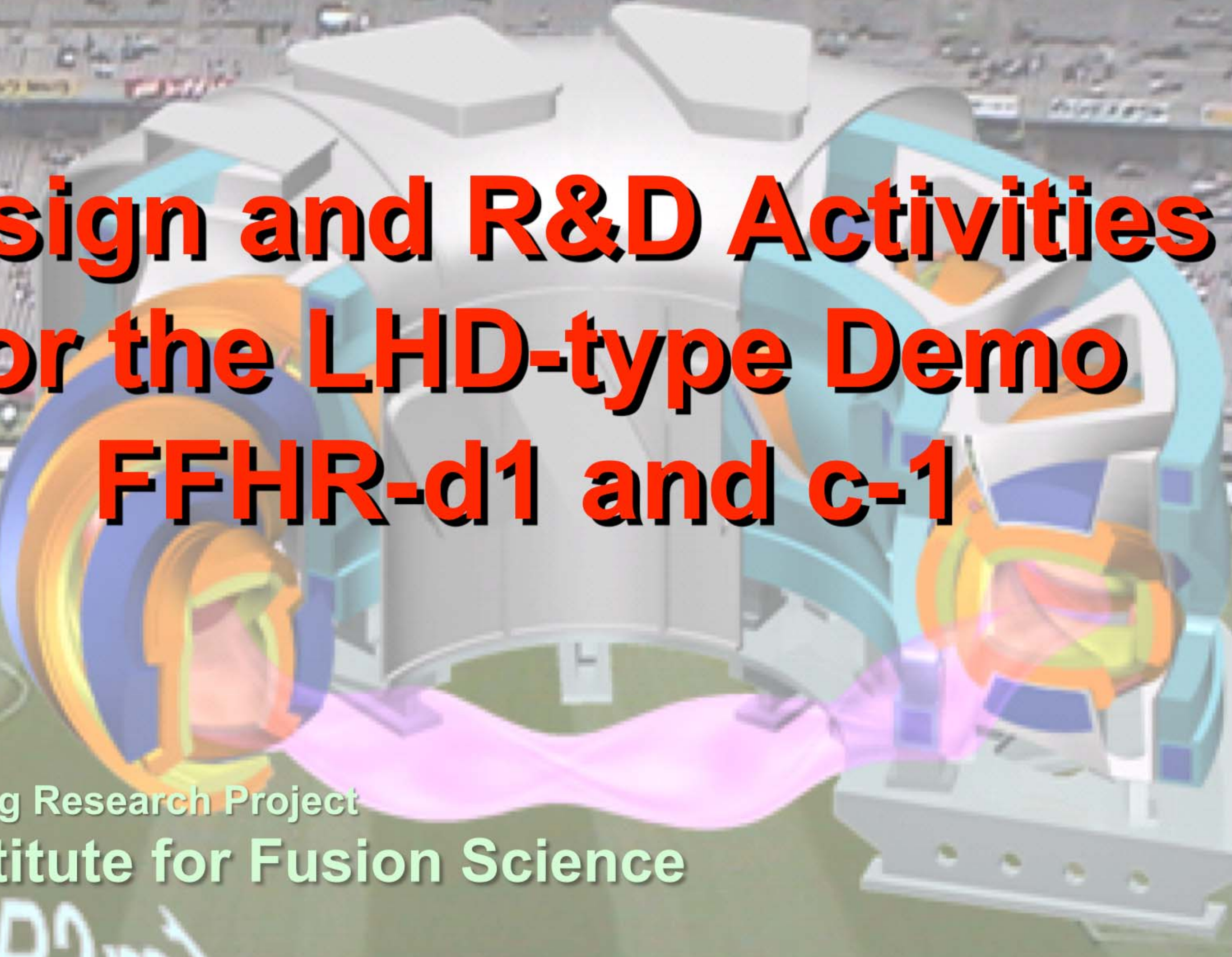
Design and R&D Activities for the LHD-type Demo FFHR-d1 and c-1



Akio Sagara
Fusion Engineering Research Project
National Institute for Fusion Science

Japan-US Workshop on
Fusion Power Plants and Related Advanced Technologies
February 26-28, 2013 at Kyoto University in Uji, JAPAN

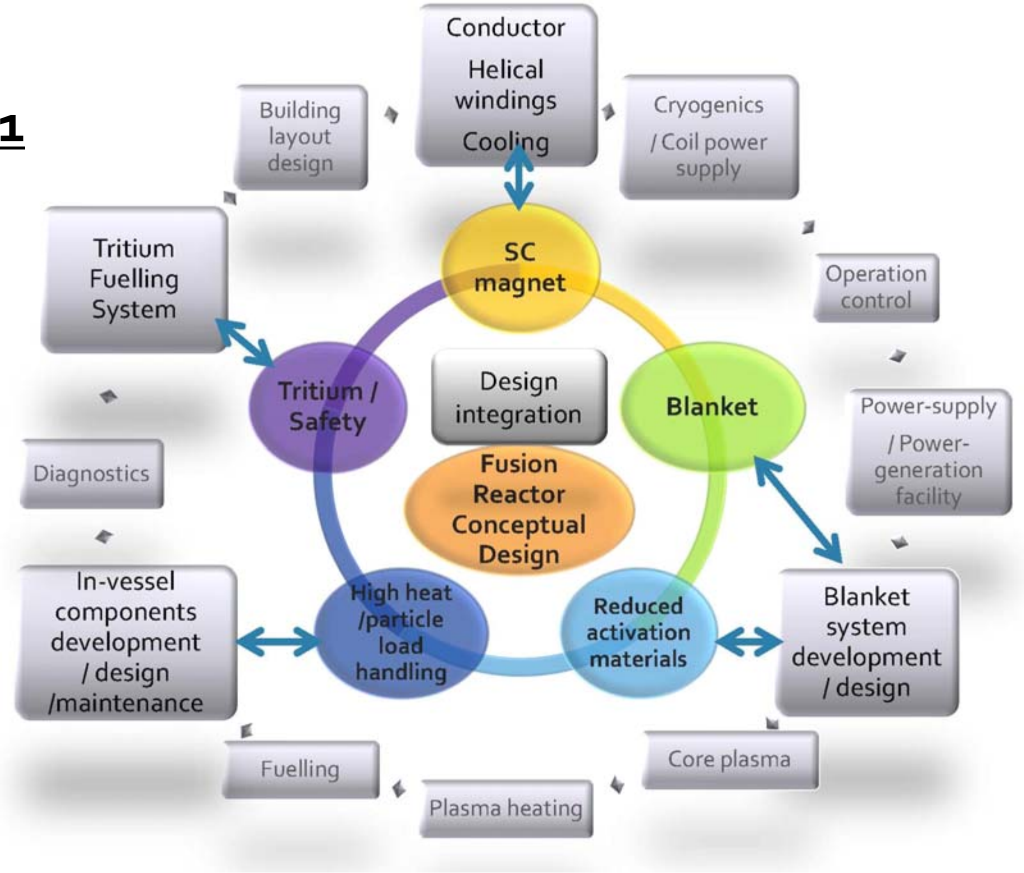
Fusion Engineering Research Project in NIFS

Conceptual design of the helical DEMO reactor FFHR-d1 is ongoing together with various R&D

- ✓ **FERP** aims at establishing the technological basis to realize the helical DEMO

  -FERP consists of 13 task groups and 44 subtask groups since 2010, and many outcomes in FY2012.

- ✓ Through the design activities of the helical fusion DEMO reactor **FFHR-d1**
  - Structural analysis using 3-D CAD
  - 3-D neutronics analysis
  - Core plasma design
  - An interim report (in Japanese) will appear in this April
- ✓ And various **R&D**
  - Superconducting magnet
  - Blanket
  - Materials and coating method
  - Divertor
  - Tritium and safety



13 task groups and 5 major R&D in FERP since 2010

13 tasks and 44 sub-tasks for Design and R&D with collaborations

Implementation structure of 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-scele high-hield 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

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

Promotion meeting
by Exc. 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	Task setting, Project managment	Sagara, Miyazawa, T. Goto
	Helical DEMO conceptual design	T. Goto, Miyazawa, Sagara
Building layout T. Goto	Layout design, process	A.Nishimura, 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	S. Nishimura
Operation control Mitarai (Tokai Univ.)	Safety analysis, control system	Uda
	Burn control	Mitarai
	Data processing	Nakanishi
Core plasma Miyazawa	High performance plasma	Miyazawa, T. Goto, Narushima
	TCT effect, α particle loss	Yokoyama, Murakami (Kyoto U)
	Ignition Scenario	Mitarai
Plasma heating Tsumori	NBI	Tsumori, Osakabe
	ECH	Igami, Yoshimura, Idei (Kyusyu U), 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	H. Yamada
	Charge exchange spectroscopy	Yoshinuma

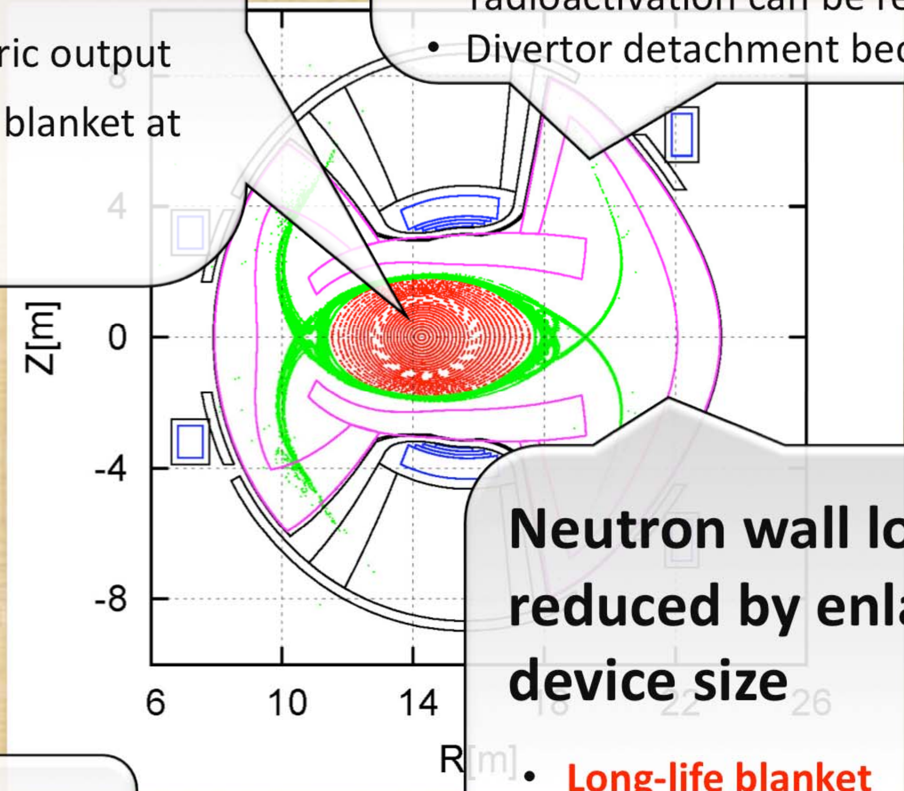
Attractive features of a helical reactor

Flux surfaces are generated by external superconducting coils

- **Plasma current drive is unnecessary**
- **Steady state operation is easy**
- Circulating energy is ~3 % of electric output
- Plasma does not contact with the blanket at start up / shut down / emergency
- No plasma current disruption

Divertors are placed behind the blankets

- **To avoid direct neutron irradiation** (both the neutron damage and radioactivation can be reduced)
- Divertor detachment becomes easy

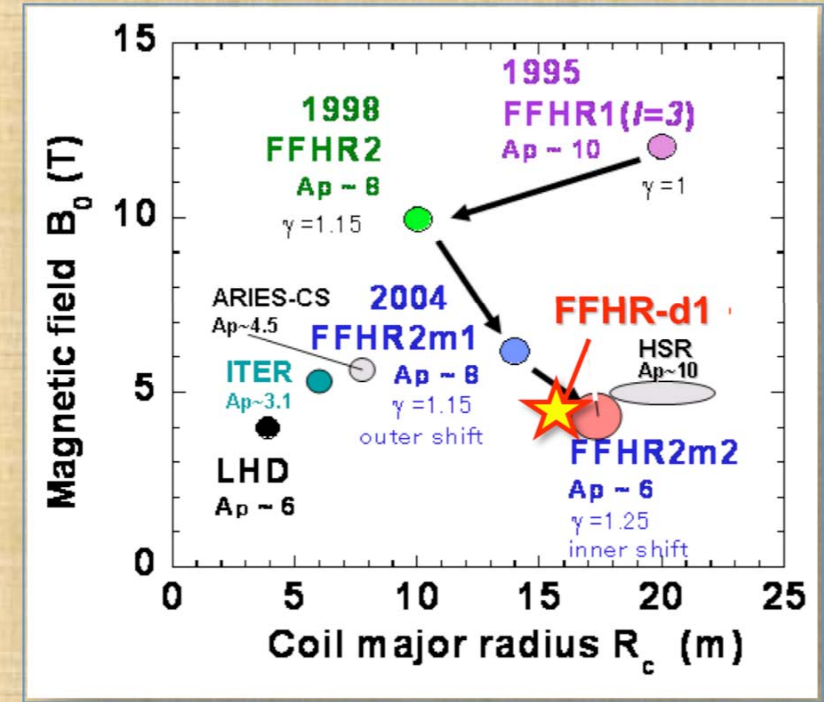


Neutron wall load can be reduced by enlarging the device size

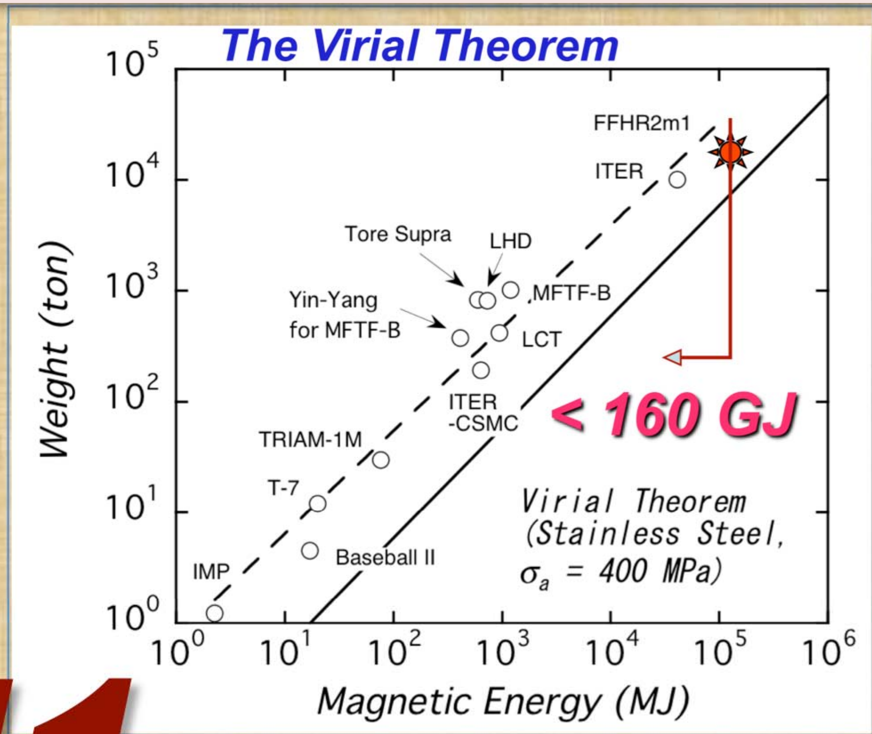
- **Long-life blanket**
- Low decay heat

Open space inside the torus

- No need of central solenoid



- Neutron wall loading 1.5 MW/m^2 (since 1995)
  for ***high availability with long-life components***
- Net TBR > 1.0
- Magnetic energy < 160 GJ



DEMO

FFHR-d1

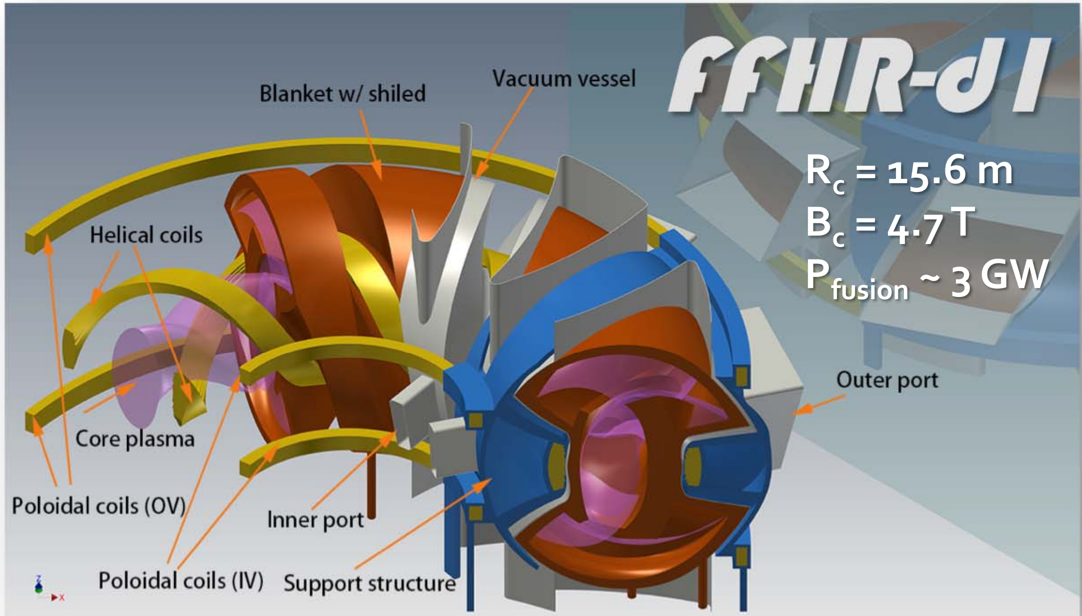
Design Parameters of FFHR-d1

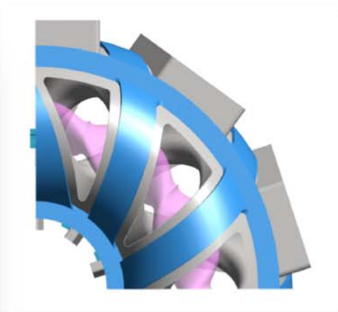
A. Sagara et al., Fusion Eng. Des. 87(2012)594.

			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)	Γ_{div}	MW/m^2			5	7.2	1.9	8.1
(on average) Total capital cost		G$(2003)		4.6	5.6	7.0		
COE		mill/kWh		155	106	93		

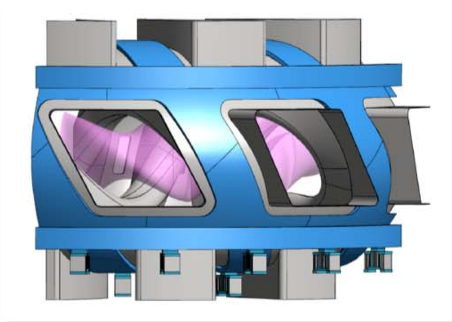
3-D CAD Design and 3-D Neutronics Analysis

Large ports can be used for maintenance, the neutron load on the divertor can be reduced

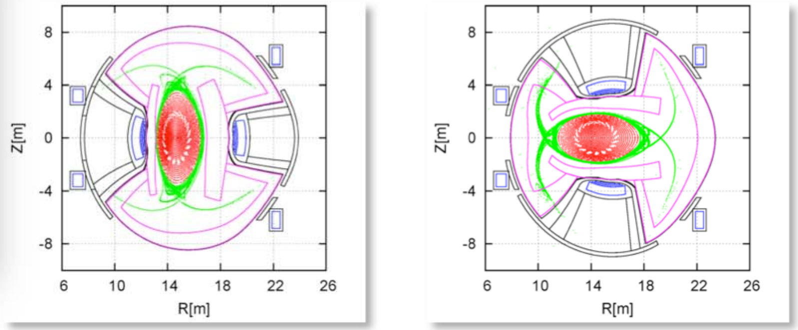




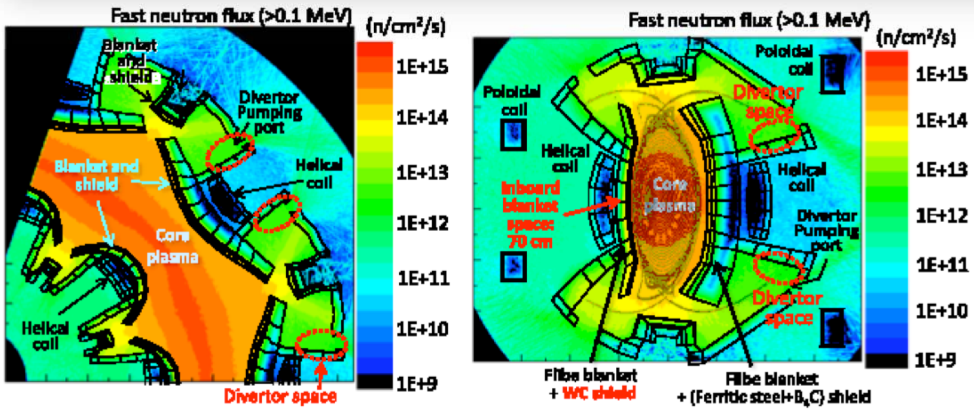
Upper Port



Outer Port



Vertical slices at a vertically elongated (left) and a horizontally elongated (right) plasma cross section



Results of 3-D neutronics analysis by MCNP

- ✓ Large upper and outer ports are included in the design
  - maintenance accessibility increases
- ✓ Divertors are located behind the blanket
  - to avoid the direct neutron irradiation

Detailed Physics Analysis of the Core Plasma

FFHR-d1 is a "semi-optimal heliotron" where Shafranov shift at high-beta is mitigated and both the neoclassical transport and the alpha loss are reduced

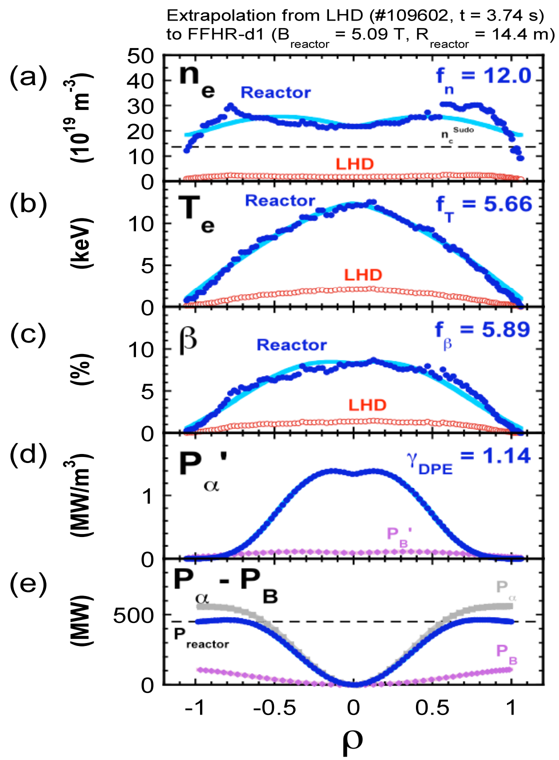
"Multifarious Physics Analyses of the Core Plasma Properties in a Helical DEMO Reactor FFHR-d1"

J. Miyazawa et al., IAEA FEC2012, FTP/P7-34

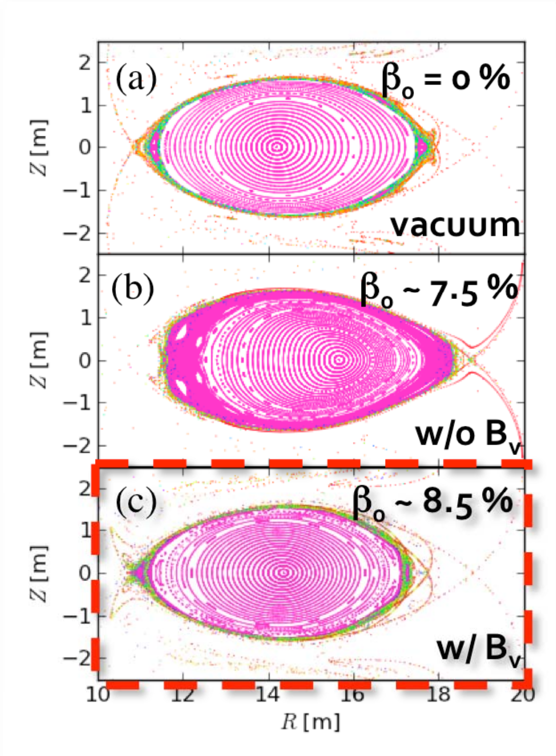
(highlighted in the summary talk of FTP by Stan Milora)

with LHD Project

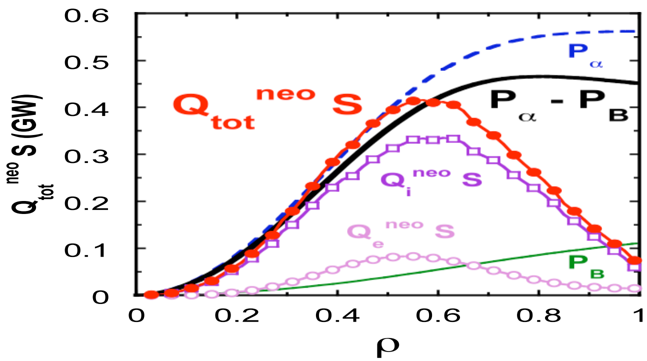
with Numerical Simulation Research Project



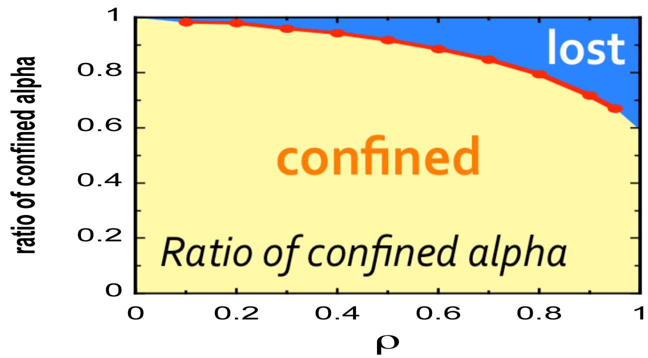
1. Radial profiles are extended directly from LHD using the gyro-Bohm model



2. Magnetic surfaces at high beta can be maintained by adjusting B_v



3. Neoclassical thermal transport allows sustained burning plasma



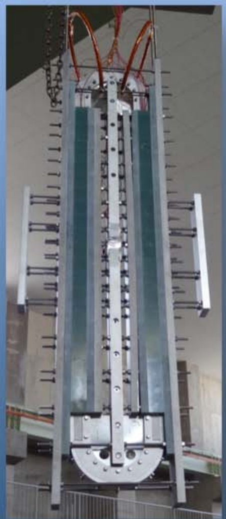
4. Direct loss of alpha particles is tolerable

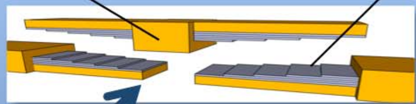
Progresses in the R&D

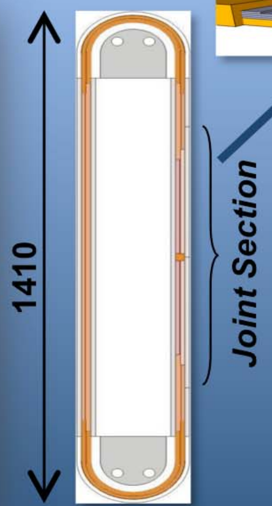
Toward the real-scale and real-environment tests in the future

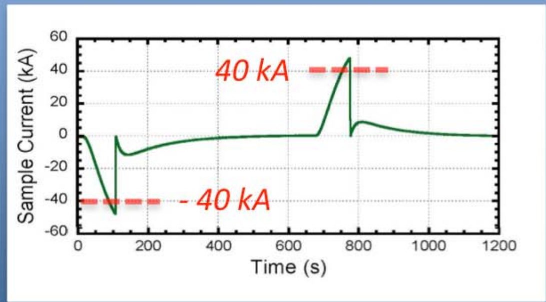
Development of High-Temperature Superconductor (HTS)

with Tohoku Univ.









✓ An HTS conductor sample has achieved 40 kA at 20 K with a mechanical joint

FLiNaK circulation loop "Orosh²i-1"

with Kyoto Univ.

The first molten salt coolant loop for
Operational
Recovery
Of
Separated
Hydrogen and **H**eat
Inquiry

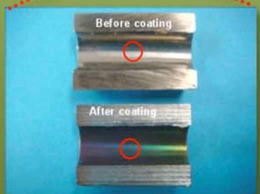
Main loop: 1/2 inch SUS 316L tube
Inventory: 3 L, Max. temp.:600 °C
Flow rate and velocity: 0.5-3 L/min, 0.2-0.5 m/s

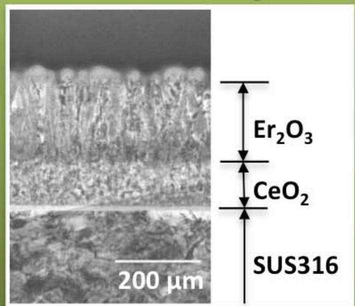
✓ Circulation of heat transfer salt (m.p. ~ 140 °C) has been demonstrated

✓ Now testing the heat control system for FLiNaK (m.p. ~450 °C) circulation

Er_2O_3 Coating by MOCVD for hydrogen permeation reduction

MOCVD: Metal Organic Chemical Vapor Deposition

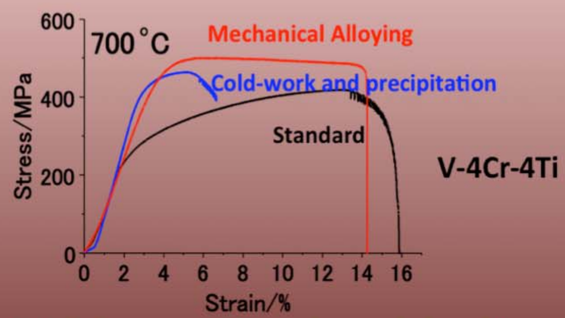


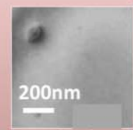


Er_2O_3 coating with a buffer layer makes thicker film formation possible

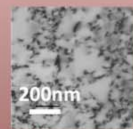
✓ A large area Er_2O_3 coating has been formed by MOCVD

Mechanically-alloyed V-4Cr-4Ti for high-temp. application

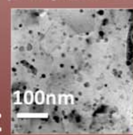




Standard



Cold-work and precipitation

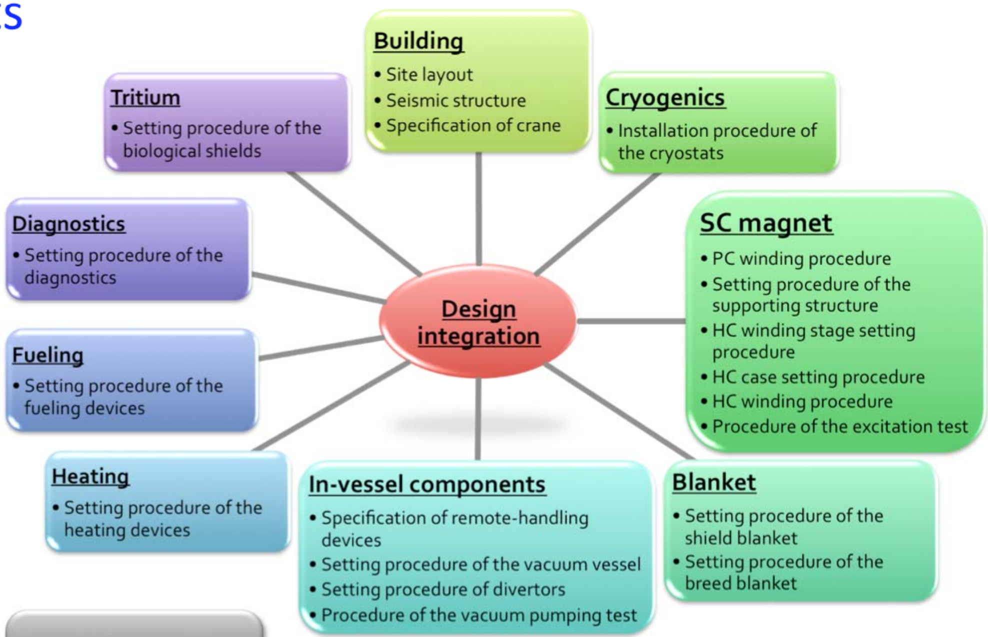


Mechanical alloying

✓ High strength and good ductility at 700 °C has been demonstrated

Future Plans

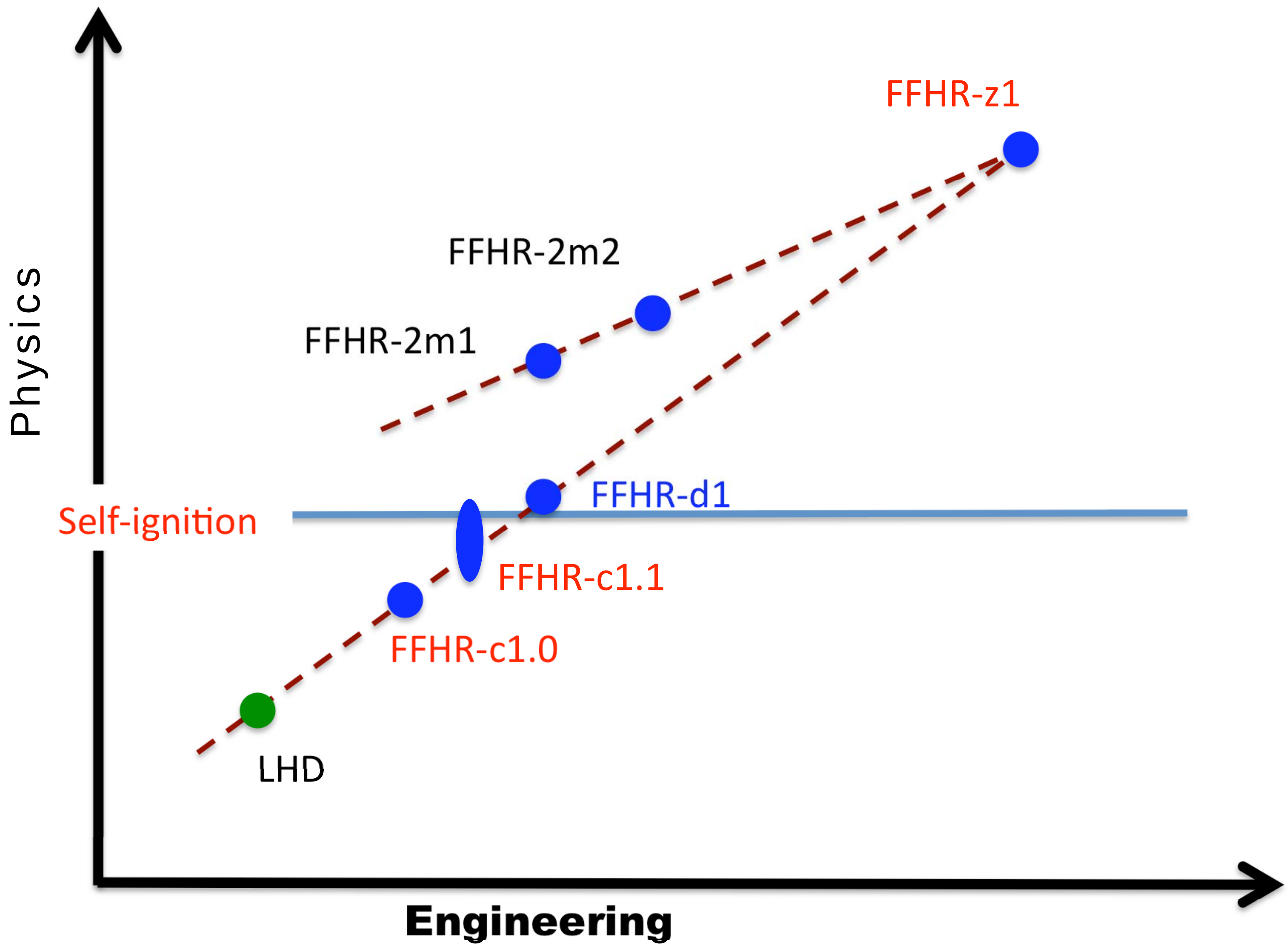
- ➢- 3-dimensional design of in-vessel components consistent with pumping and neutron shielding【Round 2 (contd.)】
- ➢- Start-up scenarios of plasma and blanket
- ➢- Construction schedule (virtual)【Round 3】←large helical coil winding
- ➢- Interim report of FFHR-d1 conceptual design activities
- ➢ Further R&D's on various components



Round 3
"Construction schedule"

Work Planned:

✓ Large SC magnet system of 3D helical continuous winding
✓ Large maintenance ports and replacement scenarios


FFHR-z1
FFHR-2m2
FFHR-2m1
Physics
FFHR-d1
Self-ignition
FFHR-c1.1
FFHR-c1.0
LHD
Engineering

Sagara 11/13

2 types of FFHR-c1 (conservative (c1.0) or challenging (c1.1))

Device Name	LHD	FFHR-c1.0	FFHR-c1.1	FFHR-d1
Target	Experiment	Q ~ 7	Self-ignition	Self-ignition
R_c helical coil major radius	3.9 m	13.0 m (10/3 times LHD)	←	15.6 m (4 times LHD)
V_p plasma volume	~30 m^3	~1,000 m^3 **(similar to ITER)**	←	~2,000 m^3
B_c magnetic field strength at the helical coil center	2.5 T	4.0 T **(design value of LHD)** **NbTiTa (He II) / HTS**	5.3 T **(similar to ITER)** **Nb3Sn / Nb3Al / HTS**	4.7 T Nb3Sn / Nb3Al / HTS
W_{mag} stored magnetic energy	1.6 GJ (at 4 T)	68 GJ	113 GJ	160 GJ
P_{aux} auxiliary heating power	25 MW (short pulse)	140 MW - CW (for sustainment)	50 MW - 1 hour (for start-up)	50 MW - 1 hour (for start-up)
P_{fusion} fusion power	–	~1 GW **(Q > 7)**	~2 GW **(Q = ∞)**	~3 GW (Q = ∞)
$\tau_{duration}$ duration time of a shot	~1 hour	~1 year	~5 month	~1 year
Φ_n dpa per shot	–	~8 dpa **(0.8 MW/m^2 x 1 year)**	~7 dpa (10 dpa at peak) **(1.7 MW/m^2 x 5 month)**	~15 dpa (1.5 MW/m^2 x 1 year)
Issues	*DD exp.*	***large HC winding*** *R&D of (NbTiTa & He II cooling) or HTS*	***large HC winding*** ***react & wind*** ***nuclear heating on SC*** ***lifetime of SC/insulator***	***large HC winding*** ***react & wind*** ***nuclear heating on SC*** ***lifetime of SC/insulator***

Summary

- **Conceptual designs of FFHR-d1 and -c1 are in progress**; major parameters are determined, features for the next step fusion facility are discussed
- The helical system **enables the long life-time blanket with a neutron wall loading of <2 MW/m^2**, which ensures long MTBF (mean time between failures) and high safety
- **Large ports** are available **for remote maintenance with short MTTR** (mean time to repair) expected
- **Divertor targets are effectively shielded from fast neutrons**, which ensures high reliability and **wide material selection for cooling pipes** (*e.g. Cu-alloy*)
- R&D's on **HTS magnet, liquid blanket, V-alloy** are in progress

Thank you