Overview of NIFS Fusion Engineering Research Project and Helical Demo FFHR-d1

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Fusion Engineering Research Project
National Institute for Fusion Science

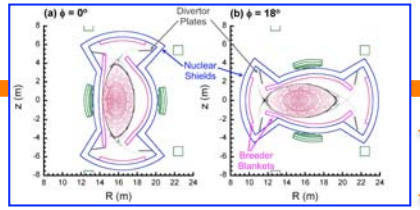
Japan-US Workshop on
Fusion Power Plants and Related Advanced Technologies with participations of EU and Korea
February 22-24, 2011 at National Institute for Fusion Science in Toki, JAPAN

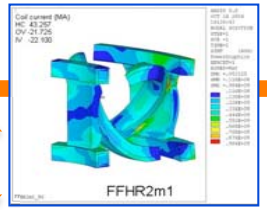
Fusion Eng. Research Project has started towards steady-state helical DEMO reactor

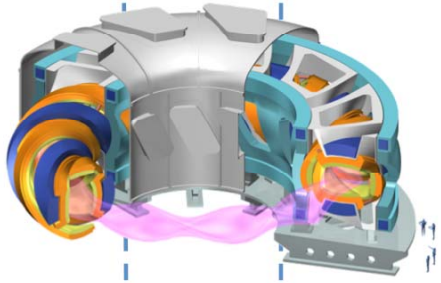
2010 — 2016 — 2022 — 2027 — 2036FY

Step by step advancement of reactor design

Conceptual design → Basic design → Improved basic design





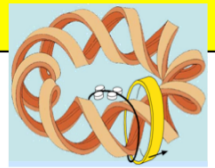


Establishment of engineering base

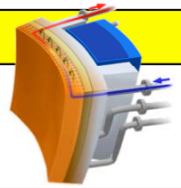
Full-scale, full-condition testing

With universities

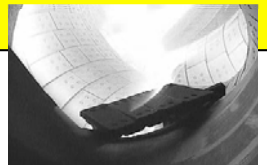
Large-scale high-field superconducting magnet



Long-life liquid blanket

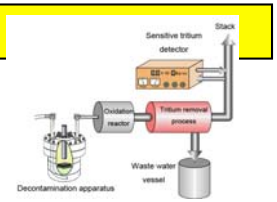


Low activation structural materials



High heat flux plasma facing wall

Tritium control



Engineering design

Construction

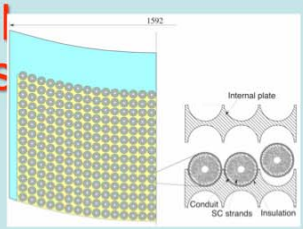
Licensing

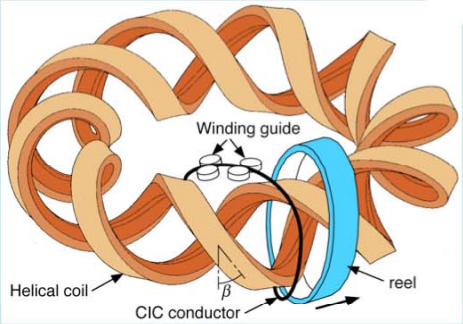
Operation

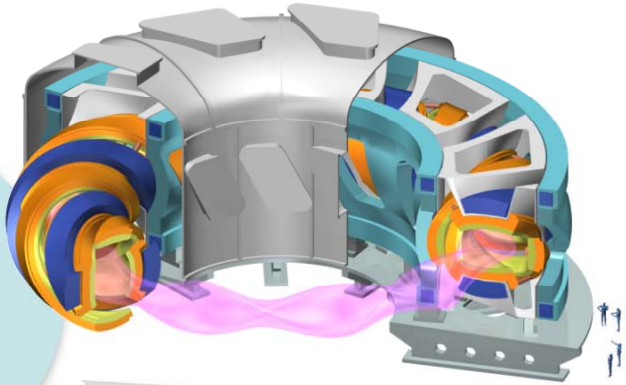
Fusion Eng. Research Project has started towards steady-state helical DEMO reactor

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

Conceptual design to wind a helical coil with CIC conductors



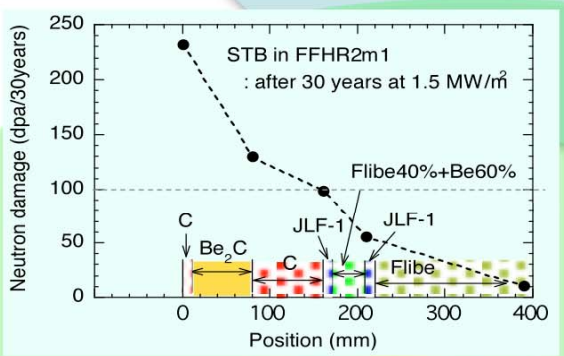


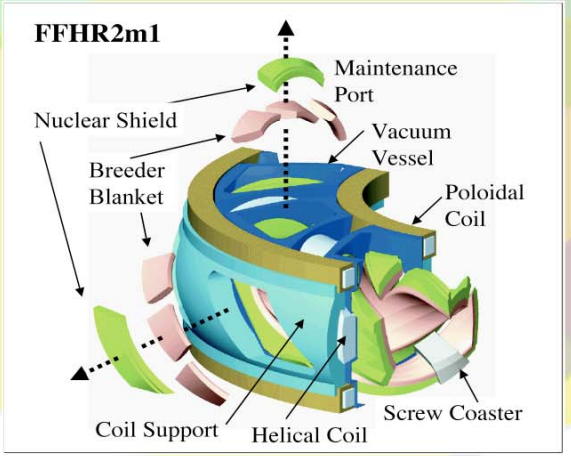


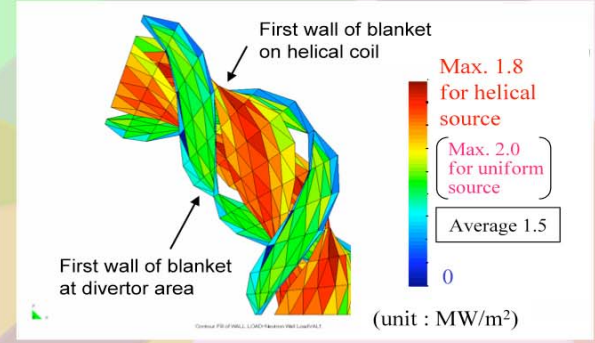
Steady-state Reactor design, 3 GWth, 30,000 ton

Step by step advancement of reactor design

Long life blanket design and wide maintenance ports

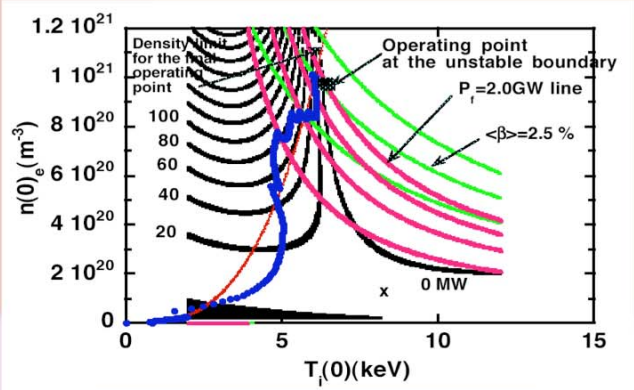






3D neutronics for non-axisymmetric helical systems

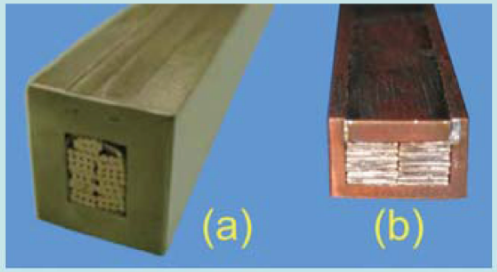
SDC ignition control by new PID method

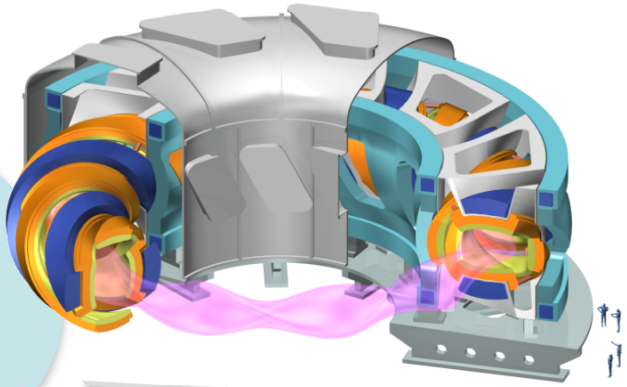


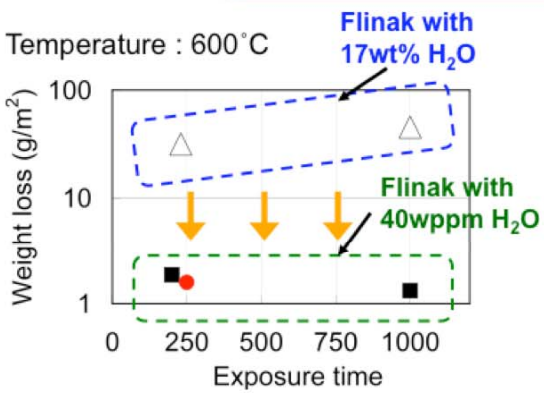
Fusion Eng. Research Project has started towards steady-state helical DEMO reactor

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

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





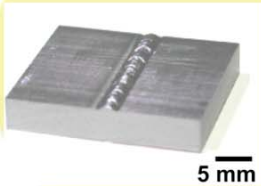


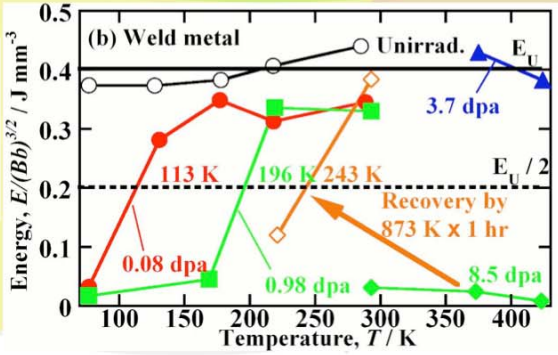
For liquid blanket, corrosion can be reduced by impurity control.

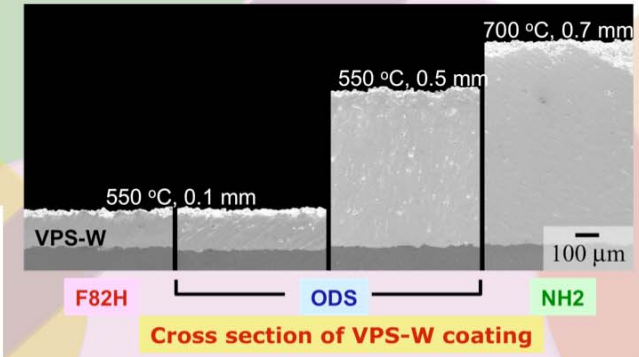
Basements for full-scale Or full-condition testing

Steady-state Reactor design, 3 GWth, 30,000 ton

Welded V-alloy can recover from neutron damage by heating.



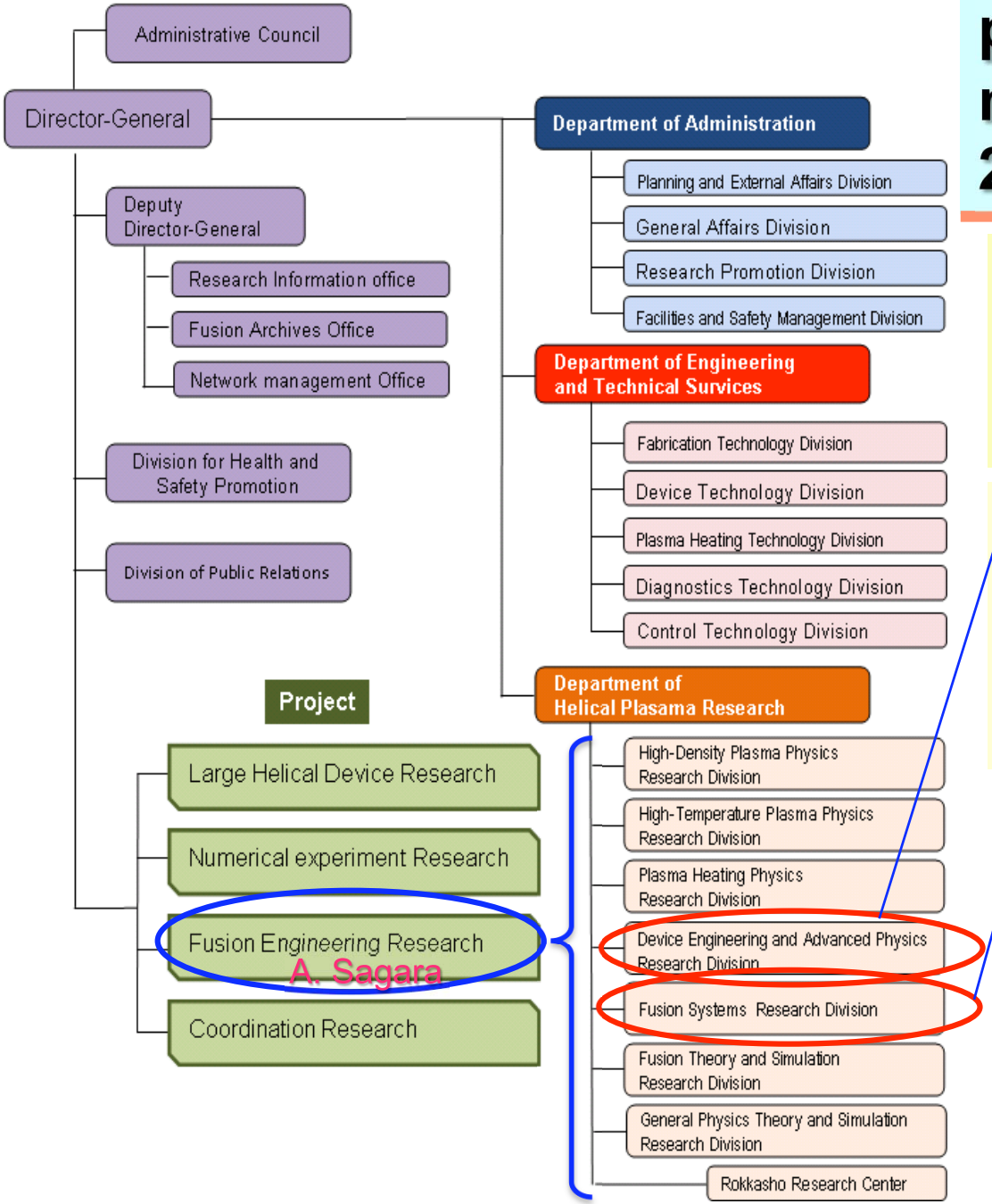




For plasma facing wall, W coating was succeeded by plasma spray.

High-sensitive tritium Monitor (with Nagoya Univ.)

New Organization (FY2010 ～)



From the FY2010, 4 projects have started in the new organization for the 2nd Mid-term goal & plan

Fusion Engineering Research Project

with about 60 staffs, mainly based on the 2 divisions:

Device Engineering and Advanced Physics Research Division (Staffs: 7,7,4)

- LHD maintenance and D-D safety devices
- Trace tritium monitoring and handling
- Large SC magnet system and applications

Fusion System Research Division (Staffs: 6,5,8)

- Helical reactor DEMO design
- Remote maintenance and replacement
- Low activation mater. and long-life liq. blanket
- PWI and high-heat flux components
- Atomic and molecular processes

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

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

- Helical reactor conceptual design
- Helical DEMO basic design

- Testing of full-scale SC conductor
- Helical winding engineering

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

- 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

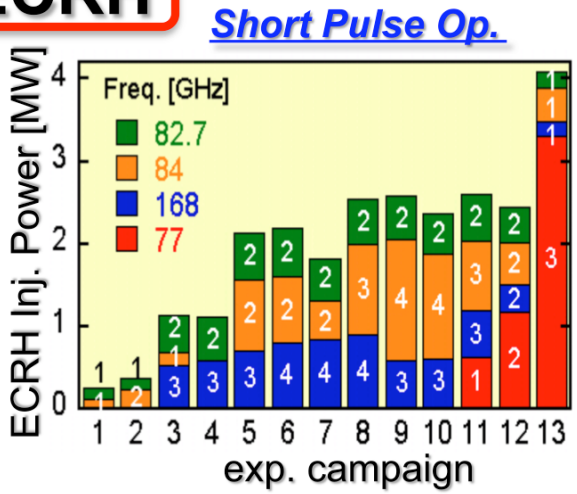
Rev. 2010.5.17

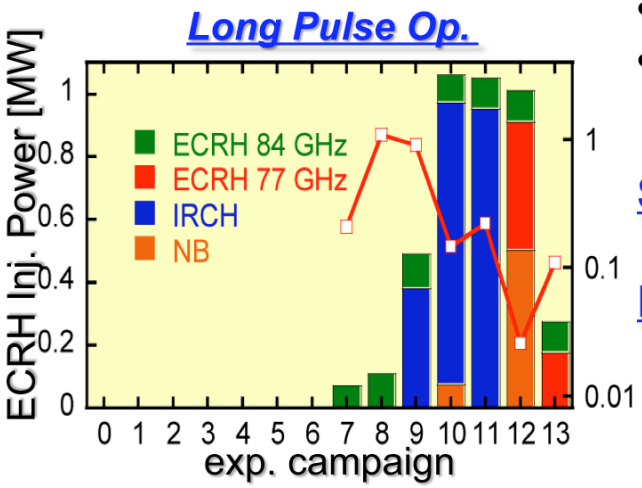
FFHR-d1

Present Status of ECRH and ICRH

Heating TGL: Tsumori

ECRH



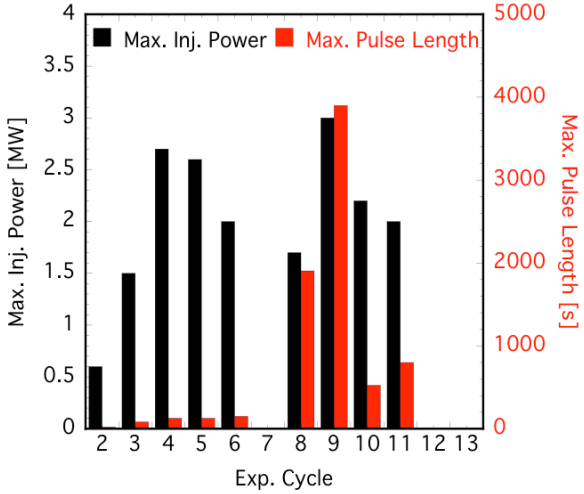


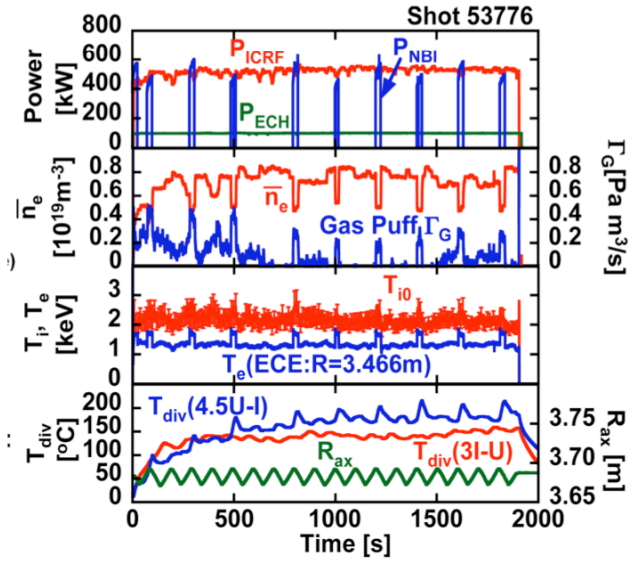
- 8 gyrotrons are operational
- 77GHz gyrotrons enhance the ECRH-injection power.

Short Pulse:
- 3.7 MW input, Te=15 keV

Long Pulse :
- 0.1 MW input, Te=1.5 keV n_e=0.15x10^{19} m^{-3}, 1 hr.
- 0.3 MW input, Te=3.0 keV n_e=0.23x10^{19} m^{-3}, 400 s.

ICRF





for short pulse operation :
- 2 MW output (50 MHz)

for long pulse operation :
- 1.6 MW output (50 MHz)

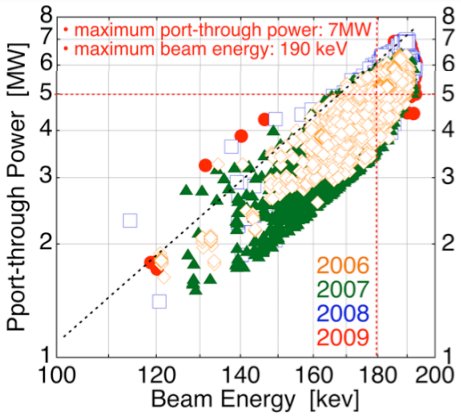
loop and wave-guide antennae

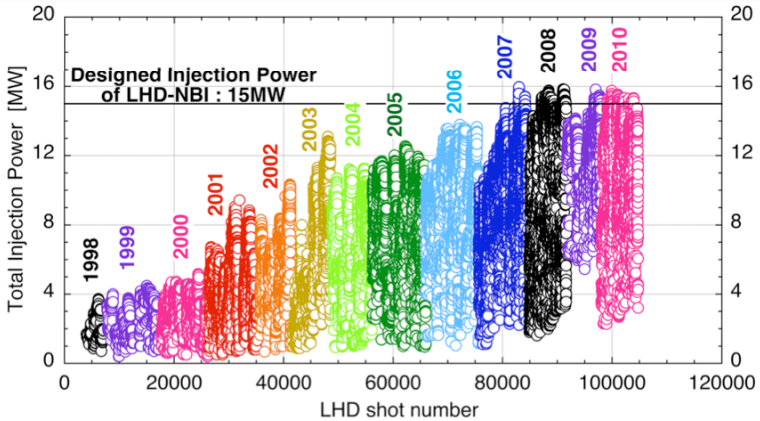
Long Pulse :
- 0.5 MW input, Te=2.0 keV n_e=0.8x10^{19} m^{-3}, 1905 sec.

With scanning magnetic-axis

Present Status of NBI and Future Issues of Heating Devices

NBI





Short Pulse:

- maximum: 190 keV, 7 MW 1.5 sec with a beamline
- total injection power: 16 MW with 3 negative-ion-based NBI systems.
- all the 3 negative NBI have reached mission values of LHD-NBI (≥ 5 MW and ≥ 180 keV)

Long Pulse: 0.1 MW, 110 sec / 0.5 MW, 80 sec

Future Issues

ECRH:

- New gyrotron should be developed, when the 2nd harmonic resonance is adopted.
- Neutron shielding for movable EBW antenna should be considered.

ICRF:

- Loop and waveguide antenna are difficult to apply for reactors due to installation space and neutron damage of the antennae.
- RF-coupling with target plasmas is necessary to be investigated.

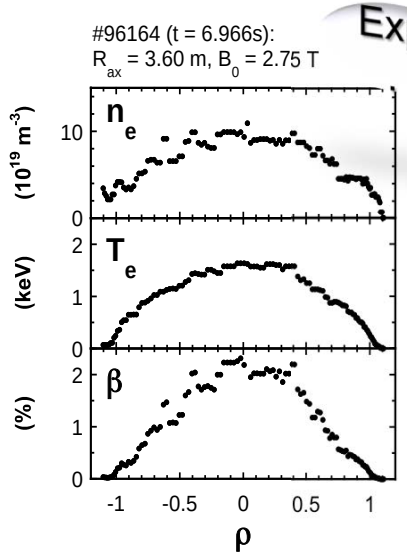
NBI:

- Control of caesium recycling for long-pulse injection more than 100 sec.
- Neutron shield for inner components of beamline is needed to be designed and developed.

Deductive Procedure using DPE Method Makes Reactor Design Reliable

Core Plasma TGL: Miyazawa

1. Direct Profile Extrapolation (DPE) method has been developed.
2. DPE enables the "deductive" fusion reactor design, where the plasma performance determines engineering parameters.



DPE

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

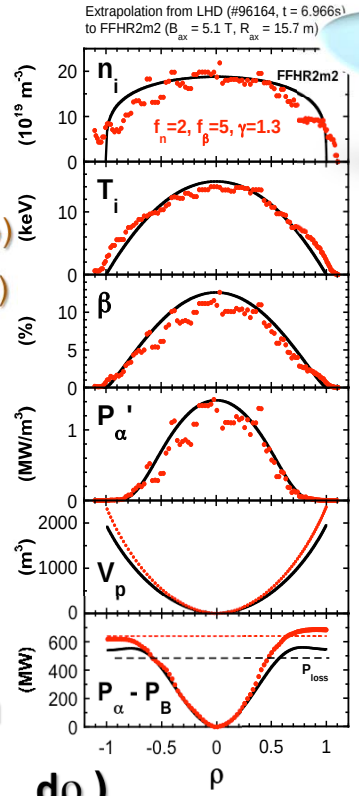
$n_{reactor}(\rho) = f_n\, n_{exp}(\rho)$

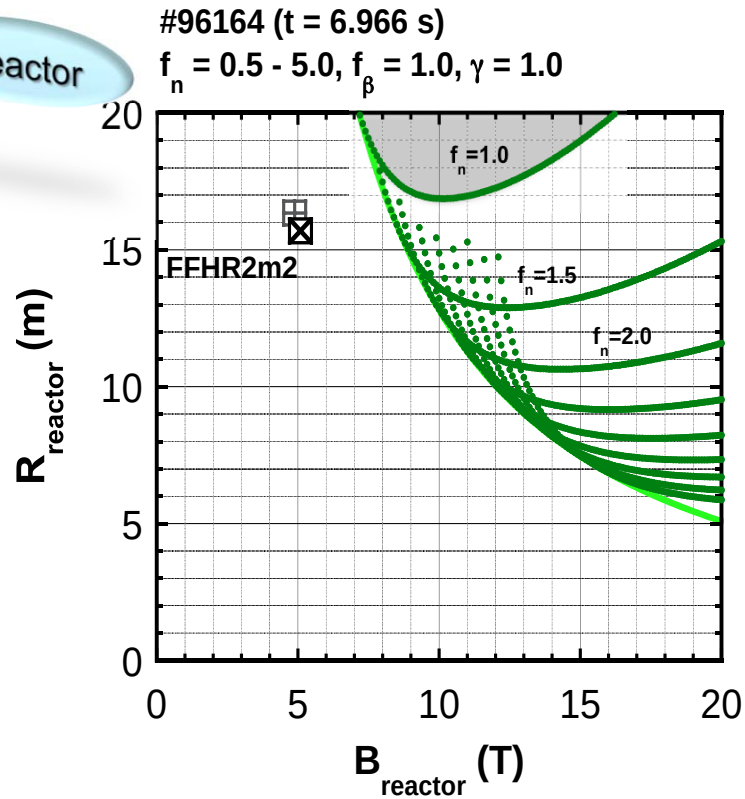
$a_{reactor} = f_a\, a_{exp}$

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

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

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





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

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

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

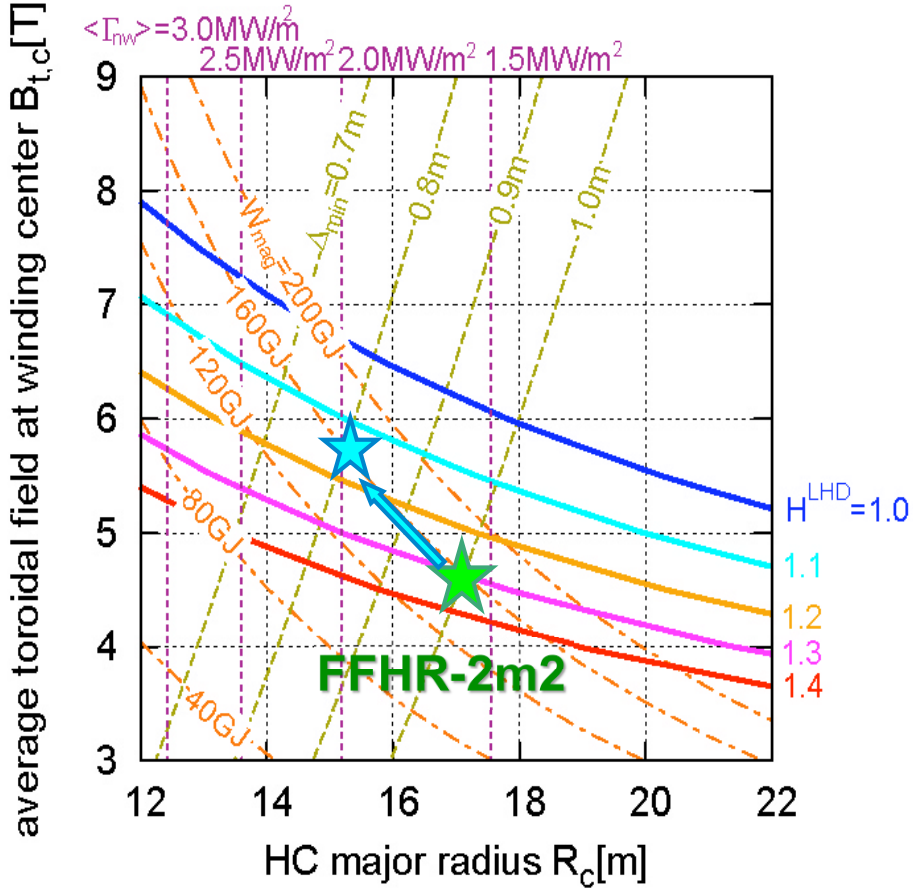
Radial profiles in the experiment are directly extrapolated to the reactor

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

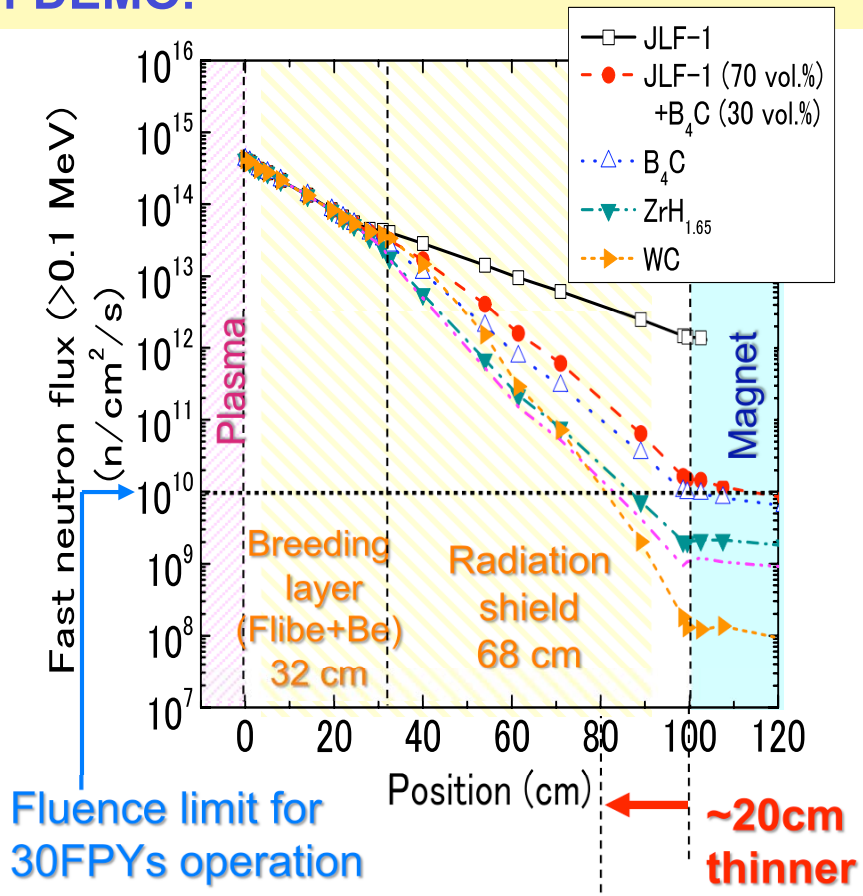
Flexible DEMO design concept can expand the design window

Reactor TGL: Goto

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 or by shortening the operation period of DEMO.



T. Goto et al., to be published in PFR.

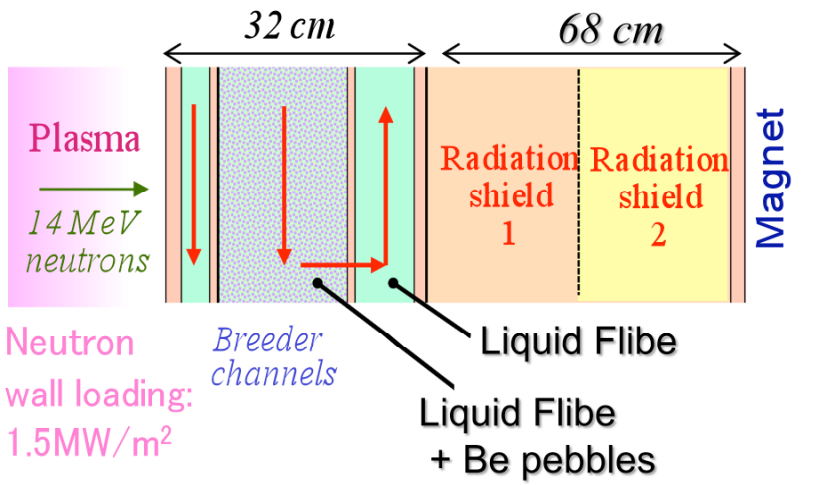


*Courtesy of Dr. Tanaka (Original figure is found in A. Sagara et al., FED **83** 1690 (2008)*

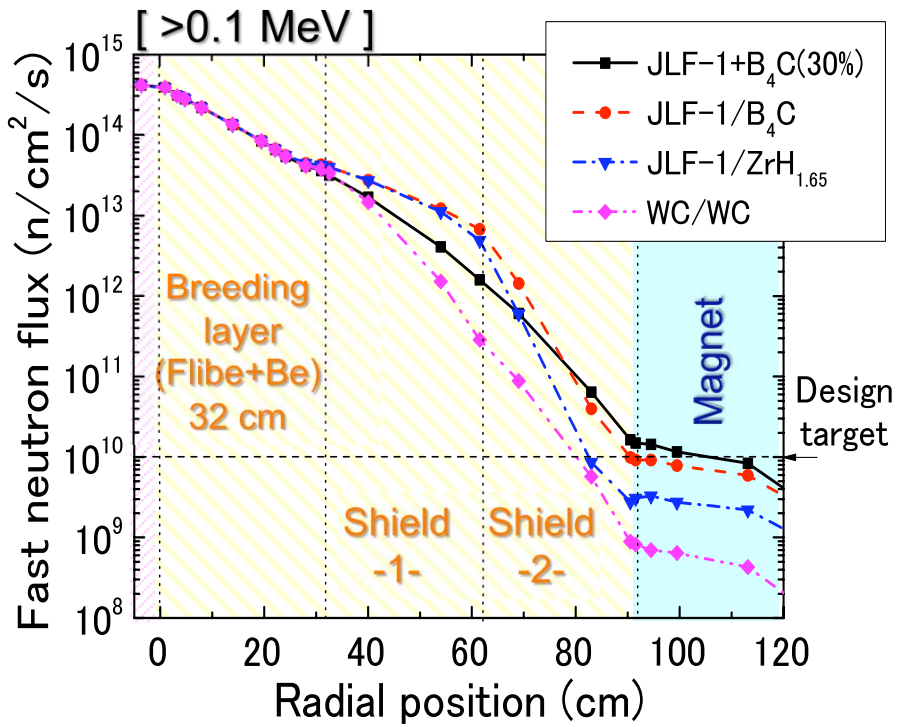
Two-layered shielding is investigated for neutron shielding in FFHR

Blanket TGL: Tanaka

1. B_4C, WC, ZrH_x etc. are considered as effective shielding materials for FFHR.
2. B_4C and ZrH_x can fulfill their shielding performances at a backside shield.



Structural material: JLF-1
(Reduced activation ferritic/martensitic steel)



T. Tanaka al., NIFS annual report 2009 Apr.-2010 Mar., p282.

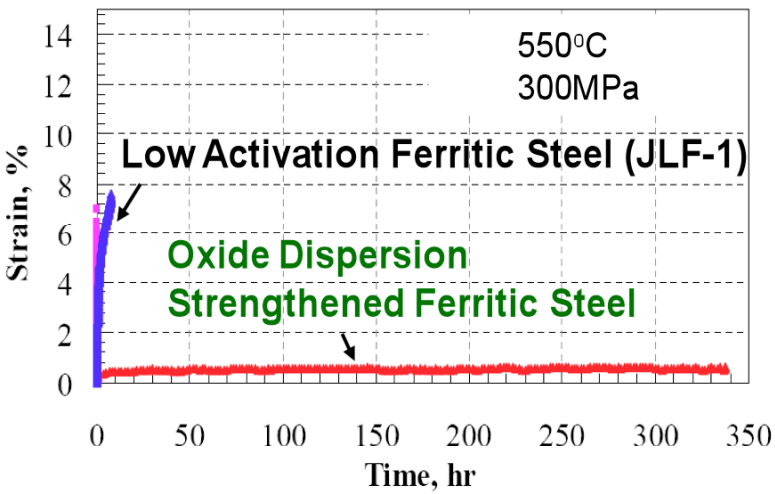
Reduction of a shielding layer thickness has a significant impact to the reactor design.

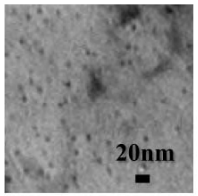
B_4C and ZrH_x could be used in a backside layer at lower temperature with lower irradiation damage.

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

Materials : Turoga

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.

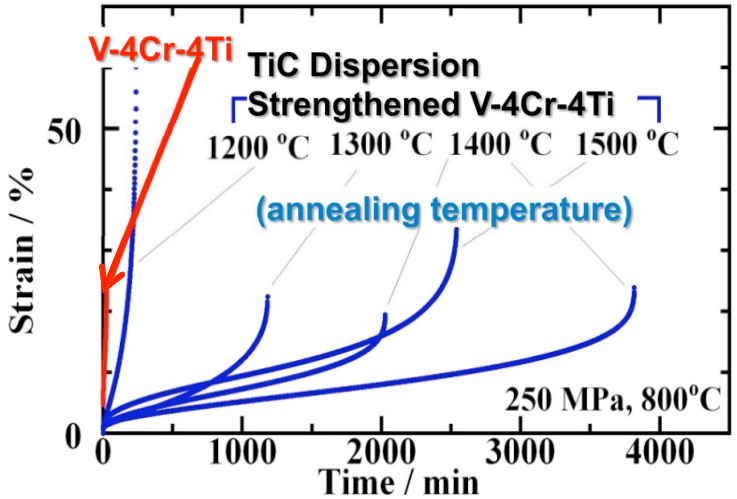


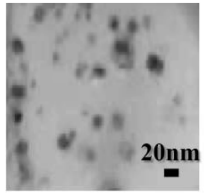


Nano-Y_2O_3 particles

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

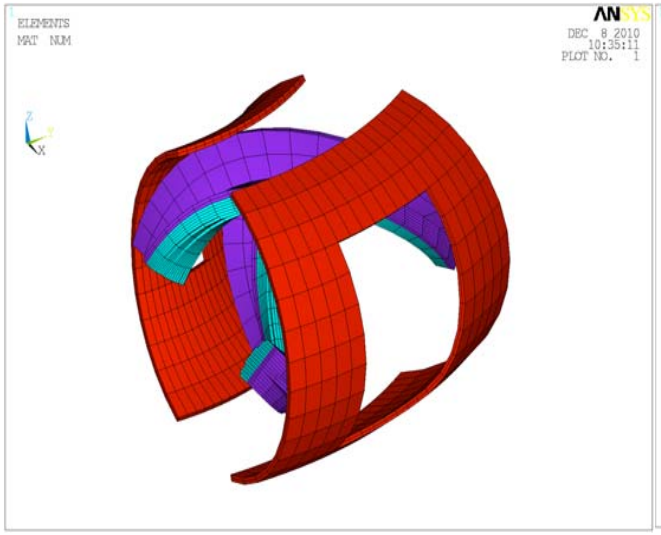
Maximum Operation Temperature
V-4Cr-4Ti : ~700°C
ODS-V alloy : ~800°C

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

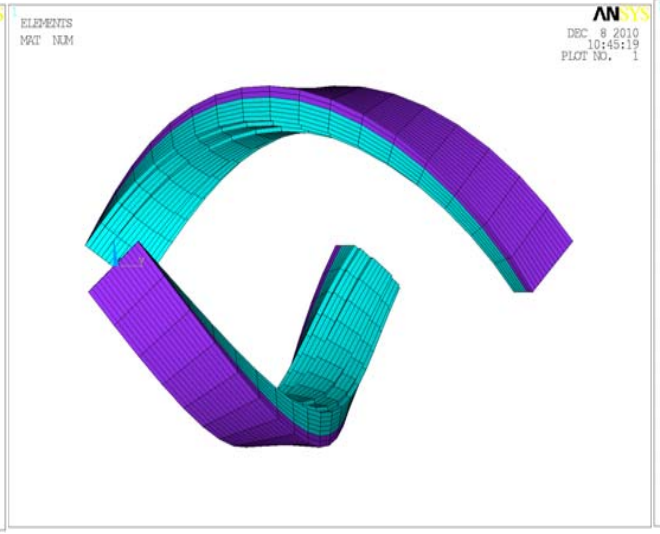
Analysis of helical coil system in FFHR

In Vessel TGL: Tamura

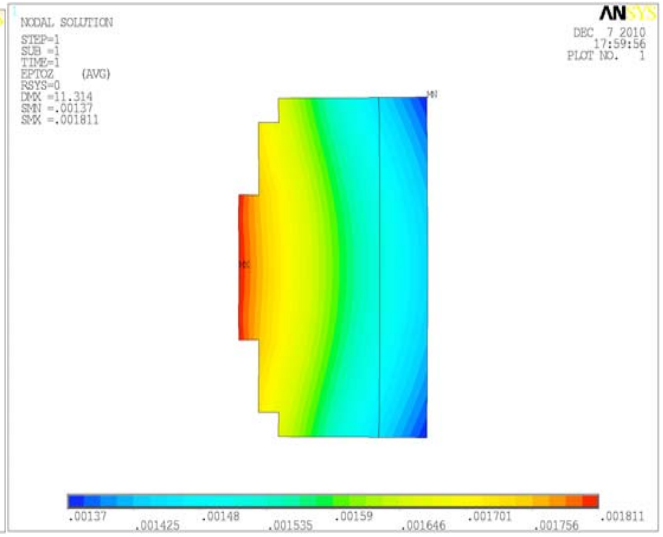
1. Differences among 3-D, quasi 3-D, and axisymmetric models were investigated.
2. Stress and strain levels can be estimated by the quasi 3-D or the axisymmetric model with about 25% margin.
3. Displacement of each component should be estimated by 3-D model since it highly depends on a supporting structure.



Example of full 3-D model; coil and supporting structure with large opening.



Quasi 3-D model; coil with specific B.C. (displacement of winding direction was set to zero.)

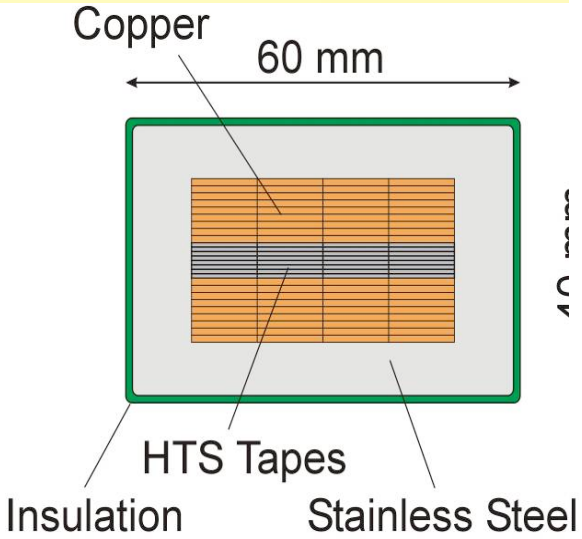


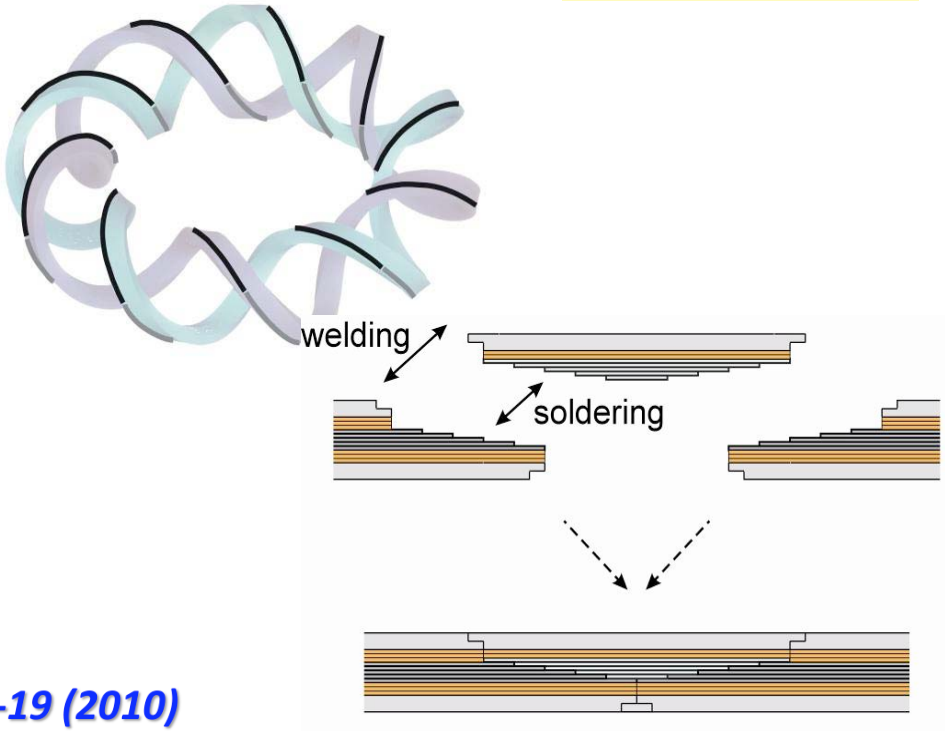
Hoop strain calculated by the axisymmetric model, where mean radius of an actual coil was set.

High-temperature superconducting coil option is feasible with segmented fabrication

Magnet TGL: Yanagi

1. High-temperature superconductor of 100-kA class will be developed, starting with a successfully tested 10-kA short sample.
2. Helical coils of FFHR can be constructed by jointing half-pitch HTS conductor segments.





N.Yanagi al. presented at TOFE-19 (2010)

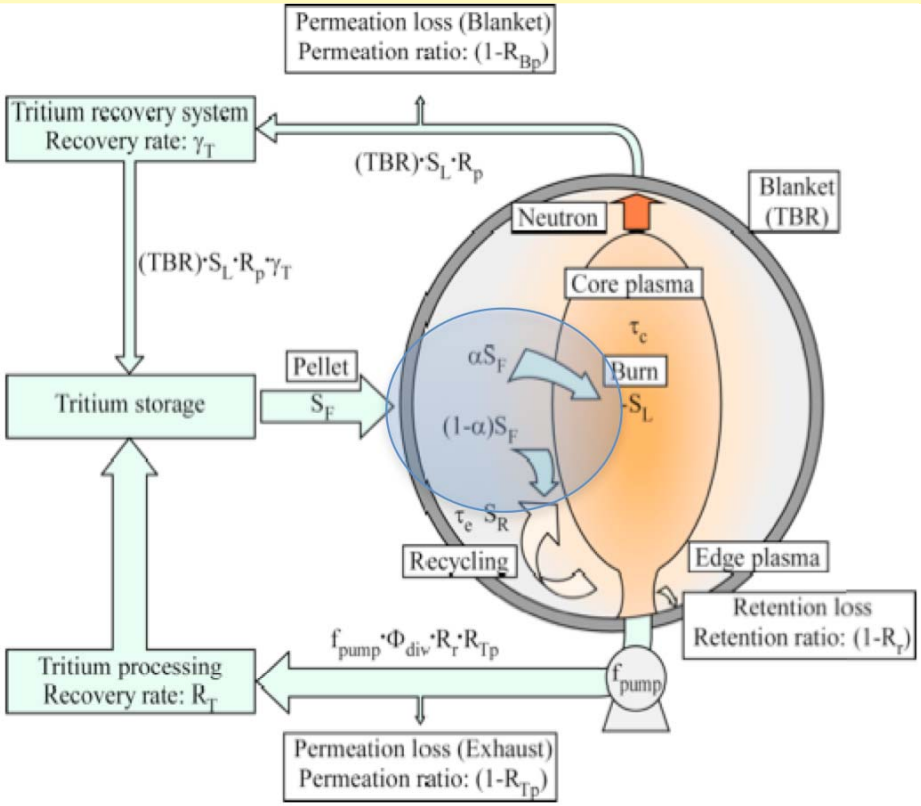
Design of 100-kA class YBCO HTS conductor and a 10-kA class sample

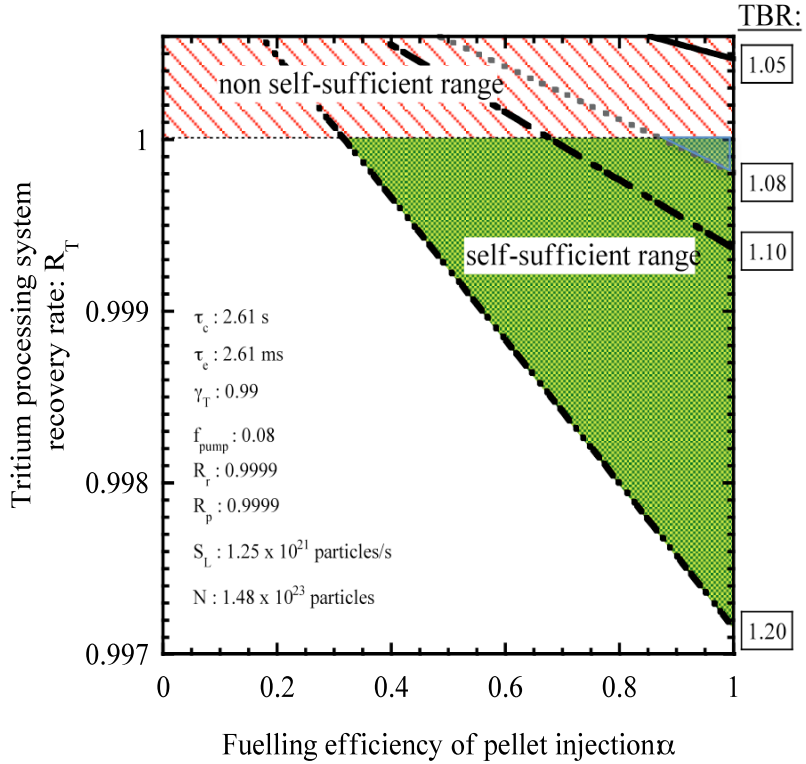
Segmented fabrication of helical coils with HTS conductor joints

Particle injection efficiency is critical issue for tritium self-sufficiency in fusion reactor

Tritium TGL: Tanaka

1. Tritium particle balance model considering the fuel injection efficiency is developing **for fuel cycle in fusion reactor**.
2. The fuel injection efficiency and TBR is critical issue for tritium self-sufficiency.





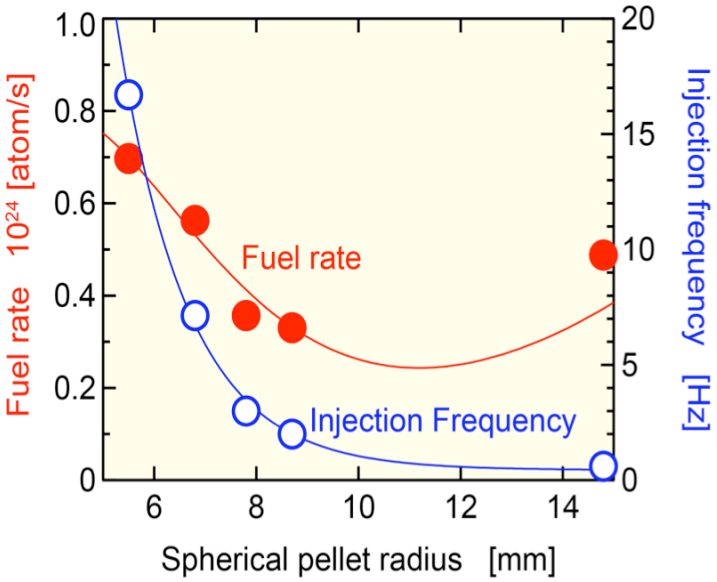
Steady state tritium particle balance model considering the fuel injection efficiency is developing for fuel cycle assessment .

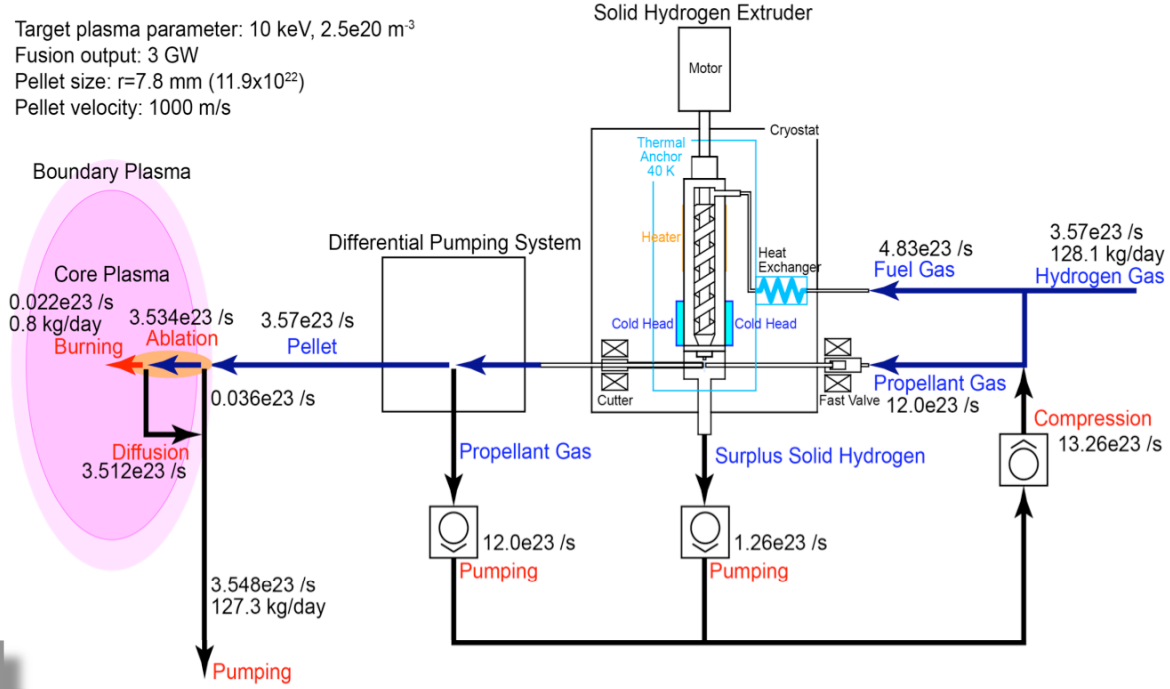
From preliminary calculation results , at least TBR 1.08 and fuelling efficiency 0.9 is required in the reactor design based on FFHR 2m2

Hydrogen isotope circulation for burning plasma sustainment

Fueling TGL: Sakamoto

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. 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.

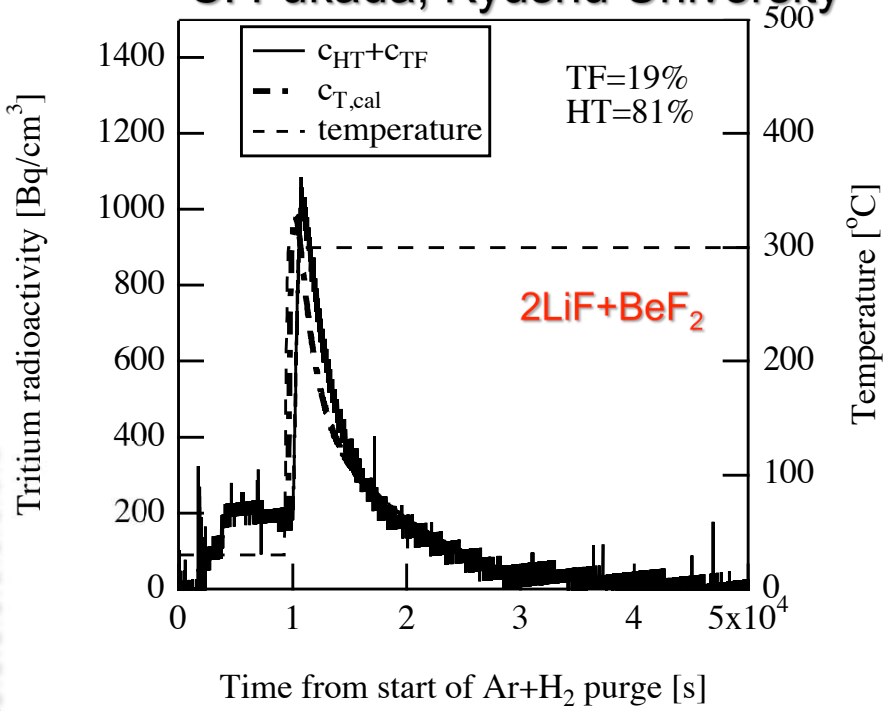
Steady-state evaluation of hydrogen isotope circulation for burning plasma sustainment

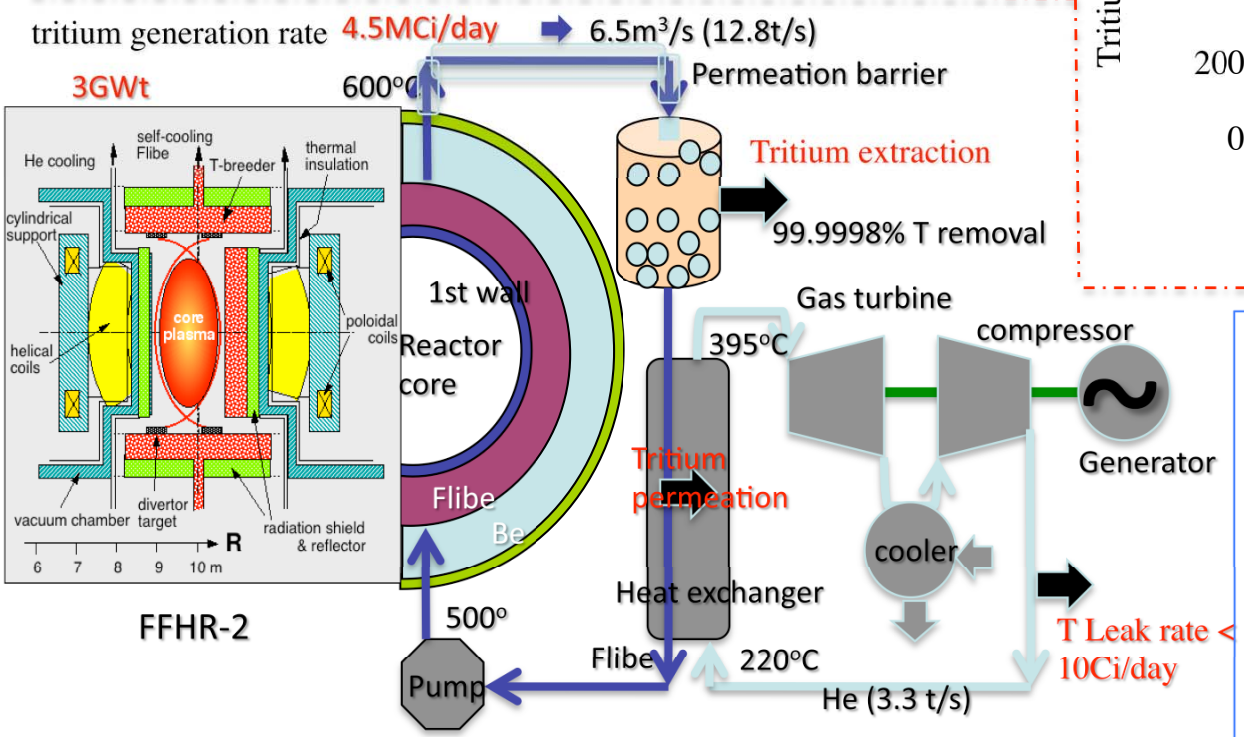
Calculation and experiment are performed for recovery of tritium and heat from FFHR-2 Flibe blanket

Experimental approach on Flibe :

Tritium is recovery from neutron irradiated Flibe by He-H_2 gas purge, and its experimental desorption rate (solid line) is well simulated by a diffusion-limiting rate process (chained line). Its tritium chemical form is a mixture of TF and HT.

S. Fukada, Kyushu University



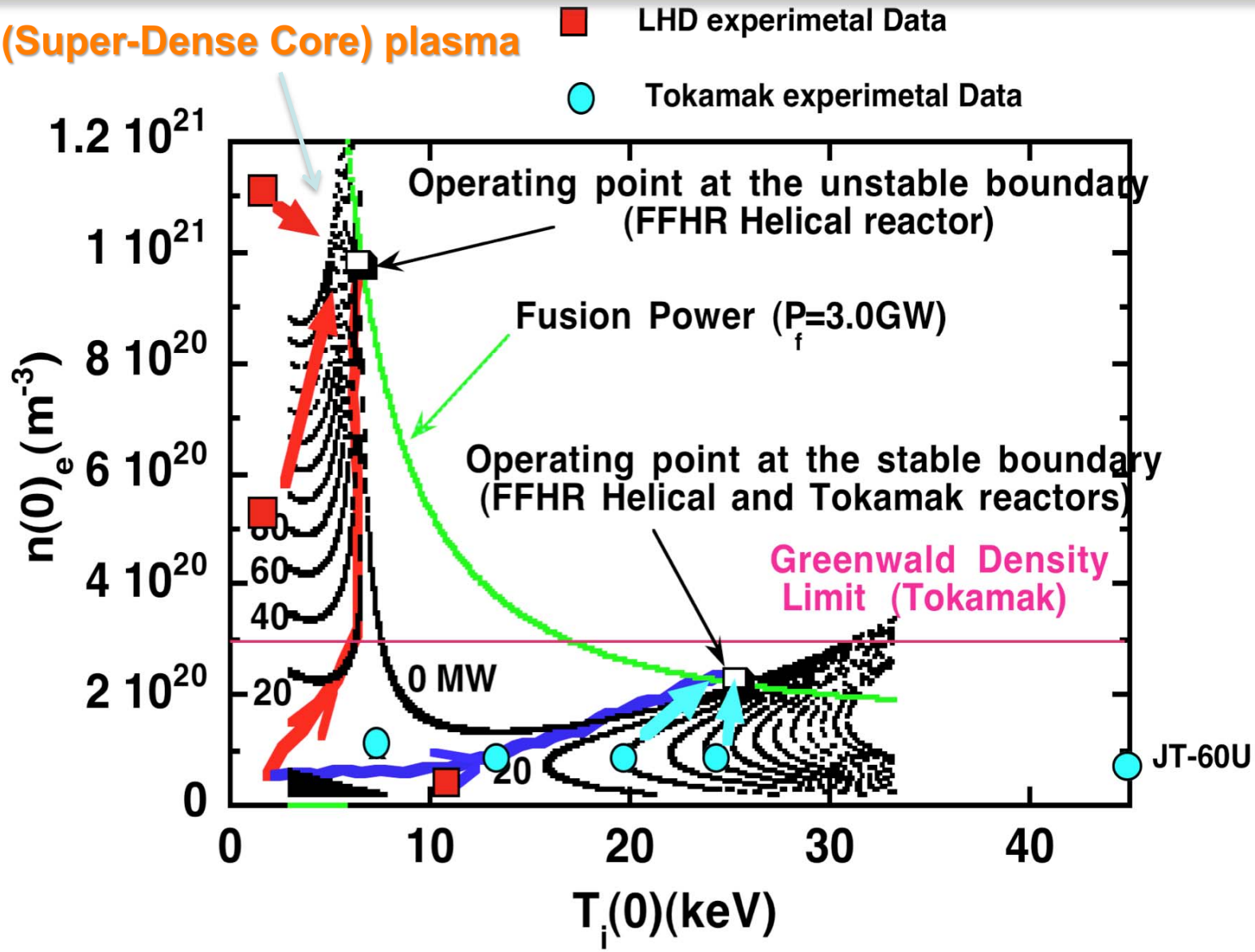


Flibe loop for FFHR-2 and He Brayton cycle

Analytical approach on Flibe blanket loop:

Flibe flow rate, temperature, overall tritium leak rate and recovery rate are estimated for FFHR-2 Flibe self-cooled blanket system

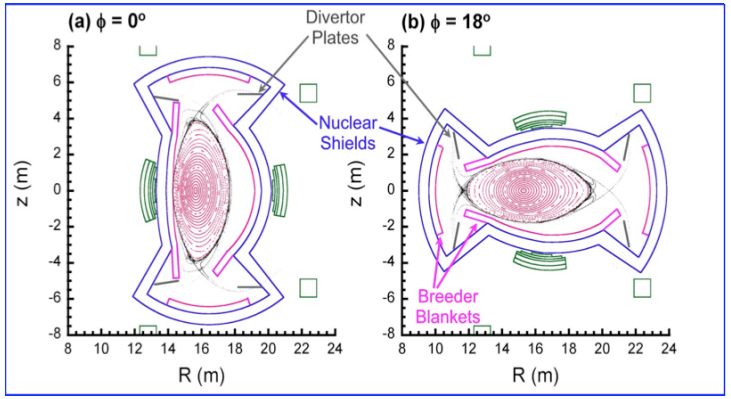
Helical system can have two operation points



POPCON plot for self-ignition access with external heating power (unit: MW)

Design parameters

Design parameters			LHD	FFHR2	FFHR2m1	FFHR2m2	SDC
Polarity	l		2	2	2	2	
Field periods	m		10	10	10	10	
Coil pitch parameter	γ		1.25	1.15	1.15	1.20	
Coil major Radius	R_c	m	3.9	10	14.0	17.3	
Coil minor radius	a_c	m	0.98	2.3	3.22	4.16	
Plasma major radius	R_p	m	3.75	10	14.0	16.0	
Plasma radius	$\langle a_p \rangle$	m	0.61	1.24	1.73	2.35	
Plasma volume	Vp	m^3	30	303	827	1744	
Blanket space	Δ	m	0.12	0.7	1.1	1.05	
Magnetic field	B_0	T	4	10	6.18	4.84	
Max. field on coils	B_{max}	T	9.2	14.8	13.3	11.9	
Coil current density	j	MA/m^2	53	25	26.6	26	
Magnetic energy		GJ	1.64	147	133	160	
Fusion power	P_F	GW		1	1.9	3	
Neutron wall load	Γ_n	MW/m^2		1.5	1.5	1.5	
External heating power	P_{ext}	MW		70	80	43	100
α heating efficiency	η_α			0.7	0.9	0.9	0.9
Density lim.improvement				1	1.5	1.5	7.5
H factor of ISS95				2.40	1.92	1.92	1.60
Confinement time	τ_E	s				2.40	6.72
He ash confinement time ratio	τ_α^*/τ_E					3	3
Effective ion charge	Z_{eff}			1.40	1.34	1.48	1.55
Electron density	$n_e(0)$	10^19 m^{-3}		27.4	26.7	17.9	83.0
Temperature	$T_i(0)$	keV		21	15.8	18	6.33
Plasma beta	$\langle\beta\rangle$	%		1.6	3.0	4.40	3.35
Radiation loss	P_r	MW				92	416
Plasma conduction loss	P_L	MW			290	453	115
Heat flux to first wall	P_W	MW/m^2				0.06	0.27
Diverter heat load (1m wet width)	Γ_{div}	MW/m^2			1.6	2.3	0.6
Total capital cost		G$(2003)		4.6	5.6	7.0	
COE		mill/kWh		155	106	93	



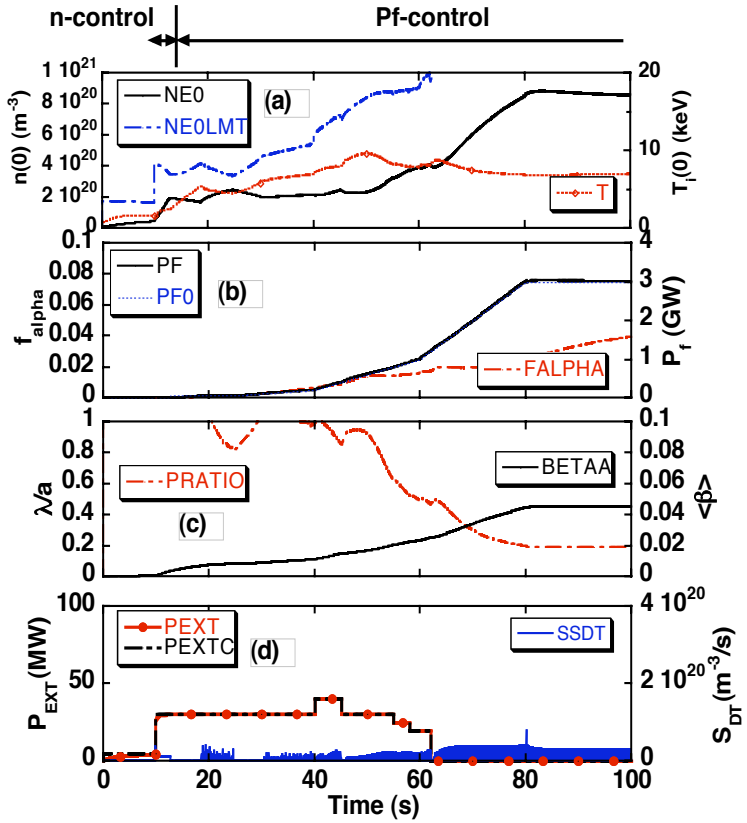
Latest design "FFHR-2m2 Type-D" with maximized magnetic surfaces by optimization of poloidal coil positions and currents. R_c = 17 m, R_p = 15.7 m, $\langle a_p \rangle$=2.5m, γ = 1.2, α = +0.1, $B_{t,c}$ = 4.45 T, j_{HC} =25 A/mm^2, W_{mag}=160Gj. $\langle\beta\rangle$=5.6% by Finite-beta equilibrium analysis using VMEC code

In SDC, divertor heat load is drastically reduced (1/3~1/4).

The fusion power control should be employed from the initial phase for high-density operation.

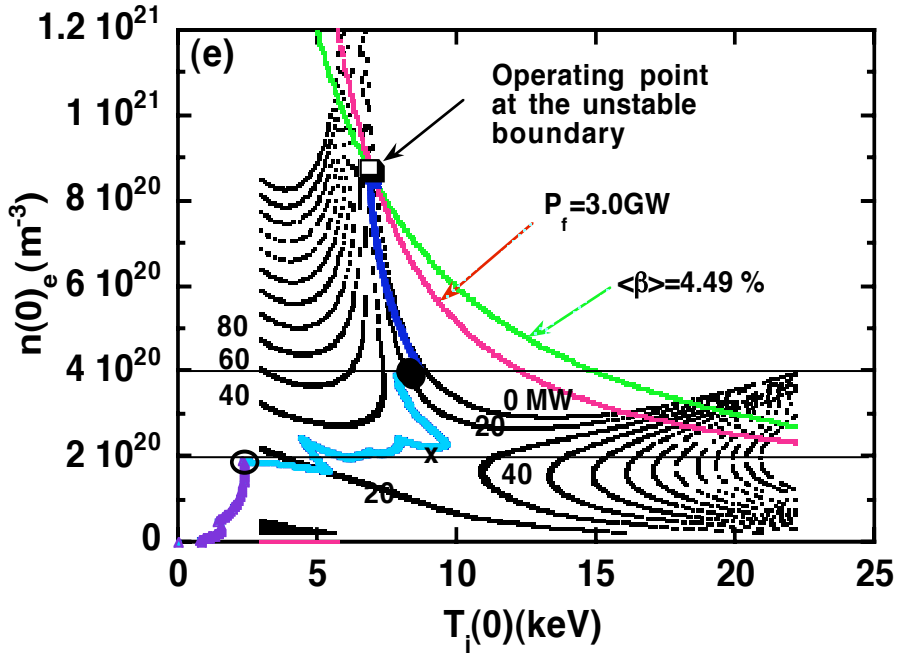
Control TGL: Mitarai

1. Fusion power is carefully controlled to keep the density at the low around 2～3x10^{20} m^{-3} to ensure the NBI penetration.



Temporal evolutions of plasma parameters. After switching off NBI, ignition is accessed by alpha heating.

γ_{ISS}=1.43, B=4.5T, α_n=3, α_T=1.0, E=1.5MeV TanNBI, n(0)=8.40x0^{20} m^{-3}, T(0)=7.1 keV, <β>=4.5 %, P_{div}=12.1MW/m^2(90°Δ=0.1m)



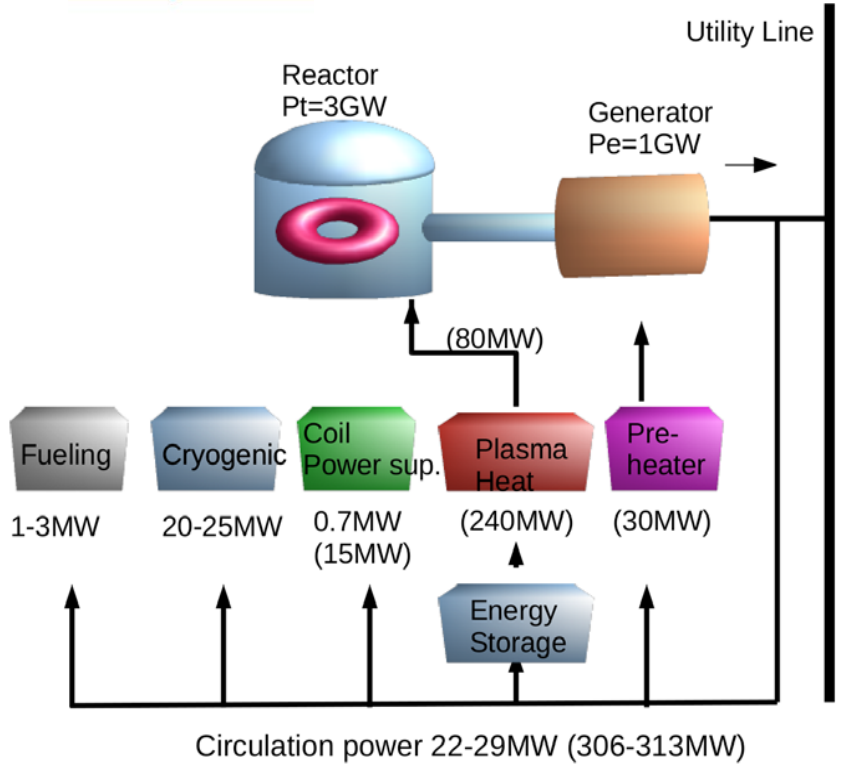
O. Mitarai al. IAEA-FEC, (2010) FTP/P6-19.

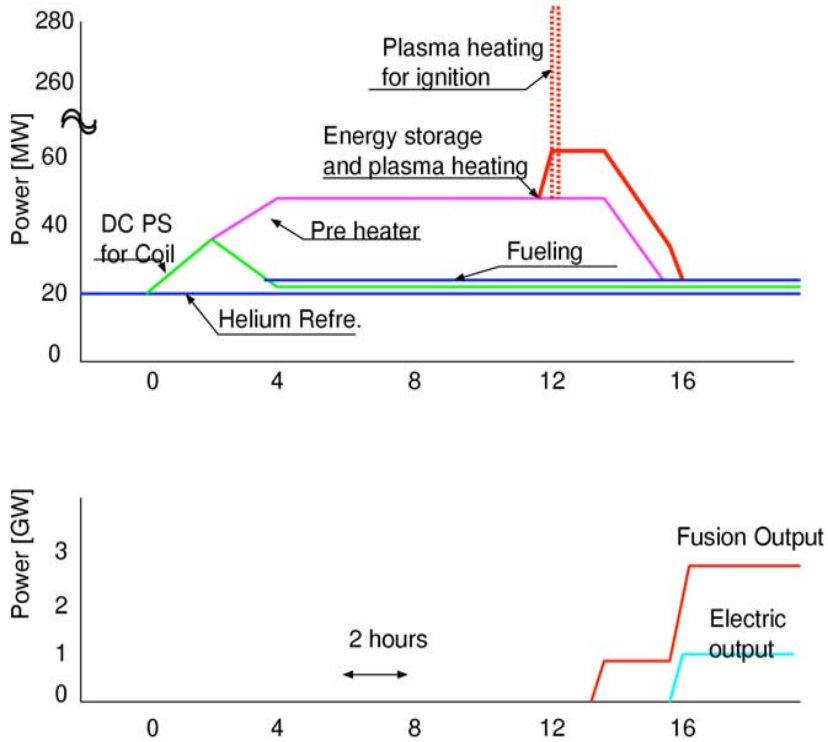
Operation path on POPCON. NBI is turned off at black point.

Operation scenario of helical type plant

Power Sys. TGL: Chikaraishi

1. The electrical power flow of the FFHR plant, which includes sub-systems, was built.
2. Start-up procedures of the FFHR as an energy plant were studied and compared.





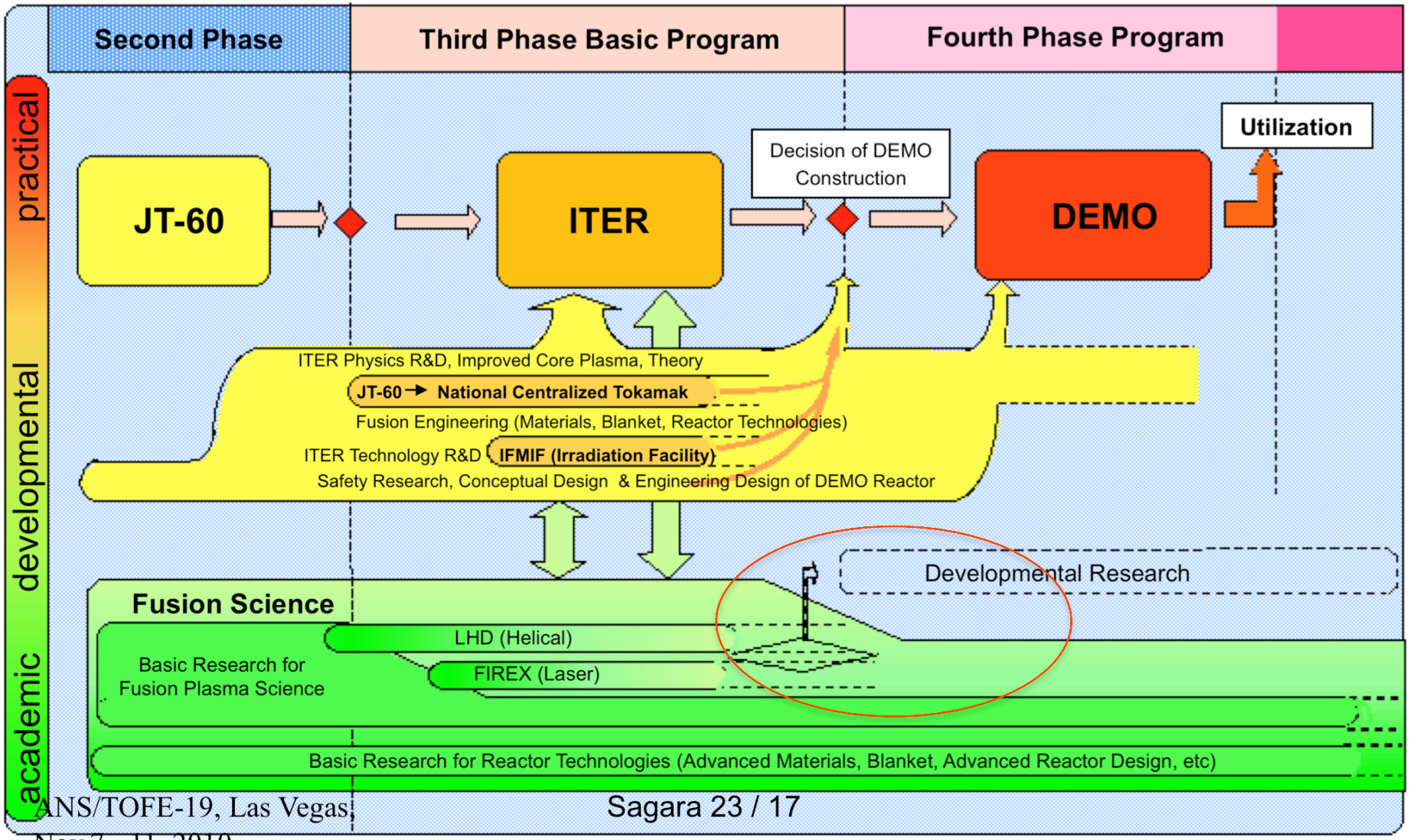
Conceptual diagram of power plant including sub-systems was made.

Start-up procedures were planned and compared.

Japan's Roadmap for fusion DEMO Reactor

Strategy for Fusion Development (Atomic Energy Commission, 2005)

http://www.aec.go.jp/jicst/NC/senmon/kakuyugo2/siryo/kettei/houkoku051026_e/betten20.htm



Concluding remarks

Main advantages of helical DEMO are

1. Current-free plasma for steady, CD-free & disruption-free,
2. High density operation with large reduction of divertor load,
3. Wide range of start-up time to minimize ex-heating power,
4. Natural or island divertor.

Key issues are

1. Reactor size optimization for blanket space and mag. energy,
2. Large SC magnet system of 3D helical continuous winding,
3. Large maintenance ports and replacement scenarios,
4. R&D's of 3D component mockup studies with design activities.