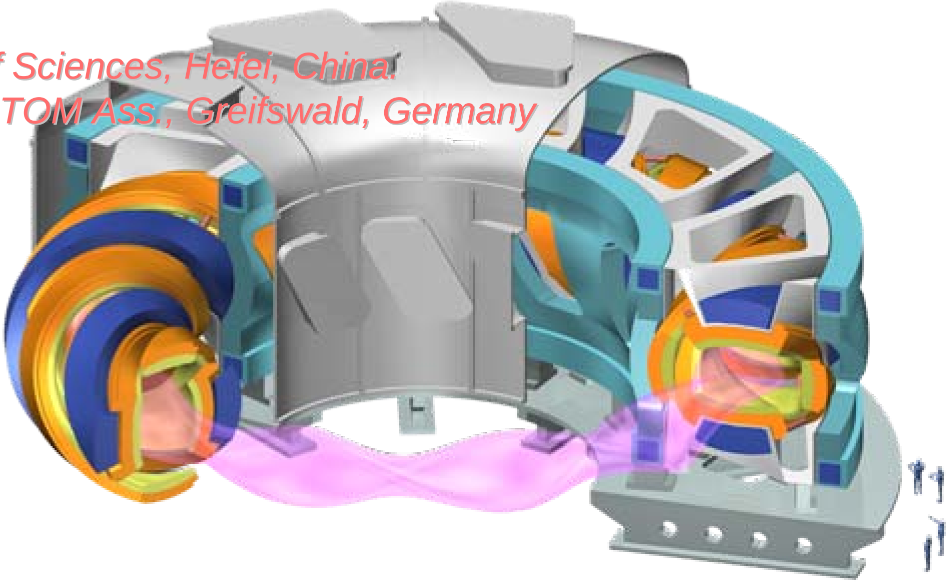
Invited in 18th ITC 2008
Dec.9-12, 2008 at Toki, Japan.

Design Integration of the LHD-type Energy Reactor FFHR2 towards Demo

A. Sagara[1], S. Imagawa[1], Y. Kozaki[1], O. Mitarai[2], T. Tanaka[1], T. Watanabe[1], N. Yanagi[1],
T. Goto[1], H. Tamura[1], K. Takahata[1], S. Masuzaki[1], M. Shoji[1], M. Kobayashi[1],
K. Nishimura[1], T. Muroga[1], T. Nagasaka[1], M. Kondo[1], A.Nishimura[1],
H. Igami[1], H. Chikaraishi[1], S. Yamada[1], T. Mito[1], N. Nakajima[1],
S. Fukada[3], H. Hashizume[4], Y. Wu[5], Y. Igitkhanov[6],
O. Motojima[1] and FFHR design group

[1]National Institute for Fusion Science, Toki, Gifu, Japan
[2]Tokai University, Kumamoto, Japan
[3]Kyushu University, Fukuoka, Japan
[4]Tohoku University, Sendai, Japan
[5]Institute of Plasma Physics, Chinese Academy of Sciences, Hefei, China
[6]Max-Planck-Institut für Plasmaphysik, IPP-EURATOM Ass., Greifswald, Germany



FFHR2m2
3GWth
5 Tesla
30,000 ton

FFHR design collaborations

Fusion Research Network

LHD

Confinement scaling
H.Yamada, Miyazawa (NIFS)

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

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)

Divertor pumping
S.Masuzaki, Kobayashi (NIFS)

Fueling
Sakamoto (NIFS)

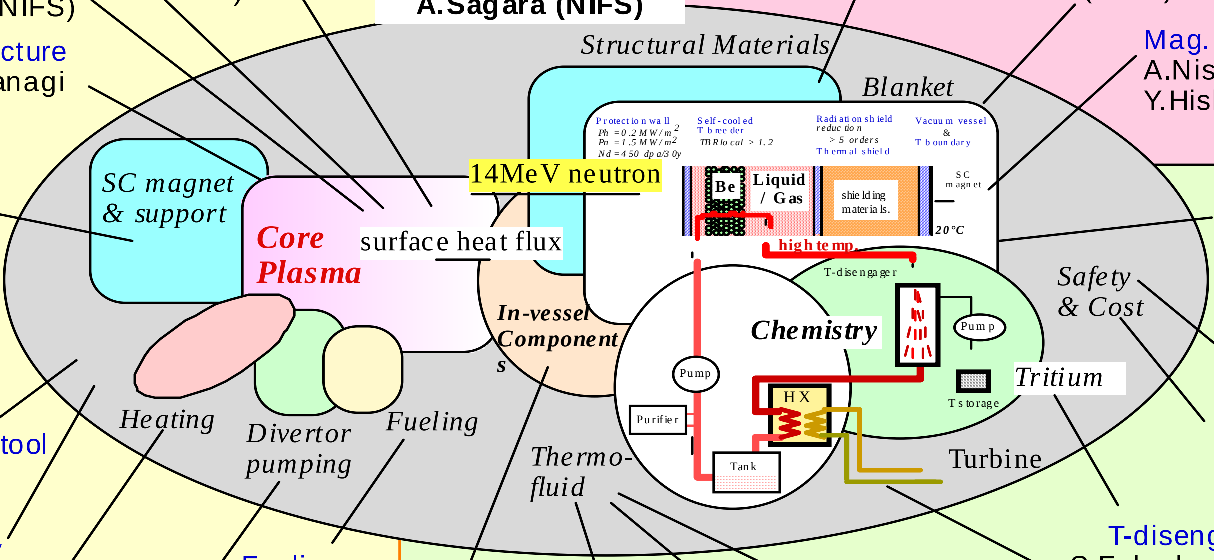
FFHR design
/ Sytem Integration
/ Replacement
A.Sagara (NIFS)

Fusion Eng.R.C.

Blanket material system
T.Muroga, T.Nagasaka, M.Kondo (NIFS)

Neutronics
T.Tanaka, A.Sagara (NIFS)

Mag. material system
A.Nishimura, Y.Hishinuma (NIFS)



Blanket system
T.Terai (Univ. of Tokyo)

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.)

Advanced thermofluid
T.Kunugi (Kyoto Univ.)

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

Advanced first wall
T.Norimatsu (Osaka Univ.)

Int. Collaboration

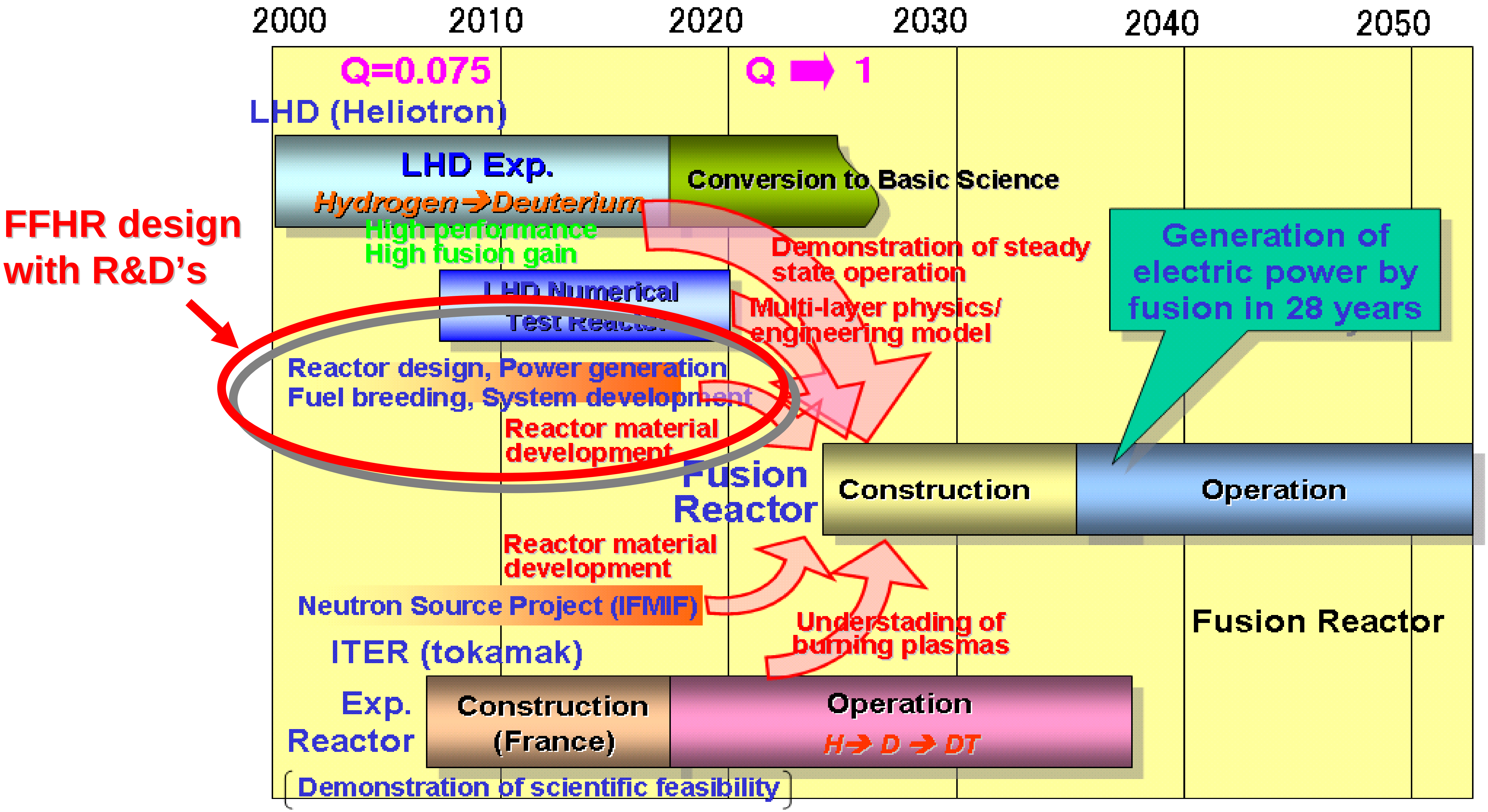
TITAN : USA (ORNL, INL, UCLA, UCSD)

CUP : China (SWIP, ASIPP)

Int. Network Program : EU (IPP), Russia, Ukraine, Korea (KAERI)

April 10, 2008, A.Sagara

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



2000 2010 2020 2030 2040 2050

Q=0.075

Q ➡ 1

LHD (Heliotron)

LHD Exp.

Hydrogen→Deuterium

High performance
High fusion gain

Conversion to Basic Science

LHD Numerical Test Reactor

Demonstration of steady state operation

Multi-layer physics/ engineering model

Generation of electric power by fusion in 28 years

FFHR design with R&D's

Reactor design, Power generation
Fuel breeding, System development

Reactor material development

Fusion Reactor

Construction

Operation

Reactor material development

Neutron Source Project (IFMIF)

Understading of burning plasmas

Fusion Reactor

ITER (tokamak)

Exp. Reactor

Construction (France)

Operation

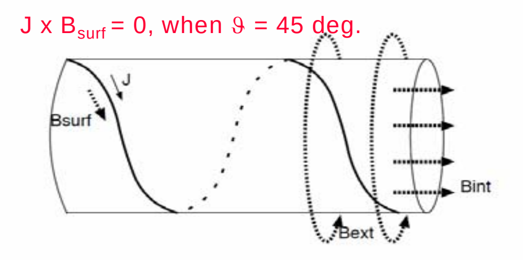
H→ D → DT

Demonstration of scientific feasibility

Two main features in FFHR

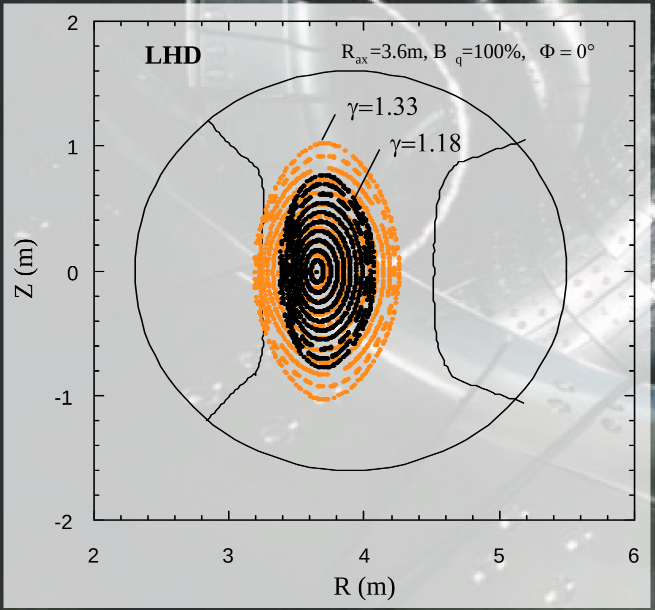
(1) Quasi-force free γ optimization on continuous helical winding

- ➢ to reduce the magnetic hoop force (Force Free Helical Reactor: FFHR)
  - ➡ *large maintenance ports*
- ➢ to expand the blanket space



J x B_{surf} = 0, when ϑ = 45 deg.

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



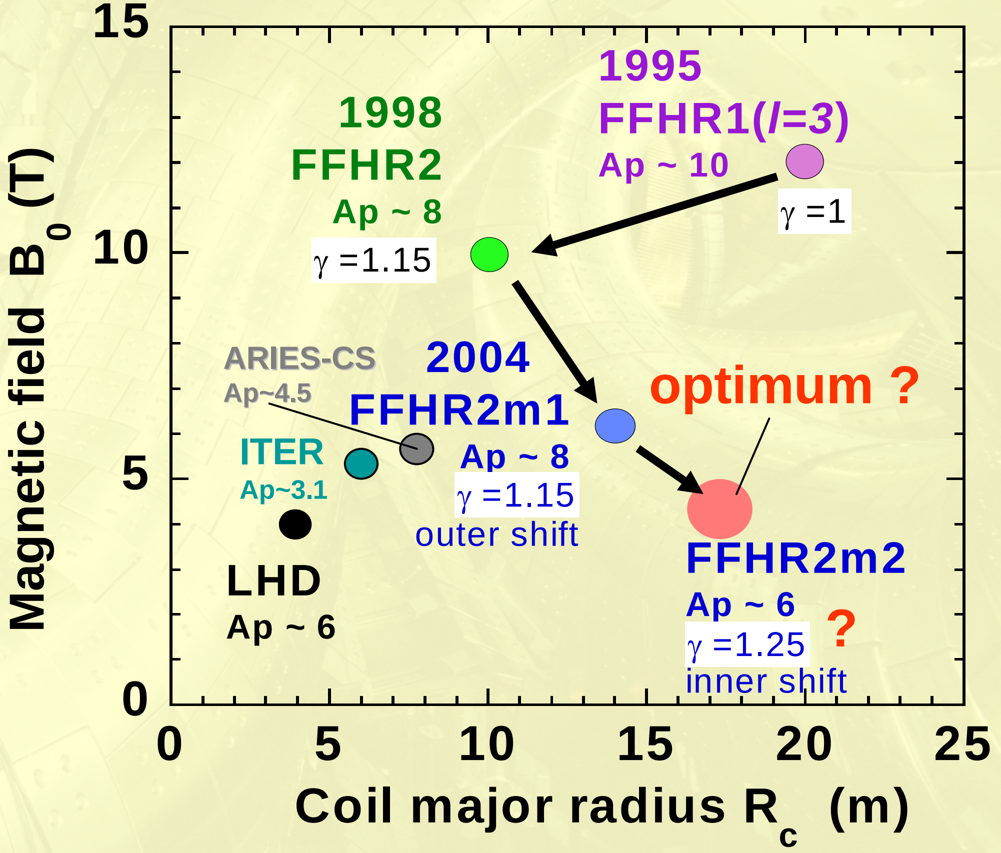
γ=tanθ

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

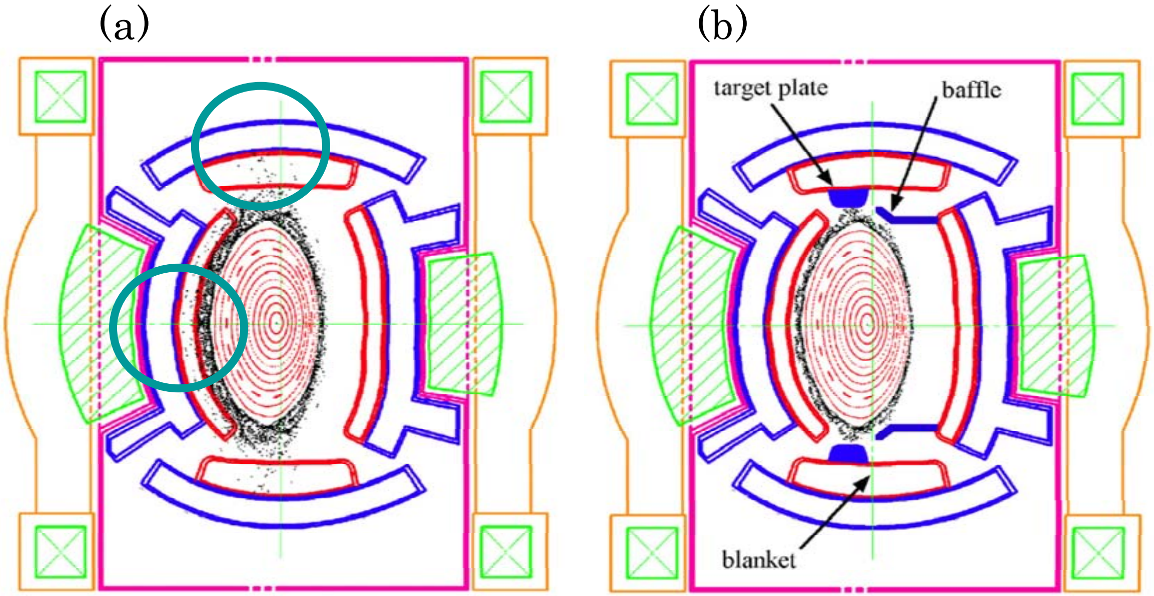
- ➢ low MHD pressure loss
- ➢ low reactivity with air
- ➢ low pressure operation
- ➢ low tritium solubility

Presentation outline

1. Blanket and divertor space
    - Design windows and cost
    - Nuclear shield on SC coils
    - Reactor size optimization
    - New ignition regime
    - Design parameters
2. SC magnet and supports
3. Blanket system integration
4. Concluding remarks



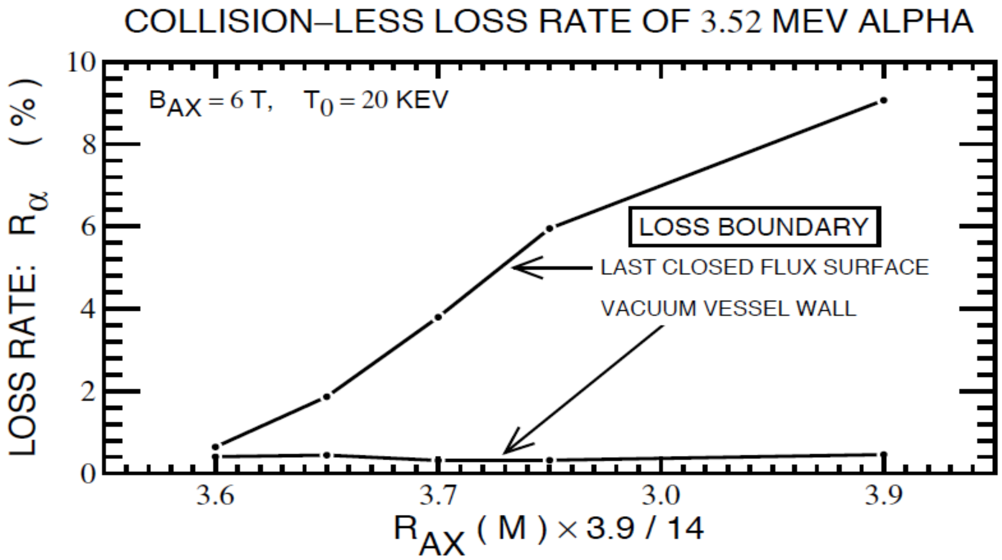
Issues on blanket and divertor space



However

For α-heating efficiency over 90%, the importance of the ergodic layers has been found by collisionless orbits simulation of 3.52MeV alpha particles.

By T. Watanabe

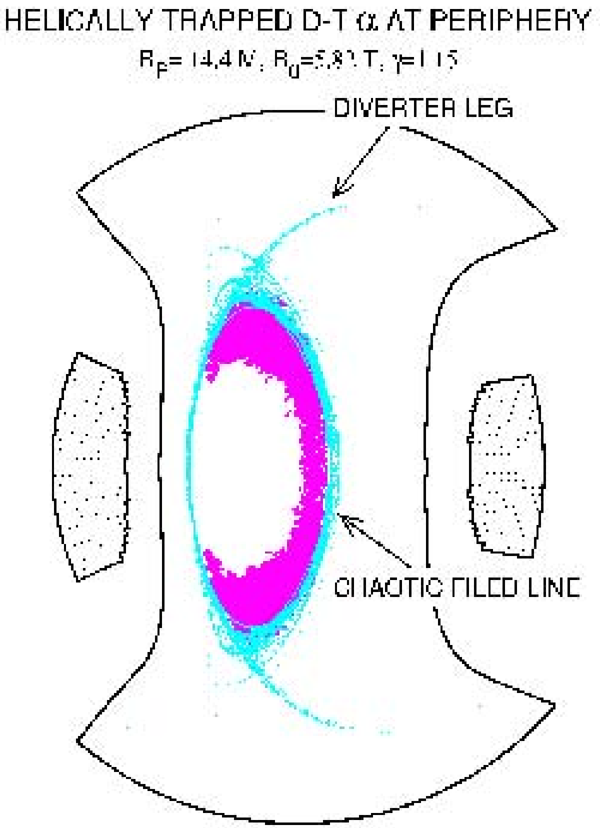


To remove the interference between the first walls and the ergodic layers surrounding the last closed flux surface, helical x-point divertor (HXD) has been proposed.

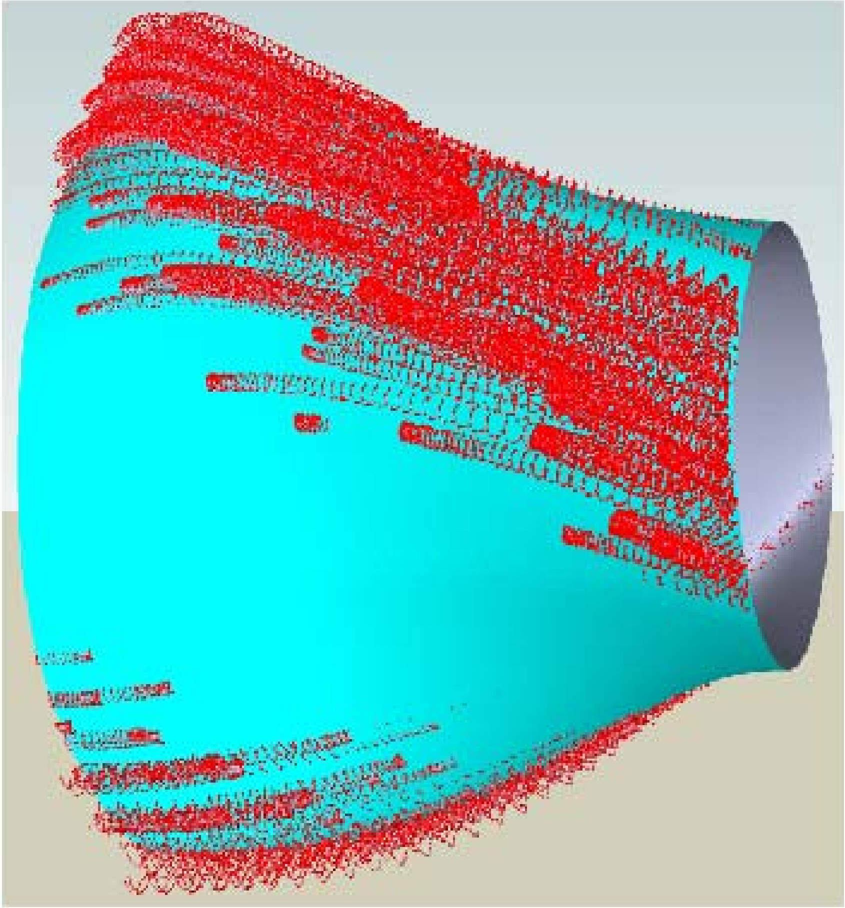
Helical X-point Divertor (HXD) *T. Morisaki et al., FED 81 (2006) 2749.*

By T. Watanabe

D-T α orbits extending to the chaotic field line region.



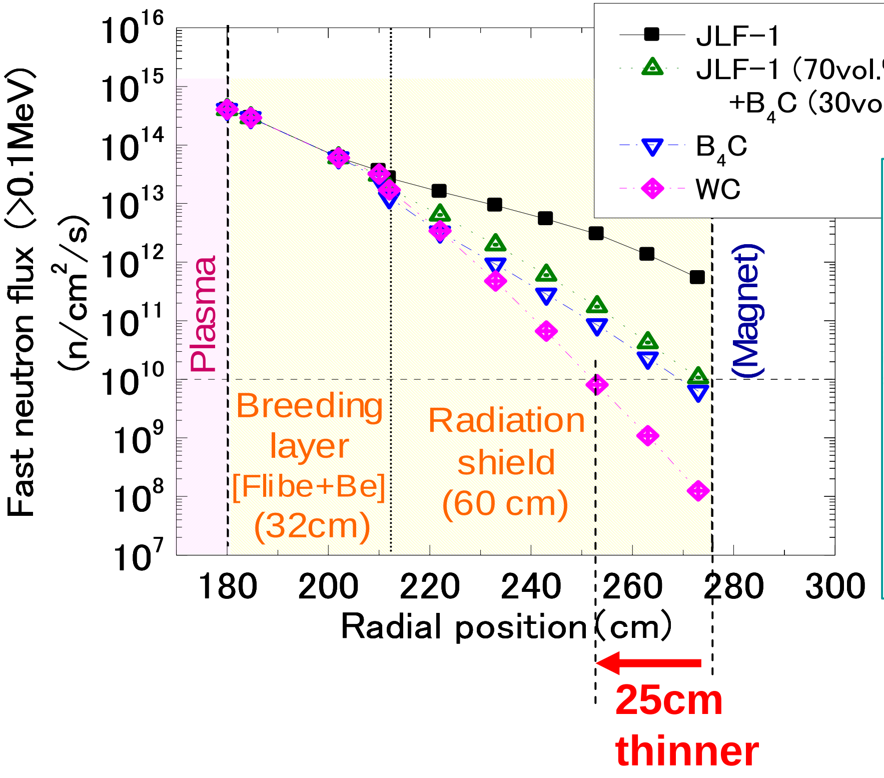
Poincare plot of helically trapped α particles (magenta dots) and the chaotic field lines (sky blue dots).



3D view of the particle orbits and the last closed flux surface(painted by sky blue).

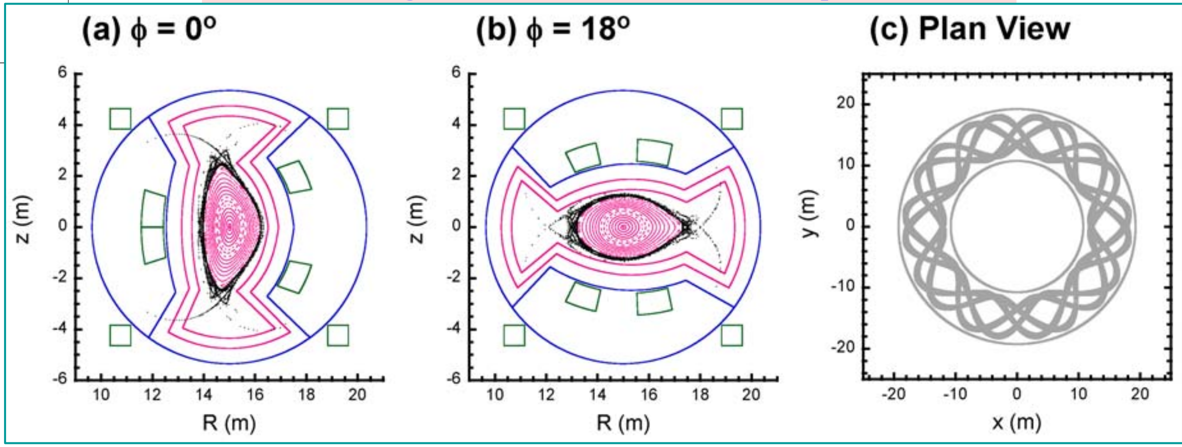
Three candidates are proposed to increase blanket space > 1.1 m

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 by splitting the helical coils.

N. Yanagi et al., in this conference.



$R_c = 15.0$ m, $a_c = 3.0$ m, $\gamma = 1.0$

$B_{axis} = 6$ T, $a_p = 1.5$ m, $W = 143.2$ GJ

Smaller size and higher field with $\gamma = 1$ (reduction of total mass)

FFHR-2S Type-I

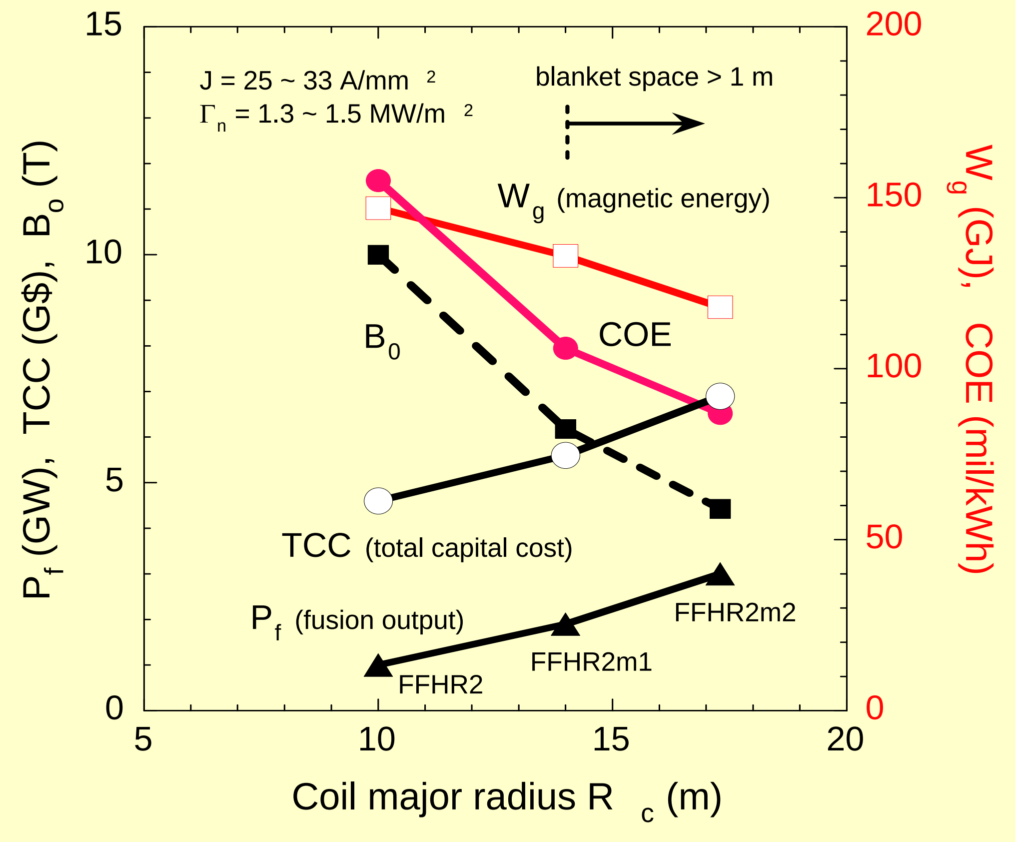
ICRF antenna

K. Saito et al., in this conference.

Three candidates are proposed to increase blanket space > 1.1 m

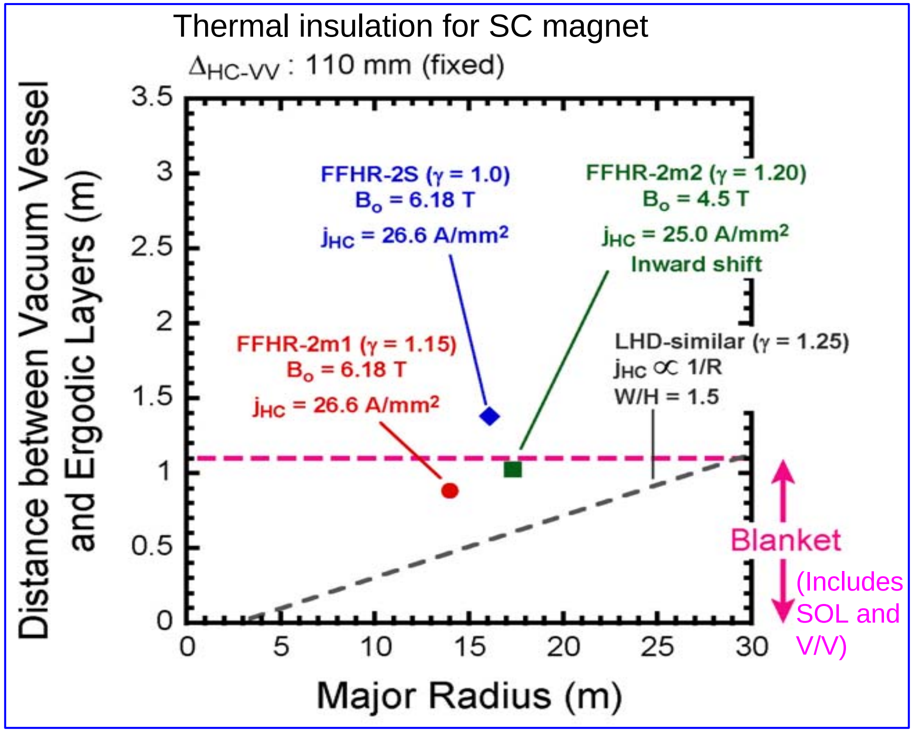
3. Enlargement of reactor size

Neutron wall loading should be kept at < 2 MW/m^2

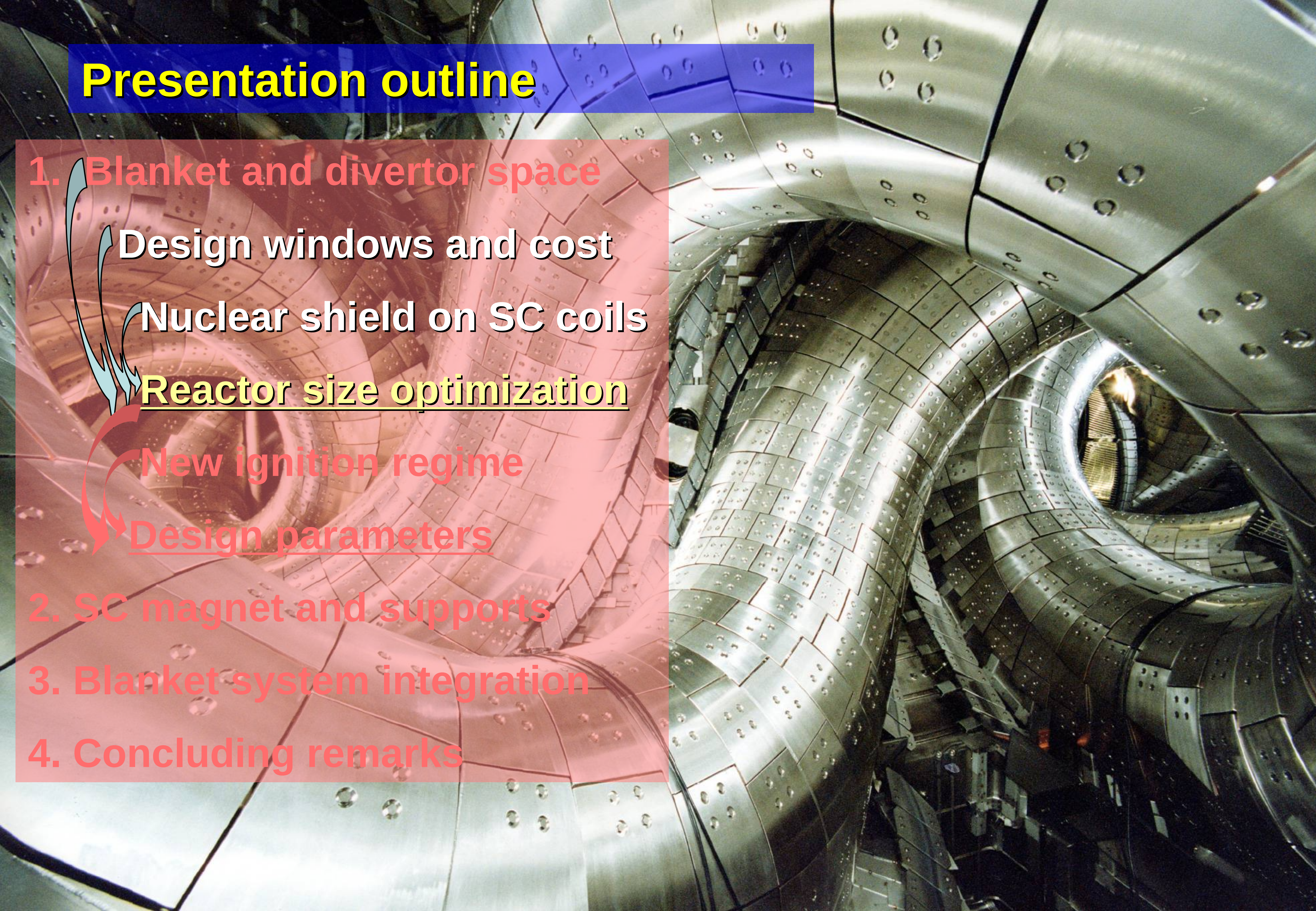


A. Sagara et al., in ISFNT-8, FED 83 (2008) 1690.

R_p=16m (R_c=17.3m) is selected.
(simple expansion of LHD gives too large R_c.)



Presentation outline

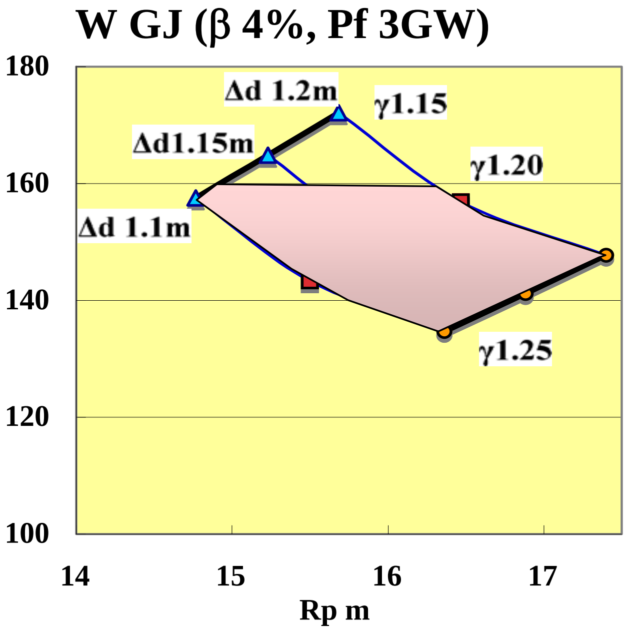


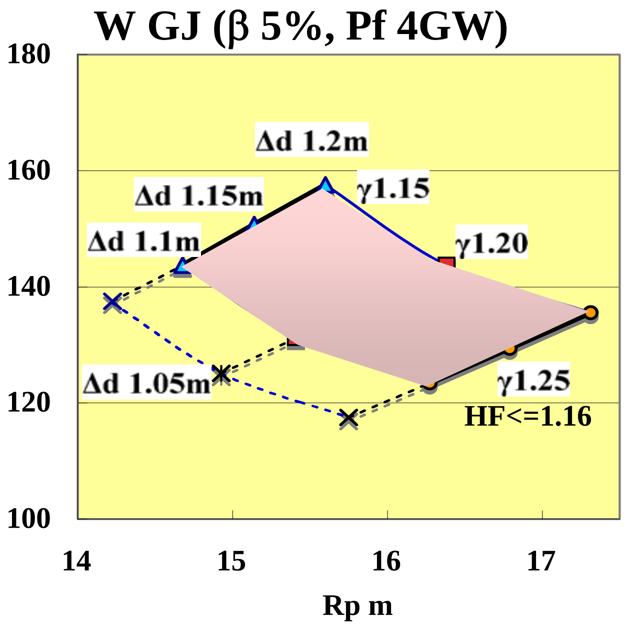
1. Blanket and divertor space
   - Design windows and cost
   - Nuclear shield on SC coils
   - Reactor size optimization
   - New ignition regime
   - Design parameters
2. SC magnet and supports
3. Blanket system integration
4. Concluding remarks

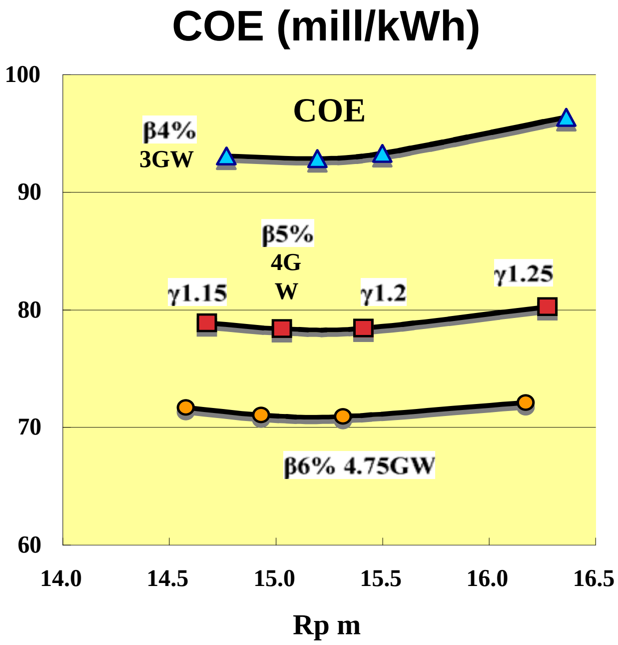
Design windows and cost

Y. Kozaki et al., in this conference.

β=5%, γ = 1.20, j=26A/mm^2 are selected


W GJ (β 4%, Pf 3GW)


W GJ (β 5%, Pf 4GW)


COE (mill/kWh)

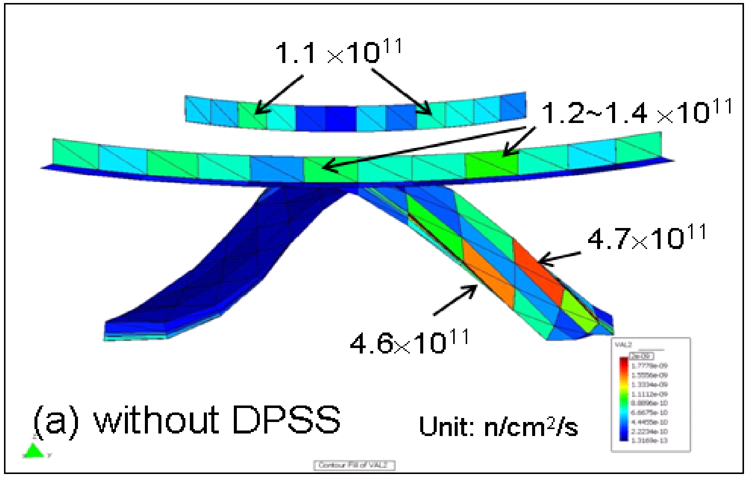
The design windows limited with $\Delta d \geq 1.1$m, $H_f \leq 1.16$, W<160GJ, depending on γ and β. H_f=1.16 means the 1.2 times value achieved in LHD experiment. j=26A/mm^2 is premised.

The COEs of helical reactors, which depend on Rp, γ and β, show the bottom as the result of the trade-off between the Cmag and Cbs, i.e., B0 versus plasma volume.

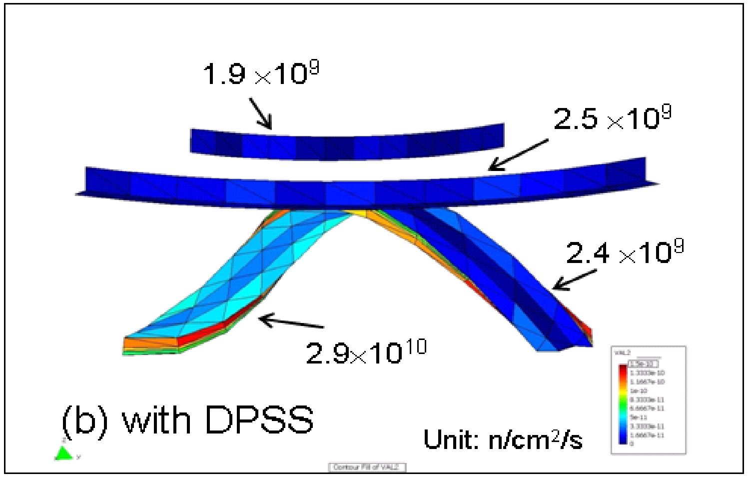
Issues on nuclear shield on SC coils

Discrete Pumping with Semi-closed Shield (DPSS) has been proposed

A. Sagara et al., in ISFNT-8, FED 83 (2008) 1690.



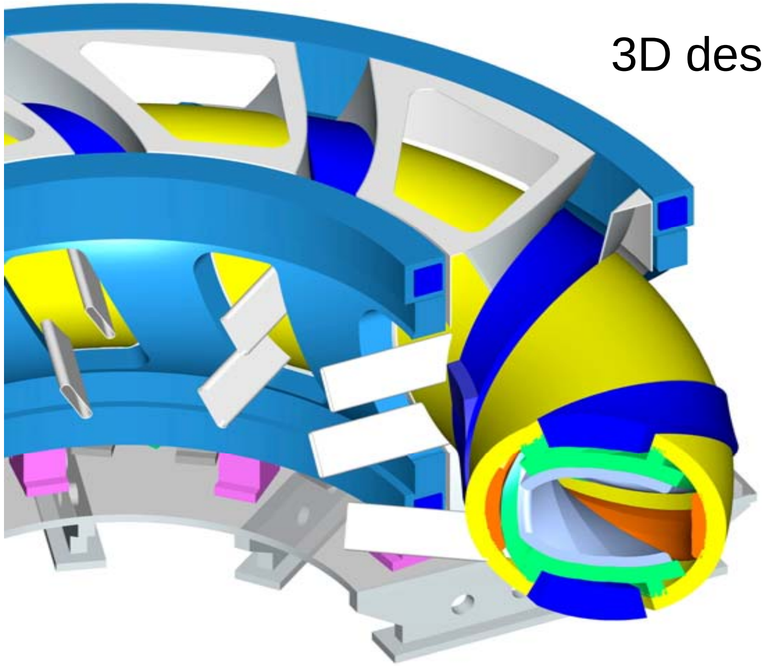
(a) without DPSS



(b) with DPSS

Acceptable level achieved

- ✓ Cover rate > 90%
- ✓ 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)



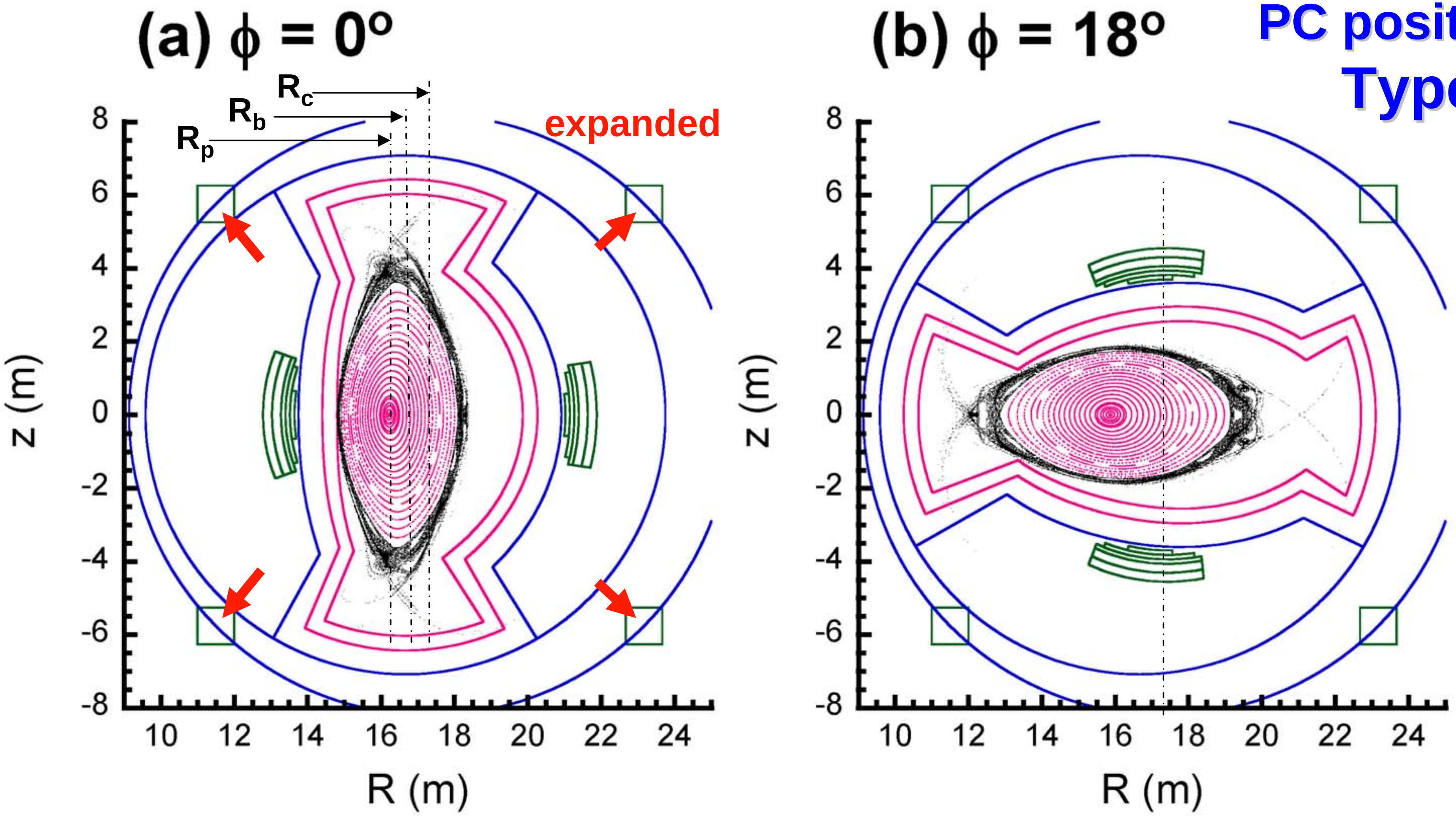
3D design with DPSS

Reactor size optimization of FFHR2m2

Vacuum Magnetic Surface

- inner shifted (*equivalent to Rax=3.6m in LHD*),
- $\gamma = 1.20$, $\alpha = +0.1$
- R_p=16m, R_c=17.3m, B_0=4.84T, j=26A/mm^2 , **W_{mag}=167.7GJ**

PC position Type A



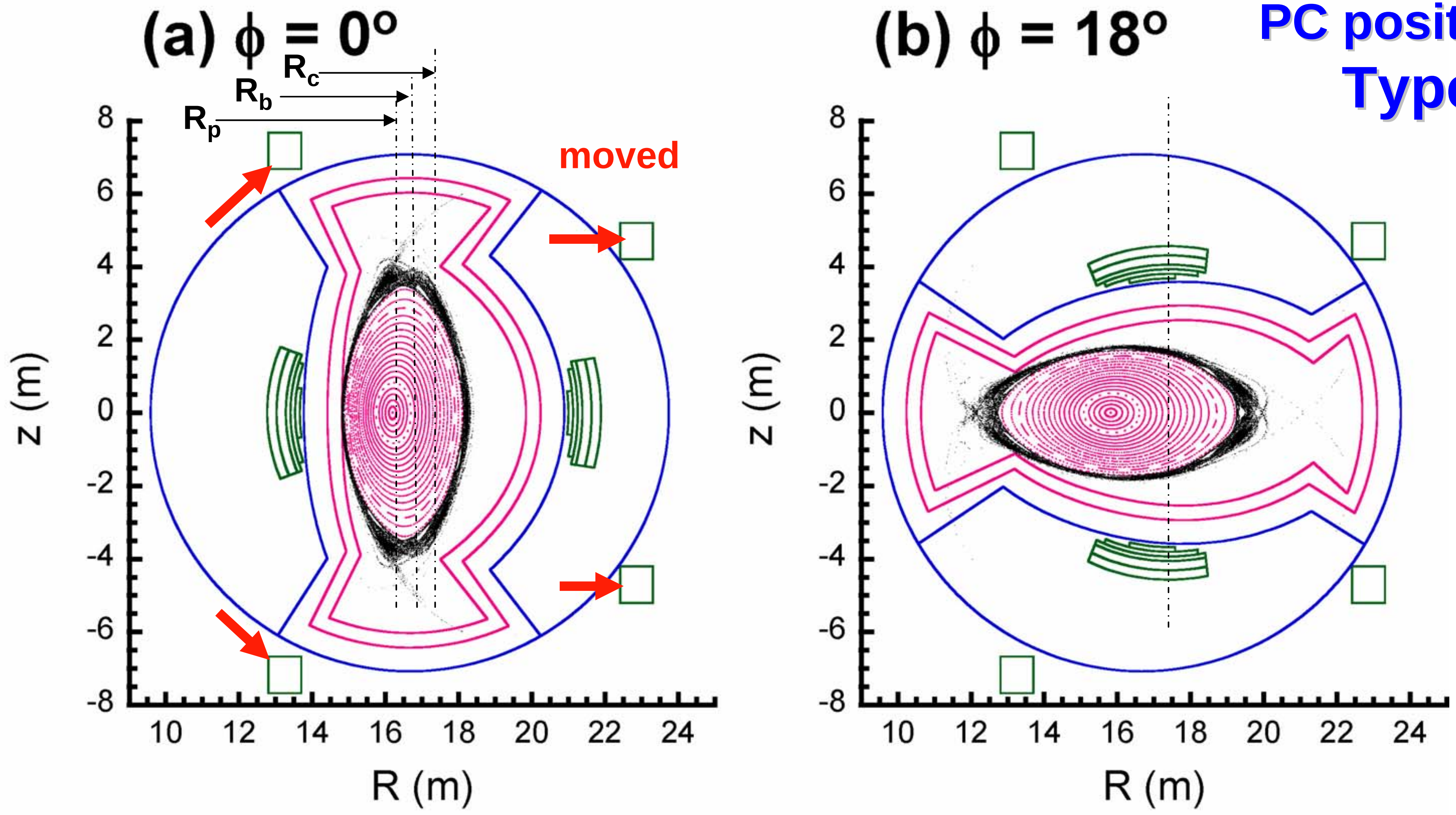
a_{PC} = 8.2 m, HC : 38.72 MA, OV : -23.47 MA, IV : -17.04 MA

Reactor size optimization of FFHR2m2

Vacuum Magnetic Surface

- inner shifted (*equivalent to Rax=3.6m in LHD*),
- γ = 1.20, α = +0.1
- R_p=16m, R_c=17.3m, B_0=4.84T, j=26A/mm² , W_{mag}=149.1GJ

PC position Type B



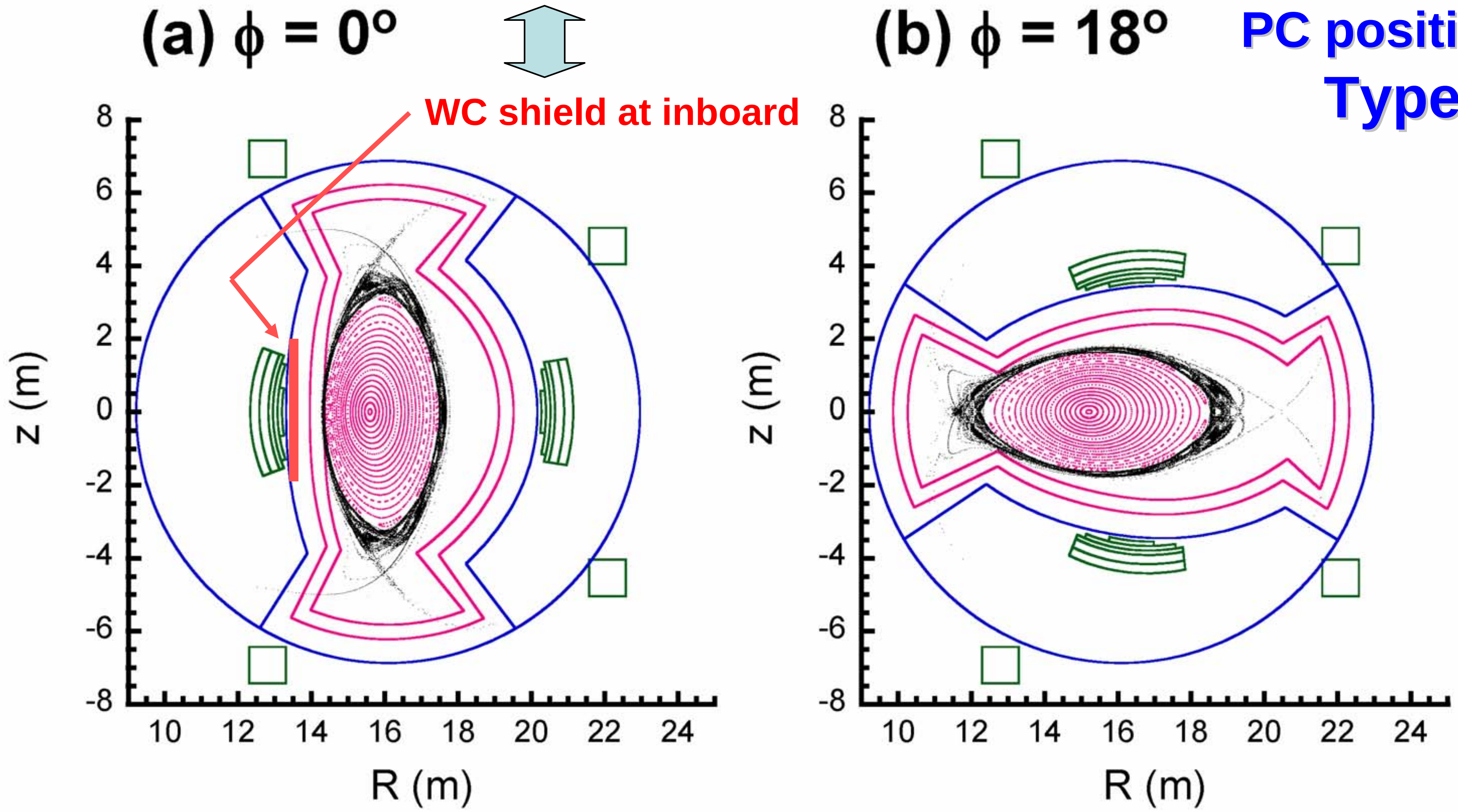
HC : 38.72 MA, OV : -18.39 MA, IV : -13.94 MA

Reactor size optimization of FFHR2m2

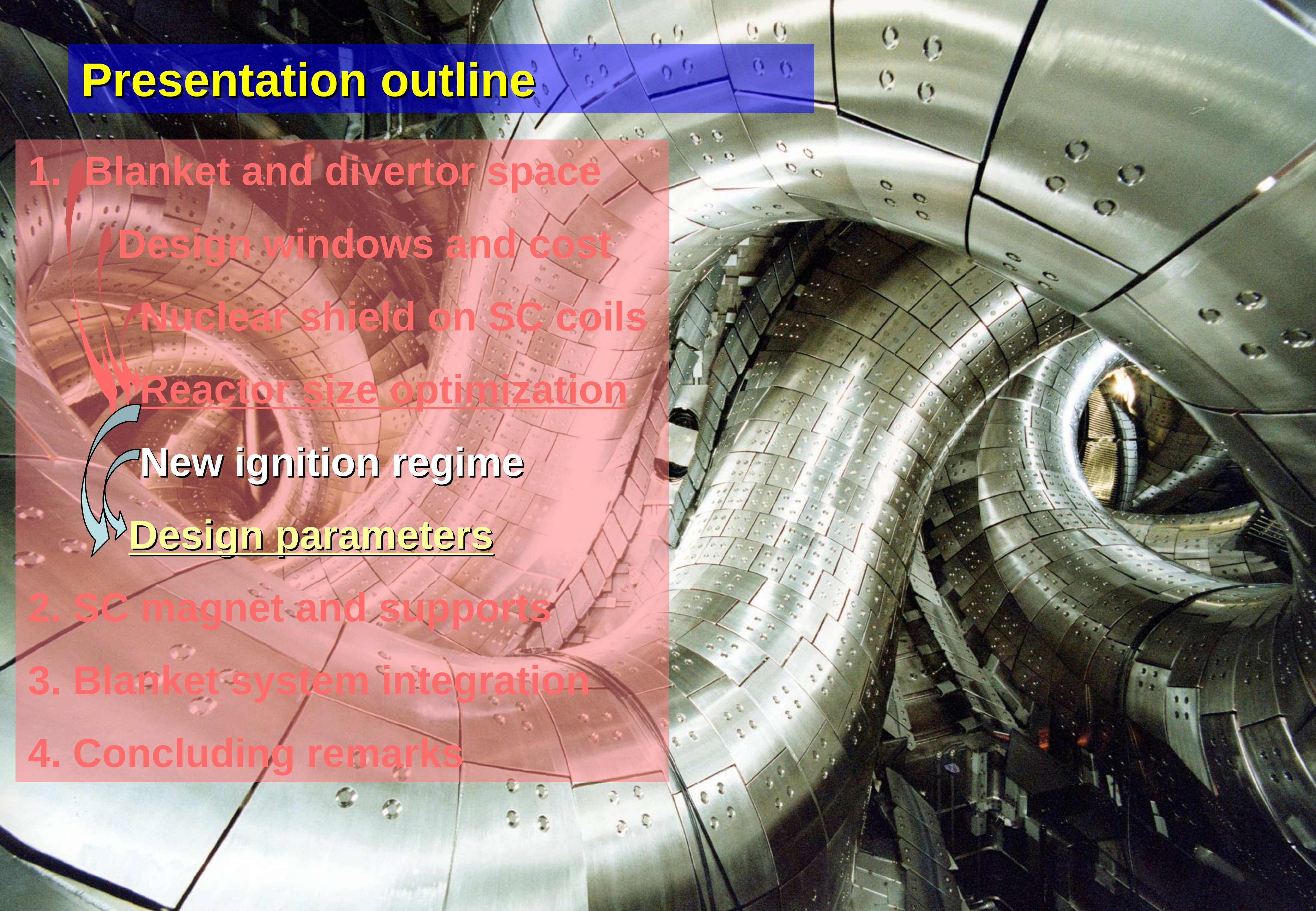
Vacuum Magnetic Surface

- inner shifted (*equivalent to Rax=3.6m in LHD*),
- γ = 1.20, α = +0.1
- R_p=15.5m, R_c=16.7m, B_0=4.90T, j=25A/mm², W_{mag}=135.6GJ

PC position Type B'



HC : 37.87 MA, OV : -18.01 MA, IV : -14.79 MA



Presentation outline

1. Blanket and divertor space

    Design windows and cost

    Nuclear shield on SC coils

    Reactor size optimization

    New ignition regime

    Design parameters

2. SC magnet and supports

3. Blanket system integration

4. Concluding remarks

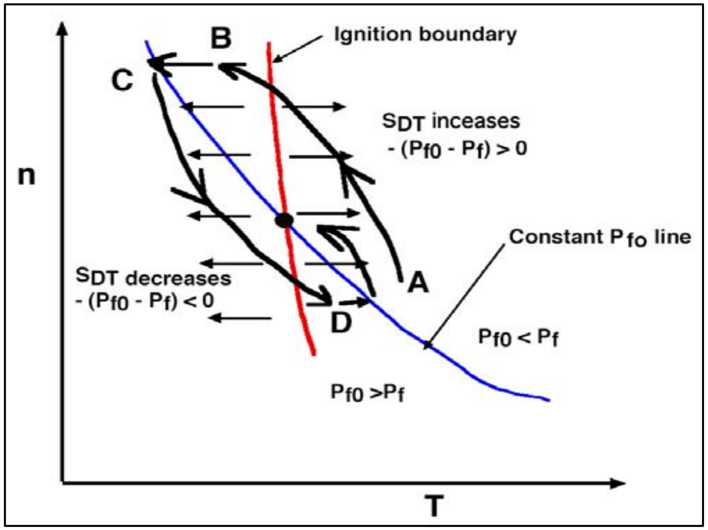
New control method in a thermally unstable ignition regime

O. Mitarai et al., in this conference.

Proportional-Integration-Derivative (PID) control

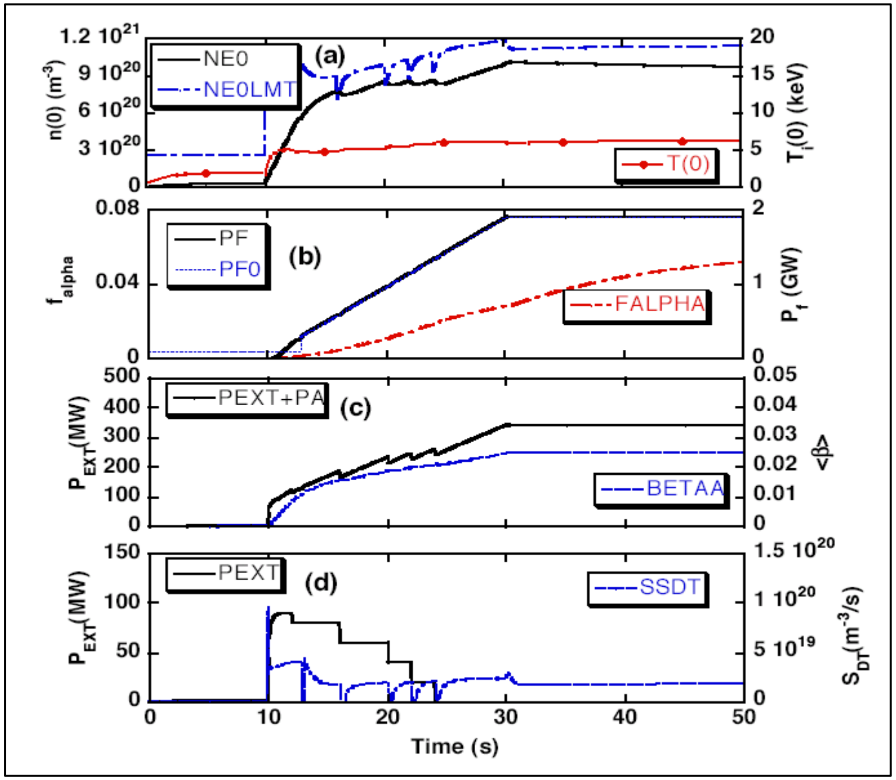
The error of the fusion power with an opposite sign of $e(P_f) = -(P_{fo} - P_f)$ can stabilize the thermal instability through fueling.

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



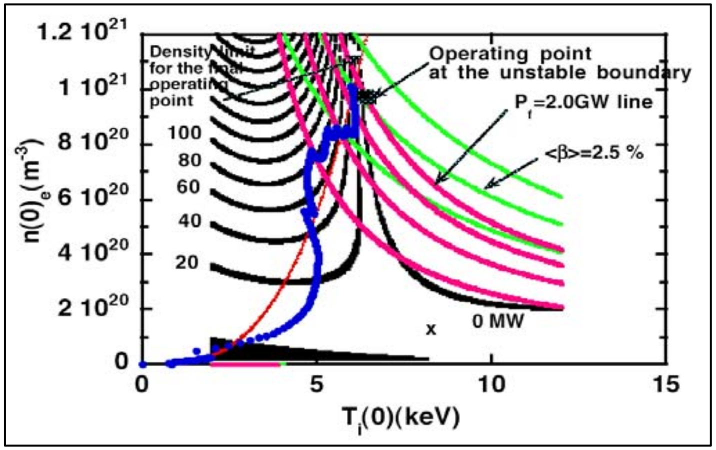
$S_{DT}(t)=0$ if $S_{DT}(t)<0$

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



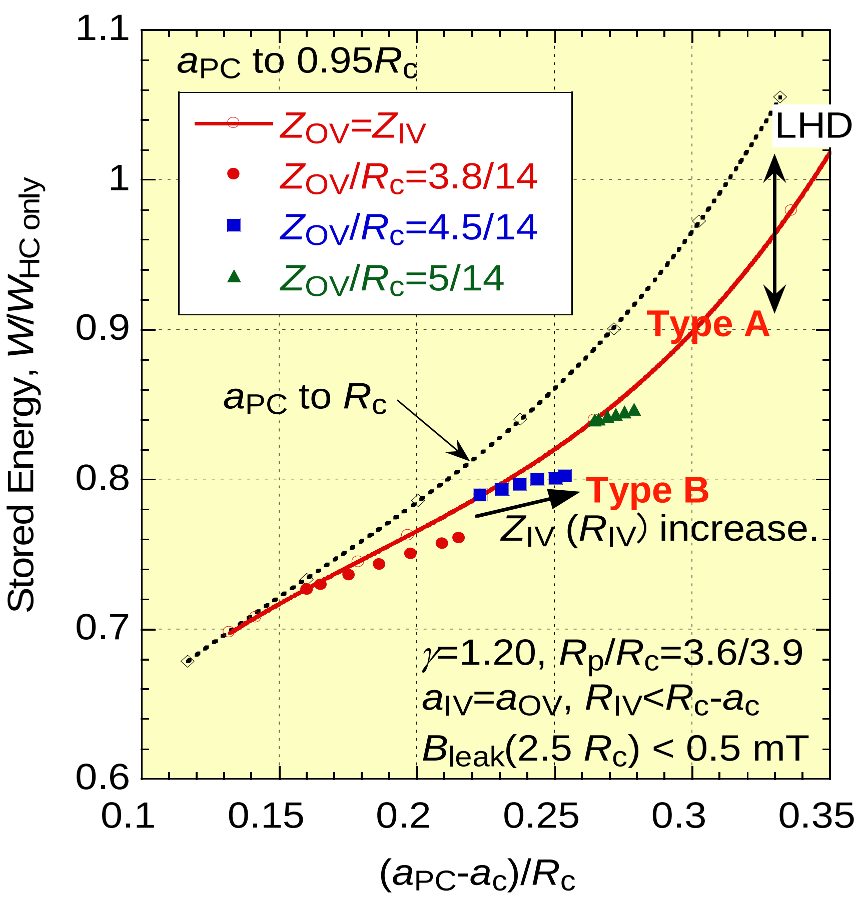
τ_α^*/τ_E=3 ~ 5
τ_p^*/τ_E=2 ~ 8

But, effectively reduced due to burning



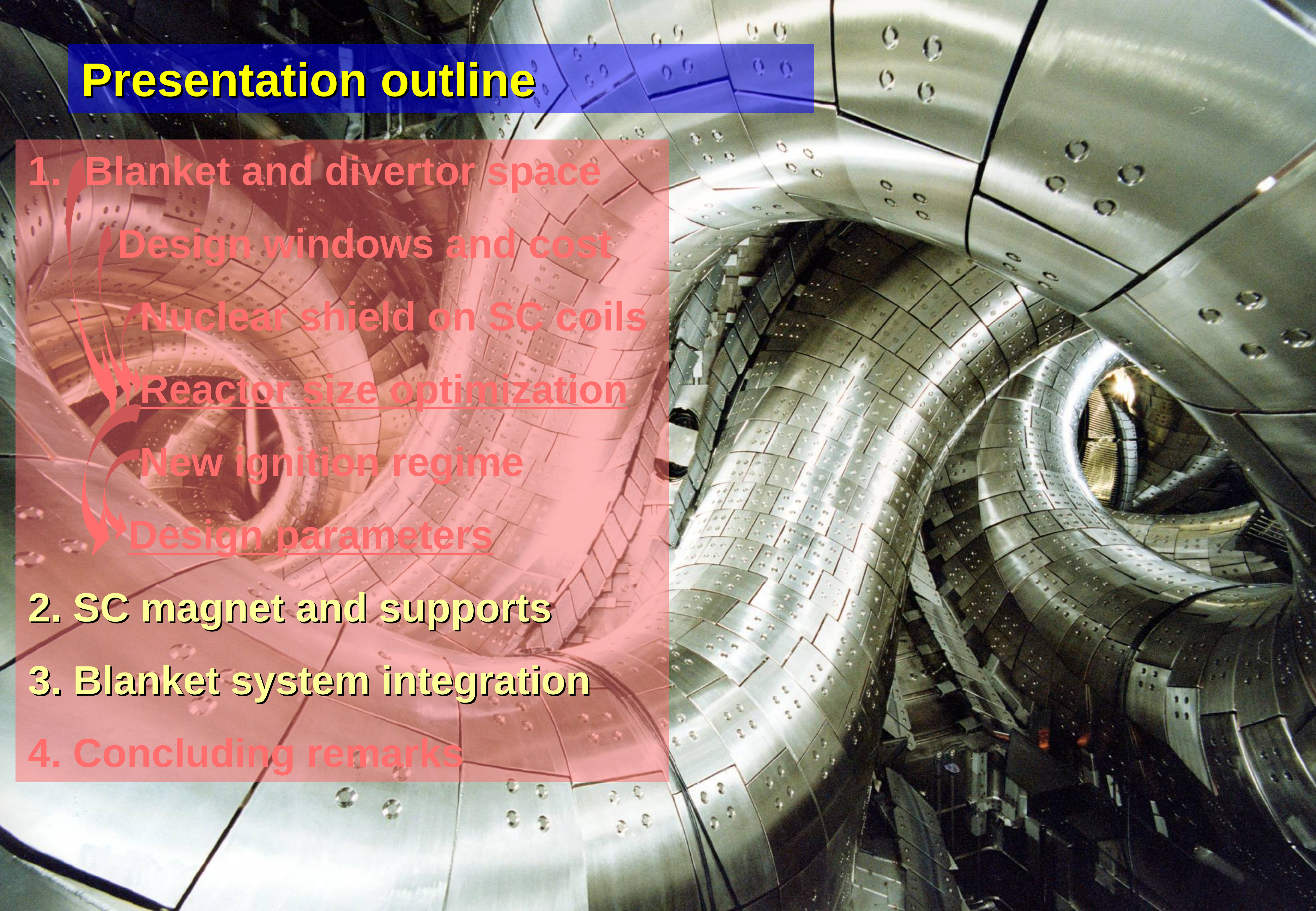
Design parameters

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	$<a_p>$	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		
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 pow	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
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	$<\beta>$	%		1.6	3.0	4.40	3.35
Plasma conduction lo	P_L	MW			290	453	115
Diverter heat load	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	



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

Presentation outline



1. Blanket and divertor space
   - Design windows and cost
   - Nuclear shield on SC coils
   - Reactor size optimization
   - New ignition regime
   - Design parameters
2. SC magnet and supports
3. Blanket system integration
4. Concluding remarks

Base design of CICC magnet system

S. Imagawa et al., in this conference. cc

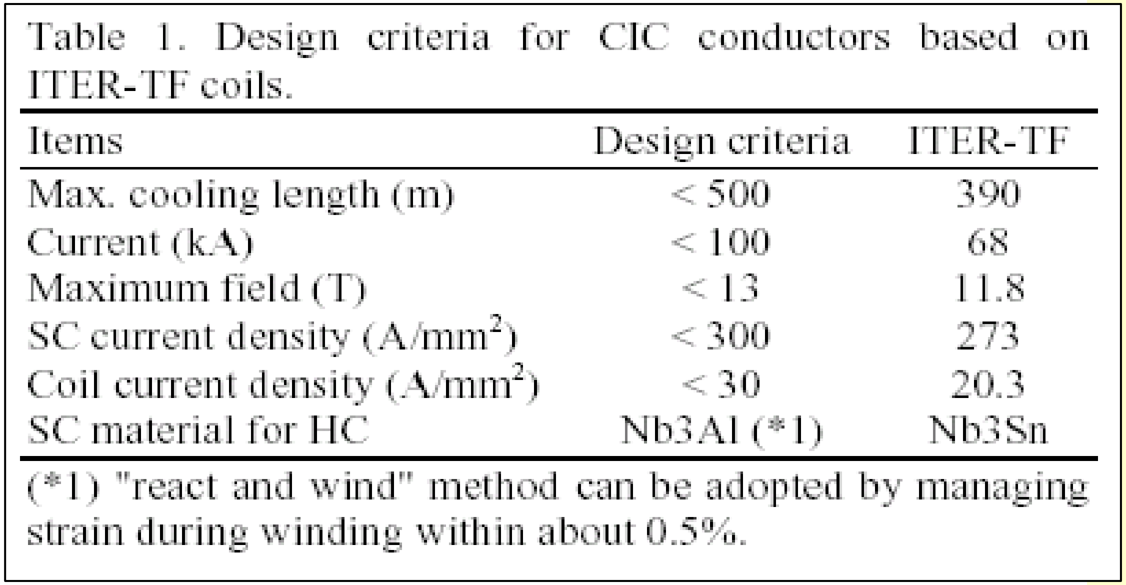
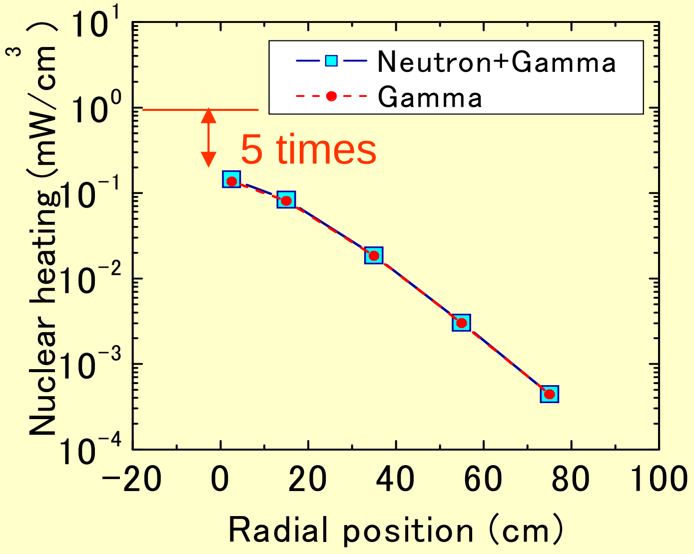


Table 1. Design criteria for CIC conductors based on ITER-TF coils.

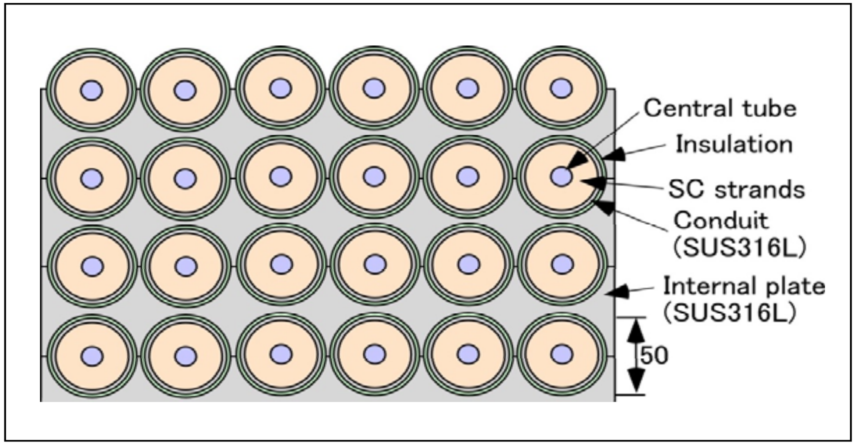
Items	Design criteria	ITER-TF
Max. cooling length (m)	< 500	390
Current (kA)	< 100	68
Maximum field (T)	< 13	11.8
SC current density (A/mm^2)	< 300	273
Coil current density (A/mm^2)	< 30	20.3
SC material for HC	Nb3Al (*1)	Nb3Sn

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

Nuclear heating in FFHR2m1

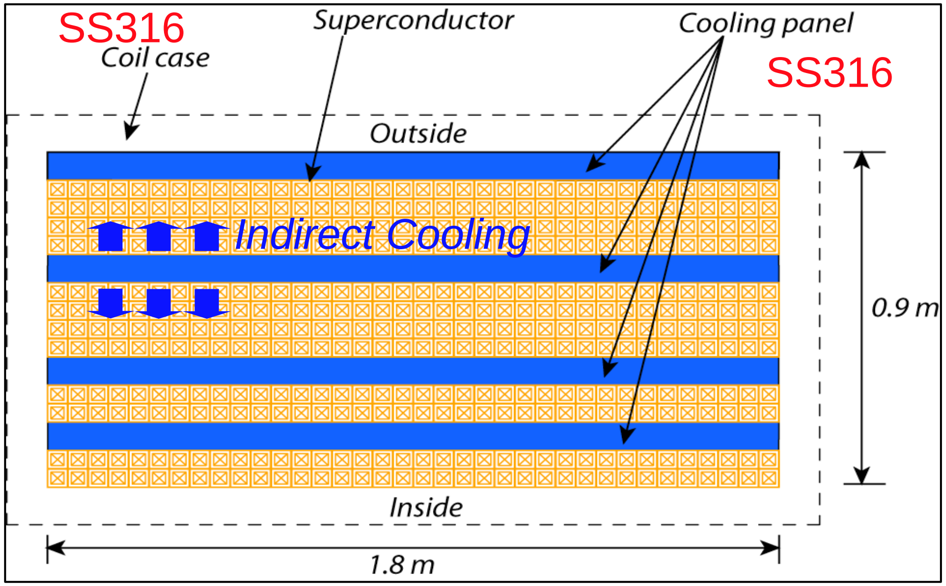


By T. Tanaka



- ✓ **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.**

Indirect-cooled helical coil system
(alternative for CICC)
with quench protection by internal dumping



Cross-sectional structure of the helical coil

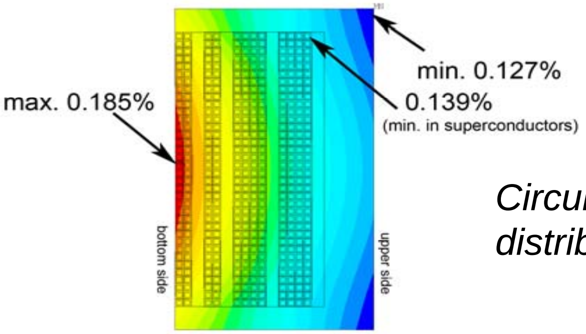
Quench protection by internal dumping

K. Takahata et al., ITC-17.

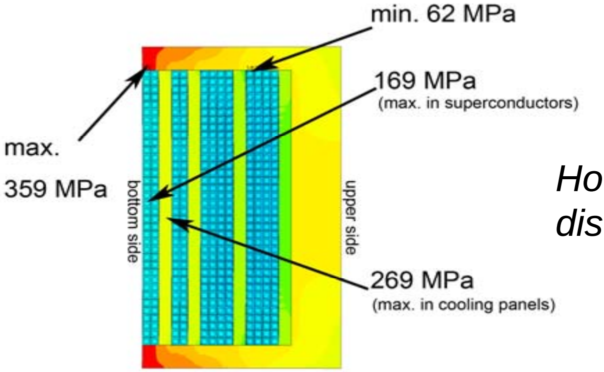
- **to increase a decay time constant > t=20 s**
- **Quench back with a secondary circuit**
- **to reduce a transient voltage V_{max}=10 kV**
- **to avoid a serious hot spot < 150 K**

Stress analysis inside of the coil

H. Tamura et al., in this conference.cc



Circumferential strain distribution



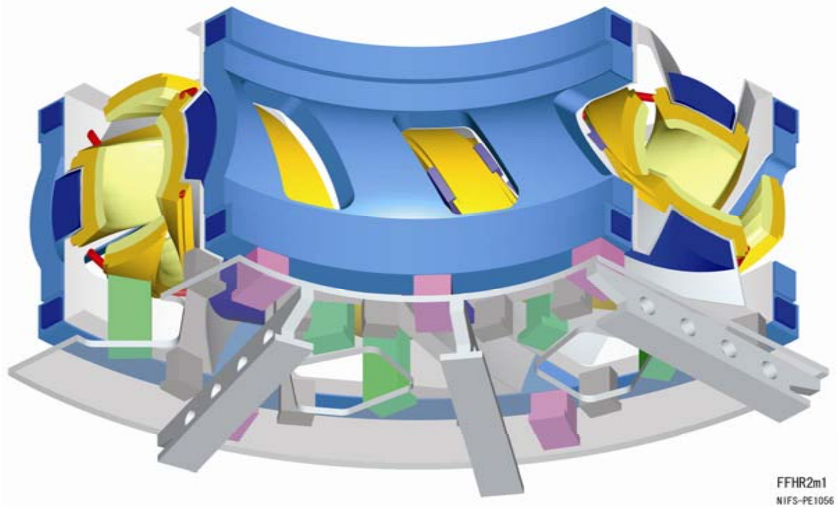
Hoop stress distribution

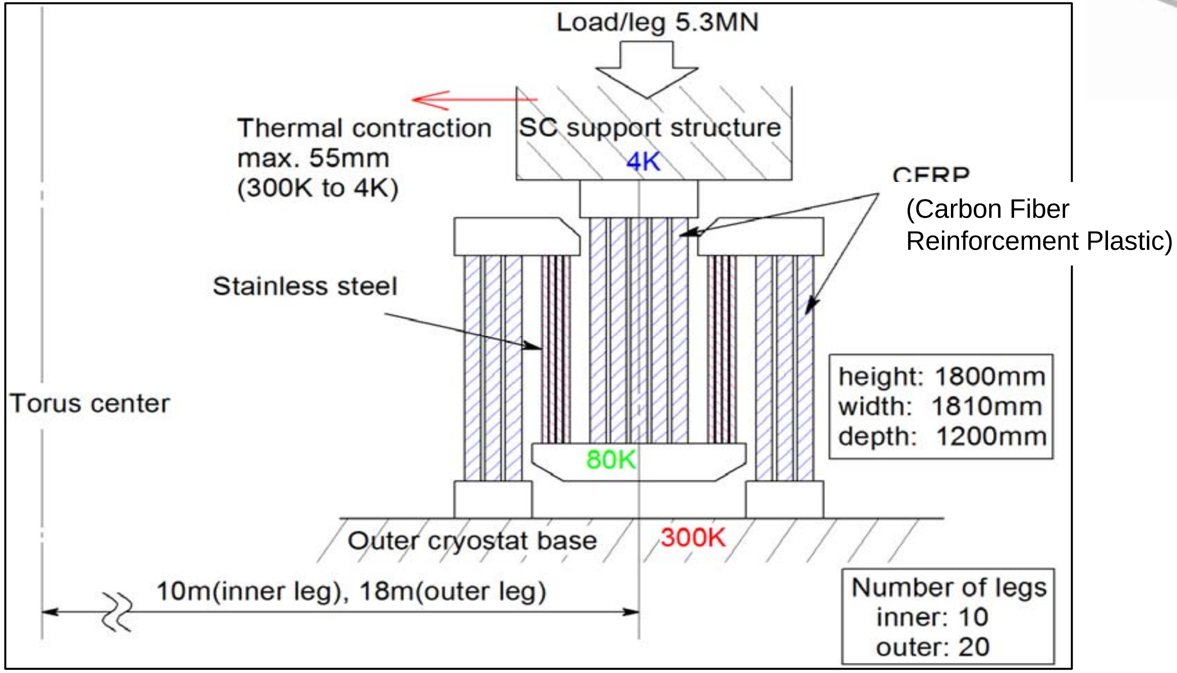
- **Confirmed to be within the permissible range.**

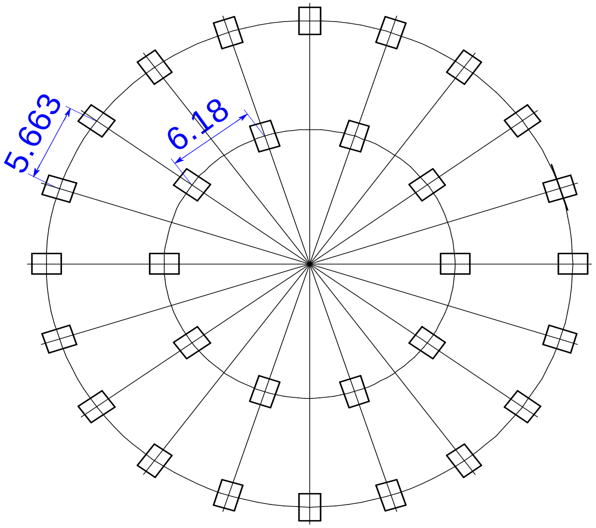
LHD-type support post for FFHR

By H. Tamura

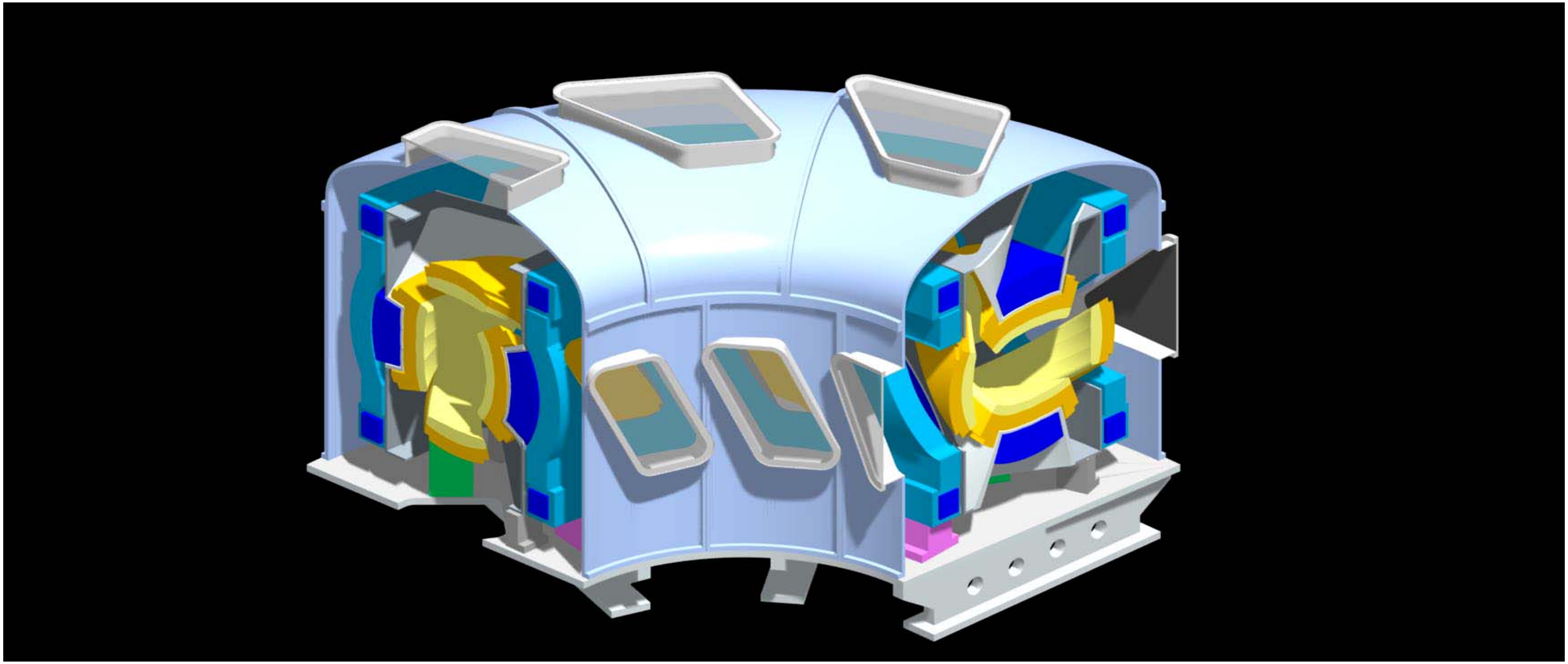
- **Gravity per support**
  = 16,000 ton / 30 legs ~ 530 ton.
- **Thermal contraction < max. 55 mm**
- **Total heat load to 4K ~ 0.34 kW**
  (1/20 of stainless steel post)







Wide maintenance ports



Blanket system integration by broad R&D collaboration activities

Fusion Research Network

LHD

Confinement scaling
H.Yamada,
Miyazawa (NIFS)

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

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)

Divertor pumping
S.Masuzaki,
Kobayashi (NIFS)

Fueling
Sakamoto
(NIFS)

FFHR design

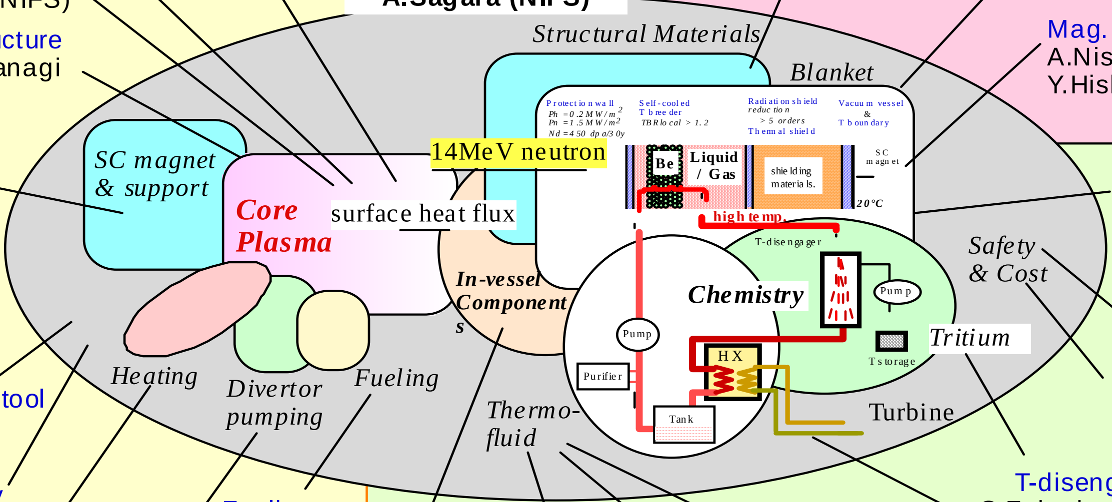
/ Sytem Integration
/ Replacement
A.Sagara (NIFS)

Fusion Eng.R.C.

Blanket material system
T.Muroga,
T.Nagasaka, M.Kondo
(NIFS)

Neutronics
T.Tanaka, A.Sagara
(NIFS)

Mag. material system
A.Nishimura,
Y.Hishinuma (NIFS)



Blanket system
T.Terai
(Univ. of Tokyo)

Safety
T.Uda(NIFS)

Cost evaluation
Y. Kozaki(NIFS)

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

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

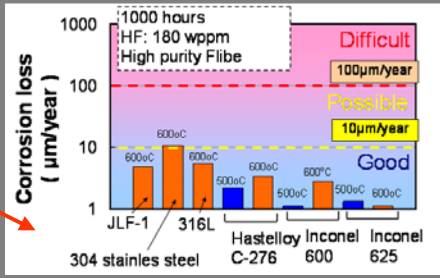
Heat exchanger & gas turbine system
A.Shimizu
(Kyusyu Univ.)

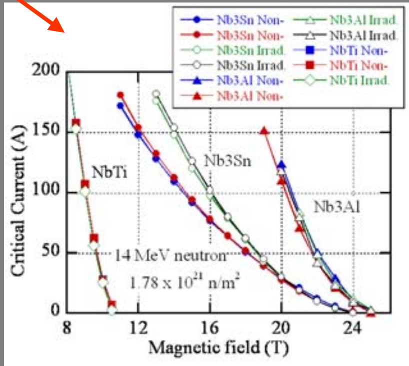
Thermofluid MHD
S.Satake
(Tokyo Univ. Sci.)

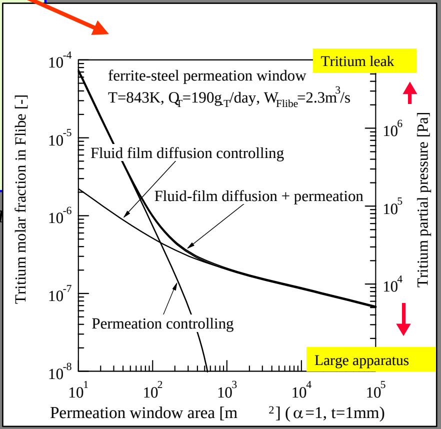
Advanced first wall
T.Norimatsu
(Osaka Univ.)

Advanced thermofluid
T.Kunugi
(Kyoto Univ.)

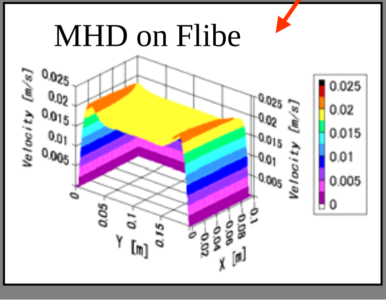
gara

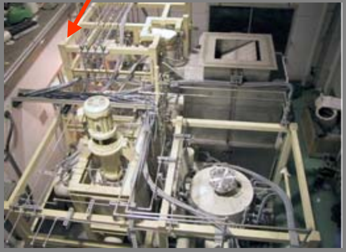






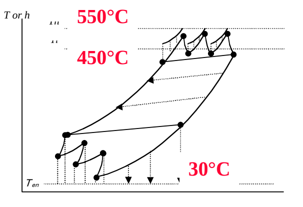
MHD on Flibe





TNT Loop : Max.20L/min @ 600° C

Closed He gas turbine cycle with three-stage compression/expansion



Concluding remarks

1. **Helical reactor is superior in steady state operation and**
    - **Reduced neutron wall loading < 2MW/m^2 with long-life blanket concept,**
    - **High density operation with reduced heat load on divertor.**
2. **On the 3D design, a simply defined blanket shape can be compatible with the ergodized magnetic layer which is essential on both of divertor and α-heating.**
3. **This simple methodology on LHD-type reactors can be used for design optimization towards DEMO, which is economically comparable to Tokamak.**
4. **The design parameters of FFHR2m2 have been totally improved, where a new ignition regime at SDC is an innovative alternative.**
5. **The large scale SC magnet system and their support posts are conceptually feasible.**
6. **Large R&D progress has been made on blanket and magnet materials.**
7. **The next key work is to realize the DPSS divertor concept compatible with replacement scenario and neutronics issues on TBR and nuclear shielding for SC magnets.**