Design Integration of the LHD-type Energy Reactor FFHR2 towards Demo

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Invited in 18th ITC 2008
Dec.9-12, 2008 at Toki, Japan.
Role of Design Study to Helical Demo-Reactor based on LHD Project

- LHD (Heliotron)
- FFHR design with R&D’s
- Reactor design, Power generation, Fuel breeding, System development, Reactor material development
- Neutron Source Project (IFMIF) ITER (tokamak)
- Fusion Reactor (France)
- Demonstration of scientific feasibility
- LHD Numerical Test Reactor
- Conversion to Basic Science
- Demonstration of steady state operation Multi-layer physics/engineering model
- Understanding of burning plasmas
- Generation of electric power by fusion in 28 years
- Q=0.075 LHD Exp. Hydrogen → Deuterium High performance, High fusion gain
- 2000 2010 2020 2030 2040 2050
(1) Quasi-force free $\gamma$ optimization on continuous helical winding

- to reduce the magnetic hoop force
  (Force Free Helical Reactor: FFHR)
  ➔ large maintenance ports
- to expand the blanket space

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

(2) Self-cooled liquid Flibe (BeF$_2$-LiF) blanket

- low MHD pressure loss
- low reactivity with air
- low pressure operation
- low tritium solubility
Presentation outline

1. Blanket and divertor space
   - Design windows and cost
   - Nuclear shield on SC coils
   - Reactor size optimization
   - New ignition regime
   - Design parameters
2. SC magnet and supports
3. Blanket system integration
4. Concluding remarks

Graphical representation:

- **1998 FFHR2**
  - $A_p \sim 8$
  - $\gamma = 1.15$

- **1995 FFHR1 ($l=3$)**
  - $A_p \sim 10$
  - $\gamma = 1$

- **2004 FFHR2m1**
  - $A_p \sim 8$
  - $\gamma = 1.15$
  - Outer shift

- **LHD**
  - $A_p \sim 6$

- **FFHR2m2**
  - $A_p \sim 6$
  - $\gamma = 1.25$
  - Inner shift

**optimum?**
Issues on blanket and divertor space

To remove the interference between the first walls and the ergodic layers surrounding the last closed flux surface, helical x-point divertor (HXD) has been proposed.


However

For $\alpha$-heating efficiency over 90%, the importance of the ergodic layers has been found by collisionless orbits simulation of 3.52MeV alpha particles.

By T. Watanabe

COLLISION–LESS LOSS RATE OF 3.52 MEV ALPHA
D-T $\alpha$ orbits extending to the chaotic field line region.

Poincare plot of helically trapped $\alpha$ particles (magenta dots) and the chaotic field lines (sky blue dots).

3D view of the particle orbits and the last closed flux surface (painted by sky blue).
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.

K. Saito et al., in this conference.

FFHR-2S Type-I

ICRF antenna

Smaller size and higher field with $\gamma = 1$
(reduction of total mass)
3. Enlargement of reactor size

Neutron wall loading should be kept at \(< 2 \text{ MW/m}^2\)

**Three candidates are proposed to increase blanket space > 1.1 m**

\[
R_p = 16 \text{m} \quad (R_c = 17.3 \text{m}) \quad \text{is selected.}
\]

(simple expansion of LHD gives too large \(R_c\))

\[\Delta_{HV-VV} : 110 \text{ mm (fixed)}\]

- FFHR-2S (\(\gamma = 1.0\))
  - \(B_o = 6.18 \text{T}\)
  - \(j_{HC} = 26.6 \text{ A/mm}^2\)
- FFHR-2m2 (\(\gamma = 1.20\))
  - \(B_o = 4.5 \text{T}\)
  - \(j_{HC} = 25.0 \text{ A/mm}^2\)
- LHD-similar (\(\gamma = 1.25\))
  - \(j_{HC} \propto 1/R\)
  - \(W/H = 1.5\)

**A. Sagara et al., in ISFNT-8, FED 83 (2008) 1690.**
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Design windows and cost

Y. Kozaki et al., in this conference.

\[ \beta = 5\%, \ \gamma = 1.20, \ j = 26A/mm^2 \] are selected

The design windows limited with \( \Delta d \geq 1.1m, \ H_f \leq 1.16, \ W < 160GJ \), depending on \( \gamma \) and \( \beta \). \( H_f = 1.16 \) means the 1.2 times value achieved in LHD experiment. \( j = 26A/mm^2 \) is premised.

The COEs of helical reactors, which depend on \( Rp, \gamma \) and \( \beta \), show the bottom as the result of the trade-off between the Cmag and Cbs, i.e., B0 versus plasma volume.
Issues on nuclear shield on SC coils

Discrete Pumping with Semi-closed Shield (DPSS) has been proposed

A. Sagara et al., in ISFNT-8, FED 83 (2008) 1690.

Acceptable level achieved
✓ Cover rate > 90%
✓ Fast neutron < 1E22 n/m² in 30 years
✓ Max. nuclear heating < 0.2 mW/cm³
✓ Total nuclear heating ~ 40 kW
Cryogenics power ~ 12 MW (1% of Pₚ)

3D design with DPSS

Cryogenics power ~ 12 MW (1% of Pₚ)
Reactor size optimization of FFHR2m2

- Inner shifted (equivalent to $R_{ax}=3.6\,\text{m}$ in LHD),
- $\gamma = 1.20$, $\alpha = +0.1$
- $R_p=16\,\text{m}$, $R_c=17.3\,\text{m}$, $B_0=4.84\,\text{T}$, $j=26\,\text{A/mm}^2$, $W_{\text{mag}}=167.7\,\text{GJ}$

Vacuum Magnetic Surface

PC position

Type A

PC position

(a) $\phi = 0^\circ$

(b) $\phi = 18^\circ$

$R_p$, $R_b$, $R_c$

Expanded

$R_{PC} = 8.2\,\text{m}$, $HC : 38.72\,\text{MA}$, $OV : -23.47\,\text{MA}$, $IV : -17.04\,\text{MA}$
Reactor size optimization of FFHR2m2

- inner shifted (equivalent to $R_{ax}=3.6m$ in LHD),
- $\gamma = 1.20$, $\alpha = +0.1$
- $R_p=16m$, $R_c=17.3m$, $B_0=4.84T$, $j=26A/mm^2$, $W_{mag}=149.1GJ$

Vacuum Magnetic Surface

PC position

Type B

(a) $\phi = 0^\circ$

(b) $\phi = 18^\circ$

Reactor size optimization of FFHR2m2

- inner shifted (equivalent to $R_{ax}=3.6m$ in LHD),
- $\gamma = 1.20$, $\alpha = 0.1$
- $R_p=15.5m$, $R_c=16.7m$, $B_0=4.90T$, $j=25A/mm^2$, $W_{mag}=135.6GJ$

Vacuum Magnetic Surface

(a) $\phi = 0^\circ$

(b) $\phi = 18^\circ$ PC position Type B’

WC shield at inboard

HC : 37.87 MA, OV : -18.01 MA, IV : -14.79 MA
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New control method in a thermally unstable ignition 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.

\[
S_{DT}(t) = S_{DT0} \left\{ e_{DT}(P_f) + \frac{1}{T_{int}} \int_{0}^{t} e_{DT}(P_f) dt + T_d \frac{de_{DT}(P_f)}{dt} \right\} G_f(t)
\]

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


\[ \tau_{\alpha}/\tau_E = 3 \sim 5 \]
\[ \tau_p*/\tau_E = 2 \sim 8 \]

But, effectively reduced due to burning.
### Design parameters

<table>
<thead>
<tr>
<th>Design parameters</th>
<th>LHD</th>
<th>FFHR2</th>
<th>FFHR2m</th>
<th>FFHR2m2</th>
<th>SDC</th>
</tr>
</thead>
<tbody>
<tr>
<td>Polarity</td>
<td>I</td>
<td>2</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>
<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>
<td>1.20</td>
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<tr>
<td>Coil major radius $R_c$</td>
<td>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 $a_c$</td>
<td>m</td>
<td>0.98</td>
<td>2.3</td>
<td>3.22</td>
<td>4.16</td>
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<tr>
<td>Plasma major radius $R_p$</td>
<td>m</td>
<td>3.75</td>
<td>10</td>
<td>14.0</td>
<td>16.0</td>
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<tr>
<td>Plasma radius $&lt;a_p&gt;$</td>
<td>m</td>
<td>0.61</td>
<td>1.24</td>
<td>1.73</td>
<td>2.35</td>
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<tr>
<td>Plasma volume $V_p$</td>
<td>m$^3$</td>
<td>30</td>
<td>303</td>
<td>827</td>
<td>1744</td>
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<tr>
<td>Blanket space $\Delta$</td>
<td>m</td>
<td>0.12</td>
<td>0.7</td>
<td>1.1</td>
<td>1.05</td>
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<tr>
<td>Magnetic field $B_0$</td>
<td>T</td>
<td>4</td>
<td>10</td>
<td>6.18</td>
<td>4.84</td>
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<tr>
<td>Max. field on coils $B_{\text{max}}$</td>
<td>T</td>
<td>9.2</td>
<td>14.8</td>
<td>13.3</td>
<td>11.9</td>
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<td>Magnetic energy $G_J$</td>
<td>GJ</td>
<td>1.64</td>
<td>147</td>
<td>133</td>
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<td>Fusion power $P_F$</td>
<td>GW</td>
<td>1</td>
<td>1.9</td>
<td>3</td>
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<td>Neutron wall load $\Gamma_n$</td>
<td>W/m$^2$</td>
<td>1.5</td>
<td>1.5</td>
<td>1.5</td>
<td></td>
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<tr>
<td>External heating power $P_{\text{ext}}$</td>
<td>MW</td>
<td>70</td>
<td>80</td>
<td>43</td>
<td>100</td>
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<td>$\alpha$ heating efficiency $\eta_\alpha$</td>
<td></td>
<td>0.7</td>
<td>0.9</td>
<td>0.9</td>
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<tr>
<td>Density lim. improvement</td>
<td></td>
<td>1</td>
<td>1.5</td>
<td>1.5</td>
<td>7.5</td>
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<td>H factor of ISS95</td>
<td></td>
<td>2.40</td>
<td>1.92</td>
<td>1.92</td>
<td>1.60</td>
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<tr>
<td>Effective ion charge $Z_{\text{eff}}$</td>
<td></td>
<td>1.40</td>
<td>1.34</td>
<td>1.48</td>
<td>1.55</td>
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<tr>
<td>Electron density $n_e(0)$</td>
<td>m$^{-3}$</td>
<td>27.4</td>
<td>26.7</td>
<td>17.9</td>
<td>83.0</td>
</tr>
<tr>
<td>Temperature $T_i(0)$</td>
<td>keV</td>
<td>21</td>
<td>15.8</td>
<td>18</td>
<td>6.33</td>
</tr>
<tr>
<td>Plasma beta $&lt;\beta&gt;$</td>
<td>%</td>
<td>1.6</td>
<td>3.0</td>
<td>4.40</td>
<td>3.35</td>
</tr>
<tr>
<td>Plasma conduction lo $P_L$</td>
<td>MW</td>
<td>290</td>
<td>453</td>
<td>115</td>
<td></td>
</tr>
<tr>
<td>Diverter heat load $\Gamma_{\text{div}}$</td>
<td>W/m$^2$</td>
<td>1.6</td>
<td>2.3</td>
<td>0.6</td>
<td></td>
</tr>
<tr>
<td>Total capital cost</td>
<td>G$(2003)$</td>
<td>4.6</td>
<td>5.6</td>
<td>7.0</td>
<td></td>
</tr>
<tr>
<td>COE</td>
<td>mill/kWh</td>
<td>155</td>
<td>106</td>
<td>93</td>
<td></td>
</tr>
</tbody>
</table>

In SDC, divertor heat load is drastically reduced (~1/4).
Presentation outline

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4. Concluding remarks
Base design of CICC magnet system

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.**

By T. Tanaka

Indirect-cooled helical coil system
(alternative for CICC)
with quench protection by internal dumping

- Quench back with a secondary circuit
- to increase a decay time constant > t=20 s
- to reduce a transient voltage $V_{\text{max}}=10$ kV
- to avoid a serious hot spot < 150 K

K. Takahata et al., ITC-17.

Stress analysis inside of the coil

H. Tamura et al., in this conference.

Cross-sectional structure of the helical coil

- Indirect Cooling

SS316
Coil case
Superconductor
Cooling panel

1.8 m

Outside

0.9 m

Inside

Circumferential strain distribution

Hoop stress distribution

- Confirmed to be within the permissible range.
LHD-type support post for FFHR

By H. Tamura

- 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)
Wide maintenance ports
Blanket system integration by broad R&D collaboration activities

Fusion Research Network

FFHR design / System Integration / Replacement
A. Sagara (NIFS)

Blanket material system
T. Muroga, T. Nagasaka, M. Kondo (NIFS)

Neutronics
T. Tanaka, A. Sagara (NIFS)

Mag. material system
A. Nishimura, Y. Hishinuma (NIFS)

Core Plasma
Surface heat flux

Structural Materials

Ignition access & heat flux
O. Mitarai (Kyusyu Tokai Univ.)

Magnetic structure
T. Morisaki, Yanagi (NIFS)

Helical core plasma
K. Yamazaki (NIFS)

SC magnet & support
S. Imagawa, T. Mito, Takahata, Yamada, Tamura, (NIFS)

Virtual Reality tool
N. Mizuguchi (NIFS)

Power supply
H. Chikaraishi (NIFS)

External heating
O. Kaneko, Igami (NIFS)

Fusion Sakamoto (NIFS)

Fueling
Sakamoto (NIFS)

Advanced first wall
T. Norimatsu (Osaka Univ.)

MHD on Flibe

Fluid-film diffusion + permeation
ferrite-steel permeation window
T=843K, QT=190g-T/day, WFlibe=2.3m3/s

Permeation window area [m2] (α=1, t=1mm)

Tritium leak
Large apparatus

Cost evaluation
Y. Kozaki (NIFS)

Safety
T. Uda (NIFS)

Fusion Eng. R.C.

Blanket system
T. Terai (Univ. of Tokyo)

Heat exchanger & gas turbine system
A. Shimizu (Kyusyu Univ.)

MHD on Flibe

Closed He gas turbine cycle with three-stage compression/expansion

TNT Loop: Max. 20L/min @ 600°C

Tritium leak:

Ferrit-steel permeation window
T=843K, Q=190g/day, Wfib=2.3m3/s

Fluid-film diffusion controlling

Permeation controlling

Large apparatus

Permeation window area [m2] (α=1, t=1mm)
Concluding remarks

1. Helical reactor is superior in steady state operation and
   - Reduced neutron wall loading < 2MW/m² with long-life blanket concept,
   - High density operation with reduced heat load on divertor.

2. On the 3D design, a simply defined blanket shape can be compatible with the ergodized magnetic layer which is essential on both of divertor and α-heating.

3. This simple methodology on LHD-type reactors can be used for design optimization towards DEMO, which is economically comparable to Tokamak.

4. The design parameters of FFHR2m2 have been totally improved, where a new ignition regime at SDC is an innovative alternative.

5. The large scale SC magnet system and their support posts are conceptually feasible.

6. Large R&D progress has been made on blanket and magnet materials.

7. The next key work is to realize the DPSS divertor concept compatible with replacement scenario and neutronics issues on TBR and nuclear shielding for SC magnets.