Key issues in LHD-type reactor FFHR and present design activities

Akio SAGARA

collaborating with

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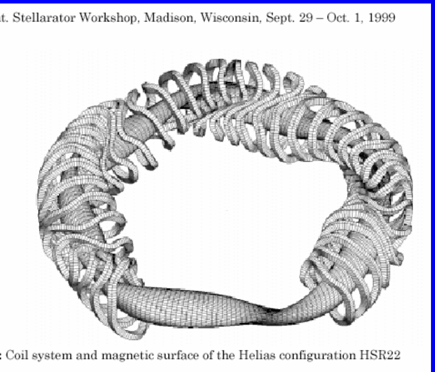
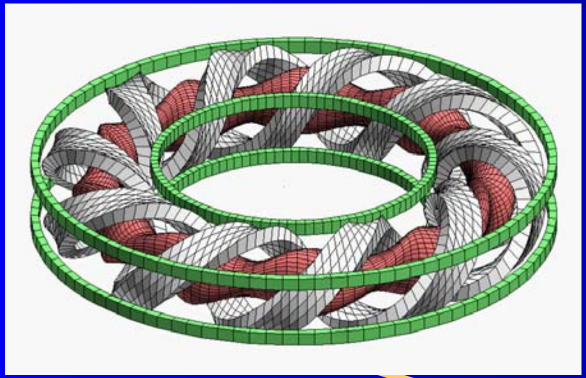
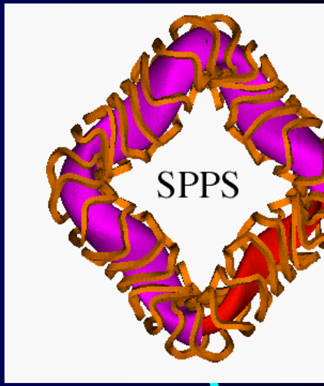
Japan-US Workshop on
Fusion Power Plants and Related Advanced Technologies with participation of EU
October 9-10, 2003 , UCSD,







	Heliotron-H	T-1	UETOR-M	MSR	CT6	HSR	FFHR-1	MHR-S	SPPS
Main field coils	2	3 (modular)	3 (modular)	2(modular)	2	50 (modular)	3	2(modular)	32(modul
oroidal field periods	15	20	6	6	6	5	18	10	4
Major radius	21m	29.2m	24.1m	20.2m	6.57m	20m	20m	16.5m	13.95m
erage plasma radius	1.8m	2.3m	1.7m	1.8m	1.74m	1.6m	2m	2.35m	1.6m
oroidal field on axis	4T	5T	5.5T	6.4T	6T	5T	12T	5T	4.95T
Average beta	6%	3.54%	5%	4%	4.70%	4-5%	0.70%	5%	5%
Fusion output	3.4GW	4.3GW	5.5GW	4GW	1.8GW	2-3GW	3GW	3.8GW	2.29GW
T-breeder	Li2O	Li	Li17Pb83		6Li17Pb83		Flibe		Li
Structure material	SUS		HT-9		HT-9		JLF-1		V
References	7, 8	9	10	11	12	13	15, 16	17	18
	1974-1982	1978	1981	1981	1989 -	1992 -	1995 -	1996 -	1997
	Kyoto-U	MIT	Wisconsin-U	Los Alamos Princeton	ORNL	IPP-Garching	NIFS	NIFS	UCSD

Issues on compactness ➡ FFHR-2 (1998 -): LHD-type, 10m, 10T, 1.8%, 1GW

A. Sagara, J. Plasma Fusion Research, 74(1998) pp. 947-951

We need High Power Density

Need Low Failure Rate

資本費　　燃料費　　運転費

$$\frac{\text{年間経費}}{\text{年間発電量}} = COE = \frac{C\,i + \boxed{\text{replacement cost}} + O\,\&\,M}{P_{fusion}\;\boxed{\text{Availability}}\;M\;\eta_{th}}$$

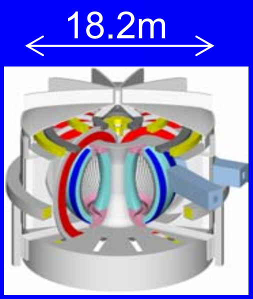
発電容量　　設備利用量

Energy Multiplication

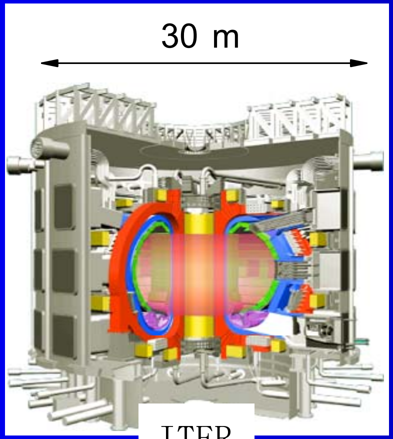
Need High Temp. Energy Extraction

After M. Abdou (in APEX)

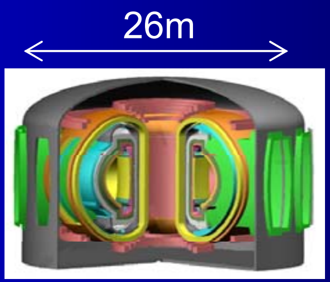
Importance of Weight Power Density



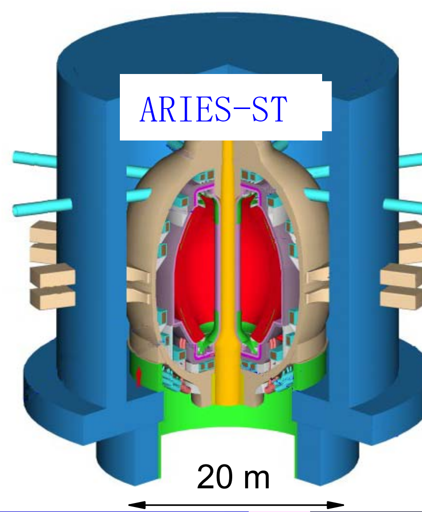
VECTOR-OPT

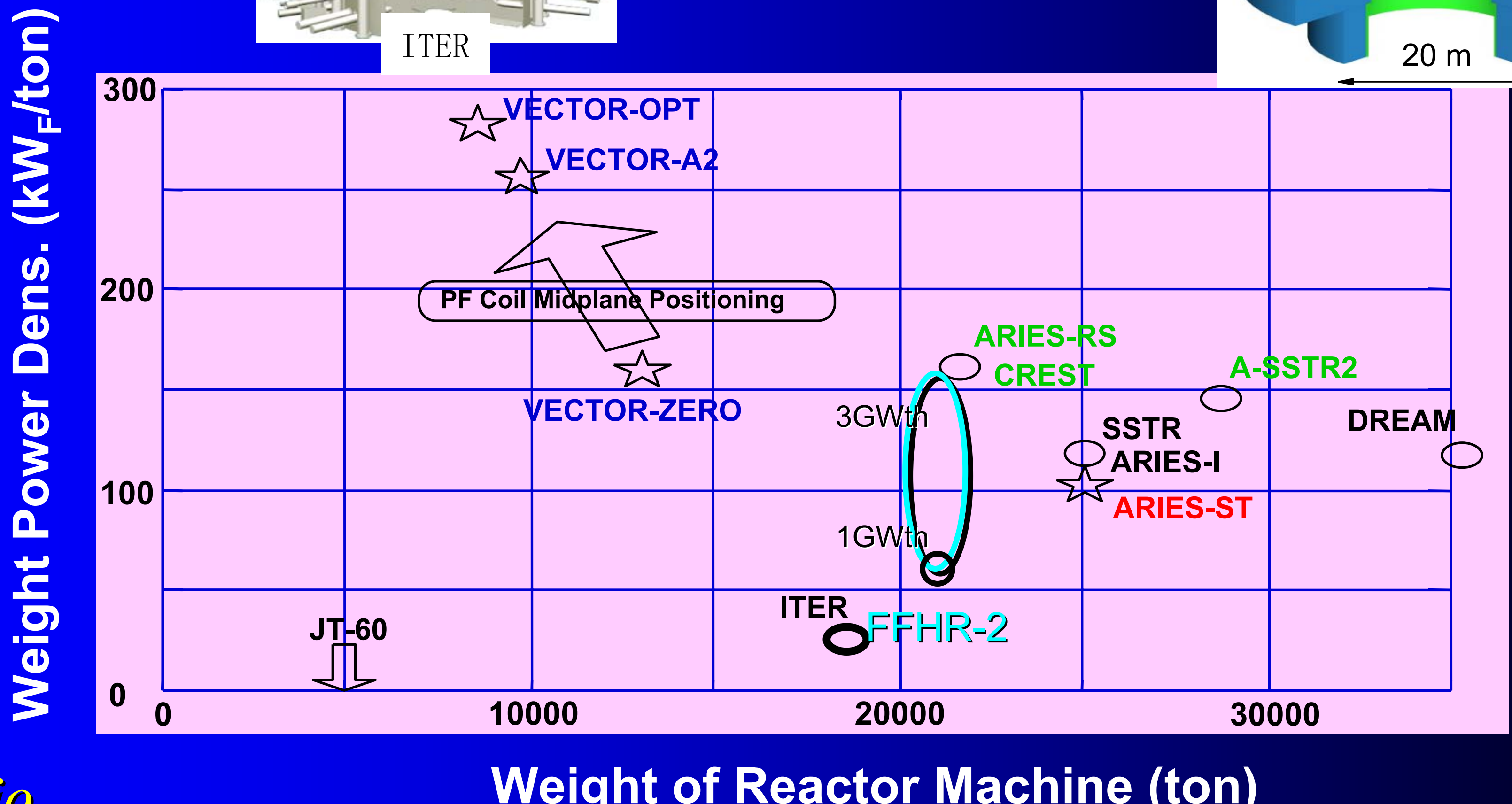


ITER



ARIES-RS





by S.Nishio

Saga

Main issues for Helical Demo

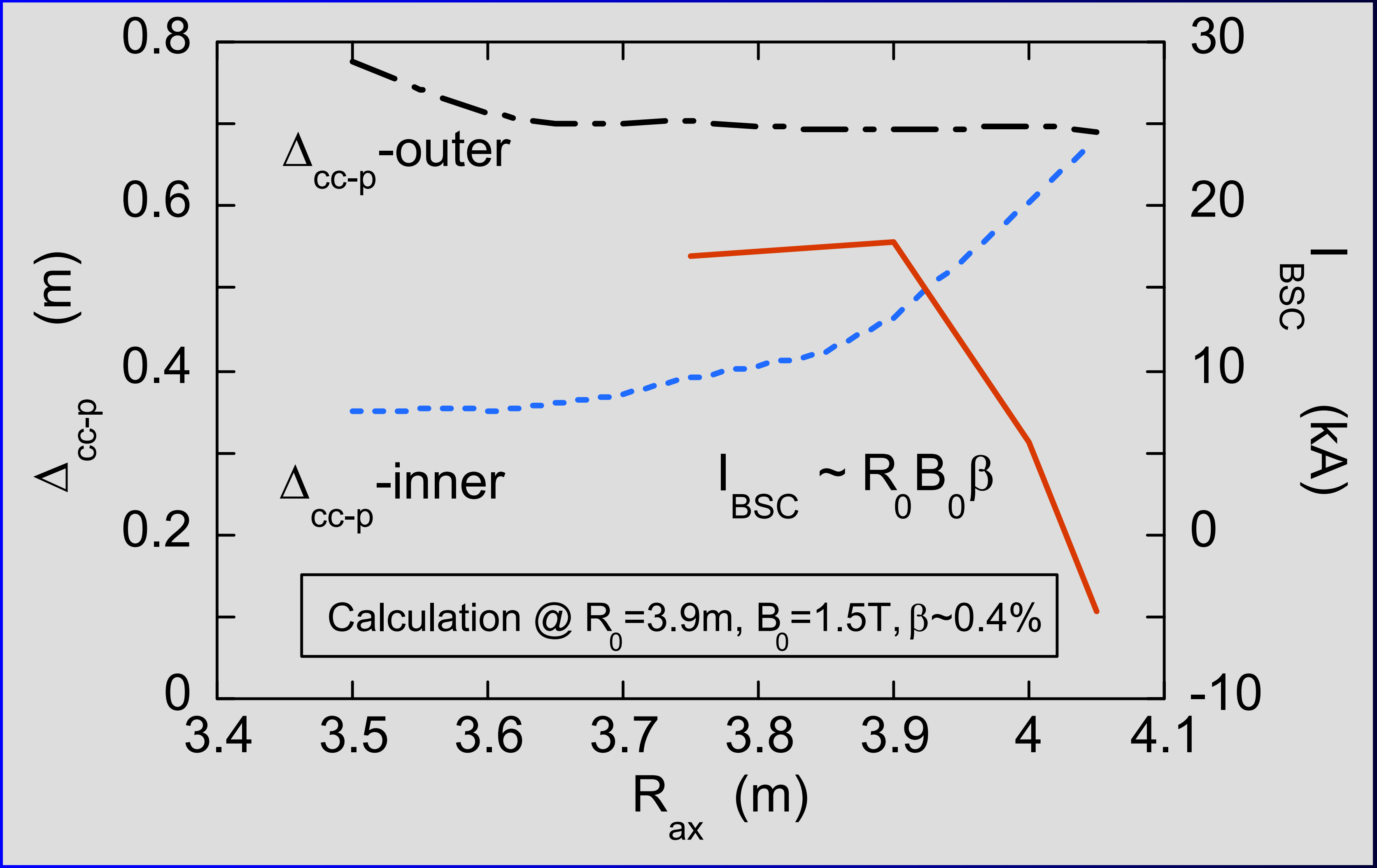
Key issue

blanket space

compact

1. Enhancement of energy confinement
2. Data base at reduced γ, outer shift, etc.
3. Reduction of heat load on the wall (α loss)
4. Divertor pumping systems (for α ash)
5. Fueling methods (to the center)
6. Heating devices (selection of method)
7. High performance blanket systems
8. Methods of maintenance & replacement
9. Advanced SC magnet systems
10. Structure materials and divertor materials
11. Reduction of construction cost
12. Reduction of COE

Outward shift of the magnetic axis

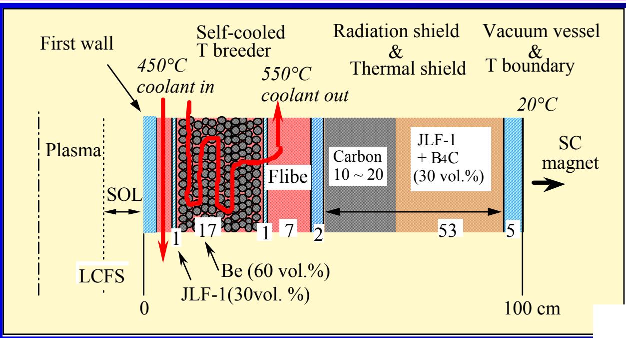


by K.Watanabe [ref. Nuclear Fusion, 41(2001) 63.]

FFHR Design Concept & Optimization

Enhancing inherent safety & Thermal efficiency

Molten-salt blanket with low MHD effect



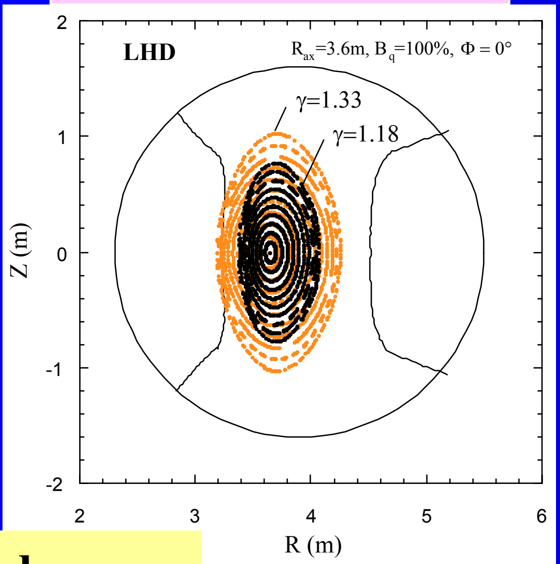
issue

eat transfer & tructure materials

issue

SC materials & Current density

Expansion of blanket area



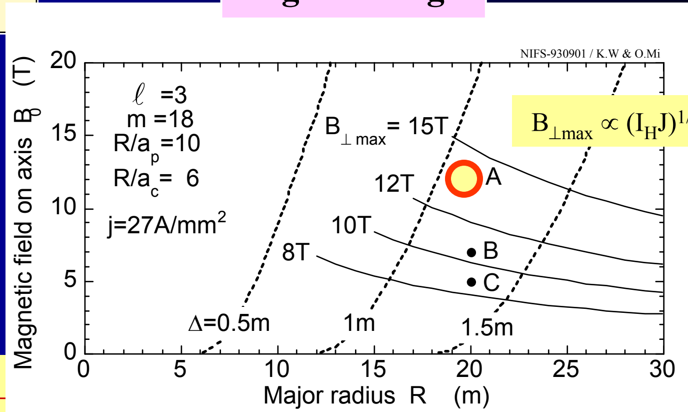
decrease

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

issue

MHD stability

High B design

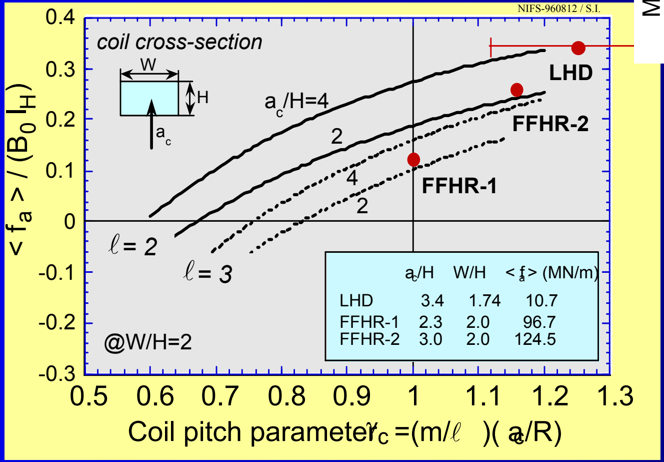


Ignition conditions for the 3GW fusion output

case A: Ti(0)=22keV, n(0)=2.0x10 ^{20}m^{-3}, < β> =0.7 %, h_H=1.5

case B: Ti(0)=24keV, n(0)=1.9x10 ^{20}m^{-3}, < β> =2.2 %, h_H=2.25

case C: Ti(0)=29keV, n(0)=1.5x10 ^{20}m^{-3}, < β> =4.5 %, h_H=3.5



	a_c/H	W/H	< f > (MN/m)
LHD	3.4	1.74	10.7
FFHR-1	2.3	2.0	96.7
FFHR-2	3.0	2.0	124.5

Reduction of magnetic force

Simplification of supporting structure

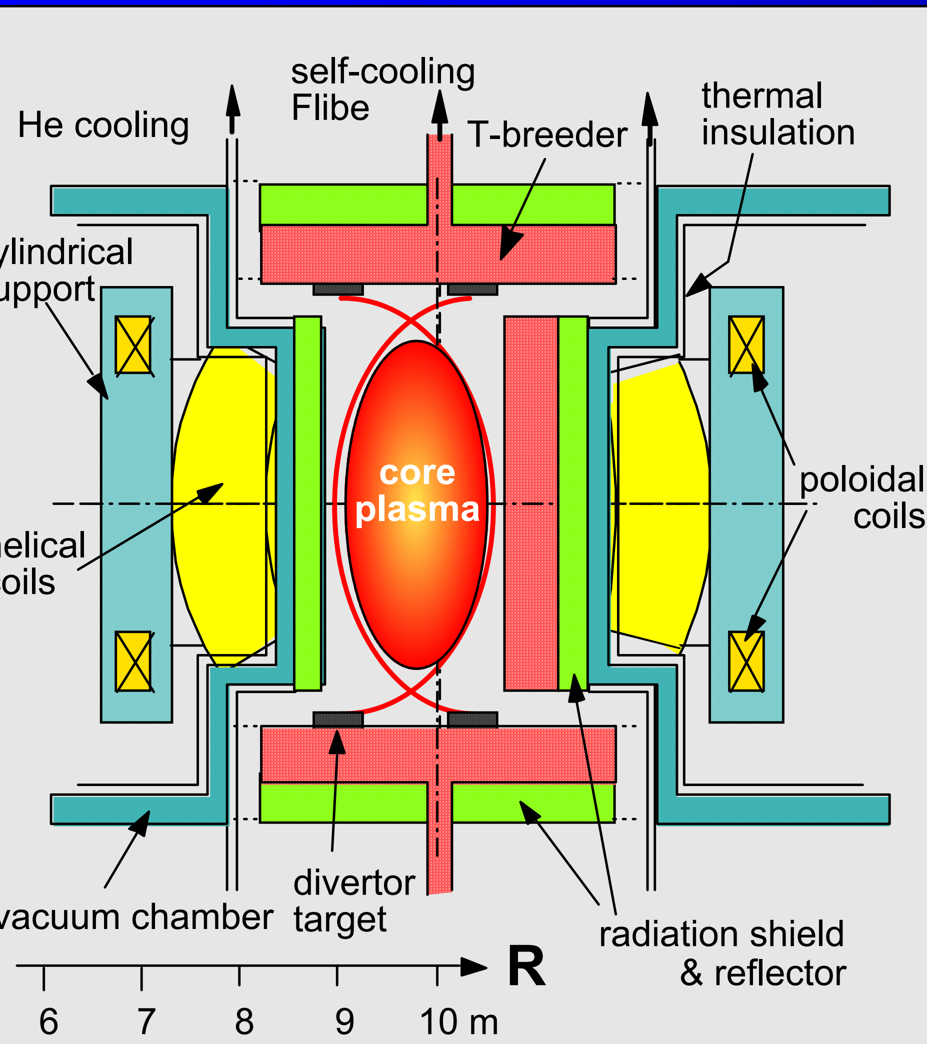
issue

Continuos winding helical coil

Saga

Neutron wall loading is reduced to $\Phi_n = 1.5 MW/m^2$ (120dpa/10

FFHR-2



Parameters	LHD	FFHR-1	FFHR-2
major radius : R	3.9	20	10 m
av. plasma radius : $\langle a_p \rangle$	< 0.65	2	1.2 m
fusion power : P(GW)	-	3	1 GW
external heating power : P_{ex}	< 20	100	100 MW
neutron wall loading : P_n	-	1.5	1.5 MW/m2
toroidal field on axis : B	4	12	10 T
average beta : $\langle\beta\rangle$	> 5	0.7	1.8 %
enhancement factor of τ_{LHD}		1.5	2.5
plasma density : $n_e(0)$	1.E2	2.E20	2.8E20 m-3
plasma temperature : $T_e(0)$	> 10	22	27 keV
effective ion charge : Z_{eff}		1.5	1.5
alpha heating efficiency : $\eta_{\alpha h}$	-	0.7	0.7
alpha density fraction : f_α	-	0.05	0.05
synchrotron reflectivity : R_{ref}	-	0.9	0.9
hole fraction : fh	-	0.1	0.1
av. heat load on divertor	< 10	1.6	1.5 MW/m2
number of pole :	2	3	2
toroidal pitch number : m	10	18	10
pitch parameter γ	1.12<1.25 <1.37	1	1.15
coil modulation α	+ 0.1	0	+ 0.1
av. helical coil radius : $\langle a_c \rangle$	0.975	3.33	2.30 m
coil to plasma clearance δL	0.03	1.1	0.70 ~ 1.25
coil current : I_H	7.8	66.6	50 MA/coil
coil current density : J	(53)	27	25 A/mm2
max. field on coils : B_{max}	(9.2)	16	13 T
stored magnetic energy	1.64	1290	147 GJ
construction cost	50 Byen		

A.Sagara et al., 17th IAEA _FTP-03 (1998) and ISFNT-5(1999, Rome)

Q_p~30, η_{th}~38%, availability~0.8

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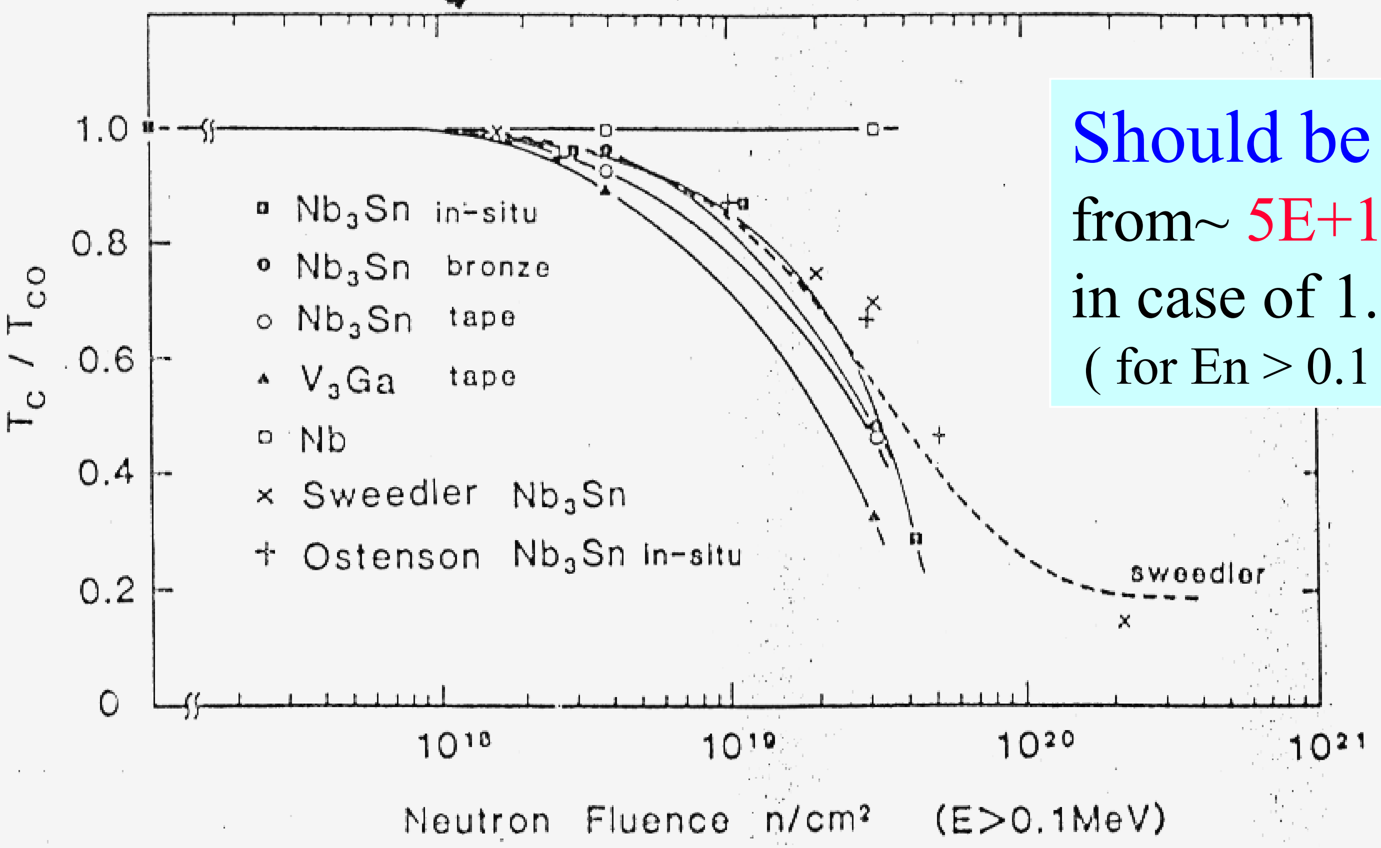
Dose effect in SC coil under fast neutron irradiation

$\sim 10^{18}$ n/cm^2 ⇓ =10year x 3E+13 n/m^{2}s



Should be reduced 5 orders

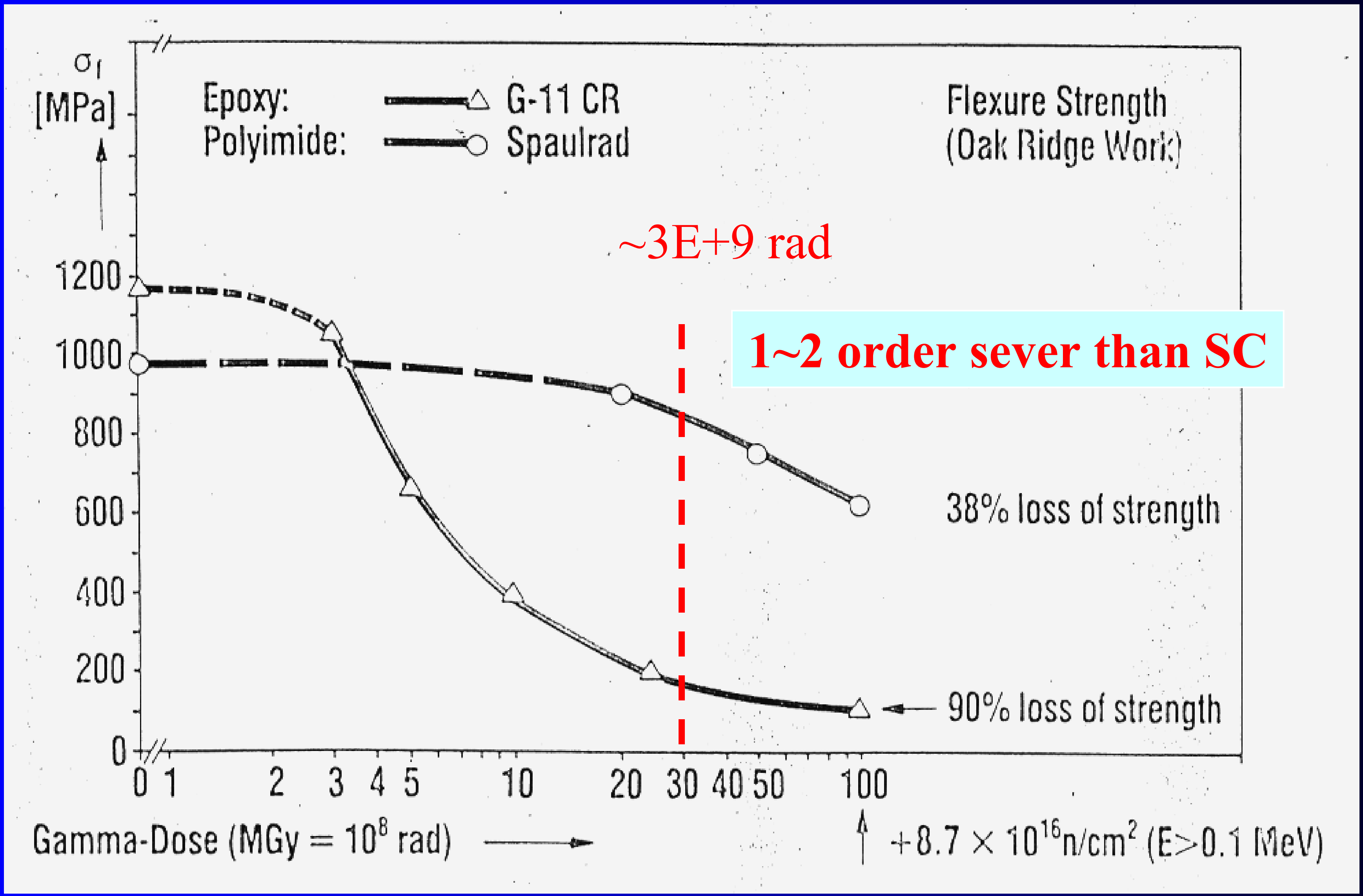
from~ 5E+18 n/m^{2}s

in case of 1.5MW/m^2

(for En > 0.1 MeV at the first wall)

H. Kodama et al., ICFRM-1, JNM 133&134 (1985) 819.

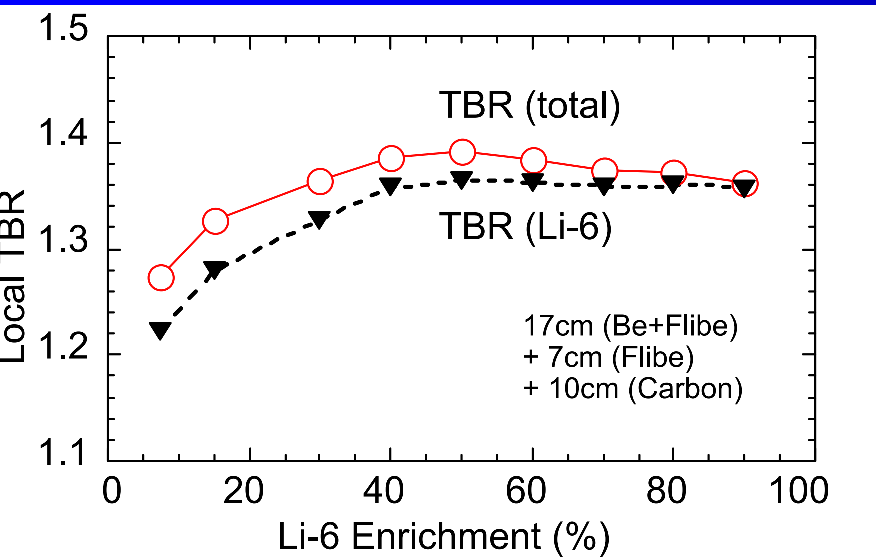
Dose effect in insulator due to gamma-ray

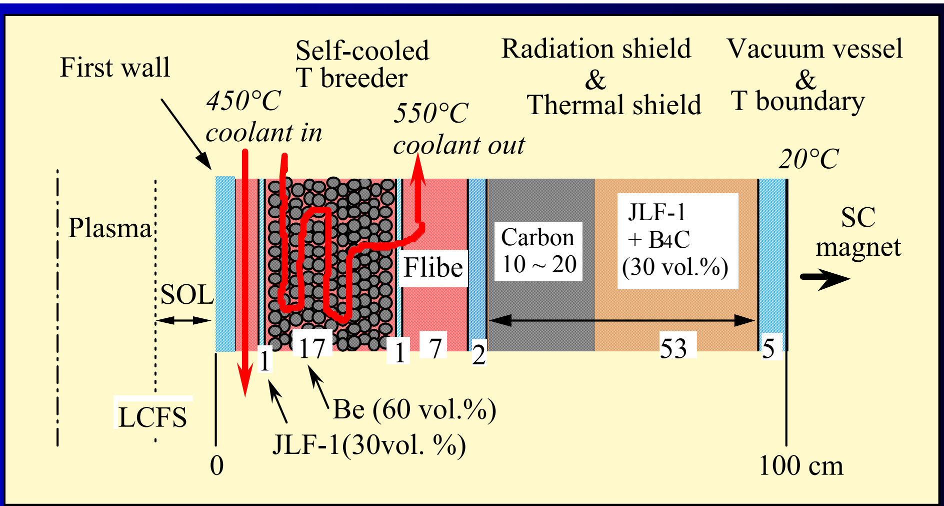


P.Komarek et al., ICFRM-3, JNM 155-157 (1988) 207. Sagar

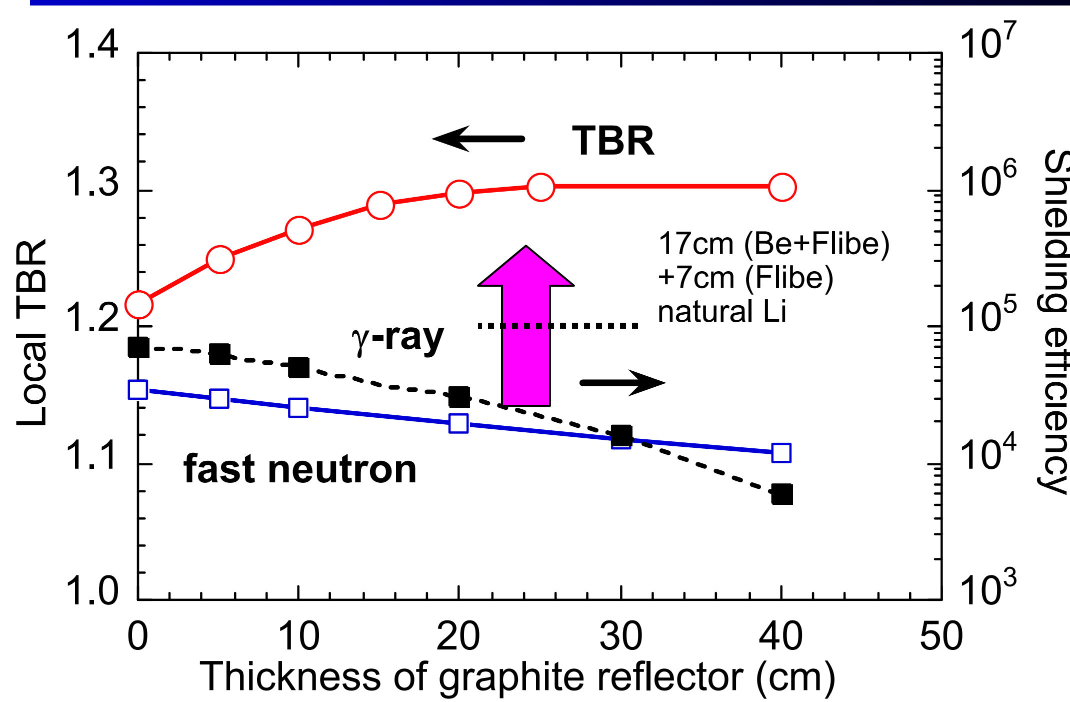
Self-cooled Tritium Breeding Blanket in FFHR

● Local TBR～1.4.





● However, shielding efficiency for SC magnet should be improved more than one order.



A.Sagara et al., Fusion Technol.,39 (2001) 753.

Another approach using HTcSC for magnet coils

Even the shielding efficiency is poor,

- **temperature margin** is wide,
- **controllability** becomes high at high T operation,
  (because specific heat ~ T^3)
- **SC joint** is promising and innovative.
  (Ohmic heat on cryo. < a few % of P_f)

Conventional SC joint design

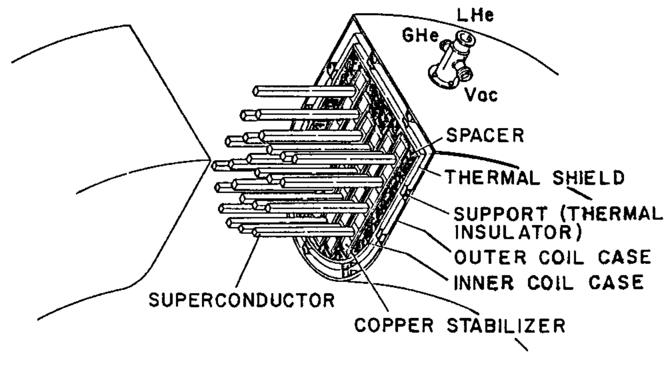
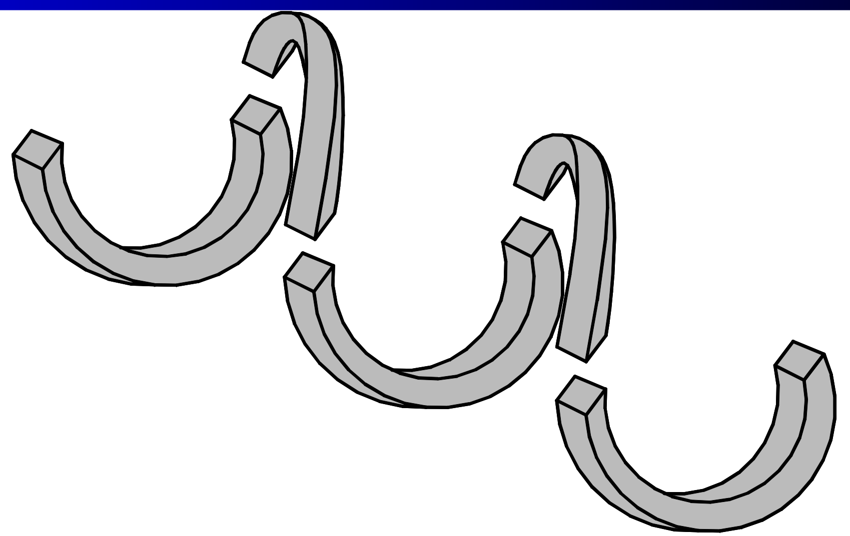


Fig. 2 Details of the joint section using the "Inserting Method".

T.Horiuchi et al., Fusion Technol. 8 (1985) 1654.

Re-mountable coils using HTcS.C. joints

H.Hashizume et al., J.Plasma Fusion Res. SERIES 5 (2001) 532.



Sagar

Modified FFHR-2

pressented in 12th Intenational Toki Conference (2002)

IPP Garhing : H. Wobig, J. Kisslinger

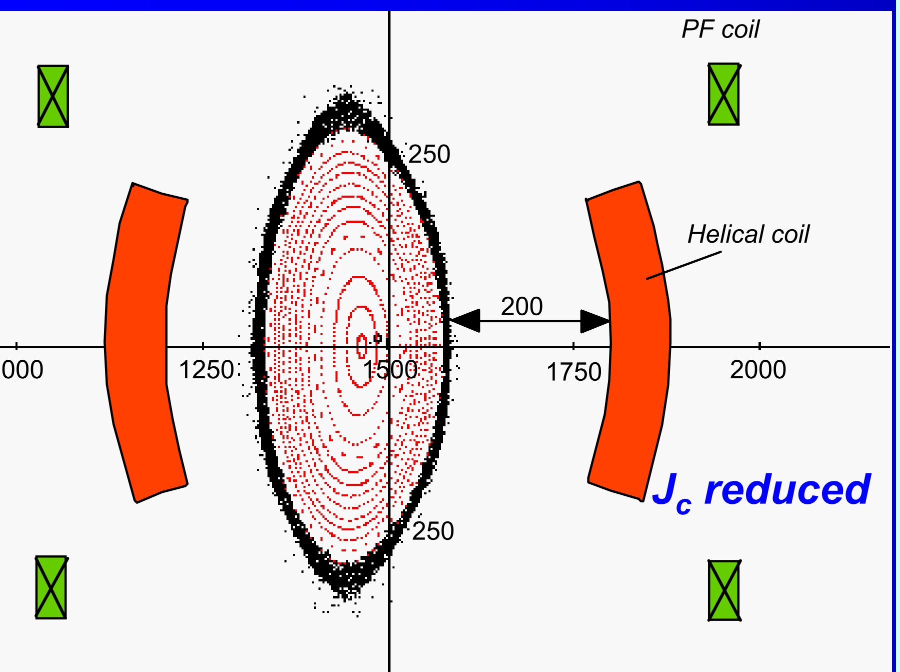
ncrease of the FFHR2 configuration by 50%

eduction of the magnetic field to 6 T on axis

o provide more space for blanket and shield

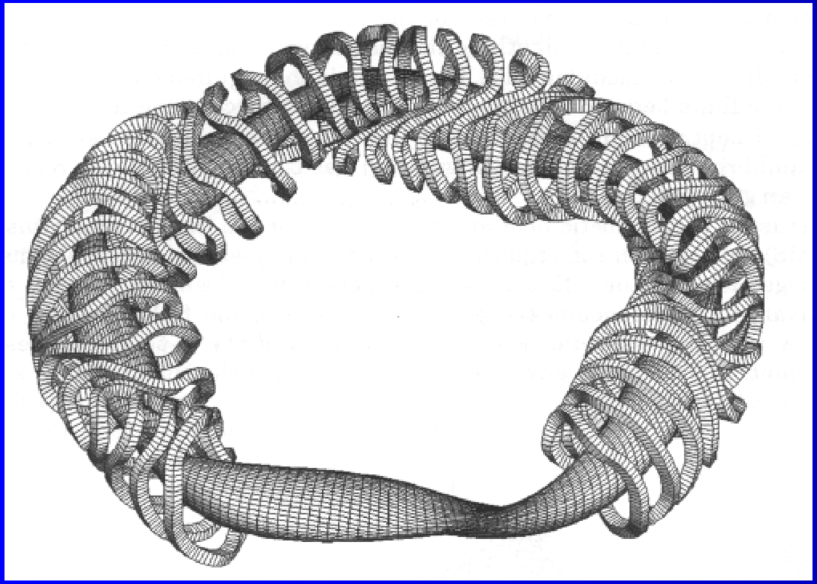
o use NbTi in the helical windings.

One way
for optimization.
But 3 times of the weig



	FFHR-2a	FFHR-2b (modified)	
Major radius	10	15	m
Average coil radius	2.30	3.45	m
Average plasma radius	1.2	1.8	m
Number of field periods	10	10	
Field on axis	10	6	T
Max. field on coils	13	7.8	T
Electron Density(0)	3.0×10^{20}	3.0×10^{20}	[m
Electron Temp(0)	15.0	15.0	[ke
Av. Beta	0.96	2.66	[%
Fusion Power	0.62	2.09	[GV
Heating Power	94	333	[MV
Energy Conf. Time	1.72	1.64	[s

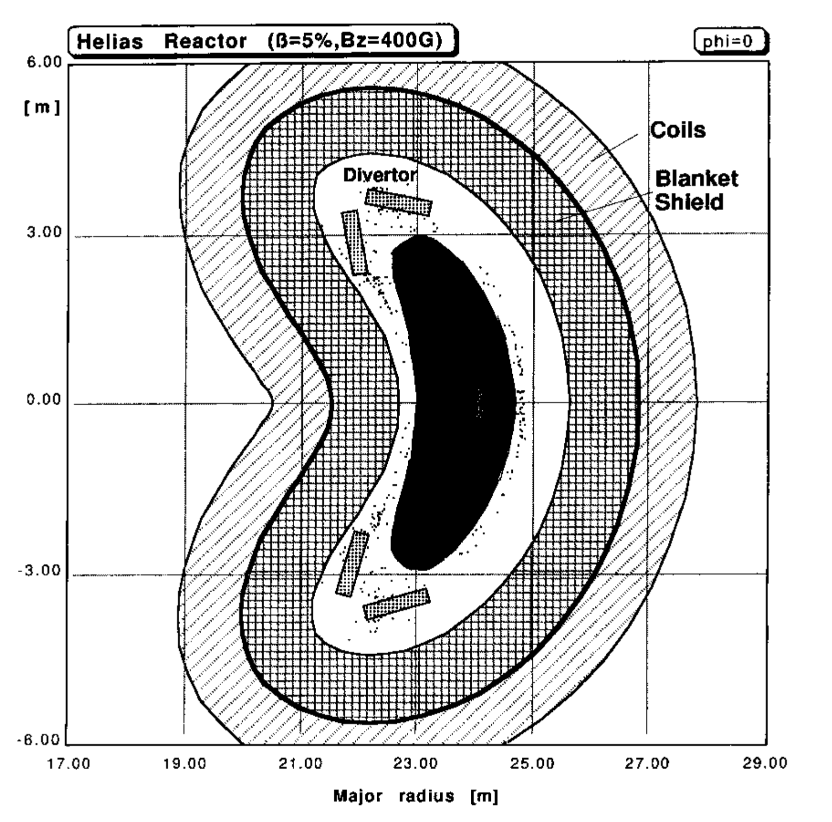
Blanket in HSR ～1.5m



- He Cooled Li_4SiO_4 (KfK design)
- He Cooled Pb-17Li (FZK design)
- Water cooled Pb-17Li (ITER design)

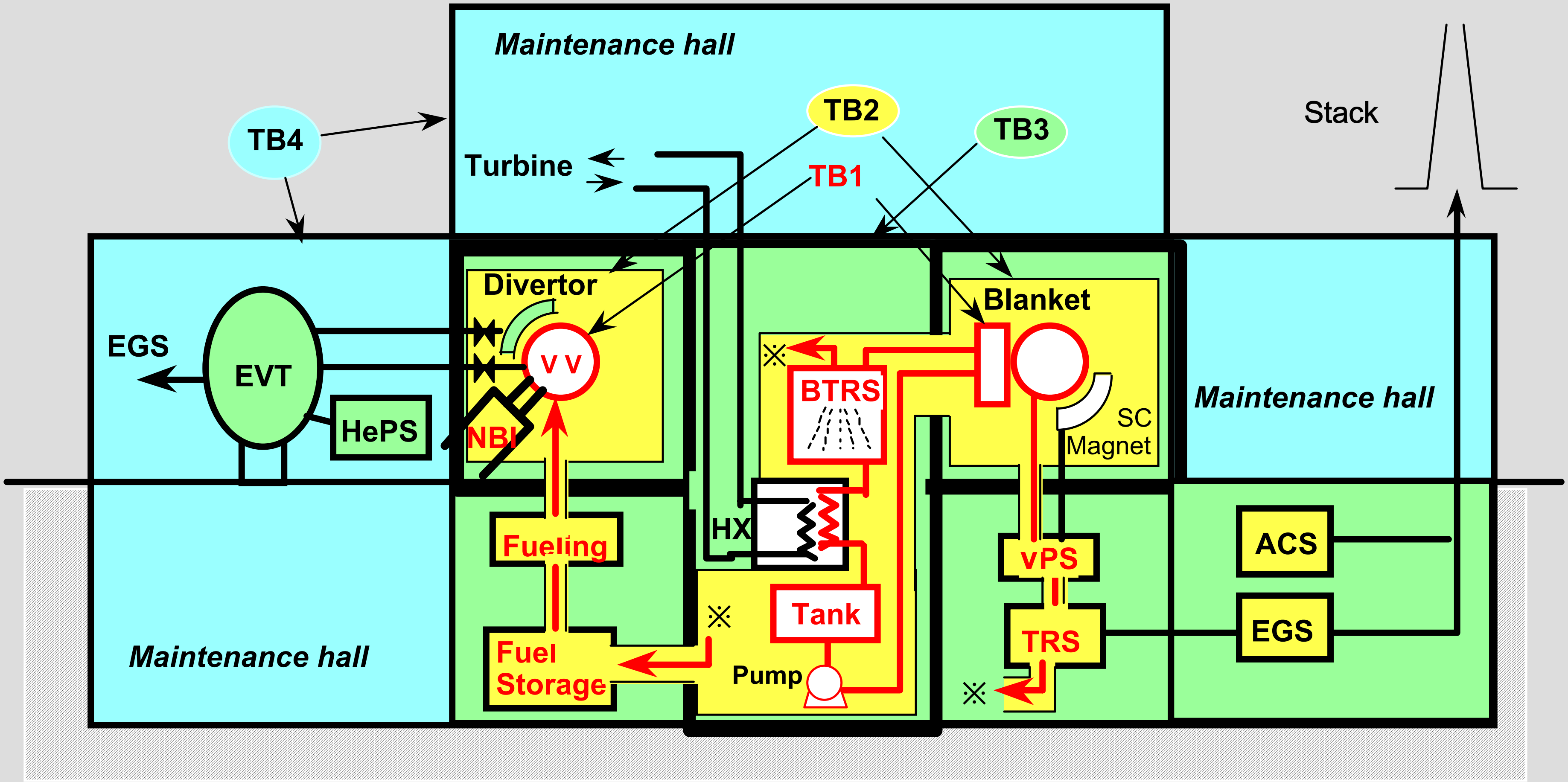
TABLE I: Main parameters of HSR4/18 and HSR5/22

	HSR4/18	HSR5/22
Major radius [m]	18	22
Average minor radius [m]	2.0	1.8
Plasma volume [m^3]	1420	1407
Iota(0)	0.83	0.84
Iota(a)	0.96	1.00
Average magnetic field on axis [T]	5.0	5.0
Maximum magnetic field on coils [T]	10	10.3
Number of coils	40	50
Magnetic energy [GJ]	80	100



Availability to use the center space in a large R helical rector

Tritium confinement boundary in FFHR



BTRS : Blanket Trium Recovery system
EVT : Expansion Volume Tank
HePS : Helium Purification System

VPS : Vacumm Pumping System
TRS : Tritiume Recorvery & Storage System
EGS : Exhaust Gas treatment System
ACS : Air Cleanup System

970321 A.S

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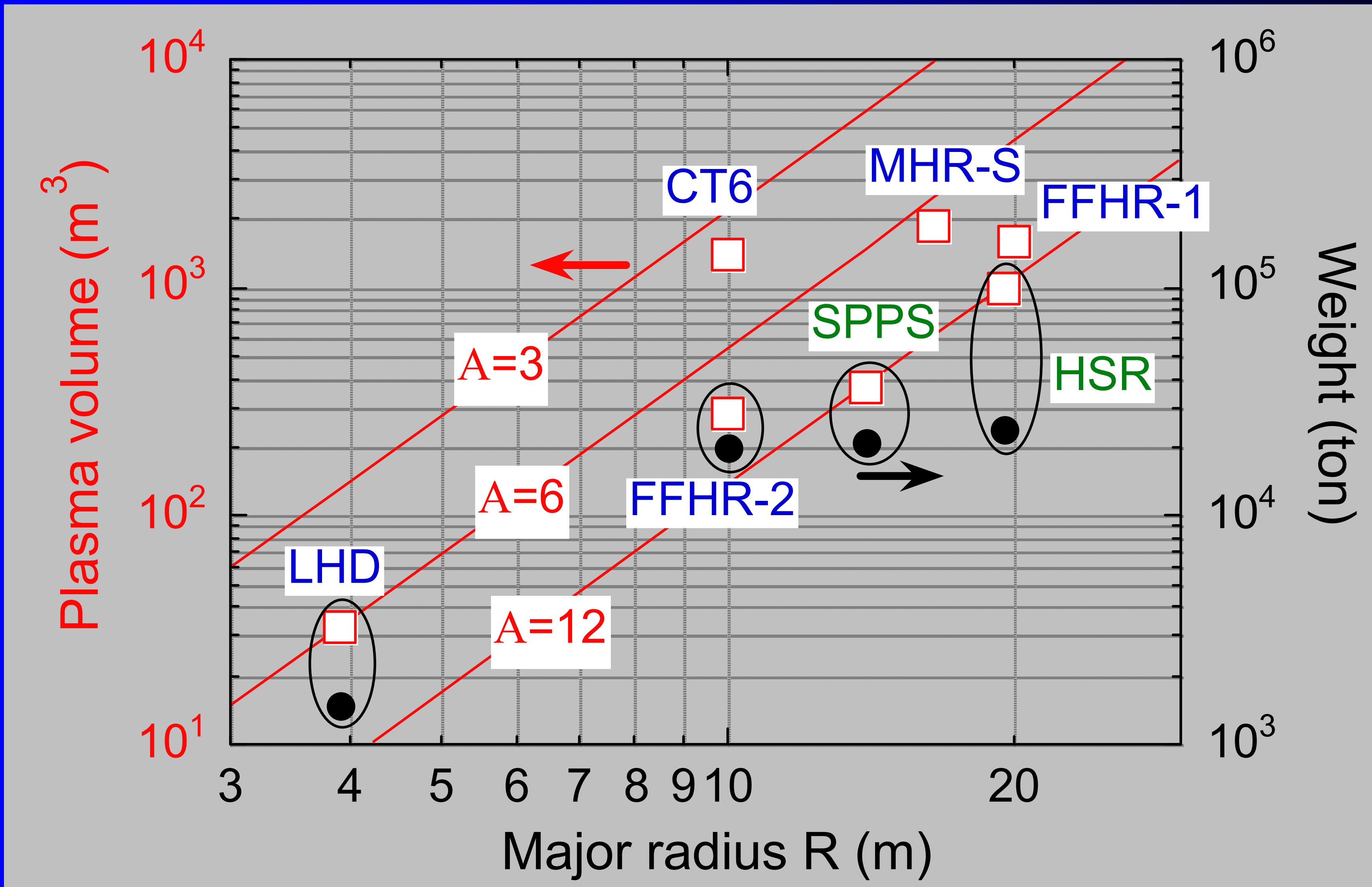
Weight $\propto V_p = 2\pi^2 R^3 / A^2$



Sagar

Reactor design activity in NIFS collaboratio

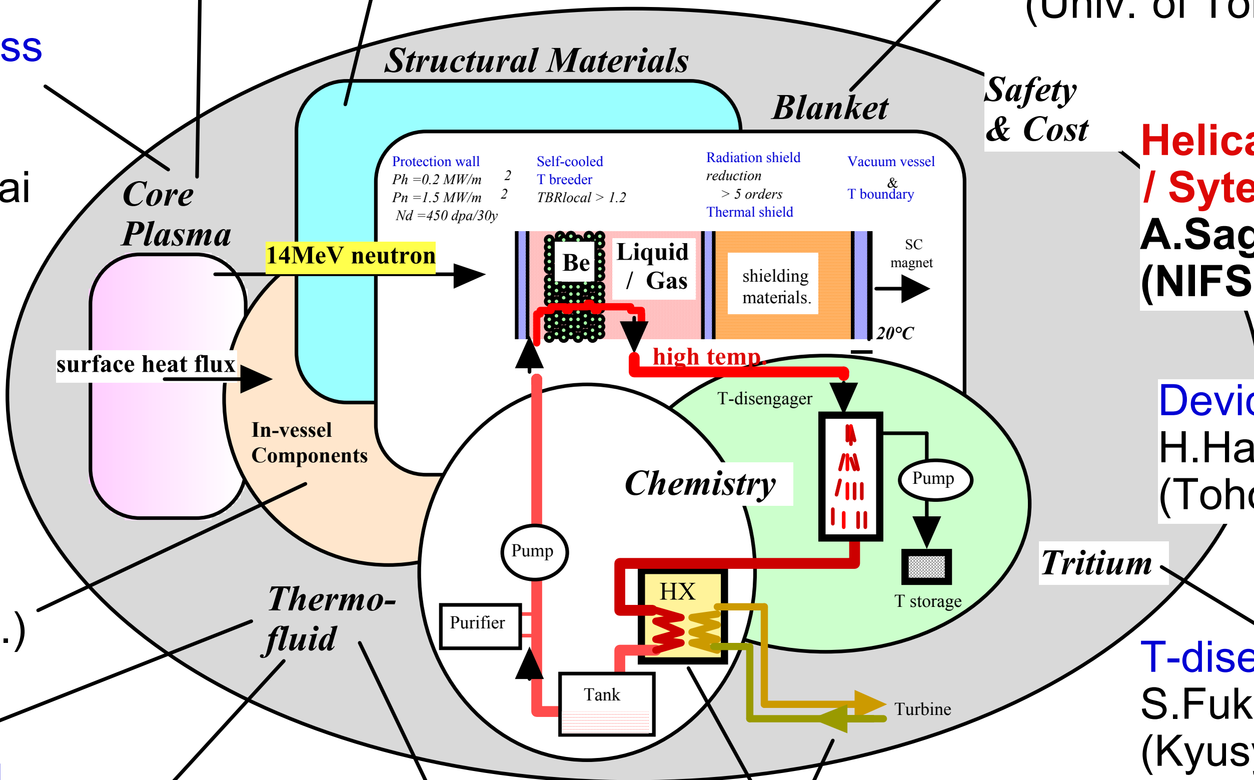
• on going
cf. NIFS annual rep

Helical core plasma
K.Yamazaki
(NIFS)

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

Blanket system
S,Tanaka
(Univ. of Tokyo)

gnition access
& heat flux
D.Mitarai
Kyusyu Tokai
Univ.)



Helical reactor design
/ Sytem Integration
A.Sagara
(NIFS)

Device system code
H.Hashizume
(Tohoku Univ.)

Advanced
first wall
T.Norimatsu
(Osaka Univ.)

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

Advanced
thermofluid
T.Kunugi
(Kyoto Univ.)

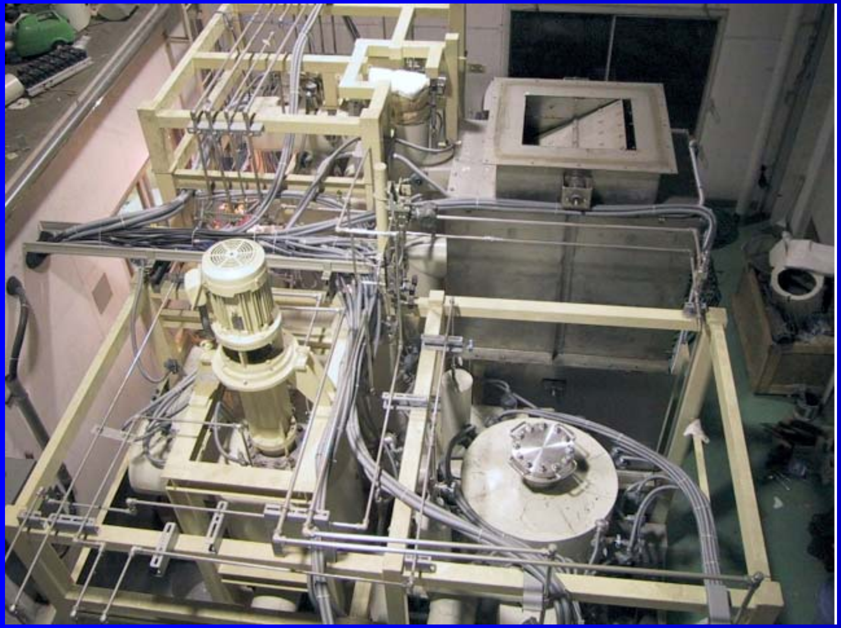
Thermofluid
MHD
S.Satake
(Tokyo Univ. Sci.)

Thermofluid
system
K.Yuki
(Tohoku Univ.)

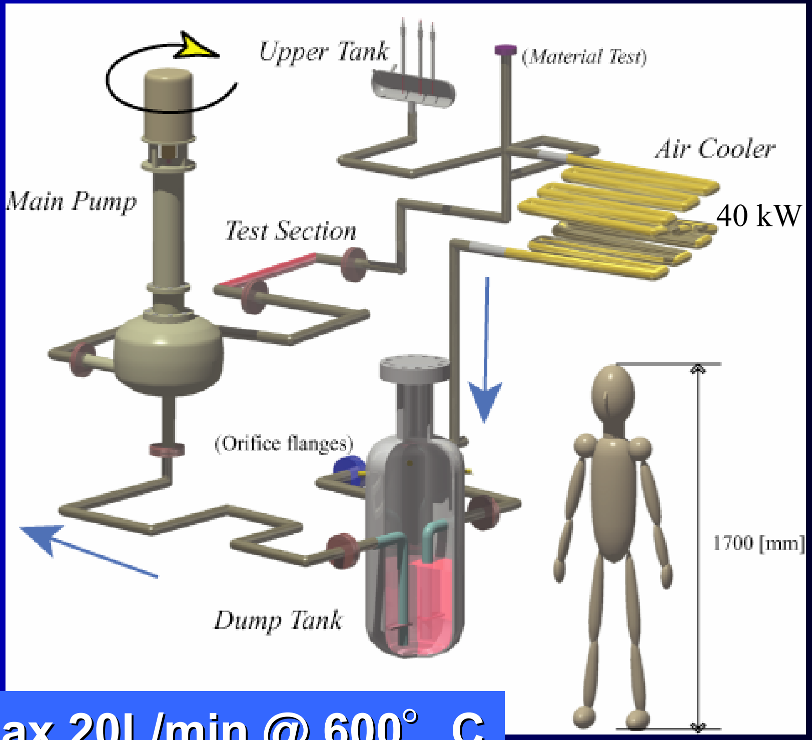
Heat exchanger & gas turbine system
A.Shimizu
(Kyusyu Univ.)

June 6. 2003, A.Sagara

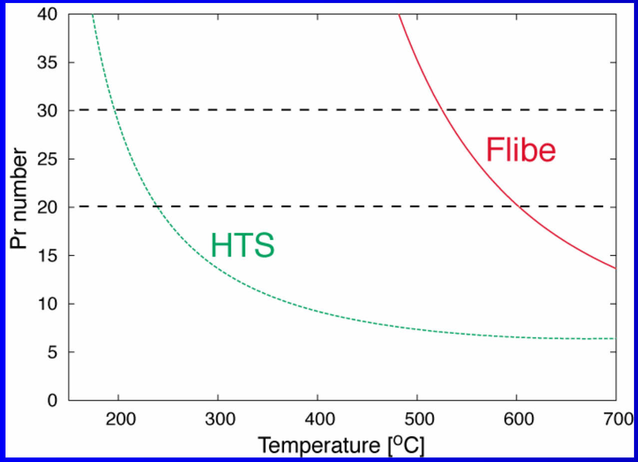
Tohoku-NIFS Thermofluid loop for molten salt



(Fabricated by IHI Co., Ltd.)



Max.20L/min @ 600° C
~ 0.1m^3 @ 0.7MPa.



Now HTS (simulant for Flibe) is used.

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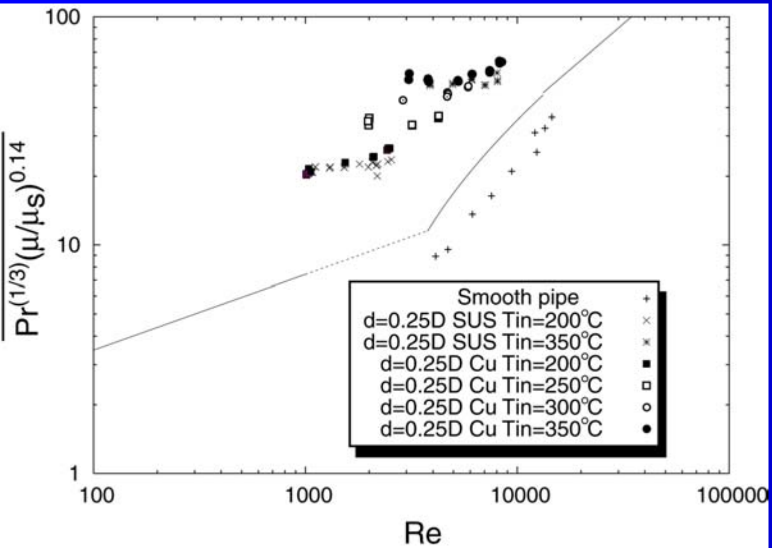
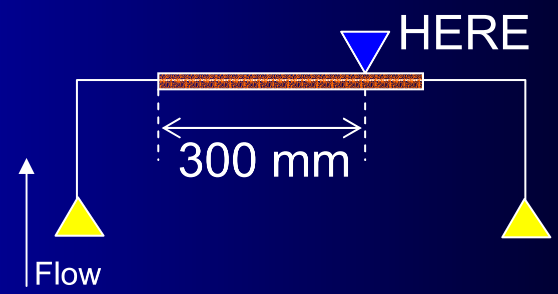
Heat transfer enhancer at low flow rate

… Packed-bed tube in TNT loop



.Re vs heat transfer characteristics (at 300 mm)



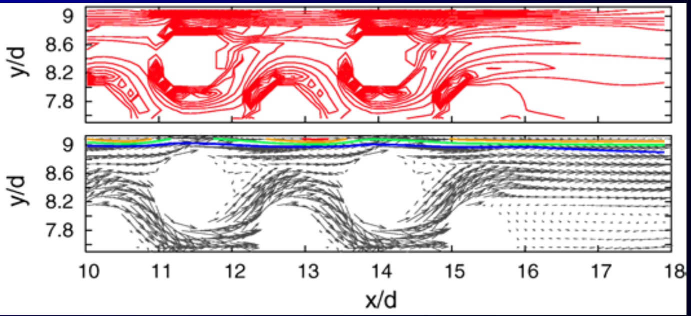
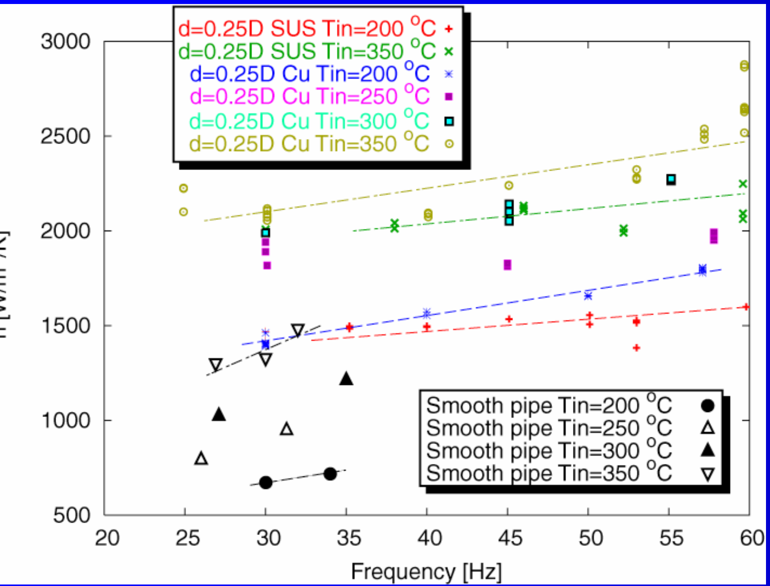


Fig. Flow structure near wall
(Result of 2D numerical simulation)



About 3 times higher than turbulent heat transfer

ig.Pump frequency vs heat transfer coefficient (at 300 mm)

MHD effects

- **Evaluation method** for degradation of the insulator, induced pressure increase.
- **Proposing new method** to reduce the MHD pressure increase.
- **Clarification of feasibility** for the liquid blanket system.

2003 NIFS Collaboration Program

H.Hashizume (Tohoku Univ.)
K.Muroga (NIFS)
A.Sagara (NIFS)
S.Tanaka (Tokyo Univ.)
H.Horiike (Osaka Univ.)
T.Kunugi (Kyoto Univ.)
Other 8 reserchers

JUPITER-II project

Task 1-1-A
Task 1-1-B
Task 1-2-A
Task 3-1

Velocity profiles (B=10[T])

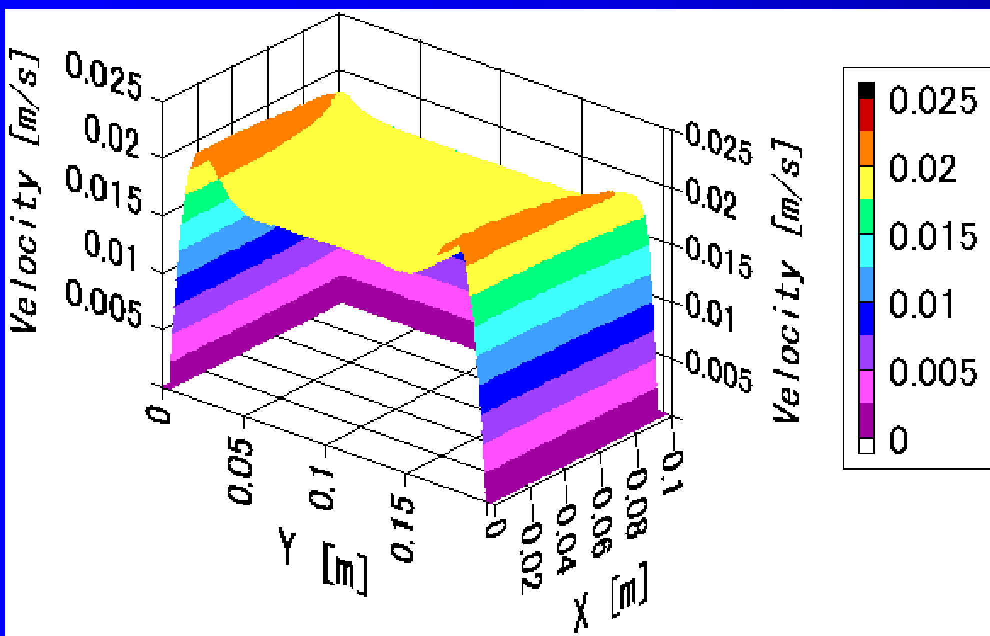
By H. Hashizume

[1] Flibe

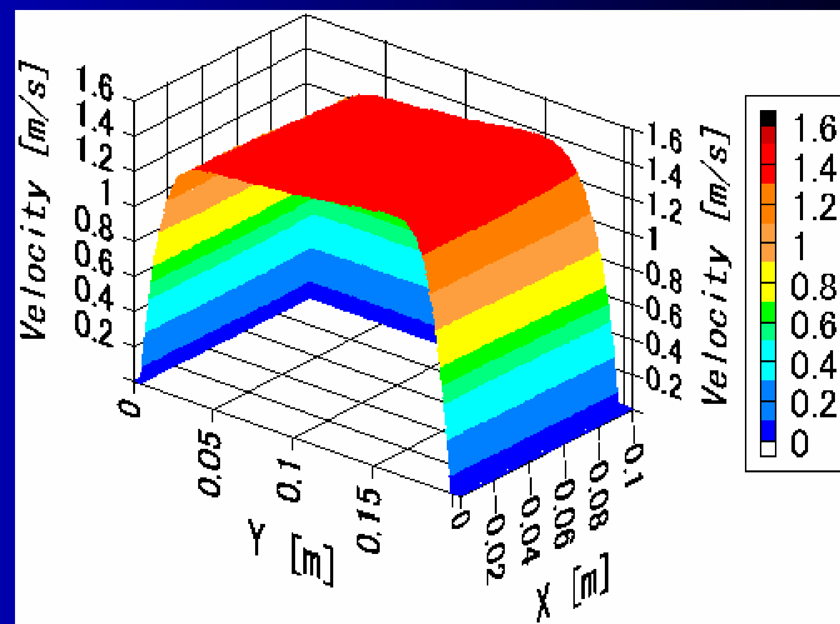
(a) Without Insulator

(b) Coated with Insulator

$$\left(\sigma_{\mathrm{i}} \big/ \sigma_{\mathrm{HT-9}} = 10^{-9} \right)$$



M-shape



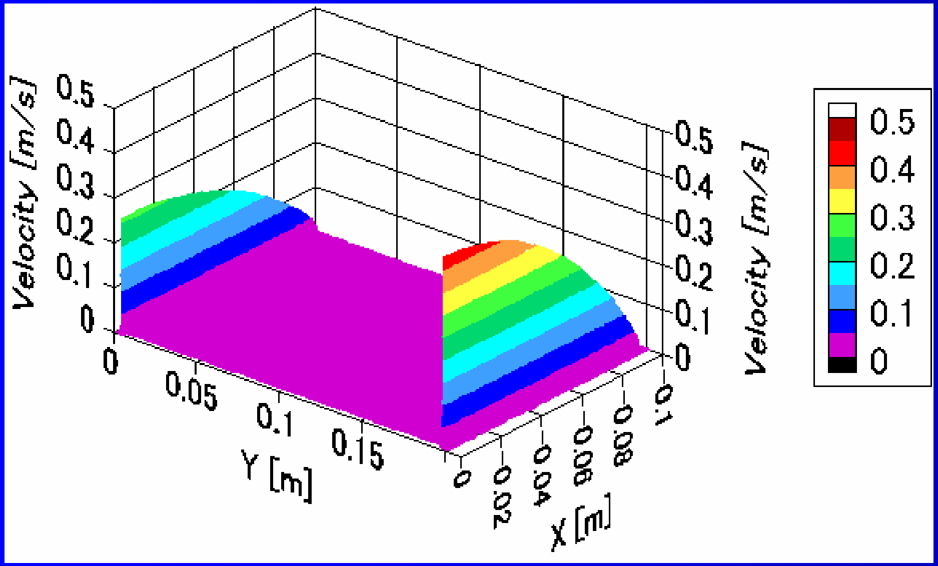
Flat-shape

[2] Lithium

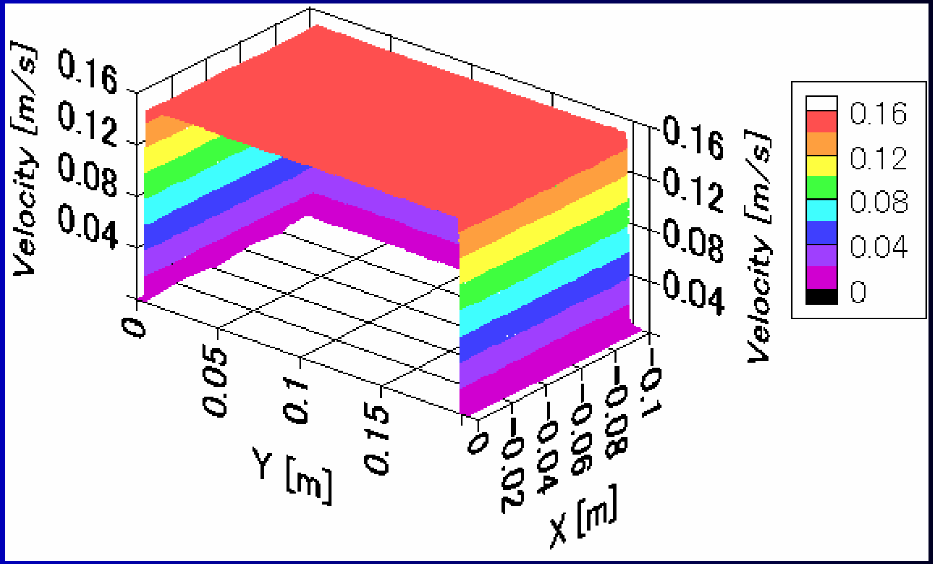
(a) Without Insulator

(b) Coated with Insulator

$\left(\sigma_{i} / \sigma_{HT-9} = 10^{-9} \right)$



M-shape



Flat-shape

H.Hashizume et al., submitted to ISEM 2003: The 11th International Symposium on Applied Electromagnetics and Mechanics, Versailles, France

Sagar

Parameter Requirements for D-^{3}He Helical Reactors

O.Mitarai [1)],

A. Sagara, S. Imagawa, Y.Tomita, K.Y.Watanabe and T.Watanabe [2)]

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Submitted to ITC-13, Dec. 2003 @Toki, Japan

D-^{3}He ignition is possible in the helical reactor with

- R=14.47m, $\langle a_p \rangle$=2.585 m, B_o=6 T, and $\langle\beta\rangle$~18 %
- for the confinement enhancement factor of γ_{HH} = 2.4
- with the external heating power of P_{EXT}= 300 MW
- for D:^{3}He fuel ratio of 2:1.

Summary

Within present-day technology, medium-size designs are one way for helical demo-reactors.

However, innovative concepts such as re-mountable coils using HTcS.C. joints are very attractive for compactness of LHD-type reactors.

Design activities in NIFS collaborations are expanding into Thermofluid exp., MHD effects, D-^{3}He, (& IFE).

ブランケット交換の各種可能性の開拓

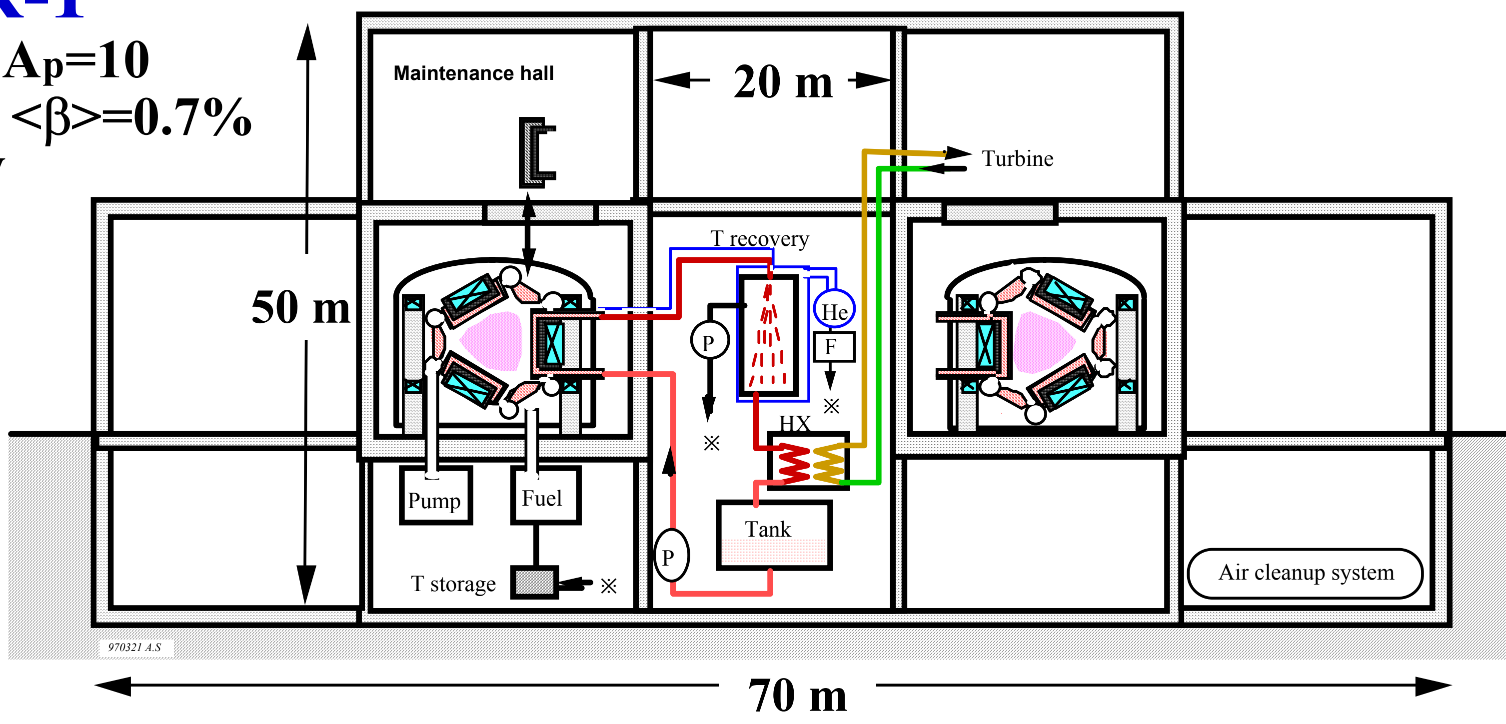
- 横ポートからの大型ユニットの取り出し（LHD実績）
- ユニットのヘリカルスライド移動
- 上ポートからの取り出し
- 中性子壁負荷下げて交換頻度〜１回／１０年

FFHR-1

$R=20$m, $A_p=10$
$B_0=12$T, $\langle\beta\rangle=0.7\%$
$P_f=3$GW

: =3
m=18
γ=1

Flibe
-JLF-1



FNT-3(1994, UCSD), ITC-6(1994, Toki), 16th IAEA(1996, Montreal), ISFNT-4(1997, Tokyo),
C-8(1997, Toki), ICFRM-8(1997, Sendai), 17th IAEA(1998, Yokohama), ISFNT-5(1999, Roma), etc.

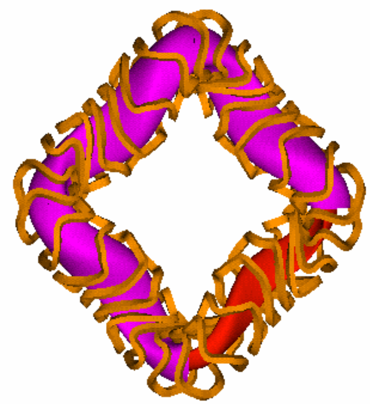
Sagar

Stellarator Power Plant Study (SPPS)

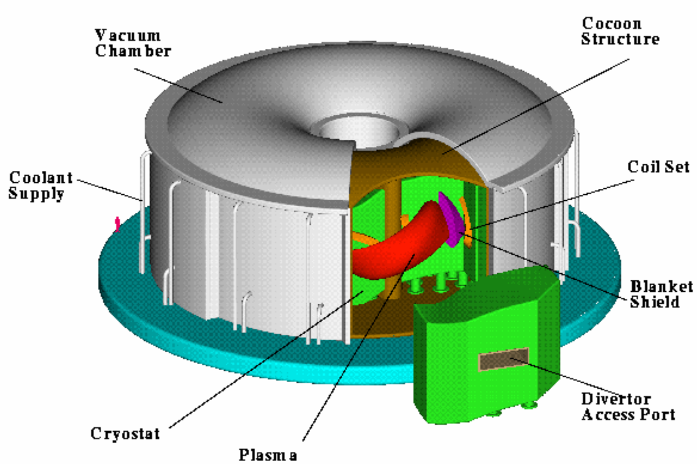
R. L. Miller (UCSD) for the SPPS Team

(1993 - 1997)

Top View of SPPS
Blanket/Shield and MHH Coils



SPPS Fusion Power Core System

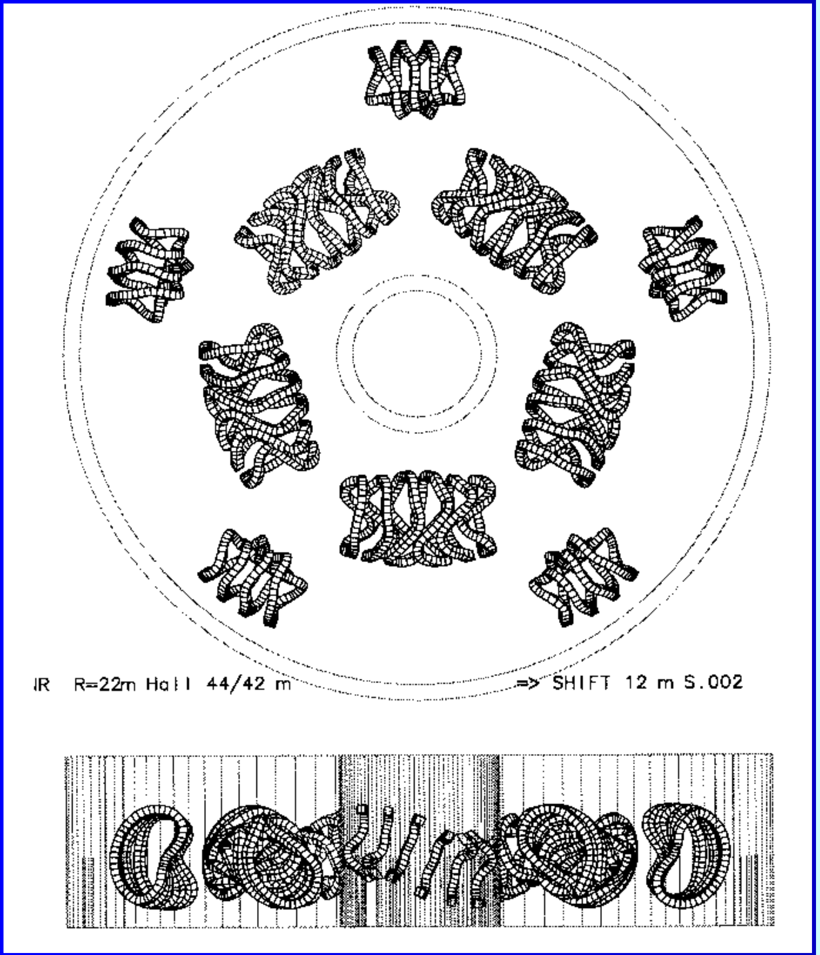


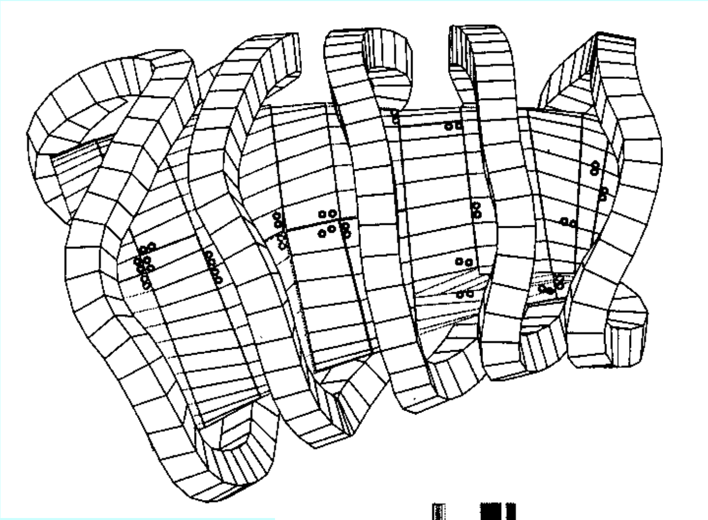
Major Parameters of the SPPS Power Plant

Number of field periods	4
Number of modular, non-planar coils	32
Plasma major toroidal radius, R_T (m)	13.95
Plasma half-width, a_p (m)	1.16
Plasma aspect ratio (cf., tokamak), $A = R_T/a_p$	12.08
Plasma aspect ratio, $A^* = R_T/r_p$	8.54
Circularized (average) plasma radius, r_p (m)	1.63
Plasma volume, V_p (m3)	734.7
Plasma beta	0.05
Lackner-Gottardi confinement multiplier, H	2.3
On-axis magnetic-field strength, Bo (T)	4.94
Peak magnetic-field strength at coil, Bc (T)	14.5
Stored magnetic energy, W_B (GJ)	80 (est.)
Fusion power, P_F (GWth)	1.73
Thermal conversion efficiency, h_{TH}	0.46
Thermal power, P_{TH} (GWth)	2.29
Gross electrical power, P_{ET} (GWe)	1.05
Net electrical power, P_E (GWe)	1.0
Recirculating power fraction, e	0.052
Plant capacity factor, p_f	0.76
Total direct cost (B$)(a)	2,249.
Total capital cost (B$)(b)	4,340.
Cost of electricity, COE (mill/kWeh)(c)	74.6

HSRにおける保守交換

以前はコイル移動方式、近年はポートからの交換方式





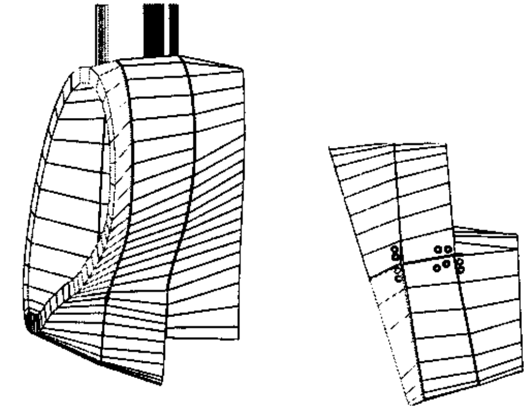
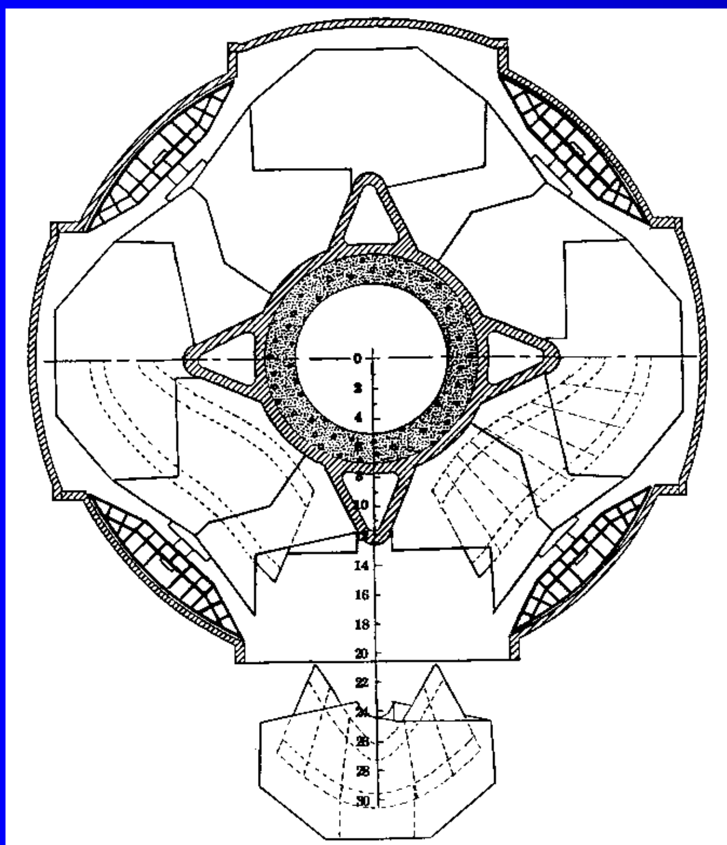


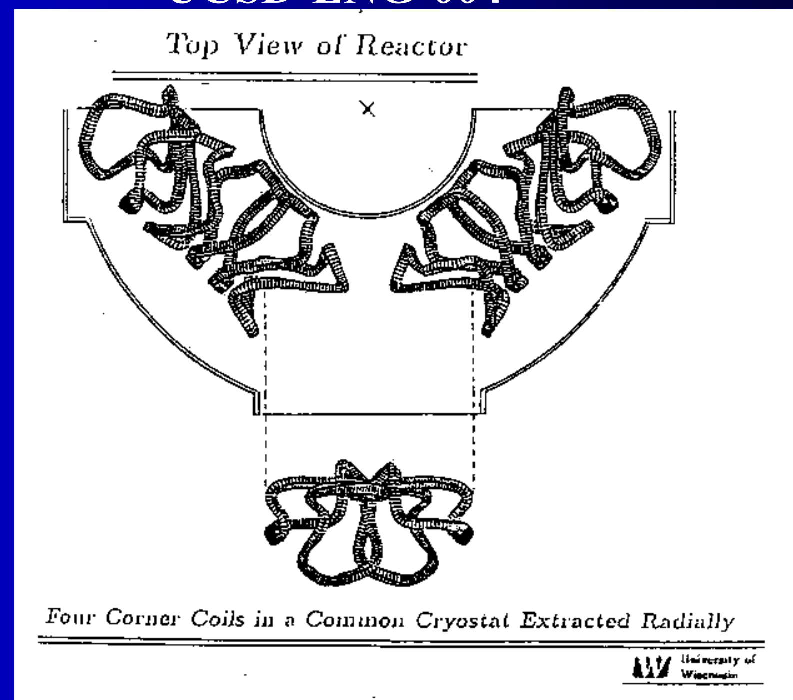
Fig. 10.12: Blanket module 3 consisting of 4 segments.

SPPSでの保守交換

コイル一体ユニットの引き出し方式

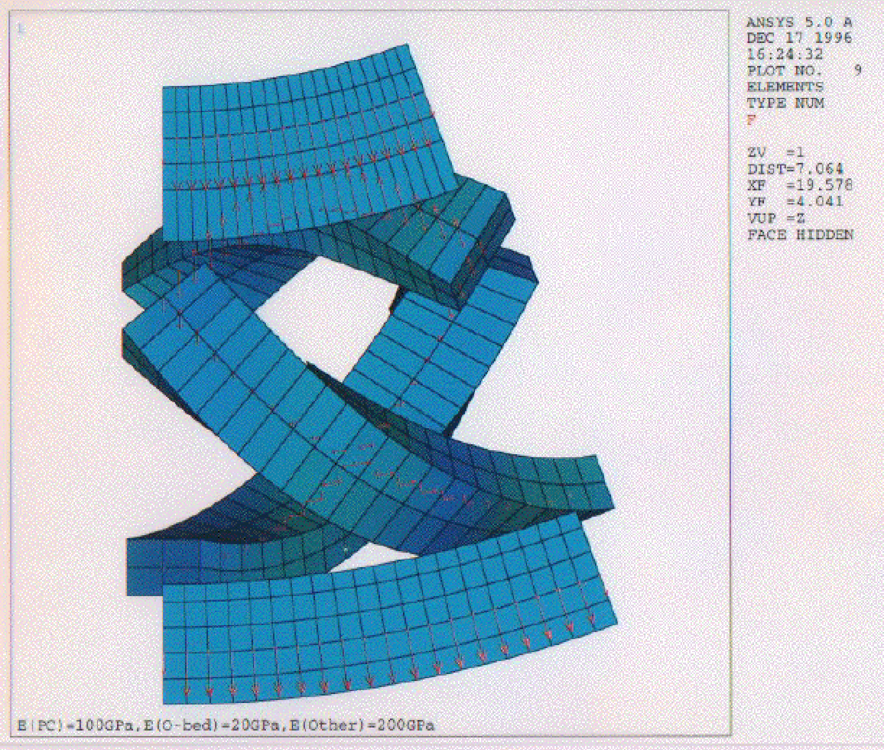


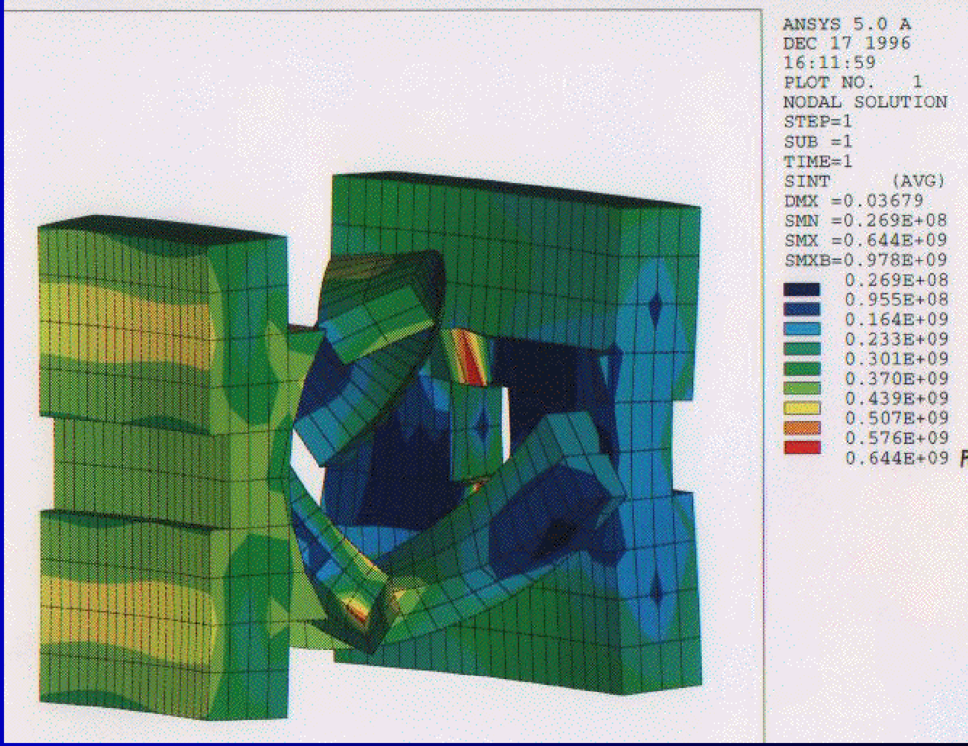
SPPS
UCSD-ENG-004



コイル支持構造の簡略化

上下シェルを解放した内外筒と自己支持構造で
SUSの許容応力レベル640MPaを実現





Present LHD experiments justify the future prospect of LHD-type reactors

Helical Reactor Designs

Name of Reactor Design	LHR/MHR-C	LHR/MHR-S	FFHR-1	FFHR-2	HSR	SPPS/MHHR
Major Radius R (m)	16.5	10.5	20	10	19.5	13.95
Average Plasm Radius a (m)	2.4	1.5	2	1.7	1.6	1.63
Toroidal Field B (T)	5	6.5	12	10	5	4.94
Maximum Coil Field Bmax (T)	14.9	14.7	16	13	10.7	14.5
Average Plasma Density <n> ($10^{20}/m^3$)	2	3.4	1	1.4	1.33	1.46
Average Plasma Temperature <T> (keV)	7.8	7.8	11	13.5	7.49	10
Volume Average Beta β (%)	5	5	0.7	0.59	4.57	5
Enhancement Factor Designed	2 (LHD)	2 (LHD)	1.5 (LHD)	2.5 (LHD)	1 (LG)	2.3 (LG)
Thermal Power P_{FT} (GW)	3.8	2.8	3	1	3	2.29
Effective Pheating Power (MW)	600	400	200	400	300	200
Energy Confinement Time τ (s)	2.67	1.5	3.7	1.8	1.2	1.75
τ _LHD (s)	1.24	0.79	2.46	0.76	0.71	0.76
τ _GRB (s)	1.30	0.69	2.42	0.75	0.64	0.74
τ _LG (s)	1.66	0.89	3.58	0.90	1.03	1.02
τ _ISS95 (s)	1.20	0.66	2.52	0.76	0.67	0.74
New Scaling (Phys. Rev. Lett.) τ (s)	2.98	1.27	4.99	1.72	1.03	1.36
New LHD #1 (heliotron-type) τ (s)	2.70	1.30	6.13	1.71	1.28	1.42
New LHD #2 (all helical) τ (s)	1.62	0.87	4.64	1.04	1.03	1.02

by K.Yamazaki et al., 18th IAEA, Sorrento (2000) FTP 2/12

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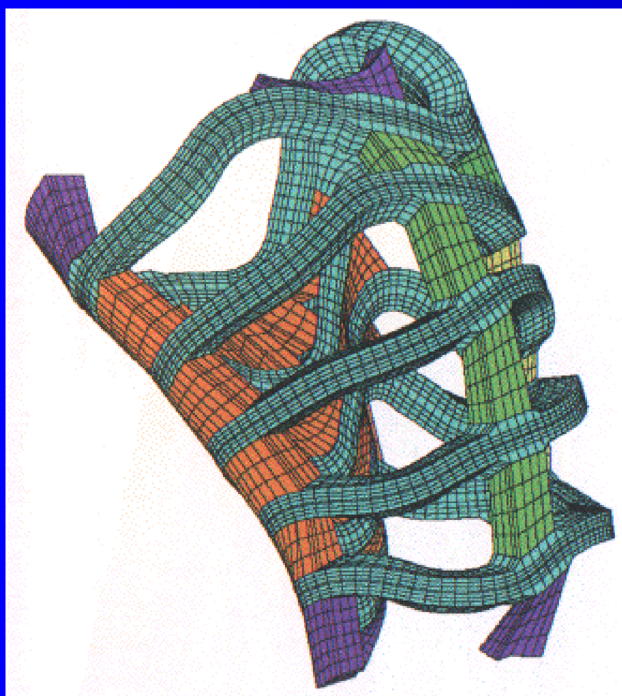
HSRのコイル支持構造の応力解析

許容応力レベル ー 650Mpa を実現している



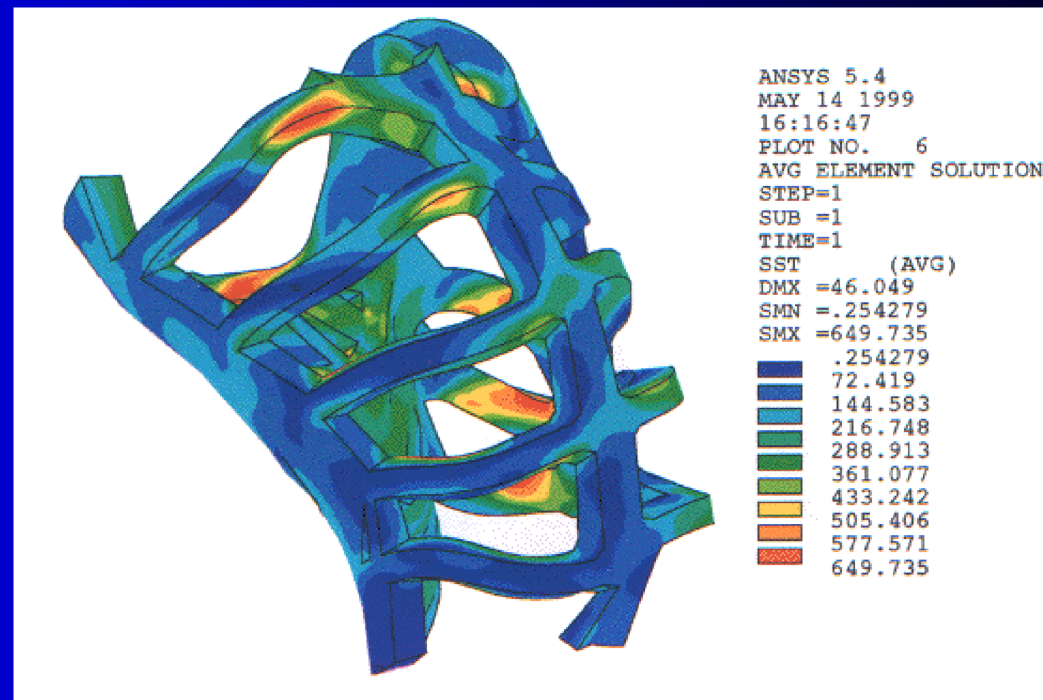
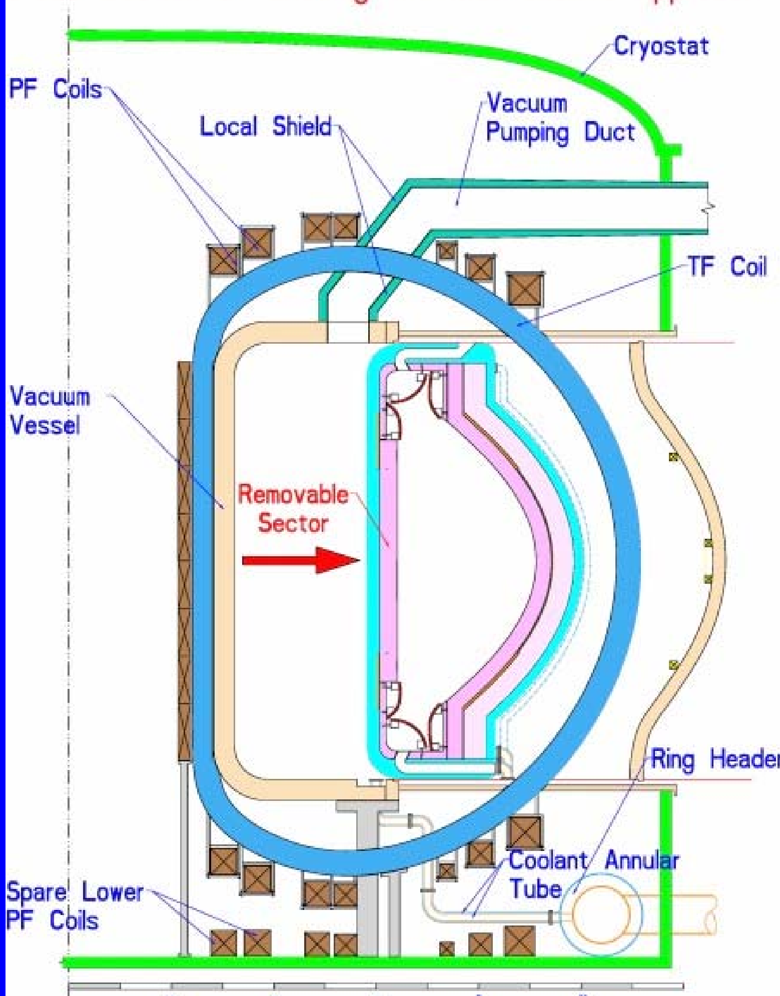


Fig. 18: Stresses in the coil support system and the coil casings. Maximum van Mises stress 650 Mpa.

最近のトカマク炉設計での保守交換

TFコイルを大きくし、縦長横ポートから一括引き抜き（A-SSTR2）

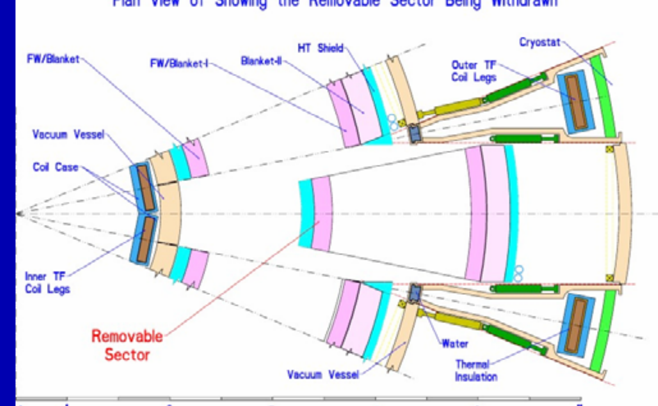
Cross Section of Showing the Maintenance Approach



ARIES-AT (based on 09/04/00 strawman)
Last Update: January 4, 2001

Major Radius	5.2 m
Minor Radius	1.3 m
Number of Sectors	16
Total useful thermal power (MW)	1968 MW
Net electrical output power (MWe)	1000 MW
Plasma current (MA)	12.825
Bootstrap-current fraction	0.915
Average neutron load (MW/m2)	3.28
Plant lifetime (full power years)	40
LSA=1 total COE (mill/kWeh)	52.5
In-vessel primary structure material	SiCfSiC
Breeder	Pb-17Li, 60% enriched 6Li
Divertor plasma-facing materials	highly pure tungsten

Plan View of Showing the Removable Sector Being Withdrawn



炉内機器の保守交換法の構築

横ポートからの大型
ユニットの組み込み

LHD での実績

Sagar