Integrated design of the device with RH
5. Blanket
6. Divertor
7. Tritium
8. Diagnostics, etc.
Three groups are working on the concept design of CFETR blanket

**Group I: Helium cooling solid blanket**

1) HC (8MPa, 300/500°C), Li$_4$SiO$_4$ (Li$_2$TiO$_3$), Be, RAFM

**Group II: Liquid blanket**

1) SLL (~150 °C), CLAM
2) DLL (~700 °C), CLAM

**Group III: Water cooling solid blanket**

1) HC, Li$_4$SiO$_4$, Be, RAFM
2) WC, Li$_2$TiO$_3$, Be$_{12}$Ti, RAFM

30th April – 2nd May 2013 Physikzentrum Bad Honnef, Germany
**Group I:**
HC (8MPa, 300/500°C), Li$_4$SiO$_4$ (Li$_2$TiO$_3$), Be, RAFM

**Group II:**
SLL (~150°C), CLAM
DLL (~700°C), CLAM

**Group III:**
1) HC, Li$_4$SiO$_4$, Be RAFM
2) WC, Li$_2$TiO$_3$, Be$_{12}$Ti, RAFM

*Helium-Cooled Pebble Breeder Concept for EU*

30th April – 2nd May 2013 Physikzentrum Bad Honnef, Germany
Main Results of Blanket Design

- **NWL** (average): 0.33 MW/m\(^2\)
- Inboard shielding thickness: ~ 46 cm
- Outboard shielding thickness: ~ 40 cm
- Inboard breeder thickness: ~ 37 cm
- Outboard breeder thickness: ~ 67 cm
- TBR can ≥ 1.2 but very sensitive by outboard windows
- TBR is impacted by first wall material and thickness;
- Difference between 1D and 3D calculations ~ 15-20%
Preliminary design consideration

1. Mission
2. Two options
3. Key parameter investigation
4. Integrated design of the device with RH
5. Blanket
6. Divertor
7. Tritium
8. Diagnostics, etc.
The power exhaust (ITER as example)

\[ P_{\text{SOL}} = P_{\text{add}} + P_a - P_{\text{rad, core}} \]
\[ = 40 + 100 - 40 = 100 \text{MW} \]

The plasma-wetted surface area \( A_w \) at the divertor plate:
\[ A_w = 4RW_{\text{SOL}} F_{\text{exp}}/\sin(\alpha) \]
\[ W_{\text{SOL}} = 5 \text{mm}, F_{\text{exp}} = 4.3, \alpha = 25 \text{ degree} \]
\[ A_w = 4 \text{m}^2 \]

For a maximum heat flux of 10 MW/m² (material and engineering constraints): 40 MW

60 MW must be radiated away before arriving at the divertor plate

Require divertor detachment to reduce heat load to < 10 MW/m²
Three types of divertor configuration can be obtained

ITER-like

Snowflake

Super-X
Considerations of Divertor Design of CFETR

CFETR divertor configurations: bottom SN

Advantages of SN:

- more simple
- larger volume of pl.
- benefit on TBR
The conceptual integrated divertor design
The Integrated Design Considerations

（1）16 TF coils, the ripple is below 0.5
（2）RH: Considering several RH schemes, vertical lifting is the first choice
（3）H&CD: The life of the NBI ion source can last about 2-3 years, under 100-200MW neutron flux. At the early stage of CFETR, all of the four kinds of H&CD can be used, so the design of H&CD system can follow the optimization of heating and drive efficiency. While at the later stage can only use ECRH and RF;
（4）Configuration: The first choice is LSN which can save large space for the lower half of the device. The VV in the part of the lower divertor can have larger curvature which saved enough space for pumping, in particular for snow-flake divertor coils;
（5）VV: the future operating temperature is within 250-300 degrees, so as to maintain tritium retention at the minimum level;
The Integrated Design Considerations

(6) Divertor: Consider ITER-like LSN, snowflake and Super-X configurations;

(7) Blanket: Consider three optimizing ways, and the blanket must be shielded completely against neutrons;

(8) Cryogenic, CODAC, power source, water-cooling system etc. will refer to ITER design;

(9) Diagnostic: Simple diagnostic system, mainly focused on the configuration, density, neutron, high-energy particles, divertor & PFC state and DUST eta. which are closely related to the safety operation;

(10) Tritium system: It is designed as the continuous operation on 200MW in the first stage.
Preliminary design consideration

1. Mission
2. Two options
3. Key parameter investigation
4. Integrated design of the device with RH
5. Blanket
6. Divertor
7. Tritium
8. Diagnostics, etc.
Deuterium/Tritium Fuel Cycle and Tritium Protection System
Estimates of Disposal Capabilities and Efficiency (1)

- 1 year tritium burning (on 200MW): 5.56kg/a
- Breeding tritium needed in 1 year (simply calculated on TBR=1.2):
  \[ 5.56\text{kg/a} \times 1.2 = 6.672\text{kg/a} \]
- First tritium input: \(~ 2\text{kg}\)
- If burning efficiency of tritium is 5%,
  - The amount of tritium into the VV/year: 5.56kg/5% = 111.2kg/a
  - Considering various tritium retention, tritium processing: 120kg/a!
  - Considering D processing (T/D=1), system processing capability: 240kg/a
- If burning efficiency is 1%,
  - tritium processing is about 600kg/a,
  - the system processing capability is 1200kg/a

Challenge suggestion:
- Endeavored toward 5% tritium burning efficiency (or even higher);
- Endeavored toward 1200kg/a fuel cycle system processing capability (tritium processing 600kg/a);
- Ensure enough redundancy between physical and techniques.
Estimates of Disposal Capabilities and Efficiency (2)

Referred to design of ITER, tritium emission limit not exceeds $1g/a$ (controlled on $0.6g/a$
- If the total recovery rate of tritium protection system can achieve $99.9\%$
- The leakage of the main technique system is not allowed to exceed $600g/a$
- For fuel cycle system that tritium operation quantity is $600kg/a$ (if burning efficiency is $1\%$), the leakage rate $= (600g/a)/(600kg/a \times 1000g/kg) = 0.1 \%$
  - Total tritium recovery rate $= 1 - 0.1 \% = 99.9\%$
  - this is also a tough technical parameter
- For fuel cycle system that tritium operation quantity is $120kg/a$ (if burning efficiency is $5\%$), the leakage rate $= (600g/a)/(120kg/a \times 1000g/kg) = 0.5 \%$
  - Total tritium recovery rate $= 1 - 0.5 \% = 99.5\%$
  - this is possible an achievable technical parameter

Challenged suggestion: endeavored toward $99.9\%$ recovery rate of Tritium protection system and fuel cycle system.
Summary of tritium inventories
(maximum instantaneous values for each system; not all simultaneous)

<table>
<thead>
<tr>
<th>Type of inventory</th>
<th>Maximum values (g T)</th>
</tr>
</thead>
<tbody>
<tr>
<td>• In-vessel</td>
<td>450</td>
</tr>
<tr>
<td>• Estimated subtotal for FC systems</td>
<td>833</td>
</tr>
<tr>
<td>✓ Fuelling system</td>
<td>55</td>
</tr>
<tr>
<td>✓ Mechanical vacuum pumps (VPS)</td>
<td>20</td>
</tr>
<tr>
<td>✓ Torus exhaust processing (TEP)</td>
<td>30</td>
</tr>
<tr>
<td>✓ Isotope separation system (ISS)</td>
<td>220</td>
</tr>
<tr>
<td>✓ Storage and delivery system (SDS)</td>
<td>480</td>
</tr>
<tr>
<td>✓ Other systems (&lt;15 g each)</td>
<td>28</td>
</tr>
<tr>
<td>• Long term storage</td>
<td>2 × 450</td>
</tr>
<tr>
<td>• Hot cell and waste treatment</td>
<td>250</td>
</tr>
<tr>
<td>• Total</td>
<td>2433</td>
</tr>
</tbody>
</table>

Assume: fraction of burn up 5% ; Reserve time: one day
First inventory of T: ~ 2000 g
Tritium Breeding and Extraction System

- Tritium breeding with solid/liquid breeders is under decision for CFETR
- Tritium extraction system for solid/liquid concepts of TBM have been designed

Tritium extraction from helium coolant: Catalytic oxidation—adsorption with molecular sieve.

Tritium extraction system both for China solid and liquid TBM: Isotopic exchange with hydrogen—adsorption with molecular sieve.
Refered to design of ITER, tritium emissions limit can’t exceed 1g/a (control on 0.6g/a)

✓ The total recovery rate of tritium protection system is designed as 99.9%

➢ Exhaust purification

✓ Include: glovebox tritium removal system, ventilation tritium removal system, apace air tritium removal system, VV emergency ventilation tritium removal system, VV maintain tritium removal system, helium cooling gas tritium removal system and so on

• Mainly use catalytic oxidation technology (partly use tritium alloy absorption technology)

➢ Water tritium removal

✓ Mainly use CECE technology

• May need to design 2~3 sets of CECE system and cascade
• According to the different hydrogen isotope composition, choose to enter into different CECE systems
Preliminary design consideration

1. Mission
2. Two options
3. Key parameter investigation
4. Integrated design of the device with RH
5. Blanket
6. Divertor
7. Tritium
8. Diagnostics, etc.
According to the needs for fusion reactor operation, the classification and arrangement of priority levels for diagnostic have been carried out.

<table>
<thead>
<tr>
<th>1 Device Operation, Safety, and Basic Control</th>
<th>2 Advanced Plasma Control</th>
<th>3 Physics Study</th>
</tr>
</thead>
<tbody>
<tr>
<td>Plasma configuration</td>
<td>Neutron and α-particle profile</td>
<td>Constraint α-particles</td>
</tr>
<tr>
<td>Plasma current, q(a)</td>
<td>Plasma rotation</td>
<td>fast ions (H, D, T and He³)</td>
</tr>
<tr>
<td>Toroidal voltage</td>
<td>Plasma current density distribution (q distribution)</td>
<td>He³ density</td>
</tr>
<tr>
<td>Fusion power</td>
<td>Electron and ion temperature profile</td>
<td>The total fast ions energy loss</td>
</tr>
<tr>
<td>β\textsubscript{N}</td>
<td>Electron density profile (core and boundary)</td>
<td>TAE mode, Fishbone mode</td>
</tr>
<tr>
<td>Line averaged electron density</td>
<td>Radiation power profile (core, X point and divertor chamber)</td>
<td>Electron and ion temperature profile (boundary)</td>
</tr>
<tr>
<td>Impurities and D,T flux (divertor and main VV)</td>
<td>Z\textsubscript{eff} profile, impurity density profile</td>
<td>Electron and ion temperature profile (near X point)</td>
</tr>
<tr>
<td>Surface temperature (divertor chamber and main VV first wall)</td>
<td>He density (divertor chamber)</td>
<td>Ion temperature in divertor chamber</td>
</tr>
<tr>
<td>Escape electrons</td>
<td>Heat deposition profile (divertor chamber)</td>
<td>The plasma flux (divertor)</td>
</tr>
<tr>
<td>Detached divertor plasma measurement (target J sat, n\textsubscript{e}, T\textsubscript{e})</td>
<td>Neutral density between plasma and first wall</td>
<td>n\textsubscript{T}/n\textsubscript{D}/n\textsubscript{H} (boundary)</td>
</tr>
<tr>
<td>Radiation power (main plasma, X point and divertor chamber)</td>
<td>The electron temperature and density within divertor</td>
<td>n\textsubscript{T}/n\textsubscript{D}/n\textsubscript{H} (divertor)</td>
</tr>
<tr>
<td>Halo current</td>
<td>The loss rate of α-particles</td>
<td>Electronic temperature perturbation</td>
</tr>
<tr>
<td>Disruption indication (m=2 mode locking)</td>
<td>The erosion rate of divertor target</td>
<td>electron density perturbation</td>
</tr>
<tr>
<td>H/L mode and ELMs characterization</td>
<td>Neutron and α-particle profile, Neutron flow rate</td>
<td>Radial electric field and its perturbation</td>
</tr>
<tr>
<td>Line averaged Z\textsubscript{eff}</td>
<td>NTM</td>
<td>Core and edge plasma turbulence</td>
</tr>
<tr>
<td>Edge pressure and mass spectrometry (divertor and piping)</td>
<td>Sawtooth</td>
<td>Core plasma MHD activities</td>
</tr>
<tr>
<td>Resistive Wall Modes</td>
<td>Dust</td>
<td></td>
</tr>
</tbody>
</table>

30th April – 2nd May 2013 Physikzentrum Bad Honnef, Germany
CFETR operation can be divided into three stages

**Preliminary Stage:** H, D, or He operation, no fusion reaction.

**Objective of diagnostic construction:**
- Obtain basic operation law and establish scaling law for fusion reactor;
- Research the internal relation and discharge priority level of all kinds of diagnostic to provide experience and data support for reducing the diagnostic.

**Transition Stage:** As required, part of diagnostics are replaced by TBM to realize the tritium breeding target.

**Objective of diagnostic construction:**
- Research and test reliability of the diagnostic under fusion conditions;
- The influence of fusion and its products on original scaling law, optimize the scaling law;
- Research the effects of fusion products on plasma performance, reduce the number and types of diagnostic gradually, and identify stage diagnostic scheme of fusion reactor.

**Operation Stage:** Most of the diagnostic windows would be occupied by TBM, leaving only necessary diagnostics for fusion reactor operation, security, fusion product monitoring and basic plasma parameters.

**Diagnostic target:** In this stage, the key point is to use the minimum number of plasma diagnostics to ensure safety, stability and normal operation of fusion reactor by using technical means.
CFETR Control System

- CFETR control system includes all hardware, software and network interface used to control the operation of CFETR.
- Divided into up and down two levels: Central System and Plant Control System.
- Divided into three levels according to functions: CODAC, Interlock Protection, Safety Controllers.
CFETR Cryogenic System (1)

CFETR Cryogenic System can be divided into three subsystems:

- Low temperature refrigeration system,
- Low temperature distribution system and
- Low temperature & room temperature transmission pipeline system.

Helium and nitrogen refrigeration system need to provide about 65kW@4.5 K and 1300kW@80K equivalent cooling capacity, respectively,

due to the estimating of the CFETR system heat load (System heat load) is close to ITER.
Power Supply System

Total electricity distribution and transformation capacity about 1200MVA
The system heat load estimation

According to the present design parameters of fusion reactor, heat load has been preliminarily estimated. Water cooling system can be divided into four parts: TCWS, CCWS, CHWS and Heat-Removal System, according to functions and heat load features of water cooling system. The total tokamak heat load is 460MW, equipment cooling 144MW, HVAC and other chilled water 40MW.

The central part of tokmak water-cooling system is divided into 6 subsystems: Power Water-Cooling Loop, Neutral Beam Water-Cooling Loop, VV Cooling Loop, CVCS, Irrigation & Drainage and Drying. Preliminarily identified the functions of these systems and parameters of main pipelines, completed the steady state simulation calculation of pipelines. Also for the equipment cooling sub-loop.
Preliminary Radioactivity Assessment

Based on Large-Scale Integrated Neutronics Calculation Analysis System, Hybrid Evaluated Data Library, IAEA Database and European Activation Database, CFETR liquid blanket neutronics model is established. Activation and nuclear thermal analysis has been completed, and the varying tendency of shut down contact dose rate of blanket radial components which changes along with cooling time are shown in the figure.
RAMI Analysis (1)

- Using IDEFØ model methods to analyze the functional decomposition of CFETR VV from the top to the bottom, the availability is 99.88% and the reliability is 99.62% in the present conceptual design and boundary conditions of CFETR VV. Referring to ITER hypothesis, CFETR availability plan is 99.43% and reliability prediction is 99.57%. The expected goals and requirements are basically achieved. After FMECA analysis, there are 13 failure modes of major risks and 2 failure modes of medium risks.

- The availability of CFETR PF is 99.89% and the reliability is 99.78% in the present conceptual design and boundary conditions of CFETR PF system. The expected goals and requirements are basically achieved. After FMECA analysis, there are 5 failure modes of major risks and 10 failure modes of medium risks. It’s difficult to repair and replace once the coils are damaged, so a large number of accurate detections are recommended to ensure normal operation of the device. Helium leakage can cause damage to device components, so should carry on strict detection on joints.
RAMI Analysis (2)

- The availability of CFETR TF is 99.89% and the reliability is 99.60% in the present conceptual design and boundary conditions of CFETR TF system. The expected goals and requirements are basically achieved. After FMECA analysis, there are 9 failure modes of major risks and 10 failure modes of medium risks. It’s difficult to repair and replace once the coils are damaged, so a large number of accurate detections are recommended to ensure normal operation of the device. Helium leakage can cause damage to device components, so should carry on strict detection on joints.
Next Step and Target for CFETR design activity

1. Promote the further international communication and cooperation;

2. Recruit talents of plasma science and fusion engineering from both home and abroad to join with CFETR design study;

3. Complete two CFETR engineering conceptual designs in 2014, based on current progress;

4. Complete two proposals in 2015: “key R&D items of CFETR” and “Construction Proposal of CFETR”, based on the design, and submit them to the government.
Summary

1. SSO or long operation with high duty cycle of the burning plasma is the most important issues for MFE development.
2. Missions required for CFETR should be more realistic and aim to the most important challenges for FE.
3. CFETR should be a good complementary with ITER and it will demonstrate the fusion energy with a minim $P_f = 50 \sim 200$ MW, long pulse or steady-state operation with duty cycle time $\geq 0.3 \sim 0.5$, demonstrating the full cycle of T self-sufficiency with TBR $\geq 1.2$, relay on the existing ITER physical and technical bases, exploring options for DEMO blanket & divertor with an easy changeable core by RH.
4. The goal of CFETR design activities is to make a proposal at the end of 2014 to government to try to get permission for construction with the key R&D items for CFETR.
5. The design choices of CFETR are still open.
Thanks for your attention!
Repeatable and stable 30 s elongated divetor H mode plasma has been achieved on EAST
531st Wilhelm and Else Heraeus Seminar on 3D versus 2D in Hot Plasmas

30th April – 2nd May 2013 Physikzentrum Bad Honnef, Germany