Optimization Studies on Conceptual Designs of LHD-type Reactor FFHR

A. Sagara[1],

O. Mitarai[2], M. Kobayashi[1], T. Morisaki[1], T. Tanaka[1], S. Imagawa[1], Y. Kozaki[1], S. Masuzaki[1], K. Takahata[1], H. Tamura[1], N. Yanagi[1], K. Nishimura[1], H. Chikaraishi[1], S. Yamada[1], S. Fukada[3], A. Shishkin[4], T. Watanabe[1], Y. Igitkhanov[5], T. Goto[6], Y. Ogawa[6], T. Muroga[1], T. Mito[1], O. Motojima[1] and FFHR design group

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[2]Kyushyu Tokai University,
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[4]Kharkov Institute of Physics and Technology, Ukraine
[5]Max-Planck-Institut für Plasmaphysik, Germany
[6]The University of Tokyo

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Fusion Power Plants and Related Advanced Technologies with participation of EU
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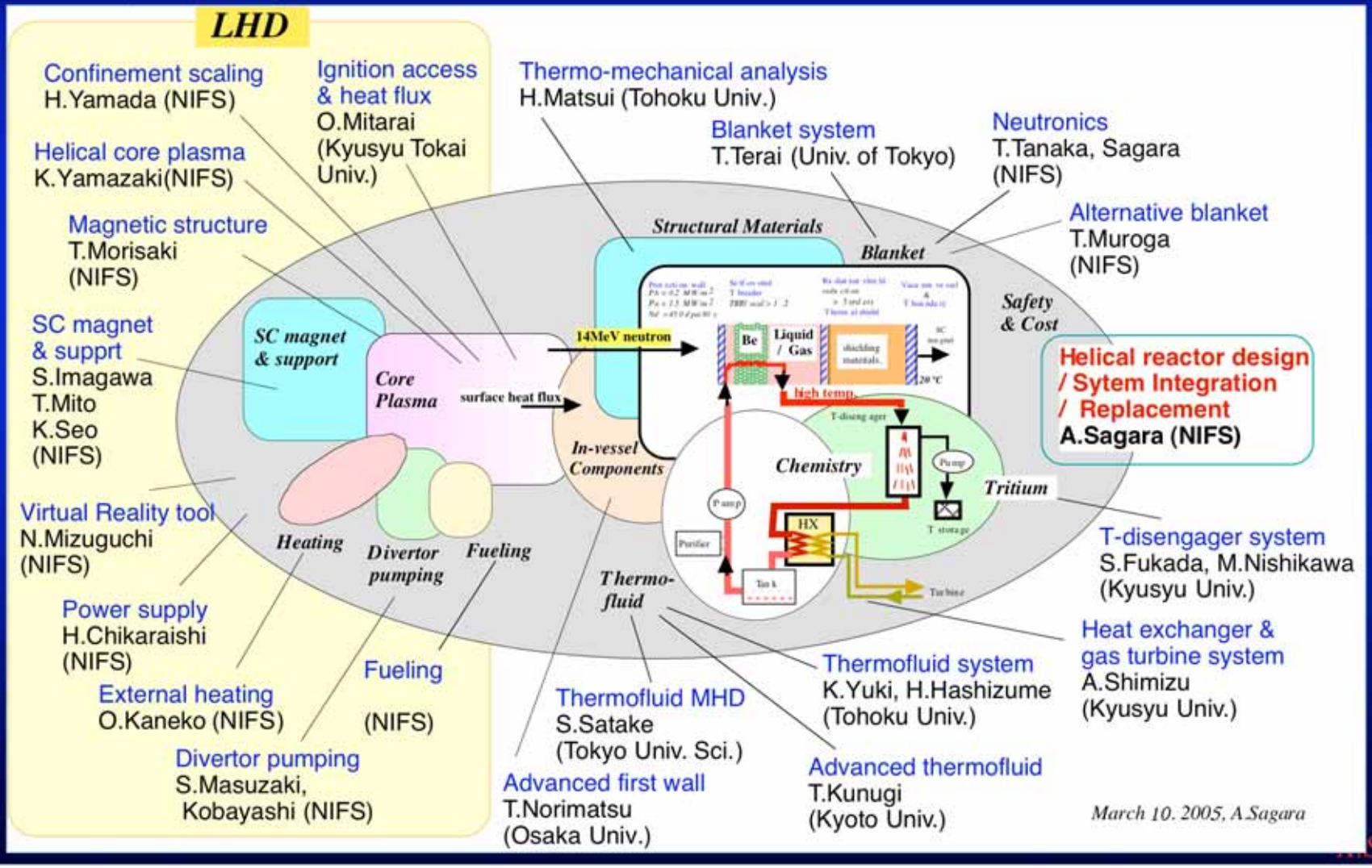
System Integration and Design Activities

From 1993, collaboration activities on LHD-type reactor-system designs have developed with connecting wide research areas on fusion science and engineering (**Fusion Research Network**) in the universities in Japan.



Fusion Engineering Network

Hokkaido Univ., Eng. (1,2,4)
Hokkaido Univ., CARET (1,3)
Muroran Inst. Tech (1,4)
Tohoku Univ., Eng. (1,2,3)
Tohoku Univ., Science (1)
Tohoku Univ., Med. (3)
Tohoku Univ., IMR (1)
Tohoku Univ., IMR-Oarai (1)
Tohoku Univ., Inst. Fluid Sci. (2)
Iwate Univ. (2)
Akita Univ., Med. (2,3)

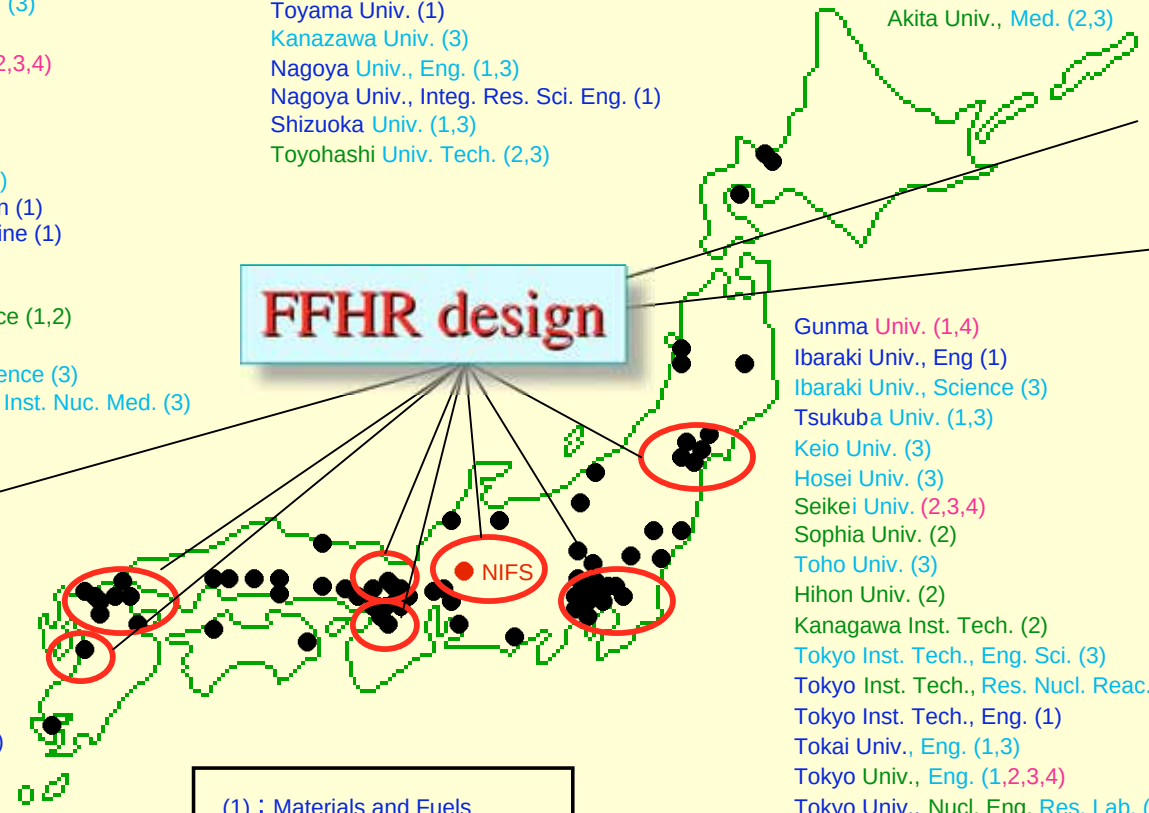
Kyoto Univ., Eng., (1,2,3,4)
Kyoto Univ., Res. Reactor Inst. (1,3)
Kyoto Univ., Inst. Adv. Energy (1,3)
Kyoto Univ., Energy Science (1,3)
Kyoto Univ. Education (3)
Osaka Univ. ILE (3,4)
Osaka Univ., Eng. (1,2,3,4)
Kinki Univ. (1,4)
Osaka Pref. Univ. (3)
Osaka City Univ. (2)
Osaka Inst. Tech. (1,3)
Hyogo Univ. Education (1)
Kobe Univ. Marc. Marine (1)
Himeji Inst. Tech (1)
Okayama Univ. (3)
Okayama Univ. Science (1,2)
Fukuyama Univ. (3)
Hiroshinma Univ., Science (3)
Hiroshima Univ., Res. Inst. Nuc. Med. (3)
Tottori Univ. (2)
Tokushima Univ. (1)
Ehime Univ. (1)

Niigta Univ. (1,3)
Nagaoka Univ. Tech. (4)
Toyama Univ. (1)
Kanazawa Univ. (3)
Nagoya Univ., Eng. (1,3)
Nagoya Univ., Integ. Res. Sci. Eng. (1)
Shizuoka Univ. (1,3)
Toyohashi Univ. Tech. (2,3)

Max-Planck IPP, Germany

Kharkov IPT, Ukraine

ASIPP, China



Gunma Univ. (1,4)
Ibaraki Univ., Eng (1)
Ibaraki Univ., Science (3)
Tsukuba Univ. (1,3)
Keio Univ. (3)
Hosei Univ. (3)
Seikei Univ. (2,3,4)
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Toho Univ. (3)
Hihon Univ. (2)
Kanagawa Inst. Tech. (2)
Tokyo Inst. Tech., Eng. Sci. (3)
Tokyo Inst. Tech., Res. Nucl. Reac. (1,2,3)
Tokyo Inst. Tech., Eng. (1)
Tokai Univ., Eng. (1,3)
Tokyo Univ., Eng. (1,2,3,4)
Tokyo Univ., Nucl. Eng. Res. Lab. (1,2,3)
Tokyo Univ., Res. Artifact (1)
Tokyo Univ., Science (2)
Tokyo Univ., Isotope (1,3)

Kyushu Univ., Eng. (1,2,3,4)
Kyushu Univ., Science (1,3)
Kyushu Univ., Res. Inst. Appl. Mech. (1)
Kyushu Univ., Eng. Sci. (1,3)
Kyushu Inst. Tech., Eng. (3)
Kyushu Inst. Tech., Comp. System (2)
Oita Univ. (2)
Kagoshima Univ. (2)
Kyushu Tokai Univ. (3)

(1)：Materials and Fuels
(2)：Magnet, Electromagnetics
(3)：Reactor System, Safety
(4)：ICF Technology

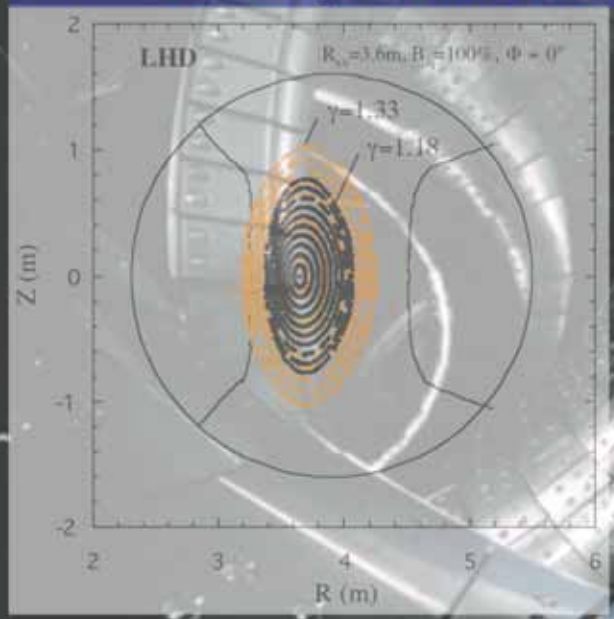
A.Sagara, ANS-FED Newsletter - June 2004, http://fed.ans.org/

Based on LHD, FFHR has 2 Main Features

(1) The coil pitch parameter γ of continuous helical winding has been adjusted

- to reduce the magnetic hoop force (Force Free Helical Reactor: FFHR),
- to expand the blanket space

By Selecting lower value of $\gamma = \left(\frac{m}{l}\frac{a_c}{R}\right)$



(2) Self-cooled liquid Flibe (BeF_2-LiF) blanket has been proposed, due to

- low MHD pressure loss,
- low reactivity with air,
- low pressure operation,
- low tritium solubility.

Main design parameters and recent modification & optimization activities

A. Sagara et al. Nucl. Fusion 45(2005) 258-263.

Sagara, 17th TOFE(2006)

Kobayashi, 17th TOFE(2006)

Morisaki, 15th ITC(2005)

Takahata, 24th SOFT(2006)

Itoh, 17th TOFE(2006)

Yamada, 24th SOFT(2006)

Mitarai, 21th IAEA(2006)

Sato, 17th TOFE(2006)

Tanaka, 21th IAEA(2006)

Muroga, 17th TOFE(2006)

Nagasaka, 17th TOFE(2006)

Kozaki, 17th TOFE(2006)

Mizuguchi, 15th ITC(2005)

Design parameters			LHD	FFHR2	FFHR2m1	FFHR2m2
Polarity	l		2	2	2	2
Field periods	m		10	10	10	10
Coil pitch parameter	γ		**1.25**	**1.15**	**1.15**	**1.25**
Coil major Radius	Rc	m	3.9	10	14.0	17.3
Coil minor radius	ac	m	0.98	2.3	3.22	4.33
Plasma major radius	Rp	m	**3.75**	**10**	**14.0**	**16.0**
Plasma radius	ap	m	0.61	1.2	1.73	2.80
Blanket space	Δ	m	0.12	**0.7**	**1.2**	**1.1**
Magnetic field	B0	T	**4**	**10**	**6.18**	**4.43**
Max. field on coils	Bmax	T	9.2	13	13.3	13
Coil current density	j	MA/m2	53	25	26.6	32.8
Weight of support		ton	400	**2880**	**3020**	**3210**
Magnetic energy		GJ	1.64	147	120	142
Fusion power	P_F	GW		**1.77**	**1.9**	**3.0**
Neutron wall load		MW/m2		**2.8**	**1.5**	**1.3**
External heating power	Pext	MW		100	80	100
α heating efficiency	ηα			0.7	0.9	0.9
Density lim.improvement				1	1.5	1.5
H factor of ISS95				**2.53**	**1.92**	**1.68**
Effective ion charge	Zeff			1.32	1.34	1.35
Electron density	ne(0)	10^19 m-3		28.0	26.7	19.0
Temperature	Ti(0)	keV		27	15.8	16.1
<β>=2*n*T/(B^2/μ) (parabolic distribution)				**1.8**	**3.0**	**4.1**
COE	Yen/kWh			**21.00**	**14.00**	**9.00**

Recent Activities in Optimization Phase

1. Heating power
2. Reactor size
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5. Magnet system
6. Blanket materials
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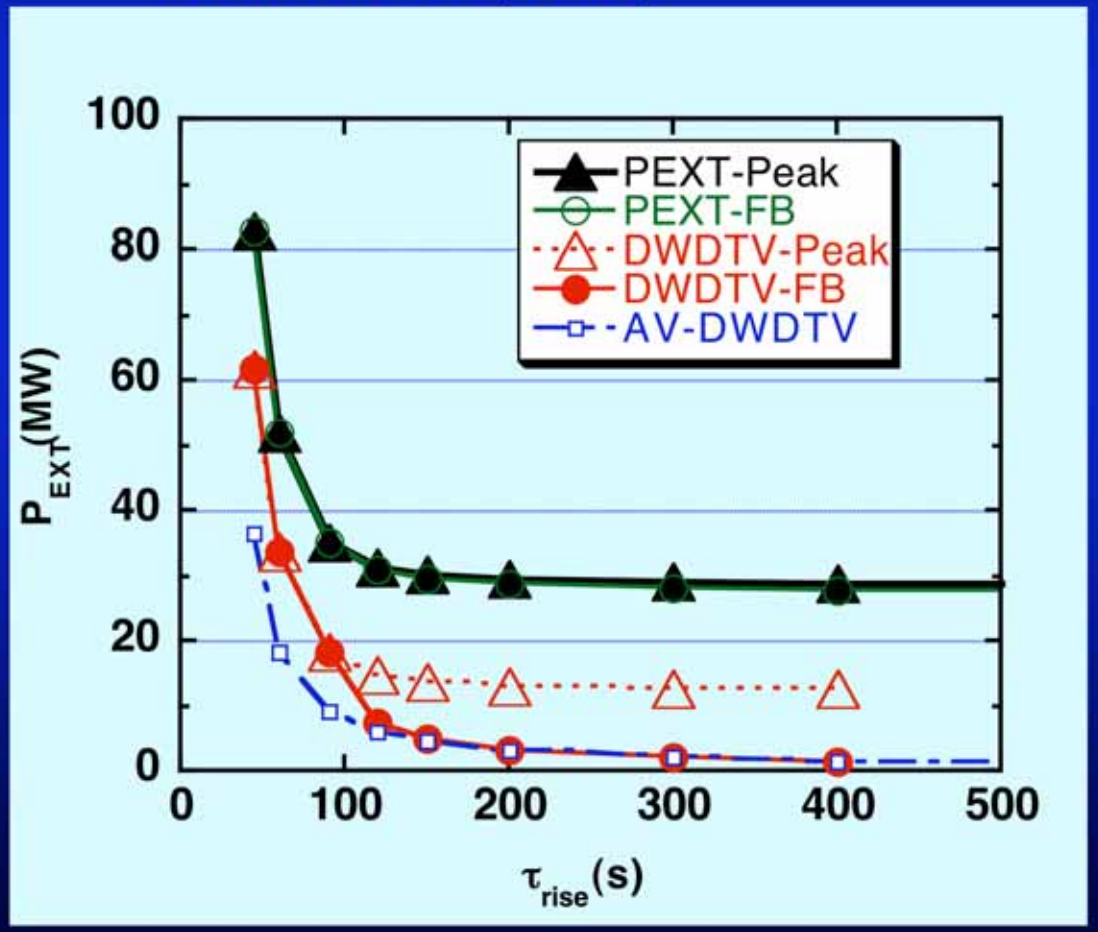
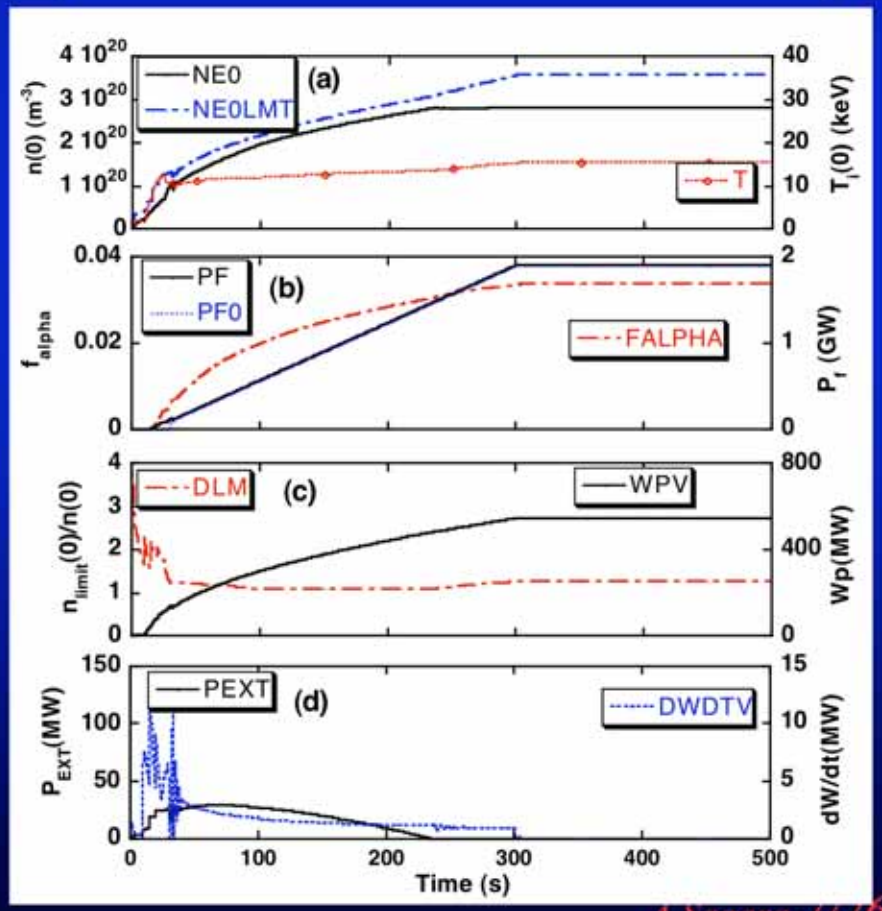
Heating Power can be Minimized to < 30MW

by longer fusion power rise-up time due to dW/dt effect.

O. Mitarai, A. Sagara et al.,
21th IAEA (2006)

Any fusion power rise-up time can be achievable in a helical reactor. [Example: t_{rise}=300 sec]





Reactor size

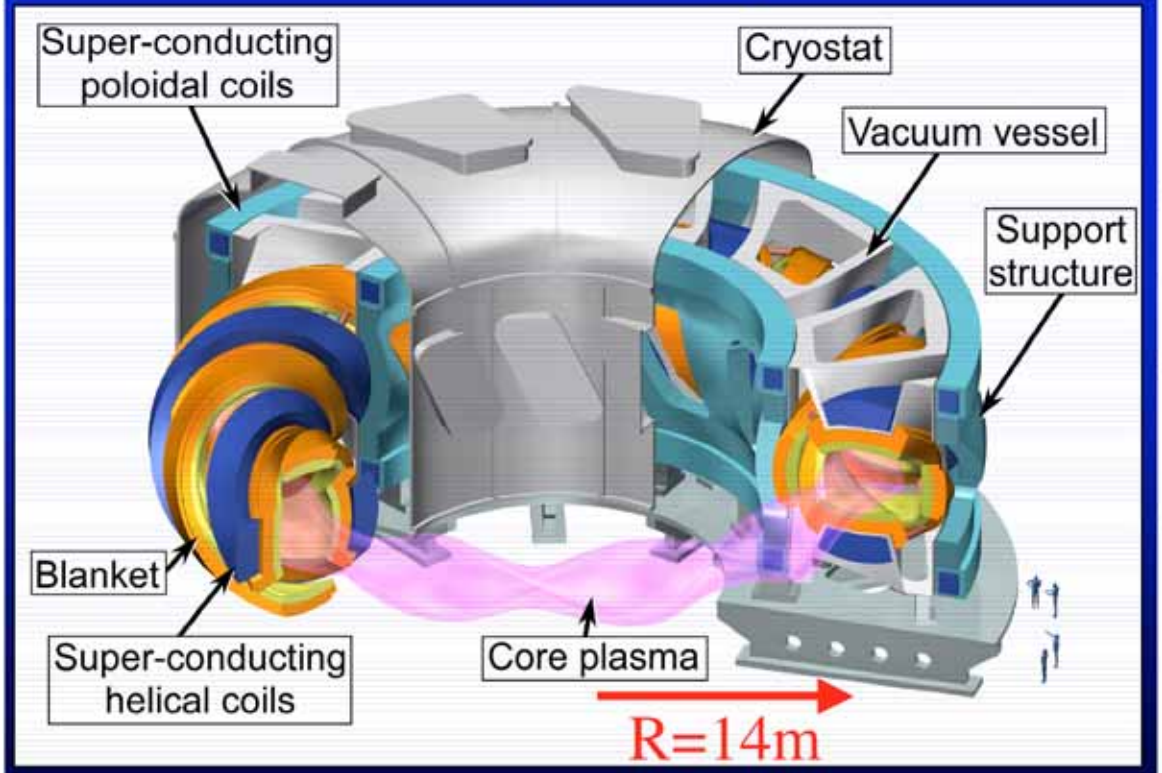
Which is strongly coupled with blanket and divertor design

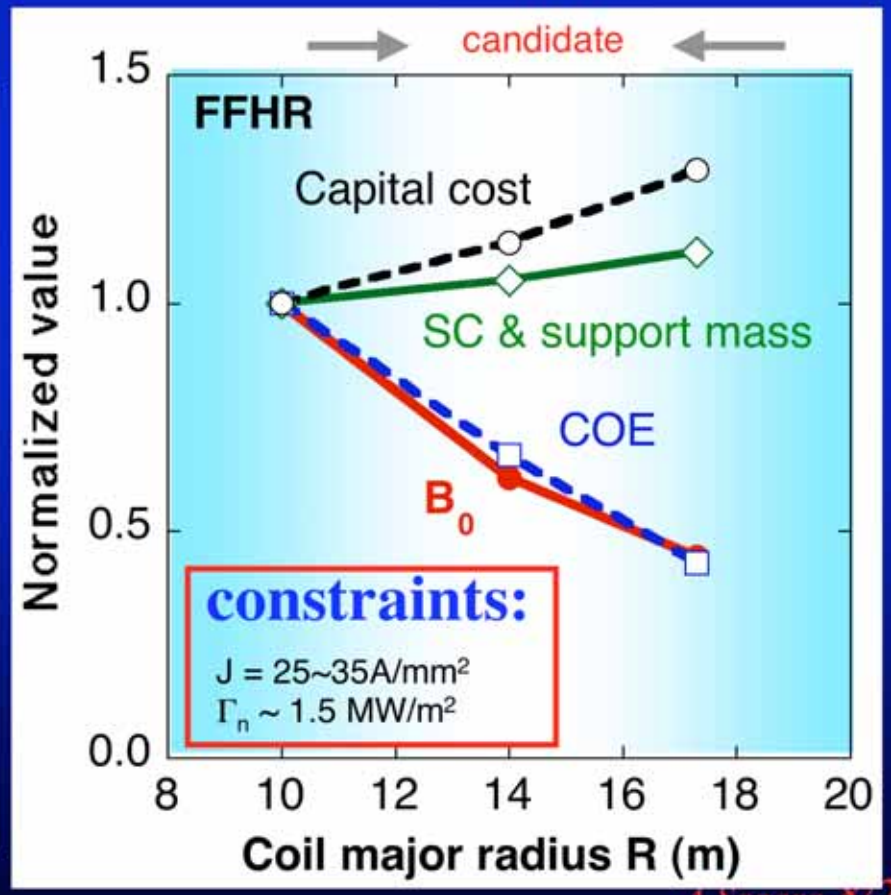
LHD-type reactor

FFHR2m1 P ~ 1GWel
W~ 25,000 ton
Q > 50 by helical

New targets based on LHD:
- Large scale SC magnet
- SC magnet of high J

Coil design by S.Imagawa

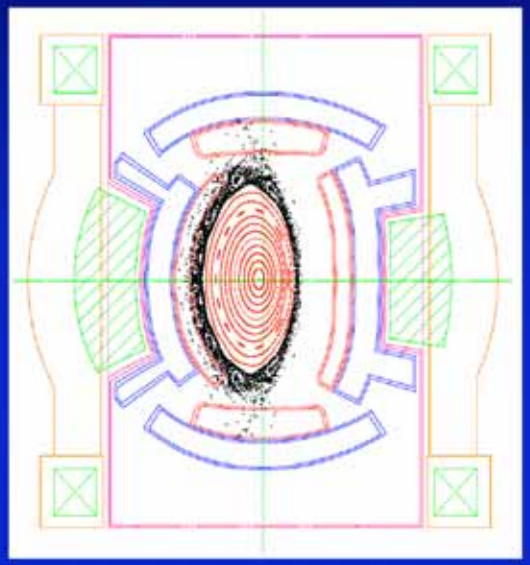


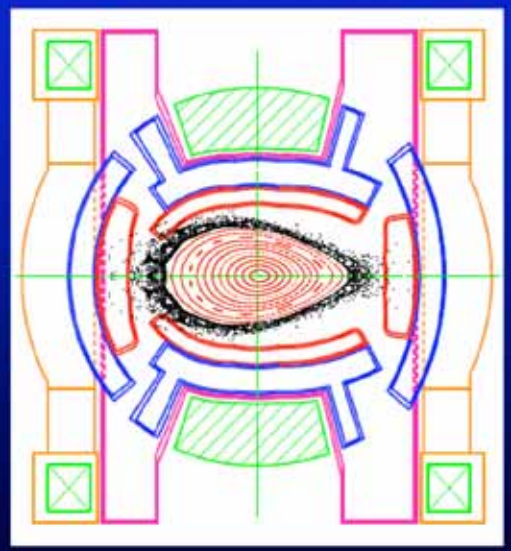


Standard design in FFHR

Conventional helical divertor

inherently equipped built-in divertor



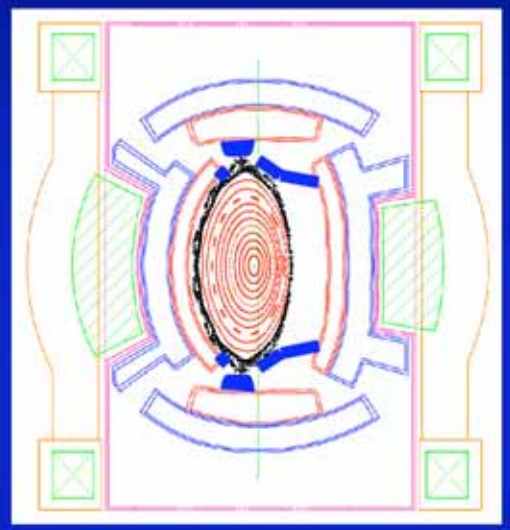


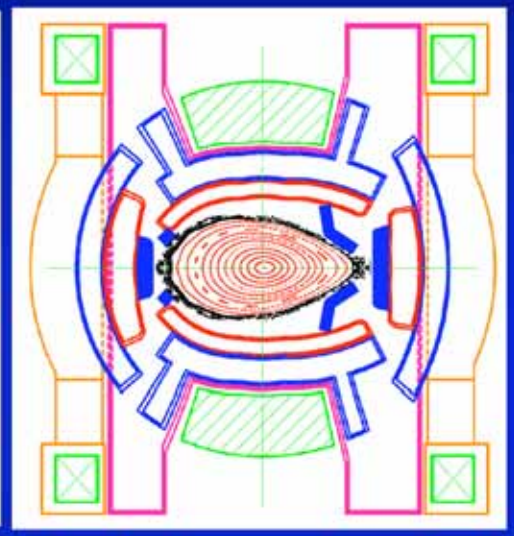
Alternative design in FFHR

Helical X-point Divertor (HXD)

new concept for the compatibility with blankets

T.Morisaki, A.Sagara et al., ITC-15, 2005 / FED 81 (2006)





Heat load to target plates

--> assuming average width ~ 0.25m

(total area: 0.25m x 230m ~ 58m^3)

--> power loss through divertor ~ 300MW

--> ~ 3.2 MW/m^2

Particle recycling

---> 99% ionized ***(Kobayashi, 17th TOFE)***

Recent Activities in Optimization Phase

1. Heating power
2. Reactor size
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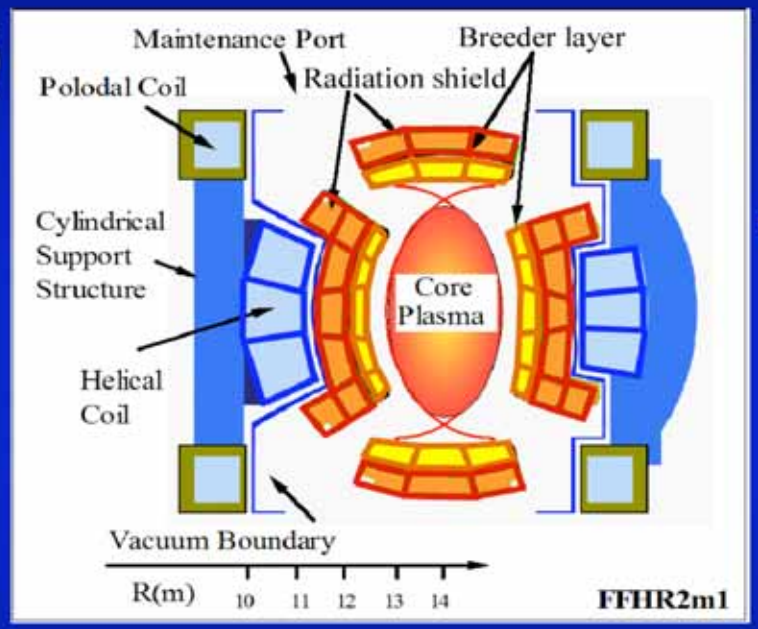
Improvement of 3D neutronics calculation system for helical structure

T. Tanaka, A. Sagara, M.Z.Youssef, et al. 21th IAEA (2006)

(a)

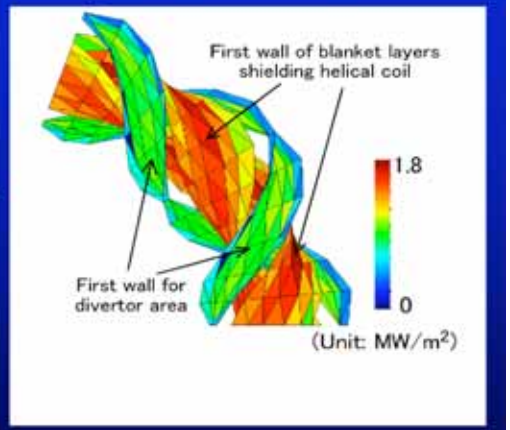


Division of cross-section of helical blanket components into quadrangular meshes.

(b)

Generation of helical structure according to numerical equations

(c)



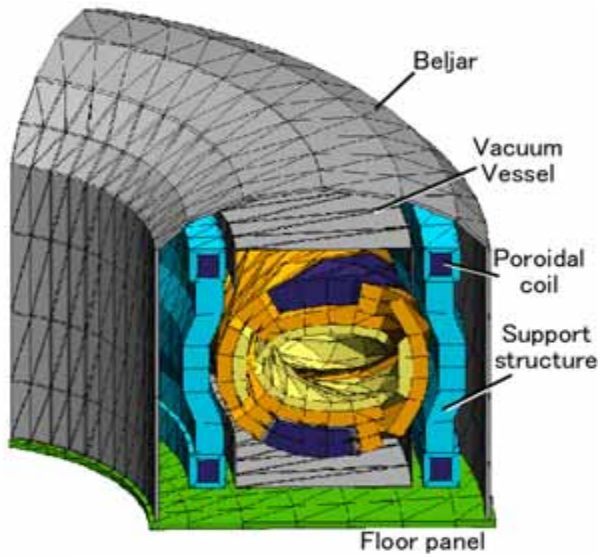
Example of 3-D geometry data with MCNP (~3,000 cells) and helical neutron wall loading

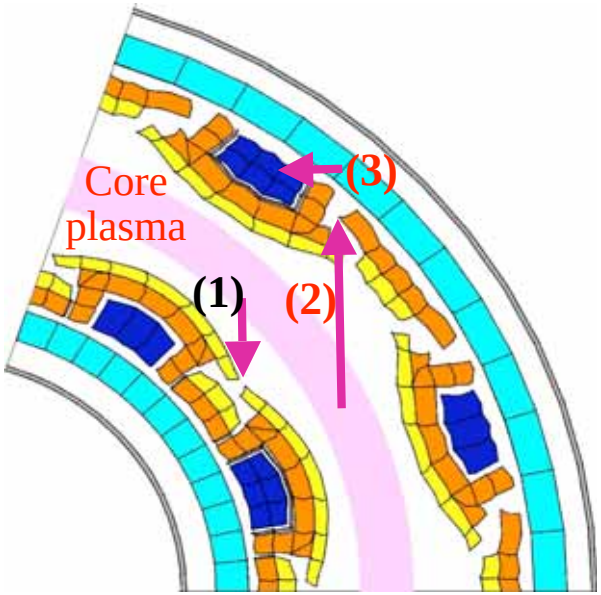
- Quick generation of 3-D helical geometry with small size input data. (Coordinates of quadrangular meshes on cross-section drawing)

Shielding issues have been made clear

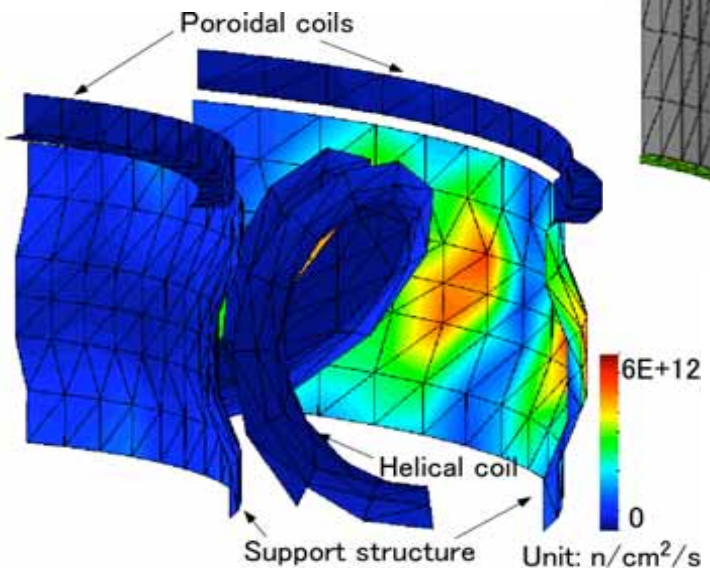
- **(1) Direct neutrons from core plasma to side surfaces of helical coils (Could be suppressed by enhancement of shielding)**
- **(2) Neutron streaming to poroidal coils**
- **(3) Neutron reflection on helical coils**

Geometry for investigation of neutron streaming and reflection





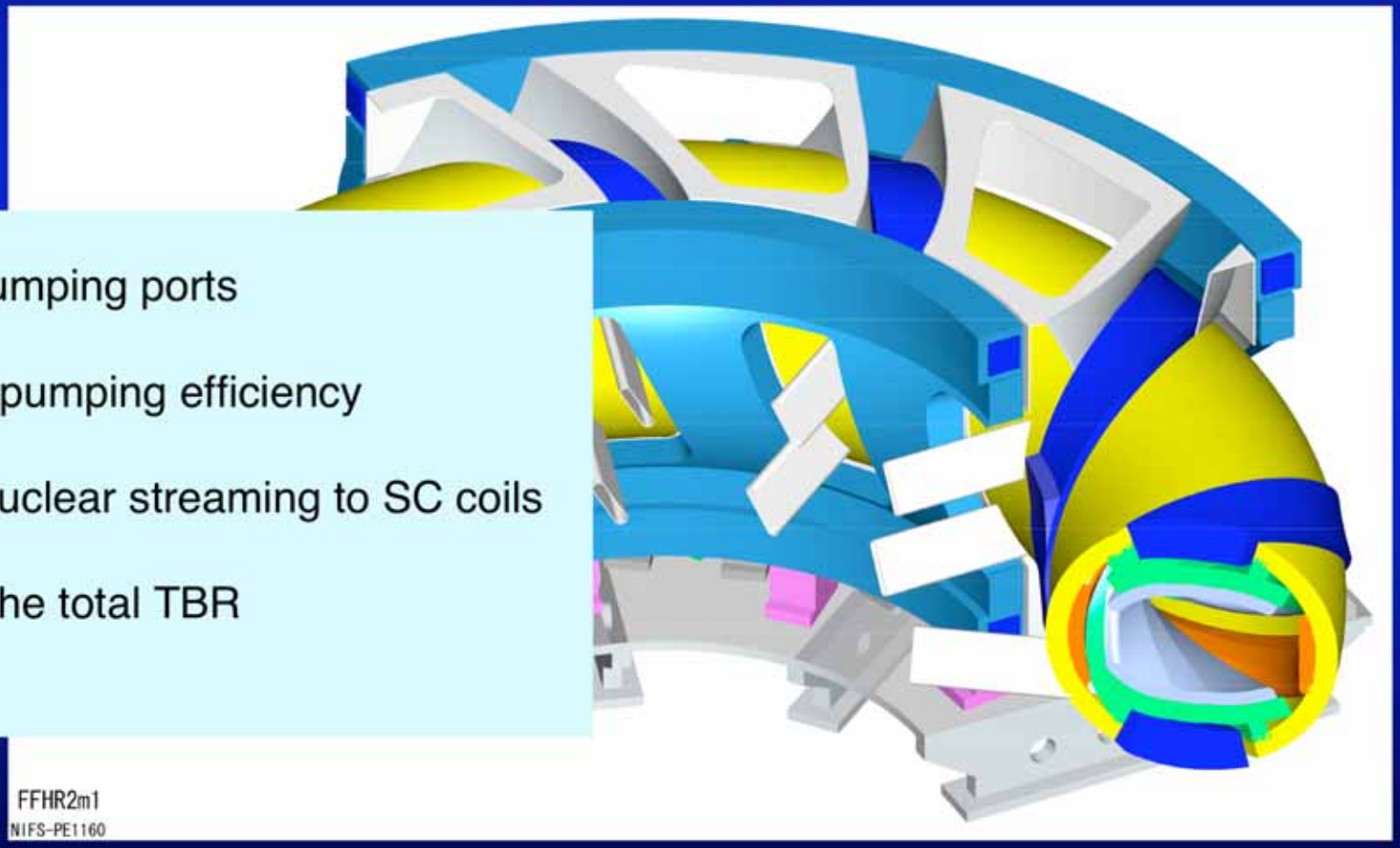
Horizontal cross-section of geometry



Distribution of fast neutron flux on helical coil, troidal coils and support structures

Discrete Pumping with Semi-closed Shield (DPSS) is proposed

- ✓ Localization of pumping ports
- ✓ Enhancement of pumping efficiency
- ✓ Suppression of nuclear streaming to SC coils
- ✓ Improvement of the total TBR



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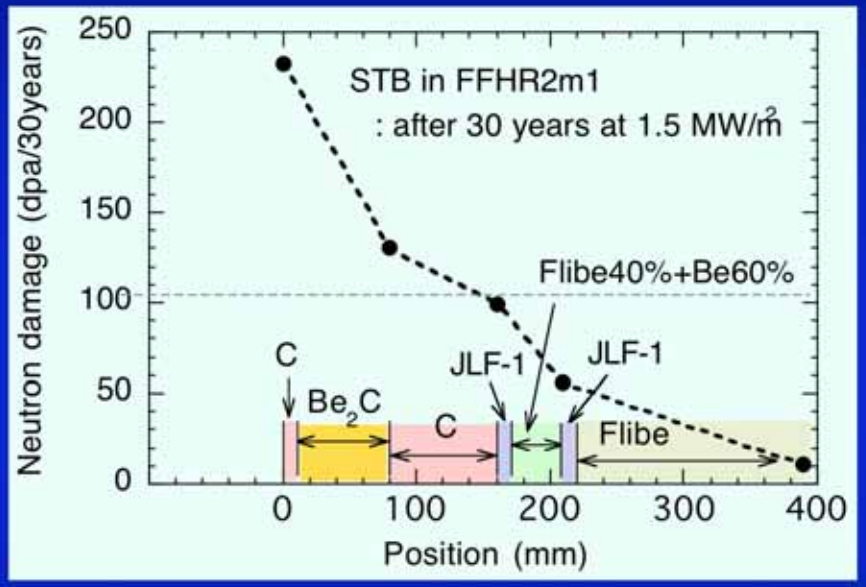
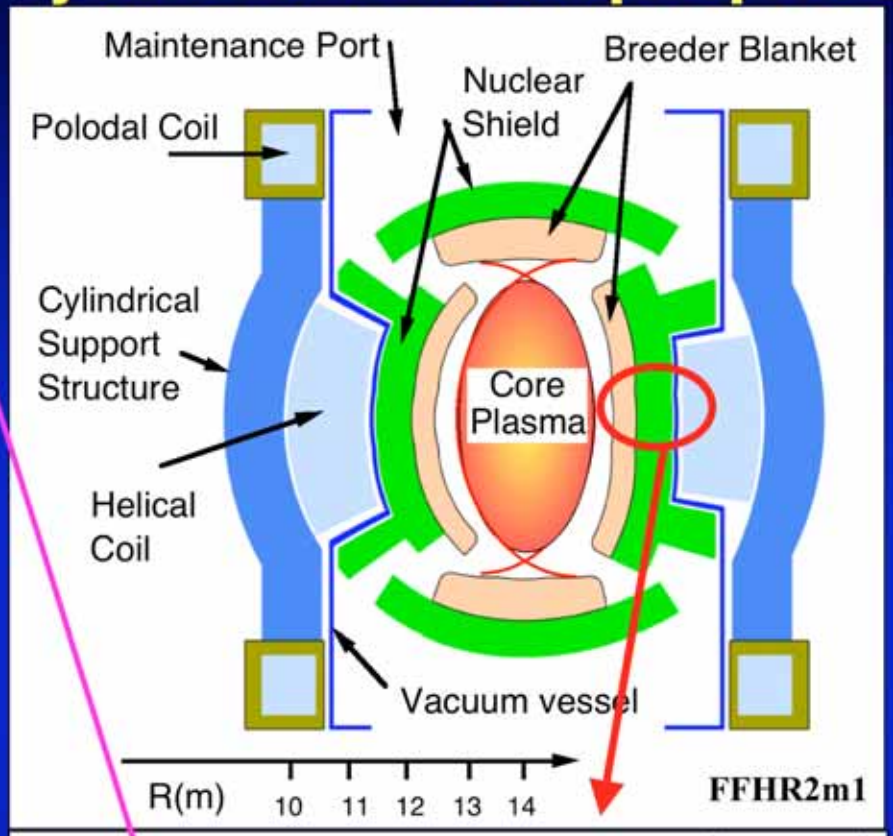
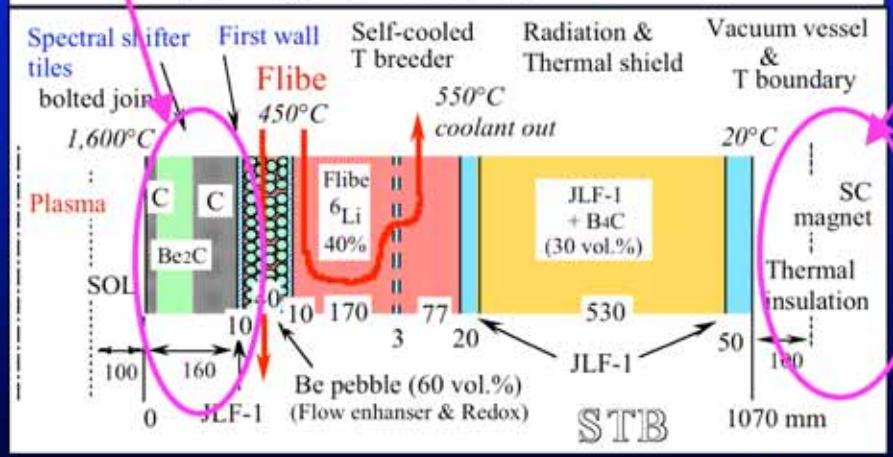
Long-life blanket STB for 30 years has been proposed (Spectral-shifter and Tritium Breeder blanket)

A. Sagara et al. Nucl. Fusion 45(2005) 258.







- **Replacement-free of blanket units**
  < 100dpa in the full-life of 30y
- **Improved nuclear shielding for SC magnets**
  ~ 5×10^{22} fast n/m^2 in 30y

Issue : Enhancement of cooling capability for one-side heating > 1MW/m^2 is requested.

Thermofluid R&D activities

17th TOFE

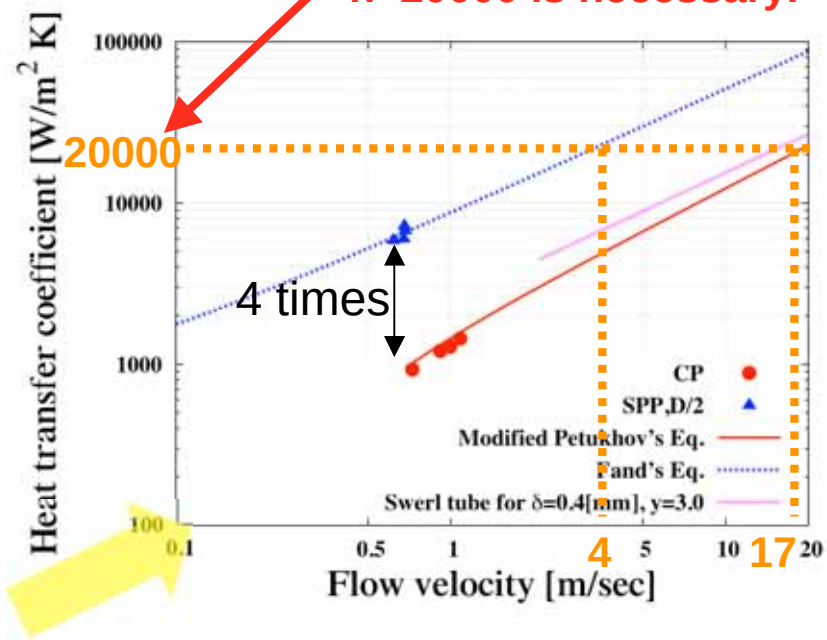
Tohoku Univ.: H.Hashizume, K.Yuki et al.

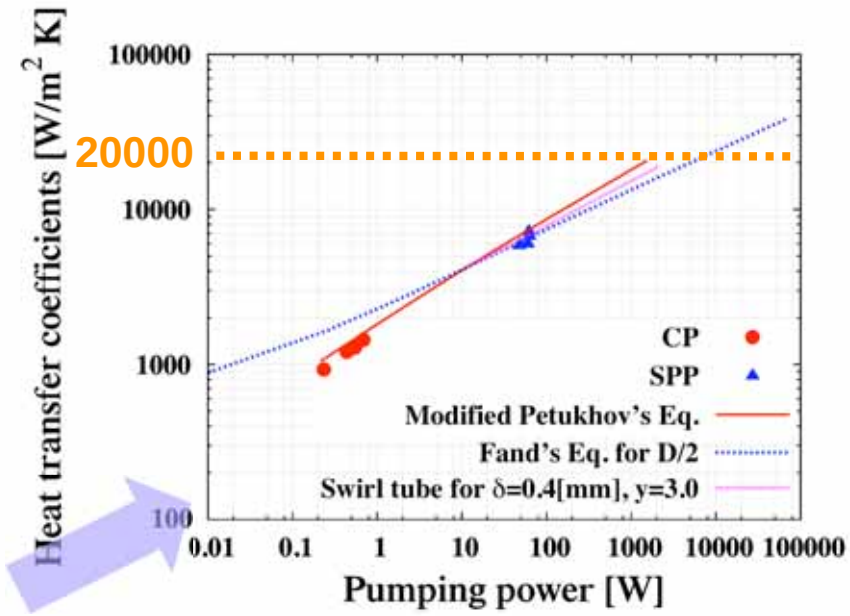
for enhancing heat-transfer in such high Prandtl-number fluid as Flibe

- The performance of SPP flow using alumina spheres (0.5D) is about 4 times higher than that of CP flow.
- Flow velocity to achieve h=20000W/m²K using SPP is just 4 m/sec, which is much lower compared with CP flow(~17m/s).
  → Reduction of MHD effect, erosion, etc.

- Pumping power to achieve h=20000 using SPP is a little higher than that of CP flow.
- However, in the case of using metal and smaller spheres, the heat transfer performance of SPP can overcome CP flow, which enables lower pumping power conditions.
  → Much lower pumping power in all piping system.

To remove 1MW/m², h=20000 is necessary.





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Superconducting Magnet System for FFHR (Low-Temperature Superconductor Option)

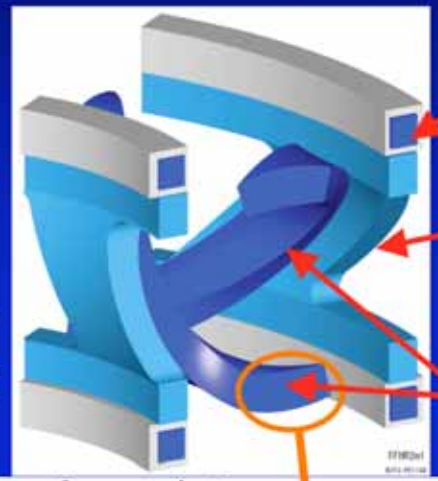
K. Takahata, T. Mito, A. Sagara, et al:

presented at 24th SOFT,11-15 Sept. 2006, Warsaw, Poland, P1-E-285

Continuous windings with a large diameter, twisted configuration and large magnetic energy

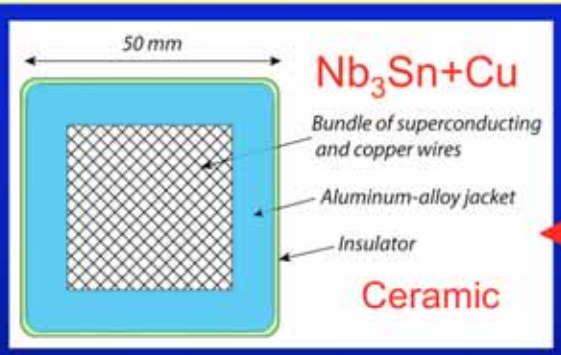
Indirect cooling using cooling panels
Aluminum-alloy jacketed Nb_3Sn superconductor

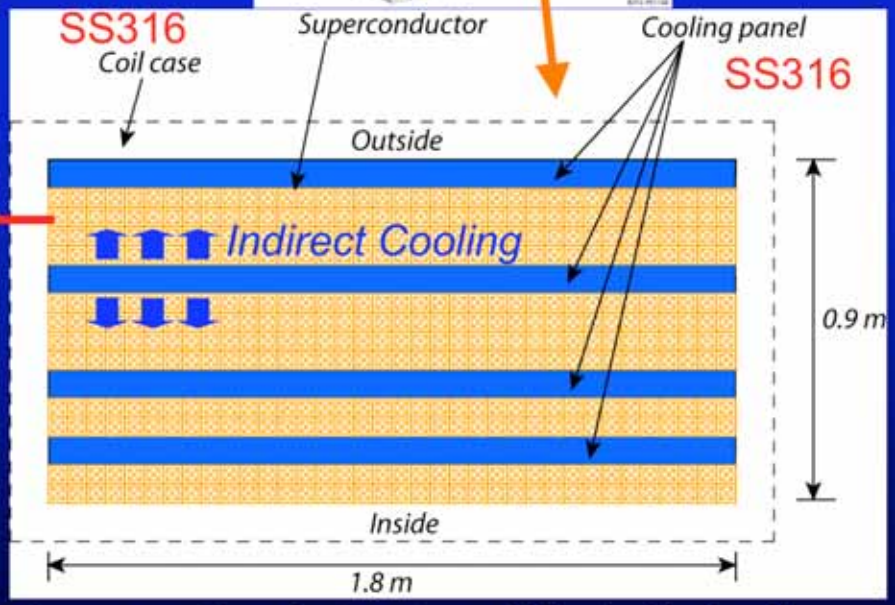
R=14 m
B_0=6.2 T
B_{max}=13.3 T
E_{coil}=120 GJ



100 kA Superconductor

- **High effective thermal conductivity**
- **High mechanical rigidity and strength**





Cross-sectional structure of the helical coil

Quench protection

- **Conventional protection circuit using an external resistor τ=20 s**
- **Six subdivisions V_{max}=10 kV**
- **Hot spot temperature < 150 K**

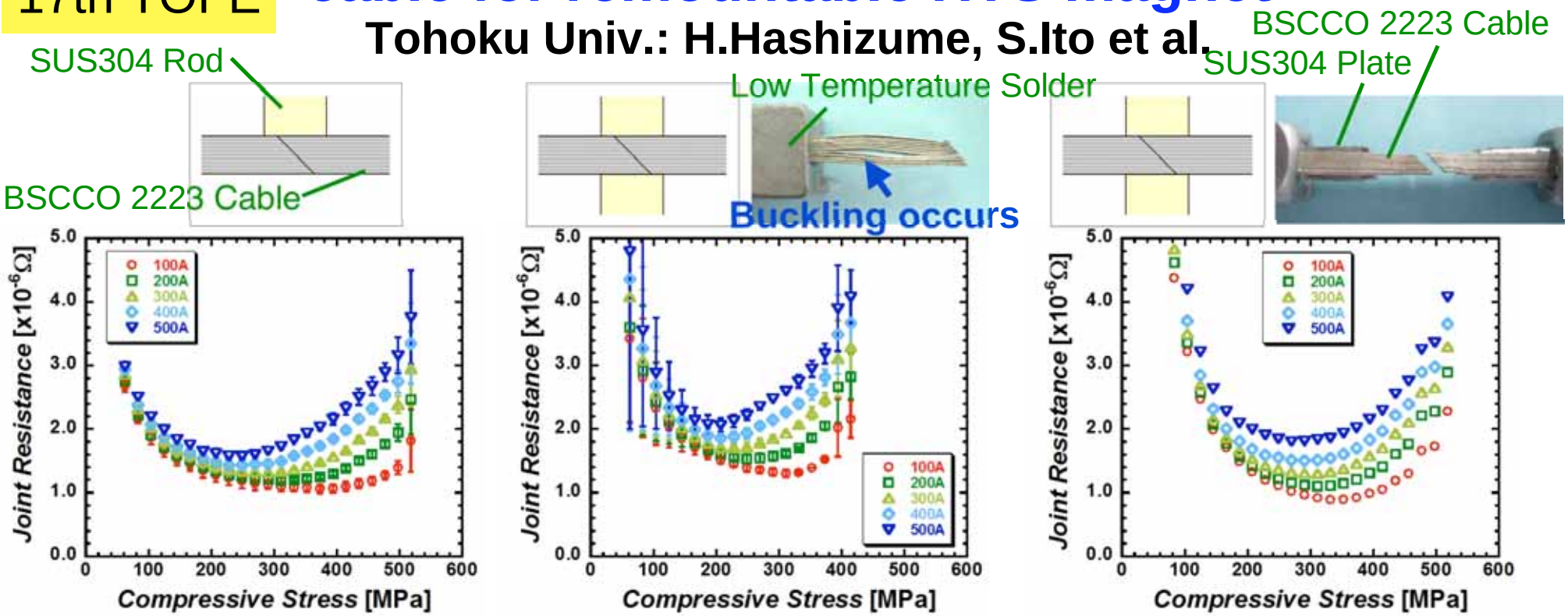
R&D plan

- **Sub-scale 10 kA-class superconductor**
- **Model coil test**

Characteristic evaluation of mechanical jointed HTS cable for remountable HTS magnet

17th TOFE

Tohoku Univ.: H.Hashizume, S.Ito et al.



Top loading | Dual loading | Dual Loading with supporting structure

Improving joint performance by optimizing loading system

- Dual loading with supporting structure can increase the optimum joint stress where joint resistance is the minimum.
- Dual loading with supporting structure can reduce joint resistance just slightly.

→ Joint surface condition must be improved to reduce joint resistance more. (Target of joint resistance: Order of 100nΩ)

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Tritium Control for V-alloy/Flibe Blanket

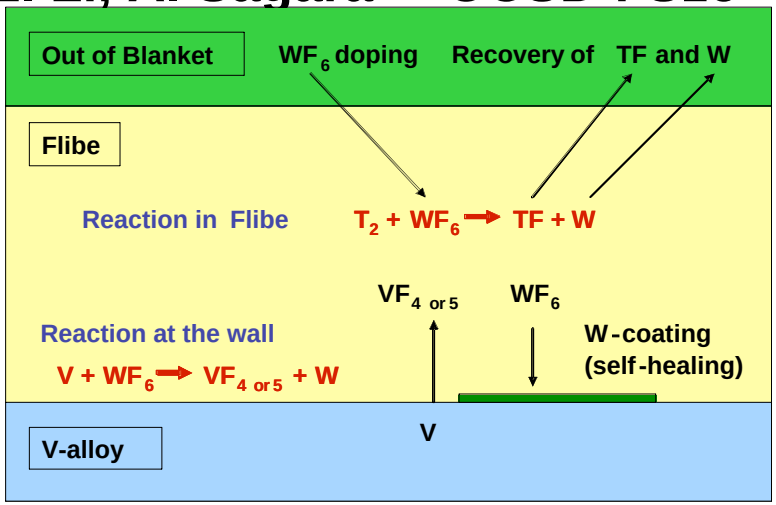
which can increase operation temperature and ΔT

17th TOFE

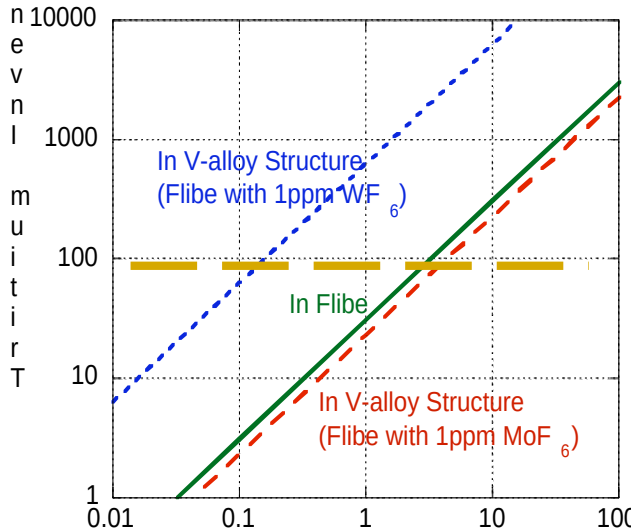
NIFS : T.Muroga, T, Tanaka, Z. Li, A. Sagara UCSD : Sze

Application of REDOX control by MoF_6 or WF_6 doping

- However, conventional REDOX control with Be (TF to T_2) results in extremely high T inventory in V-alloy structure
- Thermodynamics analysis were carried out for another REDOX control (T_2 to TF) by MoF_6 or WF_6 doping
- This reaction can form corrosion –protective Mo or W coating on V-alloy surfaces with self-healing capability
- Doping of 1ppm MoF_6 or WF_6 can reduce T inventory in blanket below 100 g.
- Among the issues of the concept are recovery of tritium in the form of TF and Mo or W out of the blanket.



Concept of REDOX control by WF_6 doping



T-inventory in blanket as a function of TF level in Flibe, for doping of 1ppm MoF_6 or 1ppm WF_6

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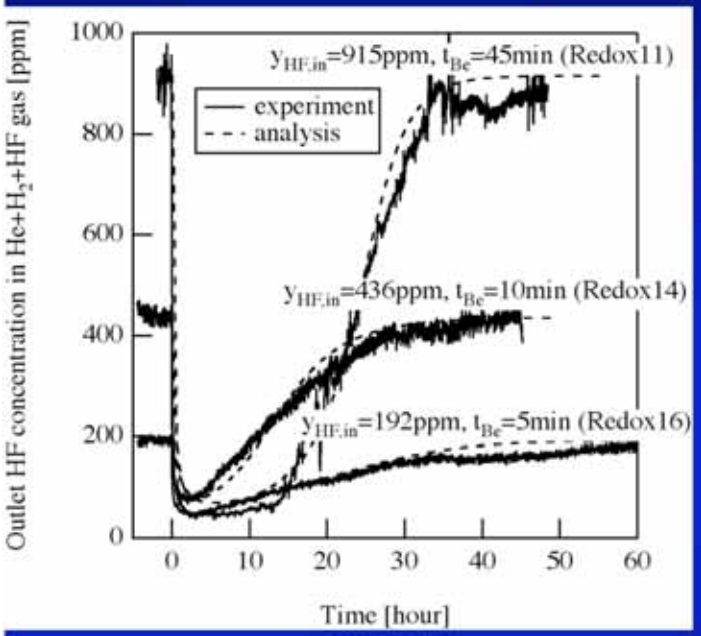
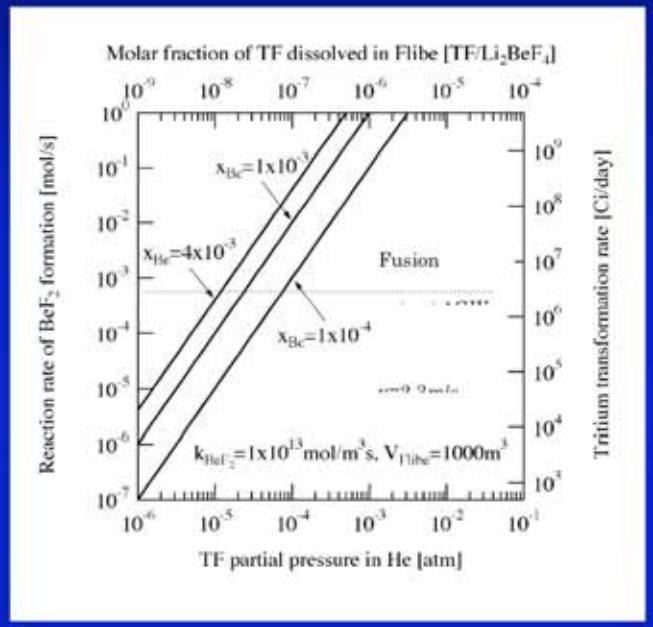
Tritium cycle of FFHR-2 Flibe self coolant system

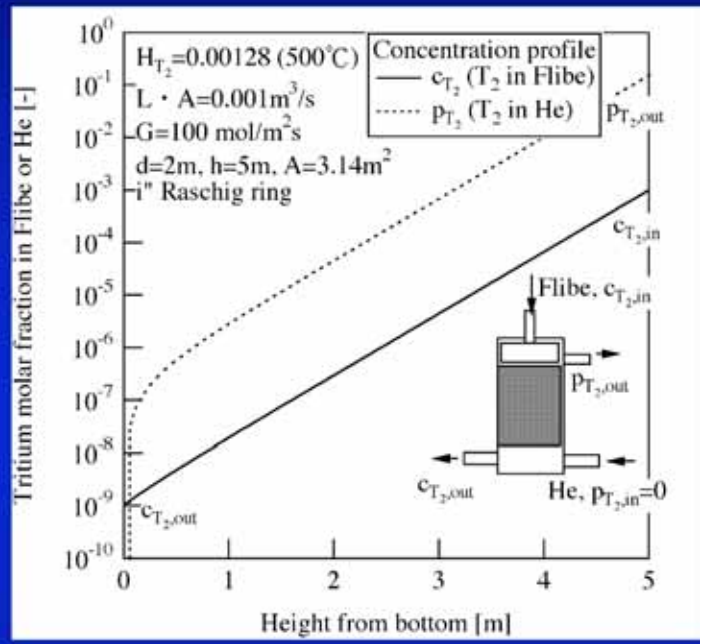
17th TOFE : S. Fukada (Kyushu University), A. Sagara (NIFS)



Flibe redox control analysis



TF concentration in a Flibe blanket of FFHR



Flibe-He extraction tower calculation

- TF (HF) can be reduced to T_2 (H_2) in contact with Be in blanket. (JUPITER-II study)
- The redox control by Be is achieved under the TF concentration expected in the FFHR-2 Flibe blanket..
- Flibe-He counter-current extraction system is designed and calculated.

Dedicated Hydrogen Production by 1 GWe (Optional Design)

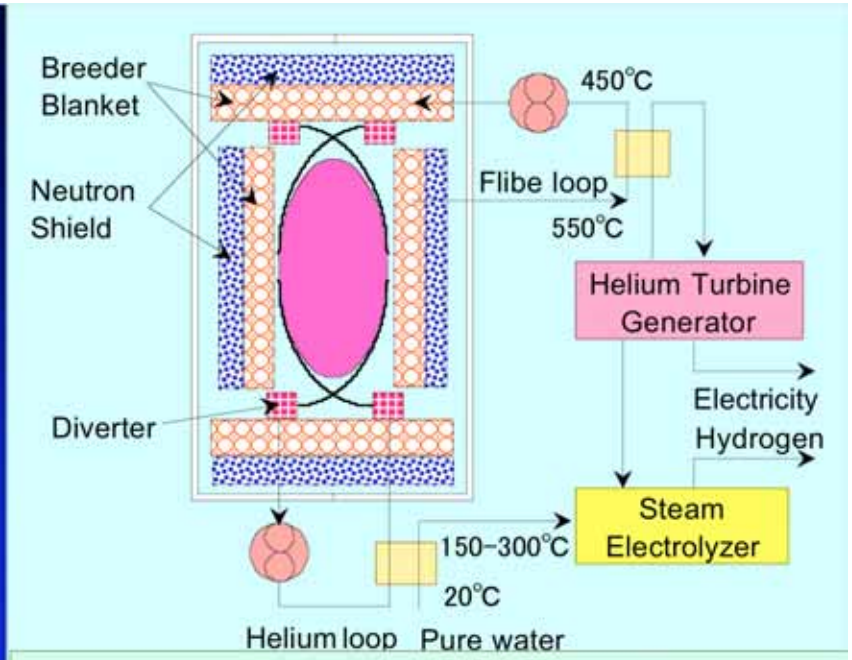
S. Yamada, A. Sagara et al: presented at 24th SOFT, 11-15 Sept. 2006, Warsaw, Poland, P3-J-333.

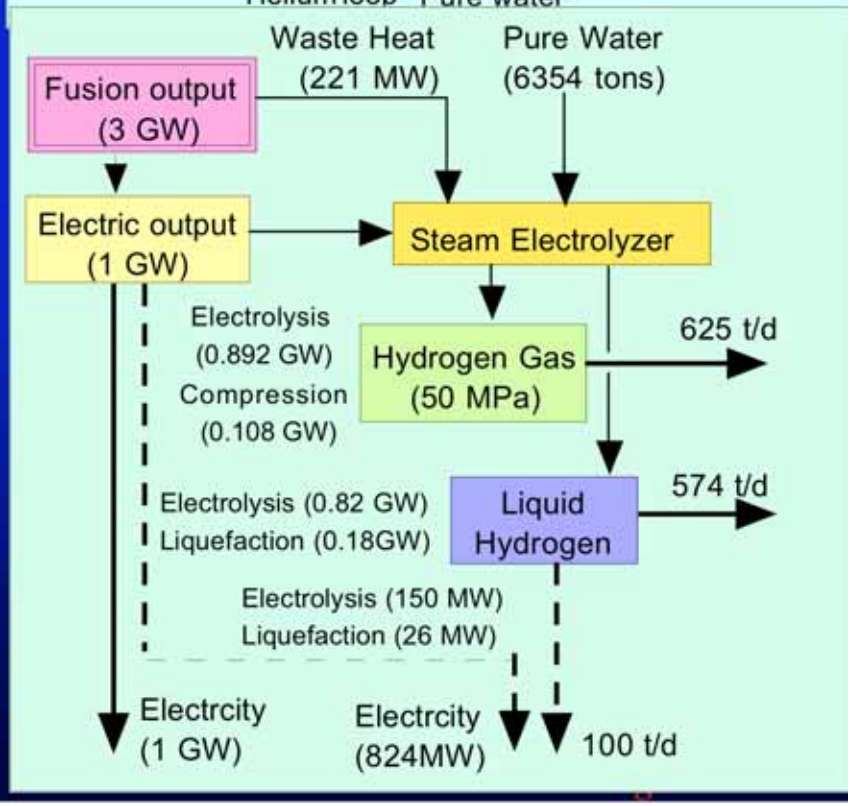
Steam production for electrolysis by the waste heat from;

- divertor plates,
- neutron shielding parts and so on.

Various types of hydrogen production;

- 24hour full-time hydrogen production,
  - High pressure gaseous hydrogen (625 t/d @50 MPa)
  - Liquid hydrogen (574 t/d)
- Hydrogen production at off-peak time. (L-H2 for 100 t/d & 824 MWe)

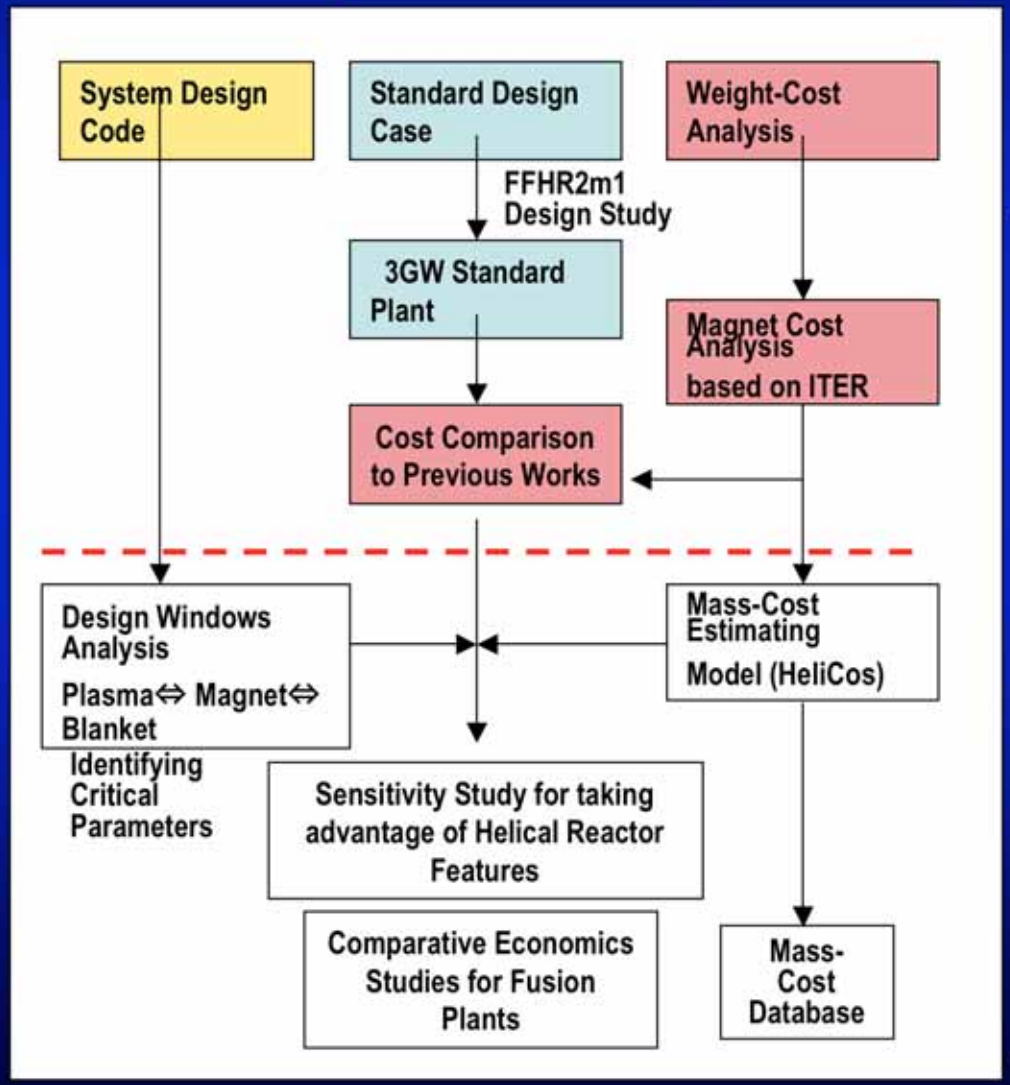




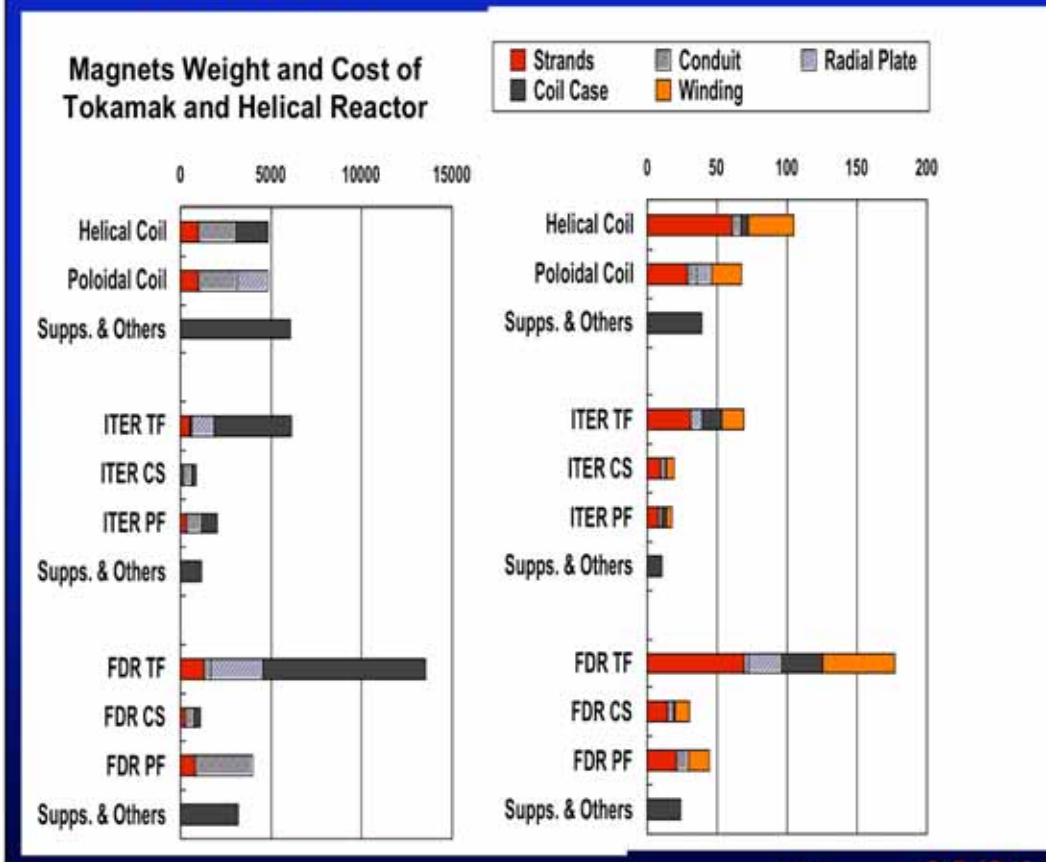
Design Windows Analysis based on a Cost Model of Helical Reactor

17th TOFE

Y. Kozaki, A. Sagara, S. Imagawa



- Using the ITER magnet cost data we evaluated the cost of helical reactor magnet,
- which show the similar coil weight and cost between Helical and Tokamak.



Summary

1. The FFHR design is now on optimization phase.
2. Heating power minimization and a flexible 3D neutronics method have been developed for helical system.
3. Blanket design and divertor design are strongly coupled in helical system.
4. New engineering targets on magnets, blanket concepts and plant system have been proposed.
5. Plasma operation devices on fueling, heating, and He ash removal etc. can be designed based on LHD results.