Design of Helical Reactor FFHR-d1 and c1 for Steady State DEMO

Akio Sagara,

H. Tamura, T. Tanaka, N. Yanagi, J. Miyazawa, T. Goto, R. Sakamoto,

J. Yagi, T. Watanabe, S. Takayama, and the FFHR design group

In Fusion Engineering Research Project (FERP)

National Institute for Fusion Science, Japan

US-Japan Workshop on Fusion Power Plants and Related Advanced Technologies
13-14 March 2014
University of California San Diego, USA

FERP consists of 13 task groups and 44 subtask groups

Young Researchers are nominated as the TG leaders!

Fusion Eng. Res. Project

High-density plasma phys., High-temp. Plasma phys. Plasma heating phys., Device eng. and advanced phys. Fusion systems, Fusion theory and simulation Collaborative study

Superconducting magnet group :Imagawa /Task/Sub task		
Conductor development, Coil winding, Cooling Yanagi	Large-scale high-field conductor testing facility	Yanagi, Mito
	CIC conductor & winding	Obana, Imagawa, Hishinuma
	Indirect cooling conductor & winding	Takahata, Tamura
	HT SC conductor & winding	Yanagi, Mito
	EM force support structure	Tamura, Imagawa
	Cryostat	Tamura
Cryogenic apparatus, Coil power supply system Iwamoto	Cryogenic system	Hamaguchi, Iwamoto
	Bus-line, Current lead	S. Yamada, Obana
	Coil power supply system	Chikaraishi, S. Yamada

In-vessel component group :Muroga /Task/Sub task		
Blanket system development, Design T. Tanaka	Radiation shield	T. Tanaka, Hishinuma Nagasaka, Kondo
	Breeding blanket	Nagasaka, Hishinuma Kondo, Tanaka, Yagi
	Heat, hydrogen isotopes recovery system	Yagi, Muroga, Nagasaka, Hishinuma, T.Tanaka
	First wall	Hirooka, Ashikawa, Nagasaka, D.Kato
In-vessel component development, Design, Maintenance Tamura	Vacuum vessel	Tamura, Masuzaki
	Divertor	Masuzaki, Tokitani, Murakami, D.Kato, Sakaue
	Remote maintenance	Ashikawa, Ohdachi, Narushima,

Promotion meeting by Exec. 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 trace tritium
- Development of Real-time detection system

Reactor system design group :Sagara / Task/Sub task		
Design integration Sagara	Task setting, Project management	Sagara, Miyazawa, T. Goto
	Helical DEMO conceptual design	T. Goto, Miyazawa, Sagara
Building layout T. Goto	Layout design, process	T. Goto
	Reactor building design	Tamura, T. Goto
Power supply, Generator Chikaraishi	Generator, Power supply system	Chikaraishi, S. Yamada
	Transmission, H production	S. Yamada, Hishinuma
Tritium fuel system M.Tanaka	Tritium processing system	M. Tanaka
	Safety control	Kawano
	Bioshield, Radioactivation	
	Legislation, Licensing	K. Nishimura
Operation control Mitarai (Tokai Univ.)	Safety analysis, control system	
	Burn control	Mitarai
	Data processing	Nakanishi
Core plasma Miyazawa	High performance plasma	Miyazawa, T. Goto, Narushima Ichiguchi, Satake, Suzuki
	TCT effect, α particle loss	Yokoyama, Murakami (Kyoto U.) Seki
	Ignition Scenario	Mitarai, Goto, Sakamoto
Plasma heating Tsumori	NBI	Tsumori, Osakabe
	ECH	Igami, Yoshimura, Idei (Kyusyu U.), Kubo, Shimozuma
	ICH	Kasahara, Saito, Muto
Fueling Sakamoto	Pellet	Sakamoto
	Gas-puff	Miyazawa
Diagnostics Isobe	Magnetic diagnostics	Sakakibara
	Neutron diagnostics	Isobe
	Divertor diagnostics	Masuzaki
	Spectroscopic diagnostics	M. Goto
	Interferometer / reflectometer	K. Tanaka, Tokuzawa, Akiyama
	Thomson scattering	I. Yamada
	Charge exchange spectroscopy	Yoshinuma

Rev. 2013.10.14, A. Sagara

Presentation Outline

1. Introduction
2. 3D Designs of FFHR-d1 and New Proposals
   1) SC Magnet and Support Structure ⇨ *Tamura*
   2) Neutronics Evaluation
   3) Flinabe Blanket mixed with Metal Powder
   4) Fabrication of Helical Coils with Segmented HTS Conductors
   5) Design of DC power supplies for superconducting magnet
3. Improvement in Ignition Core Plasma and Design Flexibility
   1) Physics Analyses of FFHR-d1 Core Plasma
   2) Start-up Scenario of Ignition Core Plasma
   3) Improvement of the Pellet Fueling Scenario
   4) Sub-Ignition Design of FFHR-c1 before Demo
4. Summary and Future Plans

1. Introduction

- From 2010, NIFS launched the Fusion Engineering Research Project (FERP) and started the re-design of the LHD-type helical reactor FFHR-d1 towards DEMO in parallel with R&D researches.

- In the first round of FERP, the main parameters of FFHR-d1 were selected, establishing the system design code HELIOSCOPE and Direct Profile Extrapolation (DPE) method based on LHD plasma data.

Design Parameters

A. Sagara et al., Fusion Eng. Des.87(2012)594.

			LHD	FFHR2	FFHR2m1	FFHR2m2		FFHR-d1
						Standard	SDC	
Coil pitch parameter	γ_c		1.25	1.15	1.15	1.2		1.25
Coil major radius	R_c	m	3.9	10	14.0	17.3		15.6
Plasma major radius	R_p	m	3.75	10	14.0	16.0		14.4
Plasma minor radius	a_p	m	0.61	1.24	1.73	2.35		2.54
Plasma volume	V_p	m^3	30	303	827	1744		1878
Blanket space	Δ	m	0.12	0.7	1.1	1.05		0.765
Magnetic field	B_0	T	4	10	6.18	4.84		4.7
Magnetic energy	W_{mag}	GJ	1.64	147	133	160		160
Fusion power	P_{fus}	GW		1	1.9	3		3
Neutron wall load	Γ_n	MW/m^2		1.5	1.5	1.5		1.5
H factor of ISS95	H^{ISS95}			2.40	1.92	1.92	1.64	2
Plasma beta (evaluated with B_{ax})	$<\beta>$	%		1.6	3.0	4.4	3.35	5
Divertor heat load (Δ 0.1m) (on average)	Γ_{div}	MW/m^2			5	7.2	1.9	8.1
Total capital cost		G$(2003)		4.6	5.6	7.0		
COE		mill/kWh		155	106	93		

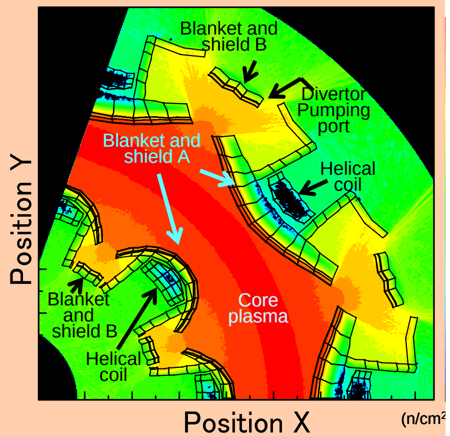
2nd Round

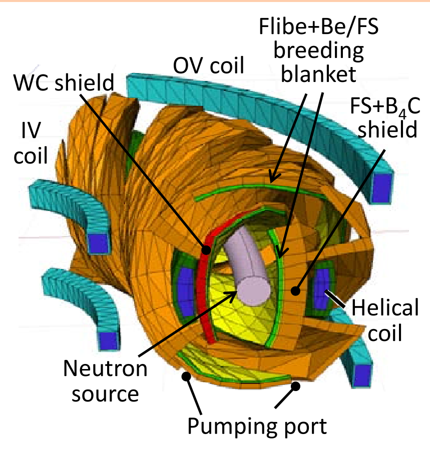
Design of in-vessel components

Blanket TG
3D blanket design
3D neutronics

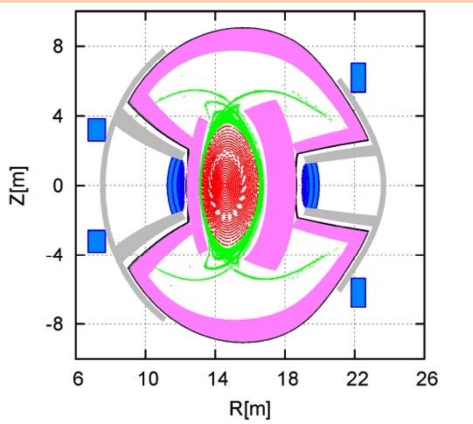
Design Integration TG
Consistency check
Numerical model

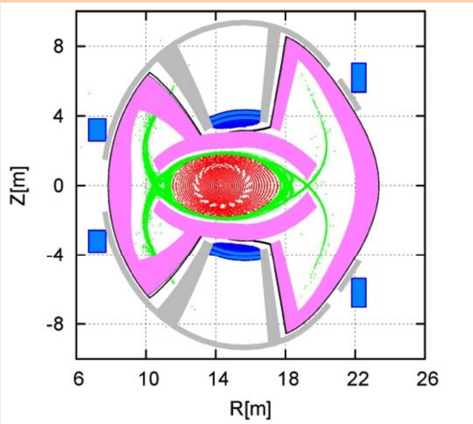
In-vessel components TG
Divertor design
Pumping

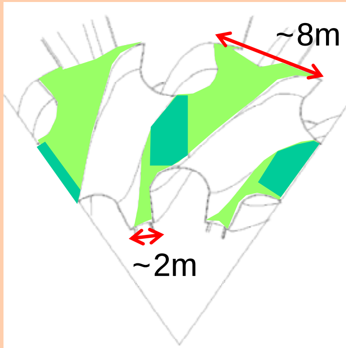




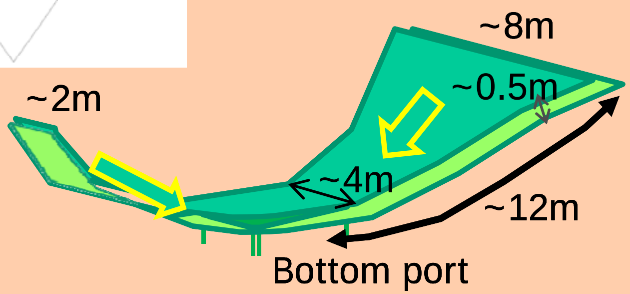
Estimation of the neutron flux distribution by the 3D simulation code MCNP【Blanket TG】







Design of the pumping ducts to realize a high-conductance【In-vessel components TG】

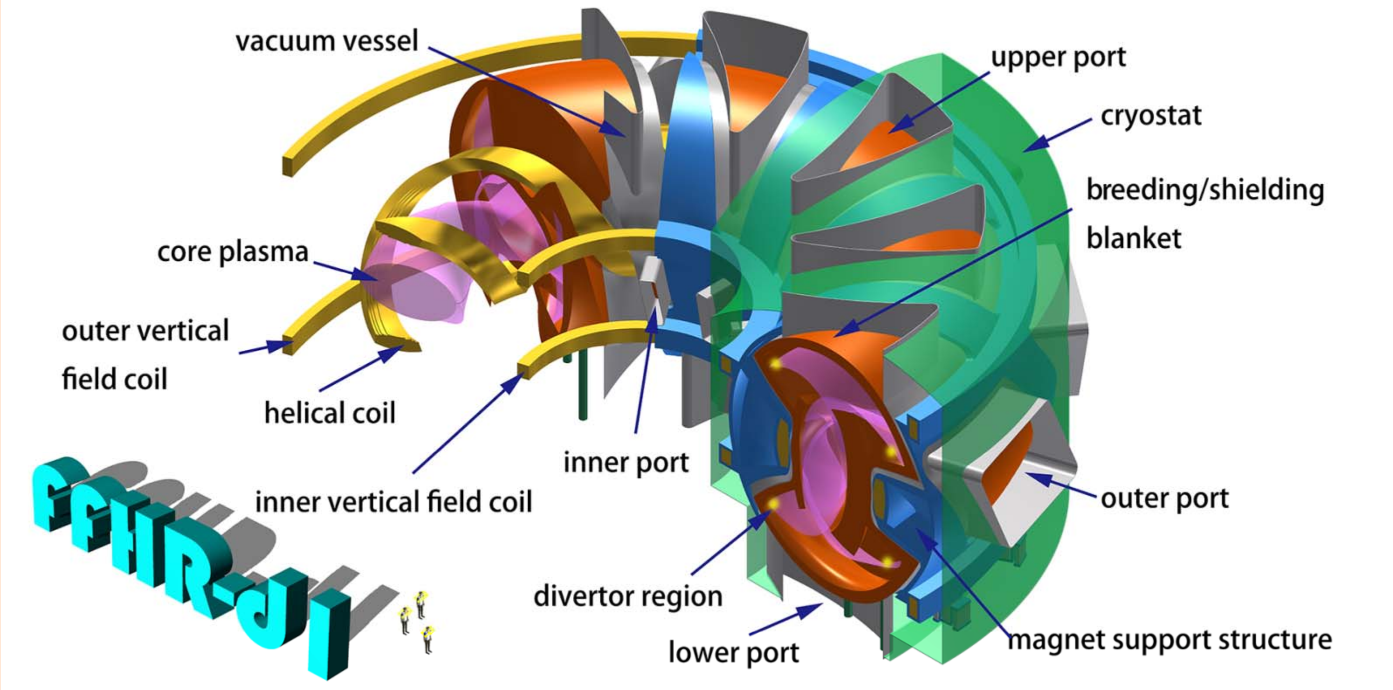


Numerical modeling of the blanket poloidal shape being consistent with plasma and divertor legs at arbitrary toroidal angle ϕ【Design integration TG】

2. 3D Designs of FFHR-d1 and New Proposals

1) SC Magnet and Support Structure

In ISFNT-11: H. Tamura



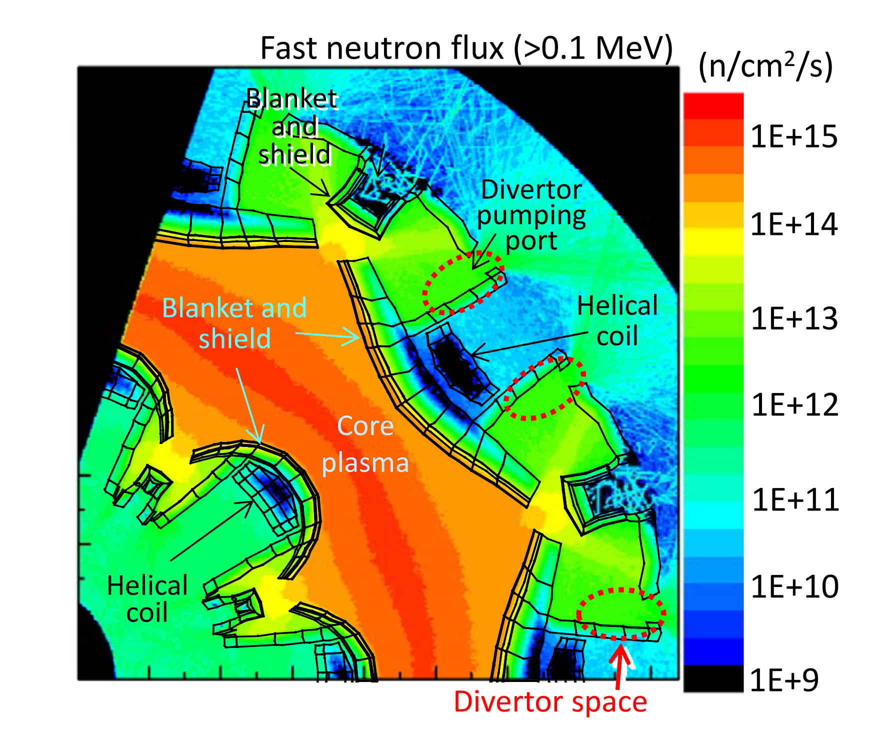
Large apertures can be secured in the coil support

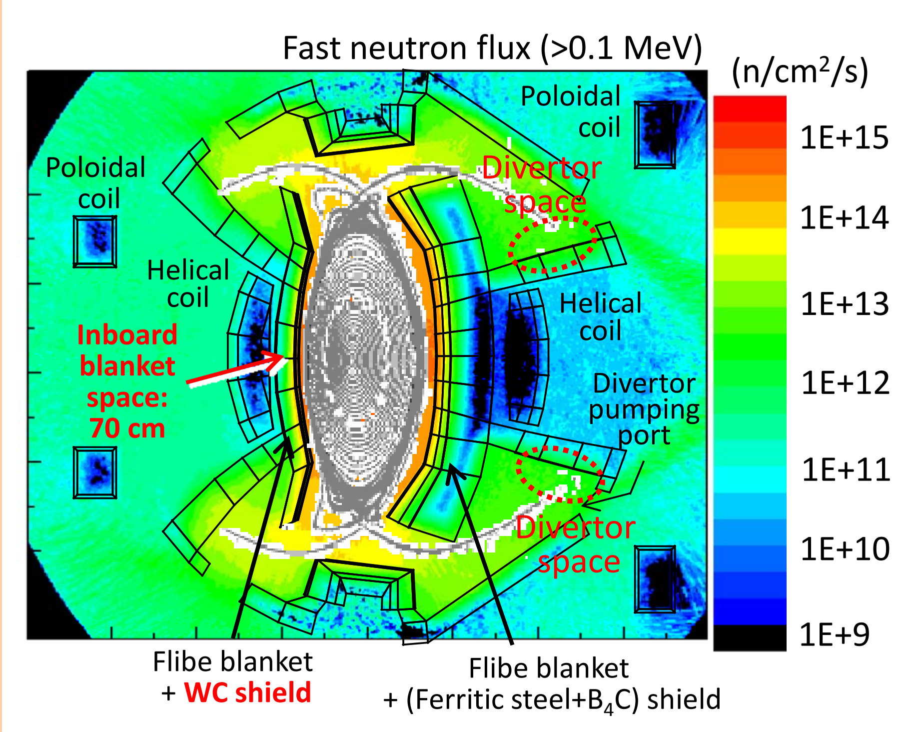
- made of 250mm-thick stainless steel 316
- with a maximum stress of 600MPa level
- for a total magnetic energy of 160GJ.

2. 3D Designs of FFHR-d1 and New Proposals

2) Neutronics evaluations

In ISFNT-11: T. Tanaka

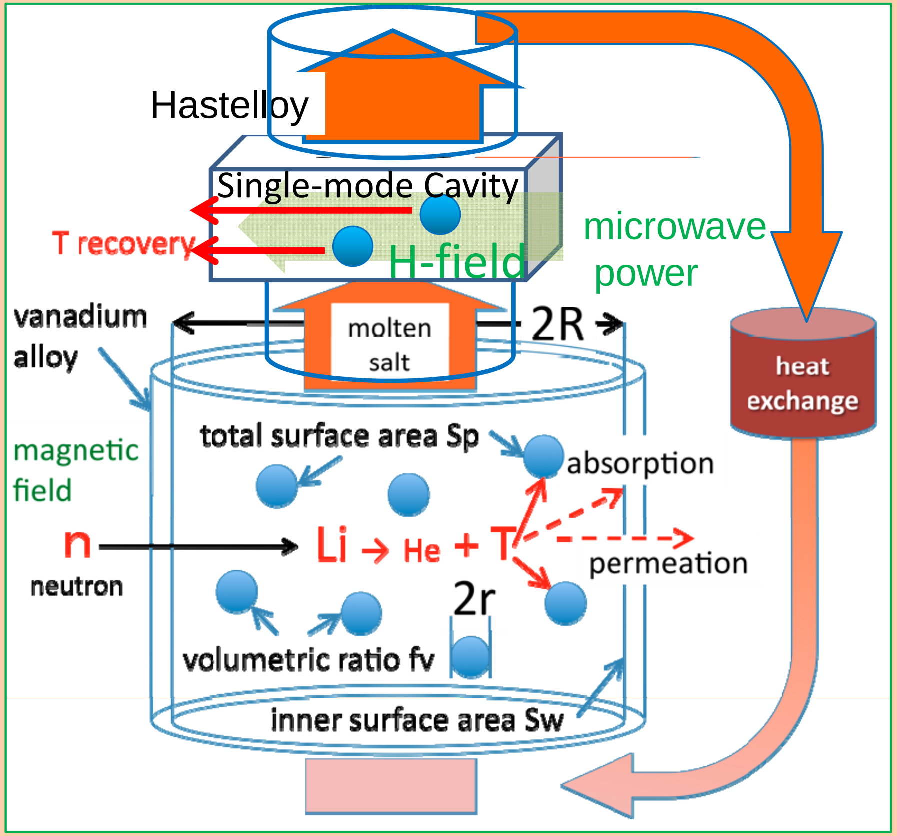




- More than 10 years' operation of magnets is feasible using the inboard side shield made with WC, where the total TBR is 1.08 with 90% Li-6 enrichment.
- The divertor targets can be efficiently shielded from fast neutrons, expanding the range of material choice (e.g., Cu alloys).

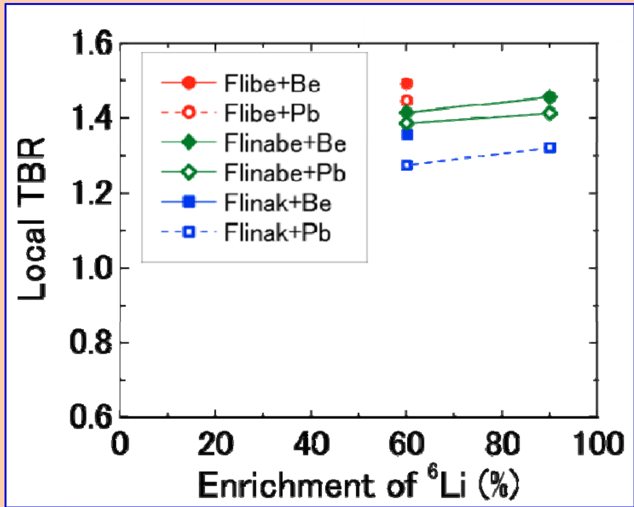
2. 3D Designs of FFHR-d1 and New Proposals

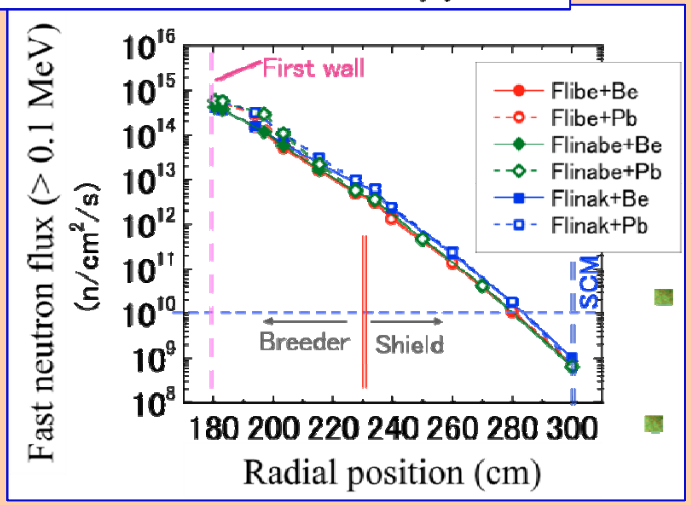
3) Flinabe Blanket mixed with Metal Powder



In FED(2014) A.Sagara

Flinabe is comparable to Flibe regarding TBR and nuclear shielding.





Newly proposed based on preliminary experiments,

- **to effectively increase hydrogen solubility**
- **to increase thermal efficiency up to 46% with V alloys or 38% with Ferrite.**

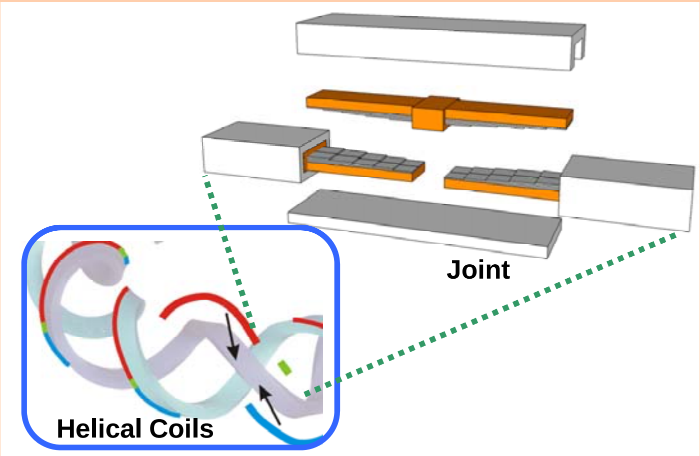
2. 3D Designs of FFHR-d1 and New Proposals

4) Fabrication of Helical Coils with Segmented HTS Conductors

proposed as a promising method by connecting half-helical-pitch segments of 100kA-class YBCO

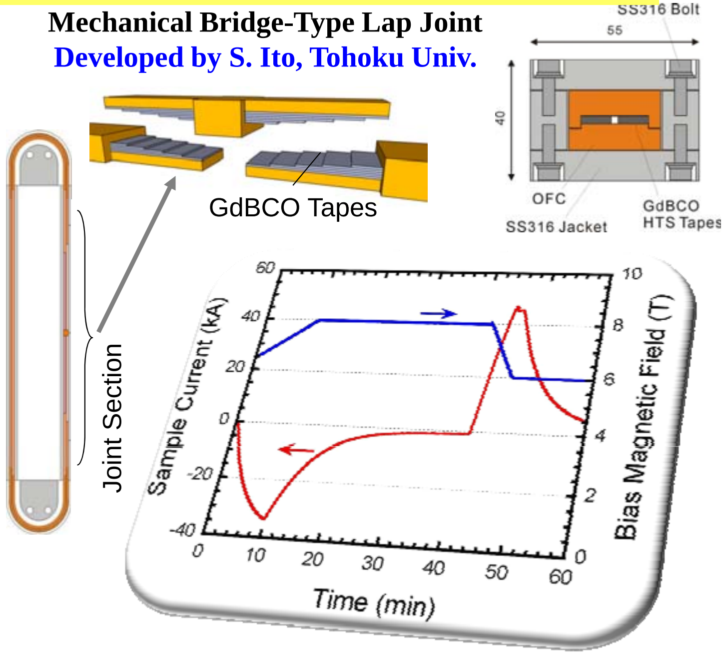
- High cryogenic stability,
- High efficiency for cooling,
- Facilitation of the winding

NIFS R&D achieved 45 kA at 20 k, 6 T, and low resistance (< 5nΩ)
A 100 KA-class conductor sample is being fabricated to be tested.



Collaboration with Tohoku Univ.

Mechanical Bridge-Type Lap Joint
Developed by S. Ito, Tohoku Univ.



The electrical power for removing the Joule heating from all 7800 joints in two helical coils is <4MW (0.1% of Pfus) for 20K operation, which is acceptable.

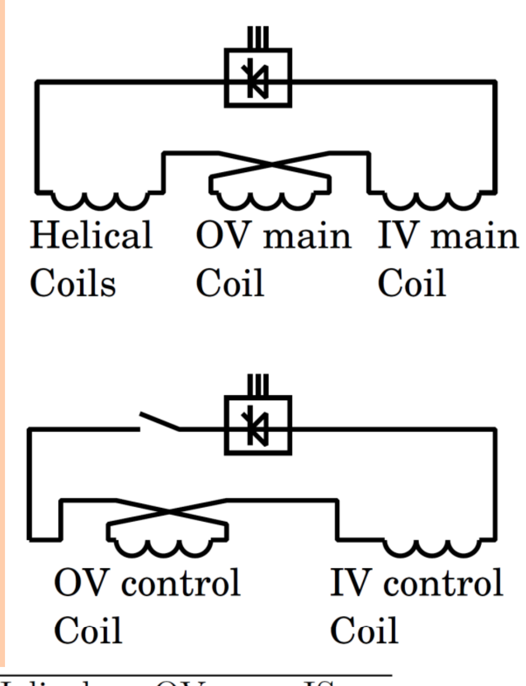
N. Yanagi, S. Ito, H. Hashizume et al., MT-23 (2013) in Boston

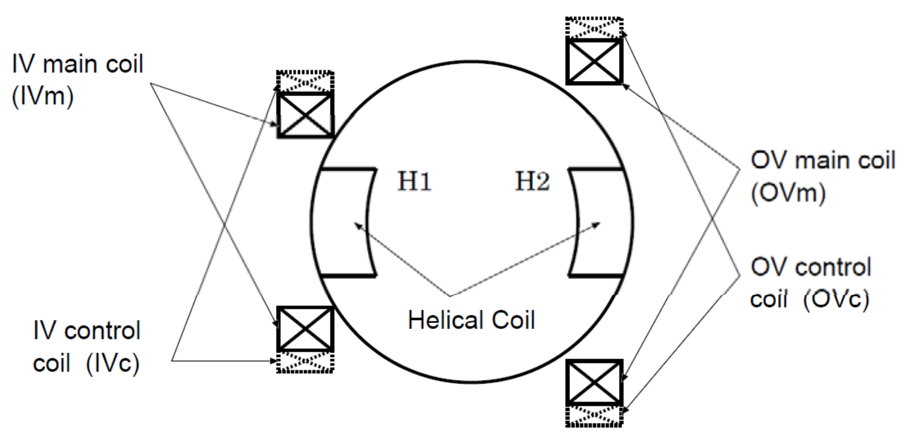
2. 3D Designs of FFHR-d1 and New Proposals

5) Design of DC power supplies for superconducting magnet

In ISFNT-11: H. Chikaraishi

- Connected in series and excited by one power supply.
- Electro-magnetic forces on the helical coil are always balanced.
- The total capacity of the power supplies is <15 MW (0.5% of P_{fus}).





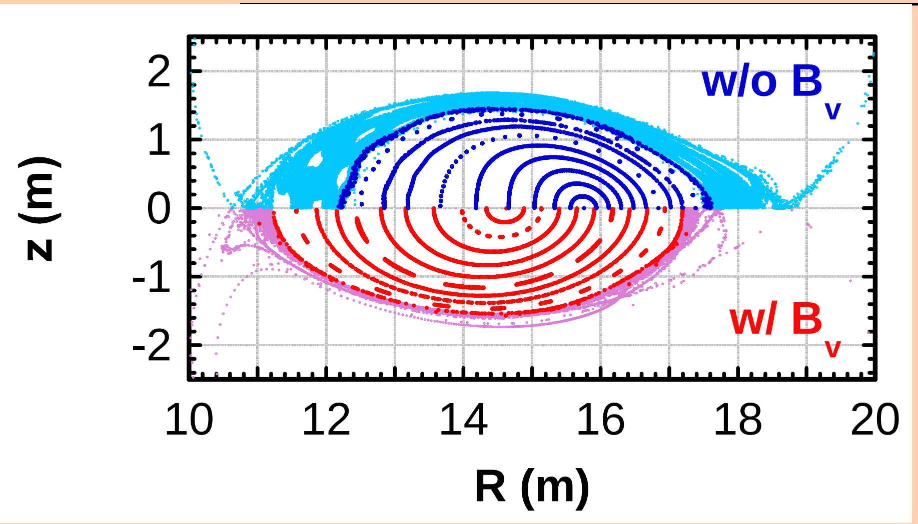
	Helical	OV	IS
Main coil			
Total current MA	36.66	20.14	21.61
Coil current kA	97.76	97.76	97.76
Turn number	375	-206	221
		(0.05)	(0.06)
Control coil			
Total current	0	0.263	3.11
Coil current kA	-	7.51	7.51
Turn numbers	-	-35	405
	-	-	(-0.06)

3. Improvement in Ignition Core Plasma and Design Flexibility

1) Physics Analyses of FFHR-d1 Core Plasma

Recently it has been found,

- **When the helical coil pith γ_c = 1.20 is selected, instead of the former 1.25, Shafranov shift of the magnetic axis in the high-β can be mitigated by controlling the vertical magnetic fields, Bv.**
- **In this case, the neoclassical heat loss is roughly ½ of the α heating power and the direct loss of α particles calculated by GNET and MORH codes can be reduced to a tolerable level of 10 ~ 20 %.**



Magnetic surface reconstructed by HINT2 code using the β profile extrapolated from LHD data by DPE method (by J. Miyazawa)

FFHR-d1
γ_c = 1.25

improved

FFHR-d1
γ_c = 1.20
(higher Aspect ratio)

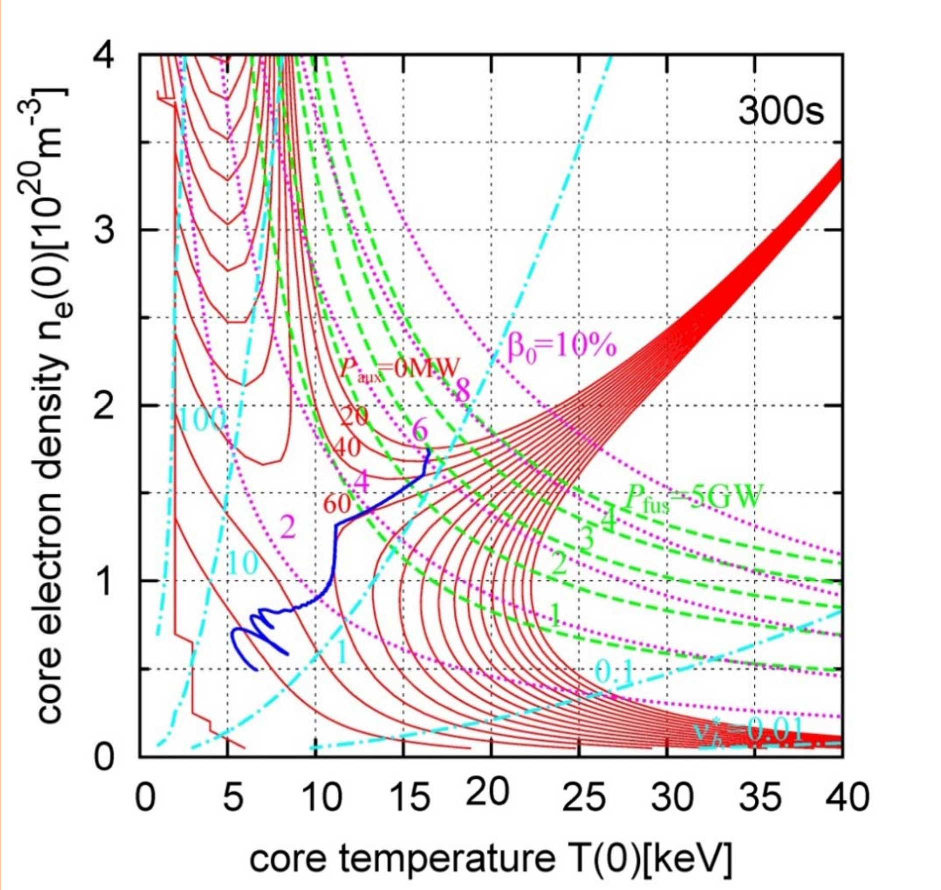
3. Improvement in Ignition Core Plasma and Design Flexibility

2) Start-up Scenario of Ignition Core Plasma

In ISFNT-11: T. Goto

In the quasi-1D particle balance model, the feedback control of the line-averaged electron density is adopted, instead of the fusion power, <u>because there is time delay</u> corresponding to the time constant of the diffusion on the density profile.

By controlling the external heating power, smooth ignition access by a simple on-off control of the fixed-size pellet injection with 10Hz repetition can be achieved in 300 sec with maximum heating power of 60MW.



3. Improvement in Ignition Core Plasma and Design Flexibility

3) Improvement of the Pellet Fueling Scenario

Re-evaluation of the particle diffusion coefficient, which degrades with increasing heating power density.

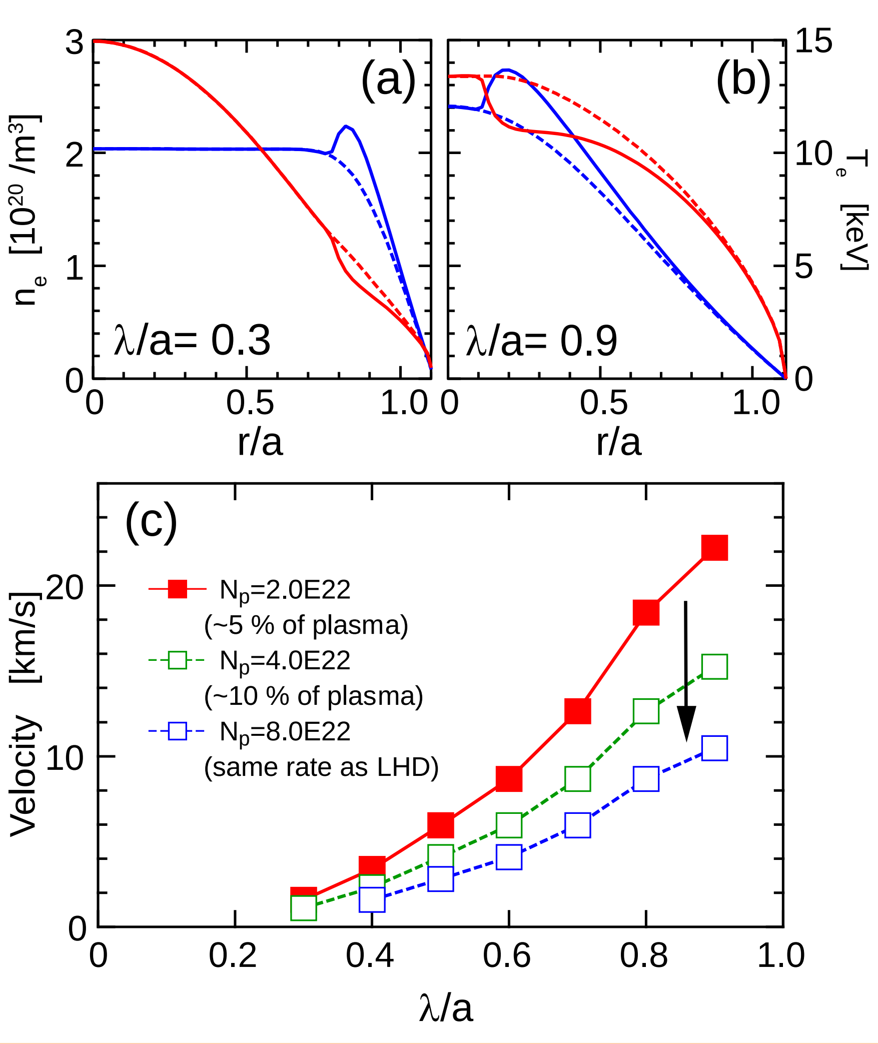
However,

- High-speed injection >15km/s cannot be attained in present-day technology.
- Large pellet size may cause unacceptable fusion output variation

Therefore,

Relatively shallow pellet penetration to λ/a = 0.3 with pellet injection velocity at 1.5 km/s is **an acceptable compromise** between pellet injection technology and burning plasma performance.

R. Sakamoto, submitted to Nucl. Fusion



3. Improvement in Ignition Core Plasma and Design Flexibility

4) Sub-Ignition Design of FFHR-c1 before Demo

In FED (2014) A. Sagara

Proposed as “before-demo, compact and component-test” to keep design flexibility on FFHR.

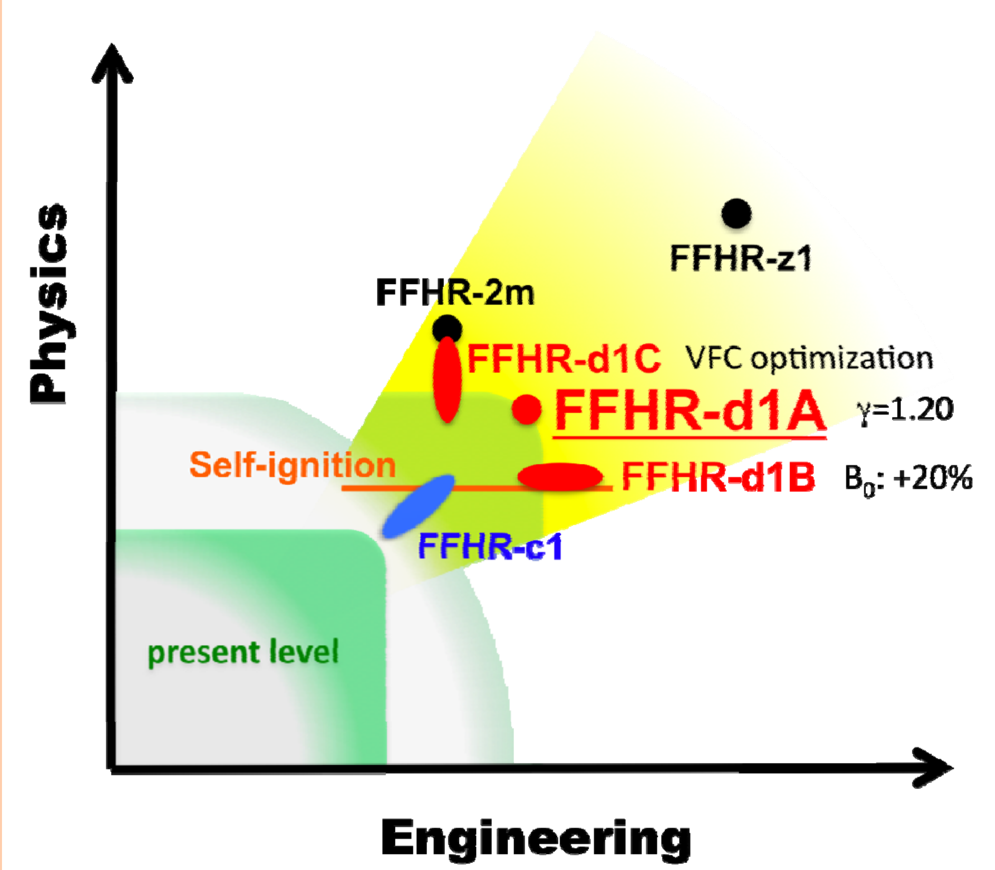
	FFHR-d1	FFHR-d1A	FFHR-d1B		FFHR-c1.0	FFHR-c1.1	FFHR-c1.2
R_c (m)	15.6	←	←	←	13.0	←	10.4
$V_{p,vac}$ (m^3)	1,877	1,421	←	←	823	823	419
B_c (T)	4.7	←	5.6	←	4.0	5.6	←
W_{mag} (GJ)	162.5	←	223.5	←	67.8	125.1	61.4
γ_c	1.25	1.20	←	←	←	←	←
η_α	1.0	0.85	←	←	←	←	←
f_β	5.1	3.7	1.9	1	←	←	←
β_0	9.1	9.1	4.5	2.4	←	←	←
P_{aux} (MW)	0	←	←	27	53	40	49
P_{fusion} (GW)	2.7	3.0	1.5	0.43	0.065	0.25	0.13
Q	∞	←	←	16	1.2	6.2	2.6
Φ_n (MW/m^2)	≤ 1.5	1.5	0.73	0.21	0.05	0.18	0.14

~ 0.5 dpa ~ 2 dpa / year is possible

As a consequence, a multi-path strategy for FFHR-d1 has been introduced

In FED (2014) A. Sagara

- FFHR-d1A is the base for 3D designs with a modified Aspect ratio by reducing the Shafranov shift,
- FFHR-d1B is a flexible design for the ignition core with the magnetic (B) field enhanced by 20%,
- FFHR-d1C is another flexible design with Configuration optimization of vertical
- field coils,
- FFHR-c1 is a sub-ignition version as "before-demo, compact and component-test"
- (FFHR-2m is a commercial reactor.)
- (FFHR-z1 is an ideal reactor.)

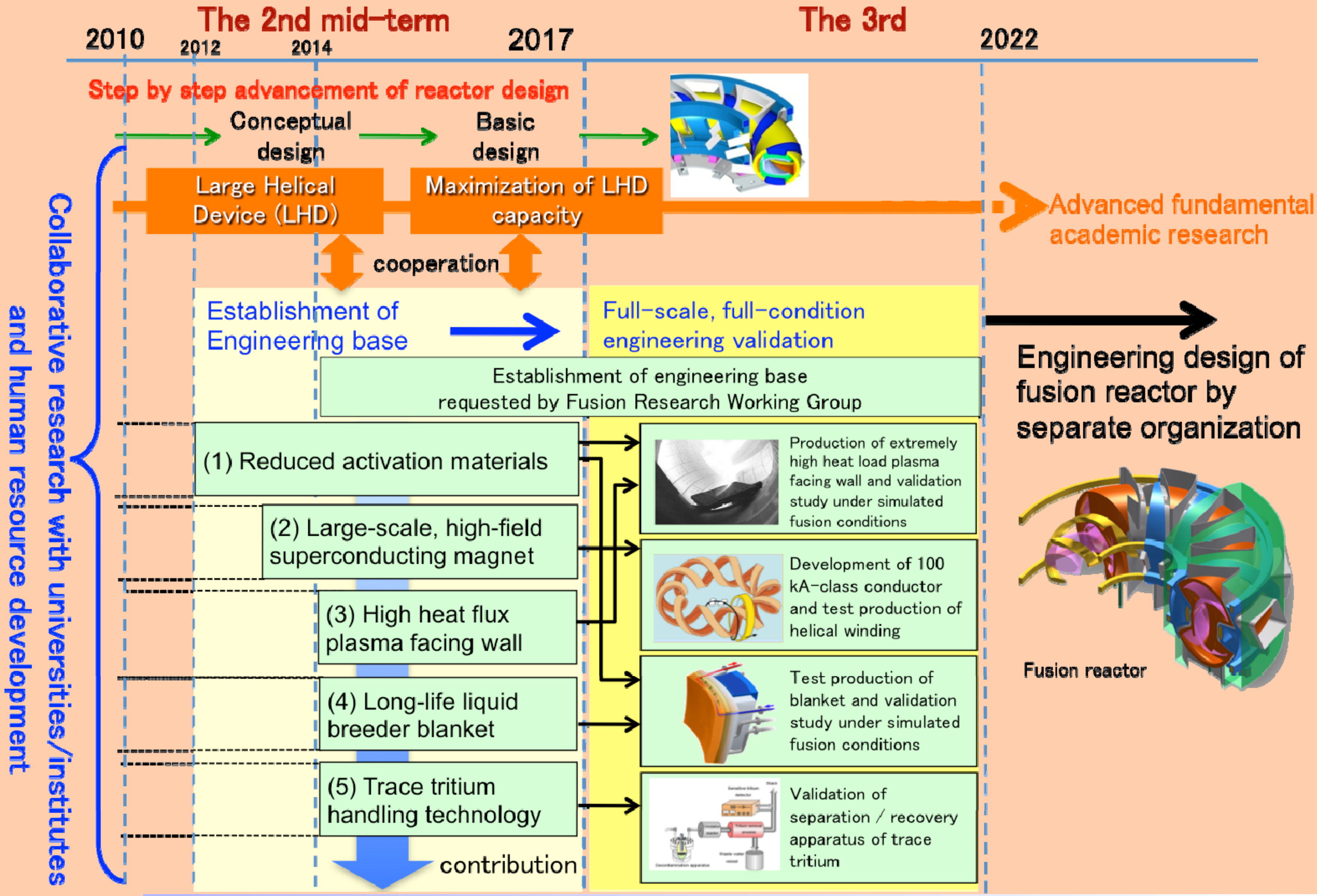


Summary

1. Fusion Engineering Research Project (FERP) in preparation for DEMO has been launched in NIFS by starting the re-design of the LHD-type helical reactor FFHR-d1.
2. In the first round, the main parameters were selected.
3. The second round is preparing detailed 3D design of the superconducting magnet support structures, and 3D neutronics analyses, where the divertor targets can be efficiently shielded from fast neutrons.
4. A new Flinabe blanket mixed with metal powder was proposed.
5. Fabrication of helical coils by connecting half-helical-pitch segments of 100kA-class YBCO high-temperature superconductors is proposed as a promising method.
6. Also in progress is improvement of the first round of the core plasma design, ignition start-up analyses, and fueling scenario.
7. As a consequence, a multi-path strategy on FFHR-d1 has been introduced with versions of -d1A, -d1B, and -d1C, where design flexibility is expanded to include sub-ignition with options FFHR-c1 for "before demo, compact, and component-test."

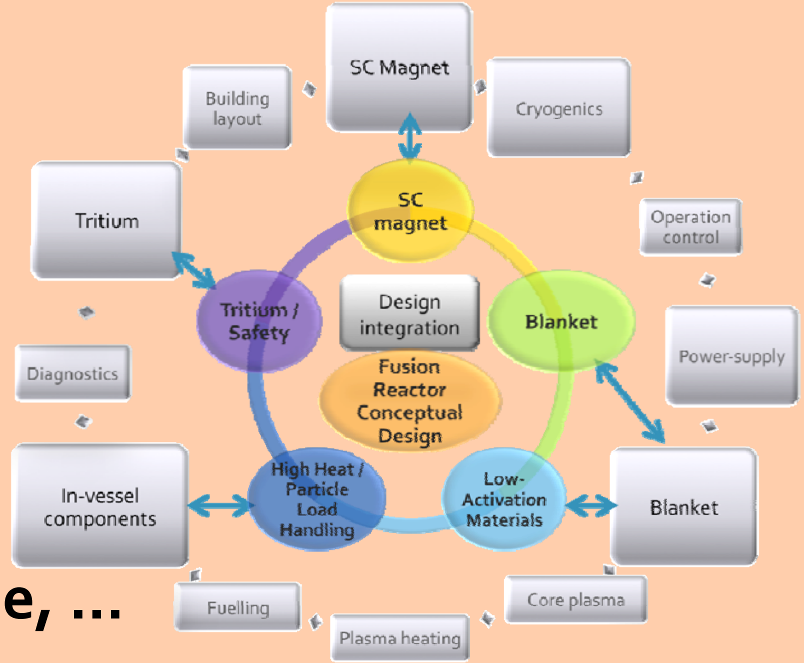
Roadmap of FERP

14, March 2014 @ UCSD



ITER / BA activity : DEMO R&D, conceptual design, JT-60SA, IFMIF/EVEDA

Large Supplementary Budget of 2.4 Billion JPY in FY2012



(1) SC Magnet

- 13 T SC magnet test facility, ...

(2) Blanket

- Twin loop with 3T SC magnet, ...

(3) Low-Activation Material

- HIP device, TEM, creep test device, ...

(4) Divertor

- High heat load test device of 10 MW/m², pelletron tandem accelerator of 1 MV, ...

(5) Tritium

- Gas / liquid analyzer, ...

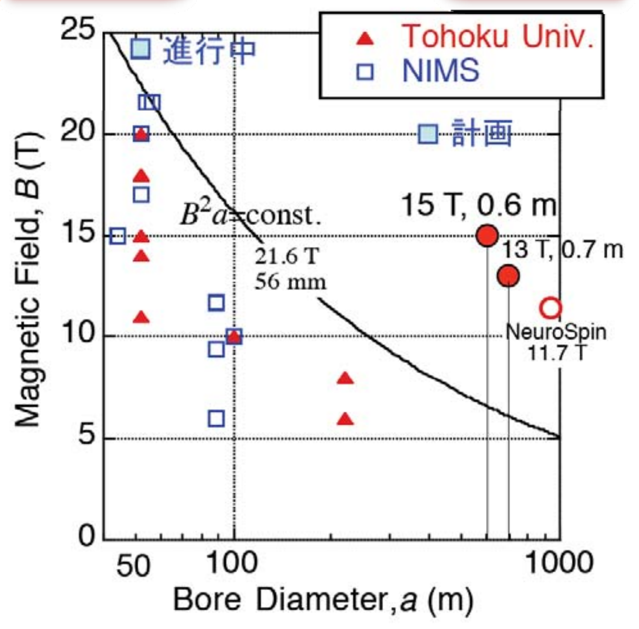
14, March 2014 @ UCSD

15 T SC Magnet Facility

R&D by SC magnet TG

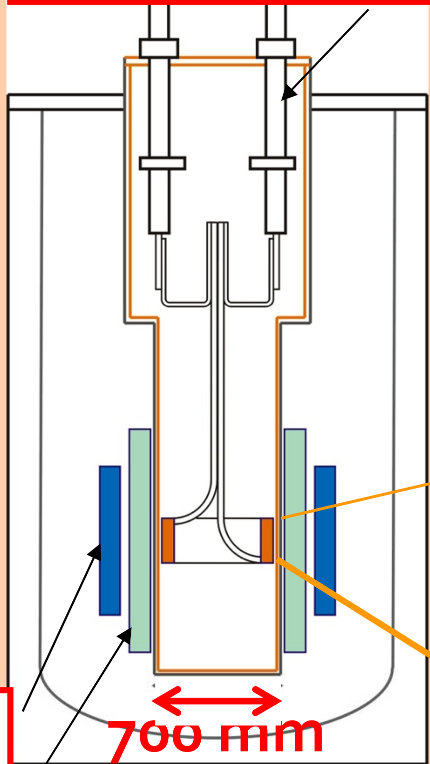
The superconductor testing facility at "Superconducting Magnet Systems Research Laboratory" will be upgraded (after 25 years operation) to increase the bias magnetic field from 9 T to 15 T so that 100 kA-class conductor samples will be tested at temperature 4 – 50 K

World's Highest B in a 700 mm Bore

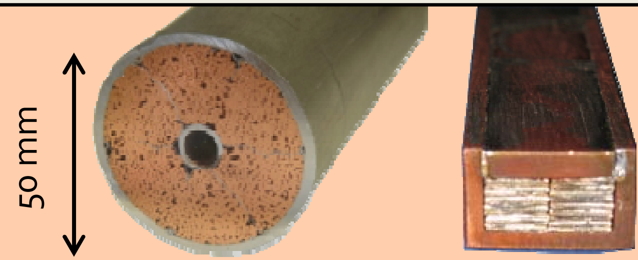


<High Field Magnets in Japan>

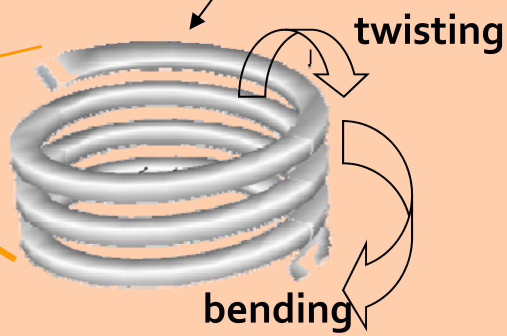
3. Temperature-variable current leads



1. Outer coil
2. Inner coil



Coil sample with 100 kA-class superconductor



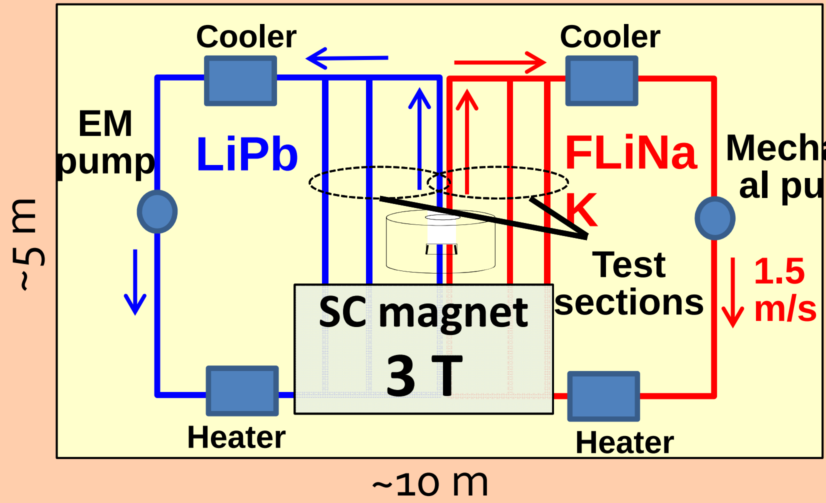
Twin Loops with 3T SC Magnet

R&D by Blanket TG

World's Highest B ⊥

✓Orosh²i-2

- Operational Recovery Of Seperated Hydrogen and Heat Inquiry
- Forced circulation loops of FLiNaK (~500 ℃) and LiPb (~300 ℃)
- Integrated test stand with a SC magnet of **3 T**



Basic configuration of ***Orosh²i-2***

Specifications of Orosh²i-2

- ***Pipe diameter :*** *1 inch*
- ***Normal operation temp.*** *: FLiNaK 500℃, LiPb 300℃*
- ***Maximum flow velocity*** *: ~1.5 m/s* ***Inventory*** *: ~100 L*
- ***Magnetic field*** *: max. ~3 T (CS magnet), 50 cm Φx 15 cm*

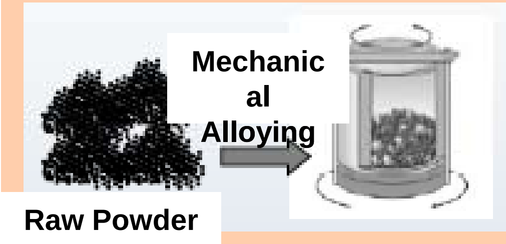
✓Simulation of temperature and flow velocity in fusion blanket

✓Integrated tests of MHD pressure drop, control of laminar and turbulence flow, hydrogen and heat recovery, corrosion behavior etc. under intense magnet field.

✓Test stand for elemental technologies developed in collaborative studies.

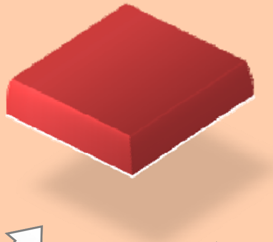
HIP Device, TEM, Creep Test

R&D on low-activation materials by Blanket TG



□ Planetary ball milling

Nano-particle dispersion strengthened materials

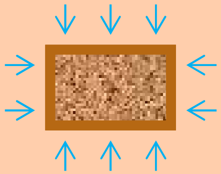


Surface morphology and composition

■ Field Emission SEM（existing）

□ TEM

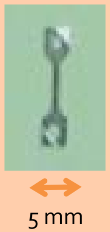
□ EDS



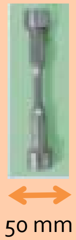
HIP

(Hot Isostatic Press)

□ High temperature isostatic sintering and joining test system（Fabrication of medium size test pieces）



Miniature size



Standard size

□ High temperature creep test facility

□ High speed impact test facility

- ✓Characterization of nano-particle dispersion
- ✓Evaluation of high temperature strength and low temperature ductility

  Target （non-irradiation）

  Yield stress : 300 MPa <

  Thermal creep : < 1 % at 100 khr

  DBTT : < RT
- ✓Establishing fabrication technology

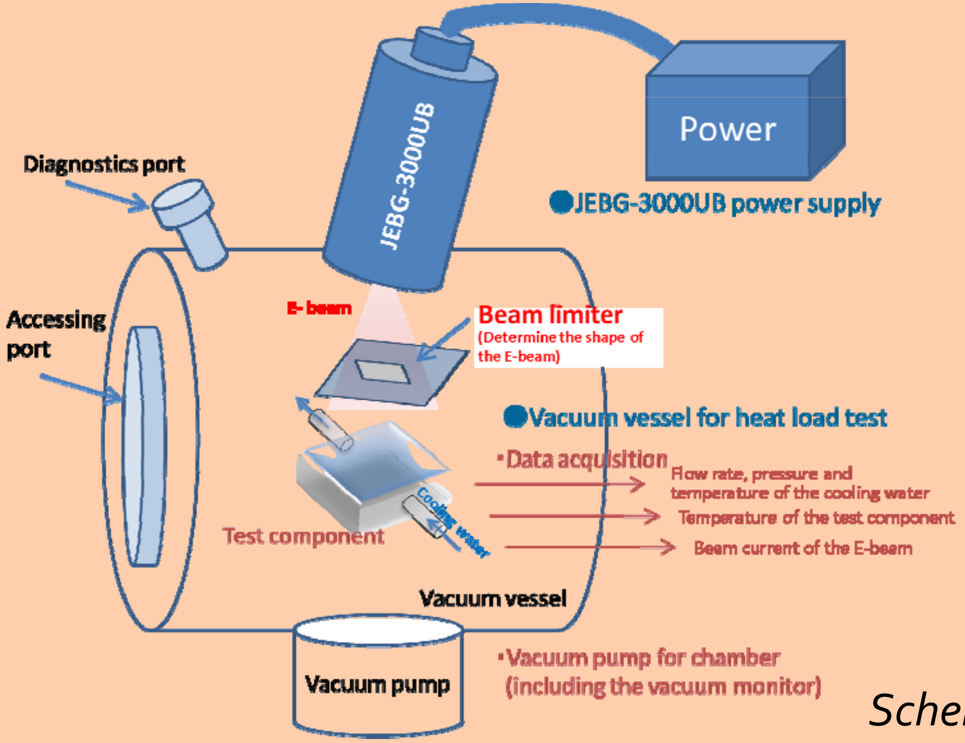
High Heat Load Test Device

R&D by In-Vessel Components TG

✓ACT-2

- Ultra high heat flux test stand
- 10 MW/m^2 of heat loading by 300 kW electron gun
- Large vacuum vessel
- R&D on material, cooling media, and bonding technique
- Realistic scale components

300 kW electron gun



*Schematic of **ACT-2***

Accelerator for Material Test

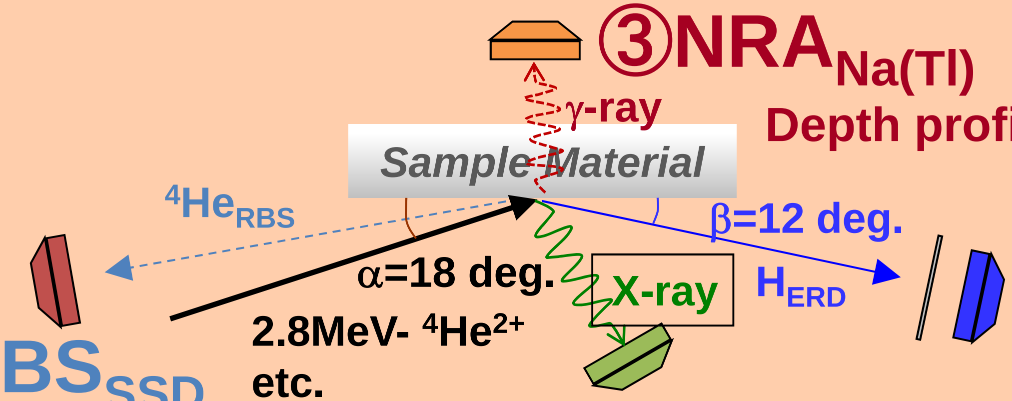
R&D by In-Vessel Components and Blanket TGs

✓Non-destructive analysis

- By 1 MV pelletron tandem accelerator
- Quantification of the retained H
- Multiple analysis of RBS, ERD, NRA, and PIXE

1 MV tandem accelerator

③NRA$_{Na(Tl)}$
Depth profile of H isotopes



①RBS$_{SSD}$
Quantitative analysis of deposition element

④PIXE$_{SDD}$
High resolution (ppm) element analysis

②ERD$_{SSD}$
Quantitative analysis of the plasma particles