Design Integration toward Optimization of LHD-type Fusion Reactor FFHR


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2Kharkov Institute of Physics and Technology, Ukraine
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FFHR
2 GWth
6 Tesla
25,000 ton
Presentation outline
toward design optimization

1. Reactor size and cost
2. Minimization of heating power
3. High density ignition and control
4. Large SC magnet system
5. Access to Demo
6. Concluding remarks
Two main features in FFHR

(1) Quasi-force free $\gamma$ optimization on continuous helical winding
- to reduce the magnetic hoop force (Force Free Helical Reactor: FFHR)
- to expand the blanket space

$$\gamma = \frac{m \ a_c}{l \ R}$$

(2) Self-cooled liquid Flibe ($\text{BeF}_2$-$\text{LiF}$) blanket
- low MHD pressure loss
- low reactivity with air
- low pressure operation
- low tritium solubility
Long life blanket and wide maintenance ports

1. Simplified support structure with wide maintenance ports due to FF concept.

2. Neutron wall loading ~ 1.5 MW/m² is a key for long life or replacement free blanket.

**STB** (Spectral-shifter and Tritium breeder Blanket)

A. Sagara, *Nuclear Fusion* 45 (2005) 258
Reactor size optimization \textit{w/o} increasing neutron wall loading in FFHR

Ergodic layer is important for alpha-heating efficiency > 90 \%,
Three candidates are proposed to increase blanket space > 1.1 m

(1) Reduction of the inboard shielding thickness using WC

(2) Improvement of the symmetry of magnetic surfaces by increasing the current density at the inboard side of the helical coils

by splitting the helical coils.

N. Yanagi et al., in this conference.
Three candidates are proposed to increase blanket space > 1.1 m

(3) Optimization of reactor size around $R_c$ of 16 m

<table>
<thead>
<tr>
<th>Design parameters</th>
<th>LHD</th>
<th>FFHR2</th>
<th>FFHR2m1</th>
<th>FFHR2m2</th>
</tr>
</thead>
<tbody>
<tr>
<td>Polarity</td>
<td>1</td>
<td>2</td>
<td>2</td>
<td>2</td>
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<tr>
<td>Field periods</td>
<td>m</td>
<td>10</td>
<td>10</td>
<td>10</td>
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<tr>
<td>Coil pitch parameter $\gamma$</td>
<td></td>
<td>1.25</td>
<td>1.15</td>
<td>1.15</td>
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<tr>
<td>Coil major Radius $R_c$ (m)</td>
<td>3.9</td>
<td>10</td>
<td>14.0</td>
<td>17.3</td>
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<tr>
<td>Coil minor radius $a_c$ (m)</td>
<td>0.98</td>
<td>2.3</td>
<td>3.22</td>
<td>4.33</td>
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<tr>
<td>Plasma major radius $R_p$ (m)</td>
<td>3.75</td>
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<tr>
<td>Plasma radius $a_p$ (m)</td>
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<td>1.73</td>
<td>2.80</td>
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<td>Plasma volume $V_p$ (m³)</td>
<td>30</td>
<td>303</td>
<td>827</td>
<td>2471</td>
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<tr>
<td>Blanket space $\Delta$ (m)</td>
<td>0.12</td>
<td>0.7</td>
<td>1.1</td>
<td>1.15</td>
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<tr>
<td>Magnetic field $B_0$ (T)</td>
<td>4</td>
<td>10</td>
<td>6.18</td>
<td>4.43</td>
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<tr>
<td>Max. field on coils $B_{\text{max}}$ (T)</td>
<td>9.2</td>
<td>14.8</td>
<td>13.3</td>
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<tr>
<td>Coil current density $j$ (MA/m²)</td>
<td>53</td>
<td>25</td>
<td>26.6</td>
<td>32.8</td>
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<td>Magnetic energy $GJ$</td>
<td>1.64</td>
<td>147</td>
<td>133</td>
<td>118</td>
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<tr>
<td>Fusion power $P_f$ (GW)</td>
<td>1</td>
<td>1.9</td>
<td>3</td>
<td></td>
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<tr>
<td>Neutron wall load $\Gamma_n$ (MW/m²)</td>
<td>1.5</td>
<td>1.5</td>
<td>1.3</td>
<td></td>
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<tr>
<td>External heating power $P_{\text{ext}}$ (MW)</td>
<td>70</td>
<td>80</td>
<td>100</td>
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<tr>
<td>$\alpha$ heating efficiency $\eta_\alpha$</td>
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<td>0.9</td>
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<tr>
<td>Density lim. improvement</td>
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<td>1.5</td>
<td>1.5</td>
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<tr>
<td>H factor of ISS95</td>
<td>2.40</td>
<td>1.92</td>
<td>1.76</td>
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<tr>
<td>Effective ion charge $Z_{\text{eff}}$</td>
<td>1.40</td>
<td>1.34</td>
<td>1.35</td>
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<tr>
<td>Electron density $n_e(0)$ (10^19 m⁻³)</td>
<td>27.4</td>
<td>26.7</td>
<td>19.0</td>
<td></td>
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<tr>
<td>Temperature $T_i(0)$ (keV)</td>
<td>21</td>
<td>15.8</td>
<td>16.1</td>
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<tr>
<td>Plasma beta $&lt;\beta&gt;$ (%)</td>
<td>1.6</td>
<td>3.0</td>
<td>4.1</td>
<td></td>
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<td>Plasma conduction loss $P_L$ (MW)</td>
<td>290</td>
<td>463</td>
<td></td>
<td></td>
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<tr>
<td>Divertor heat load $\Gamma_{\text{div}}$ (MW/m²)</td>
<td>1.6</td>
<td>2.3</td>
<td></td>
<td></td>
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<tr>
<td>Total capital cost G$ (2003)</td>
<td>4.6</td>
<td>5.6</td>
<td>6.9</td>
<td></td>
</tr>
<tr>
<td>COE (mill/kWh)</td>
<td>155</td>
<td>106</td>
<td>87</td>
<td></td>
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</tbody>
</table>

(Cost evaluation based on the ITER (2002) report)

J = 25 ~ 33 A/mm²  
$\Gamma_n$ = 1.3 ~ 1.5 MW/m²
ITER based cost analyses show similar weights and costs between Helical and Tokamak reactor magnet systems (FFHR2m1).

<table>
<thead>
<tr>
<th>Component</th>
<th>Weight (ton)</th>
<th>Cost (B Yen)</th>
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<tbody>
<tr>
<td>Helical Coil</td>
<td>16,000</td>
<td>210</td>
</tr>
<tr>
<td>Poloidal Coil</td>
<td>10,200</td>
<td>110</td>
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<tr>
<td>Supps. &amp; Others</td>
<td>21,900</td>
<td>275</td>
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<tr>
<td>ITER TF</td>
<td></td>
<td></td>
</tr>
<tr>
<td>ITER CS</td>
<td></td>
<td></td>
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<tr>
<td>ITER PF</td>
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Graph showing weight and cost distribution for different components.
Presentation outline

toward design optimization

1. Reactor size and cost
2. Minimization of heating power
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Minimized heating power ~ 30 MW

\[ P_{\text{EXT}} + P_{\alpha} = \frac{dW}{dt} + (P_L + P_B + P_S) \]

Reduction of thermal stress

\( P_{\text{EXT}} + P_{\alpha} = \frac{dW}{dt} + (P_L + P_B + P_S) \)

Minimized heating power ~ 30 MW

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Reduction of thermal stress

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### Design parameters

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<td>Fusion power $P_F$</td>
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</tr>
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</tr>
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<td>keV</td>
<td>15.8</td>
</tr>
<tr>
<td>Effective ion charge $Z_{eff}$</td>
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<td>1.48</td>
</tr>
<tr>
<td>Plasma beta $\langle \beta \rangle$</td>
<td>%</td>
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</tr>
<tr>
<td>Energy confinement time $\tau_E$</td>
<td>s</td>
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<tr>
<td>Bremsstrahlung loss $P_B$</td>
<td>MW</td>
<td>57</td>
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<tr>
<td>Plasma conduction loss $P_L$</td>
<td>MW</td>
<td>282</td>
</tr>
<tr>
<td>Heat load to first wall $\Gamma_{\text{div}}$</td>
<td>MW/m$^2$</td>
<td>0.06</td>
</tr>
<tr>
<td>Heat load to divertor $\Gamma_{\text{div}}$</td>
<td>MW/m$^2$</td>
<td>1.6</td>
</tr>
</tbody>
</table>

**Divertor heat load is drastically reduced.**
New control method in a thermally unstable regime

Proportional-Integration-Derivative (PID) control

The error of the fusion power with an opposite sign of \( e(P_f) = - (P_{fo} - P_f) \) can stabilize the thermal instability through fueling.

\[
S_{DF}(t) = S_{DF0} \left\{ e_{DF}(P_f) + \frac{1}{T_{int}} \int_0^t e_{DF}(P_f) dt + T_d \frac{de_{DF}(P_f)}{dt} \right\} G_{fo}(t)
\]

\( S_{DT}(t) = 0 \) if \( S_{DT}(t) < 0 \)

But, effectively reduced due to burning

\( \tau_{\alpha}/\tau_E = 3 \sim 5 \)  
\( \tau_p/\tau_E = 2 \sim 8 \)

Neutron wall loading is ave. 1.5 MW/m² and peaking factor < 1.3

Evaluated using the recently developed 3D neutronics calculation system for non-axisymmetric helical systems

T. Tanaka et al. 21th IAEA (2006)

Quick feedback between 3D CAD and 3D Monte-Carlo code MCNP5, using numerical helical equations.

Ion density: \( n \propto (1-r^2) \)
Temperature: \( T \propto (1-r^2) \)
Max. Temp.: 16 keV

\[ p(r) \propto n^2 \langle \sigma v \rangle(T) \]

Neutron wall loading is ave. 1.5 MW/m² and peaking factor < 1.3

Two cases of neutron sources

Source distribution (a.u.)

Position (cm)

Helical source
Torus uniform source

Max. Temp.: 16 keV

Total TBR > 1.1

First wall of blanket on helical coil
Max. 1.8 for helical source
Max. 2.0 for uniform source
Average 1.5

( unit : MW/m² )
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Base design of CICC magnet system

S. Imagawa et al., in this conference.

Table 1. Design criteria for CIC conductors based on ITER-TF coils.

<table>
<thead>
<tr>
<th>Items</th>
<th>Design criteria</th>
<th>ITER-TF</th>
</tr>
</thead>
<tbody>
<tr>
<td>Max. cooling length (m)</td>
<td>&lt; 500</td>
<td>390</td>
</tr>
<tr>
<td>Current (kA)</td>
<td>&lt; 100</td>
<td>68</td>
</tr>
<tr>
<td>Maximum field (T)</td>
<td>&lt; 13</td>
<td>11.8</td>
</tr>
<tr>
<td>SC current density (A/mm²)</td>
<td>&lt; 300</td>
<td>273</td>
</tr>
<tr>
<td>Coil current density (A/mm²)</td>
<td>&lt; 30</td>
<td>20.3</td>
</tr>
<tr>
<td>SC material for HC</td>
<td>Nb3Al (*1)</td>
<td>Nb3Sn</td>
</tr>
</tbody>
</table>

(*1) "react and wind" method can be adopted by managing strain during winding within about 0.5%.

Nuclear heating in FFHR2m1

- Max. cooling path is 500 m for the nuclear heat of 1 mW/cm³.
- This value is 5 times larger on the FFHR magnets.
- Gamma-ray heating is dominant.

Central tube
Insulation
SC strands
Conduit (SUS318L)
Internal plate (SUS316L)

5 times
Nuclear shielding of SC magnets by Discrete Pumping with Semi-closed Shield (DPSS)

- Cover rate > 90%
- Acceptable level achieved
  - Fast neutron < $1 \times 10^{22}$ n/m$^2$ in 30 years
  - Max. nuclear heating < 0.2 mW/cm$^3$
  - Total nuclear heating ~ 40 kW
  - Cryogenics power ~ 12 MW (1% of $P_f$)

3D design with DPSS

(a) without DPSS

(b) with DPSS

Unit: n/cm$^2$/s
Indirect cooling system as an alternative with quench protection candidates

K. Takahata et al., 24th SOFT, 2006.

100 kA Superconductor
- High effective thermal conductivity
- High mechanical rigidity and strength

Quench protection by external dumping
- Conventional protection circuit using an external resistor $\tau = 20$ s
- Six subdivisions $V_{max} = 10$ kV
- Hot spot temperature < 150 K

Quench protection by internal dumping
- Quench back with a secondary circuit to increase a decay time constant
- to reduce a transient voltage
- to avoid a serious hot spot

Cross-sectional structure of the helical coil

K. Takahata et al., in this conference.
LHD-type support post for FFHR

- Gravity per support
  - 16,000 ton / 30 legs ~ 530 ton.
- Thermal contraction < max. 55 mm
- Total heat load to 4K ~ 0.34 kW
  (1/20 of stainless steel post)

H. Tamura et al., in this conference.
### Design parameters

<table>
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<tr>
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<th>FFHR2m1</th>
<th>SDC</th>
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<td>Polarity</td>
<td>l</td>
<td>2</td>
<td>2</td>
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<tr>
<td>Field periods</td>
<td>m</td>
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<tr>
<td>Coil pitch parameter</td>
<td>γ</td>
<td>1.25</td>
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<tr>
<td>Coil major Radius</td>
<td>Rc</td>
<td>m</td>
<td>3.9</td>
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<tr>
<td>Coil minor radius</td>
<td>a_c</td>
<td>m</td>
<td>0.98</td>
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<tr>
<td>Plasma major radius</td>
<td>Rp</td>
<td>m</td>
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<tr>
<td>Plasma radius</td>
<td>a_p</td>
<td>m</td>
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<tr>
<td>Plasma volume</td>
<td>Vp</td>
<td>m³</td>
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<td>Blanket space</td>
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<tr>
<td>Magnetic field</td>
<td>B₀</td>
<td>T</td>
<td>4</td>
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<tr>
<td>Max. field on coils</td>
<td>Bₘₐₓ</td>
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<td>Coil current density</td>
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<tr>
<td>Magnetic energy</td>
<td>GJ</td>
<td></td>
<td>1.64</td>
</tr>
</tbody>
</table>

### Fusion power

| P_F | GW  | 1.9 |

### Neutron wall load

| Γₙ  | MW/m² | 1.5 |

### External heating power

| Pₚₑₓₜ | MW  | 80  |

### α heating efficiency

| ηₐ | 0.9 |

### Density limit improvement

| 1.5 | 7.5 |

### H factor of ISS95

| 1.92 |

### Electron density

| nₑ(0) | 10^20 m³ | 2.4 | 9.8 |

### Temperature

| Tₑ(0) | keV   | 15.8 | 6.27 |

### Effective ion charge

| Zₑff | 1.48 | 1.52 |

### Plasma beta

| <β> | %    | 3.0  | 2.5  |

### Energy confinement time

| τₑ  | s    | 1.9  | 4.7  |

### α ash confinement

| τₐ* / τₑ | 3 < 7 |

### Bremsstrahlung loss

| P_B | MW  | 57  | 248 |

### Plasma conduction loss

| P_L | MW  | 282 | 96  |

### Heat load to first wall

| Γₐᶠᵥ | MW/m² | 0.06 | 0.25 |

### Heat load to divertor

| Γₐfv | MW/m² | 1.6  | 0.54 |

### Diverter heat load

| Γₐdv | MW/m² | 1.6  | 0.5  |

### Total capital cost

| GS(2003) | 5.6 |

### COE

| mill/kWh | 106 |
Role of Design Study to Helical Demo-Reactor based on LHD Project

LHD-type Helical Reactor FFHR
- Electric Power: 1GW
- Weight: 25,000 ton
- Magnetic Field: 6T

Helical Demo Reactor (29 years to go)

Multi-layer models covering physics and engineering

Basic Science

LHD-NT
LHD Numerical Test Reactor

Demonstration of steady-state, high-density, high beta by net-current free plasma

Physics of burning plasmas

ITER

Tokamak Experimental Reactor

LHD
Concluding remarks

1. Helical reactor is superior in steady state operation and
   - Reduced neutron wall loading by optimization of large reactor size,
   - Minimized heating power by long access time to ignition,
   - High density operation with reduced heat load on divertor.

2. Large SC magnet system is conceptually feasible.
3. Helical reactor is economically comparable to Tokamak.
4. Numerical Test Reactor is planned to Helical Demo.
5. Large sized construction is important R&D issue.
6. LHD experiments can open new reactor regimes.