Present activities on LHD-type Reactor Designs FFHR

Akio SAGARA,
National Institute for Fusion Science, Japan

Contributed with
S. Imagawa, O. Mitarai, T. Dolan, T. Tanaka,
Y. Kubota, K. Yamazaki, K. Y. Watanabe, N. Mizuguchi,
T. Muroga, N. Noda, O. Kaneko, H. Yamada, N. Ohyabu,
T. Uda, A. Komori, S. Sudo, and O. Motojima

Japan-US Workshop on
Fusion Power Plants and Related Advanced Technologies
with participation of EU
January 11-13, 2005 at Tokyo, JAPAN
1. Introduction of FFHR

2. New design approach
   - The size is increased
   - Why?

3. Conclusions

Presented in 20th IAEA Fusion Energy Conference
1 - 6 November 2004
Vilamoura, Portugal
プラズマ保持時間 (秒)

入力エネルギー値

Injected Energy (GJ)

Plasma Duration [sec]
Reactor design collaborations in NIFS

- Helical core plasma
  K. Yamazaki (NIFS)

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

- Advanced first wall
  T. Norimatsu (Osaka Univ.)

- Advanced thermo-fluid
  T. Kunugi (Kyoto Univ.)

- Thermo-fluid
  MHD
  S. Satake (Tokyo Univ. Sci.)

- Blanket system
  S. Tanaka (Univ. of Tokyo)

- Safety & Cost

- Blanket

- Helical reactor design / System Integration
  A. Sagara (NIFS)

- Device system code
  H. Hashizume (Tohoku Univ.)

- T-disengager system
  S. Fukada, M. Nishikawa (Kyusyu Univ.)

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

- June 6, 2003, A. Sagara
Tritium recovery systems
Kyushyu Univ.: S.Fukada

Permeation leak through the recovery system is a crucial problem

- Small amount of Flibe or He gas flow in the double tube are good as permeation barrier to reduce < 10Ci/day.

- The most serious problem is permeation leak of ~34 kCi/day through the heat exchanger to the He loop.

**FFHR tritium recovery system (1GWth)**

Tritium : 190 g-T/day = 1.8 MCi/day

- Double tube
- Flibe blanket
- Pump
  - Flow rate 2.3m³/s
- Heat exchanger
  - Permeation barrier < 10Ci/day
- Permeation barrier
- Tritium storage
- Tritium recovery
- First wall
- LCFS
- SOL
- Thermal shield
- Self-cooled T breeder
- Radiant shield
- Vacuum vessel
- SC magnet
Energy conversion systems
for Flibe in/out temperature of 450°C and 550°C
Kyushyu Univ.: A. Shimizu

- $\eta_{\text{max}} \sim 37\%$ for compression ratio of 1.5,
- However, $\eta_{\text{max}}$ decreases rapidly with the increase of pressure drop.
- Therefore the layout of energy conversion system is a key design issue.

Three-stage compression-expansion He-GT system was newly proposed
Micro grooves are made on the first wall to use capillary force to withstand the gravity force.

Numerical simulation has explored the formation of a pair of symmetrical spiral flow,

which enhances heat transfer efficiency about one order.
Thermofluid R&D activities
Tohoku Univ.: H.Hashizume
for enhancing heat-transfer in such
high Prandtl-number fluid as Flibe

➢ “TNT loop” (Tohoku-NIFS Thermofluid loop) has been operated using HTS (Heat Transfer Salt, $T_m = 142^\circ C$)

➢ Results are converted into Flibe case at the same $Pr=28.5$ ($T_{in}=200^\circ C$ for HTS and $536^\circ C$ for Flibe)

➢ Same performance as turbulent flow is obtained at one order lower flow rate.

➢ This is a big advantage for MHD effects and the pumping power.
Self-cooled Be-free Li/V blanket

NIFS : T.Tanaka, T.Muroga
based on R&D progress on in-situ MHD coatings and high purity V fabrication

- Simple models are evaluated as alternatives for FFHR2 blanket.
- Balance of TBR and the shielding performance is examined, because shielding is poor w/o Be.
- TBR of Li/V is higher than 1.3 at about 50 cm with an acceptable shielding efficiency for superconducting magnets.
Modeling to Evaluate MHD pressure drop is established for self-cooled lithium blanket.

Tohoku Univ. : H.Hashizume

- Three-layered wall is proposed, where the inner thin metal layer protects permeation of lithium into the crack of coated layer
- Extremely good agreement between FEM and theory has been obtained

The performance required to the insulator is evaluated to be

\[
\frac{\sigma_{\text{insulator}}}{\sigma_V} \approx 10^{-8} - 10^{-9}
\]
Present R & D activities on Flibe blanket in Japan

Presented by A. Sagara (NIFS), Feb.’04


- Helical reactor
  - FFHR design
  - with R&D

- R&D/LHD
  - TNT loop
  - Ultrahigh HT

- JUPITER-II

- ITER-TBM
  - Frame?
  - Resouce?
Flibe/RAF blanket, Li/V blanket, SB-He/SiC Blanket R&D in Japan-US joint project JUPITER-II (FY’01~’06)

INNEEL
1-1-A: FLiBe Handling/Tritium Chemistry
1-1-B: FLiBe Thermofluid Flow Simulation
2-2 : SiC System Thermomechanics

Japan

3-1: Design-based Integration Modeling
3-2: Materials Systems Modeling

UCLA

ORNL
1-2-B: V Alloy Capsule Irradiation
2-1 : SiC Fundamental Issues, Fabrication, and Materials Supply
2-3 : SiC Capsule Irradiation
1-2-A: Coatings for MHD Reduction

http://jupiter2.iae.kyoto-u.ac.jp/index-j.html
Many advantages:
- Current-less
- Steady state
- No current drive power
- Intrinsic divertor

LHD operation: 1998 ~

Selection of lower

- To reduce mag. foop force
- To expand blanket space

1993 FFHR-1 (l=3, m=18)
  \[ R=20, B_t=12T, \beta=0.7\% \]

1995 FFHR-2 (l=2, m=10)
  \[ R=10, B_t=10T, \beta=1.8\% \]
However, direction of compact design has engineering issues

- Insufficient tritium breeding ratio (TBR)
- Insufficient nuclear shielding for superconducting (SC) magnets,
- Replacement of blanket due to high neutron wall loading
- Narrowed maintenance ports due to the support structure for high magnetic field
New design approach is proposed to overcome all these issues

- Introducing a long–life & thicker breeder blanket
- Increasing the reactor size with decreasing the magnetic field
- Improving the coils-support structure
Proposal and Optimization of STB (Spectral-shifter and Tritium breeder Blanket)

• Lifetime of Flibe/RAFS liquid blanket in FFHR ~ 15MWa/m²

• Neutron wall loading in FFHR2 in 30 years 1.5MW/m² x 30y = 45MWa/m²

Neutron wall loading factor 3 operation

Carbon First wall Breeder

Fast neutron this work

ISSEC : by Kulchinski, ‘75

Reduced Activation Ferritic Steel

Thermal creep (1% creep strain under applied stress σ/3)

Void swelling (>1%)

He effect (?)

Reduced Activation Ferritic Steel operation

He effect (?)

DBTT increase by irradiation

0 2 4 6 8 10 12 14 16 18 20
Neutron wall loading MWa/m²

0 200 400 600 800 1000 1200 1400
Temperature °C
**Tritium breeding**

- Design point: 2nd C = 80 mm
- Thickness of Be\(_2\)C: 100 mm

**Shielding efficiency**

- Fast Neutron Flux: \(x 10^{18}/m^2s\)
- Neutron energy: MeV
- Neutron flux: \(n/m^2s/\text{lethargy}\)

**Graphs:**

- Local TBR vs. thickness of Be\(_2\)C (mm)
- Fast Neutron Flux vs. thickness of 2nd carbon (mm)
- Neutron flux vs. Neutron energy (MeV)
Results and Key R&D Issues

Results:

✓ Fast neutron flux at the first wall is factor 3 reduced.
✓ Local TBR > 1.2 is possible.
✓ Fast neutron fluence to SC is reduced to 5x10^{22}n/m^2 (Tc/Tco>90% in 30y)
✓ Surface temperature < 2000°C (~mPa of C) is possible

Key R&D issues:

(1) Impurity shielding in edge plasma
(2) Neutron irradiation effects on tiles at high temperature
(3) Heat transfer enhancement in Flibe flow

Required conditions:

- Neutron wall loading < 1.5MW/m^2
- Blanket thickness > 1100 mm
- λ for C-Be-C tiles > 100W/mK
- Super-G sheet for tiles joint > 6kW/m^2K
- Heat removal by Flibe > 1MW/m^2
## Improved design parameters

<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>l</td>
<td>2</td>
<td>2</td>
<td>2</td>
</tr>
<tr>
<td>Field periods</td>
<td>m</td>
<td>10</td>
<td>10</td>
<td>10</td>
</tr>
<tr>
<td>Coil pitch parameter</td>
<td>γ</td>
<td>1.25</td>
<td>1.15</td>
<td>1.15</td>
</tr>
<tr>
<td>Coils major Radius</td>
<td>Rc m</td>
<td>3.9</td>
<td>10</td>
<td>14.0</td>
</tr>
<tr>
<td>Coils minor radius</td>
<td>ac m</td>
<td>0.98</td>
<td>2.3</td>
<td>3.22</td>
</tr>
<tr>
<td>Plasma major radius</td>
<td>Rp m</td>
<td>3.75</td>
<td>10</td>
<td>14.0</td>
</tr>
<tr>
<td>Plasma radius</td>
<td>ap m</td>
<td>0.61</td>
<td>1.2</td>
<td>1.73</td>
</tr>
<tr>
<td>Blanket space</td>
<td>Δ m</td>
<td>0.12</td>
<td>0.7</td>
<td>1.2</td>
</tr>
<tr>
<td>Magnetic field</td>
<td>B0 T</td>
<td>4</td>
<td>10</td>
<td>6.18</td>
</tr>
<tr>
<td>Max. field on coils</td>
<td>Bmax T</td>
<td>9.2</td>
<td>13</td>
<td>13.3</td>
</tr>
<tr>
<td>Coils current density</td>
<td>j MA/m2</td>
<td>53</td>
<td>25</td>
<td>26.6</td>
</tr>
<tr>
<td>Weight of support</td>
<td>ton</td>
<td>400</td>
<td>2880</td>
<td>3020</td>
</tr>
<tr>
<td>Magnetic energy</td>
<td>GJ</td>
<td>1.64</td>
<td>147</td>
<td>120</td>
</tr>
<tr>
<td>Fusion power</td>
<td>PF GW</td>
<td>1.77</td>
<td>1.9</td>
<td>3.0</td>
</tr>
<tr>
<td>Neutron wall load</td>
<td>MW/m2</td>
<td>2.8</td>
<td>1.5</td>
<td>1.3</td>
</tr>
<tr>
<td>External heating power</td>
<td>Pext</td>
<td>100</td>
<td>80</td>
<td>100</td>
</tr>
<tr>
<td>α heating efficiency</td>
<td>ηα</td>
<td>0.7</td>
<td>0.9</td>
<td>0.9</td>
</tr>
<tr>
<td>Density lim.improvement</td>
<td></td>
<td>1</td>
<td>1.5</td>
<td>1.5</td>
</tr>
<tr>
<td>H factor of ISS95</td>
<td></td>
<td>2.53</td>
<td>1.92</td>
<td>1.68</td>
</tr>
<tr>
<td>Effective ion charge</td>
<td>Zeff</td>
<td>1.32</td>
<td>1.34</td>
<td>1.35</td>
</tr>
<tr>
<td>Electron density</td>
<td>ne(0) 10^19 m-3</td>
<td>28.0</td>
<td>26.7</td>
<td>19.0</td>
</tr>
<tr>
<td>Temperature</td>
<td>Ti(0) keV</td>
<td>27</td>
<td>15.8</td>
<td>16.1</td>
</tr>
<tr>
<td>&lt;β&gt;=2<em>n</em>T/(B^2/μ) (parabolic distribution)</td>
<td></td>
<td>1.8</td>
<td>3.0</td>
<td>4.1</td>
</tr>
<tr>
<td>COE</td>
<td>Yen/kWh</td>
<td>21.00</td>
<td>14.00</td>
<td>9.00</td>
</tr>
</tbody>
</table>

\[ \tau_{ISS95} = 0.26P^{-0.59−0.51}n_e^{-0.83}R^{-0.65}a^{-0.21}r_{2/3}^{-0.4} \]
Self-ignition access in FFHR2m1

Zero-dimensional analysis

- $H_{ISS95} = 1.2 \times 1.6$
- $\frac{\tau^*_{\alpha}}{\tau_E} = 3\; (< 7)$
- parabolic profiles
- $n_e < 1.5 \times $ Sudo limit
- $\alpha$ heating efficiency = 0.9

*O.Mitarai et al., Fusion Eng. Design 70 (‘04) 247.*

Self-ignition access in FFHR2m2

Zero-dimensional analysis
- $H_{ISS95} = 1.1 \times 1.6$
- $\tau^*_{\alpha} / \tau_E = 3 ( < 7 )$
- parabolic profiles
- $n_e < 1.5 \times Sudo\ limit$
- $\alpha$ heating efficiency = 0.9

Improved Design of
Coil-supporting Structure (1/2)

- Cylindrical supporting structure under reduced magnetic force, which facilitates expansion of the maintenance ports.
- Helical coils supported at inner, outer and bottom only.

- W/H = 2 & H/a_c determined by
- B_{max} \sim 13 \text{T} for such as Nb_3Sn or Nb_3Al.
- Then J=25\sim35\text{A/mm}^2
- Poloidal coils layout as for stored magnetic energy and stray field

![Diagram of cylindrical supporting structure with helical coils supported at inner, outer, and bottom.

Graph showing stored energy (GJ) vs. Z of IV Coil (m) with R_{IV}=9.5\text{m}, Z_{OV}=3.6\text{m}.]
**Improved Design of Coil-supporting Structure (2/2)**

- The maximum stress can be reduced less than 1000 MPa (<1.5Sm)
- This value is allowable for strengthened stainless steel.

Electromagnetic forces on a helical coil of FFHR2m1.

- Coil current (MA)
  - HC 43.257
  - OV -21.725
  - IV -22.100

Electromagnetic forces on poloidal coils
Replacement of In-Vessel Components (1/2)

- **Large size maintenance ports** at top, bottom, outer and inner sides.
- **The vacuum boundary** located just inside of the helical coils and supporting structure.
- **Blanket units** supported on the permanent shielding structures, which are mainly supported at their helical bottom position.
Replacement of In-Vessel Components (2/2)

Proposal of the “Screw coaster” concept to replace STB armor tiles

- Replacement of bolted tiles during the planned inspection period.
- Using the merit of helical structure, where the normal cross section of blanket is constant.
- Toroidal effects can be adjusted with flexible actuators.
Cost Estimation
Using PEC code developed in NIFS.
Calibrated with ARIES-AT, ARIES-SPPS, resulting in good agreement within 5%.
(T.J.Dolan, K. Yamazaki, A. Sagara, in press in Fusion Science & Tech.)

- COE’s for FFHR2, FFHR2m1 and FFHR2m2 decreases with increasing the reactor size, because the fusion output increases in ~ R^2, while the weight of coil supporting structure increases in ~ R^{0.4} (not ~ R^3).

- When the blanket lifetime ~ 30y, the COE decreases ~20% due to higher availability and lower cost for replacement.
Conclusions

Design studies on FFHR have focused on new design approaches to solve the key engineering issues of blanket space limitation and replacement difficulty.

(1) The combination of improved support structure and long-life breeder blanket STB is quite successful.

(2) The “screw coaster” concept is advantageous in heliotron reactors to replace in-vessel components.

(3) The COE can be largely reduced by those improved designs.

(4) The key R&D issues to develop the STB concept are elucidated.