Design and R&D Activities for the LHD-type Demo FFHR-d1

Akio Sagara
Executive Director of Fusion Engineering Research Project
National Institute for Fusion Science

Fusion Power Plants and Related Advanced Technologies
March 8-9, 2012 at UCSD

Key subjects are

Design of React & Wind large size, high field SC helical coil with CIC conductors.

Long life blanket with inherent safety and wide maintenance ports

Step by step advancement of reactor design

Steady-state helical DEMO reactor FFHR-d1

High density self-ignition, heating, fueling and diverter pumping

TBR and shielding with 3D neutronics for non-axisymmetric helical systems

Sagara- 2/32
Fusion Eng. Research Project includes R&D activities towards steady-state helical DEMO reactor

Based on the 1st mid-term results (2004~2009)

Developments of
(a) Nb₃Sn or Nb₃Al superconductor with Al-alloy jacket,
(b) 10 kA-class high-temp. superconductor HTS.

Reduction of corrosion and T permeation barrier for liquid blanket (with U. of Tokyo.)

Recovery of welded V-alloy from neutron damage by heating.

W coating for plasma facing wall by plasma spray (with Kyoto Univ.)

Steady-state helical DEMO reactor FFHR-d1

High-sensitive tritium Monitor (with Nagoya Univ.)

Sagara- 3/32
13 tasks and 44 sub-tasks for Design and R&D with collaborations

**Promotion meeting by Exe. Dir. Sagara & Dirs. Imagawa, Muroga, Task leaders**

- Helical reactor conceptual design
- Helical DEMO basic design
- Testing of full-scale SC conductor
- Helical winding engineering
- Testing for lifetime expansion of liquid blanket
- Thermo-fluid dynamics under high magnetic field
- Test fabrication of high temperature low activation material
- Surface modification for heat-resistance
- Prototype testing of 3D divertor
- Hydrogen retention in LHD irradiation
- Removal and recovery of scrape tritium
- Development of Real-time detection system

**Reactor system design group : Sagara / Task/Sub task**

<table>
<thead>
<tr>
<th>Design Integration</th>
<th>Task setting, Project management</th>
<th>Sagara, Miyazawa, T. Goto</th>
</tr>
</thead>
<tbody>
<tr>
<td>Building layout</td>
<td>Layout design, process</td>
<td>T. Goto, Miyazawa, Sagara</td>
</tr>
<tr>
<td>T. Goto</td>
<td>Reactor building design</td>
<td>Tamura, T. Goto</td>
</tr>
<tr>
<td>Power supply, Generator</td>
<td>Power supply system</td>
<td>Chikaraishi, S. Yamada</td>
</tr>
<tr>
<td>Chikaraishi</td>
<td>Transmission, H production</td>
<td>S. Yamada, Hoshinuma</td>
</tr>
<tr>
<td>T. Tanaka</td>
<td>Tritium processing system</td>
<td>M. Tanaka</td>
</tr>
<tr>
<td>M. Tanaka</td>
<td>Safety control</td>
<td>Kawano</td>
</tr>
<tr>
<td></td>
<td>Bioshield-Radioactivation</td>
<td>Yamanishi</td>
</tr>
<tr>
<td></td>
<td>Legislation, Licensing</td>
<td>S. Nishimura</td>
</tr>
<tr>
<td>Mitsarai (Tokai Univ.)</td>
<td>Safety analysis, control system</td>
<td>Uda</td>
</tr>
<tr>
<td></td>
<td>Burn control</td>
<td>Mitsarai</td>
</tr>
<tr>
<td></td>
<td>Data processing</td>
<td>Nakanishi</td>
</tr>
<tr>
<td>Miyazawa</td>
<td>High performance plasma</td>
<td>Miyazawa, T. Goto, Narusim</td>
</tr>
<tr>
<td></td>
<td>TCT effect, &amp; particle loss</td>
<td>Yokoyama, Murakami (Kyoto U)</td>
</tr>
<tr>
<td></td>
<td>Ignition Scenario</td>
<td>Mitsarai</td>
</tr>
<tr>
<td>Tsunori</td>
<td>NBI</td>
<td>Tsumori, Osakabe</td>
</tr>
<tr>
<td></td>
<td>ECH</td>
<td>Iizumi, Yoshimura, Idei (Kyusyu U), Shimozuma</td>
</tr>
<tr>
<td></td>
<td>ICH</td>
<td>Kasahara, Saito, Muto</td>
</tr>
<tr>
<td>Sakamoto</td>
<td>Pellet</td>
<td>Sakamoto</td>
</tr>
<tr>
<td></td>
<td>Gas-puff</td>
<td>Miyazawa</td>
</tr>
<tr>
<td>Isobe</td>
<td>Magnetic diagnostics</td>
<td>Sakikakura</td>
</tr>
<tr>
<td></td>
<td>Neutron diagnostics</td>
<td>Isobe</td>
</tr>
<tr>
<td></td>
<td>Divertor diagnostics</td>
<td>Masuzaki</td>
</tr>
<tr>
<td></td>
<td>Spectroscopic diagnostics</td>
<td>M. Goto</td>
</tr>
<tr>
<td></td>
<td>Interferometer / reflectometer</td>
<td>K. Tanaka, Tokuzawa, Akiyama</td>
</tr>
<tr>
<td></td>
<td>Thomson scattering</td>
<td>H. Yamada</td>
</tr>
<tr>
<td></td>
<td>Charge exchange spectroscopy</td>
<td>Yoshinuma</td>
</tr>
</tbody>
</table>

**Rev. 2010.5.17**

Sagara- 4/32
Mid-term plan towards helical DEMO

2010
The 2nd mid-term
Step by step advancement of reactor design
Conceptual design → Basic design → Improved basic design

2016
The 3rd

2022
2027
2036FY

Establishment of engineering base

Full-scale, full-condition testing

With universities

Engineering design
Construction Licensing Operation

Large-scale high-field superconducting magnet

Low activation structural materials

Tritium control

Long-life liquid blanket

High heat flux plasma facing wall
FFHR-d1
Re-design, enhancing robustness, construction and safety

Based on FFHR2m1

Superconducting helical coils
Superconducting poloidal coils
Vacuum vessel
Support structure
Blanket
Cryostat
Core plasma
Support post

$R_{ax} = 14\ m$

Radial build between coil and plasma

This space includes:
- helical coil bottom plate
- thermal insulation gap
- vacuum vessel
- blanket system
- plasma-FW clearance

\( \Delta_{c-p} \)
1st cycle
~Decision of main design parameters~

Extrapolation of LHD exp. results (ex. #96164 @6.966s)

Design integration TG
- Reconsideration of the design window

Superconducting coil TG
- Consideration of the maximum allowable nuclear heating in magnets (cooling capability calculation)

In-vessel components TG
- Consideration of radial build

Blanket TG
- Optimization of breeding/shielding layer thickness (neutronics calc.)

$R_c=15.6\,\text{m}$, $B_c=4.7\,\text{T}$,
$\Delta_{c-p}=89\,\text{cm}$, $<I_{nw}>=1.5\,\text{MW/m}^2$

$\Delta_{blk}=70\,\text{cm}$

Full cover $TBR=1.27$
$q=0.5\,\text{mW/cc}$
Direct Profile Extrapolation from LHD data

J. Miyazawa et al., FED, 86 (2011) 2879.

DPE enables “deductive” fusion reactor design, where the plasma performance determines engineering parameters.

Radial profiles in experiment are directly extrapolated to reactor

The reactor size is determined as a function of magnetic field strength
1. Required confinement improvement can be relaxed by **reducing the minimum inboard blanket space** without the increase in the stored magnetic energy.

2. The blanket space can be reduced by the use of **advanced shielding material**.

---

T. Goto et al., 21st Int. Toki Conf., 2011, P2-94, and submitted to PFR.

Revised from A. Sagara et al., FED 83 1690 (2008)

Sagara- 11/32
Design of the inboard space $\Delta_{cp}$ (this case 890 mm) to ensure the gap at R.T. (this case 10 mm)

H. Tamura et al., in ITC-21 (2011)
Inboard Breeder: 15 cm WC Shield: 55 cm
(Nuclear heating: \( \sim 0.5 \text{ mW/cm}^3 \), Fast neutron flux: \( \sim 3 \times 10^{10} \text{ n/s/cm}^2 \) at SCM)

Outboard Breeder: 60 cm (Multiplier 15 cm) FS+B\(_4\)C Shield: >50 cm

Neutron wall loading: 1.5 MW/m\(^2\)

TBR > 1.15 with n multiplier and shielding for SC are possible

In case of Li / V, shielding with WC etc. would be necessary.

Simple torus model for neutronics analysis

T. Tanaka et al., submitted to IAEA-FEC24
1-D thermal hydraulic analysis of CIC conductor

There is a possible operation window for CIC conductor with supercritical helium coolant

Acceptable range: 41-47g/s
$\Delta T<1K$, $\Delta P<0.1MPa$

S. Hamaguchi et al., in ITC-21 (2011)
<table>
<thead>
<tr>
<th></th>
<th>LHD</th>
<th>FFHR2</th>
<th>FFHR2m1</th>
<th>FFHR2m2</th>
<th>FFHR-d1</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Coil pitch parameter</td>
<td>$\gamma_c$</td>
<td>1.25</td>
<td>1.15</td>
<td>1.15</td>
<td>1.2</td>
</tr>
<tr>
<td>Coil major radius</td>
<td>$R_c$ m</td>
<td>3.9</td>
<td>10</td>
<td>14.0</td>
<td>17.3</td>
</tr>
<tr>
<td>Plasma major radius</td>
<td>$R_p$ m</td>
<td>3.75</td>
<td>10</td>
<td>14.0</td>
<td>16.0</td>
</tr>
<tr>
<td>Plasma minor radius</td>
<td>$\alpha_p$ m</td>
<td>0.61</td>
<td>1.24</td>
<td>1.73</td>
<td>2.35</td>
</tr>
<tr>
<td>Plasma volume</td>
<td>$V_p$ m$^3$</td>
<td>30</td>
<td>303</td>
<td>827</td>
<td>1744</td>
</tr>
<tr>
<td>Blanket space</td>
<td>$\Delta$ m</td>
<td>0.12</td>
<td>0.7</td>
<td>1.1</td>
<td>1.05</td>
</tr>
<tr>
<td>Magnetic field</td>
<td>$B_0$ T</td>
<td>4</td>
<td>10</td>
<td>6.18</td>
<td>4.84</td>
</tr>
<tr>
<td>Magnetic energy</td>
<td>$W_{mag}$ GJ</td>
<td>1.64</td>
<td>147</td>
<td>133</td>
<td>160</td>
</tr>
<tr>
<td>Fusion power</td>
<td>$P_{fus}$ MW</td>
<td>1</td>
<td>1.9</td>
<td>3</td>
<td>3</td>
</tr>
<tr>
<td>Neutron wall load</td>
<td>$\Gamma_n$ MW/m$^2$</td>
<td>1.5</td>
<td>1.5</td>
<td>1.5</td>
<td>1.5</td>
</tr>
<tr>
<td>$H$ factor of ISS95</td>
<td>$H^{ISS95}$</td>
<td>2.40</td>
<td>1.92</td>
<td>1.92</td>
<td>1.64</td>
</tr>
<tr>
<td>Plasma beta (evaluated with $B_{ax}$)</td>
<td>$&lt;\beta&gt;$ %</td>
<td>1.6</td>
<td>3.0</td>
<td>4.4</td>
<td>3.35</td>
</tr>
<tr>
<td>Divertor heat load ((\Delta 0.1m))</td>
<td>$\Gamma_{div}$ MW/m$^2$</td>
<td>5</td>
<td>7.2</td>
<td>1.9</td>
<td>8.1</td>
</tr>
<tr>
<td>Total capital cost</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></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></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>mill/kWh</td>
<td>155</td>
<td>106</td>
<td>93</td>
<td></td>
</tr>
</tbody>
</table>
2nd cycle
~Design of in-vessel components~

Design integration TG
- Consistency check (divertor trace calculation, basic shape determination)

Blanket TG
- Optimization of 3D blanket shaping (3D neutronics calc.)

Divertor Sub TG (In-vessel components TG)
- Divertor/port design
- Pumping performance calc.
[1] Radiation shielding

Evaluated shielding performance

(a) WC shield
(b) WC + B$_4$C shield

Configuration of inboard radiation shield
(SCM*: Superconducting Magnet)

- Main shielding material: WC
- Coolant: Water or Helium

- The blanket space at the inboard side could be reduced to 70 cm for FFHR-d1 (DEMO).
- Fully-covered TBR: ~1.3* (with Flibe+Be/Ferritic Steel blanket)

*N. Sagara et al., in this conference.

Fast neutron flux (>0.1 MeV) at SCM

Nuclear heating at SCM (Nb$_3$Al)
[2] Support of helical blanket  

**Thickness of vacuum vessel in FFHR**

- **Vacuum vessel (5 cm thick)**

- **Thickness of vacuum vessel in FFHR2 design:** ~5 cm

  - *It would be difficult to support all the blanket component by the thin vacuum vessel.*

  - *Possible to support by radiation shielding layers?*
Two-layered radiation shield

Structure of Flibe+Be/Ferric Steel blanket

- Main materials for outboard shield: Ferric steel and $B_4C$
- A two-layered shield has a similar performance as an uniformly mixed shield.

A ferric steel layer could be used for a main support structure of blanket and shield.
[2] Support of helical blanket  

Mechanical stress on helical support layer

- Mechanical stress calculation for helical radiation shield.
  - 72° section model (1/5 of torus)
  - 20 cm + 10 cm thick; two layers
  - Weight load:
    Own weight + shield (295 t) + blanket (100 t)

- Estimated maximum stress of \(\sim 30 \text{ MPa}\) would be acceptable.
- More detailed investigation and optimization of the structure will be performed.
Size of helical blanket modules

Shape of helical blanket system

- Size of maintenance port has been investigated by mechanical stress analyses** for the support structure of the magnet system.

- Maximum size of helical blanket modules would be ~2.5 m x ~4.5 m.

Heat removal in Flibe blanket

- Heat removal of a Be pebble layer has been investigated in the present study.
- Temperature of ~0.1 m/s Flibe flow in a Be pebble layer increases from 450 °C to 550 °C during go through ~4.5 m long channel.
- Nuclear heating in Be is lower than that in a Flibe coolant.

- In a ~550 °C Flibe coolant, a temperature of Be will be ~700 °C
- Control of heating in a Flibe+Be pebble layer would not be a severe issue.

Geometry and result of heat transfer calculation for Flibe+Be pebble layer ➞
(for neutron wall loading of 1.5 MW/m²)
[3] Cooling of helical blanket

Magnetic field in helical blanket

- Directions of magnetic fields are different in the blanket A and B.
- Optimum flow directions for liquid coolants will be decided by the magnetic field distribution.
  - For Li coolant: Parallel to magnetic fields to reduce MHD pressure drop.
  - For Flibe coolant: Magnetic fields would affect the heat transfer performance*

Magnetic field distribution in helical breeding blanket

1. Burning rate of DT fuel is less than 1%, i.e. burnup/fuel ≈ 1×10^{21} s^{-1}/3×10^{23} s^{-1}.
2. Large amount of DT fuel circulation is required in pellet injector.
3. But, the hydrogen inventory is within tolerance level.
   - 5 g in the gas exhaust loop assuming τ = 1 s for pumping and compression.
   - 80 g in solid hydrogen reservoir assuming τ = 40 s to solidify hydrogen gas.

Minimum requirement of DT fuel is at least 3×10^{23} s^{-1} to sustain burning plasma.

Steady-state evaluation of hydrogen isotope circulation for burning plasma sustainment.
2nd cycle

Joint works between NIFS Projects

1 Proposal of experiment with FFHR-d1 like magnetic configuration

Core plasma TG

Direct Profile Extrapolation

LHD project

Experiment

Numerical project

Physics analysis

4 Detailed physics studies become possible

3 Reconstruction of MHD equilibrium using HINT2 and VMEC

Radial profiles extrapolated from LHD to FFHR-d1

Sagara- 25/32
R&D for DEMO
Option: Helical coil winding by joining HTS conductors

- Large-current high-Tc superconductor (HTS) using YBCO is being developed
- Continuous helical coils by joining half-pitch conductors (0.36 nΩ × 8000 @20 K \(\Rightarrow\) 4.5 MW@R.T.)
- 10-kA conductors with joints have been successful and 100-kA will be tested next year

N. Yanagi et al., Fusion Science and Technology 60 (2011) 648
N. Yanagi, HTS^4Fusion Conductor Workshop (2011) KIT, Germany
Integration for steady-state blanket system in NIFS: Orosh^2i-1 (Operational Recovery Of Separated Hydrogen and Heat Inquiry – 1)

- Forced convection loop up to 873K
- Flinak (LiF + NaF + KF) as a candidate, and as a stimulant of Flibe (LiF + BeF_2)
- Compatibility of hydrogen recovery and heat recovery
1. Thermal creep deformation was suppressed by nano-particle dispersion for reduced activation ferritic/martensitic steels (RAFM) and vanadium alloys.

2. The blanket operation temperature could be enhanced by 100-150K by the used of Dispersion Strengthened Materials.

**Enhancement of Blanket Operation Temperature by Dispersion Strengthening with Nano-Particles**

![Diagram showing comparison of operation temperatures between Low Activation Ferritic Steel (JLF-1) and Vanadium alloy (V-4Cr-4Ti)]

**Ferritic Steel**

**Vanadium alloy**

**Maximum Operation Temperature**
- RAFM: ~550°C
- ODS-Steel: 650~700°C
- V-4Cr-4Ti: ~700°C
- ODS-V alloy: ~800°C

**Fabrication technology, fracture toughness and isotropy of strength are the key issues**
Irradiation effects and recovery on weld joint of NIFS-HEAT-2 V-4Cr-4Ti alloy

- Base metal maintained DBTT below RT after 8.5 dpa irradiation
- Degradation of impact energy and DBTT shift by irradiation was enhanced in weld metal
- Weld metal maintained DBTT below RT after 0.98 dpa irradiation, however impact energy was lost at 8.5 dpa
- The fracture energy was recovered by post-irradiation annealing at 600°C for 1 hr
VPS-W coating on low activation materials

- W coating for the 1st wall was fabricated on various low activation materials by vacuum plasma spray process.

- Substrate temperature was controlled around 550°C for F82H, 700°C for ODS and NH2(V-alloy), respectively.

![Surface after VPS-W coating](image)

![Cross section of VPS-W coating](image)

![W-coated low activation materials](image)
Work Performed:
• Reactor size optimization for blanket space and mag. energy
  ➢ Core plasma design by the Direct Profile Extrapolation from LHD exp.
  ➢ Reduction in blanket thickness with advanced shielding materials

Work in Progress:
• 3D design of in-vessel components consistent with divertor pumping and neutron shielding
• Start-up scenarios of self-ignition plasma and blanket system

Work Planned:
• Large SC magnet system of 3D helical continuous winding
• Large maintenance ports and replacement scenarios
• R&D’s for 3D components mockup with design activities