Design Integration toward Optimization of LHD-type Fusion Reactor FFHR

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National Institute for Fusion Science, Japan

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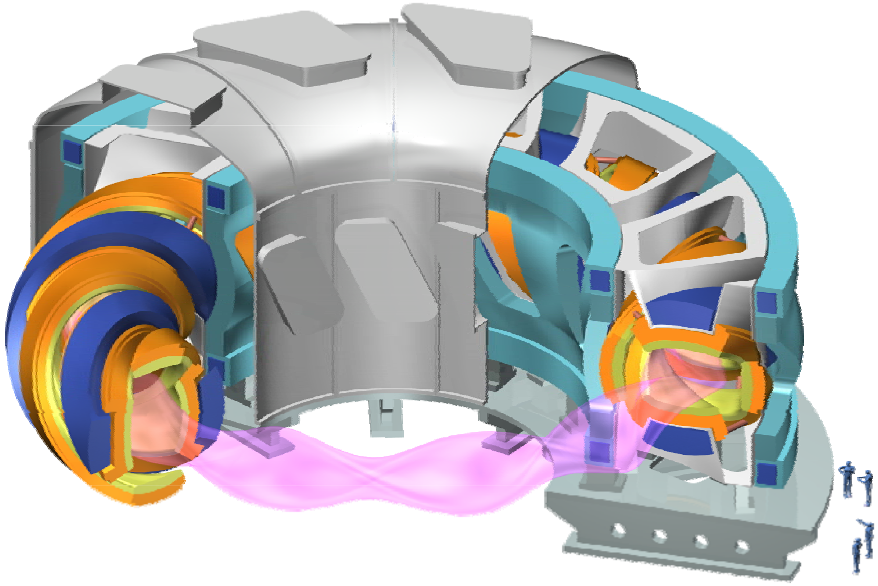
[3]Max-Planck-Institut fur Plasmaphysik, Germany

FFHR
2 GWth
6 Tesla
25,000 ton



FFHR design collaborations

LHD

Confinement scaling
H.Yamada, Miyazawa (NIFS)

Helical core plasma
K.Yamazaki(NIFS)

Magnetic structure
T.Morisaki, Yanagi (NIFS)

SC magnet & supprt
S.Imagawa
T.Mito
Takahata,
Yamada,
Tamura,
(NIFS)

Virtual Reality tool
N.Mizuguchi (NIFS)

Power supply
H.Chikaraishi (NIFS)

External heating
O.Kaneko, Igami (NIFS)

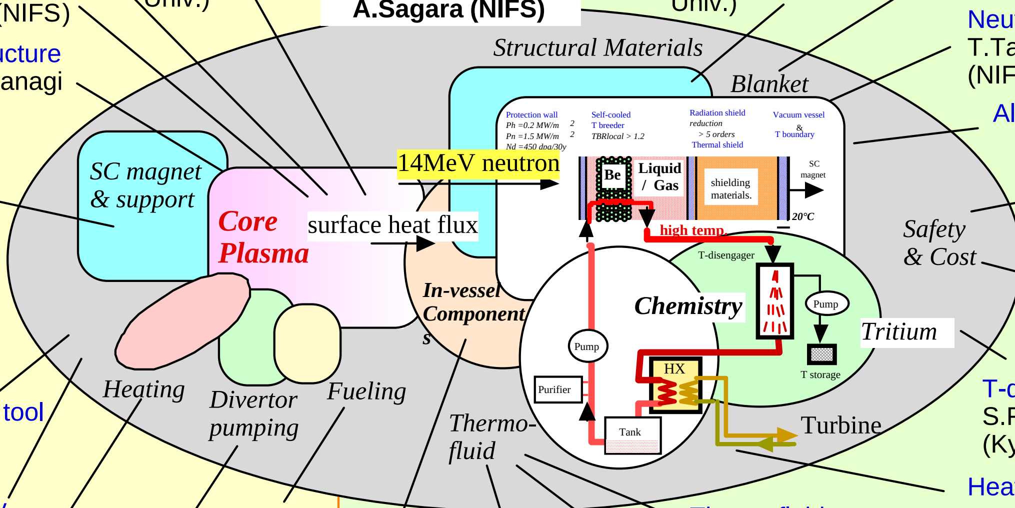
Ignition access & heat flux
O.Mitarai (Kyusyu Tokai Univ.)

Fueling
Sakamoto (NIFS)

Divertor pumping
S.Masuzaki, Kobayashi (NIFS)

FFHR design / Sytem Integration / Replacement
A.Sagara (NIFS)

Fusion Engineering



Thermo-mechanical analysis
H.Matsui (Tohoku Univ.)

Blanket system
T.Terai (Univ. of Tokyo)

Neutronics
T.Tanaka, Sagara (NIFS)

Alternative blanket
T.Muroga (NIFS)

Safety
T. Uda (NIFS)

Cost evaluation
Y. Kozaki (NIFS)

T-disengager system
S.Fukada, M.Nishikawa (Kyusyu Univ.)

Heat exchanger & gas turbine system
A.Shimizu (Kyusyu Univ.)

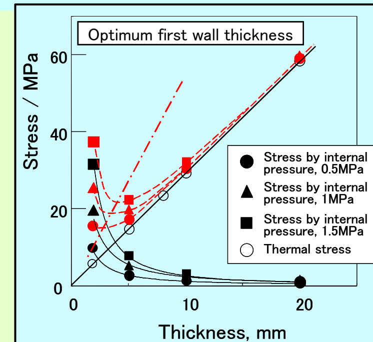
Thermofluid system
K.Yuki, H.Hashizume (Tohoku Univ.)

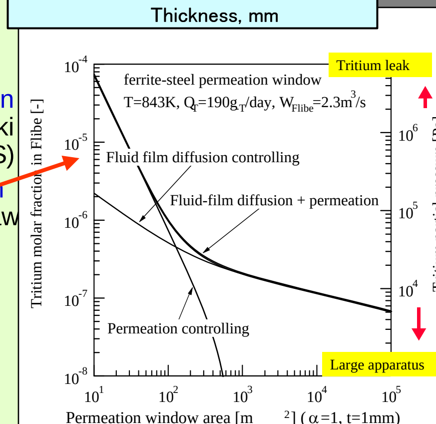
Thermofluid MHD
S.Satake (Tokyo Univ. Sci.)

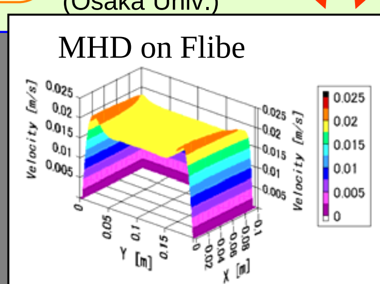
Advanced thermofluid
T.Kunugi (Kyoto Univ.)

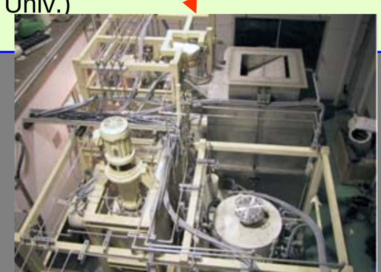
Advanced first wall
T.Norimatsu (Osaka Univ.)

Thermo-mechanical analysis

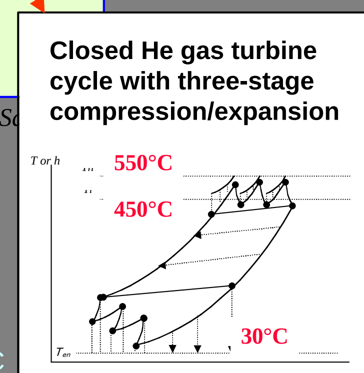








TNT Loop : Max.20L/min @ 600° C



A.Sa

Presentation outline toward design optimization

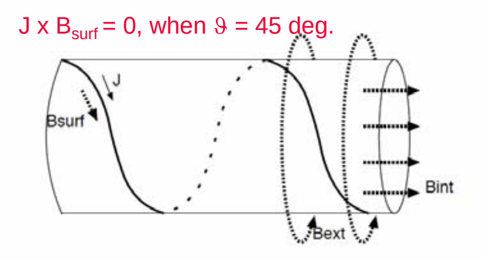
1. Reactor size and cost
2. Minimization of heating power
3. High density ignition and control
4. Large SC magnet system
5. Access to Demo
6. Concluding remarks

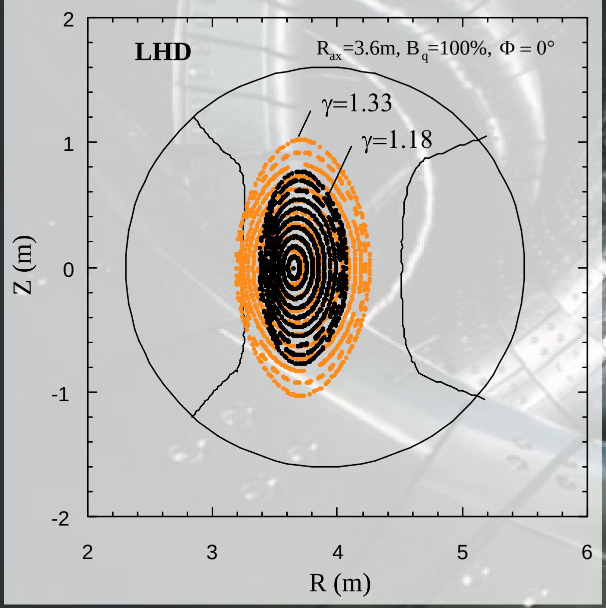
Two main features in FFHR

(1) Quasi-force free γ optimization on continuous helical winding

- to reduce the magnetic hoop force (Force Free Helical Reactor: FFHR)
- to expand the blanket space

$$\gamma = \left(\frac{m}{l} \frac{a_c}{R} \right)$$





γ=tanθ

(2) Self-cooled liquid Flibe (BeF_2-LiF) blanket

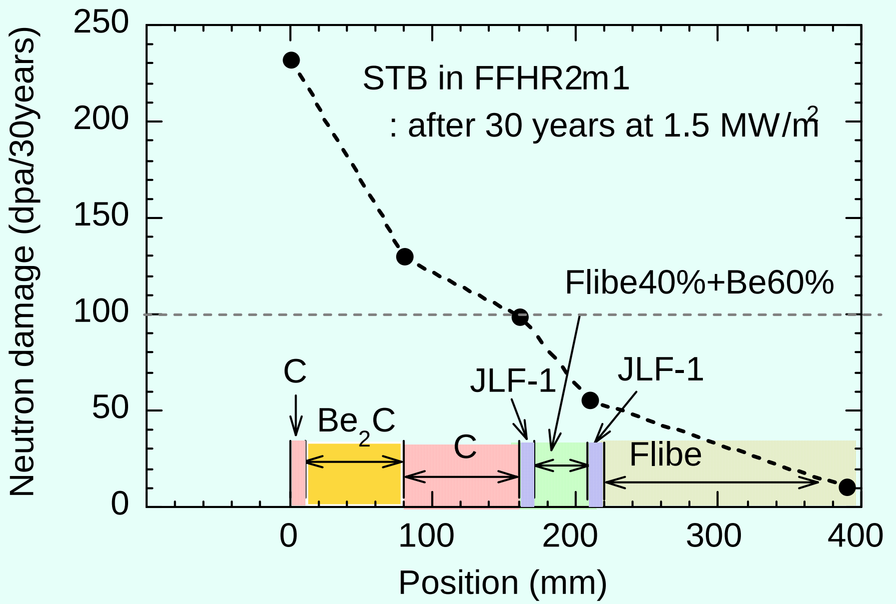
- low MHD pressure loss
- low reactivity with air
- low pressure operation
- low tritium solubility

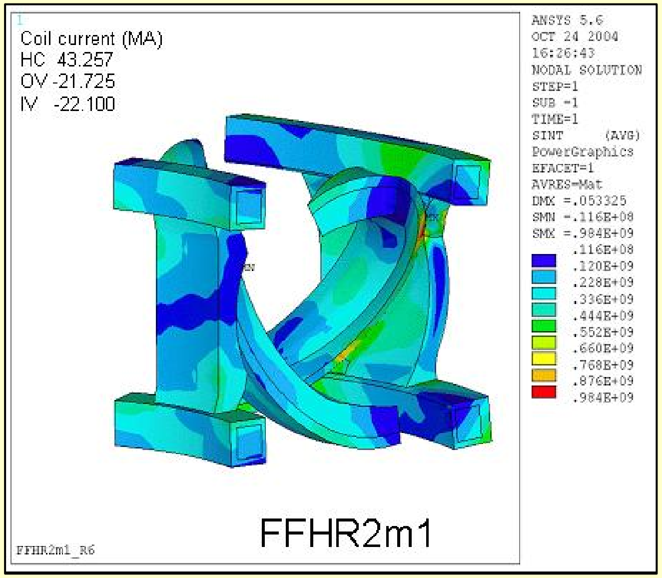
Long life blanket and wide maintenance ports

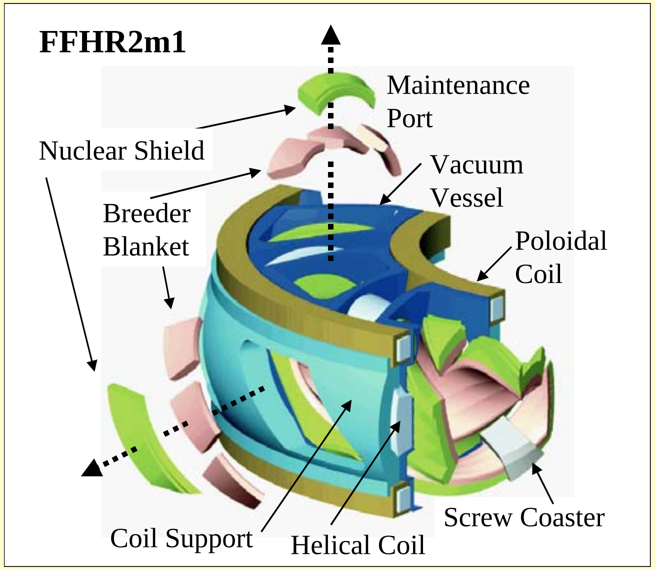
1. **Simplified support structure with wide maintenance ports due to FF concept.**
2. **Neutron wall loading ~ 1.5 MW/m^2 is a key for long life or replacement free blanket.**

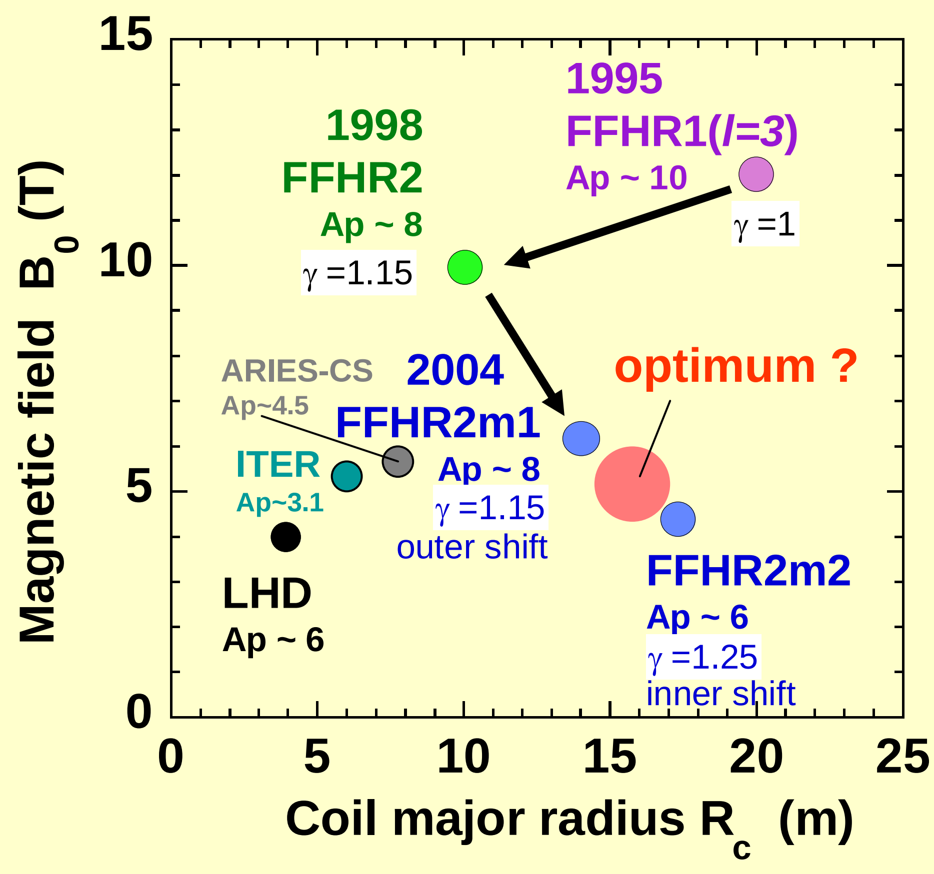
   STB (Spectral-shifter and Tritium breeder Blanket)

   A.Sagara, Nuclear Fusion 45 (2005) 258





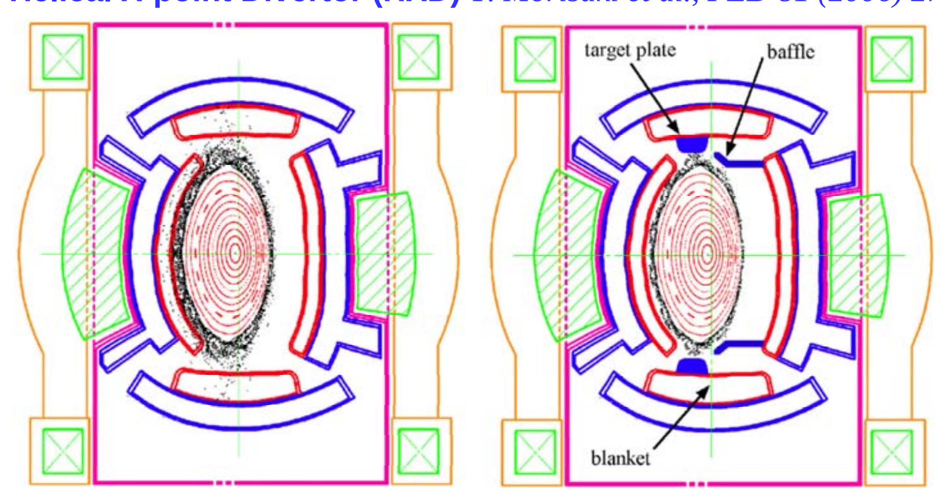


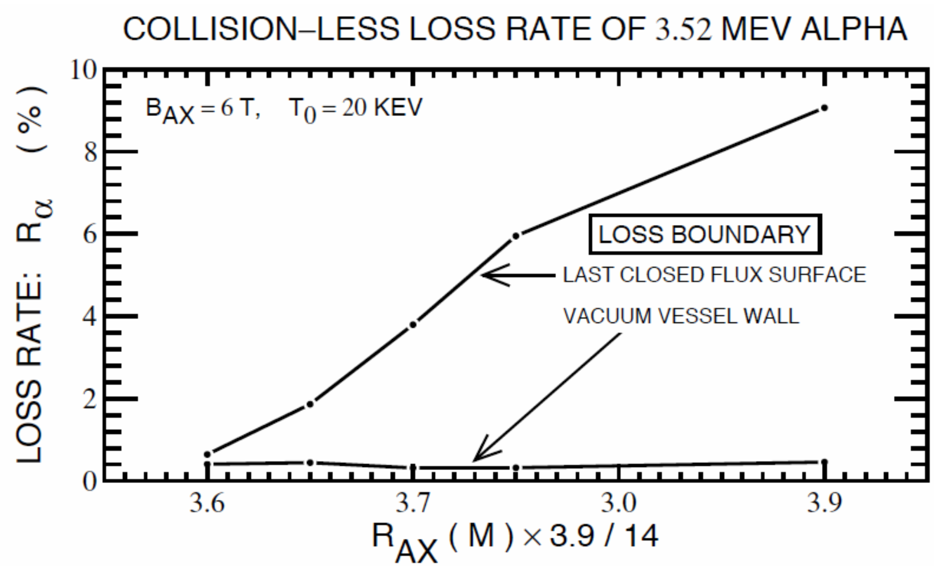


Reactor size optimization w/o increasing neutron wall loading in FFHR

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

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

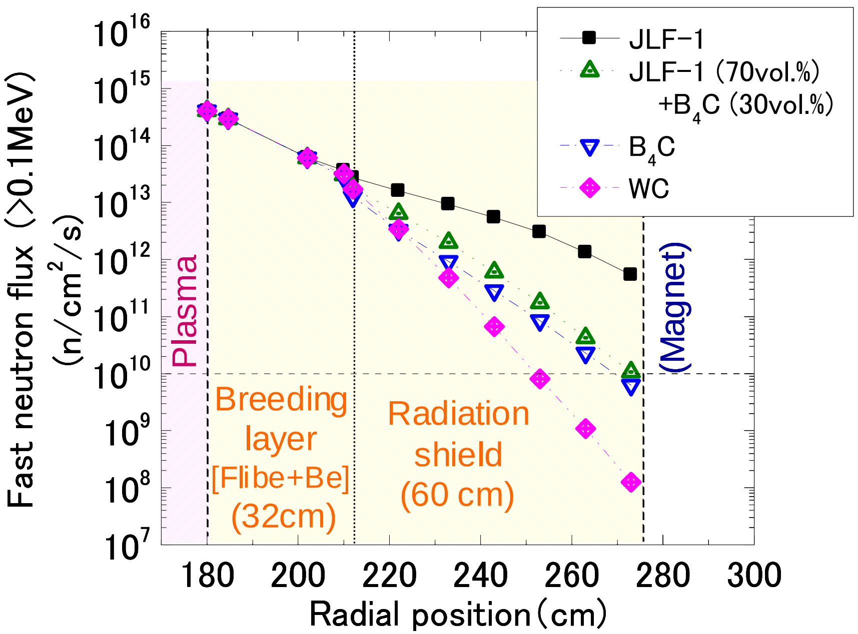




by T. Watanabe

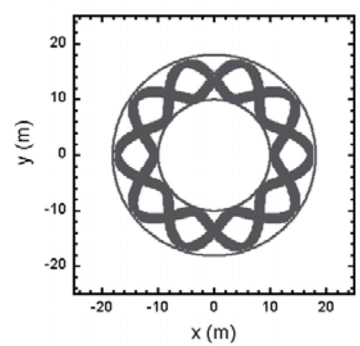
Three candidates are proposed to increase blanket space > 1.1 m

(1) Reduction of the inboard shielding thickness **using WC**

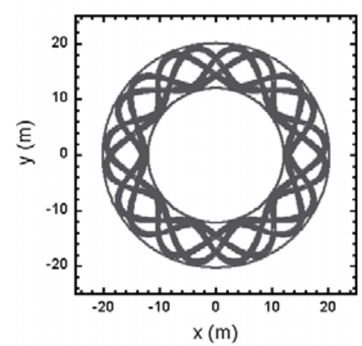


(2) Improvement of the symmetry of magnetic surfaces

<u>by increasing the current density at the inboard side of the helical coils</u>



LHD



FFHR-2S

by splitting the helical coils.

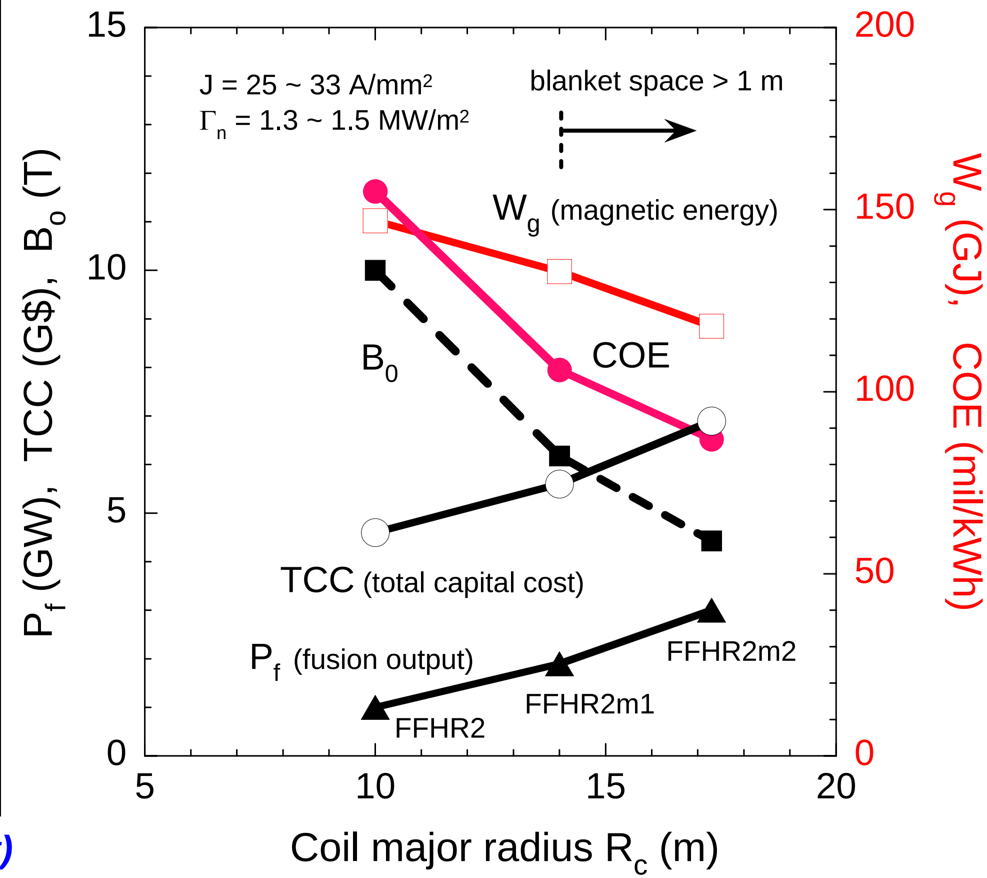
N. Yanagi et al., in this conference.

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	$\langle\beta\rangle$	%		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

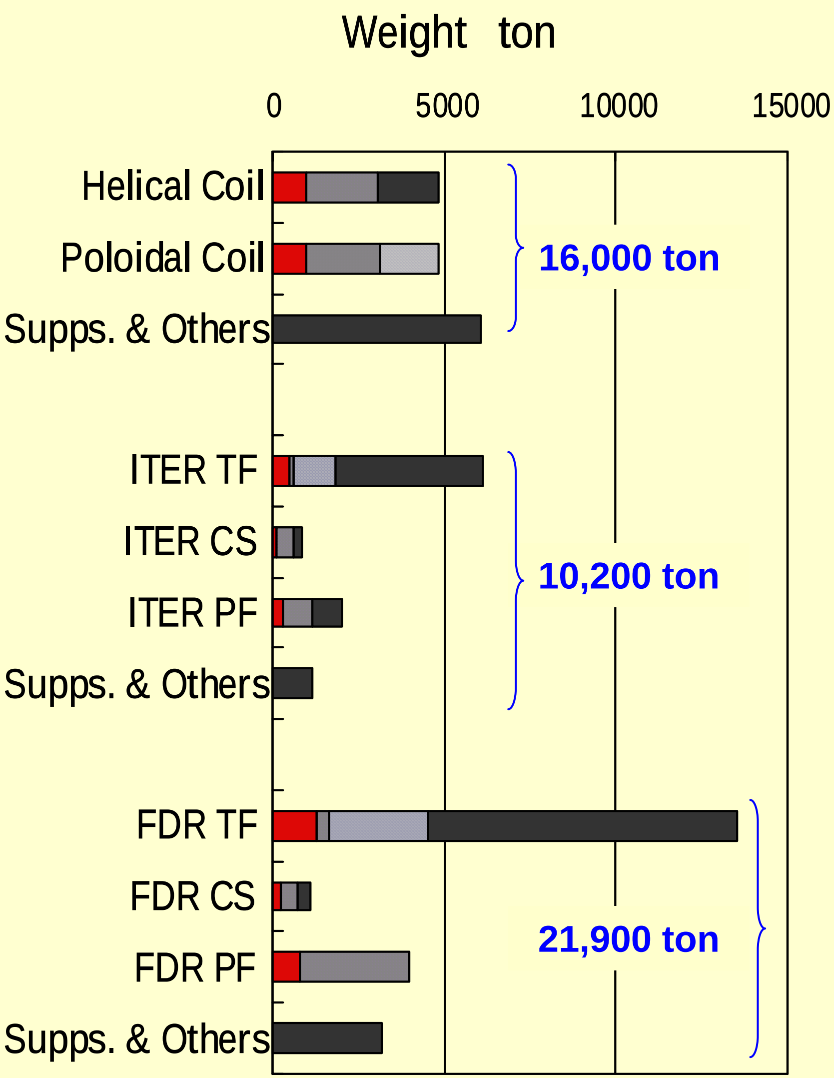
(Cost evaluation based on the ITER (2002) report)

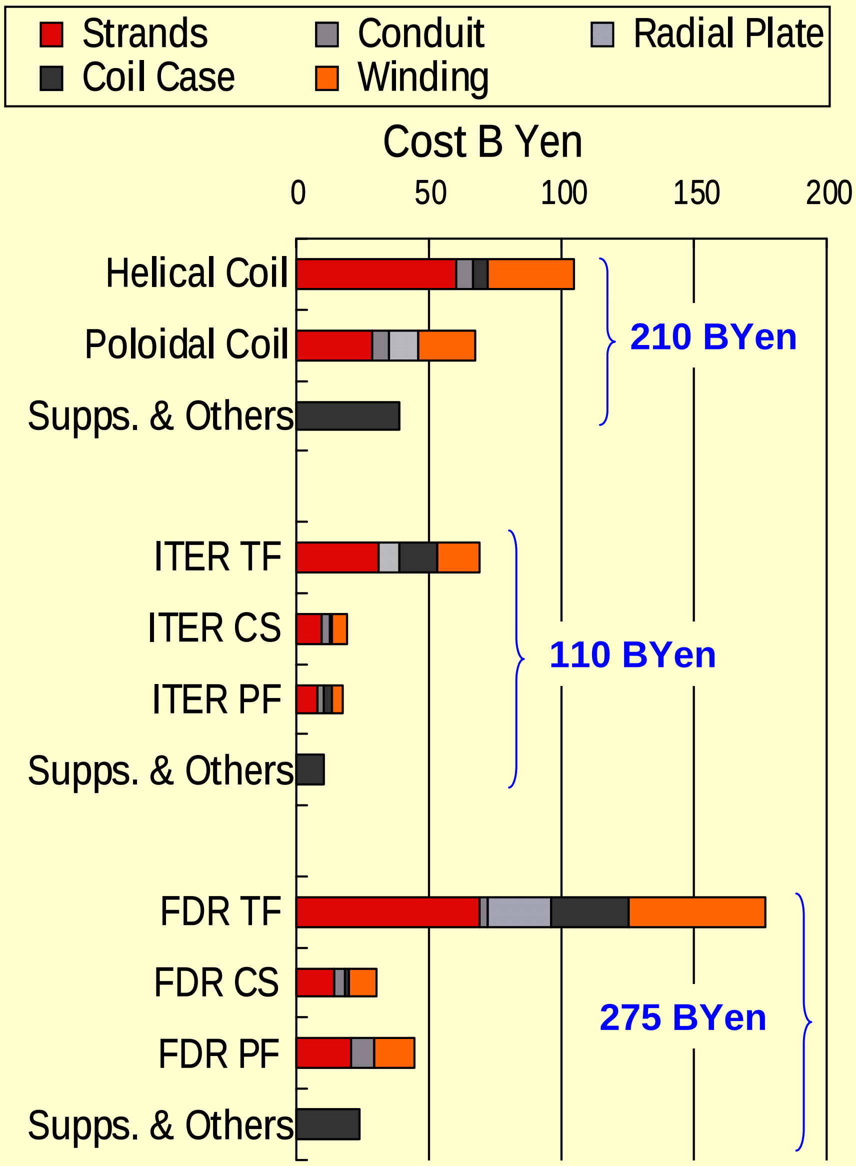
(3) Optimization of reactor size around *Rc* of 16 m



ITER based cost analyses show

Similar weights and costs between Helical and Tokamak reactor magnet systems (FFHR2m1)





Presentation outline
toward design optimization

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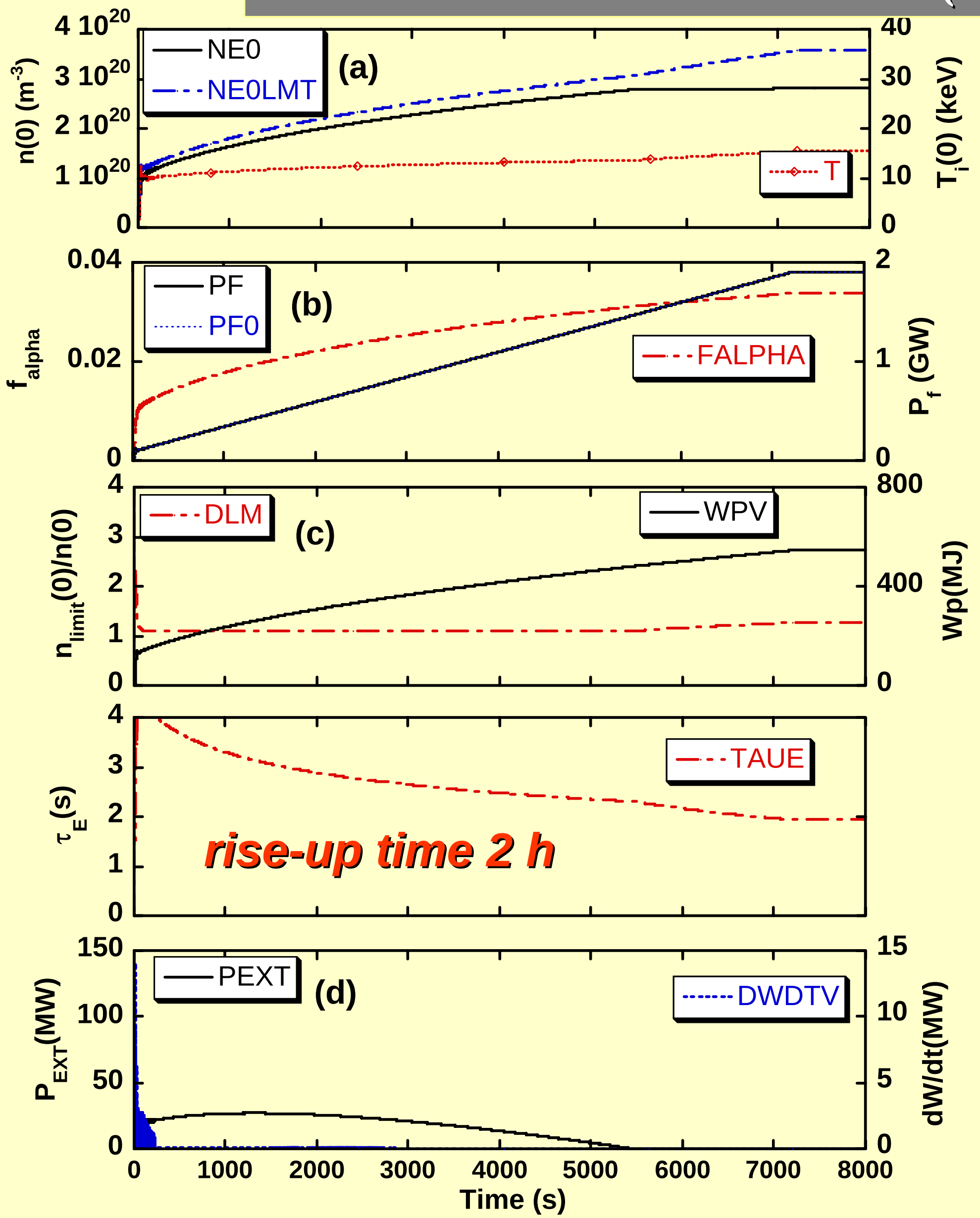
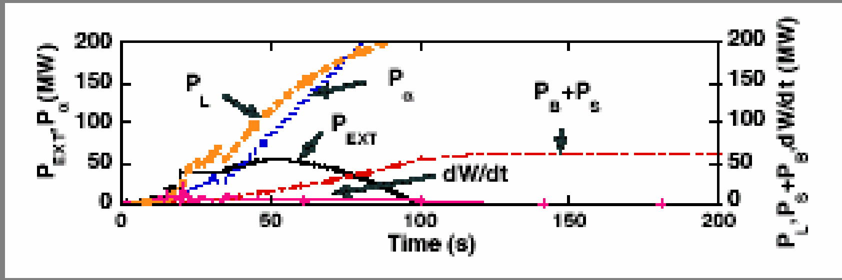
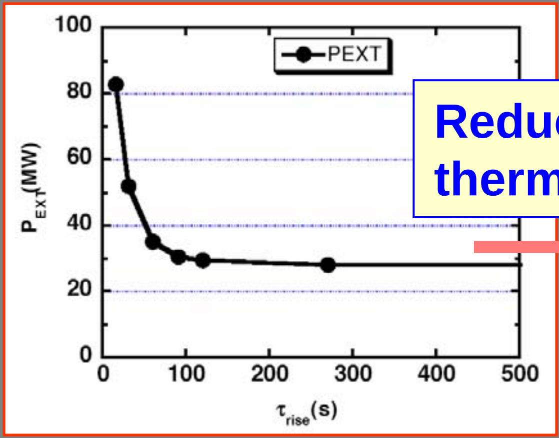
Minimized heating power ~ 30 MW

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

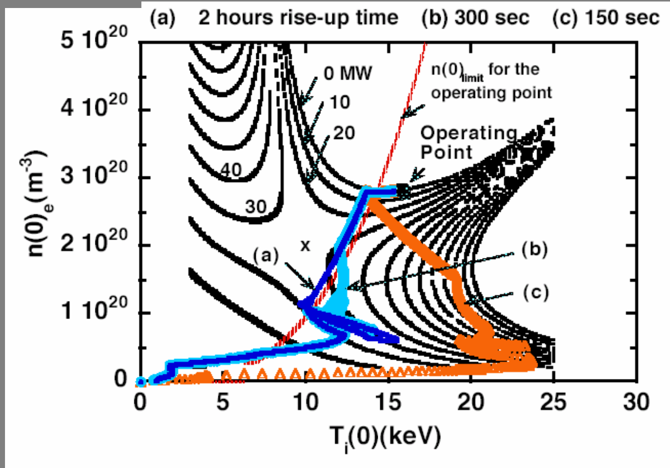
$$P_{EXT} + P_{\alpha} = \frac{dW}{dt} + (P_L + P_B + P_S)$$







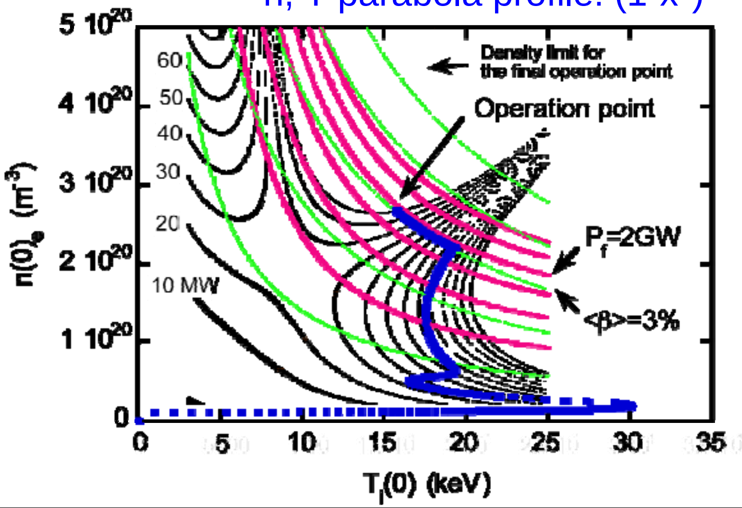
Reduction of thermal stress



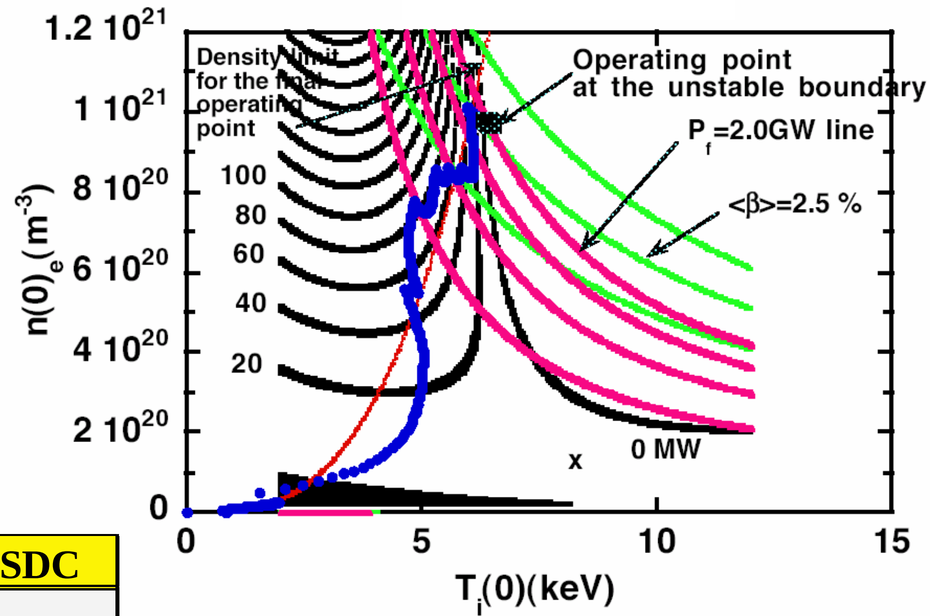
IDB Ignition Scenario with SDC

Conventional design

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

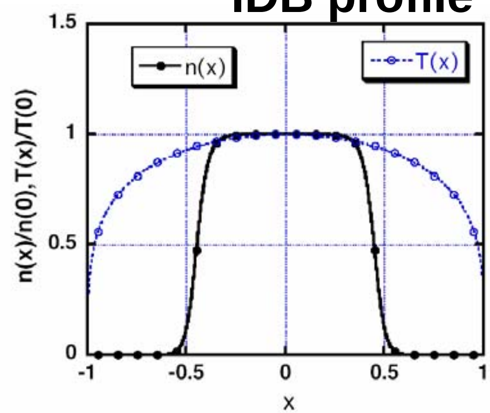


IDB scenario



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	$\langle\beta\rangle$	%	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



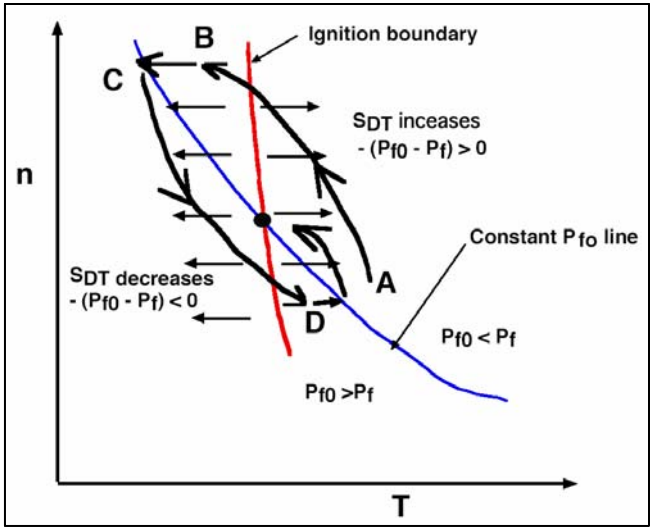
Divertor heat load is drastically reduced.

New control method in a thermally unstable regime

Proportional-Integration-Derivative (PID) control

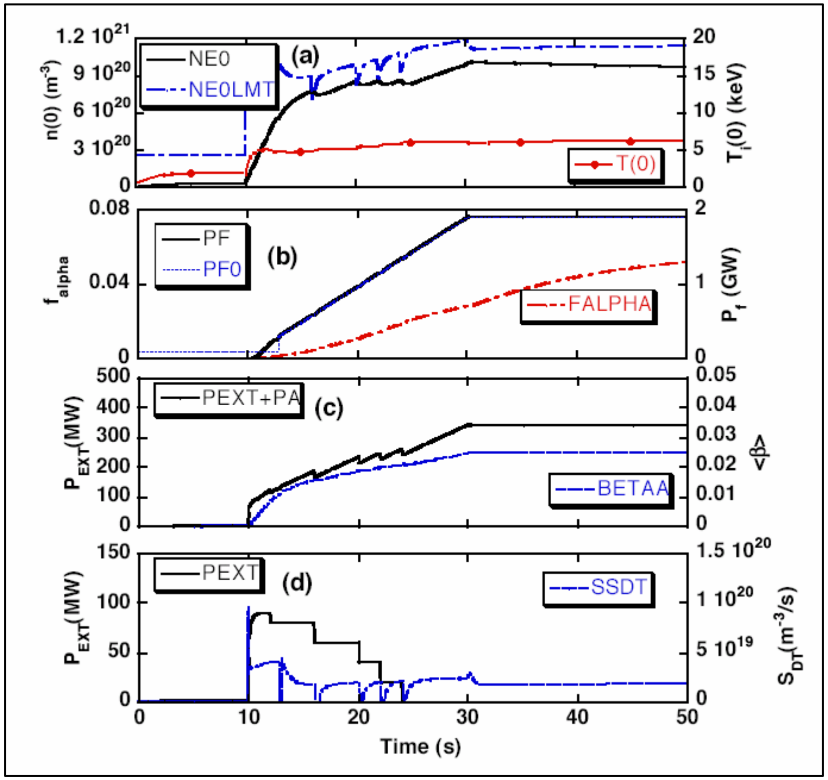
The error of the fusion power with an opposite sign of $e(P_f) = -(P_{fo} - P_f)$ can stabilize the thermal instability through fueling.

$$S_{DT}(t) = S_{DT0}\left\{ e_{DT}(P_f) + \frac{1}{T_{\rm int}}\int_0^t e_{DT}(P_f)dt + T_d \frac{de_{DT}(P_f)}{dt} \right\} G_{fo}(t)$$



$S_{DT}(t)=0$ if $S_{DT}(t)<0$

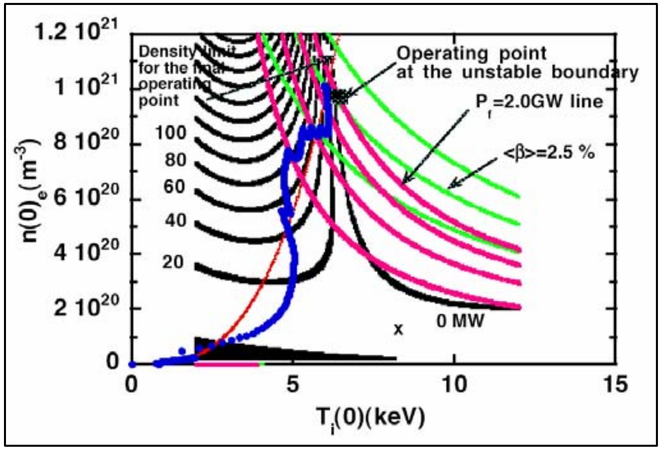
O. Mitarai al. Plasma and Fusion Research, Rapid Communications, 2, 021 (2007) .



$\tau_\alpha^*/\tau_E = 3 \sim 5$

$\tau_p^*/\tau_E = 2 \sim 8$

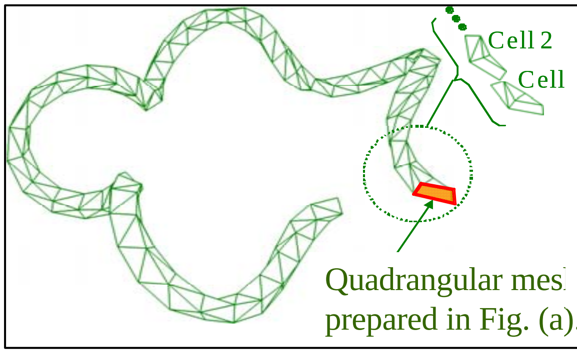
But, effectively reduced due to burning



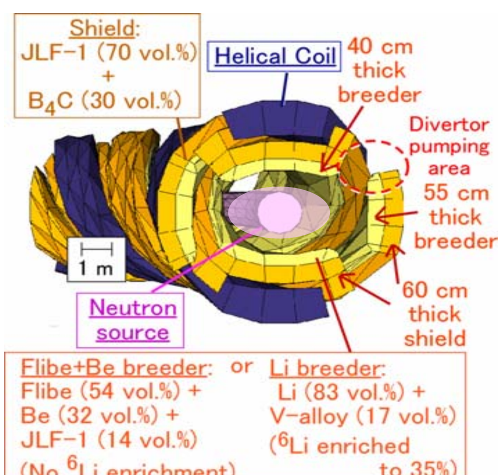
Neutron wall loading is ave. 1.5 MW/m^2 and peaking factor < 1.3

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

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

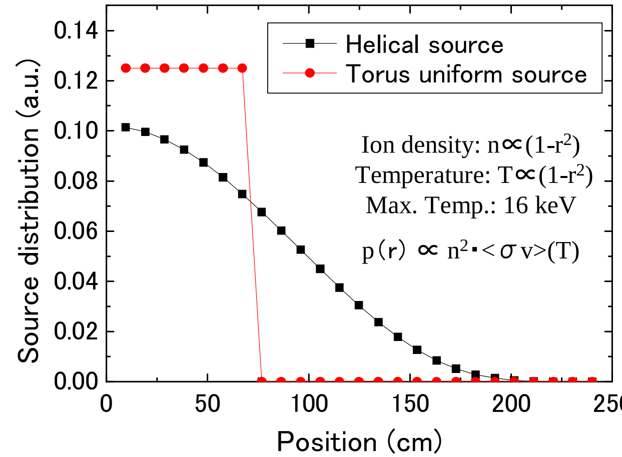


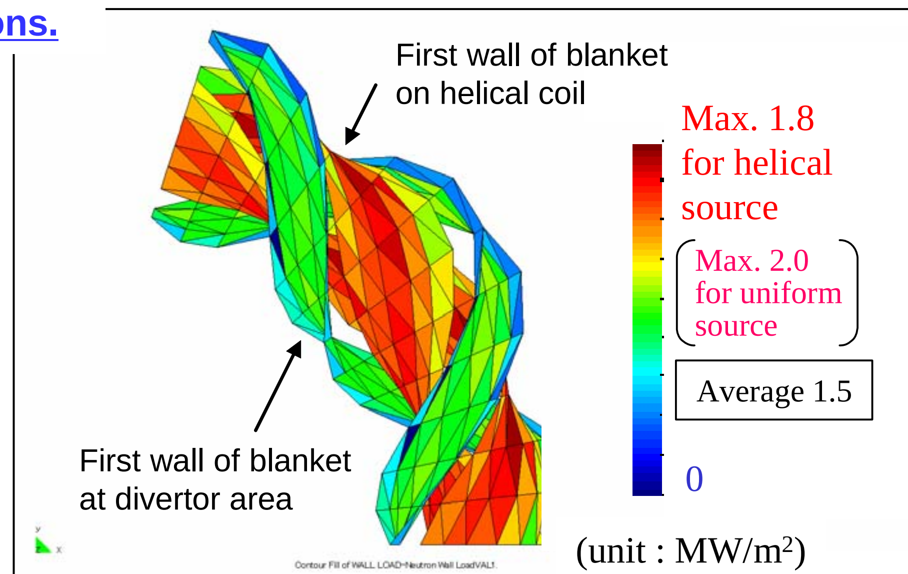
Quick feedback between 3D CAD and 3D Monte-Carlo code MCNP5, **using numerical helical equations.**



Total TBR > 1.1

Two cases of neutron sources





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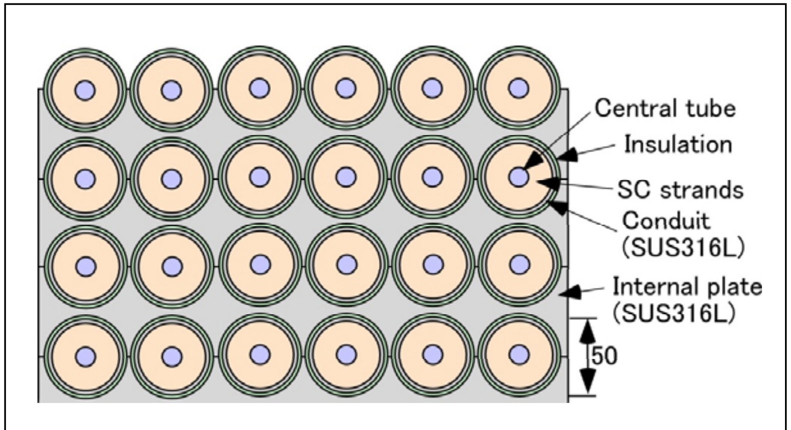
Base design of CICC magnet system

S. Imagawa et al., in this conference.

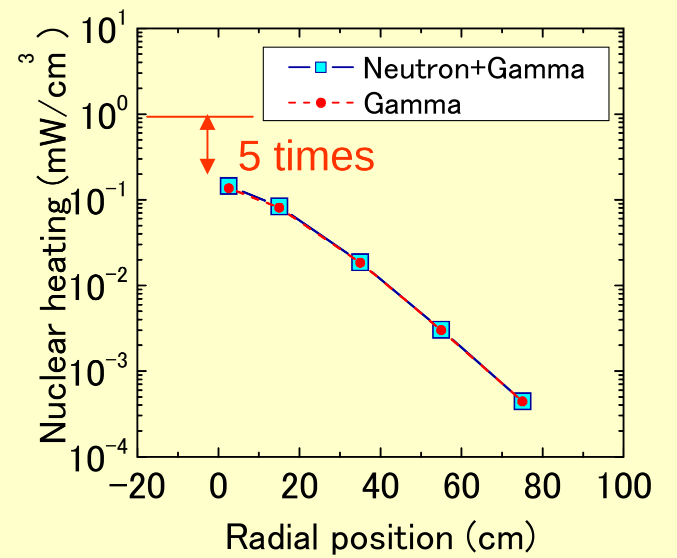
Table 1. Design criteria for CIC conductors based on ITER-TF coils.

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

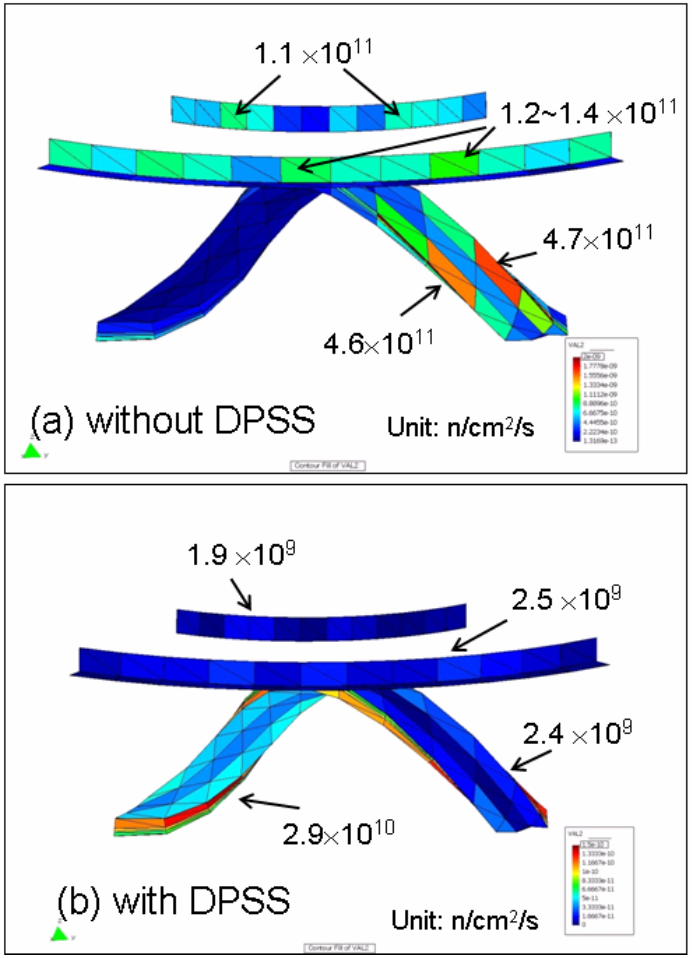


- ✓ Max. cooling path is 500 m for the nuclear heat of 1 mW/cm^3.
- ✓ This value is 5 times larger on the FFHR magnets.
- ✓ Gamma-ray heating is dominant.

Nuclear shielding of SC magnets

by Discrete Pumping with Semi-closed Shield (DPSS)

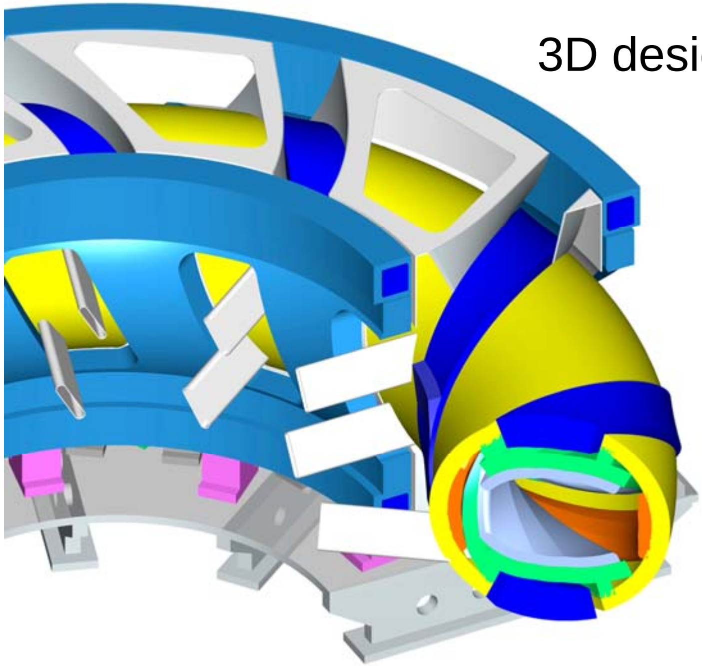
cover rate > 90%



Acceptable level achieved

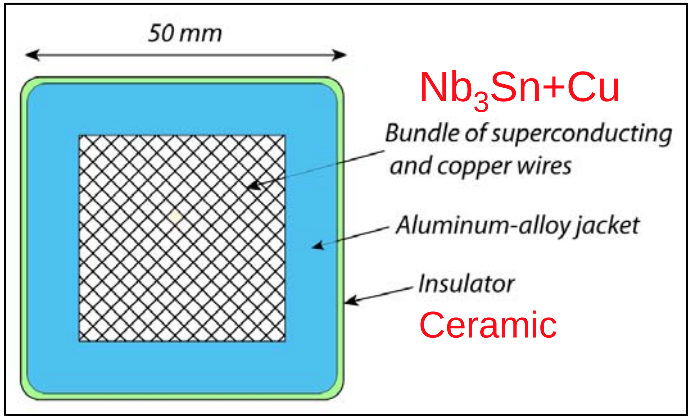
- ✓ Fast neutron < 1E22 n/m² in 30 years
- ✓ Max. nuclear heating < 0.2 mW/cm³
- ✓ Total nuclear heating ~ 40 kW

  Cryogenics power ~ 12 MW (1% of P_f)

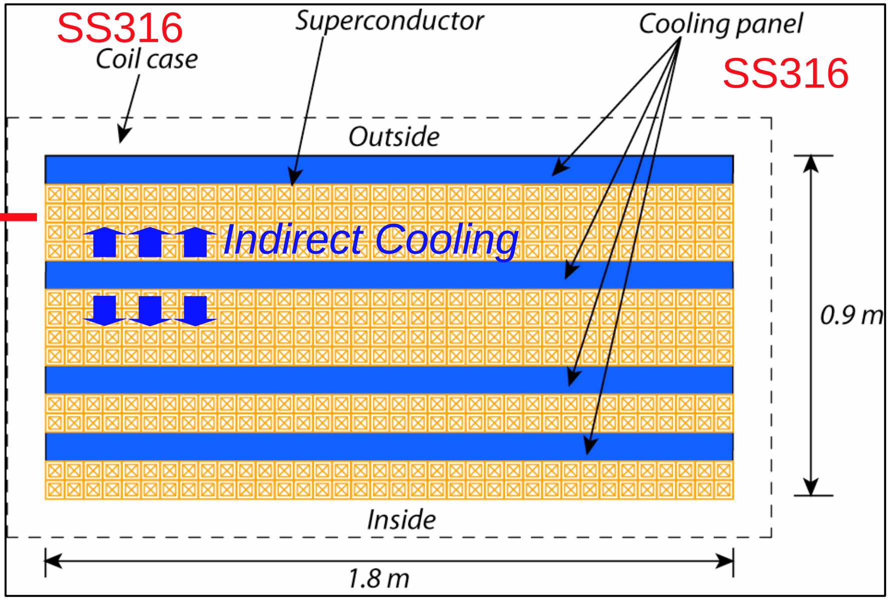


Indirect cooling system
as an alternative
with quench protection candidates

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



100 kA Superconductor
- High effective thermal conductivity
- High mechanical rigidity and strength



Cross-sectional structure of the helical coil

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

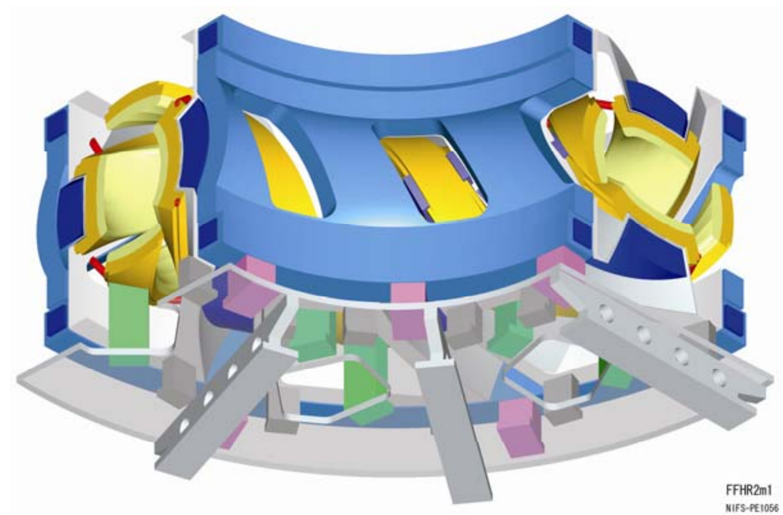
K. Takahata et al., in this conference.

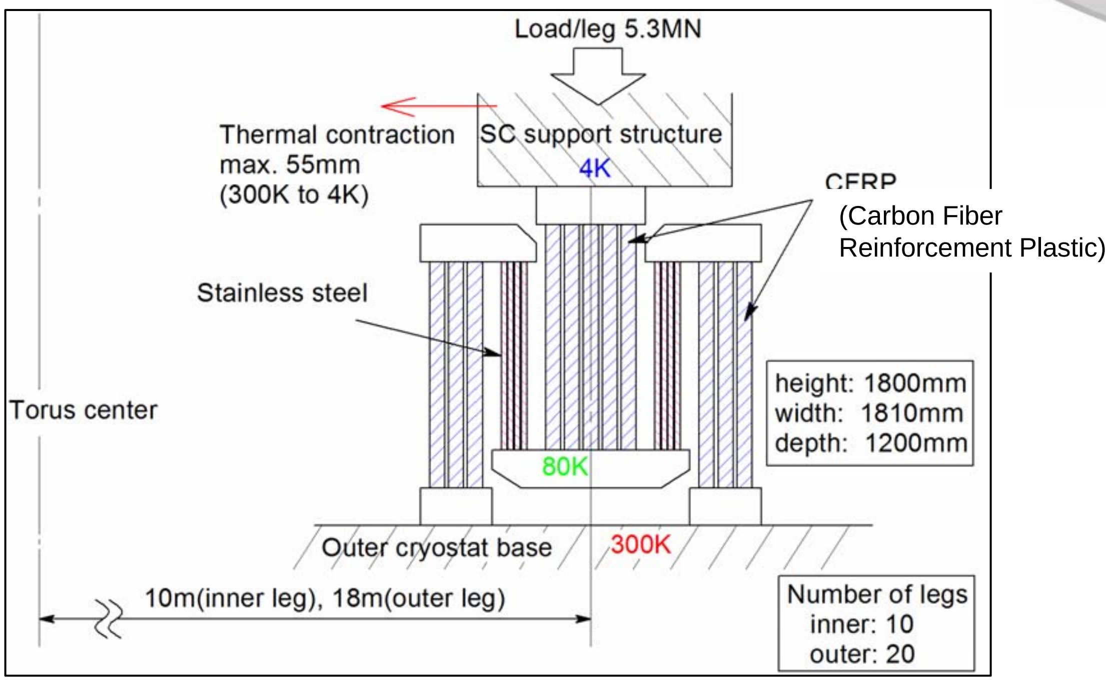
- Quench back with a secondary circuit
- to increase a decay time constant
- to reduce a transient voltage
- to avoid a serious hot spot

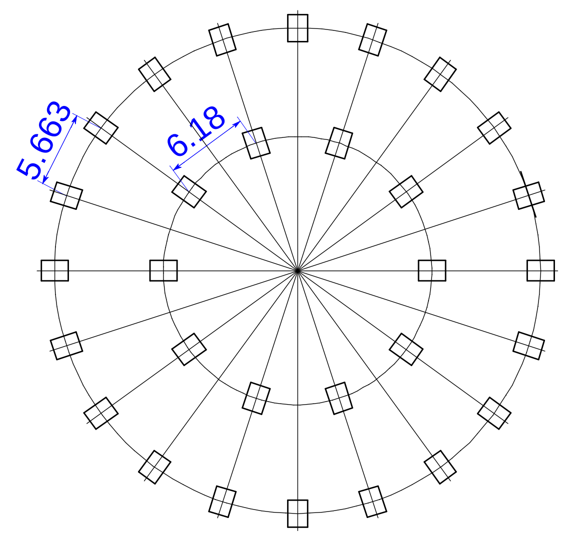
LHD-type support post for FFHR

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)







Design parameters			LHD	FFHR2m1	SDC
Polarity	l		2	2	
Field periods	m		10	10	
Coil pitch parameter	γ		1.25	1.15	
Coil major Radius	R_c	m	3.9	14.0	
Coil minor radius	a_c	m	0.98	3.22	
Plasma major radius	R_p	m	3.75	14.0	
Plasma radius	a_p	m	0.61	1.73	
Plasma volume	Vp	m^3	30	827	
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	
Coil current density	j	MA/m^2	53	26.6	
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	
Electron density	$n_e(0)$	$10\text{^}20\ m^{-3}$		2.4	9.8
Temperature	$T_i(0)$	keV		15.8	6.27
Effective ion charge	Zeff			1.48	1.52
Plasma beta	$<\beta>$	%		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

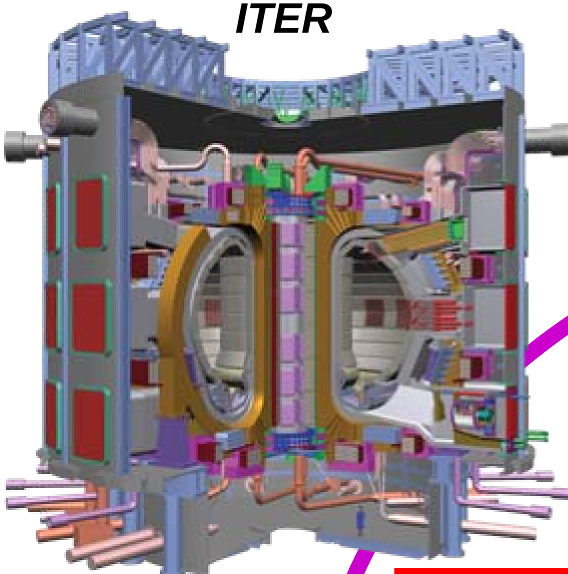
Impurity shielding

He exhaust

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

Tokamak Experimental Rector

ITER



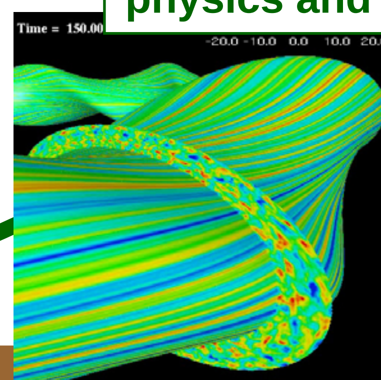
LHD-type Helical Reactor FFHR
Electric Power 1GW
Weight 25,000ton
Magnetic Field 6T

Helical Demo Rector
(29 years to go)

Physics of burning plasmas

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

Multi-layer models covering physics and engineering



LHD-NT
LHD Numerical Test Reactor

LHD

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.