Present activities on LHD-type Reactor Designs FFHR

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Contributed with

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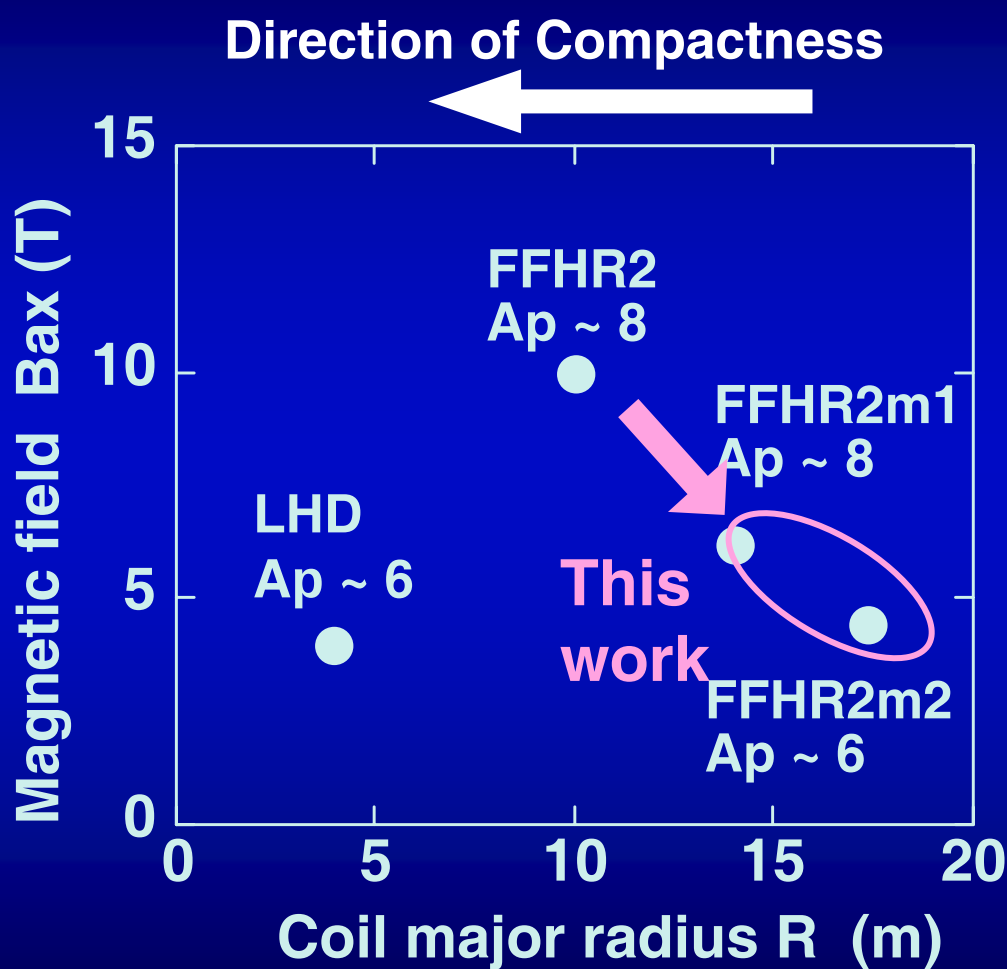
**Japan-US Workshop on
Fusion Power Plants and Related Advanced Technologies
with participation of EU
January 11-13, 2005 at Tokyo, JAPAN**

Presentation Outline

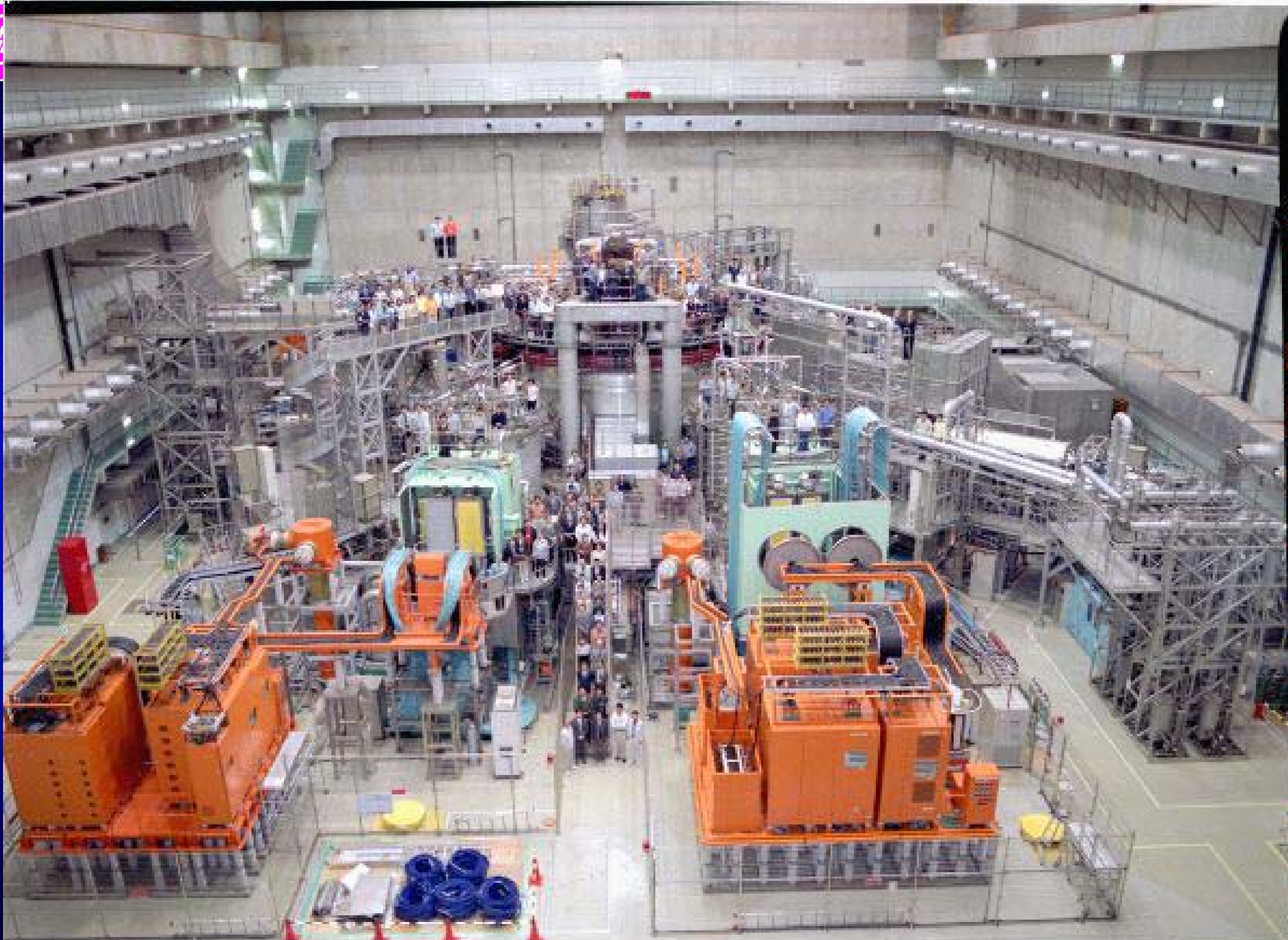


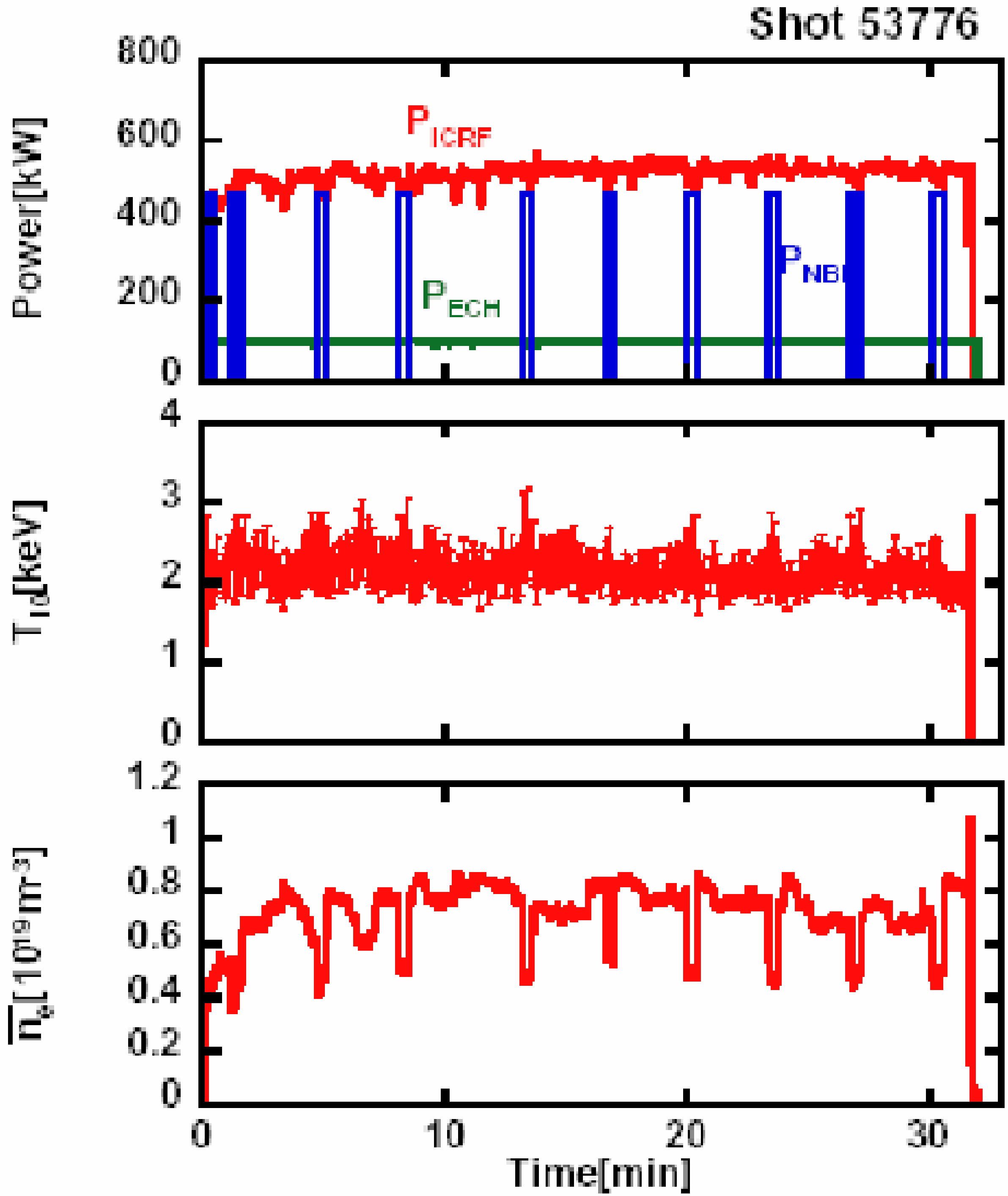
1. Introduction of FFHR

2. New design approach
   - The size is increased
   - Why ?

3. Conclusions

Presented in 20th IAEA Fusion Energy Conference
1 - 6 November 2004
Vilamoura, Portugal





Shot 53776
Power[kW]
800
600
400
200
0
P ICRF
P ECH
P NBI
T id[keV]
4
3
2
1
0
n e[10¹⁹m⁻³]
1.2
1
0.8
0.6
0.4
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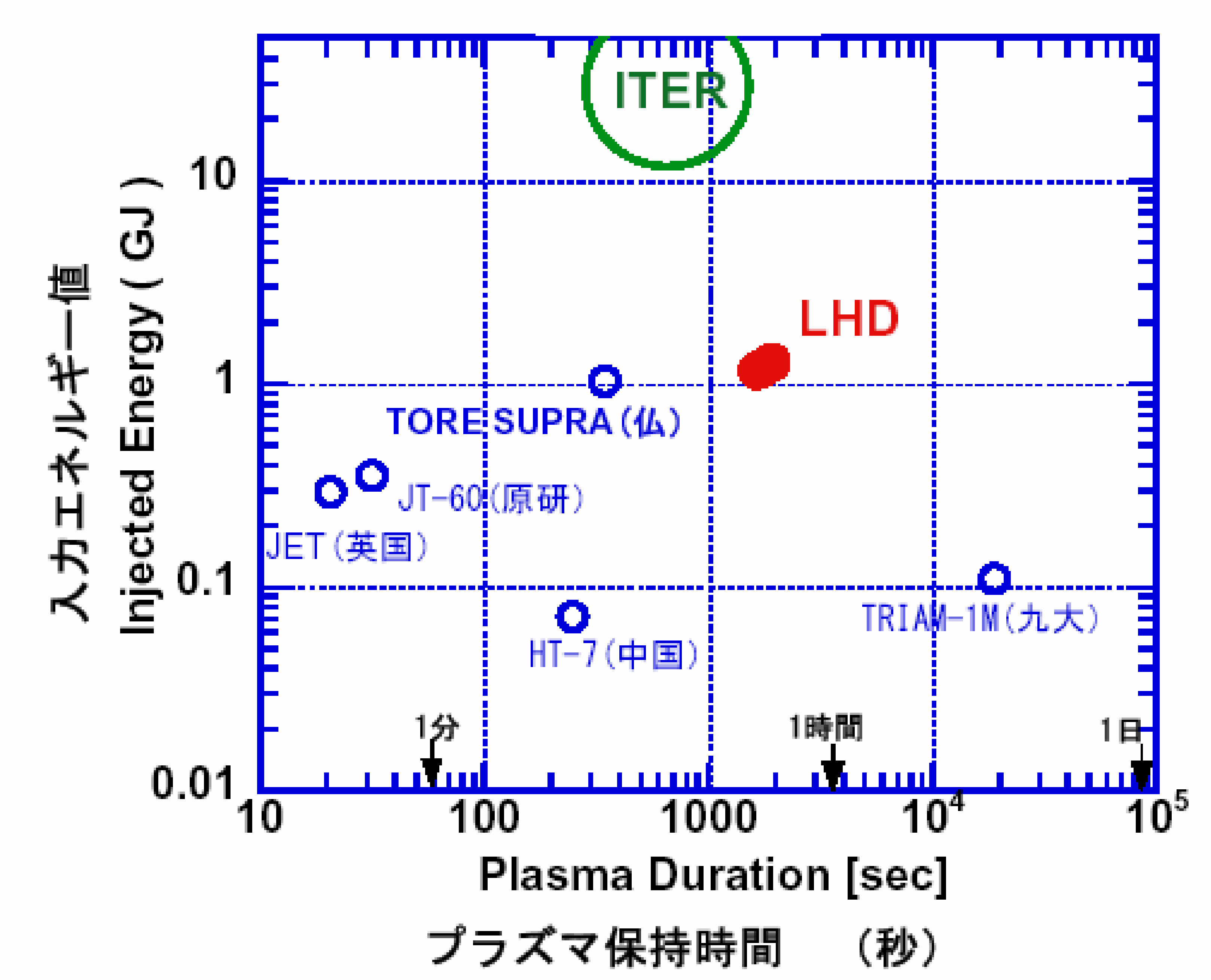
0
0
10
20
30
Time[min]


ITER
LHD
TORE SUPRA (仏)
JT-60 (原研)
JET (英国)
HT-7 (中国)
TRIAM-1M (九大)
1分
1時間
1日
10
1
0.1
0.01
10
100
1000
10^4
10^5
入力エネルギー値
Injected Energy (GJ)
Plasma Duration [sec]
プラズマ保持時間 (秒)

Reactor design collaborations in NIFS

1993〜

FFHR

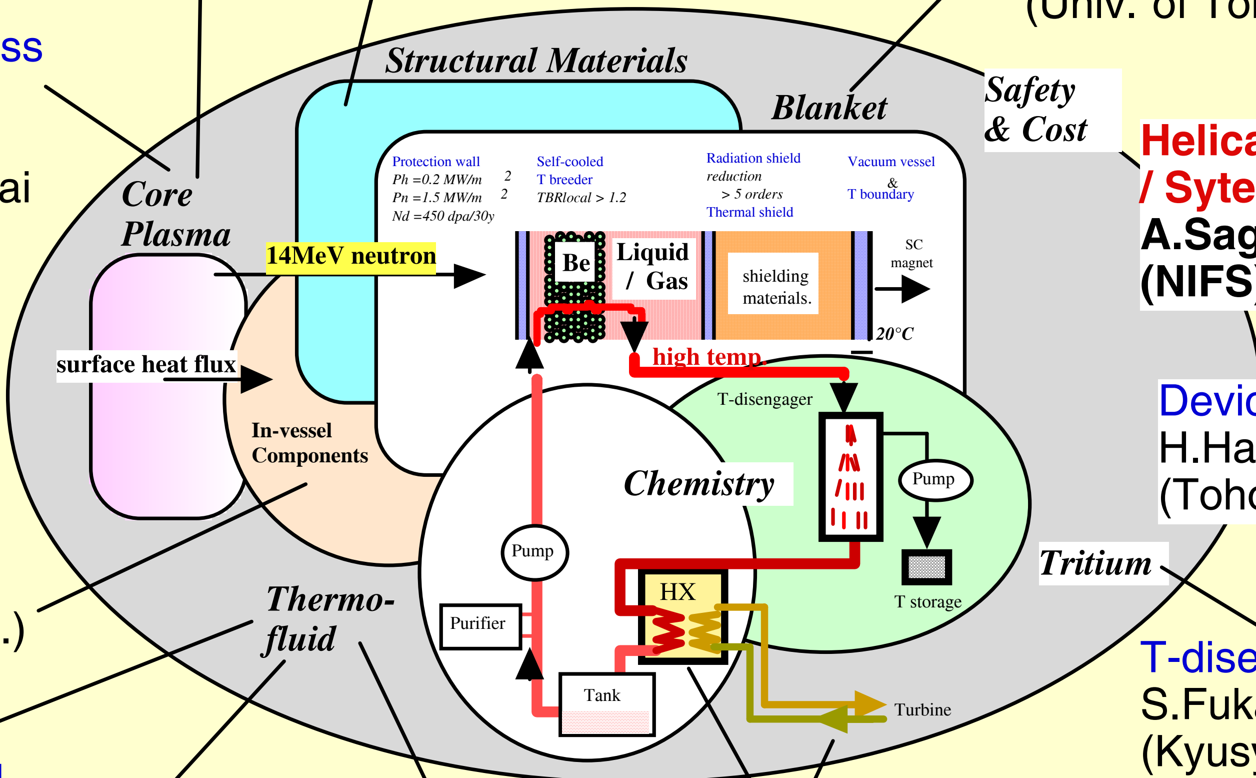
Helical core plasma
K.Yamazaki
(NIFS)

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

Blanket system
S,Tanaka
(Univ. of Tokyo)

Ignition access
& heat flux
O.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.)

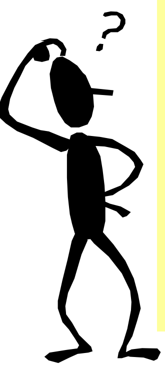
June 6. 2003, A.Sagara

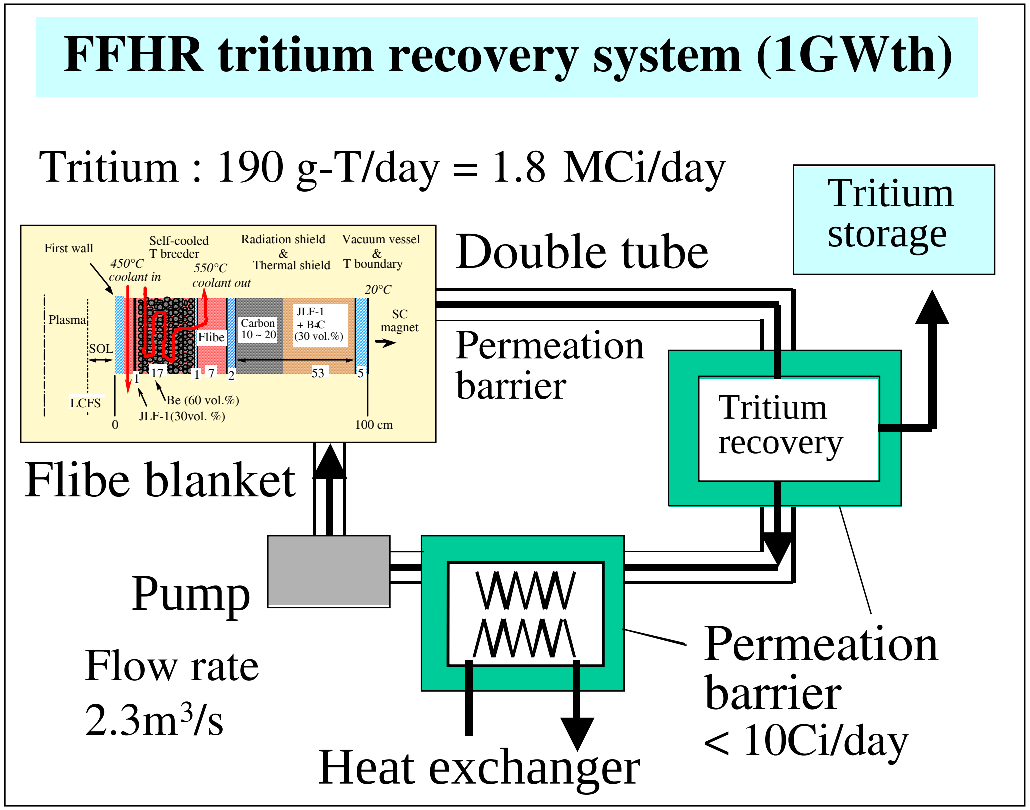
Tritium recovery systems

Kyushyu Univ.: S.Fukada

Permeation leak through the recovery system
is a crucial problem

- **Small amount of Flibe or He gas flow in the double tube are good as permeation barrier to reduce < 10Ci/day.**
- **The most serious problem is permeation leak of ~34 kCi/day through the heat exchanger to the He loop**





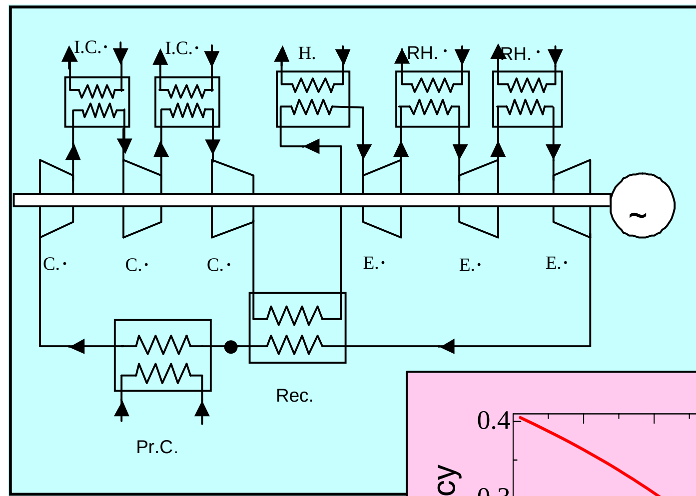
Energy conversion systems

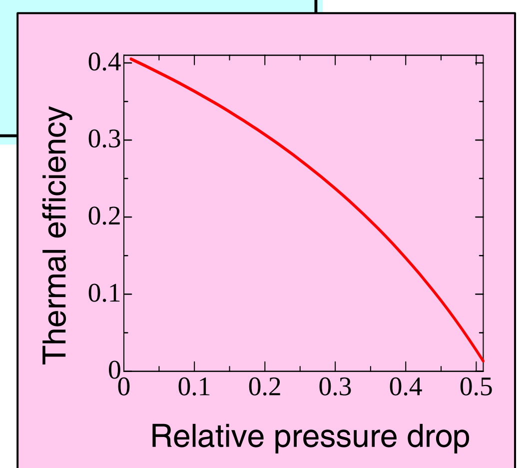
for Flibe in/out temperature of 450˚C and 550˚C

Kyushyu Univ.: A.Shimizu

Three-stage compression-expansion He-GT system was newly proposed

- **η_{max} ~ 37% for compression ratio of 1.5,**
- **However, η_{max} decreases rapidly with the increase of pressure drop.**
- **Therefore the layout of energy conversion system is a key design issue.**

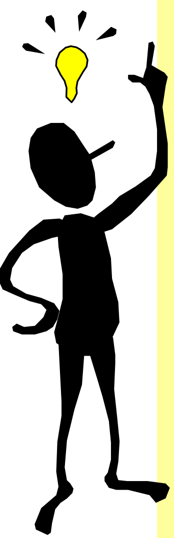




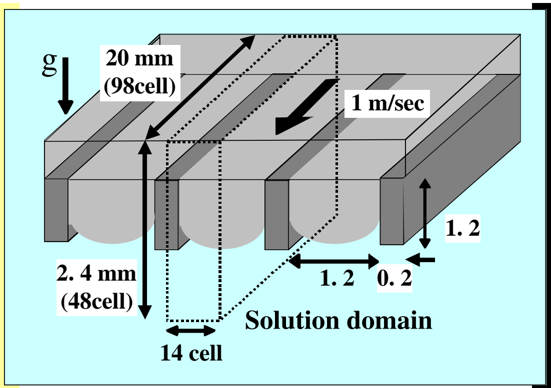
innovative free surface wall* design

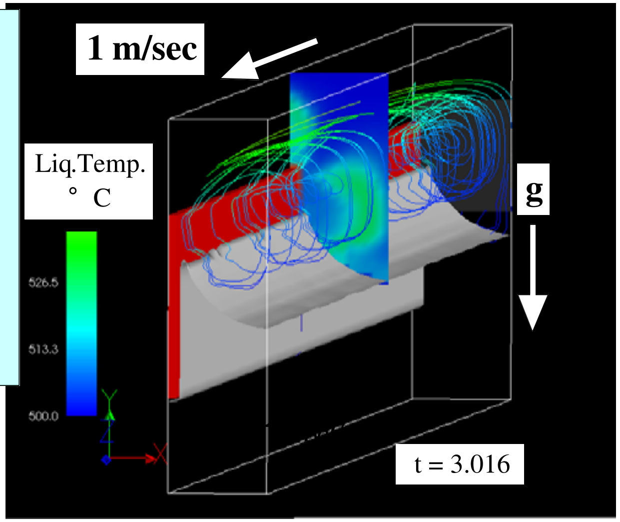
Kyoto Univ.: T.Kunugi

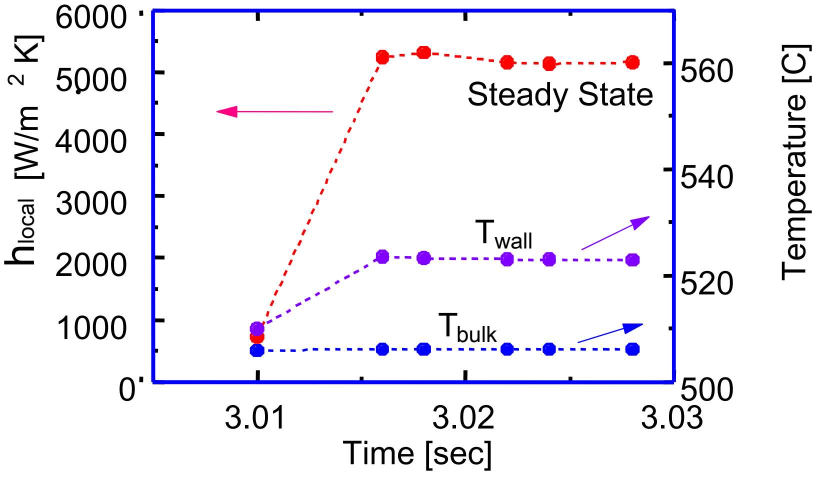
*KSF wall (Kunugi-Sagara type Free surface wall)



- Micro grooves are made on the first wall to use capillary force to withstand the gravity force
- Numerical simulation has explored the formation of a pair of symmetrical spiral flow,
- which enhances heat transfer efficiency about one order.







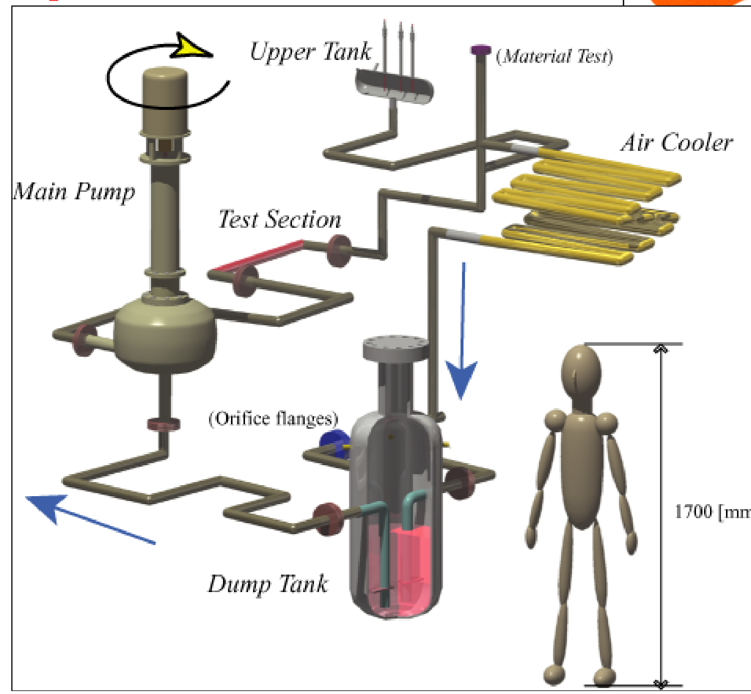
Thermofluid R&D activities

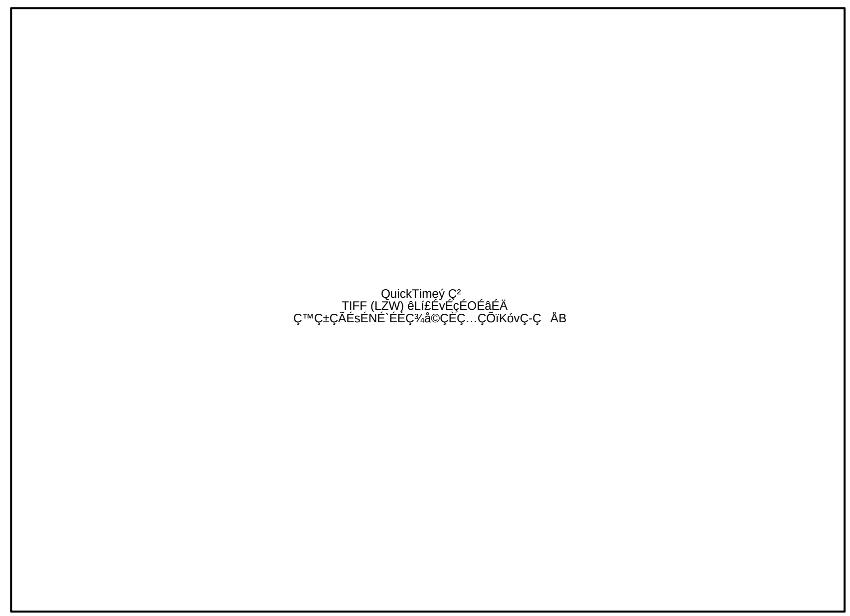
Tohoku Univ.: H.Hashizume

for enhancing heat-transfer in such high Prandtl-number fluid as Flibe

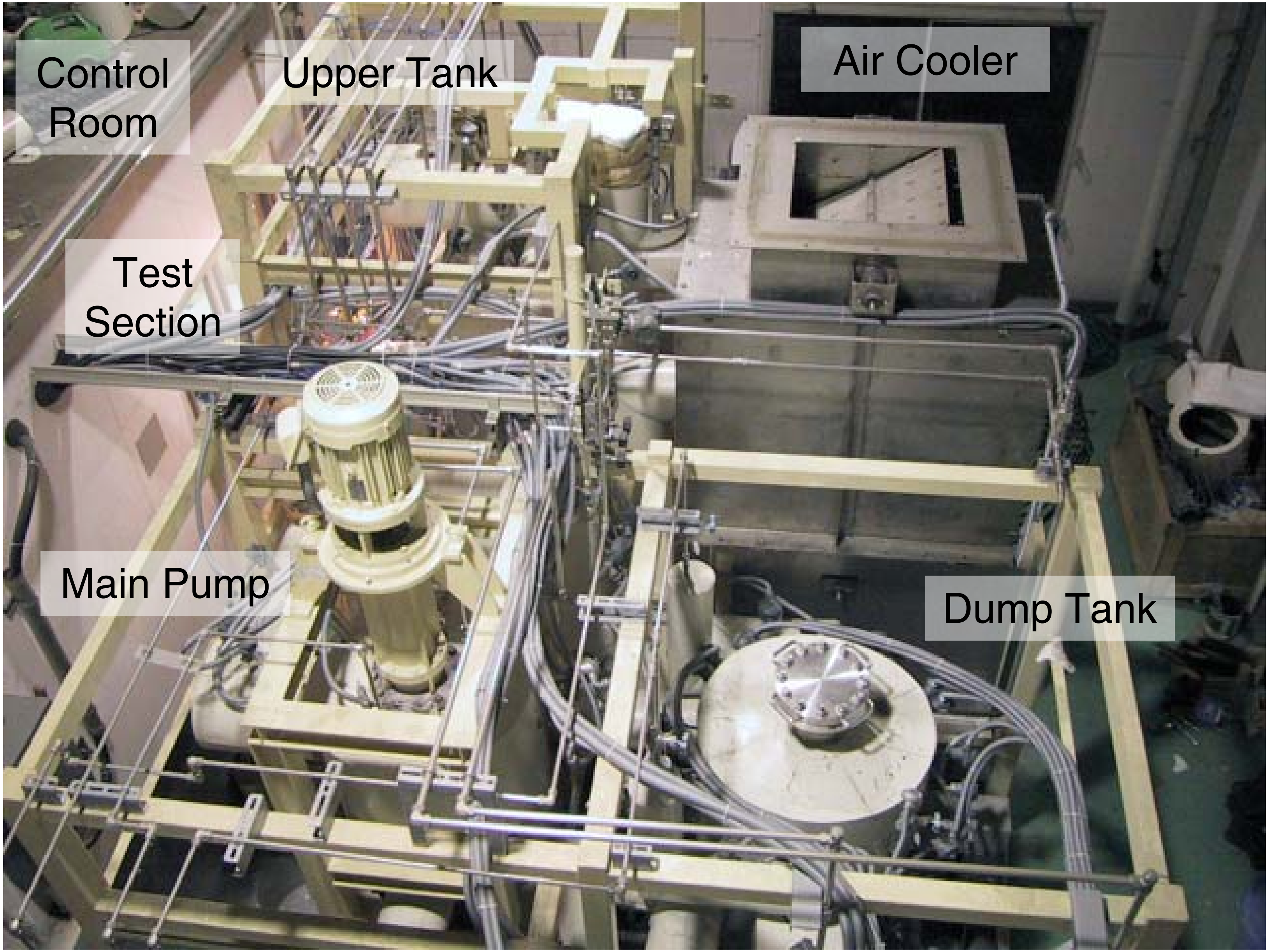
- "TNT loop" (Tohoku-NIFS Thermofluid loop) has been operated using HTS
  (Heat Transfer Salt, T_m= 142°C)
- Results are converted into Flibe case at the same Pr=28.5
  (Tin=200°C for HTS and 536°C for Flibe)
- Same performance as turbulent flow is obtained at one order lower flow rate.
- This is a big advantage for MHD effects and the pumping power.

TNT loop ~ $0.1m^3$, < 600°C





Bird's eye view of TNT loop

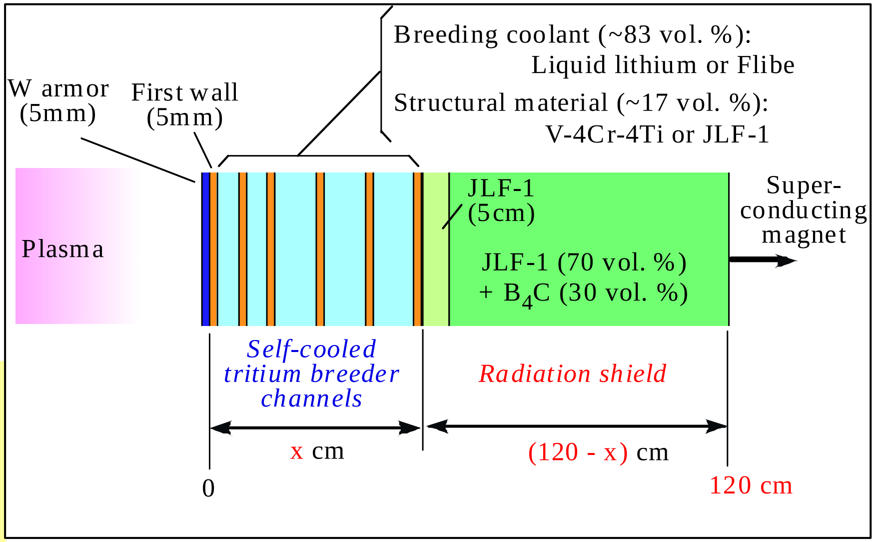


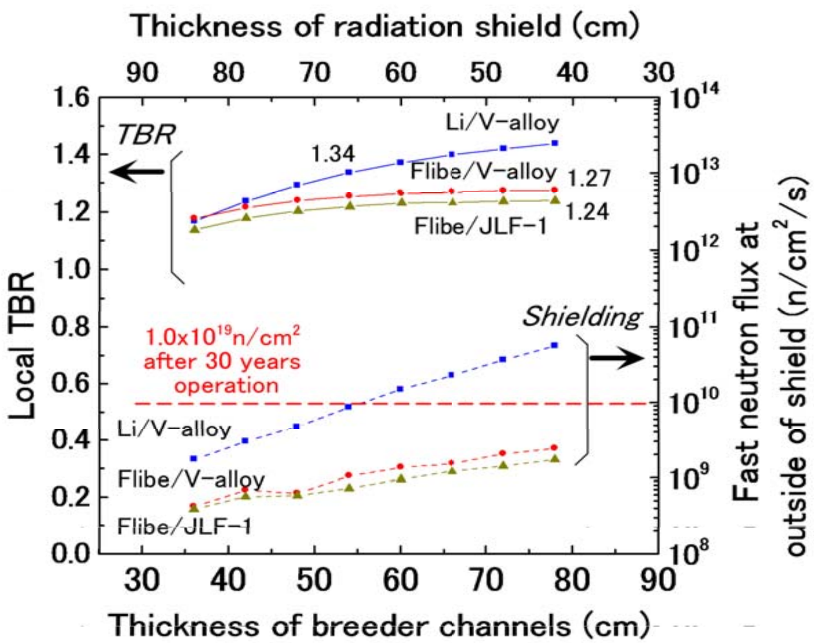
Self-cooled Be-free Li/V blanket

NIFS : T.Tanaka, T.Muroga

based on R&D progress on in-situ MHD coatings and high purity V fabrication

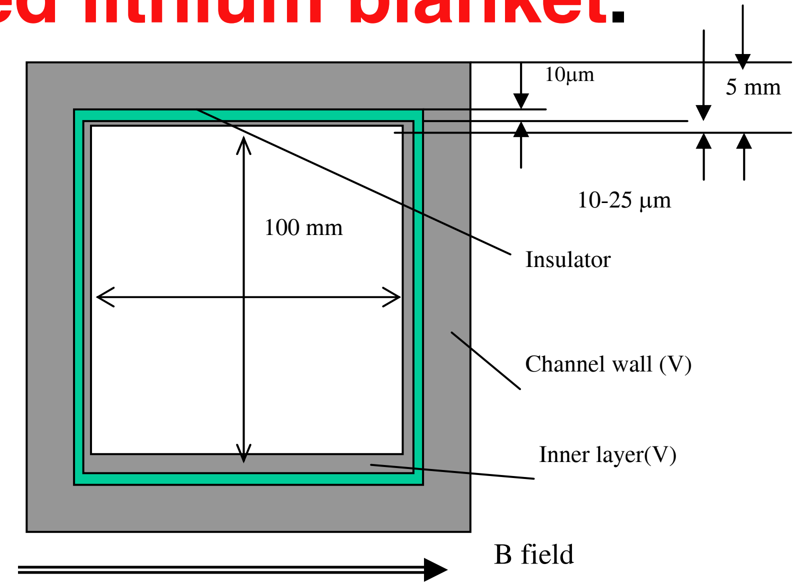
- Simple models are evaluated as alternatives for FFHR2 blanket..
- Balance of TBR and the shielding performance is examined, because shielding is poor w/o Be.
- TBR of Li/V is higher than 1.3 at about 50 cm with an acceptable shielding efficiency for super-conducting magnets.





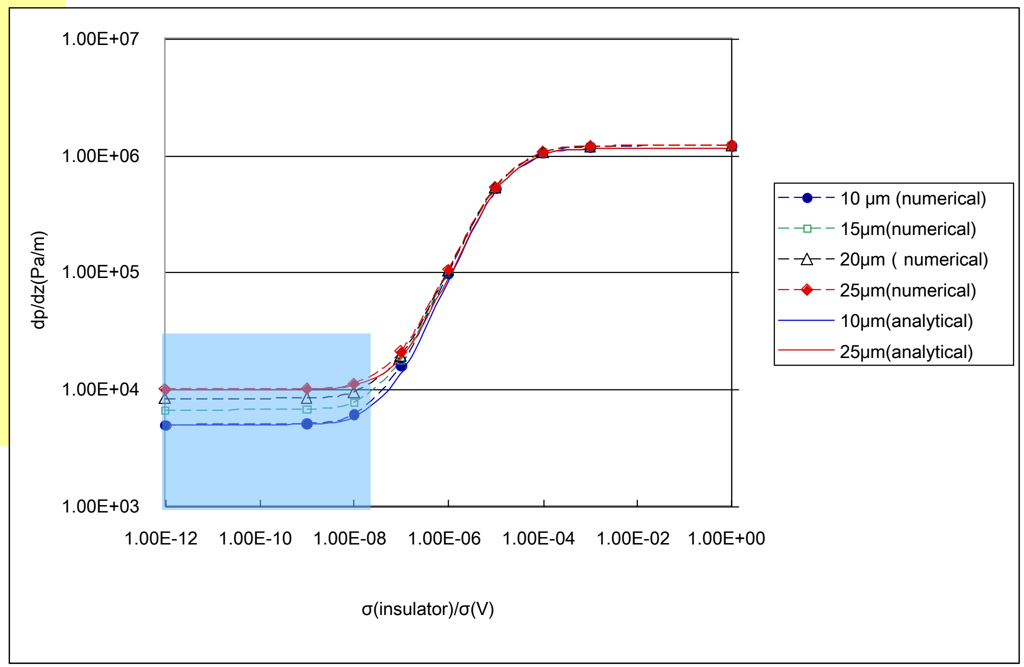
Modeling to Evaluate MHD pressure drop is established for self-cooled lithium blanket.

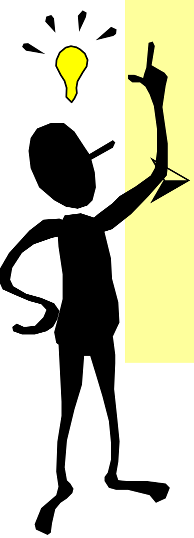
Tohoku Univ. : H.Hashizume



- Three-layered wall is proposed, where the inner thin metal layer protects permeation of lithium into the crack of coated layer
- Extremely good agreement between FEM and theory has been obtained
- The performance required to the insulator is evaluated to be

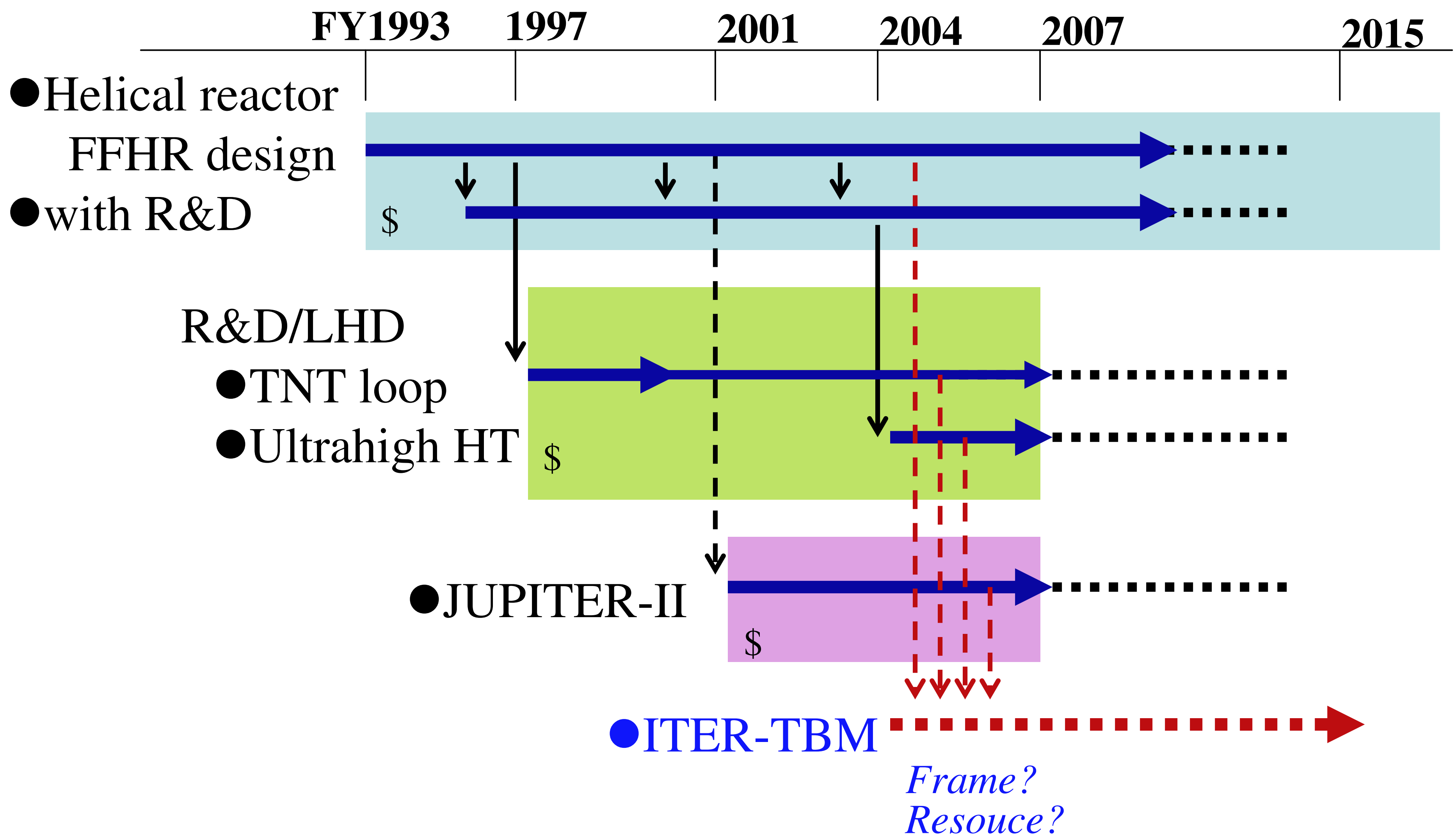
$$\frac{\sigma_{insulator}}{\sigma_V} \approx 10^{-8} - 10^{-9}$$





Present R & D activities on Flibe blanket in Japan

Presented by A.Sagara(NIFS) , Feb.'04



Flibe/RAF blanket, Li/V blanket, SB-He/SiC Blanket

R&D in Japan-US joint project JUPITER-II(FY'01~'06)

INEEL *1-1-A: FLiBe Handling/Tritium.Chemistry*

1-1-B: FLiBe Thermofluid Flow Simulation
2-2 : SiC System Thermomechanics

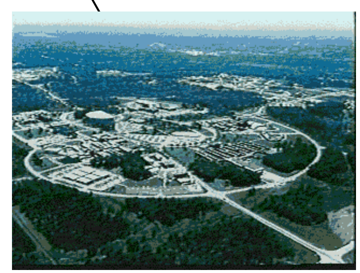
Japan

UCLA

3-1: Design-based Integration Modeling
3-2: Materials Systems Modeling

ORNL

1-2-B: V Alloy Capsule Irradiation
2-1 : SiC Fundamental Issues, Fabrication, and Materials Supply
2-3 : SiC Capsule Irradiation



ANL
(2001)

1-2-A: Coatings for MHD Reduction

http://jupiter2.iae.kyoto-u.ac.jp/index-j.html

LHD-type D-T Reactor FFHR

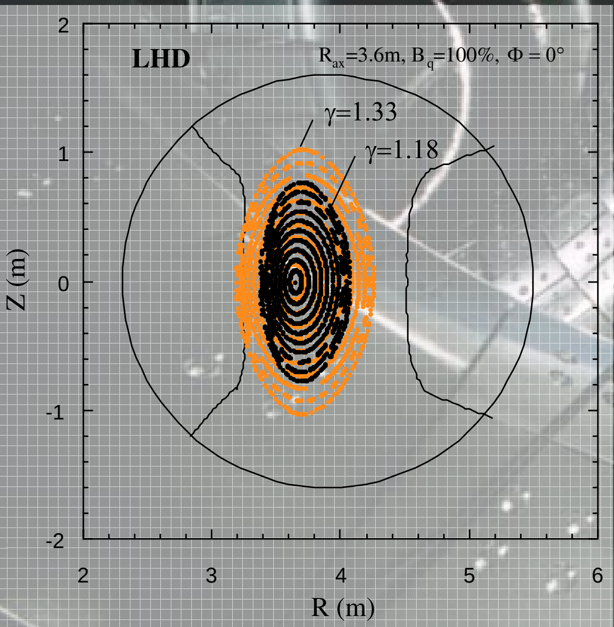
Many advantages :
- current-less
- Steady state
- no current drive power
- Intrinsic divertor

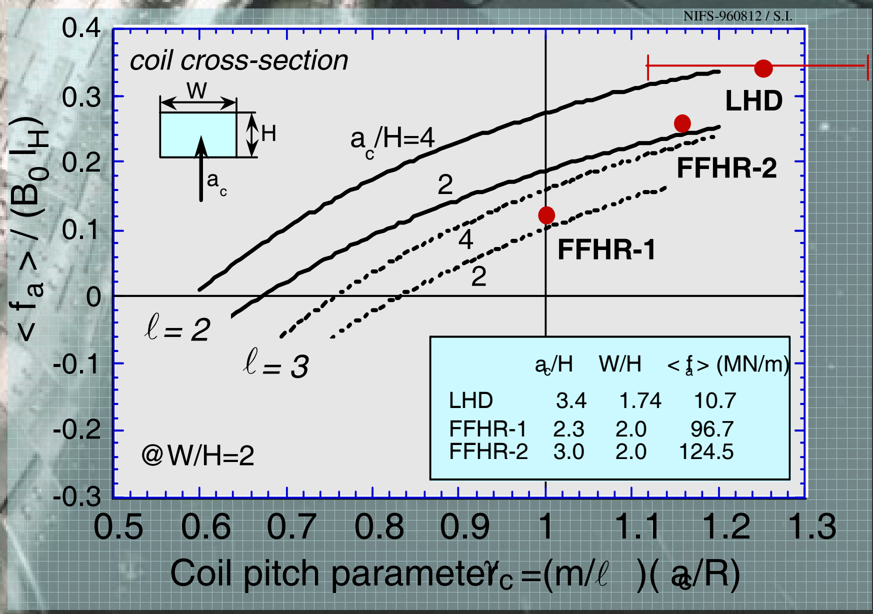
LHD operation: 1998 ~

Selection of lower $\gamma = \left(\frac{m}{l}\frac{a_c}{R}\right)$
- To reduce mag. foop force
- To expand blanket space

1993 FFHR-1 (*l=3, m=18*)
R=20, Bt=12T, β=0.7%

1995 FFHR-2 (*l=2, m=10*)
R=10, Bt=10T, β=1.8%





	a_c/H	W/H	< f_a > (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

However, direction of compact design has engineering issues

- Insufficient tritium breeding ratio (TBR)
- Insufficient nuclear shielding for superconducting (SC) magnets,

Blanket space limitation

- Replacement of blanket due to high neutron wall loading
- Narrowed maintenance ports due to the support structure for high magnetic field

Replacement difficulty

New design approach is proposed to overcome all these issues

- Introducing a long–life & thicker breeder blanket
- Increasing the reactor size with decreasing the magnetic field
- Improving the coils-support structure

Blanket space

Replacement

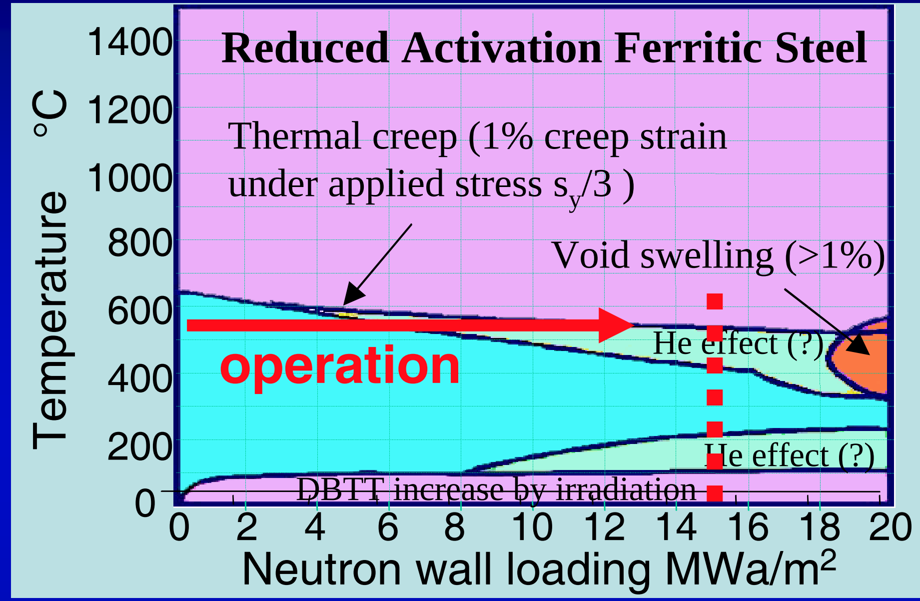
Proposal and Optimization of STB (Spectral-shifter and Tritium breeder Blanket)

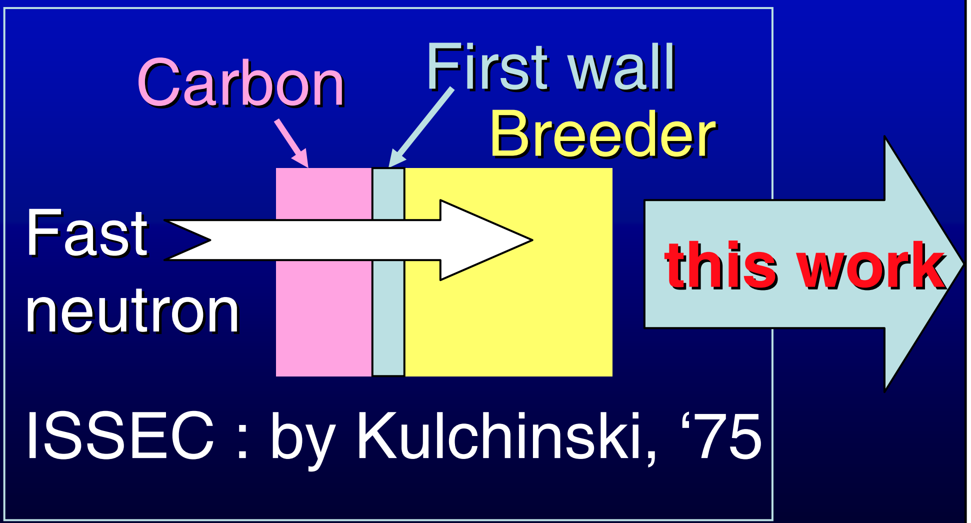
- Lifetime of Flibe/RAFS liquid blanket in FFHR ~ 15MWa/m^2

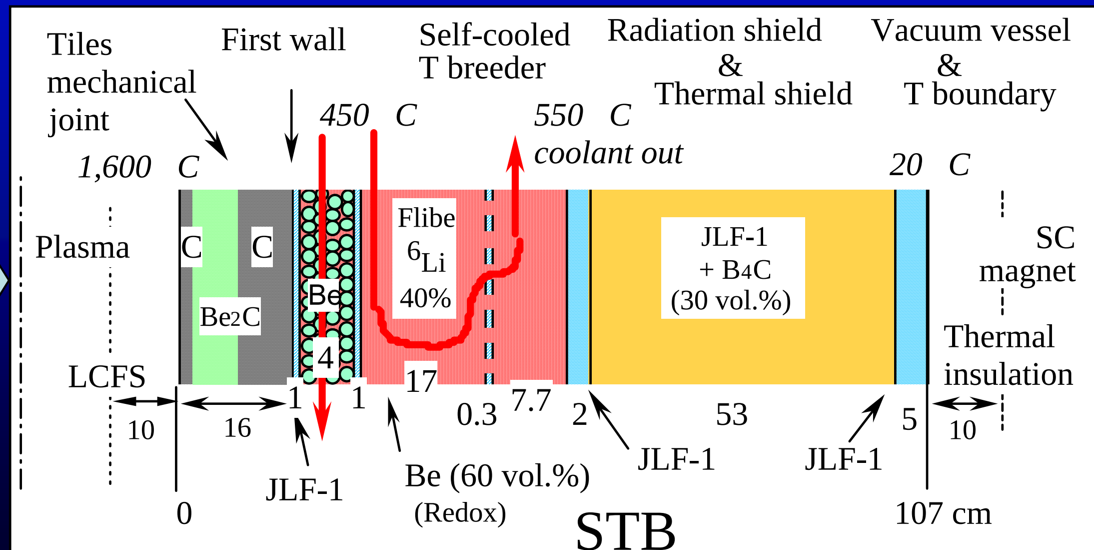
factor 3

- Neutron wall loading in FFHR2 in 30 years

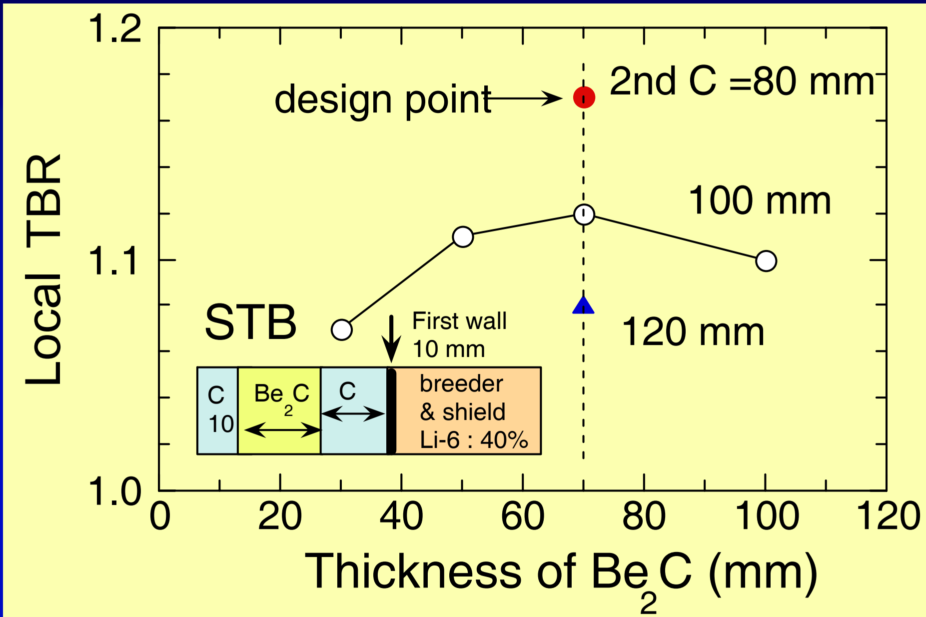
1.5MW/m^2 x 30y = 45MWa/m^2

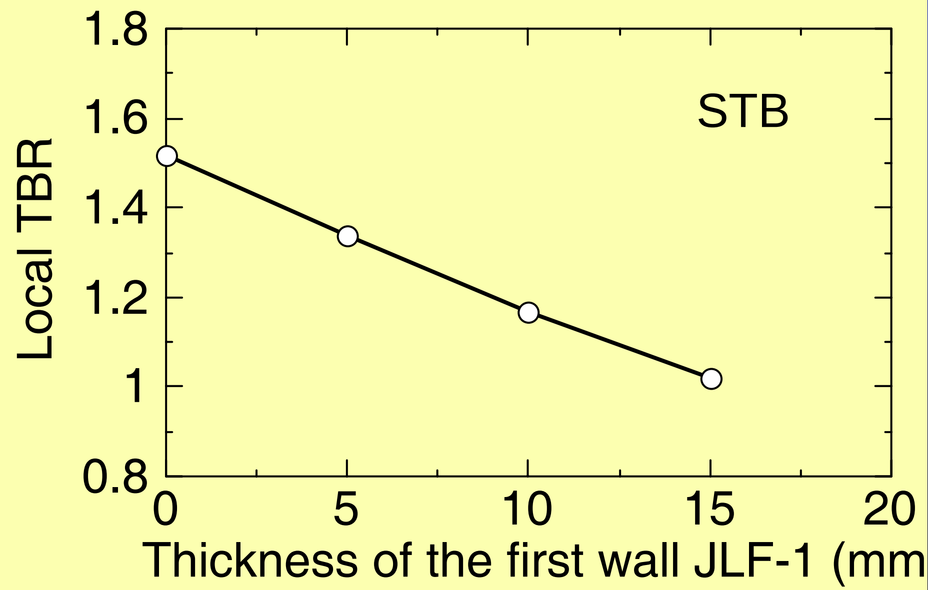




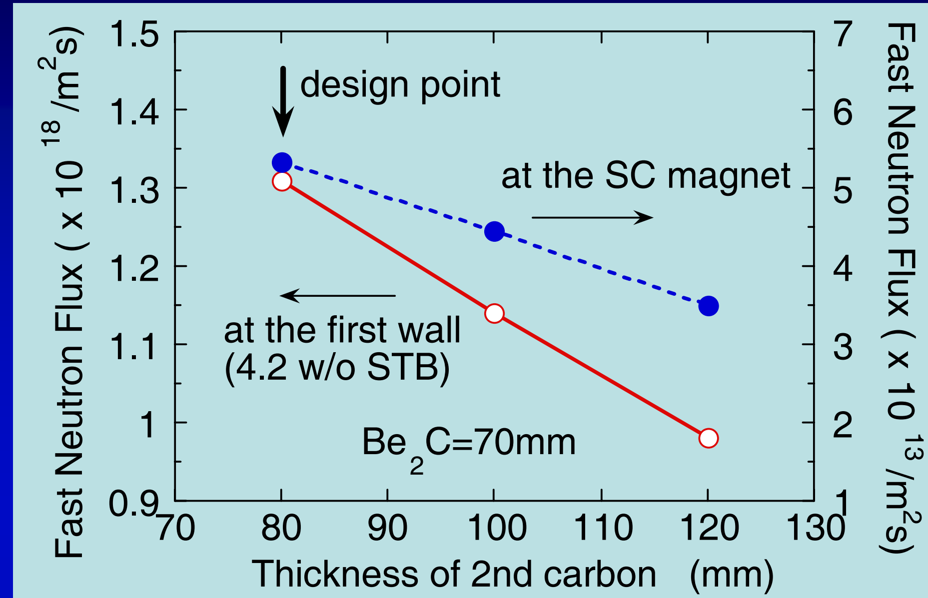


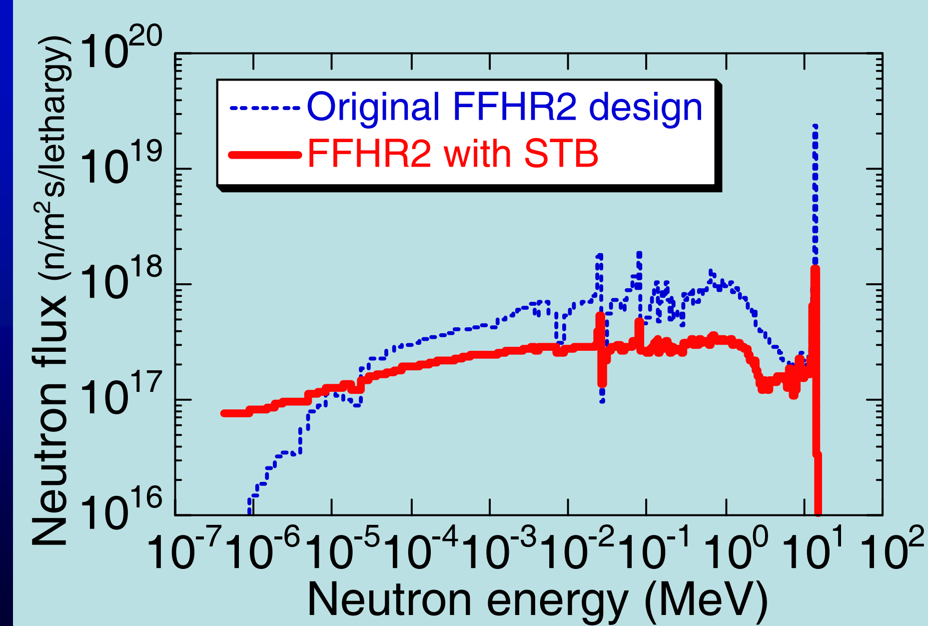
Tritium breeding





Shielding efficiency

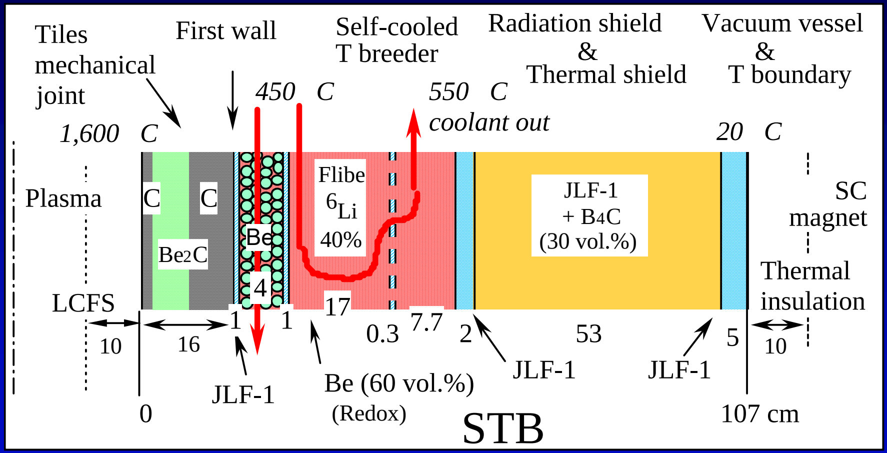




Results and Key R&D Issues

Results :

- ✓ Fast neutron flux at the first wall is factor 3 reduced.
- ✓ Local TBR > 1.2 is possible.
- ✓ Fast neutron fluence to SC is reduced to $5 \times 10^{22} n/m^2$ (Tc/Tco>90% in 30y)
- ✓ Surface temperature < 2000°C(~mPa of C) is possible



Required conditions :

- ➢ Neutron wall loading < $1.5 MW/m^2$
- ➢ Blanket thickness > 1100 mm

Design parameter

- ➢ λ for C-Be-C tiles > 100W/mK
- ➢ Super-G sheet for tiles joint > $6 kW/m^2K$
- ➢ Heat removal by Flibe > $1 MW/m^2$

Key R&D issues :

(1) Impurity shielding in edge plasma

(2) Neutron irradiation effects on tiles at high temperature

(3) Heat transfer enhancement in Flibe flow

Improved design parameters

Improved input

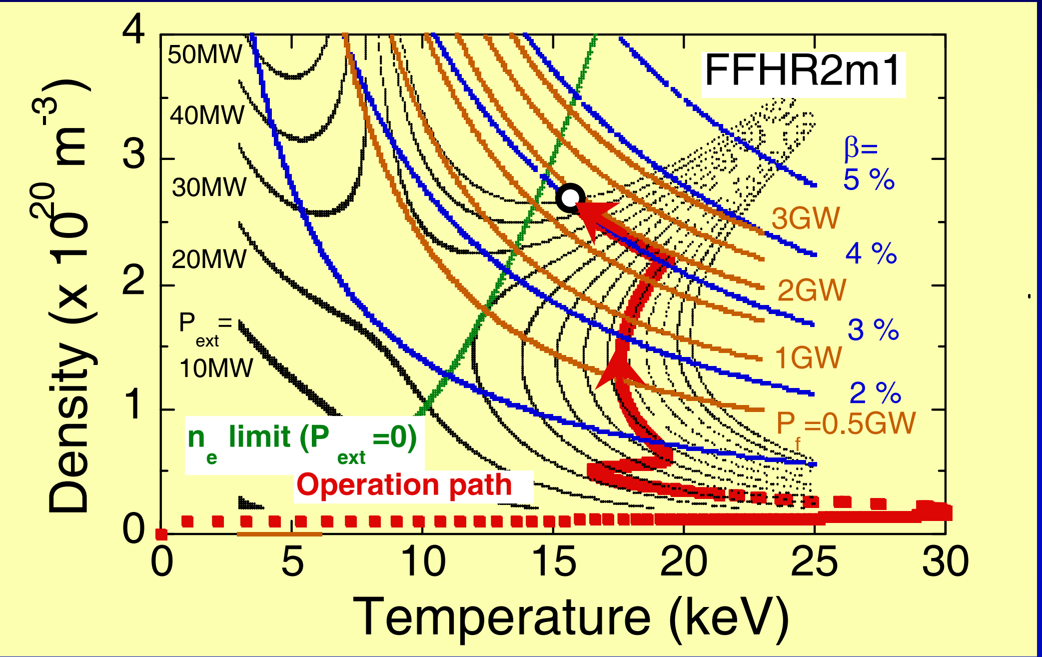
Required Blanket space

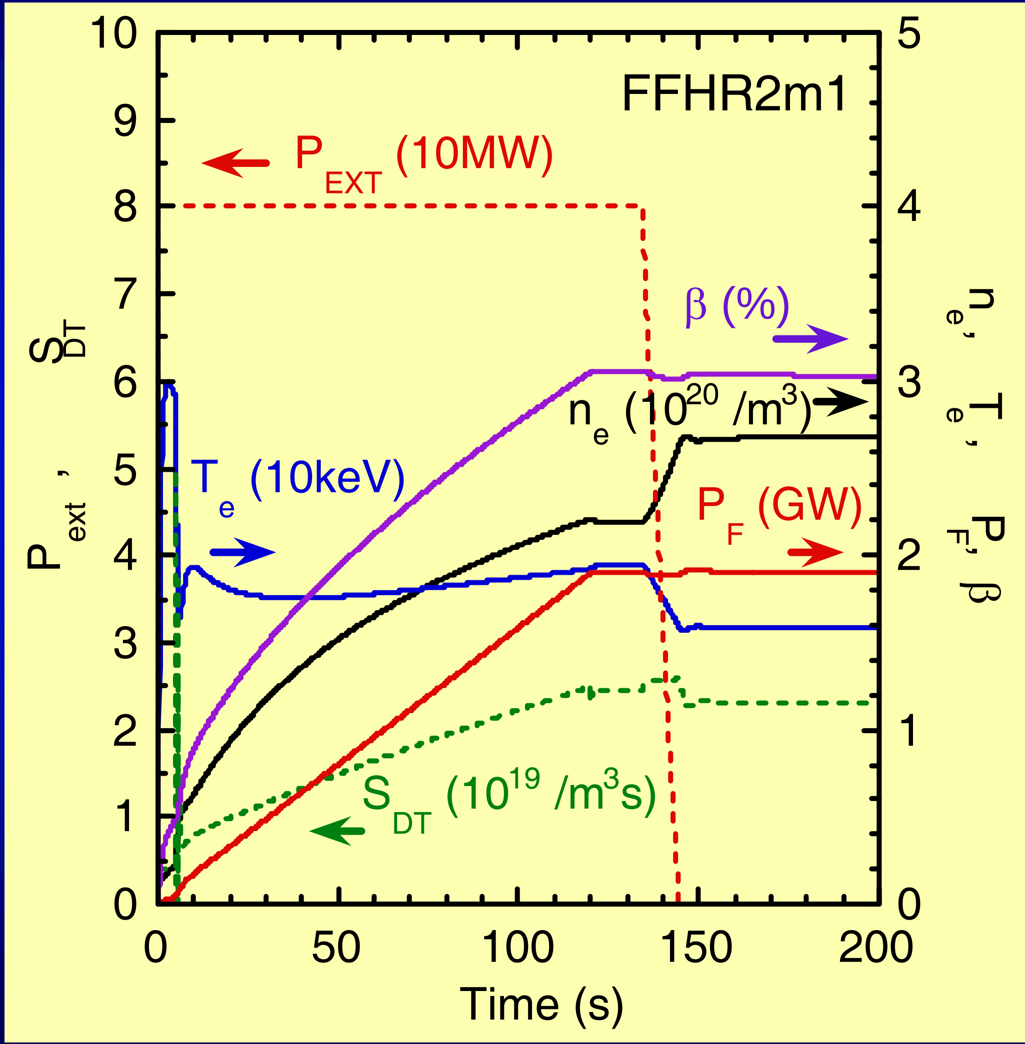
Required Wall loading

$$\tau_{ISS95} = 0.26 P^{-0.59} \bar{n}_e^{\,0.51} B^{0.83} R^{0.65} a^{2.21} \iota_{2/3}^{\,0.4}$$

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	Rc	m	3.9	10	14.0	17.3
Coil minor radius	ac	m	0.98	2.3	3.22	4.33
Plasma major radius	Rp	m	**3.75**	**10**	**14.0**	**16.0**
Plasma radius	ap	m	0.61	1.2	1.73	2.80
Blanket space	Δ	m	0.12	**0.7**	**1.2**	**1.1**
Magnetic field	B0	T	**4**	**10**	**6.18**	**4.43**
Max. field on coils	Bmax	T	9.2	13	13.3	13
Coil current density	j	MA/m2	53	25	26.6	32.8
Weight of support		ton	400	**2880**	**3020**	**3210**
Magnetic energy		GJ	1.64	147	120	142
Fusion power	P_F	GW		**1.77**	**1.9**	**3.0**
Neutron wall load		MW/m2		**2.8**	**1.5**	**1.3**
External heating power	Pext	MW		100	80	100
α heating efficiency	ηα			0.7	0.9	0.9
Density lim.improvement				1	1.5	1.5
H factor of ISS95				**2.53**	**1.92**	**1.68**
Effective ion charge	Zeff			1.32	1.34	1.35
Electron density	ne(0)	10^19 m-3		28.0	26.7	19.0
Temperature	Ti(0)	keV		27	15.8	16.1
<β>=2*n*T/(B^2/ μ) (parabolic distribution)				**1.8**	**3.0**	**4.1**
COE		Yen/kWh		**21.00**	**14.00**	**9.00**

Self-ignition access in FFHR2m1





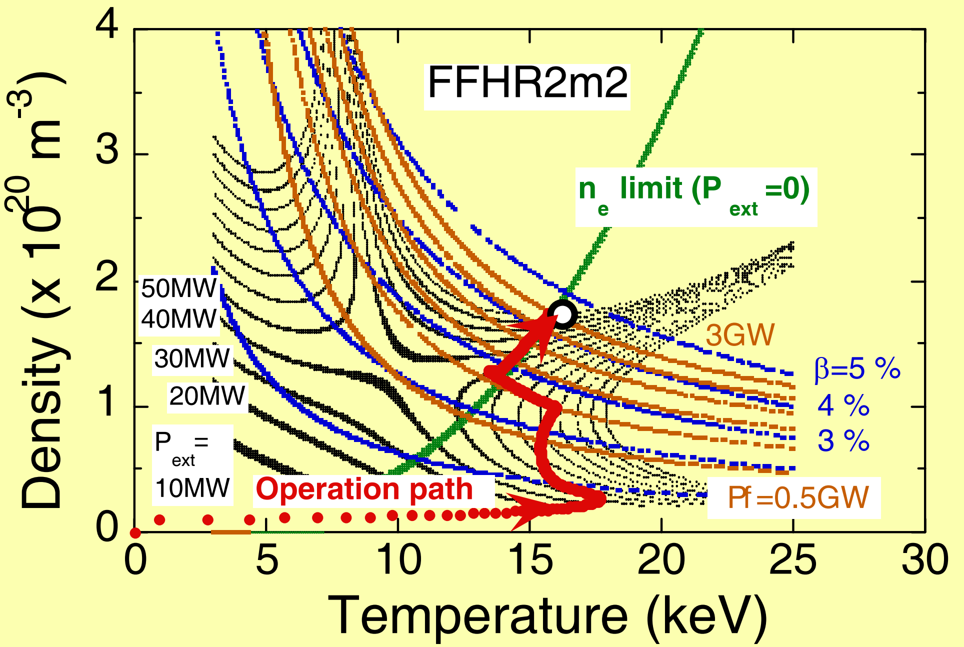
Zero-dimensional analysis

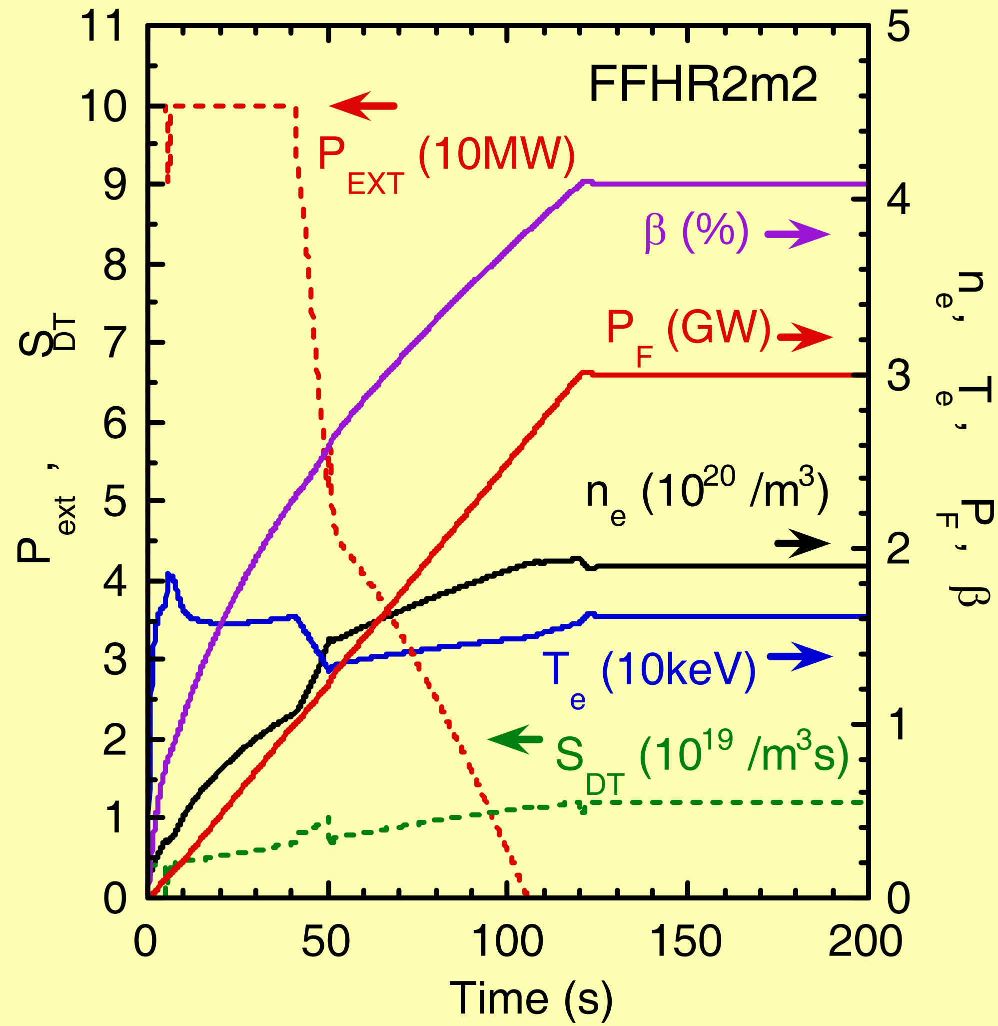
- H_{ISS95} = 1.2 x 1.6
- $\tau^*_\alpha / \tau_E = 3\ (< 7)$
- parabolic profiles
- n_e < 1.5 x Sudo limit
- α heating efficiency = 0.9

Already achieved in LHD

O.Mitarai et al., Fusion Eng. Design 70 ('04) 247.

Self-ignition access in FFHR2m2





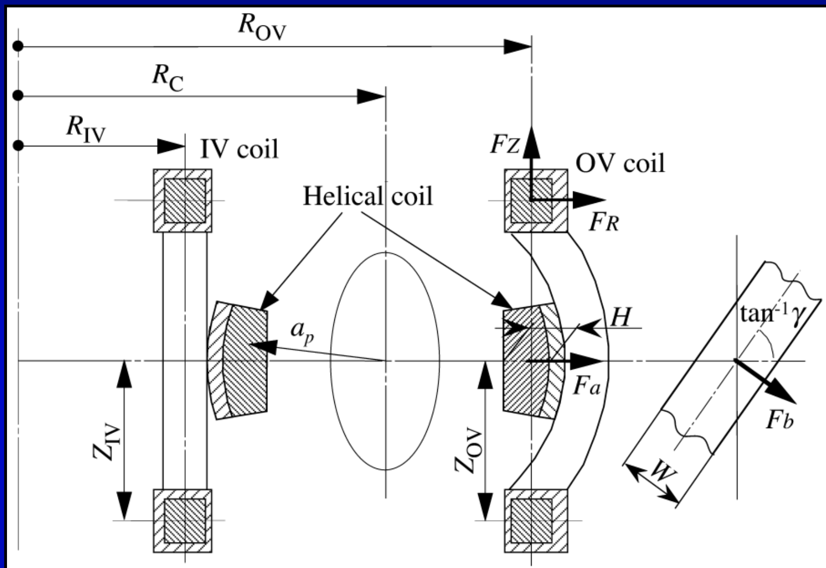
Zero-dimensional analysis

- H_{ISS95} = 1.1 x 1.6
- τ^*_α / τ_E = 3 (< 7)
- parabolic profiles
- n_e < 1.5 x Sudo limit
- α heating efficiency = 0.9

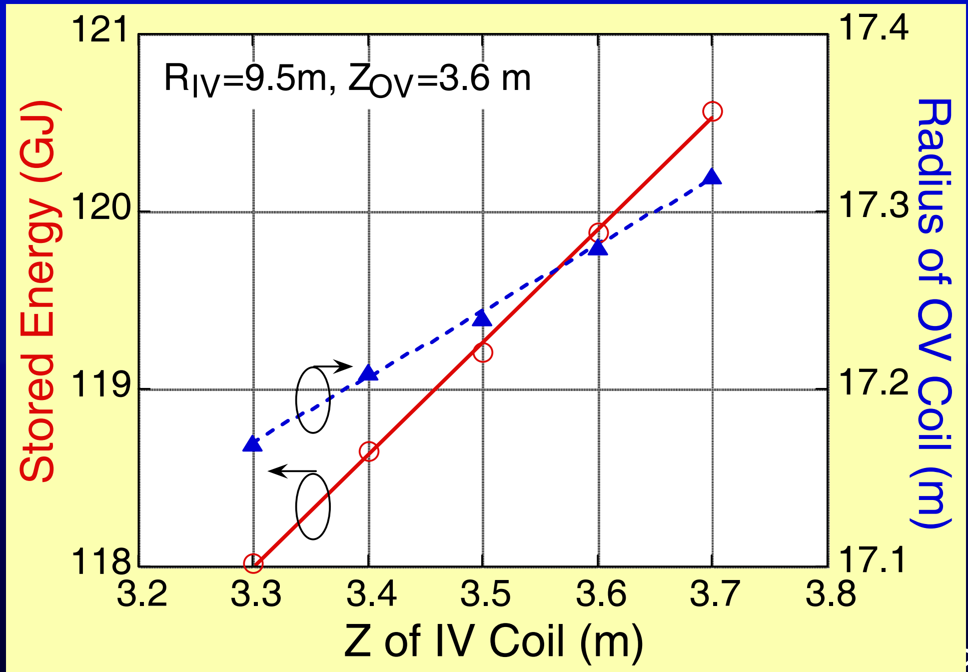
Already achieved in LHD

O.Mitarai et al., Fusion Eng. Design 70 ('04) 247.

Improved Design of Coil-supporting Structure (1/2)



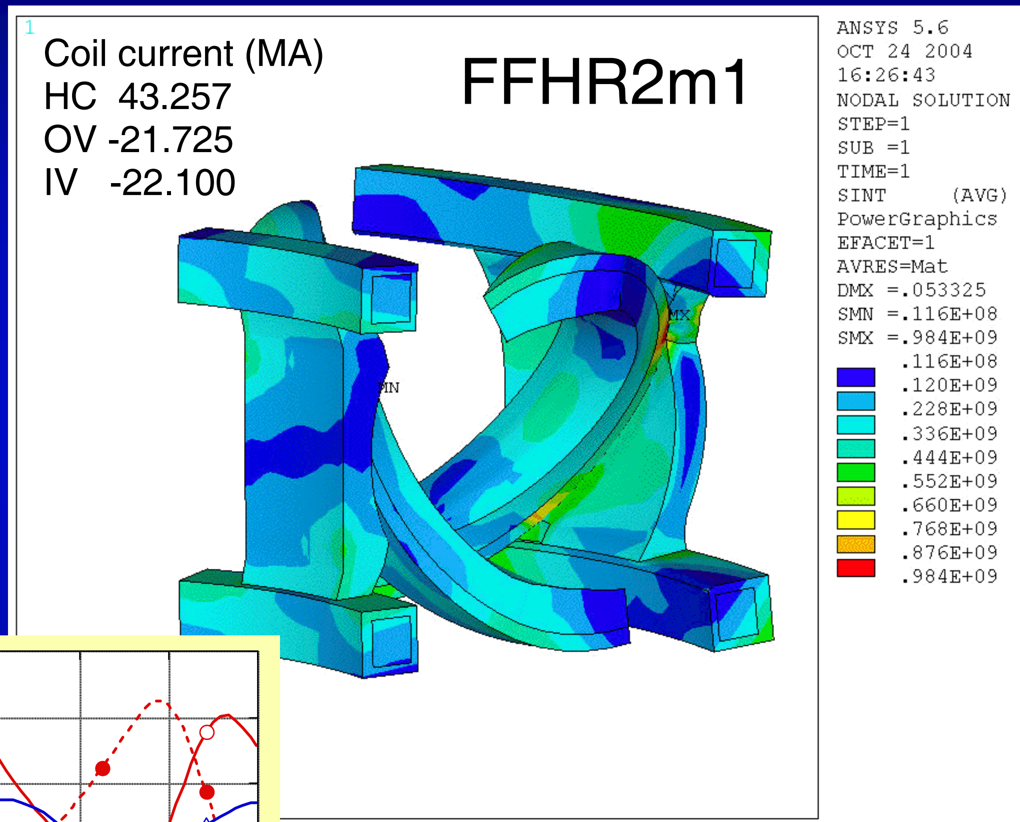
- **Cylindrical supporting structure under reduced magnetic force,** which facilitates expansion of the maintenance ports.
- **Helical coils supported at inner, outer and bottom only.**



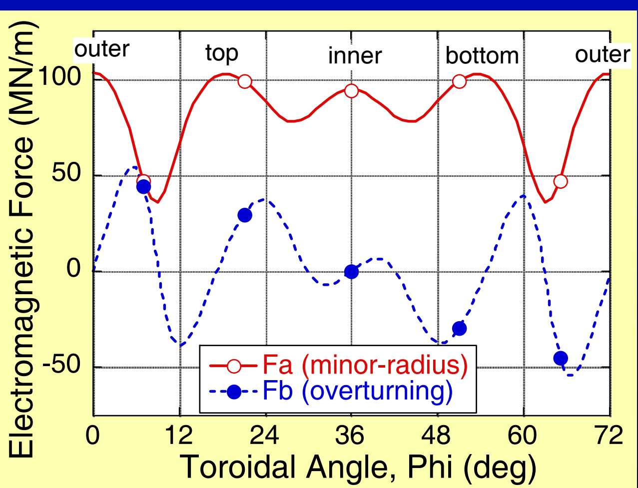
- **W/H = 2 & H/a_c determined by**
- **Bmax ~ 13 T for such as Nb_3Sn or Nb_3Al.**
- **Then J=25~35A/mm^2**
- **Poloidal coils layout as for stored magnetic energy and stray field**

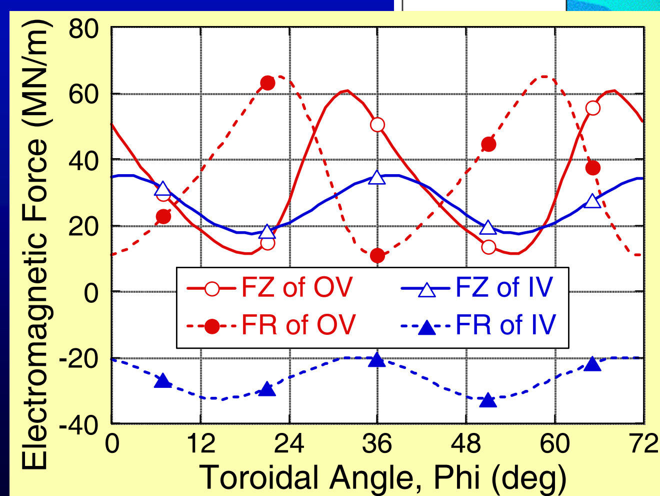
Improved Design of Coil-supporting Structure (2/2)

- The maximum stress can be reduced less than 1000 MPa (<1.5Sm)
- This value is allowable for strengthened stainless steel.



Electromagnetic forces on a helical coil of FFHR2m1.

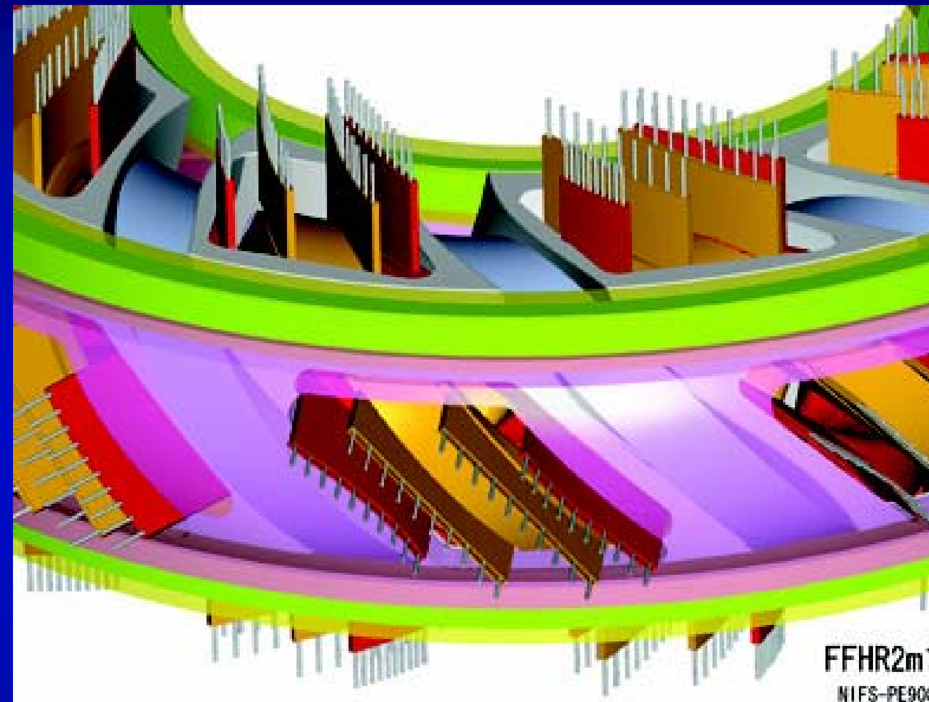


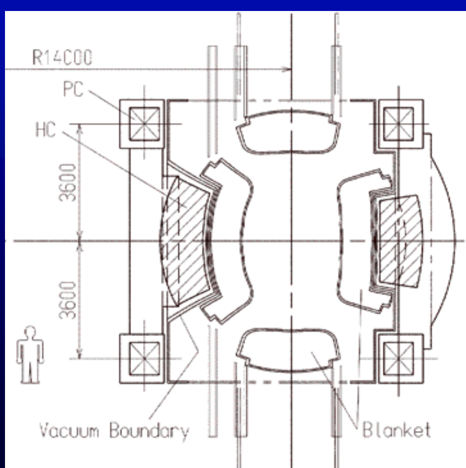


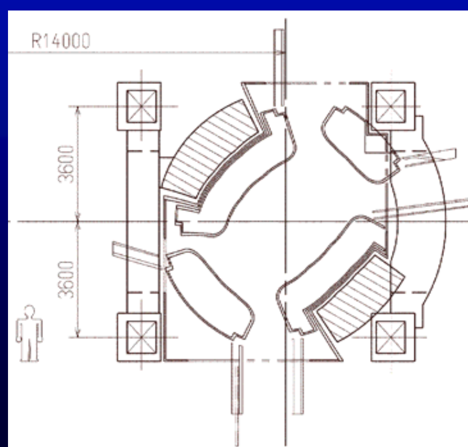
on poloidal coils

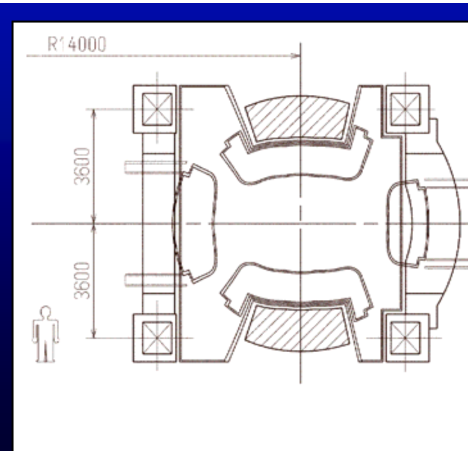
Replacement of In-Vessel Components (1/2)

- **Large size maintenance ports** at top, bottom, outer and inner sides.
- **The vacuum boundary located** just inside of the helical coils and supporting structure.
- **Blanket units supported on the permanent shielding structures**, which are mainly supported at their helical bottom position.





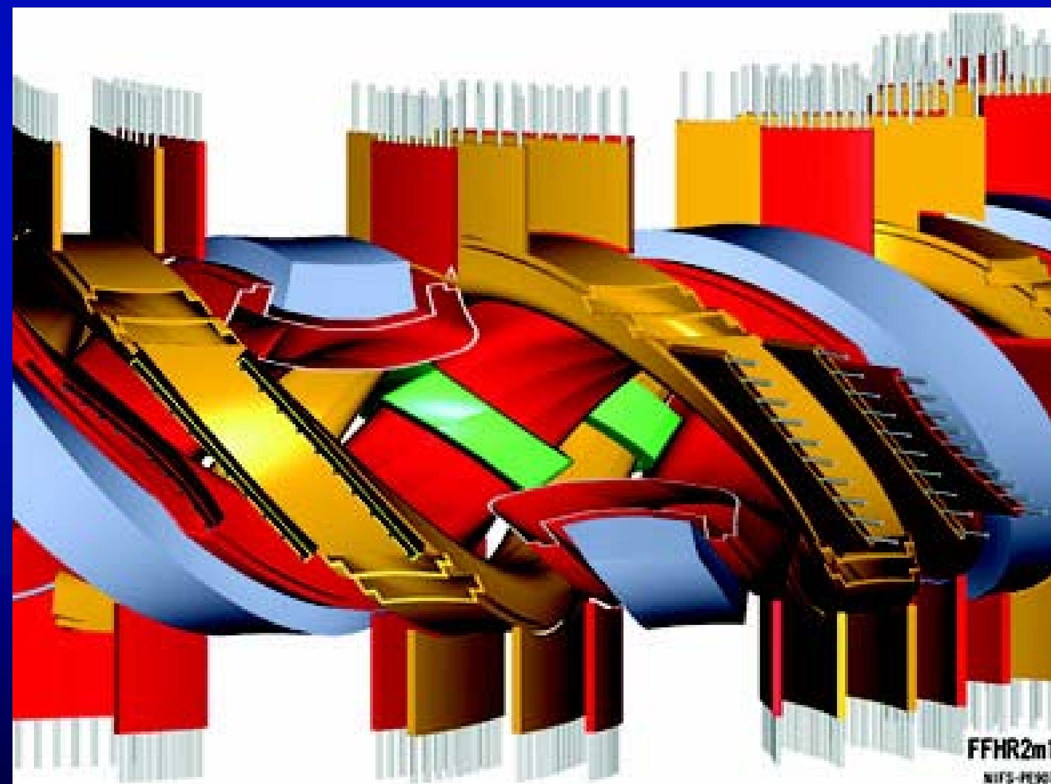




Replacement of In-Vessel Components (2/2)

Proposal of the "Screw coaster" concept to replace STB armor tiles

- **Replacement of bolted tiles** during the planned inspection period.
- **Using the merit of helical structure,** where the normal cross section of blanket is constant.
- **Toroidal effects can be adjusted** with flexible actuators.



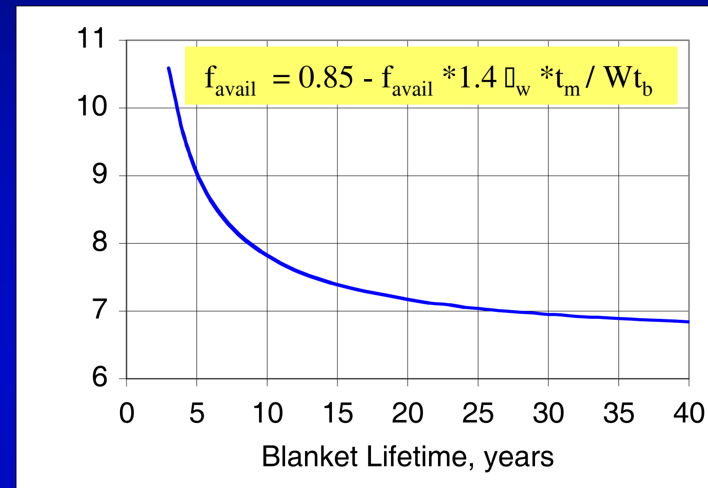
Cost Estimation
Using PEC code developed in NIFS.

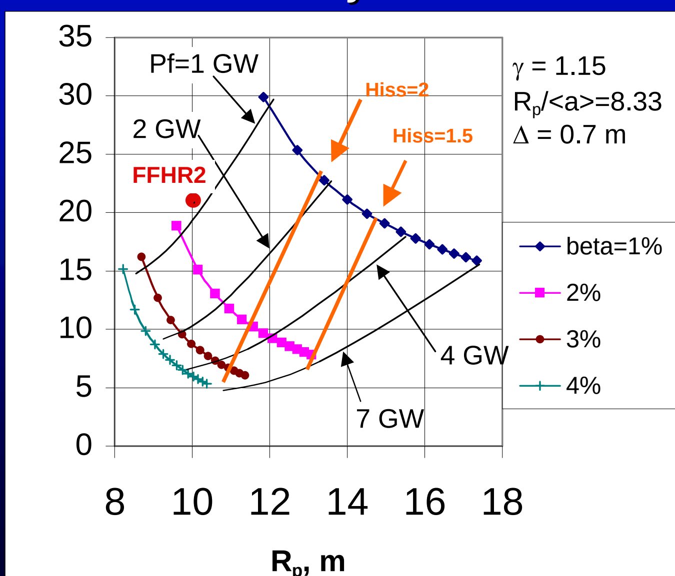
Calibrated with ARIES-AT, ARIES-SPPS, resulting in good agreement within 5 %.

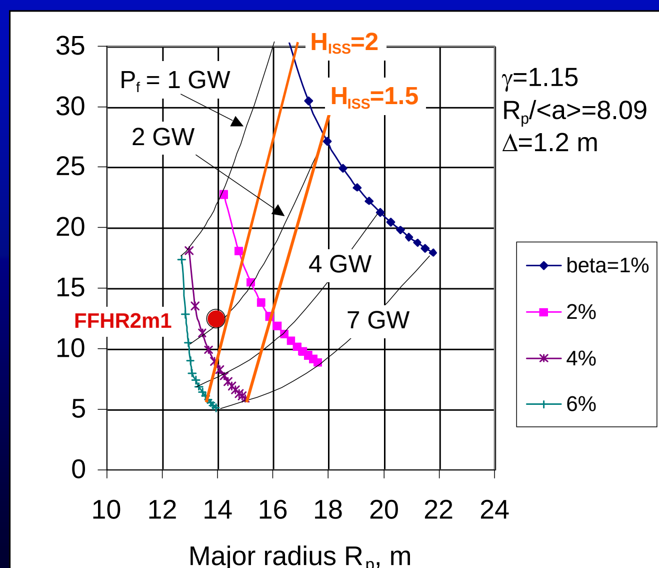
(T.J.Dolan,K.Yamazaki, A.Sagara, in press in Fusion Science & Tech.)

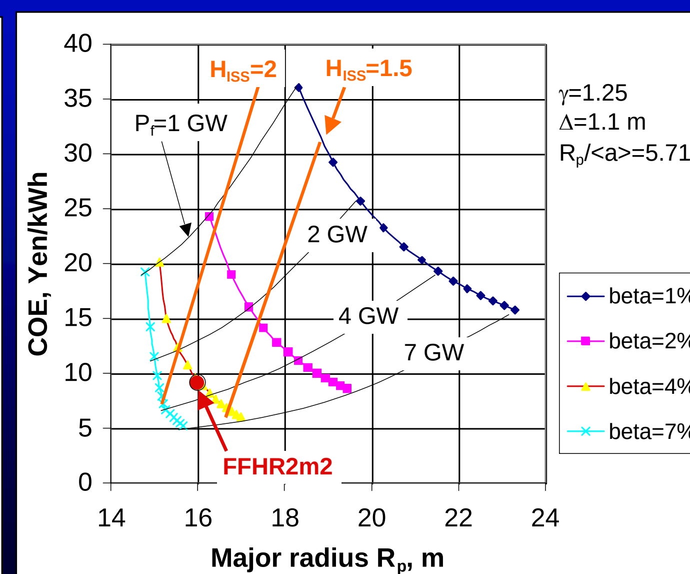
- **COE's for FFHR2, FFHR2m1 and FFHR2m2 decreases with increasing the reactor size,** because the fusion output increases in ~ R^2, while the weight of coil supporting structure increases in~ $R^{0.4}$ (not ~R^3).

- **When the blanket lifetime ~ 30y, the COE decreases ~20%** due to higher availability and lower cost for replacement.









Conclusions

Design studies on FFHR have focused on new design approaches to solve the key engineering issues of blanket space limitation and replacement difficulty.

(1) The combination of improved support structure and long–life breeder blanket STB is quite successful.

(2) The “screw coaster” concept is advantageous in heliotron reactors to replace in-vessel components.

(3) The COE can be largely reduced by those improved designs.

(4) The key R&D issues to develop the STB concept are elucidated.