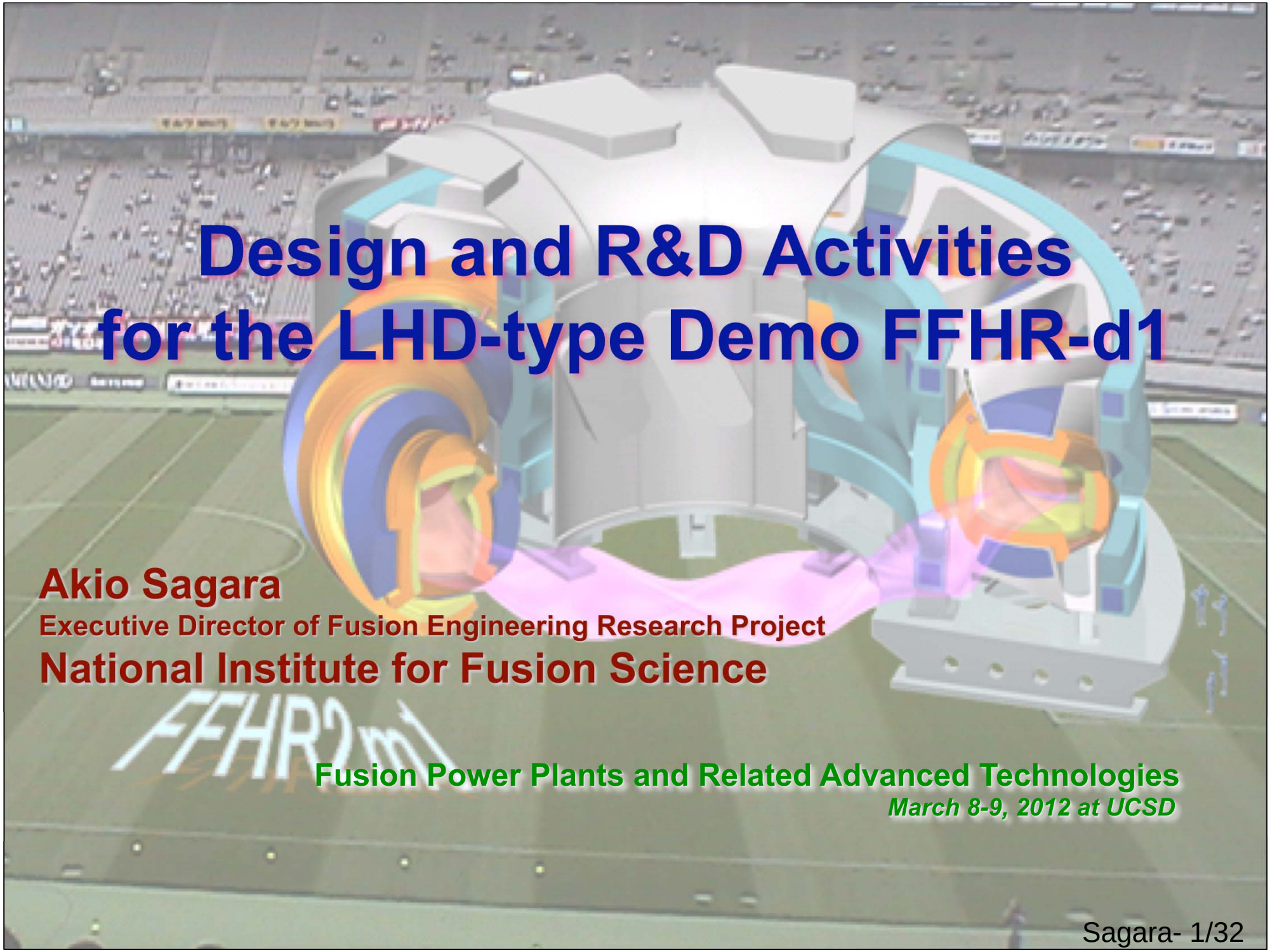


Design and R&D Activities for the LHD-type Demo FFHR-d1

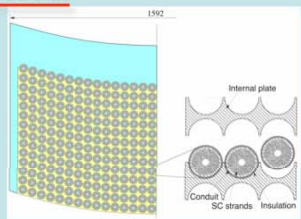
Akio Sagara
Executive Director of Fusion Engineering Research Project
National Institute for Fusion Science

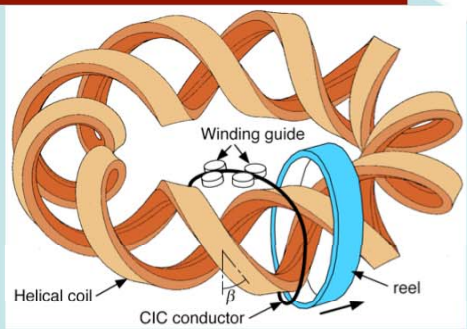
Fusion Power Plants and Related Advanced Technologies
March 8-9, 2012 at UCSD

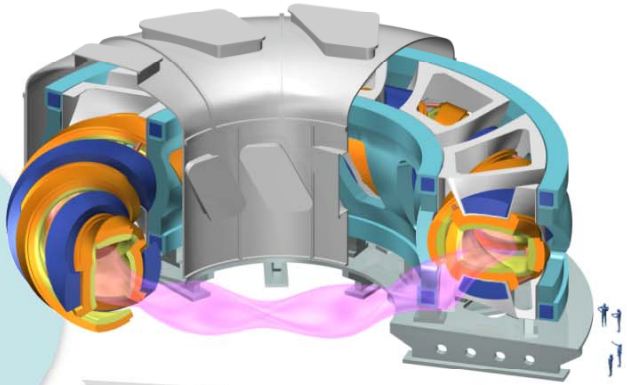
From FY2010, NIFS Fusion Eng. Research Project
has started towards steady-state helical DEMO reactor

based on the 1st mid-term results (2004~2009)
Key subjects are ------

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

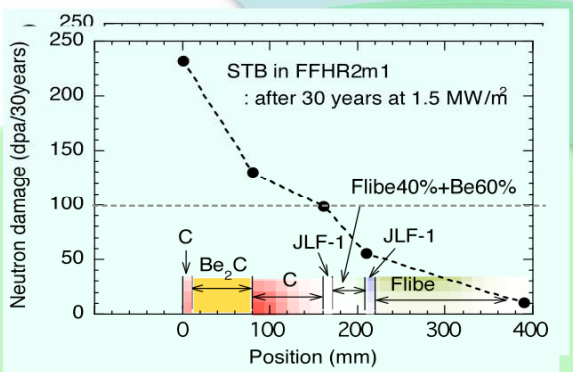




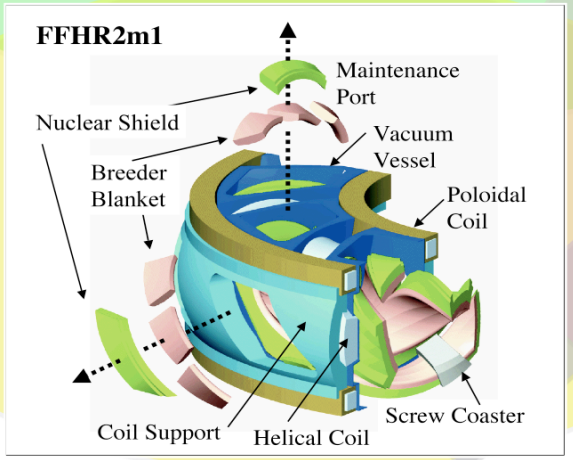


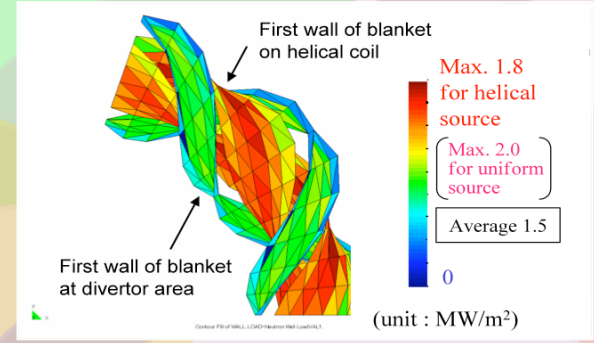
Steady-state helical DEMO reactor FFHR-d1

Step by step advancement of reactor design



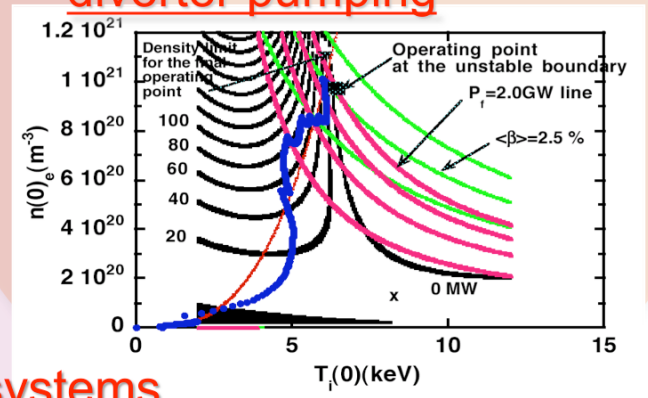
Long life blanket with inherent safety and wide maintenance ports





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

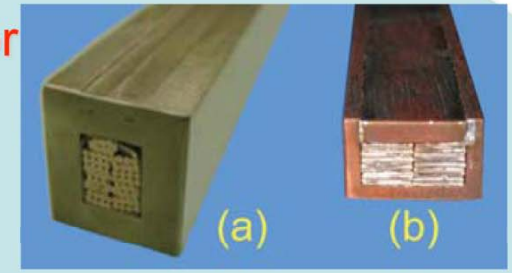
High density self-ignition, heating, fueling and diverter pumping

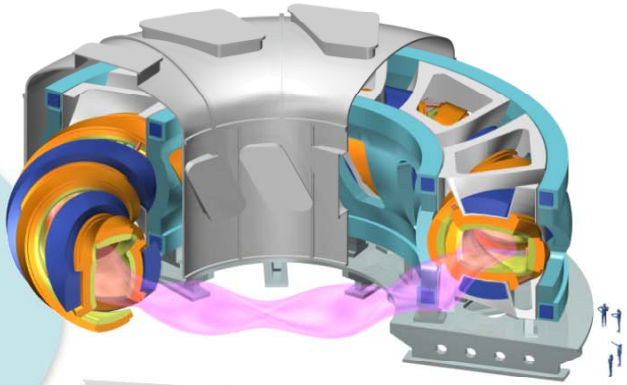


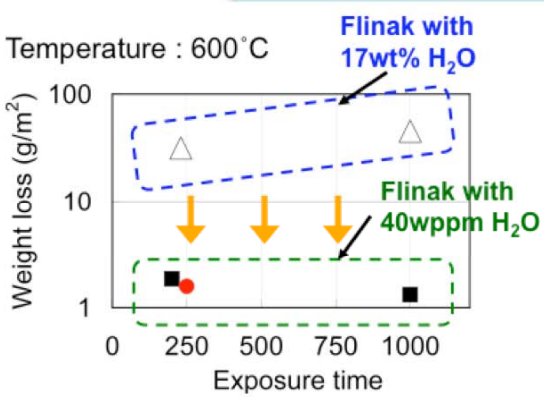
Fusion Eng. Research Project ***includes R& D activities***
towards steady-state helical DEMO reactor

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

Developments of
(a) Nb_3Sn or Nb_3Al superconductor with Al-alloy jacket,
(b) 10 kA-class high-temp. superconductor HTS.





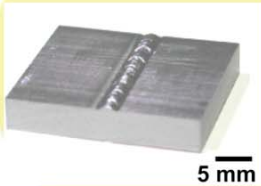


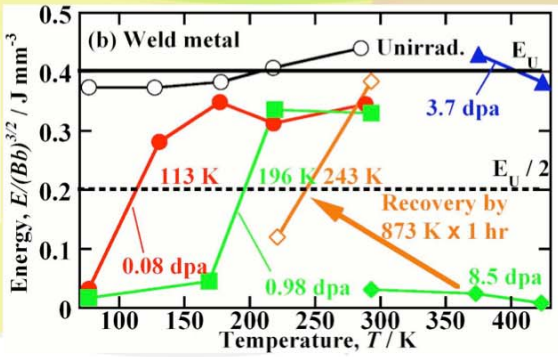
Reduction of corrosion and T permeation barrier for liquid blanket (with U. of Tokyo.)

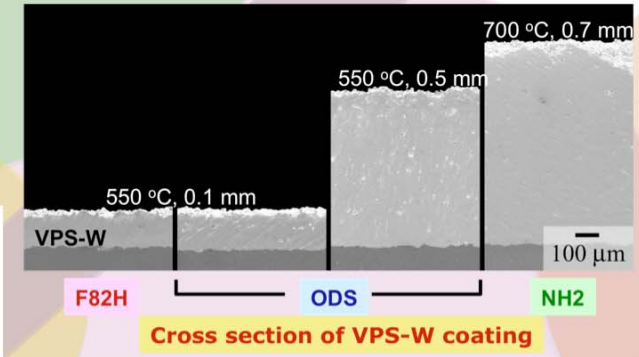
Basements for full-scale or full-condition testing

Steady-state helical DEMO reactor FFHR-d1

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







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

High-sensitive tritium Monitor (with Nagoya Univ.)

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

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

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

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	Kondo, Muroga
	First wall	Hirooka, Ashikawa
In-vessel componet development, Design, Maintenance Tamura	Vacuum vessel	Tamura, Masuzaki
	Divertor	Masuzaki, Tokitani
	Remote maintenance	Nishimura, Ashikawa

- ・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

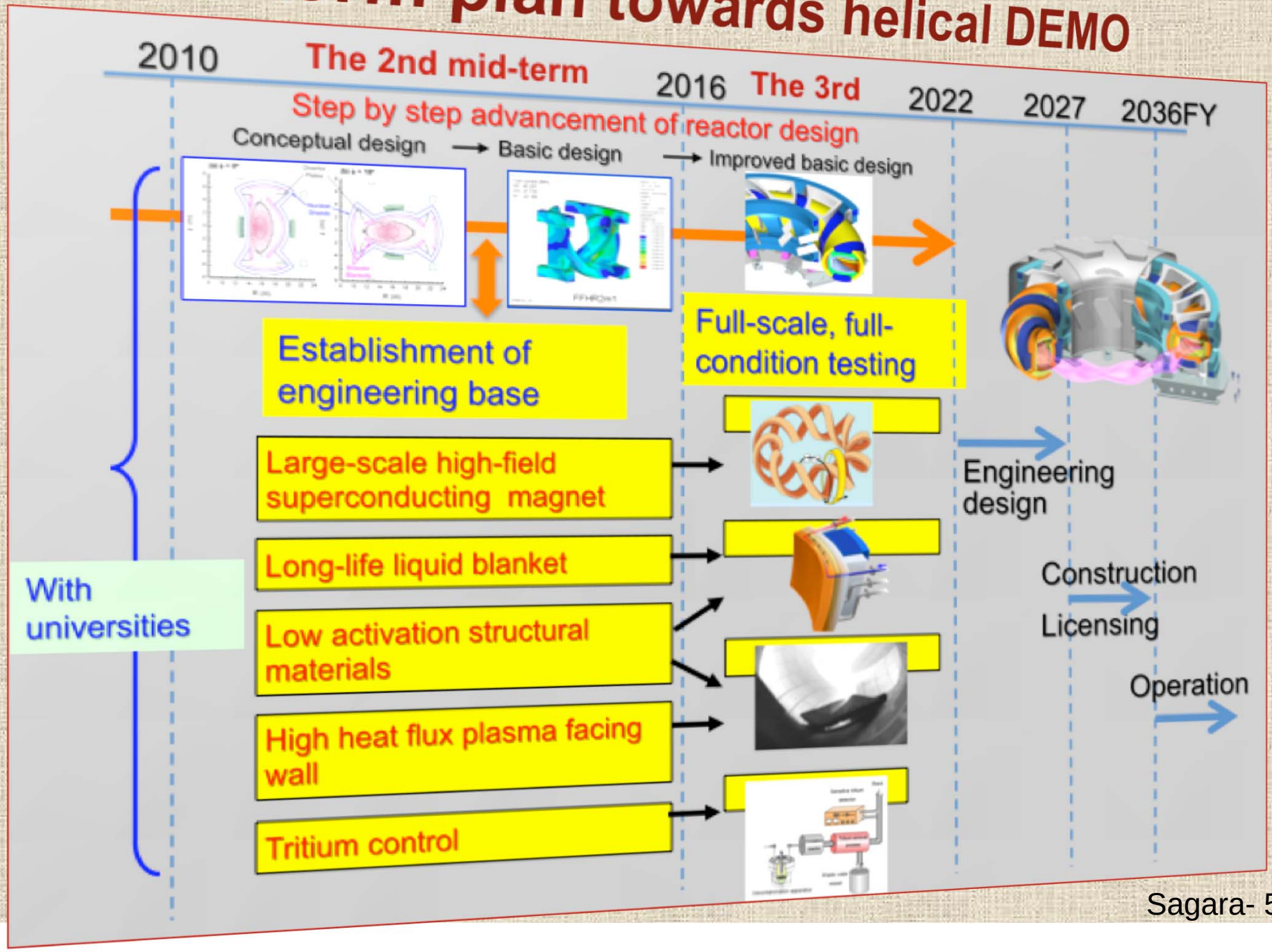
- ・ Prototype testing of 3D divertor
- ・Hydrogen retention in LHD irradiation

- ・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 managiment	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	Yamanishi
	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

Rev. 2010.5.17

Mid-term plan towards helical DEMO

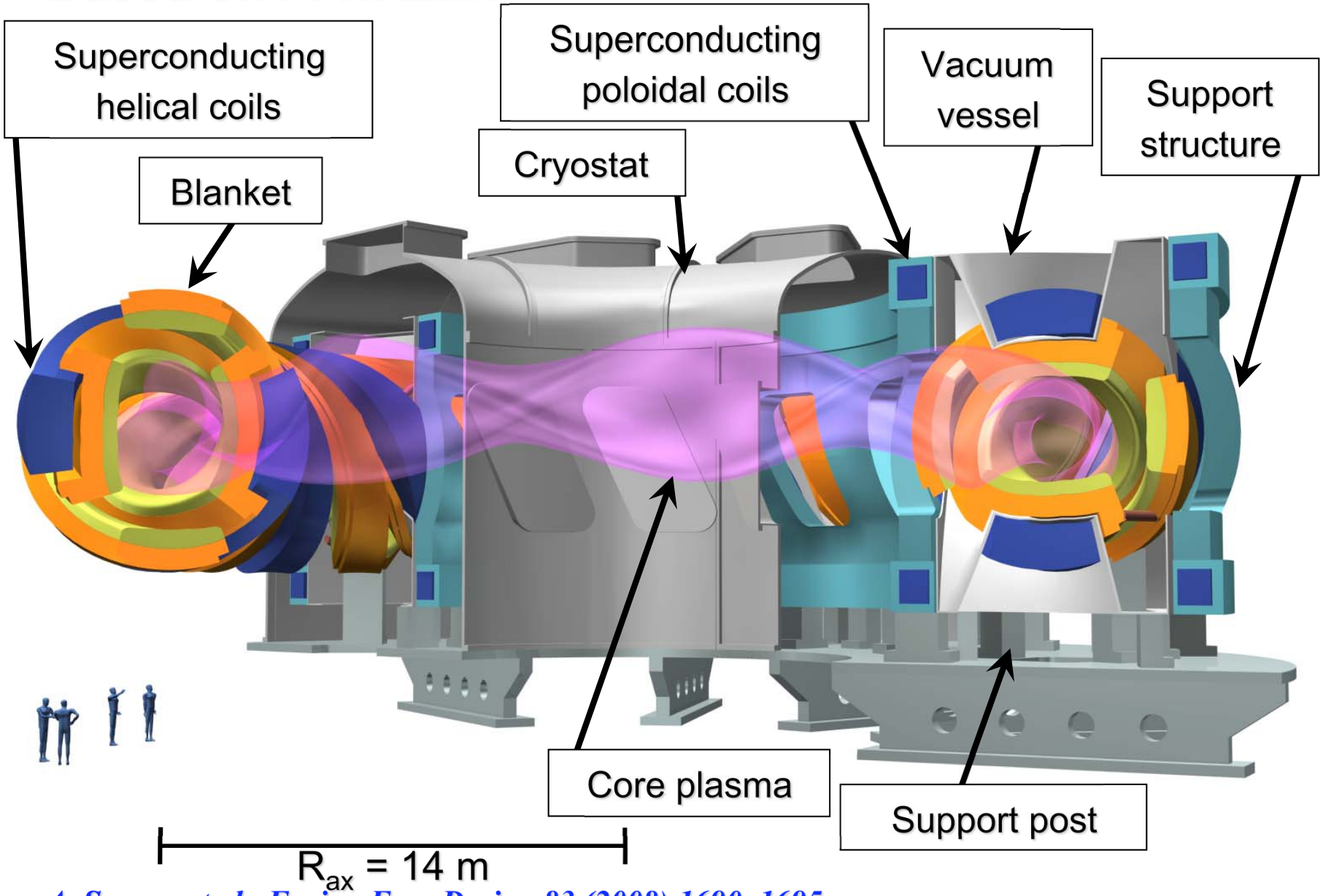


DEMO

FFHR-d1

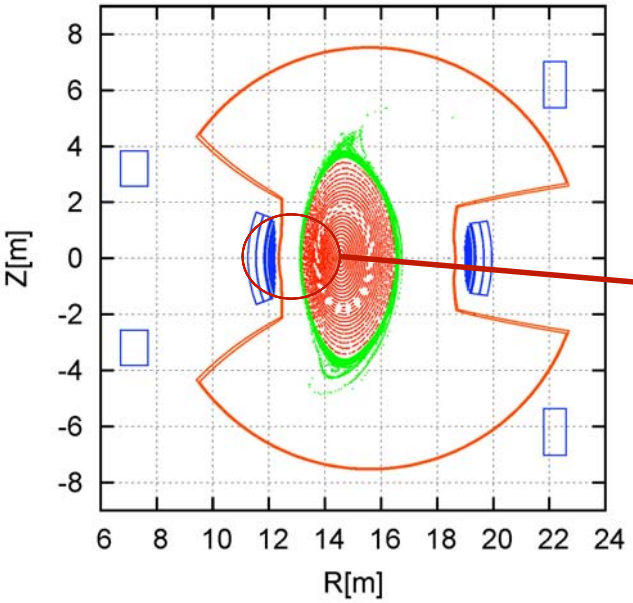
Re-design, enhancing robustness, construction and safety

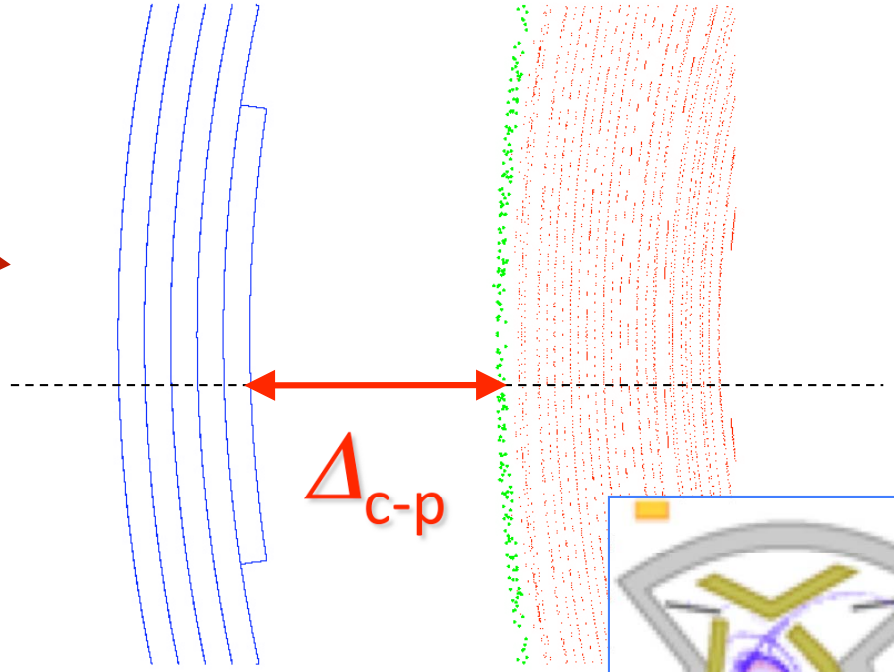
Based on FFHR2m1

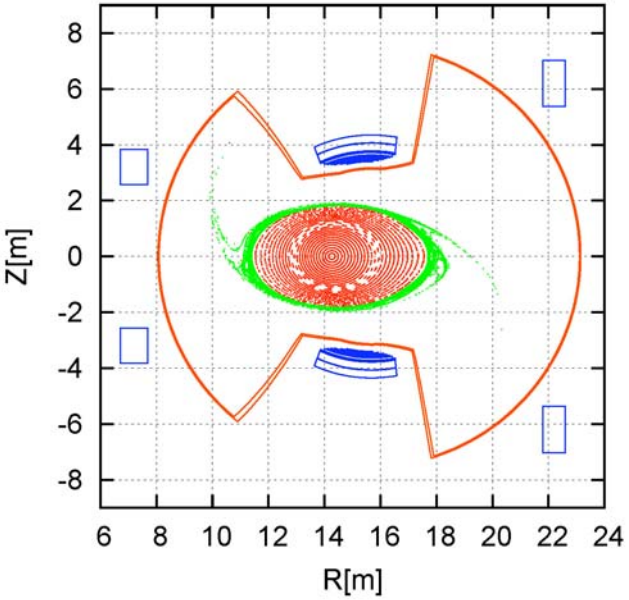


A. Sagara et al., Fusion Eng. Design 83 (2008) 1690–1695

Radial build between coil and plasma

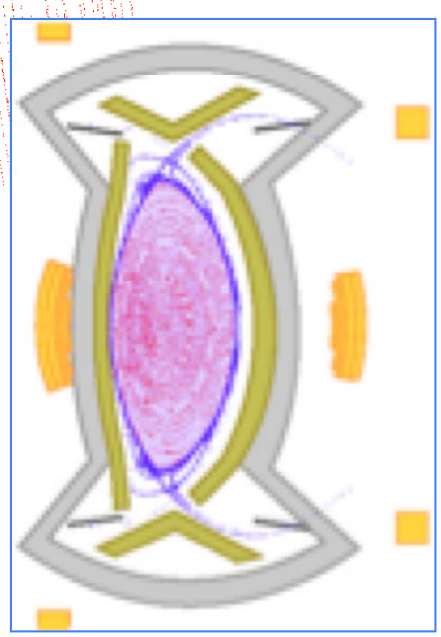




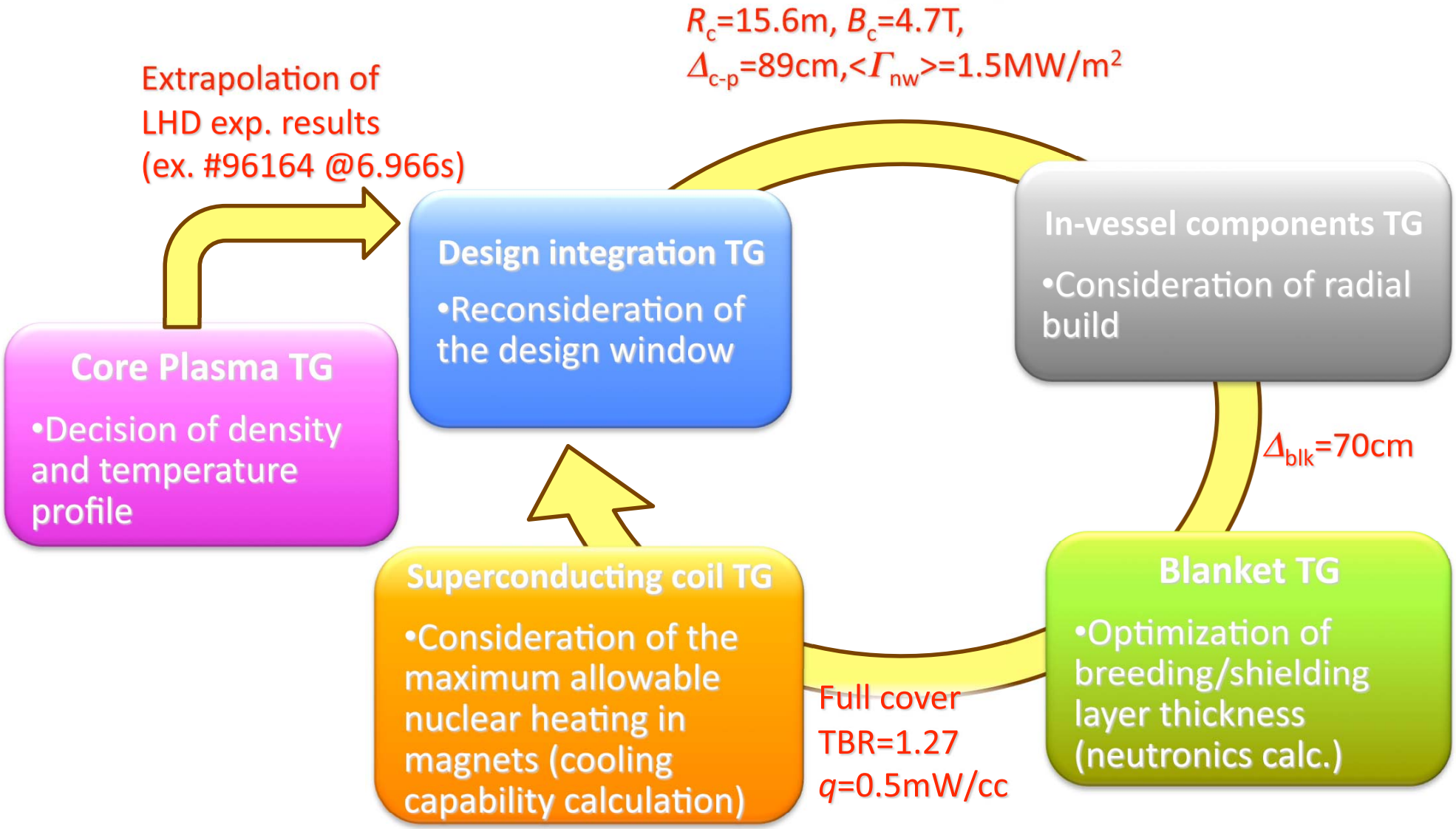


This space includes:

- helical coil bottom plate
- thermal insulation gap
- vacuum vessel
- blanket system
- plasma-FW clearance



1st cycle
~Decision of main design parameters~

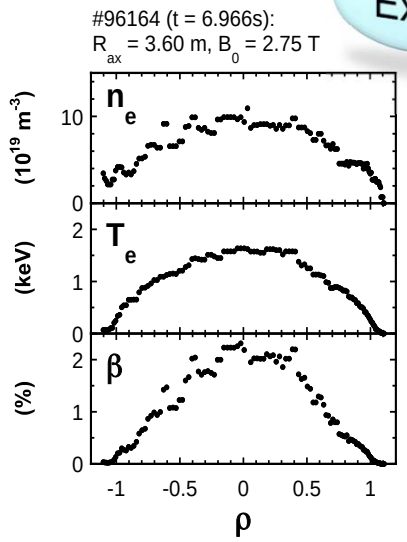


Direct Profile Extrapolation from LHD data

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

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

Exp



DPE

$T_{reactor}(\rho) = f_T\, T_{exp}(\rho)$

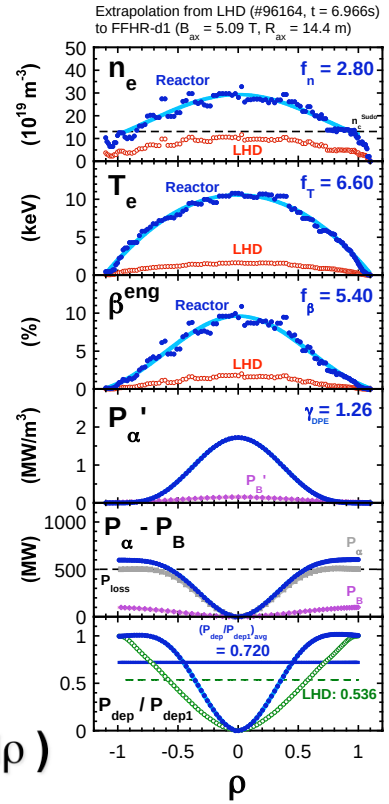
$n_{reactor}(\rho) = f_n\, n_{exp}(\rho)$

$a_{reactor} = f_a\, a_{exp}$

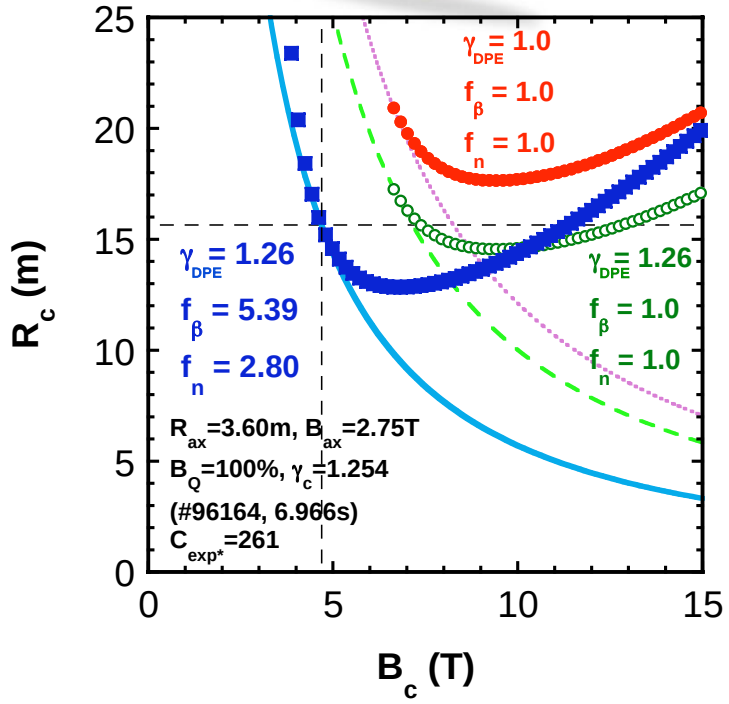
$R_{reactor} = f_R\, R_{exp}$

$B_{reactor} = f_B\, B_{exp}$

$P_{reactor} = f_P\, P_{exp}$



Reactor



$$f_T = f_\beta\, f_n^{-1}\, f_B^{2}$$

$$f_P = \gamma_{DPE}^{-2.5}\, f_\beta^{2.5}\, f_B^{3}\, f_n^{-1.5}$$ ← Gyro-Bohm

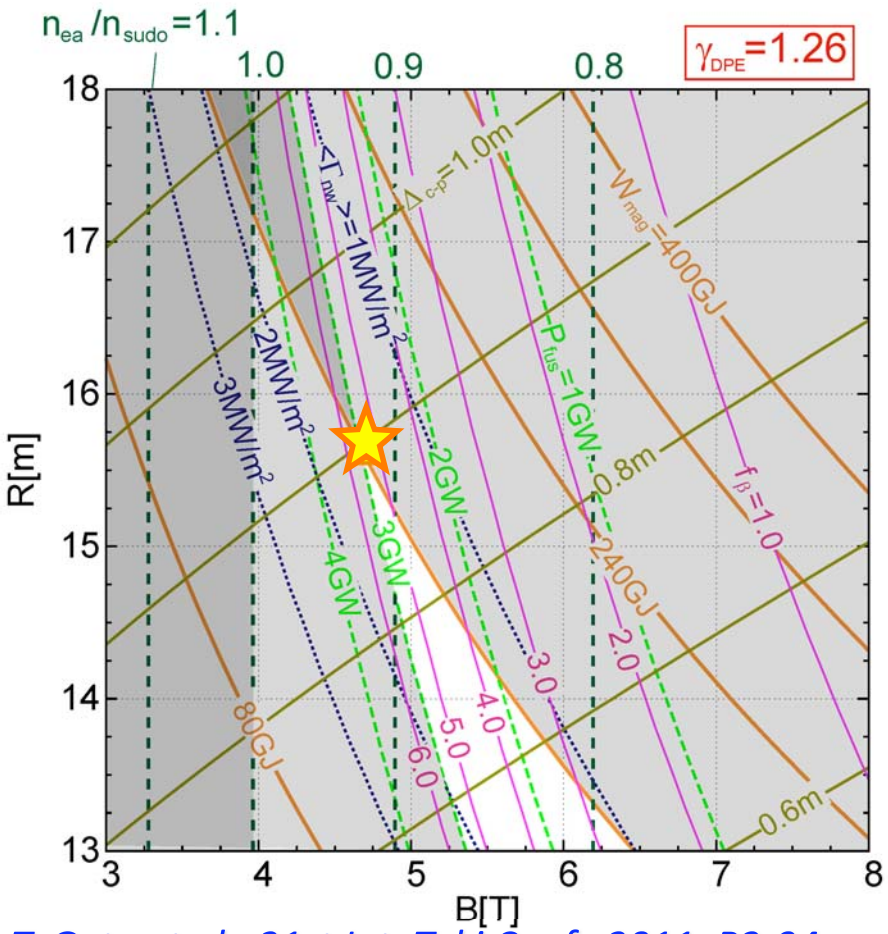
$$f_R^{3} = f_P\, P_{exp} \Big/ \left(f_n^{2} \int_0^1 (P_\alpha' - P_B')\,(dV/d\rho)_{exp}\, d\rho\right)$$

Radial profiles in experiment are directly extrapolated to reactor

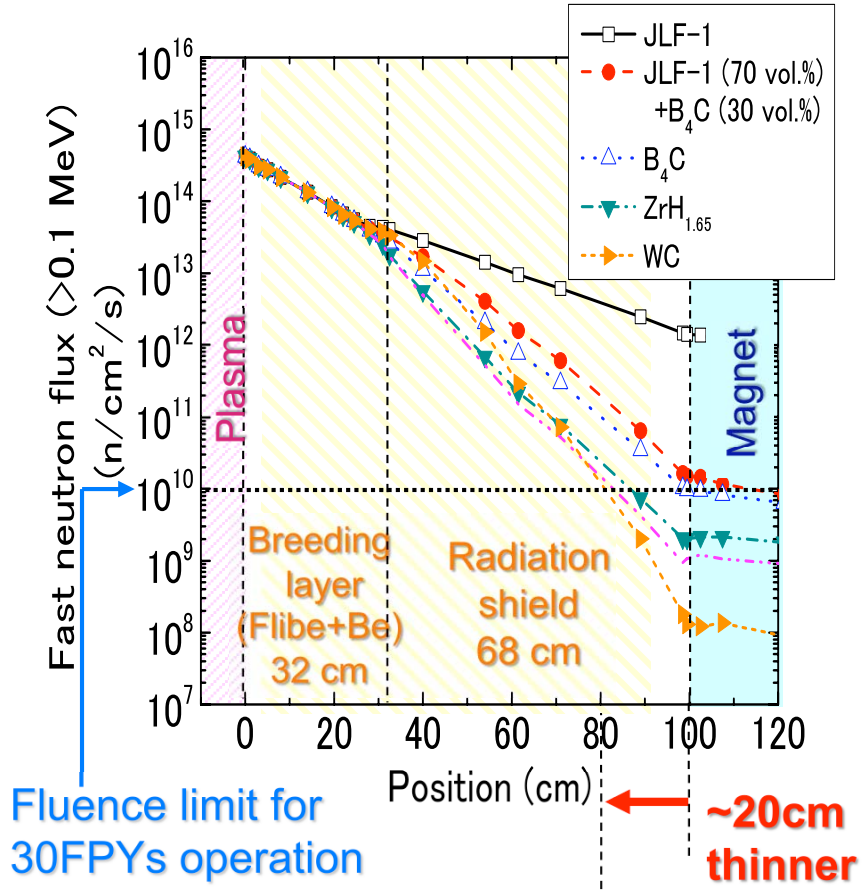
The reactor size is determined as a function of magnetic field strength

Design window by the system code

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**.



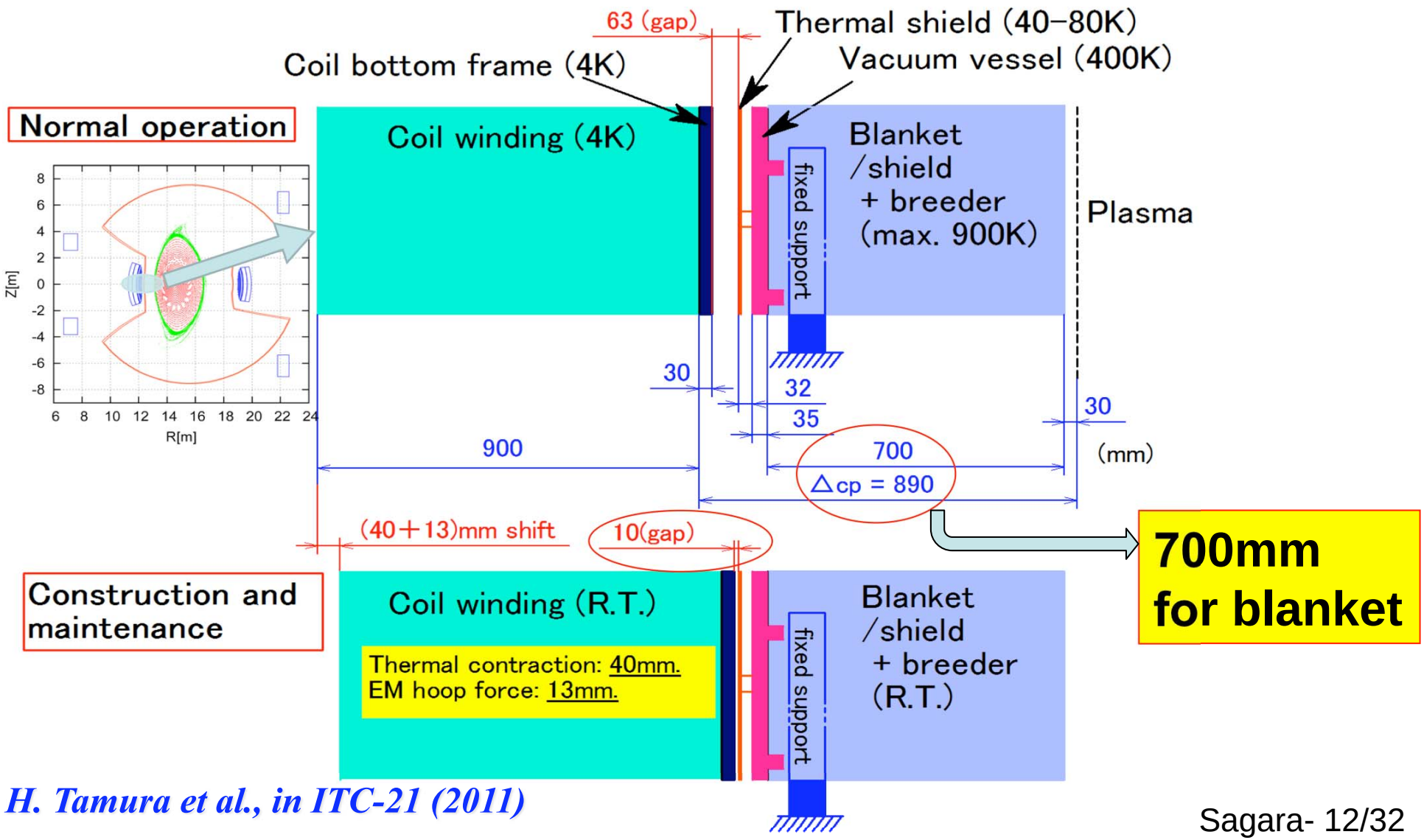
T. Goto et al., 21st Int. Toki Conf., 2011, P2-94, and submitted to PFR.



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

Design of the inboard space Δ_{cp} (this case 890 mm)

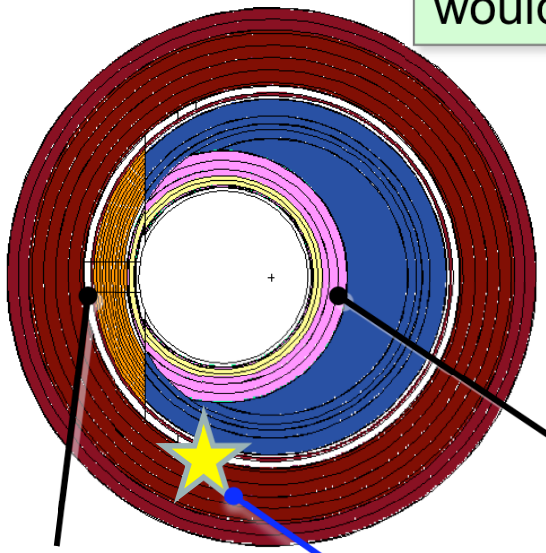
to ensure the gap at R.T. (this case 10 mm)



H. Tamura et al., in ITC-21 (2011)

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

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



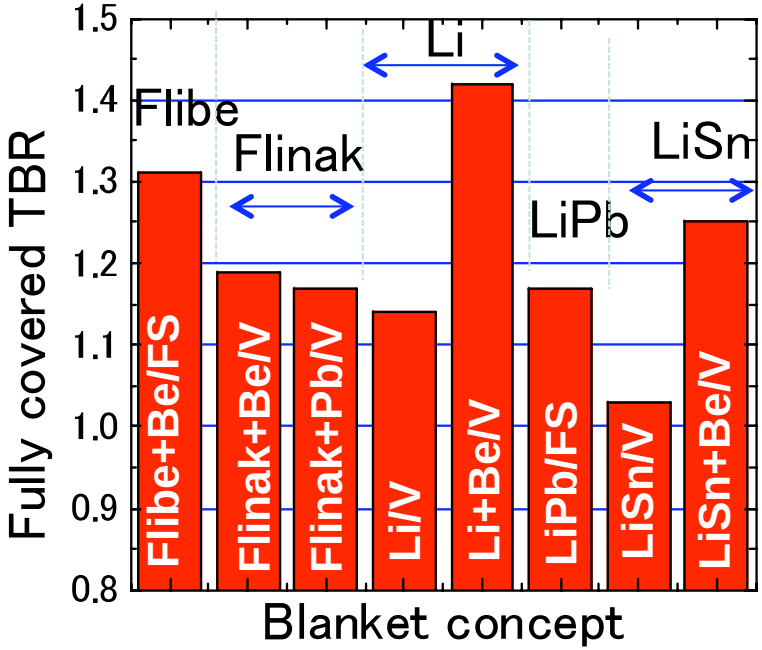
Neutron wall loading: 1.5 MW/m²

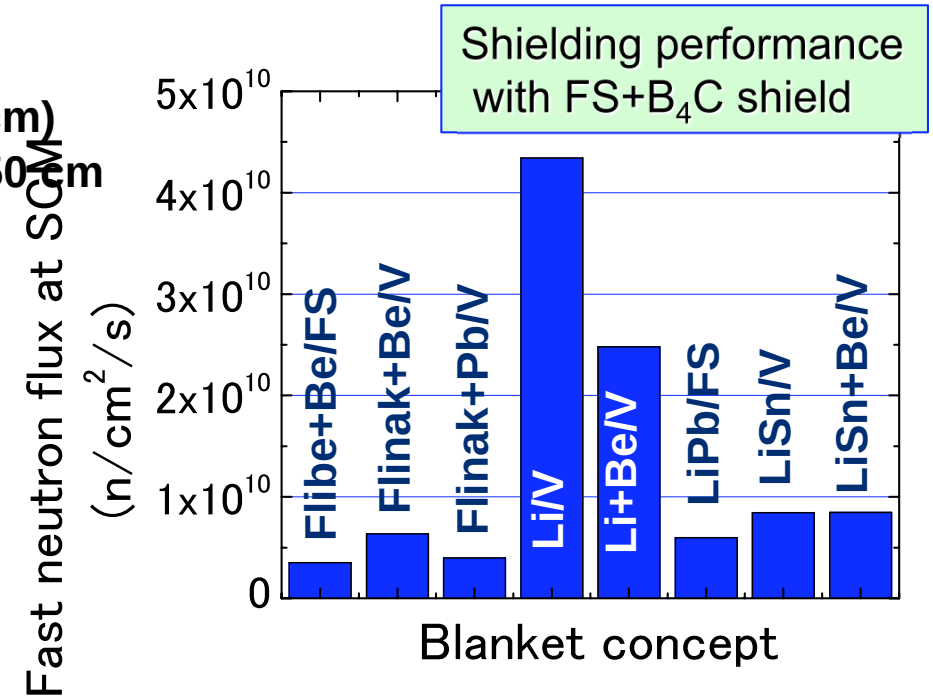
[Outboard]
Breeder: 60cm
(Multiplier 15cm)
FS+B_4C Shield: >50cm

[Inboard]
Breeder:15 cm
WC Shield: 55 cm

(Nuclear heating: ～0.5 mW/cm³,
Fast neutron flux: ～3x10¹⁰ n/s/cm² at SCM)

Simple torus model for neutronics analysis

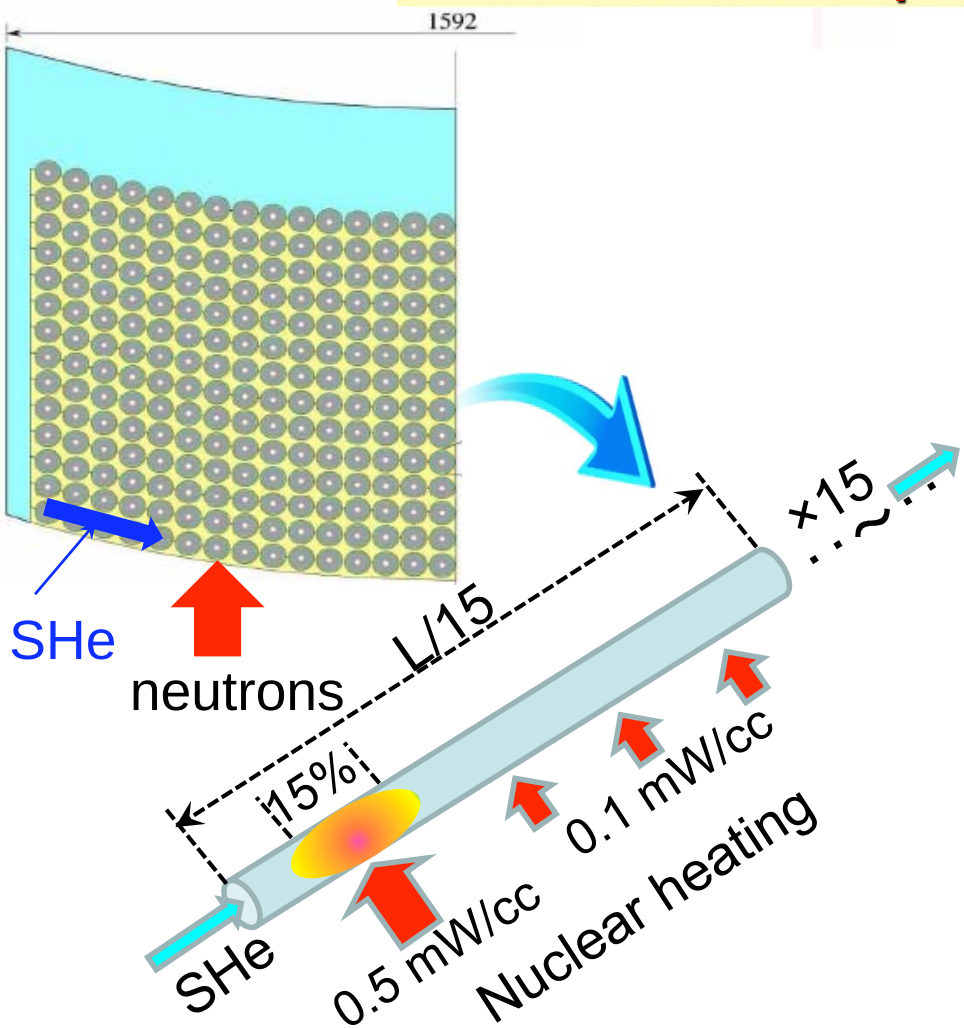


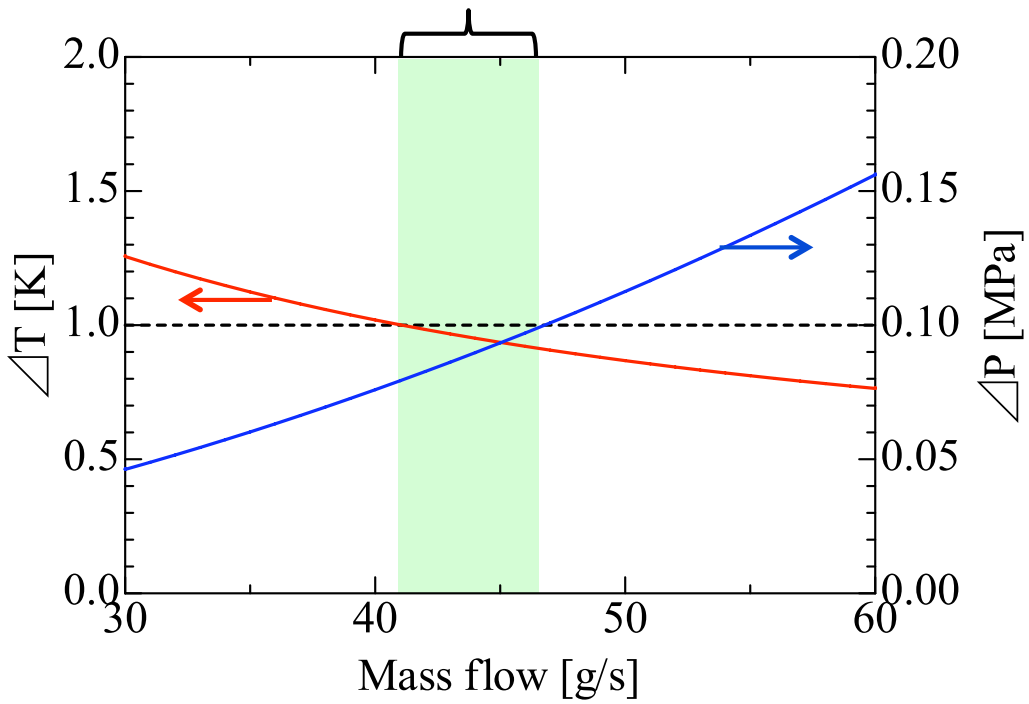


T. Tanaka et al., submitted to IAEA-FEC24

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
ΔT<1K, ΔP<0.1MPa

S. Hamaguchi et al., in ITC-21 (2011)

Design parameters

			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}	MW		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
Total capital cost		G$(2003)		4.6	5.6	7.0		
COE		mill/kWh		155	106	93		

2nd cycle
~Design of in-vessel components~

Design integration TG

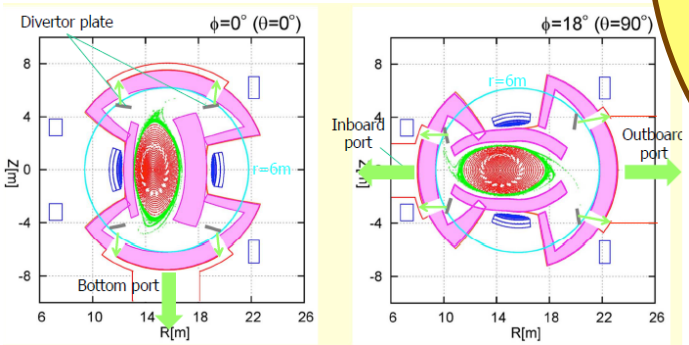
- Consistency check (divertor trace calculation, basic shape determination

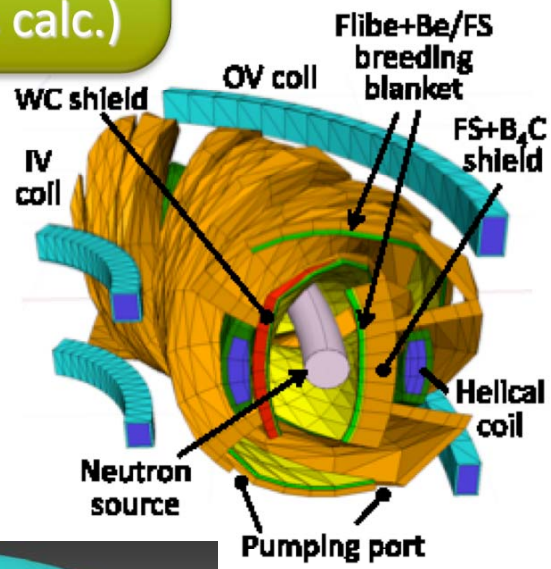
Blanket TG

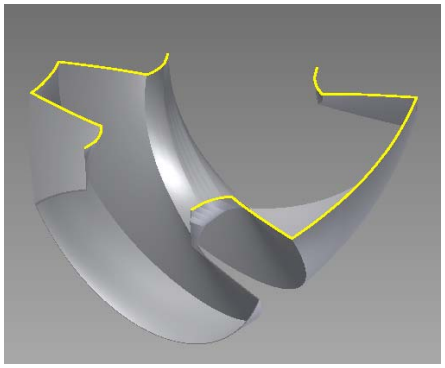
- Optimization of 3D blanket shaping (3D neutronics calc.)

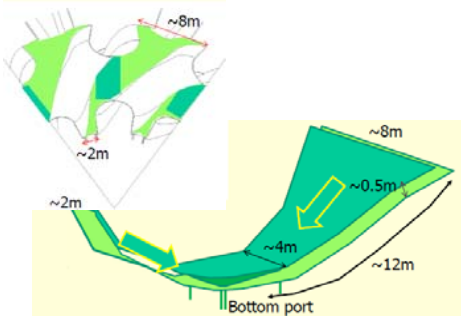
Divertor Sub TG (In-vessel components TG)

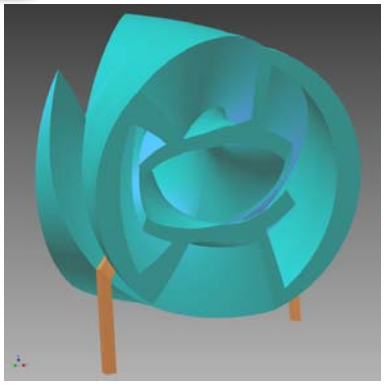
- Divertor/port design
- Pumping performance calc.











[1] Radiation shielding

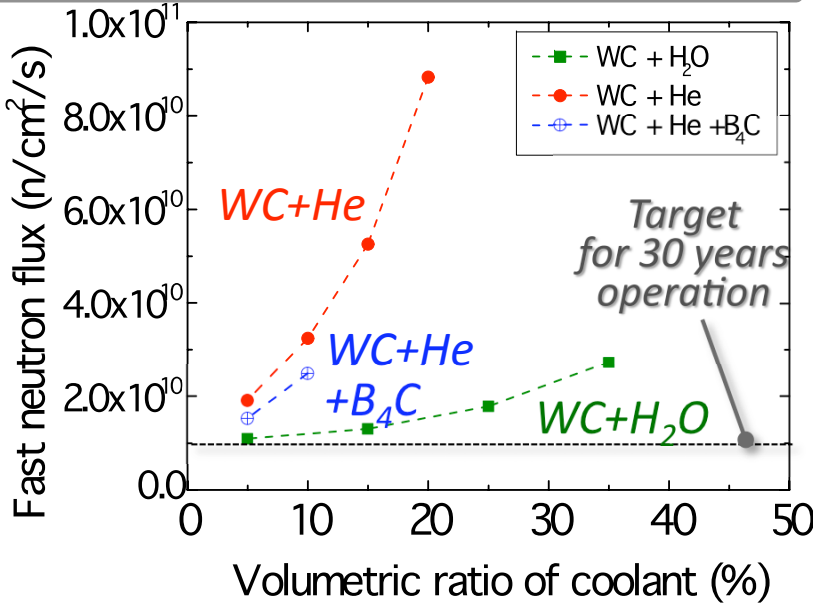
T. Tanaka, ISFNT-10

Evaluated shielding performance

(a) WC shield **(b) WC +B_4C shield**

Configuration of inboard radiation shield

(SCM: Superconducting Magnet)*



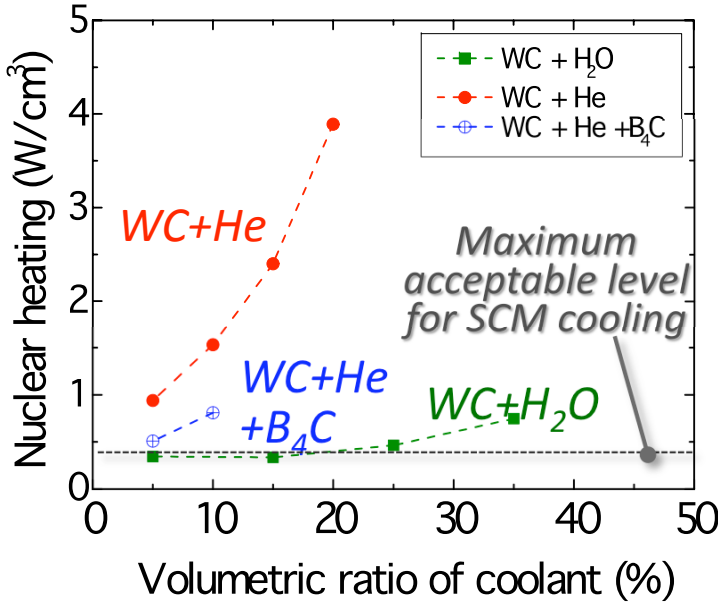
Fast neutron flux (>0.1 MeV) at SCM

- **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)***

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

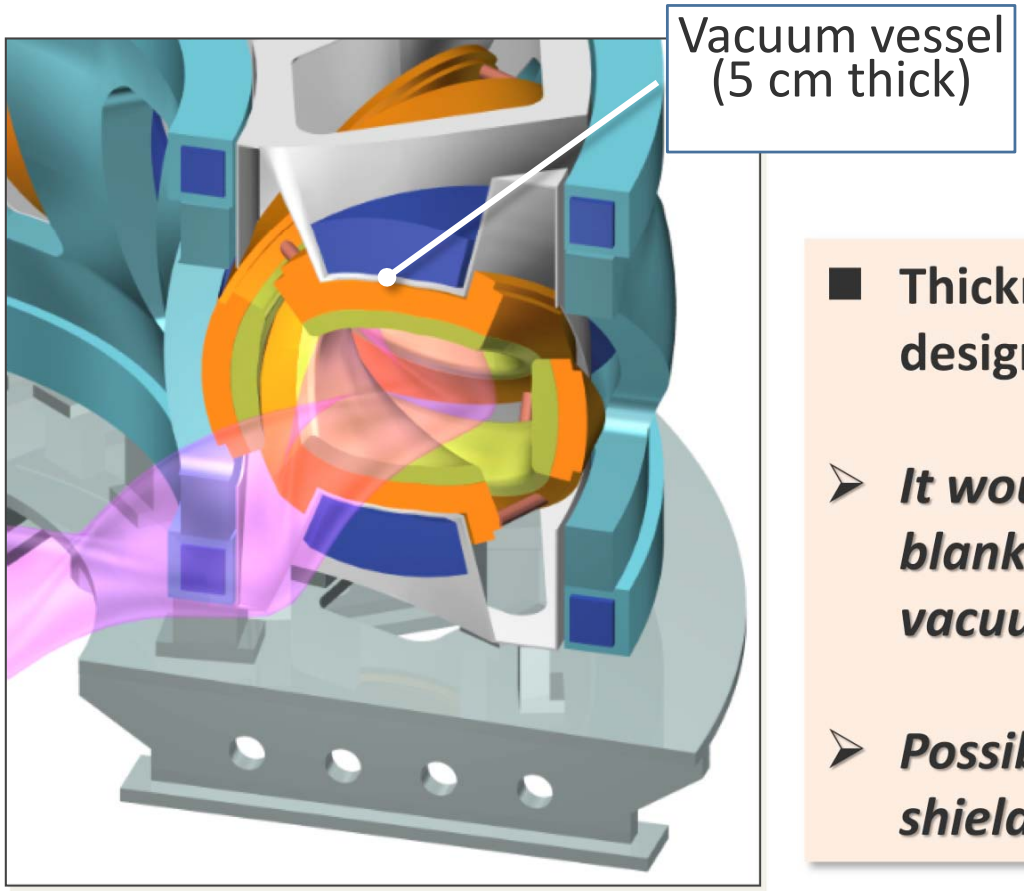


Nuclear heating at SCM (Nb_3Al)

[2] Support of helical blanket

T. Tanaka, ISFNT-10

Thickness of vacuum vessel in FFHR



Cross section of FFHR2-m1

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

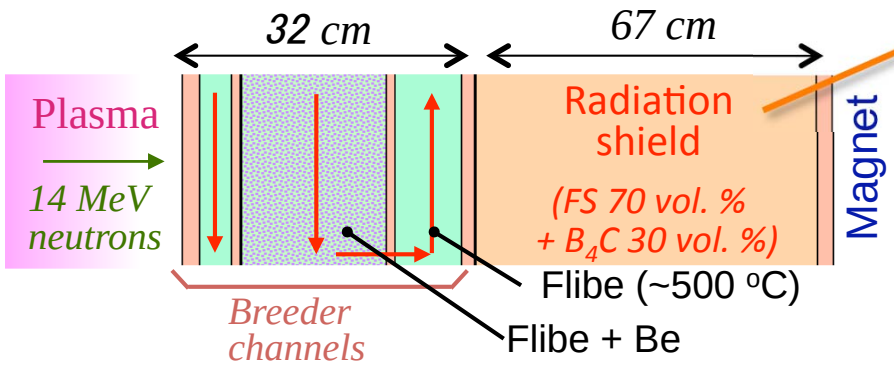
➢ ***It would be difficult to support all the blanket component by the thin vacuum vessel.***

➢ ***Possible to support by radiation shielding layers?***

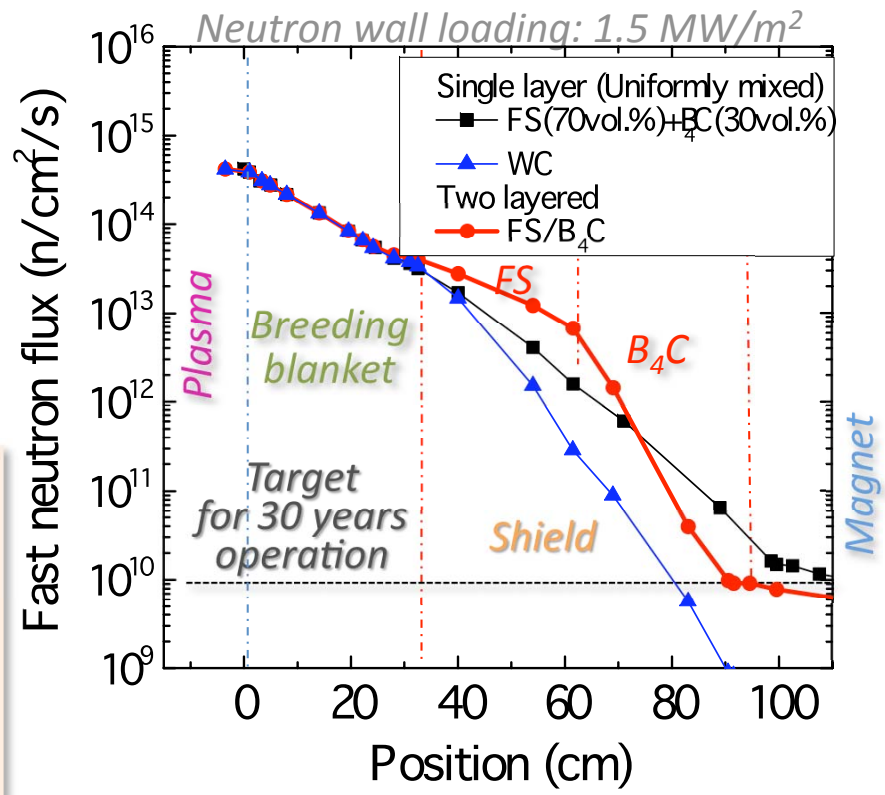
[2] Support of helical blanket

T. Tanaka, ISFNT-10

Two-layered radiation shield



Structure of Flibe+Be/Ferric Steel blanket



Attenuation of fast neutron flux by two-layered shield

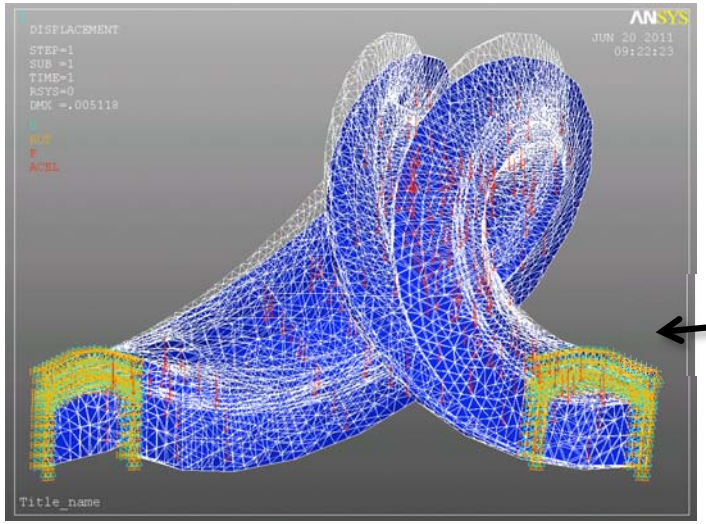
- **Main materials for outboard shield: Ferric steel and B_4C**
- **A two-layered shield has a similar performance as an uniformly mixed shield.**

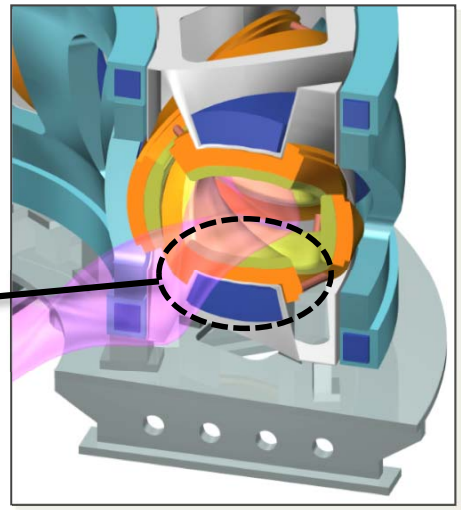
➢ ***A ferric steel layer could be used for a main support structure of blanket and shield.***

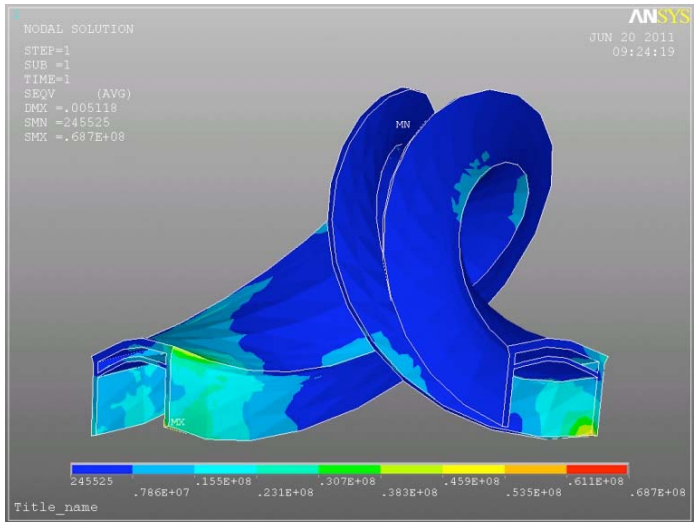
[2] Support of helical blanket

T. Tanaka, ISFNT-10

Mechanical stress on helical support layer







Calculation of mechanical stress in radiation shield 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)

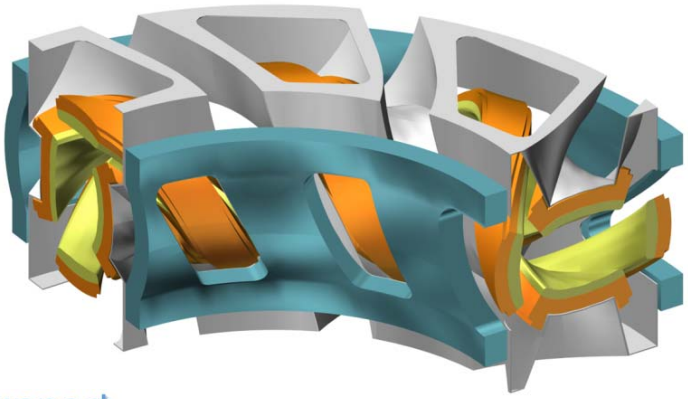
➢ ***Estimated maximum stress of ~30 MPa would be acceptable.***

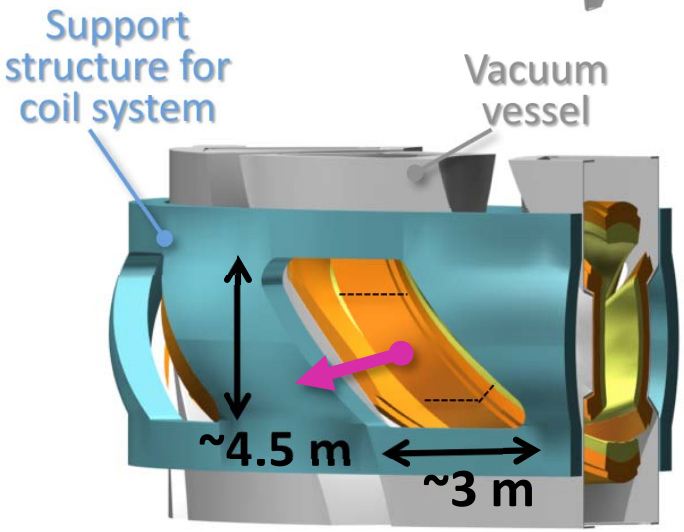
➢ ***More detailed investigation and optimization of the structure will be performed.***

[3] Cooling of helical blanket

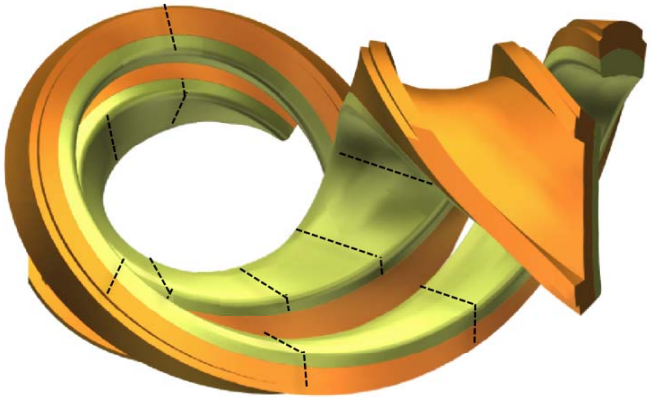
T. Tanaka, ISFNT-10

Size of helical blanket modules





Maintenance port openings in FFHR-2m1*



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.***

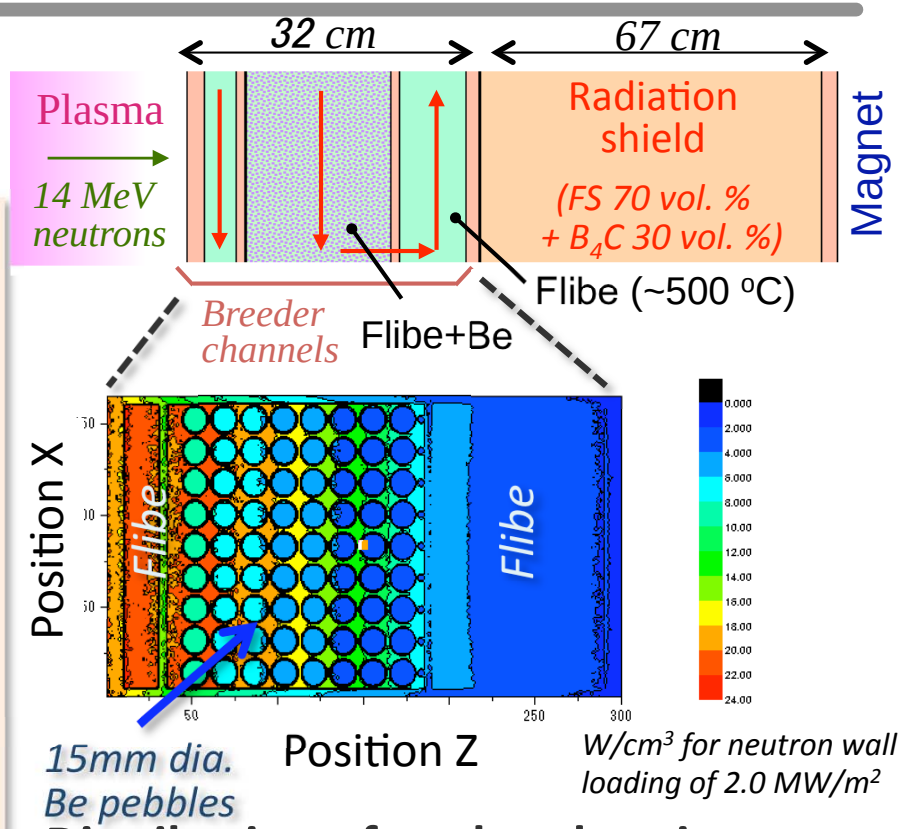
*A. Sagara et al., Fusion Engineering and Design 49-50 (2000) 661-666.
**S. Imagawa et al., Annual Report of NIFS, April 2005-March 2006, 274.

[3] Cooling of helical blanket

T. Tanaka, ISFNT-10

Heat removal in Flibe blanket

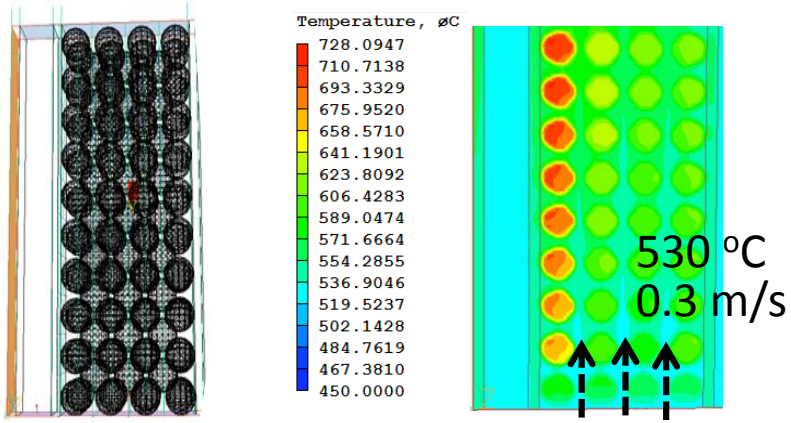
- **Heat removal of a Be pebble layer has been investigated in the present study.**
- **Temperature of ~0.1 m/s Flibe flow in a Be pebble layer increases from 450 °C to 550 °C during go through ~4.5 m long channel.**
- **Nuclear heating in Be is lower than that in a Flibe coolant.**
- ***In a ~550 °C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



Distribution of nuclear heating in Flibe+Be/Ferritic Steel blanket

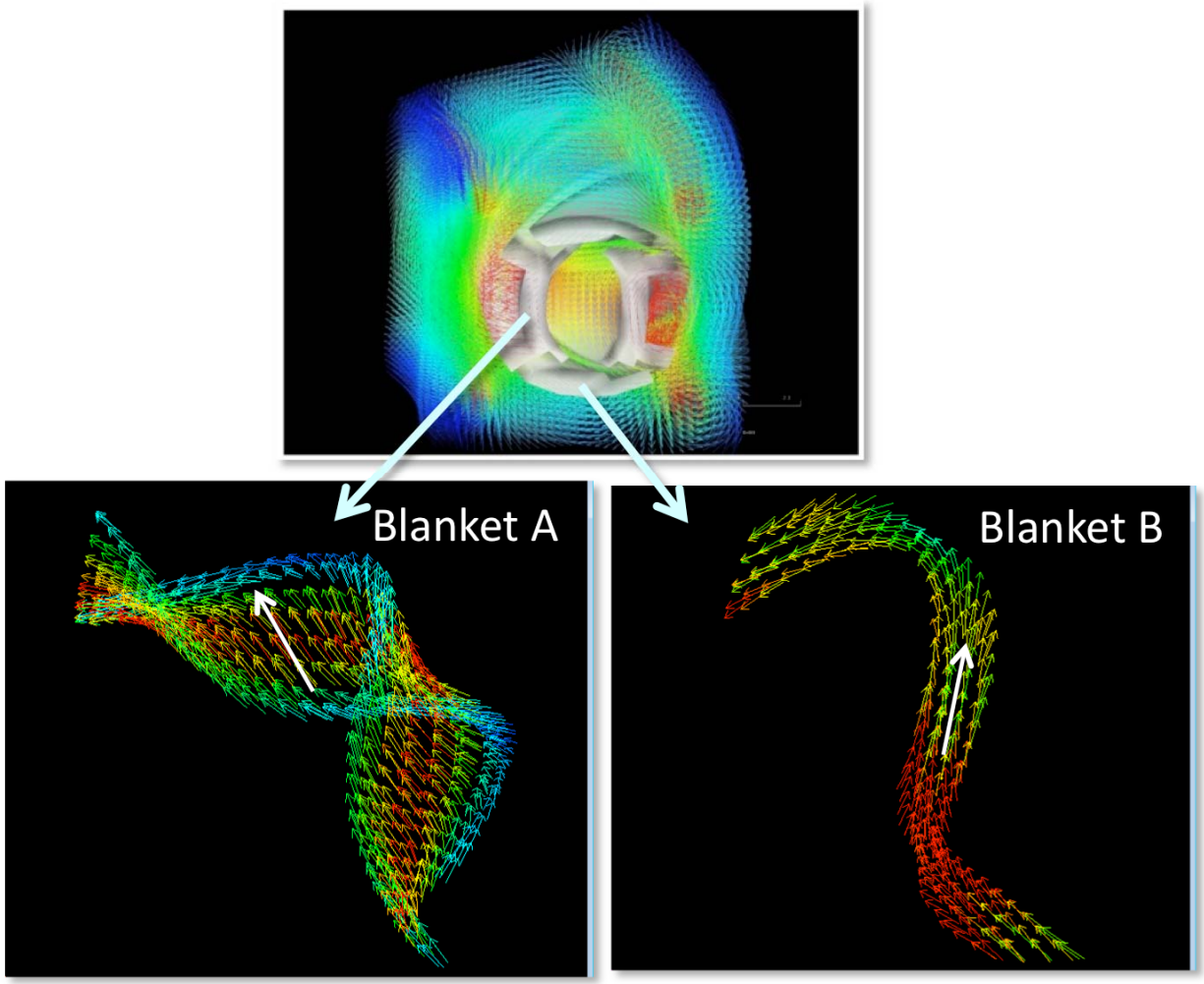
Geometry and result of heat transfer calculation for Flibe+Be pebble layer ➔

(for neutron wall loading of 1.5 MW/m²)



[3] Cooling of helical blanket

Magnetic field in helical blanket



Magnetic field distribution in helical breeding blanket

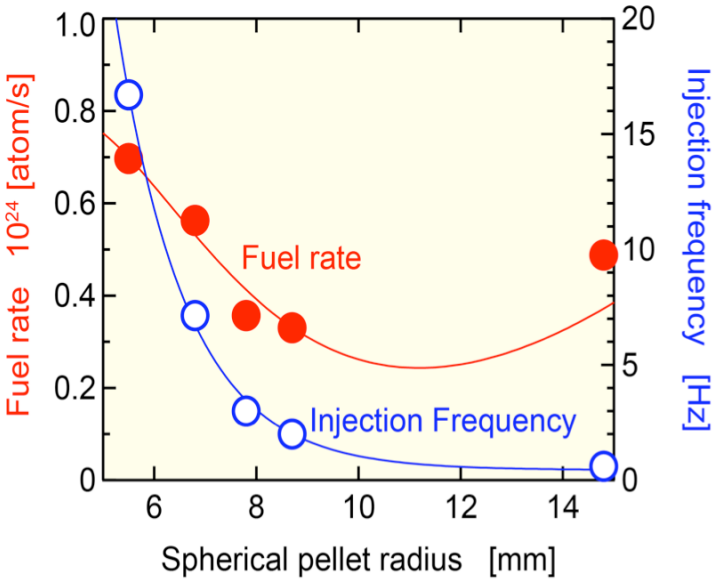
- **Directions of magnetic fields are different in the blanket A and B.**
- **Optimum flow directions for liquid coolants will be decided by the magnetic field distribution.**
  - ***For Li coolant:*** ***Parallel to magnetic fields to reduce MHD pressure drop.***
  - ***For Flibe coolant:*** ***Magnetic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

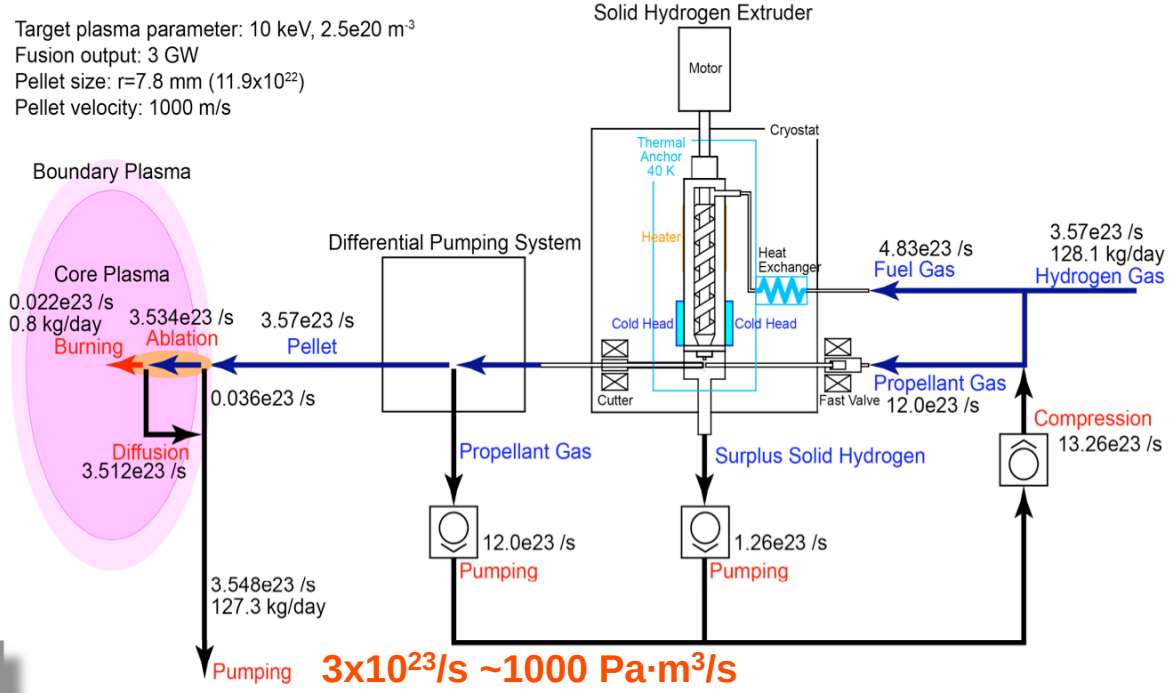
*S. Satake et al., Annual Report of NIFS, April 2007-March 2008, 282.

Hydrogen isotope circulation for burning plasma sustainment

R. Sakamoto, NIFS Annual repo. 2010-2011.

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.

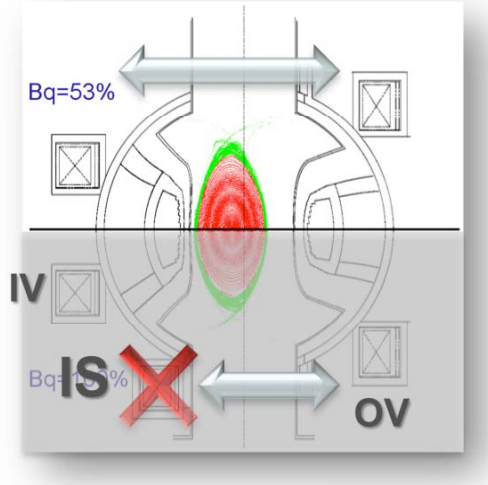
Steady-state evaluation of hydrogen isotope circulation for burning plasma sustainment

2nd cycle

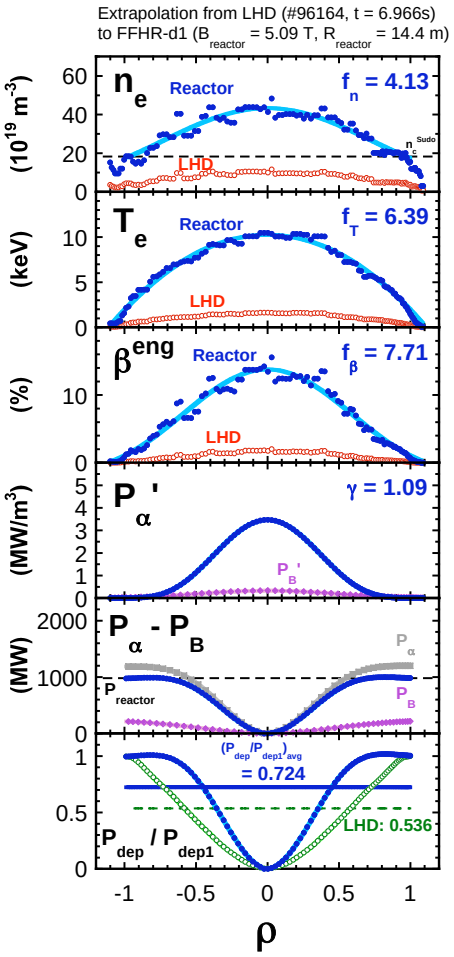
Joint works between NIFS Projects

Large maintenance ports are available w/o IS coils

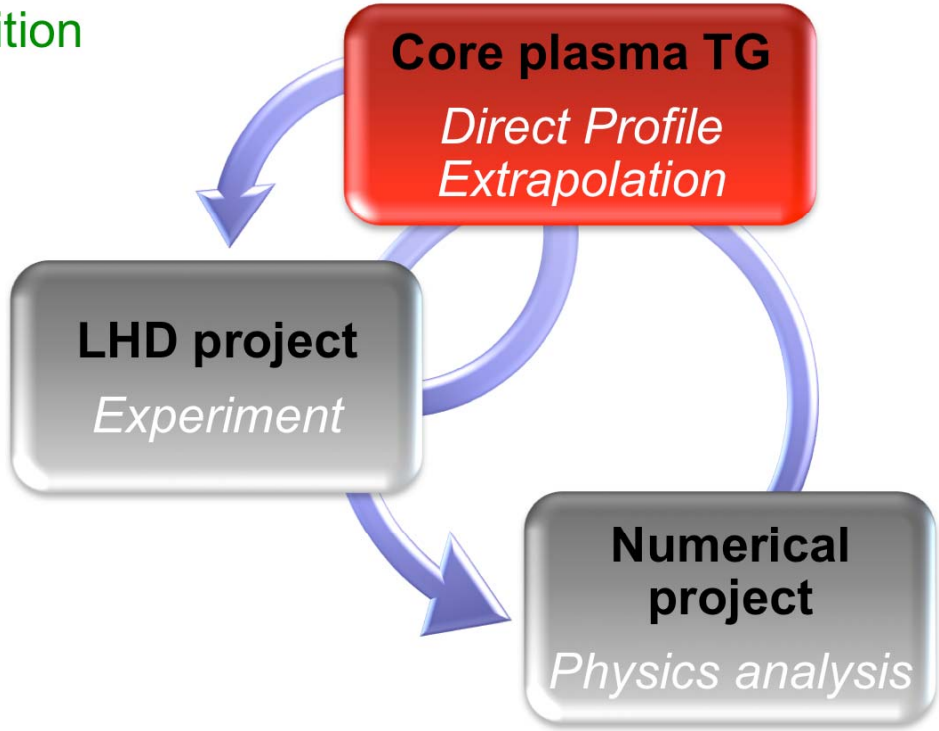
1 Proposal of experiment with FFHR-d1 like magnetic configuration

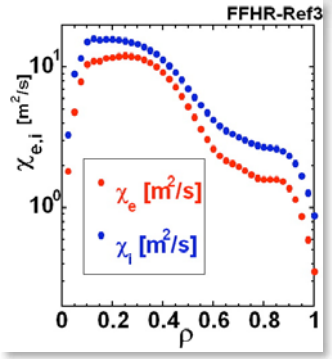


2 Experiment and DPE to self-ignition condition



Radial profiles extrapolated from LHD to FFHR-d1

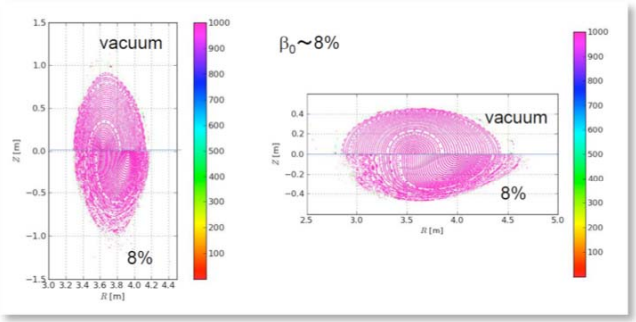




GSRAKE results

4 Detailed physics studies become possible

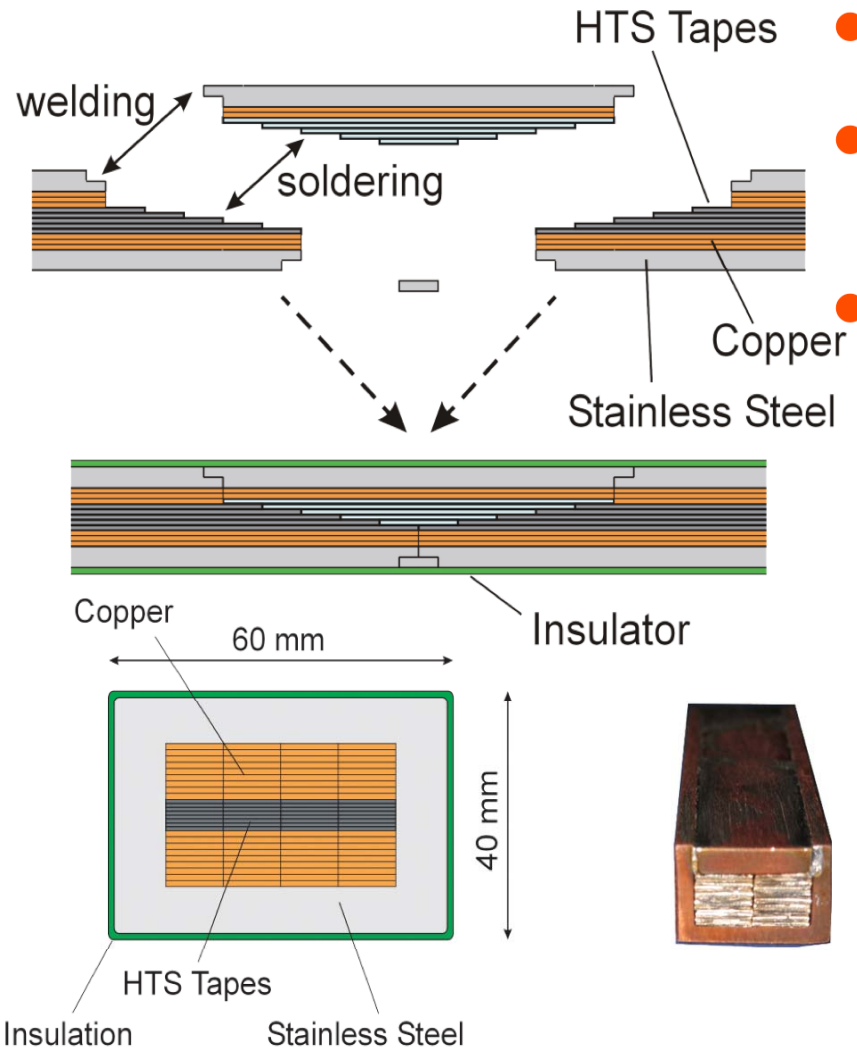
3 Reconstruction of MHD equilibrium using HINT2 and VMEC



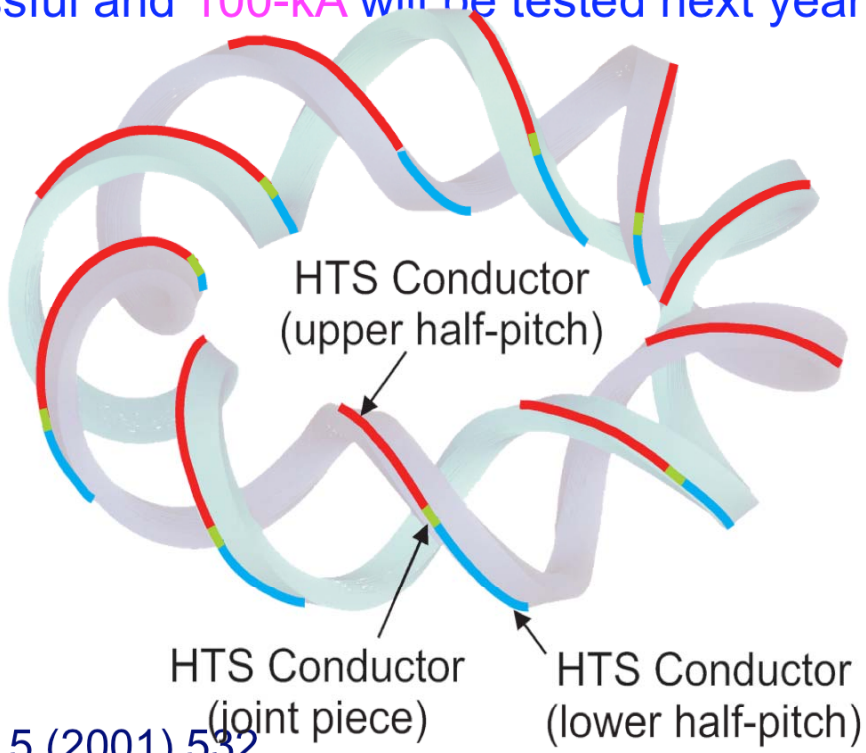
HINT2 results

R&D for DEMO

Option: Helical coil winding by joining HTS conductors



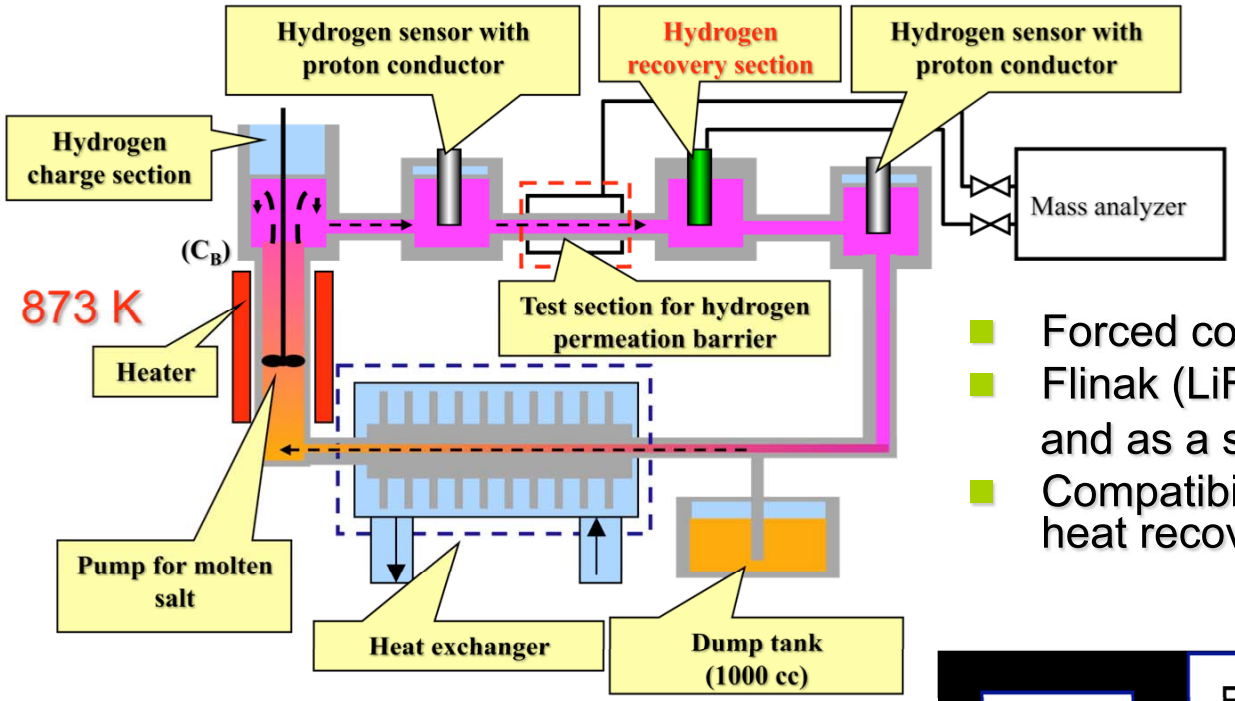
- Large-current high-Tc superconductor (HTS) using YBCO is being developed
- Continuous helical coils by joining half-pitch conductors (0.36 nΩ×8000 @20 K ➔ 4.5 MW@R.T.)
- 10-kA conductors with joints have been successful and 100-kA will be tested next year



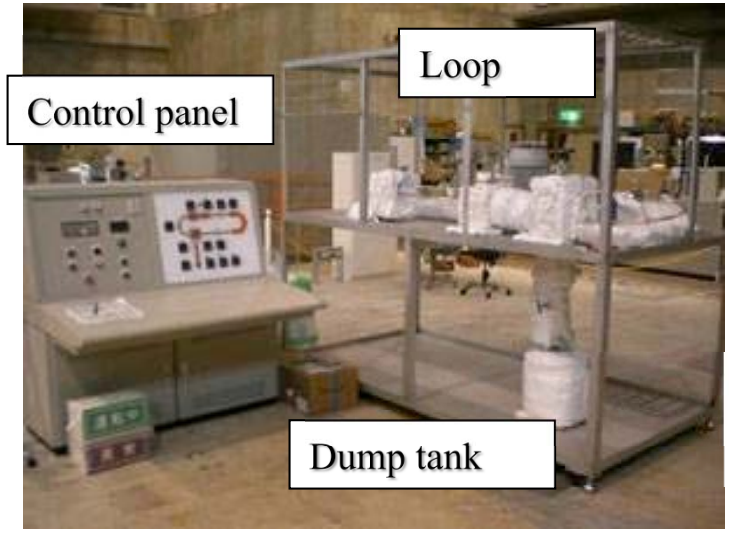
H. Hashizume *et al.*, *J. Plasma Fusion Res. SERIES* 5 (2001) 532
N. Yanagi *et al.*, Fusion Science and Technology 60 (2011) 648
N. Yanagi, HTS^{4}Fusion Conductor Workshop (2011) KIT, Germany

Integration for steady-state blanket system in NIFS: Orosh^{2}i-1 (Operational Recovery Of Separated Hydrogen and Heat Inquiry – 1)

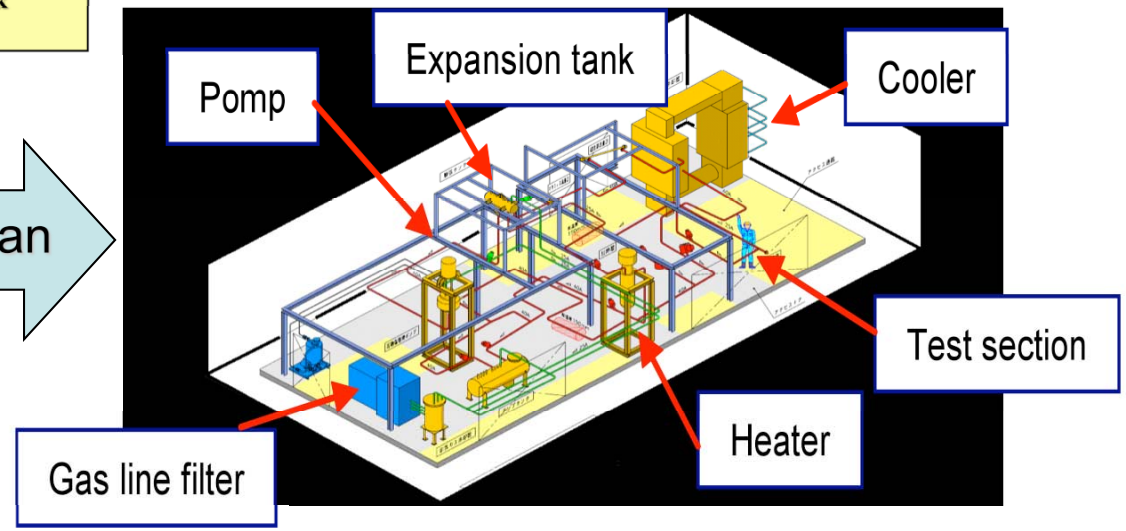
A. Sagara et al., Fus. Sci. Technol., 60 (2011) 3-10.



- 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

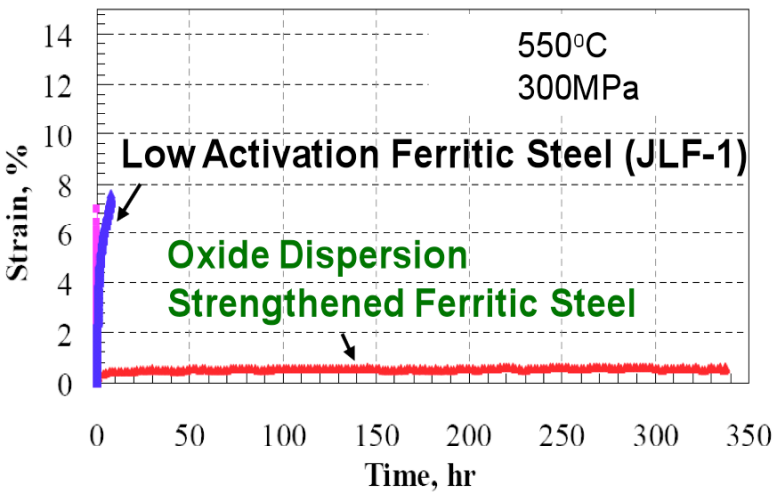


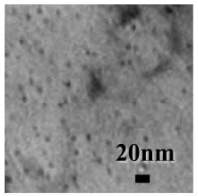
Plan



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

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.

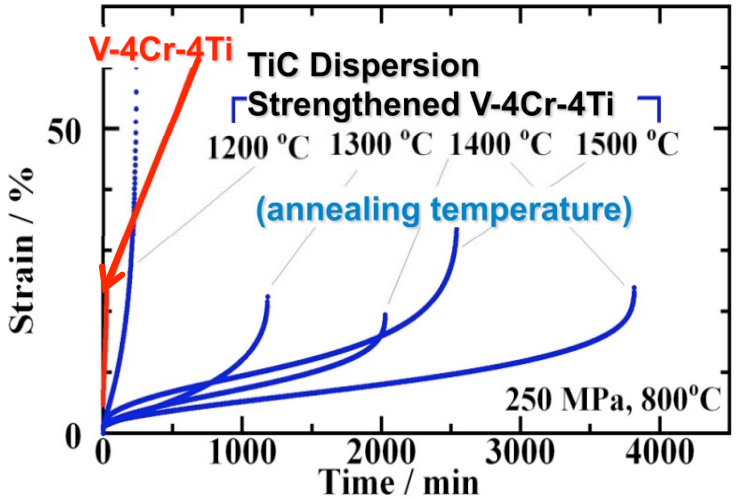


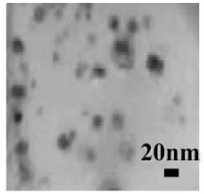


Nano-Y_2O_3 particles

Ferritic Steel

Y.F. Li et al. SOFT (2010)





Nano-TiC particles

Vanadium alloy

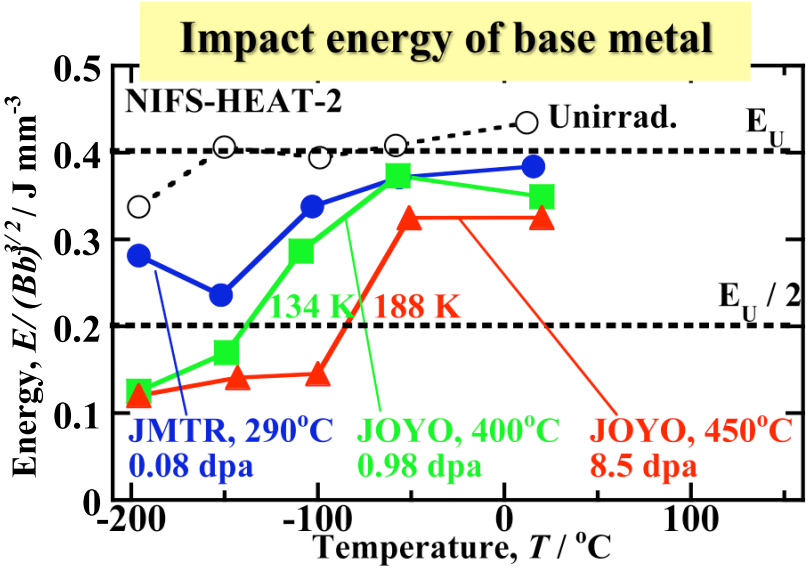
H. Kurishita et al. ICFRM-14 (2009)

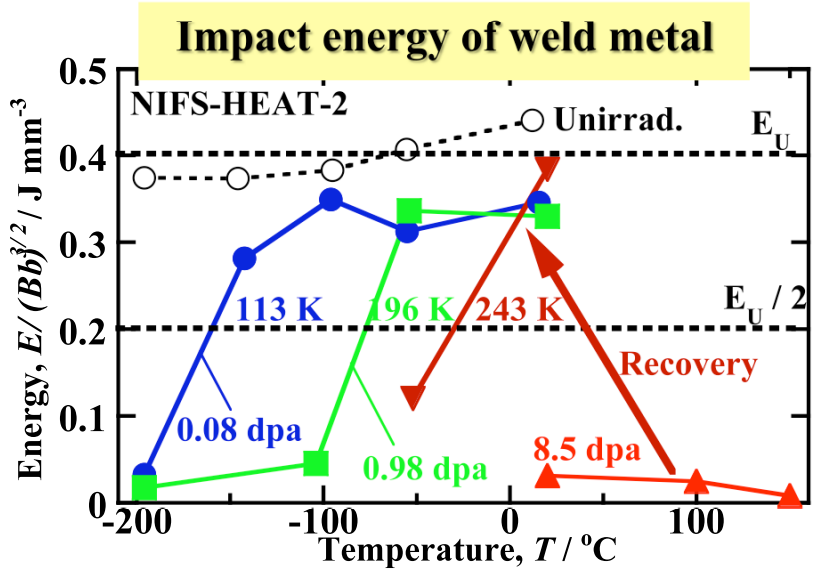
Maximum Operation Temperature
RAFM : ~550°C
ODS-Steel : 650~700°C

Maximum Operation Temperature
V-4Cr-4Ti : ~700°C
ODS-V alloy : ~800°C

Fabrication technology, fracture toughness and isotropy of strength are the key issues

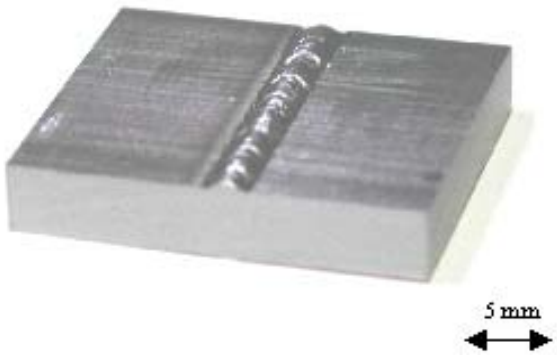
Irradiation effects and recovery on weld joint of NIFS-HEAT-2 V-4Cr-4Ti alloy



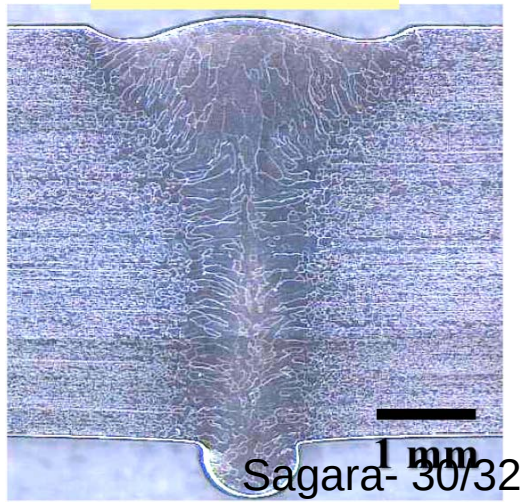


- Base metal maintained DBTT below RT after 8.5 dpa irradiation
- Degradation of impact energy and DBTT shift by irradiation was enhanced in weld metal
- Weld metal maintained DBTT below RT after 0.98 dpa irradiation, however impact energy was lost at 8.5 dpa
- The fracture energy was recovered by post-irradiation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

NH2 Weld

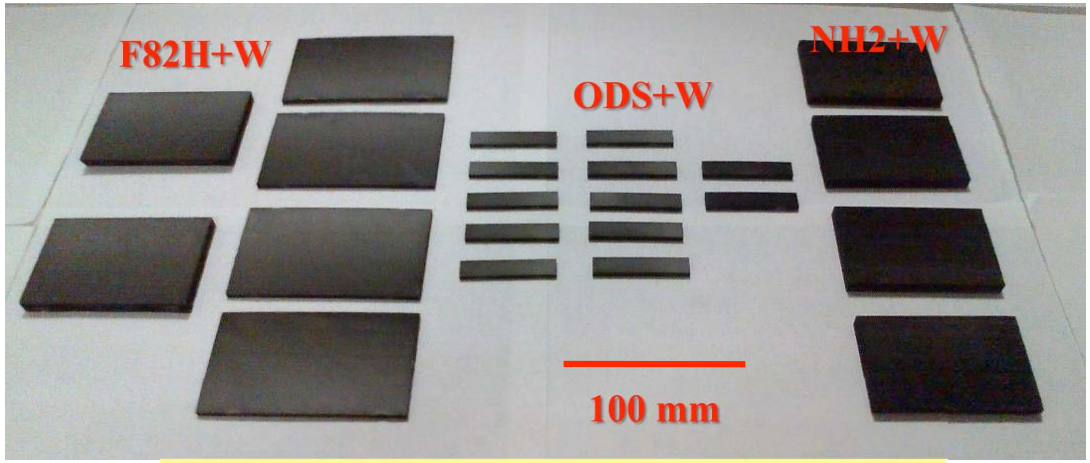


Cross section

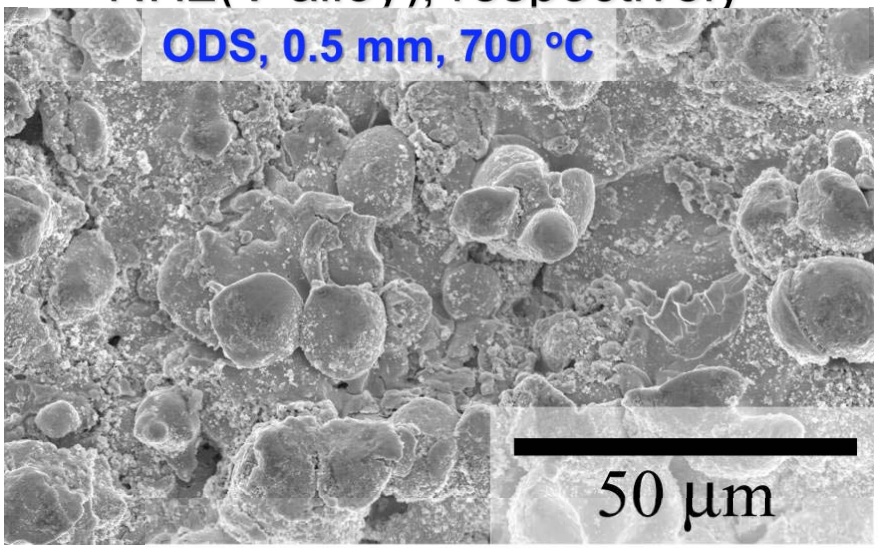


VPS-W coating on low activation materials

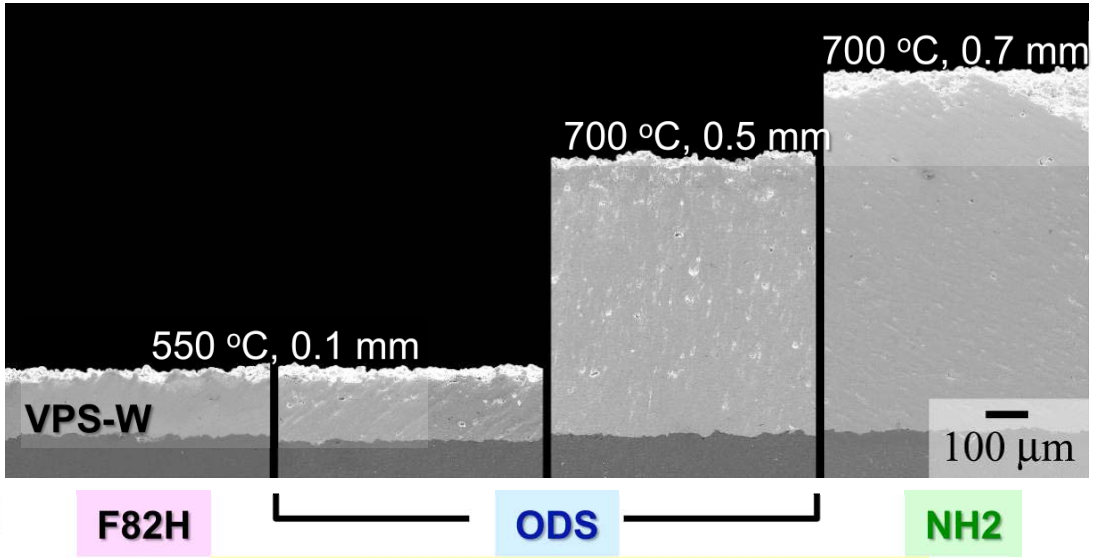
- W coating for the 1st wall was fabricated on various low activation materials by vacuum plasma spray process
- Substrate temperature was controlled around 550°C for F82H, 700°C for ODS and NH2(V-alloy), respectively



W-coated low activation materials



Surface after VPS-W coating



Cross section of VPS-W coating

Summary

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