US-Japan workshop on Fusion Power Reactor Design
and Related Advanced Technologies, March 5-7 2008 at UCSD.

Overview of Design Integration toward Optimization of LHD-type Fusion Reactor FFHR

Presented by Y. Kozaki

Akio Sagara, O. Mitarai[1], S. Iamagawa, Y. Kozaki, T. Watanabe, T. Tanaka, N. Yanagi, K. Takahata, H. Tamura, H. Chikaraishi, S. Yamada, T. Mito, A. Shishkin[2], Y. Igitkhanov[3], H. Igami, T. Morisaki, M. Kobayashi, R. Sakamoto, H. Yamada, O. Motojima and FFHR design group

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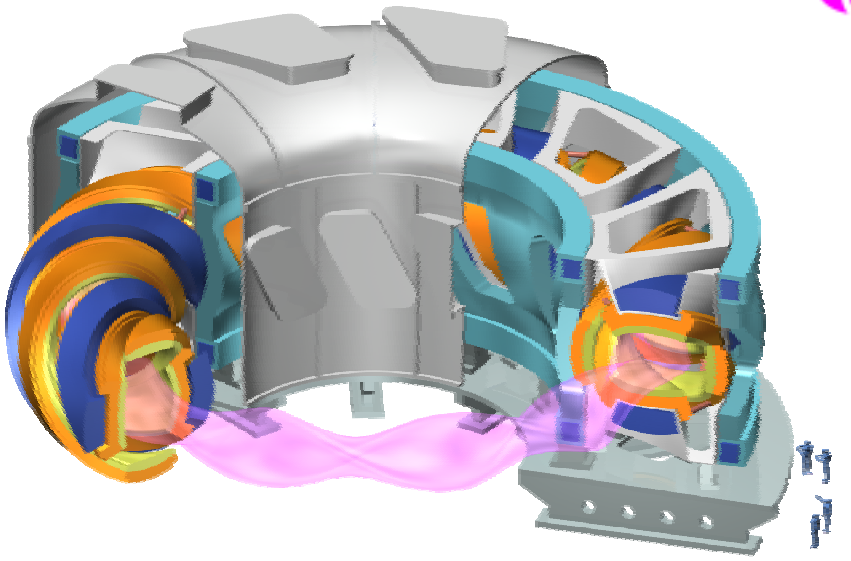
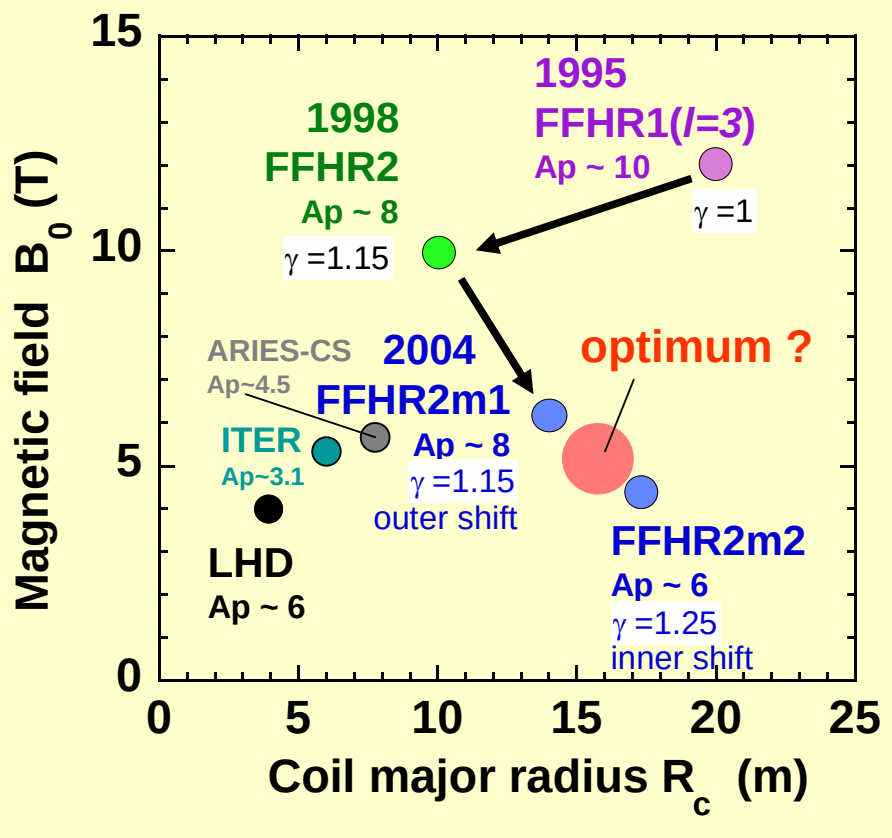


FFHR
2 GWth
6 Tesla
25,000 ton

Presentation outline
toward design optimization

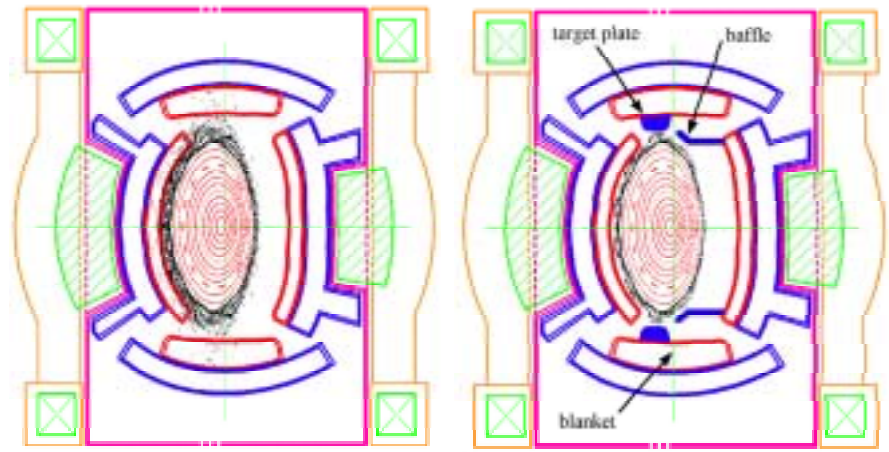
1. Reactor size and cost
2. Minimization of heating power
3. High density ignition and control
4. Large SC magnet system
5. Access to Demo
6. Concluding remarks

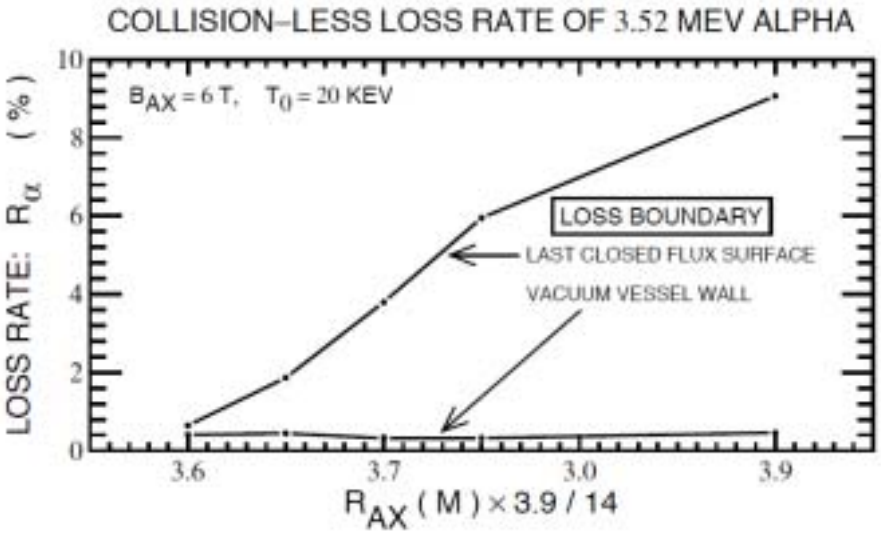


Reactor size optimization w/o increasing neutron wall loading in FFHR

Ergodic layer is important for alpha-heating efficiency > 90 %,

Helical X-point Divertor (HXD) *T. Morisaki et al., FED 81 (2006) 274*

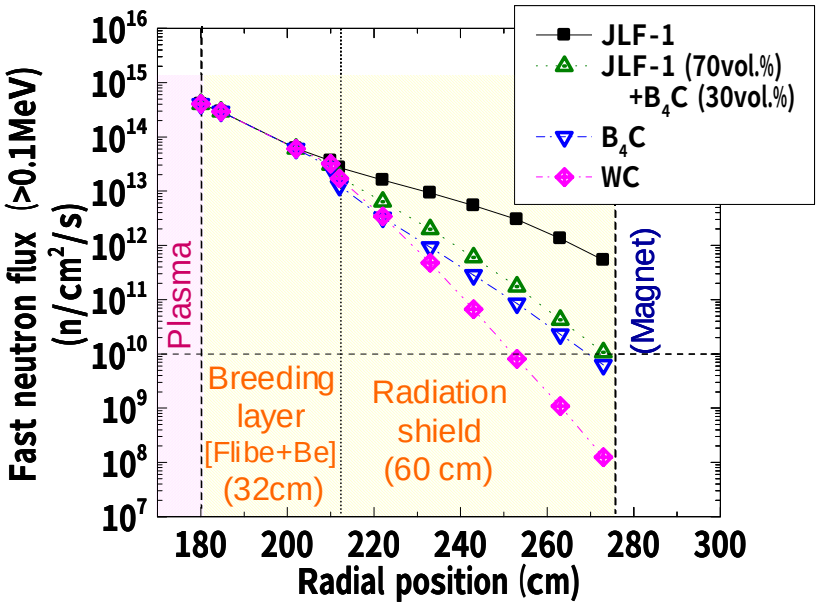




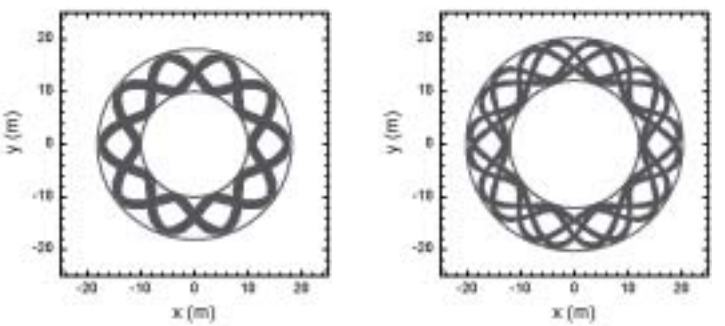
by T. Watanabe

Three candidates are proposed to increase blanket space > 1.1 m

(1)
Reduction of the inboard shielding thickness using WC



(2)
Improvement of the symmetry of magnetic surfaces
by increasing the current density at the inboard side of the helical coils



LHD FFHR-2S

by splitting the helical coils.

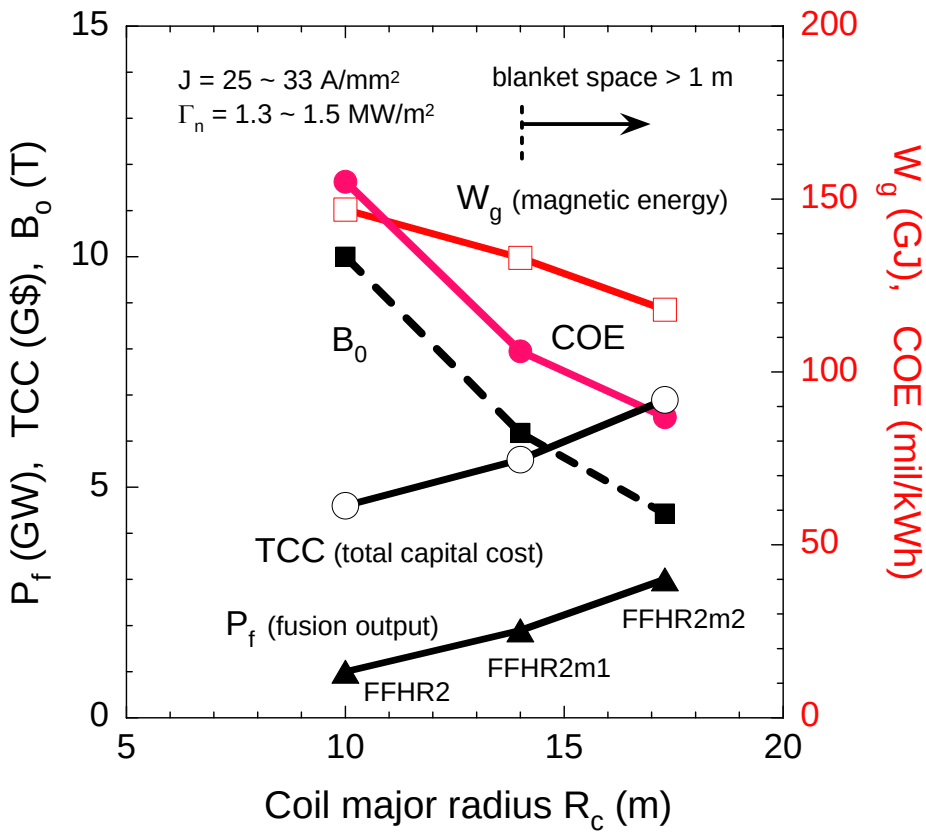
N. Yanagi et al., in this conference.

Three candidates are proposed to increase blanket space > 1.1 m

Design parameters			LHD	FFHR2	FFHR2m1	FFHR2m2
Polarity	l		2	2	2	2
Field periods	m		10	10	10	10
Coil pitch parameter	γ		1.25	1.15	1.15	1.25
Coil major Radius	R_c	m	3.9	10	14.0	17.3
Coil minor radius	a_c	m	0.98	2.3	3.22	4.33
Plasma major radius	R_p	m	3.75	10	14.0	16.0
Plasma radius	a_p	m	0.61	1.24	1.73	2.80
Plasma volume	Vp	m^3	30	303	827	2471
Blanket space	Δ	m	0.12	0.7	1.1	1.15
Magnetic field	B_0	T	4	10	6.18	4.43
Max. field on coils	B_{max}	T	9.2	14.8	13.3	13.0
Coil current density	j	MA/m^2	53	25	26.6	32.8
Magnetic energy		GJ	1.64	147	133	118
Fusion power	P_F	GW		1	1.9	3
Neutron wall load	Γ_n	MW/m^2		1.5	1.5	1.3
External heating power	P_{ext}	MW		70	80	100
α heating efficiency	η_α			0.7	0.9	0.9
Density lim.improvement				1	1.5	1.5
H factor of ISS95				2.40	1.92	1.76
Effective ion charge	Z_{eff}			1.40	1.34	1.35
Electron density	$n_e(0)$	10^19 m^{-3}		27.4	26.7	19.0
Temperature	$T_i(0)$	keV		21	15.8	16.1
Plasma beta	<β>	%		1.6	3.0	4.1
Plasma conduction loss	P_L	MW			290	463
Diverter heat load	Γ_{div}	MW/m^2			1.6	2.3
Total capital cost		G$(2003)		4.6	5.6	6.9
COE		mill/kWh		155	106	87

(Cost evaluation based on the ITER (2002) report)

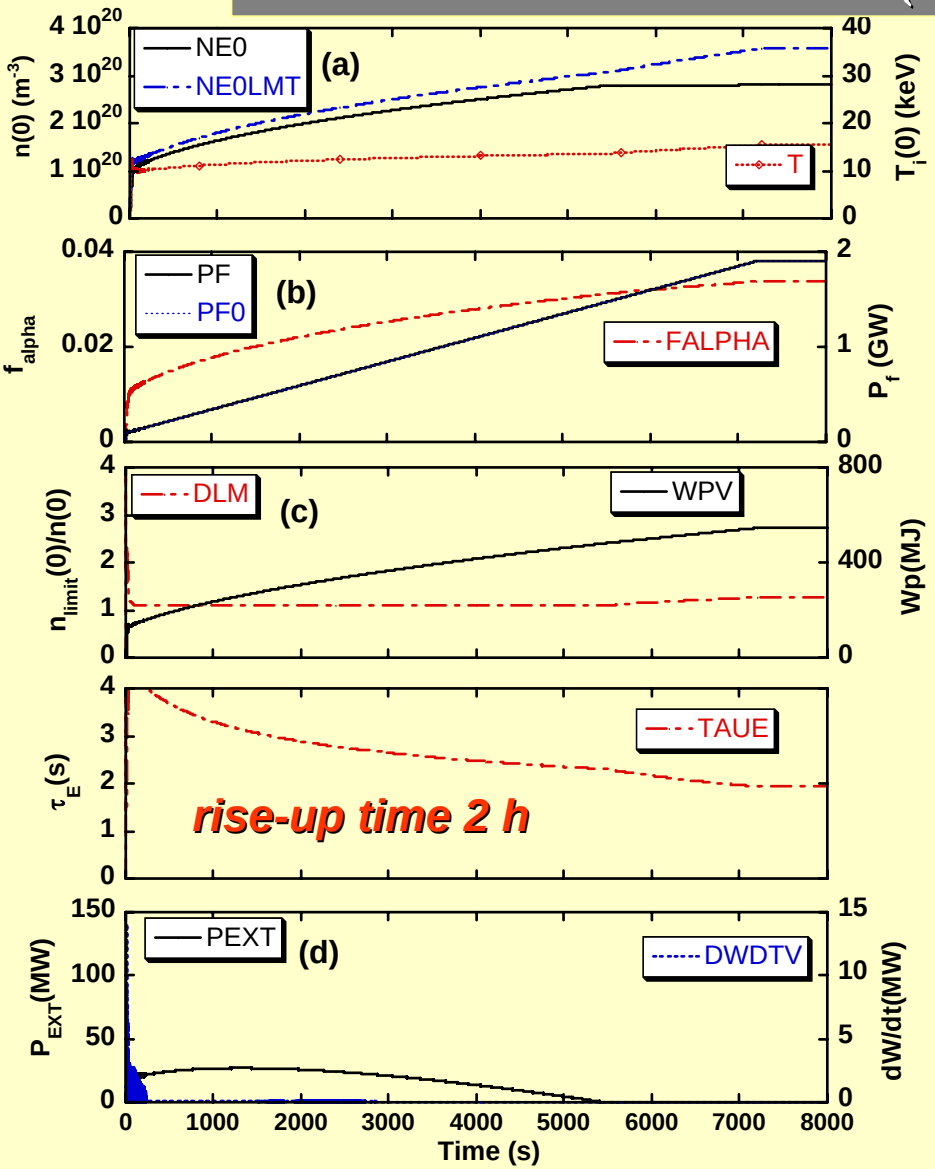
(3) Optimization of reactor size around *Rc* of 16 m

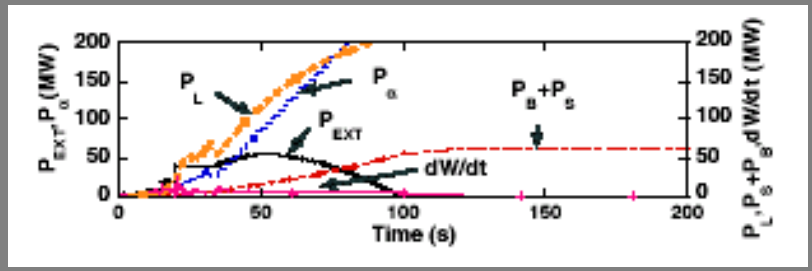


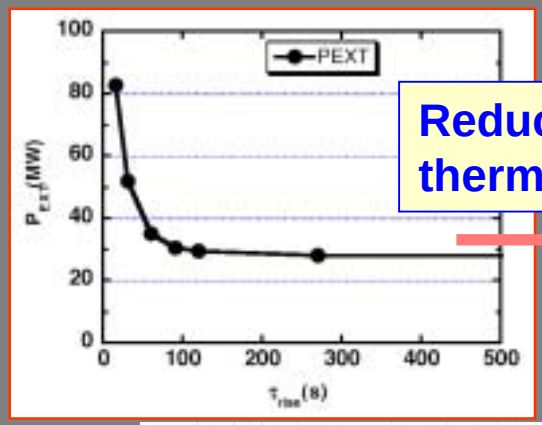
Minimized heating power ~ 30 MW

O. Mitarai al. Nucl. Fusion 47 (2007) 1411.

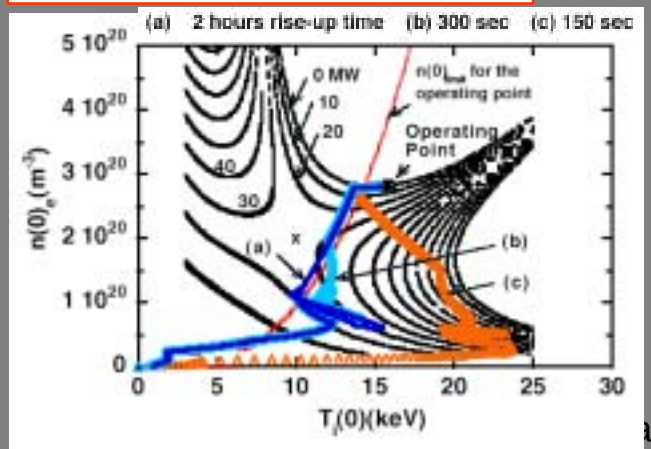
$$P_{EXT} + P_{\alpha} = \frac{dW}{dt} + (P_L + P_B + P_S)$$







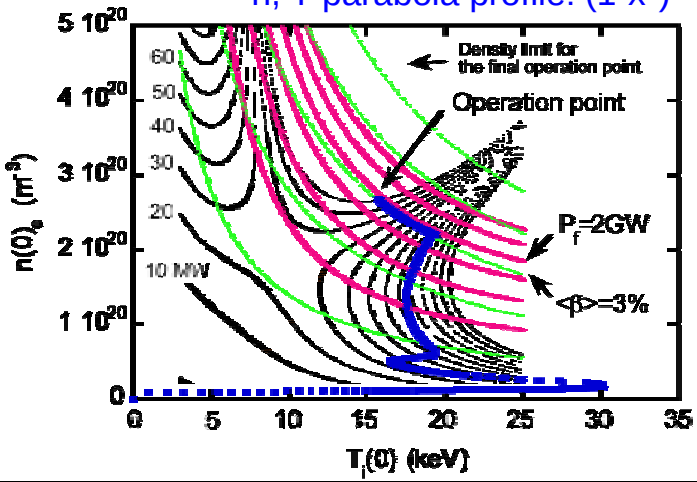
Reduction of thermal stress



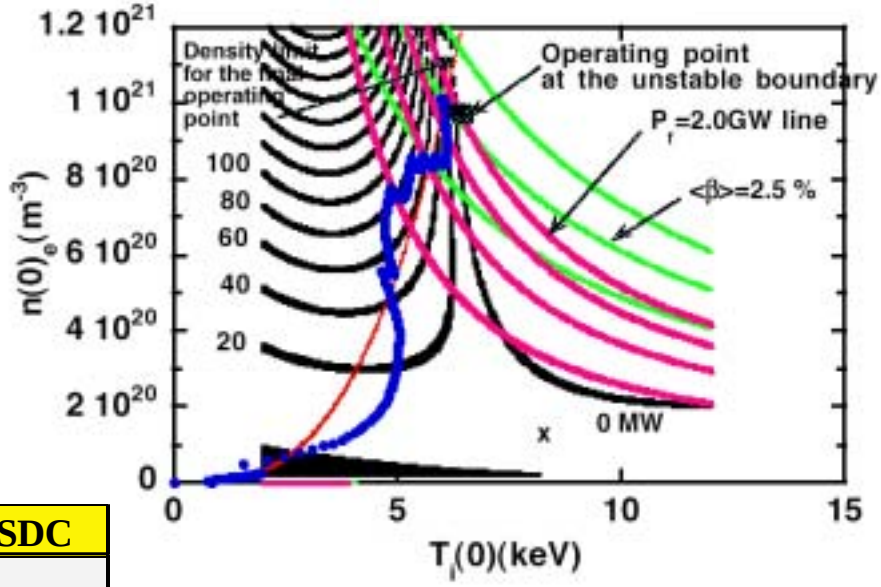
IDB Ignition Scenario with SDC

Conventional design

n, T parabola profile: $(1-x^2)$

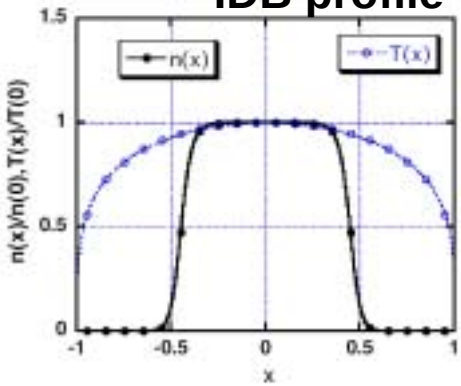


IDB scenario



Design parameters			FFHR2m1	SDC
Fusion power	P_F	GW	1.9	
Density lim.improvement			1.5	7.5
H factor of ISS95			1.92	
Electron density	$n_e(0)$	10^20 m^{-3}	2.4	9.8
Temperature	$T_i(0)$	keV	15.8	6.27
Effective ion charge	Zeff		1.48	1.52
Plasma beta	<β>	%	3.0	2.5
Energy confinem. time	τ_E	s	1.9	4.7
Bremsstrahlung loss	P_B	MW	57	248
Plasma conduction loss	P_L	MW	282	96
Heat load to first wall	Γ_{div}	MW/m^2	0.06	0.25
Heat load to divertor	Γ_{div}	MW/m^2	1.6	0.54

IDB profile

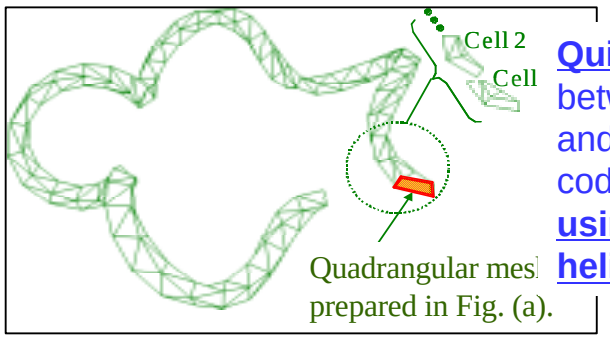


Divertor heat load is drastically reduced.

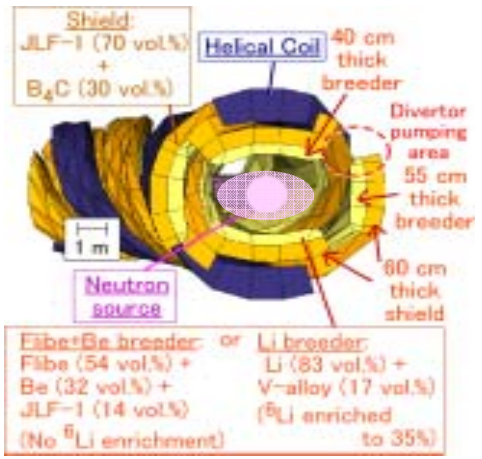
Neutron wall loading is ave. 1.5 MW/m^2 and peaking factor < 1.3

Evaluated using the recently developed 3D neutronics calculation system for non-axisymmetric helical systems

T. Tanaka et al. 21th IAEA (2006)

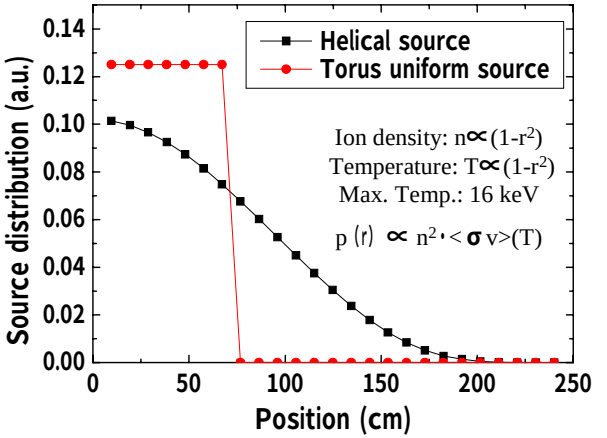


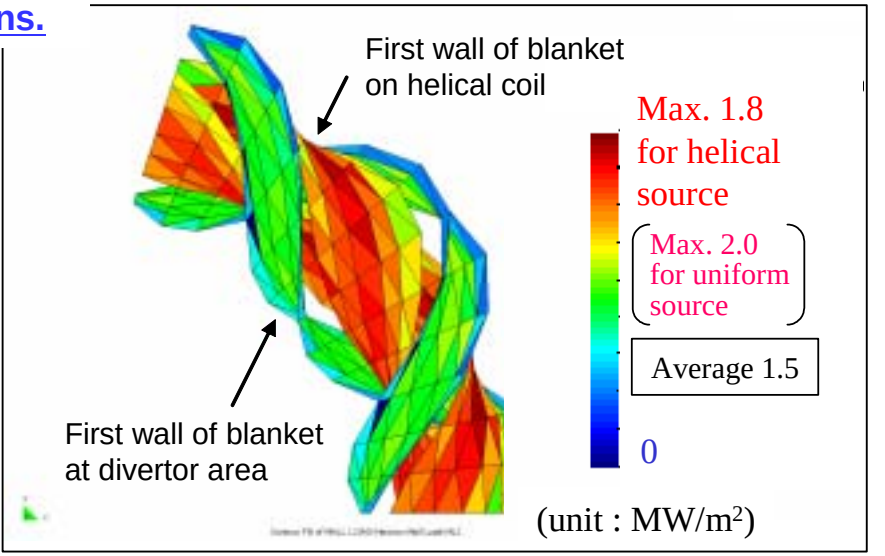
Quick feedback between 3D CAD and 3D Monte-Carlo code MCNP5, **using numerical helical equations.**



Total TBR > 1.1

Two cases of neutron sources





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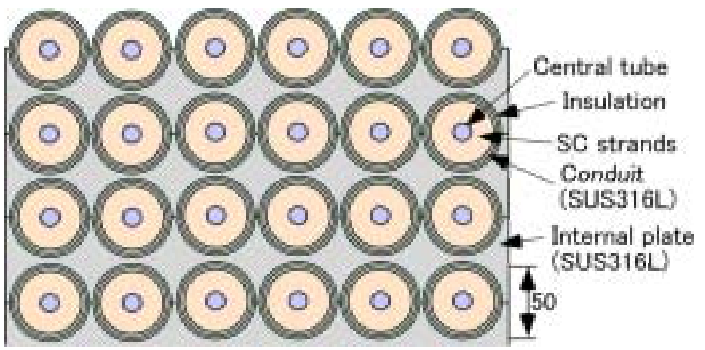
Base design of CICC magnet system

S. Imagawa et al., in this conference.

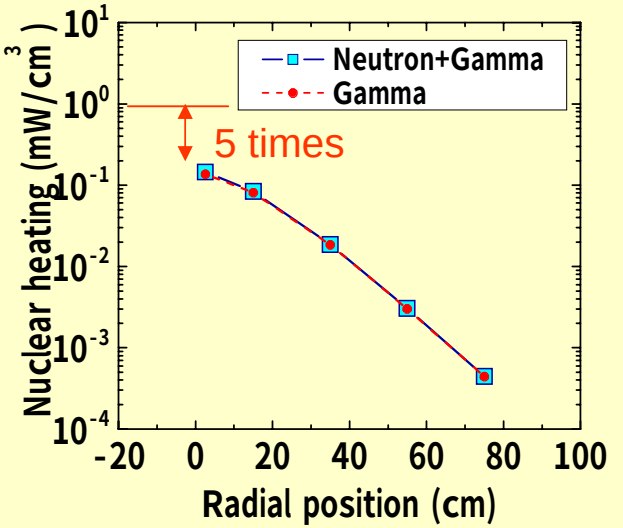
Table 1. Design criteria for CIC conductors based on ITER-TF coils.

Items	Design criteria	ITER-TF
Max. cooling length (m)	< 500	390
Current (kA)	< 100	68
Maximum field (T)	< 13	11.8
SC current density (A/mm^2)	< 300	273
Coil current density (A/mm^2)	< 30	20.3
SC material for HC	Nb3Al (*1)	Nb3Sn

(*1) "react and wind" method can be adopted by managing strain during winding within about 0.5%.



Nuclear heating in FFHR2m1

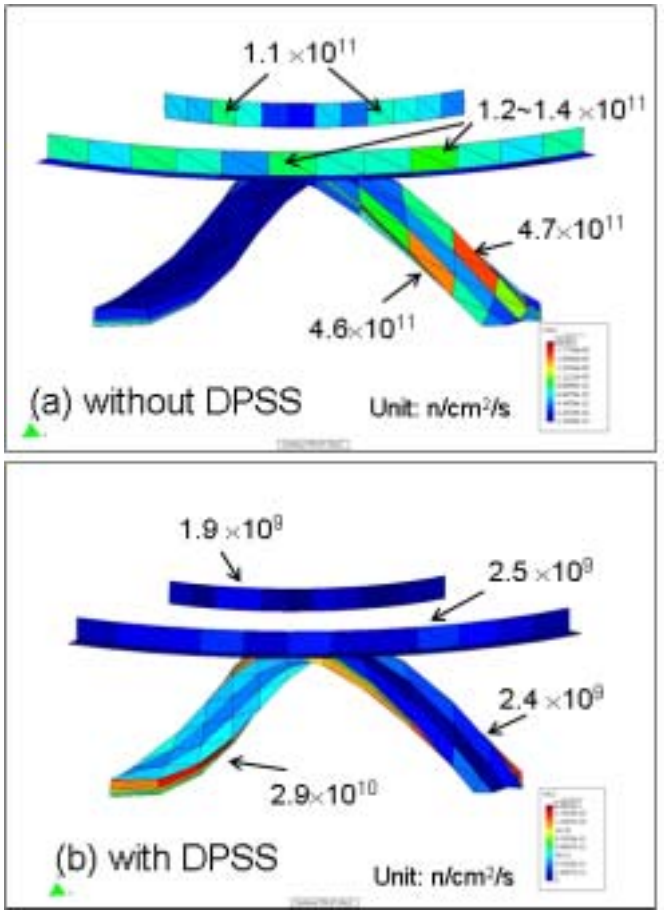


- ✓ Max. cooling path is 500 m for the nuclear heat of 1 mW/cm^3.
- ✓ This value is 5 times larger on the FFHR magnets.
- ✓ Gamma-ray heating is dominant.

Nuclear shielding of SC magnets

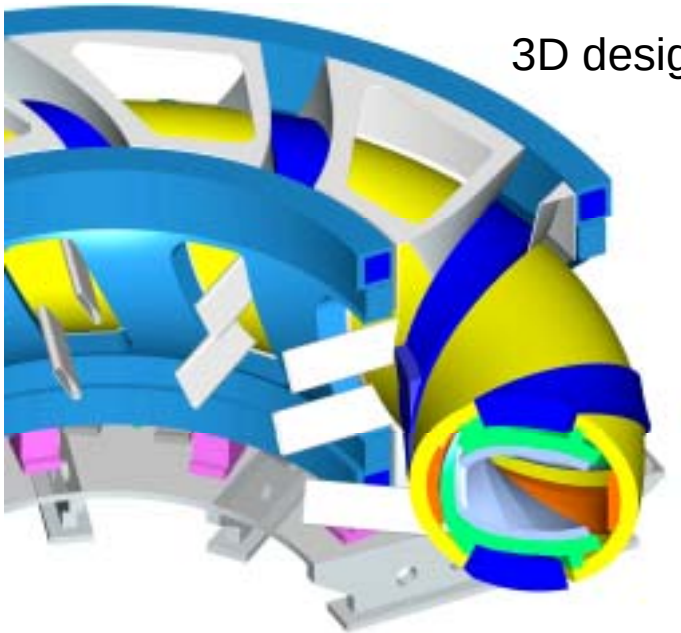
by Discrete Pumping with Semi-closed Shield (DPSS)

cover rate > 90%



Acceptable level achieved

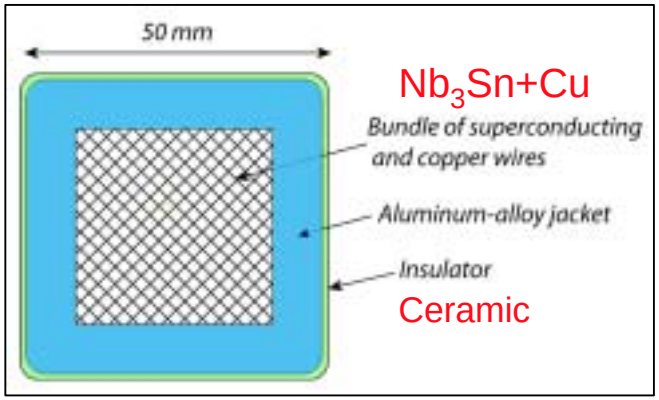
- ✓ Fast neutron < 1E22 n/m^2 in 30 years
- ✓ Max. nuclear heating < 0.2 mW/cm^3
- ✓ Total nuclear heating ~ 40 kW
  Cryogenics power ~ 12 MW (1% of P_f)

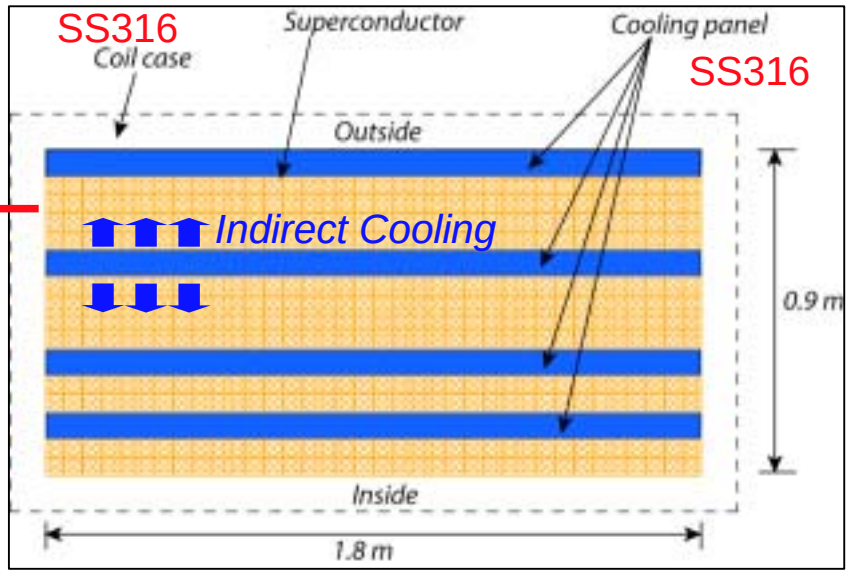


3D design with DPSS

Indirect cooling system
as an alternative with quench protection candidates

K. Takahata et al., 24th SOFT, 2006.





Cross-sectional structure of the helical coil

100 kA Superconductor

- High effective thermal conductivity
- High mechanical rigidity and strength

Quench protection by external dumping

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

Quench protection by internal dumping

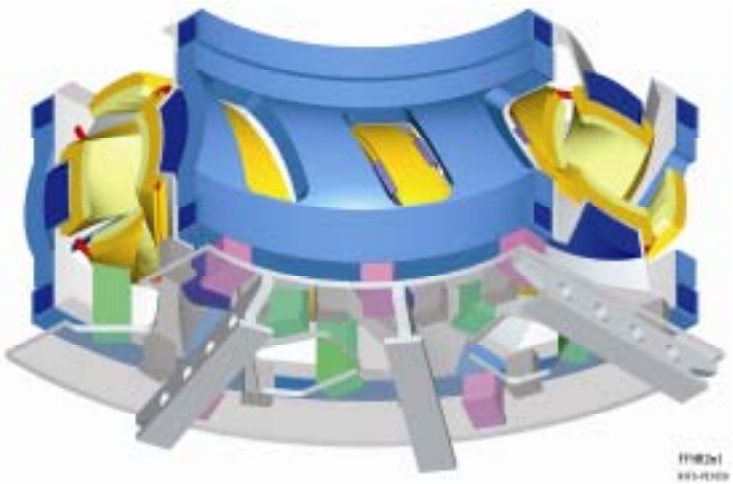
K. Takahata et al., in this conference.

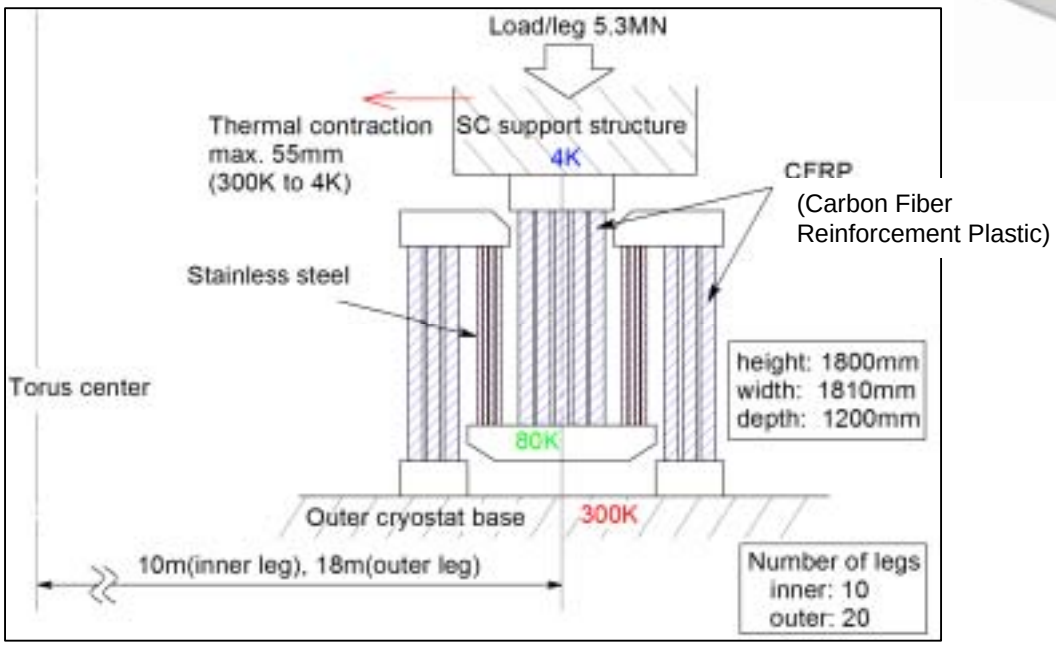
- Quench back with a secondary circuit
- to increase a decay time constant
- to reduce a transient voltage
- to avoid a serious hot spot

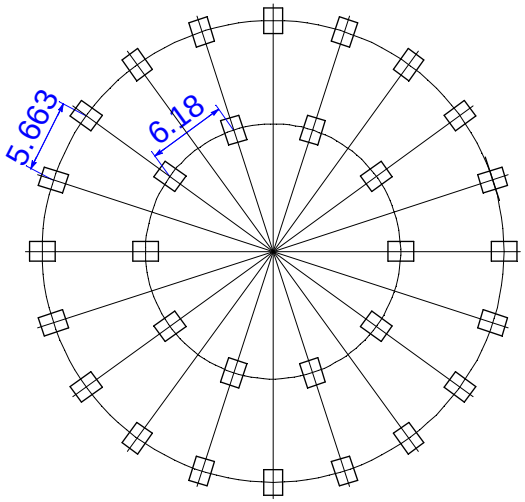
LHD-type support post for FFHR

H. Tamura et al., in this conference.

- Gravity per support
  = 16,000 ton / 30 legs ~ 530 ton.
- Thermal contraction < max. 55 mm
- Total heat load to 4K ~ 0.34 kW
  (1/20 of stainless steel post)







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Blanket space	Δ	m	0.12	1.1	
Magnetic field	B_0	T	4	6.18	
Max. field on coils	B_{max}	T	9.2	13.3	
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Magnetic energy		GJ	1.64	133	
Fusion power	P_F	GW		1.9	
Neutron wall load	Γ_n	MW/m^2		1.5	
External heating power	P_{ext}	MW		80	
α heating efficiency	η_α			0.9	
Density lim.improvement				1.5	7.5
H factor of ISS95				1.92	
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Effective ion charge	Zeff			1.48	1.52
Plasma beta	$<\beta>$	%		3.0	2.5
Energy confinem. time	τ_E	s		1.9	4.7
α ash confiment	τ_α^*/τ_E			3 < 7	
Bremsstrahlung loss	P_B	MW		57	248
Plasma conduction loss	P_L	MW		282	96
Heat load to first wall	Γ_{div}	MW/m^2		0.06	0.25
Heat load to divertor	Γ_{div}	MW/m^2		1.6	0.54
Diverter heat load	Γ_{div}	MW/m^2		1.6	0.5
Total capital cost		G$(2003)		5.6	
COE		mill/kWh		106	

LHD experiments

External heating

High energy particle

New density limit

Enhancement of τ_E

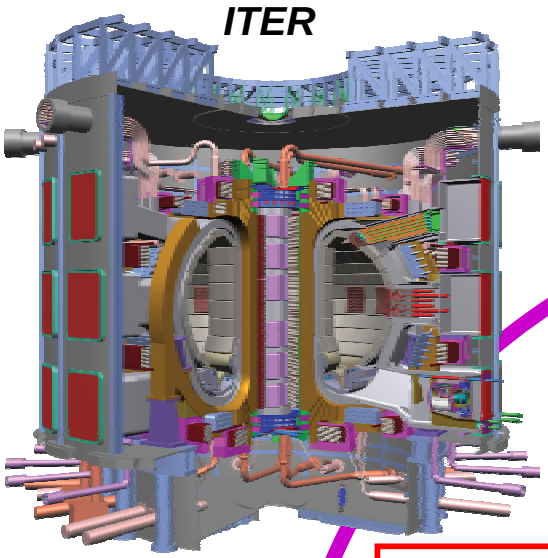
Impurity shielding

He exhaust

Role of Design Study to Helical Demo-Reactor based on LHD Project

Tokamak Experimental Rector

ITER



LHD-type Helical Reactor FFHR

Electric Power 1GW
Weight 25,000ton
Magnetic Field 6T

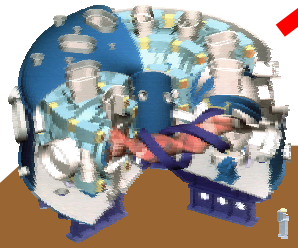
Helical Demo Rector (29 years to go)

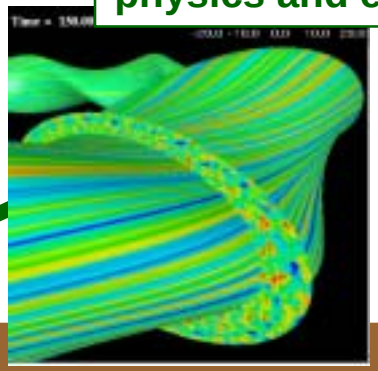
Physics of burning plasmas

Multi-layer models covering physics and engineering

Demonstration of steady-state, high-density, high beta by net-current free plasma

LHD





LHD-NT
LHD Numerical Test Reactor

Basic Science

Concluding remarks

1. Helical reactor is superior in steady state operation and
   - Reduced neutron wall loading by optimization of large reactor size,
   - Minimized heating power by long access time to ignition,
   - High density operation with reduced heat load on divertor.
2. Large SC magnet system is conceptually feasible.
3. Helical reactor is economically comparable to Tokamak.
4. Numerical Test Reactor is planned to Helical Demo.
5. Large sized construction is important R&D issue.
6. LHD experiments can open new reactor regimes.