Overview of LHD Experiments and Helical Reactor Design

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Japan-US Workshop on
Fusion Power Plants and Related Advanced Technologies
with participation of EU

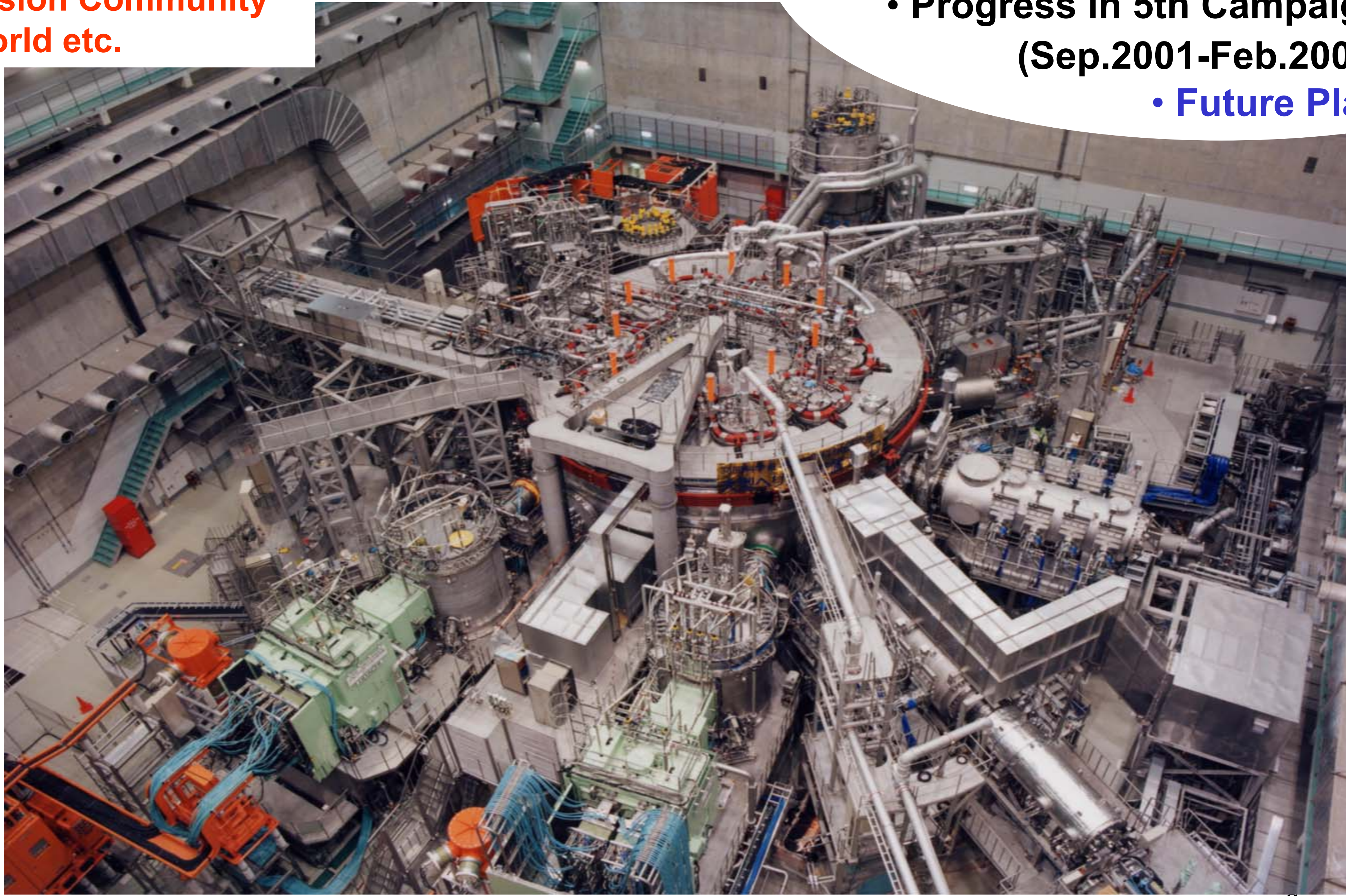
April 6-7, 2002, Hotel Hyatt Islandia, San Diego, CA

Collaboration for
Universities
Fusion Community
World etc.

Part 1 LHD Experiments

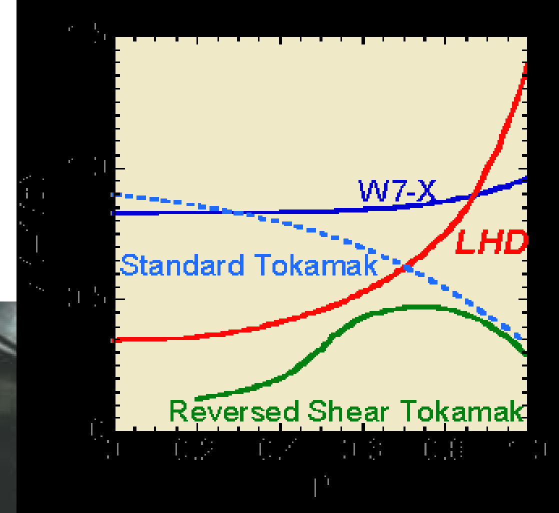
• Progress in 5th Campaign
(Sep.2001-Feb.2002)

• Future Plan

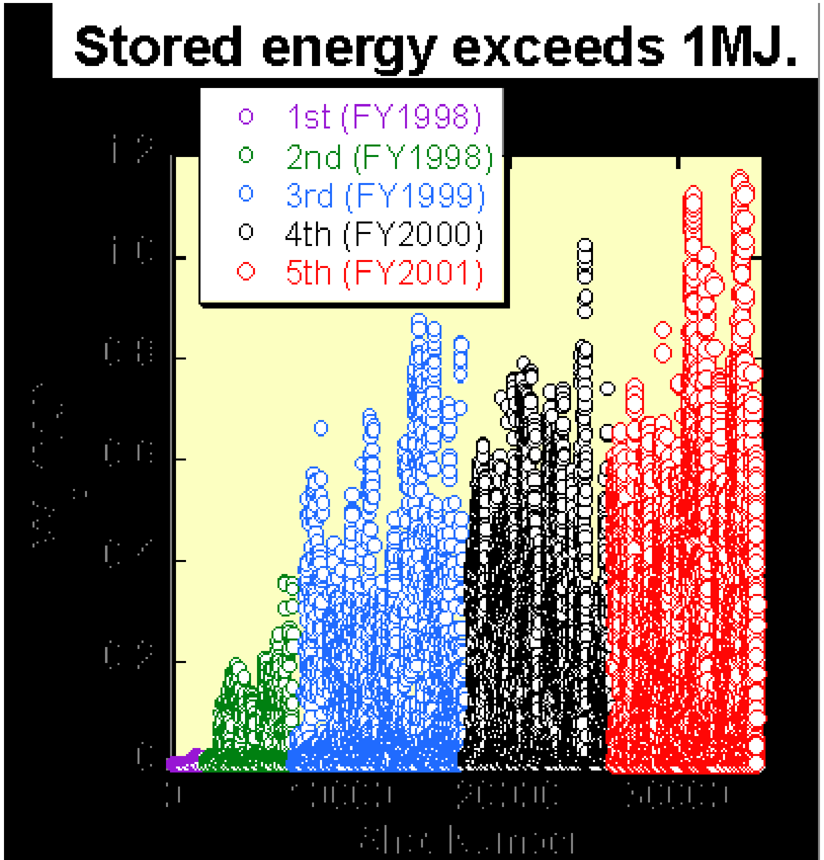


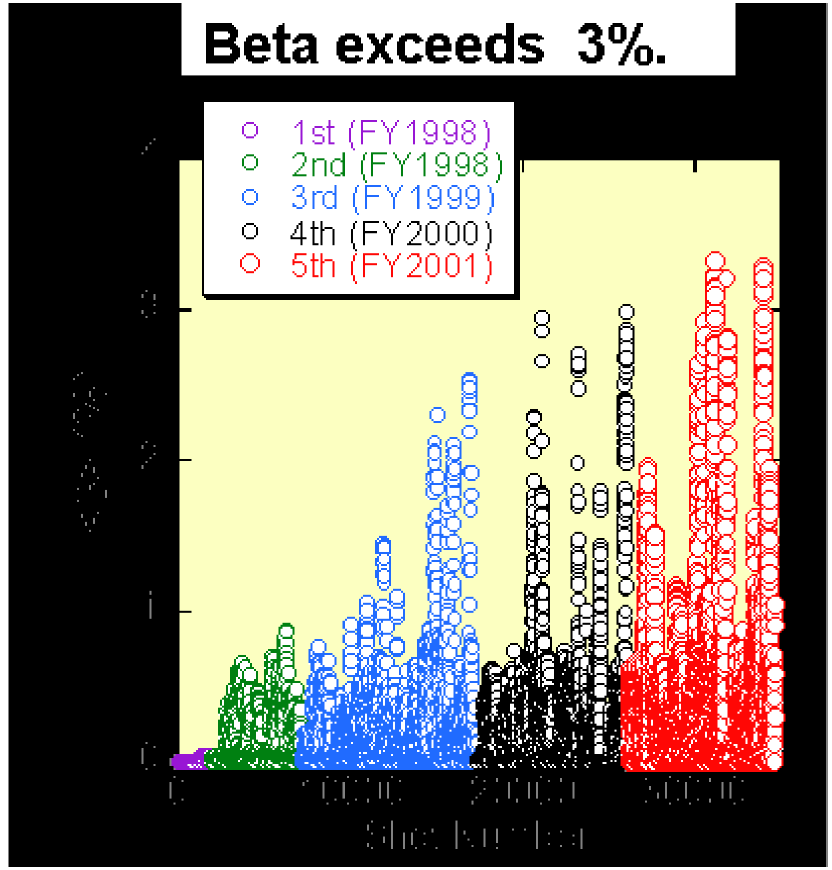
Targets			(Achieved in the 5th camp.)
NBI	9	MW	(9 MW)
ECH	1.5	MW	(1.8 MW)
ICH	3	MW	(2.7 MW)

B @ Rax=3.5 m : 2.976 T

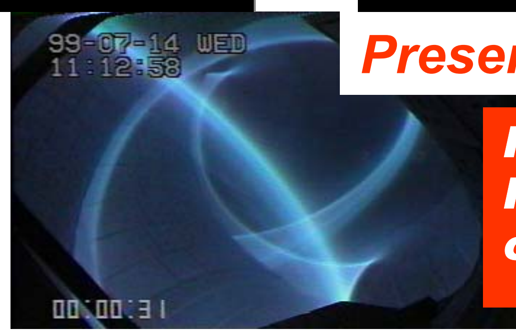


Progress of LHD Experiment





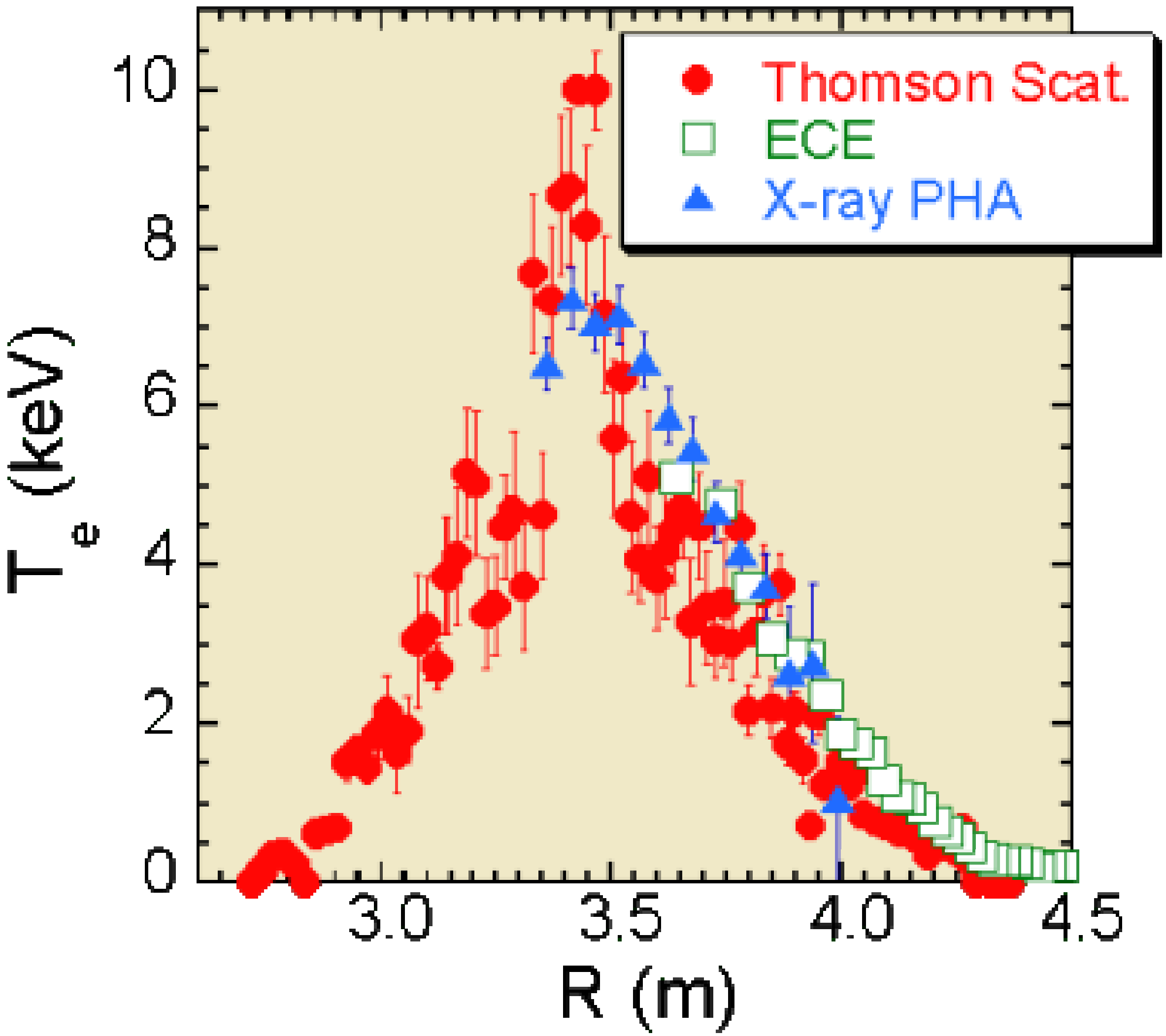
First plasma



Present plasma

Progress of LHD experiment is Rapidly Improving plasma capability.

Electron temperature has reached 10 keV



Achieved parameters

$T_e(0)$	10 keV
$T_i(0)$	2 keV
$<n_e>$	5×10^{18} m^{-3}
τ_E	0.06 s
Discharge duration	0.4s

Experimental conditions

ECRH	1.2 MW
B	2.976 T
R_{ax}	3.5 m

Comparison of achieved maximum parameters

	Tokamaks	Helical systems other than LHD	LHD
Ion temp. (keV)	45 (JT-60U・J)	1.6 (Heliotron E・J)	**5.0**
Electron temp. (keV)	25 (ASDEX-U・FRG) 21(JT-60U・J)	6.1(W7-AS・FRG)	**10**
Density ($10^{20}m^{-3}$)	20 (AlcatorC・US) 2.7 (JT-60・J)	3.5(W7-AS・FRG)	**1.5**
Beta (%)	40 (START・UK) 12.5 (DIII-D・USA)	3.1(W7-AS・FRG)	**3.2**
Energy conf. time (s)	1.2 (JET・EU) 1.1 (JT-60U・J)	0.055 (W7-AS・FRG)	**0.36**
Fusion triple product ($10^{20}m^{-3}$・s・keV)	15 (JT-60U・J)	0.035 (W7-AS・FRG)	**0.22**
Stored energy (MJ)	11 (JET・EU) 10.9 (JT-60U・J)	0.02(W7-AS・FRG)	**1.16**

Accumulation of Database for Stellarators

Heliotron E (Kyoto Univ.)
ATF (ORNL)
W7-AS (IPP Garching)
CHS (NIFS)
R/a<2.2m/0.28m

LHD 1999 ⇒

in plan

W7-X 2006-
(IPP Greifswald)
R/a=5.5m/0.53m
B=3T

NCSX 2007-
(PPPL)
R/a=1.42m/0.33m
B>2T

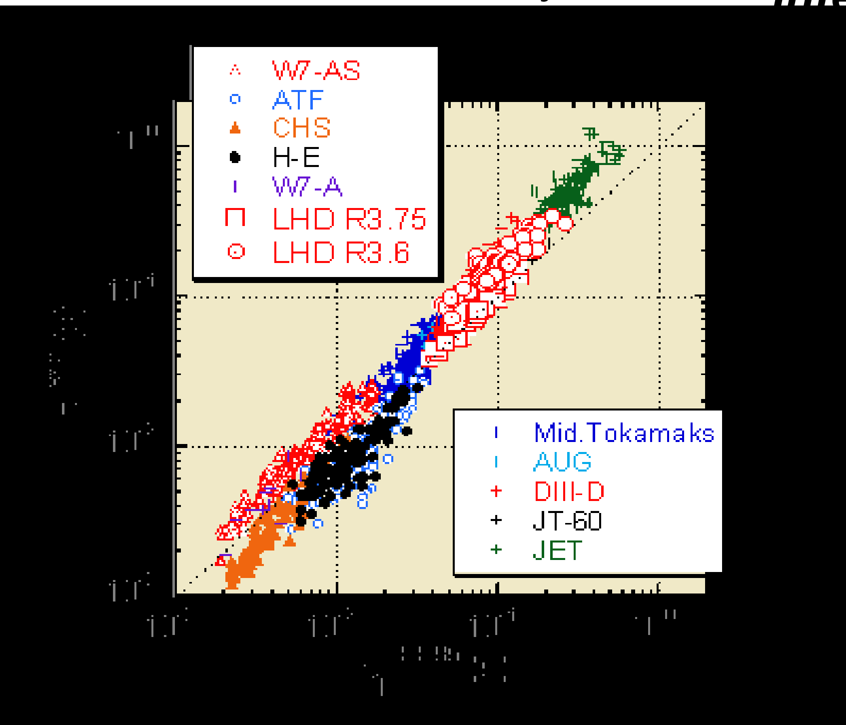
International stellarator database
Nucl. Fusion 36 (1996) 1063

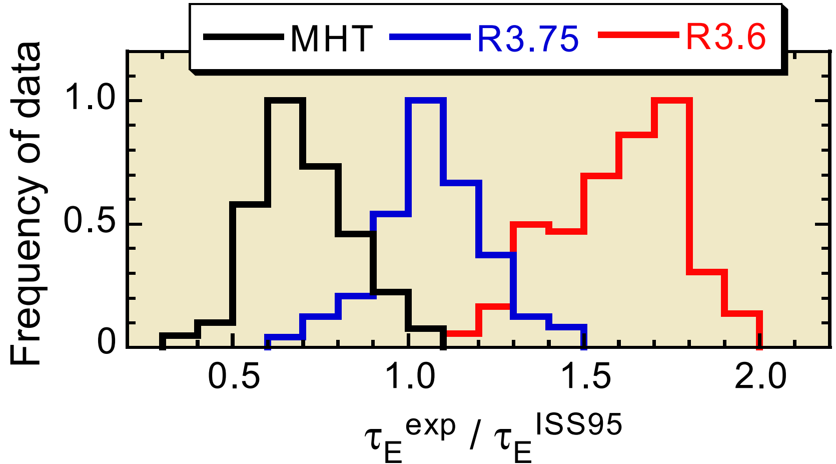
- Description and format in conformity of ITER DB.
- Scalar data only

International Stellarator Scaling 95 (ISS95)

$$\tau_E^{ISS95} = 0.26\, B^{0.80} P^{-0.59} \overline{n}_e^{0.51} R^{0.65} a^{2.21} q_{2/3}^{-0.40}$$

$$\propto \tau_B \rho_*^{-0.71} \beta^{-0.16} \nu_*^{-0.04}$$





Reliable long pulse operation

LHD plasma is confined by stable steady-state magnetic field.

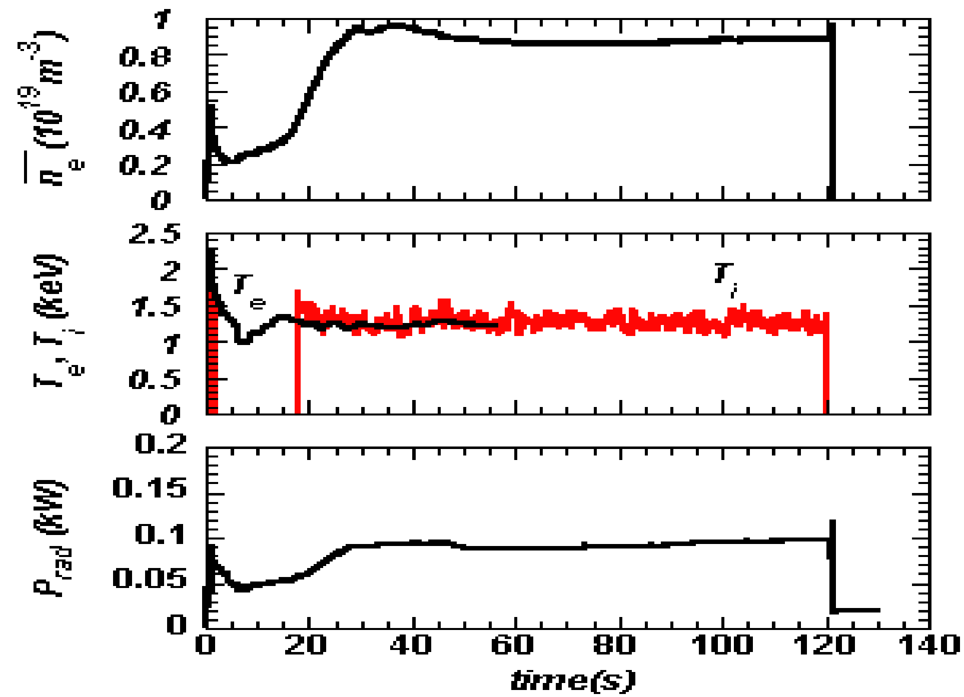
➔ Intrinsic reproducibility and reliability has been demonstrated in experiments up to 2 min.

(1) Disruption has never been observed.

Note: When the radiation power exceeds 50 % of heating power, radiation collapse in the time scale of energy confinement time occurs.

(2) Long pulse in a couple of minutes is easily available

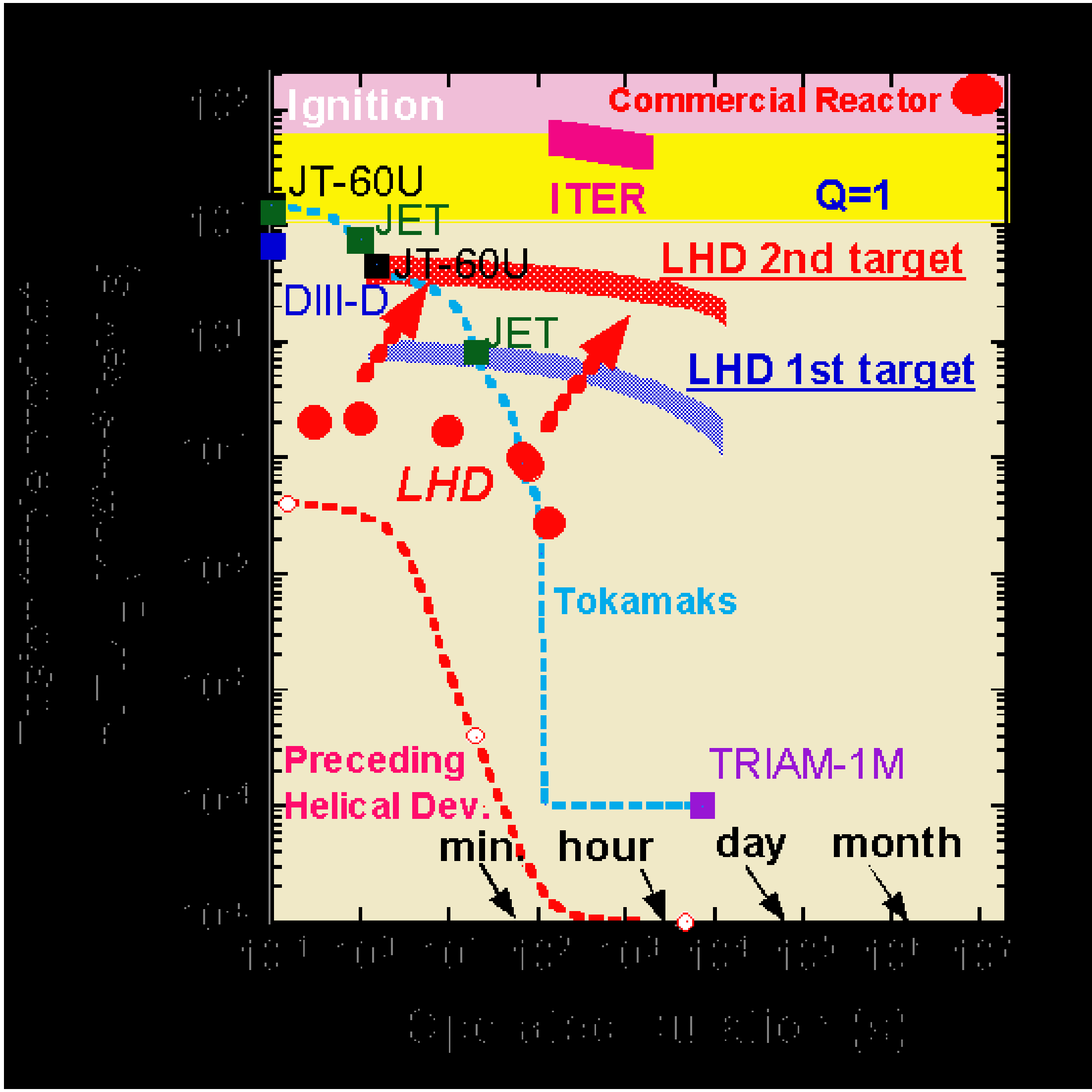
No significant change during the pulse. Plasma performance such as energy confinement time is the same as in the short-pulse discharge.



ICRH(0.8MW) sustains 2 min. discharge.

(80sec in case of NBI/0.5MW)

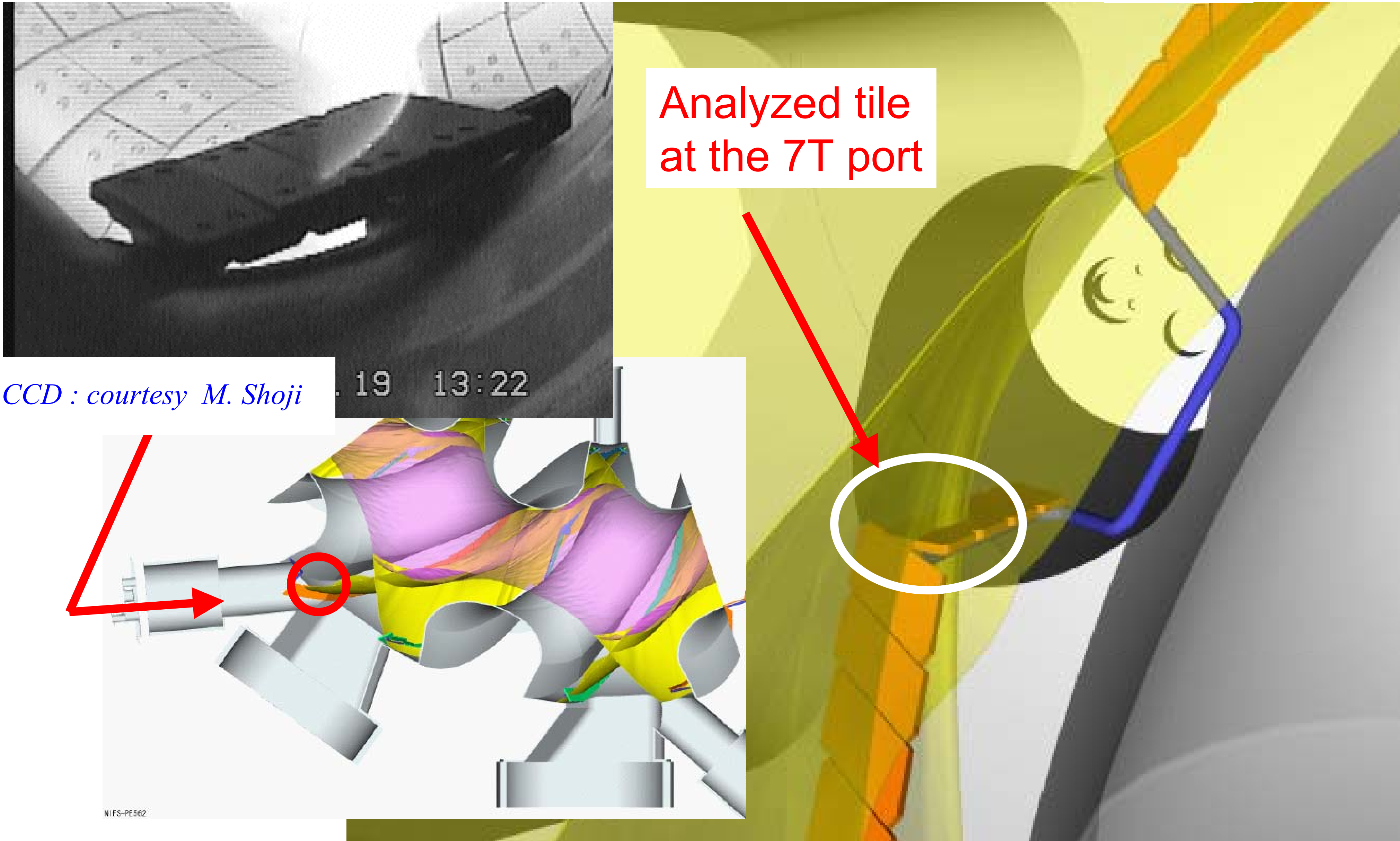
LHD is extending the frontier on Currentless steady-state plasmas



1. Transport studies in sufficiently high $n\tau_E T$ regime relevant to reactor condition.
2. MHD studies beyond β of 5%.
3. Fundamental research for steady-state operation with employing divertor.
4. Confinement studies on high energetic particles and simulation experiments of alpha particles.
5. Complimentary study to tokamaks leading to comprehensive understanding of toroidal plasmas.

Surface Analysis of Divertor Tiles

A.Sagara et al., to be submitted to J.Nucl.Mater.



LHD dust collection on March ‘01

Collaboration with INEEL

Idaho National Engineering and Environmental Laboratory

Preliminary results

- total amount of dust collected (16.5 mg) much less than amounts collected in other fusion devices-

  LHD is very clean!

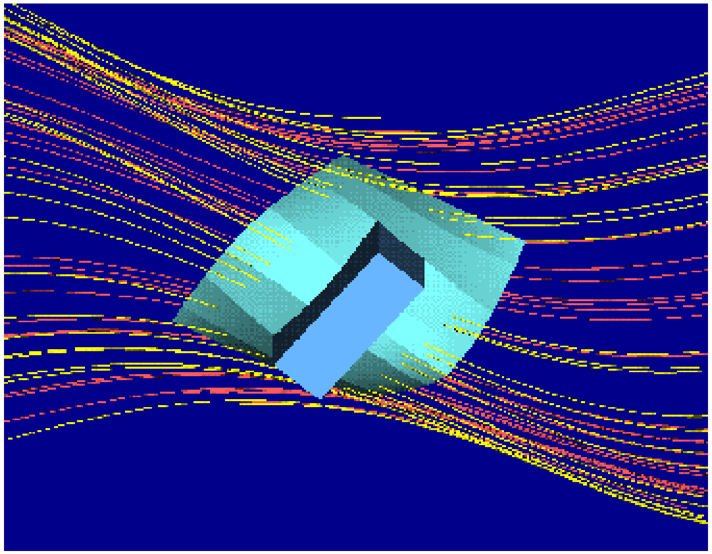
- median particle diameters are larger than those of other devices; mechanisms generating the smaller particulate may not be present in LHD? (important for safety)

J.P.Sharpe, A.Sagara et al., to be submitted to PSI-15

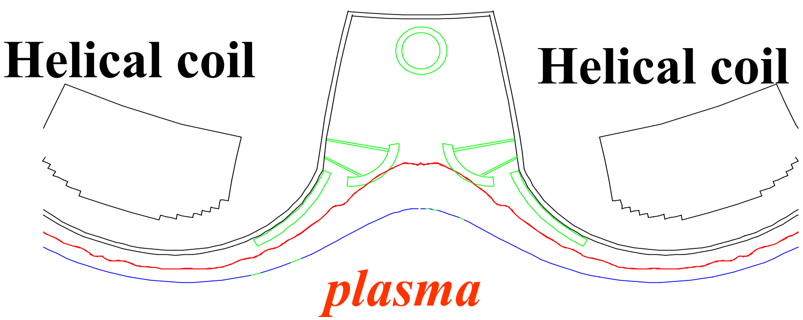
next 6th Experimental Campaign in this year
Divertor and Fueling

LID (2002)

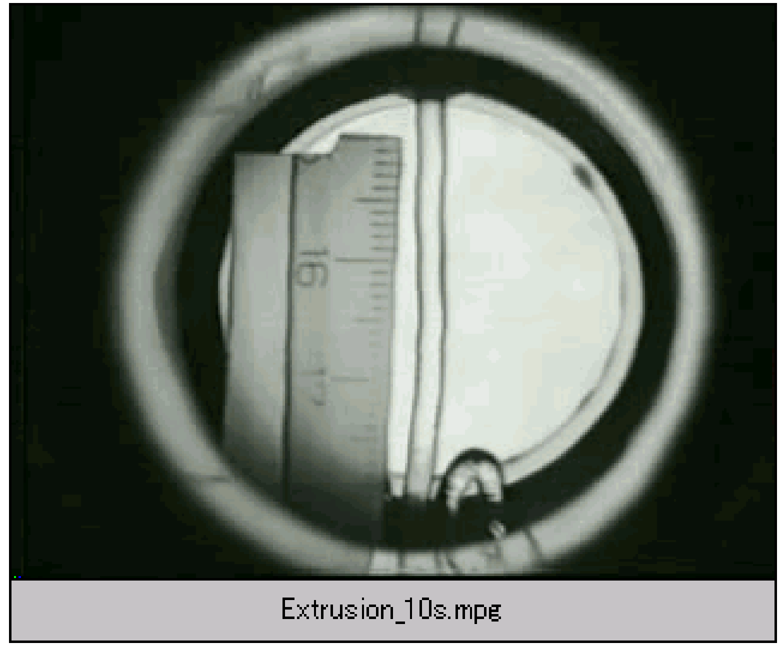
→ Closed divertor (2007)



10MW in steady-state

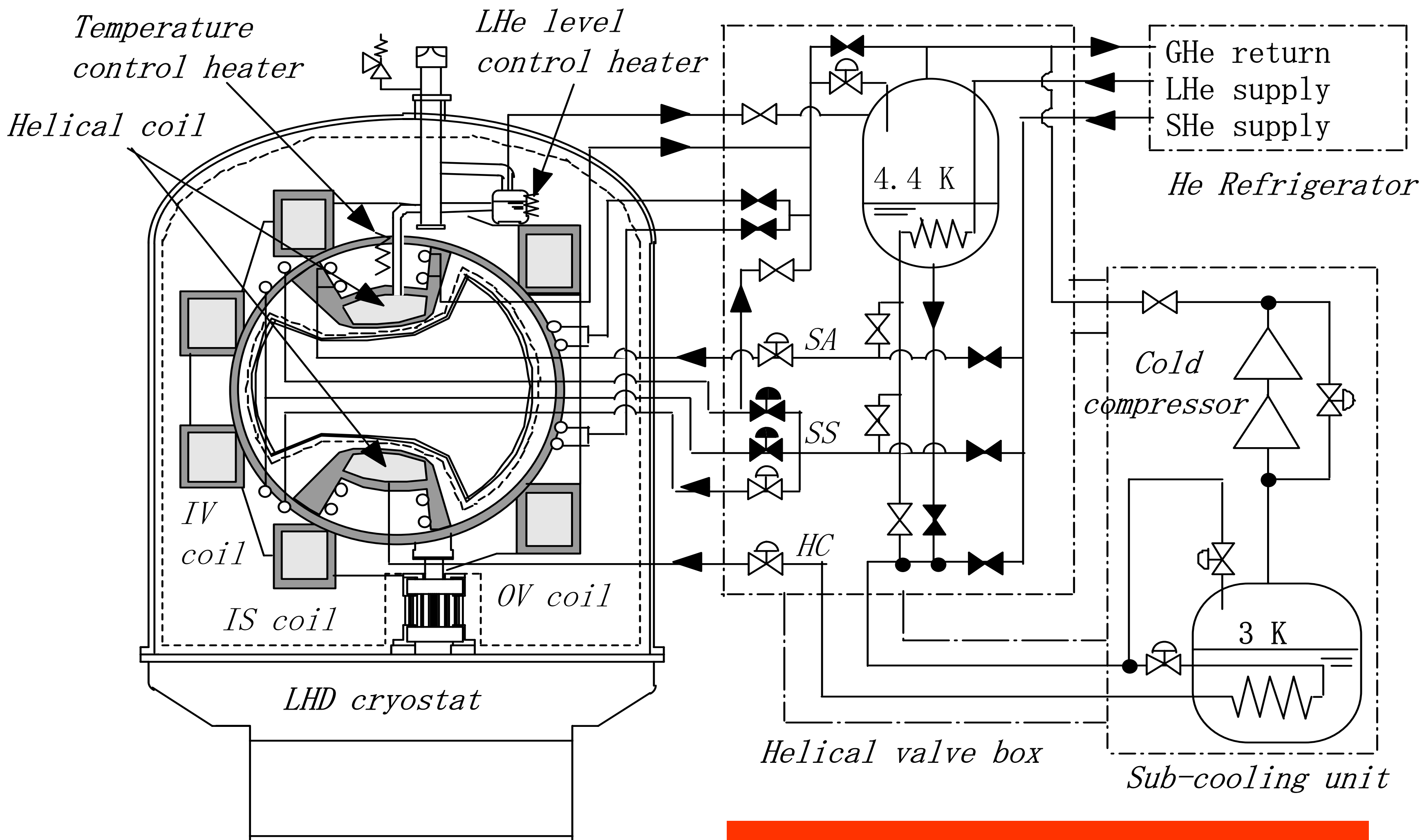


10 pellets/shot → Infinite (2002)



1 hour-long extrusion of hydrogen ice has been already demonstrated in the proto-type without use of liquid helium. Cooled by GM-cycle refrigerator.

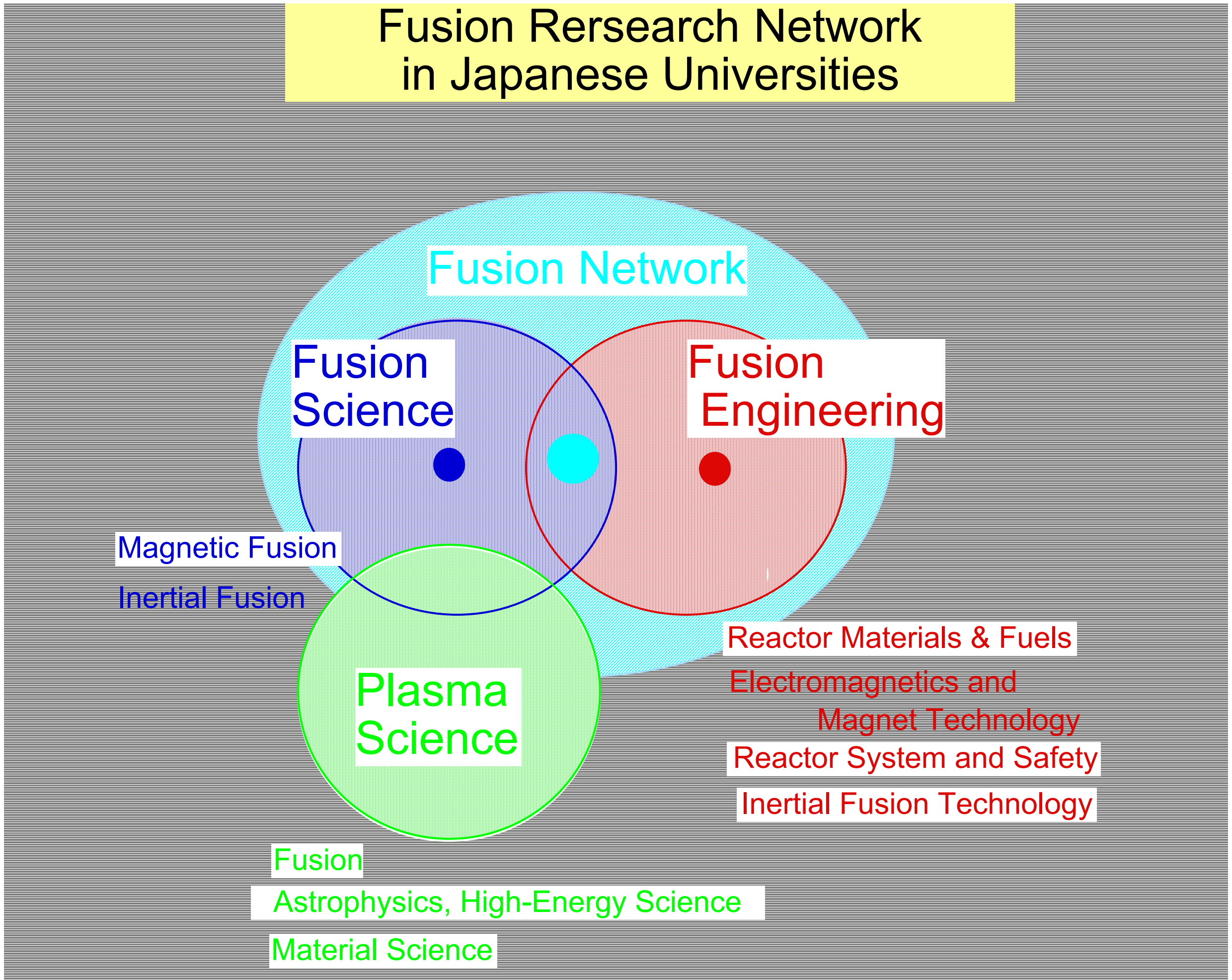
Near Term Up-Grading
Sub-cooling system for LHD helical coils



Realize 3Tesla Operation

Part 2 Helical Reactor design

- **Fusion Research Network in Japan Universities**
- **Design activities in NIFS collaboration**



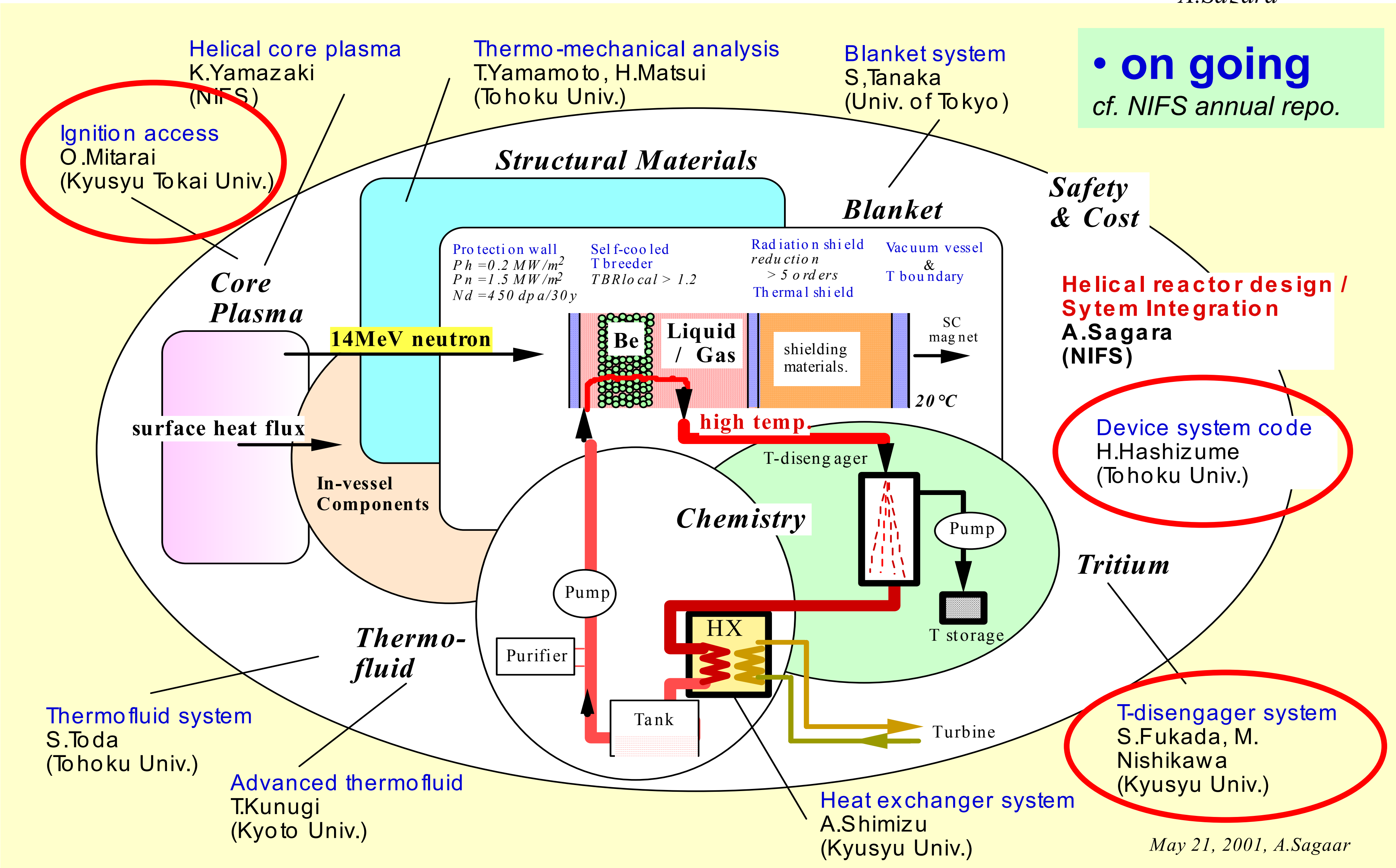
Fusion Engineering Network in Japanese University, 2001

Working Group	Research Fields
Material & Fuel *(K.Abe)*	**Plasma Facing Materials** *: N.Noda (NIFS)* **Structural Materials** *: K.Abe (Tohoku U.), T.Muroga (NIFS)* **Blanket Technology** *: S.Tanaka (U.Tokyo)* **Tritium Science & Technology** *: M.Nishikawa (Kyusyu U.)*
Magnet & Electromagnetics *(M.Takeo)*	**Superconducting Magnet** *: M.Takeo (Kyusyu U.), T.Satow (NIFS)* **Electromagnetic Engineering** *: H.Hashizume (Tohoku u.)* **Fusion Electrical Power Engineering** *: R.Shimada (Tokyo Inst. Tech.)*
Reactor System & Safety *(T.Uda)*	**Tritium Biological Effects** *: Y.Ichimasa (Ibaragi U.), K.Komatsu (hiroshima U.)* **Thermal & Mechanical Engineering** *: S.Toda (Tohoku U.), A.Shimizu (Kyushyu U.)* **Reactor Design Engineering** *: A.Sagara (NIFS), Y.Ogawa (U.Tokyo)* **System Safety Engineering** *: T.Uda (NIFS)* **Neutronics** *: T.Iguchi (Nagoya U.)*
Inertial Confinement Fusion *(S.Nakai)*	**Inertial Fusin Engineering** *: S.Nakai (Osaka U.), K.Mima (Osaka U.)*

Steering Committee : N.Yoshida (Chair.)(Kyusyu U.), A.Sgara, T.Iguchi
Advisory Committee : T.Yamashina (Sapporo Int. U.), O.Motojima (NIFS), A.Kohyama (Kyoto U.), M.Nishikawa (Osaka U.), H.Matsui (Tohiku U.), H.Moriyama (Kyoto U.)

NIFS Collaboration on Reactor Design

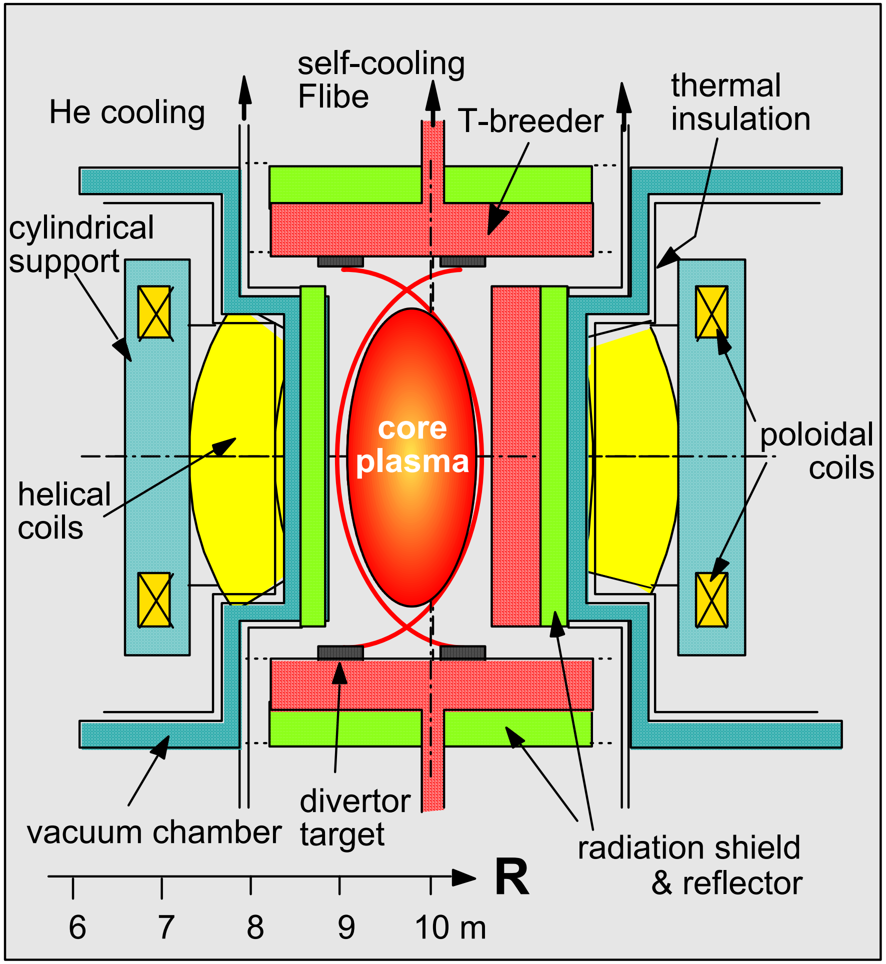
A.Sagara



Phase I ----- Concept definition

1993 **Design of FFHR-1 (l=3)**
1994 ---> *NIFS Collaboration*
1995 **Design of FFHR-2 (l=2)**
1998 ---> *Fusion Eng. Network*
2001 ---> *Liq. Blanket in UPITER-II*

FFHR-2



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

Φ_n = 1.5MW/m^2 (120dpa/10y)

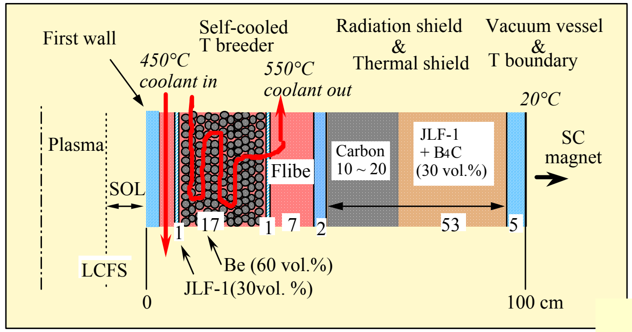
Parameters	LHDFFHR-1 FFHR-2
major radius : R	3.9 2010 m
av. plasma radius : <a_p>	< 0.65 1.2 m2
fusion power : P_f (GW)	– 3 1 GW
external heating power : P_ex	< 20 100 100 MW
neutron wall loading : P_n	– 1.51.5 MW/m2
toroidal field on axis : B_0	4 1210 T
average beta <β>	> 5 0.7 1.[illegible]
enhancement factor of τ_LHD	2.51.5
plasma density : n_e(0)	1.E2 2.8E20 2.E20 m-3
plasma temperature : T_e(0)	> 10 2227 keV
effective ion charge : Z_eff	1.51.5
alpha heating efficiency : h_α	– 0.70.7
alpha density fraction : f_α	– 0.05 0.05
synchrotron reflectivity : R_ef	– 0.90.9
hole fraction : fh	– 0.10.1
av. heat load on divertor	< 10 1.61.5 MW/m2
number of pole : l	2 3 2
toroidal pitch number : m	10 1810
pitch parameter γ :	1.12<1.25 <1.37 1.115
coil modulation α :	+ 0.1 0 0.1
av. helical coil radius : <a_c>	<0.975 3.33 2.30 m
coil to plasma clearance δ :	0.03 1.10.70 ~ 1.25 [illegible]
coil current : I_H	7.8 66.6 50 MA/coil
coil current density : J	(53) 2725 A/mm2
max. field on coils : B_max	(9.2) 1613 T
stored magnetic energy	1.64 1290 147 GJ
construction cost	50 Byen

Q~30, η_{th}~38%, availability~0.8

FFHR Design Concept & Optimization

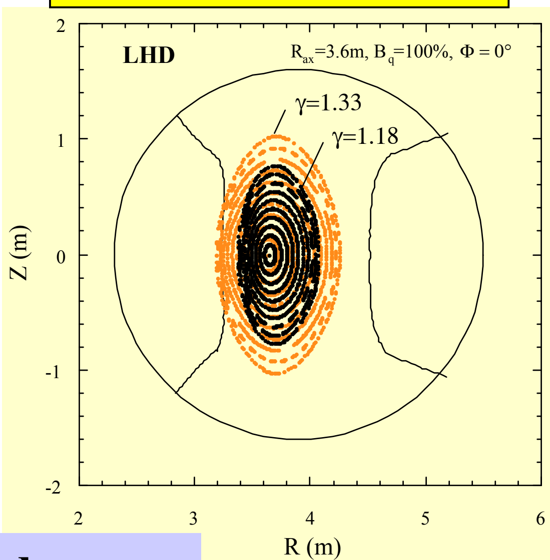
Enhancing inherent safety & Thermal efficiency

Molten-salt blanket with low MHD effect



Heat transfer & structure materials

Expansion of blanket area



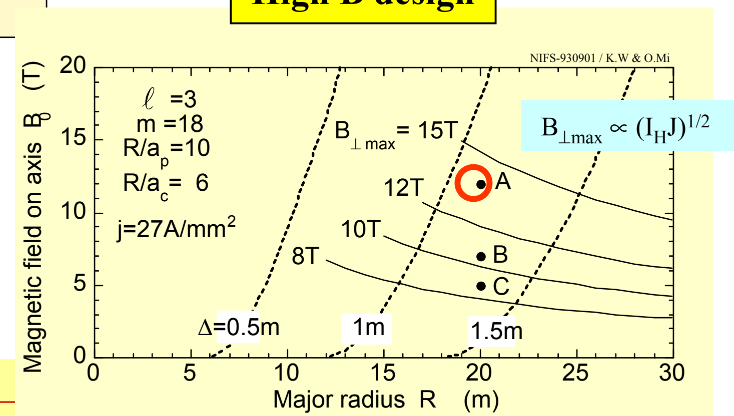
decrease

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

MHD stability

SC materials & Current density

High B design

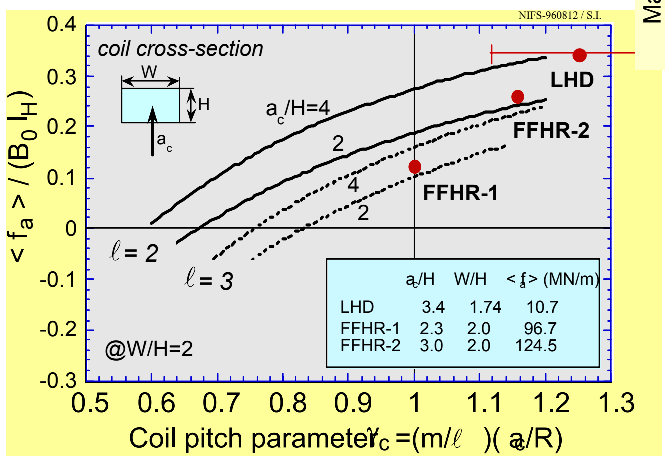


Ignition conditions for the 3GW fusion output

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



Reduction of magnetic force

Simplification of supporting structure

Continuos winding helical coil

Continuous Winding Helical Coil

- **R&D and operation in LHD**
- **Development of SC joint**
  (a few % of power load)

  → Remountable HTcS.C.

  H.Hashizume et al., ITC-12
- **Reduction of magnetic force**

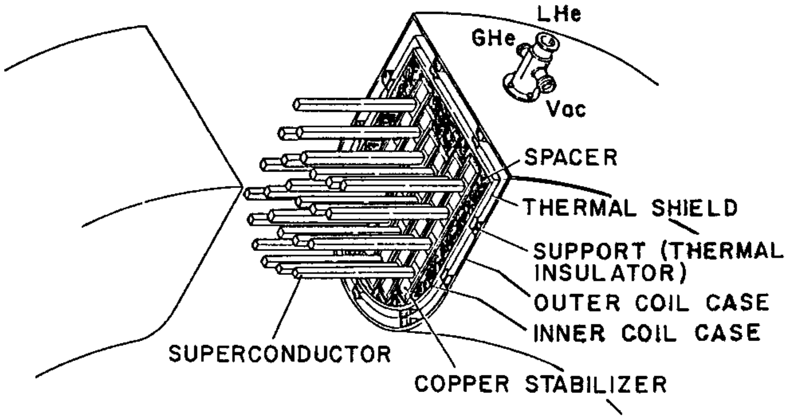
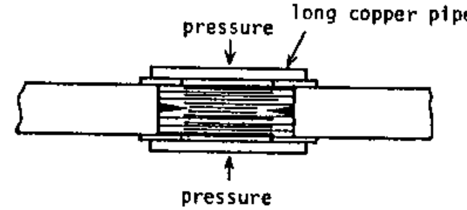
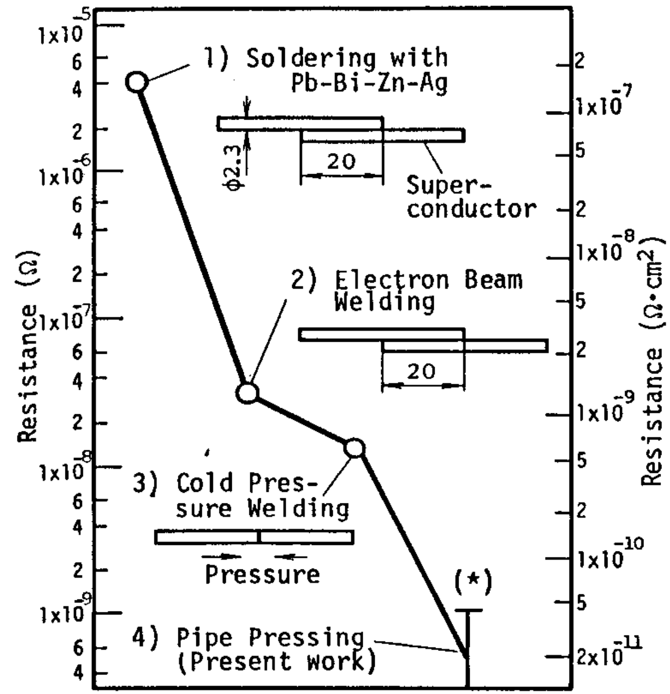


Fig. 2 Details of the joint section using the "Inserting Method".

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





H.Kita et al., Fusion Technol. 8 (1985) 1698

Modified FFHR-2

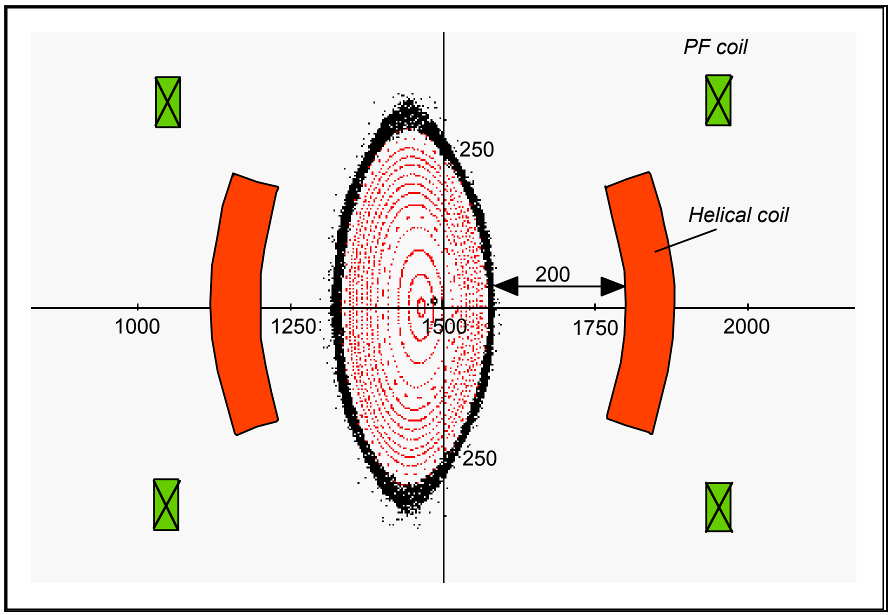
pressented in 12th Intenational Toki Conference (2002)

One way
for optimization

IPP Garhing : H. Wobig, J. Kisslinger

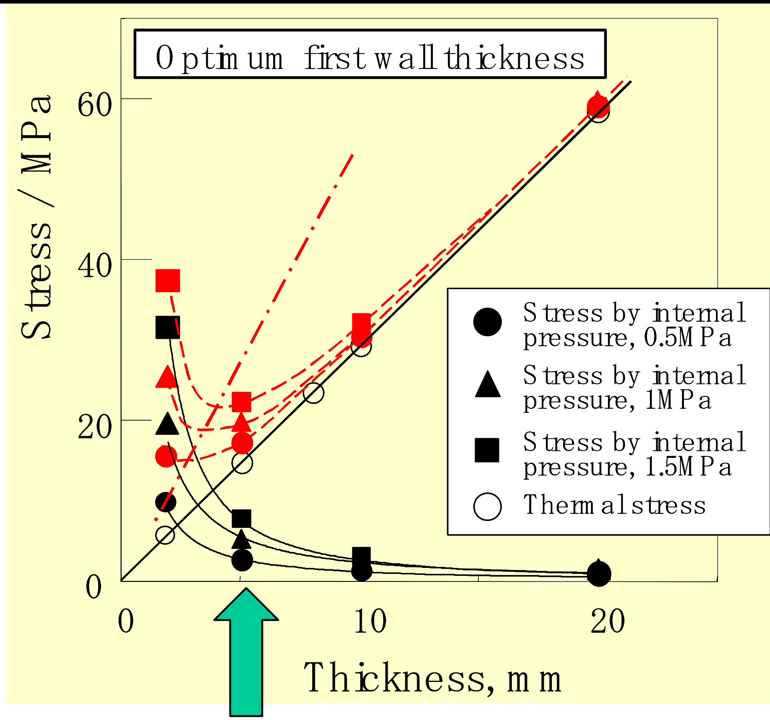
Increase of the FFHR2 configuration by 50% and
Reduction of the magnetic field to 6 T on axis

- to provide more space for blanket and shield
- to use NbTi superconductor in the helical windings.



	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^{-3}]
Electron Temp(0)	15.0	15.0	[keV]
Av. Beta	0.96	2.66	[%]
Fusion Power	0.62	2.09	[GW]
Heating Power	94	333	[MW]
Energy Conf. Time	1.72	1.64	[s]

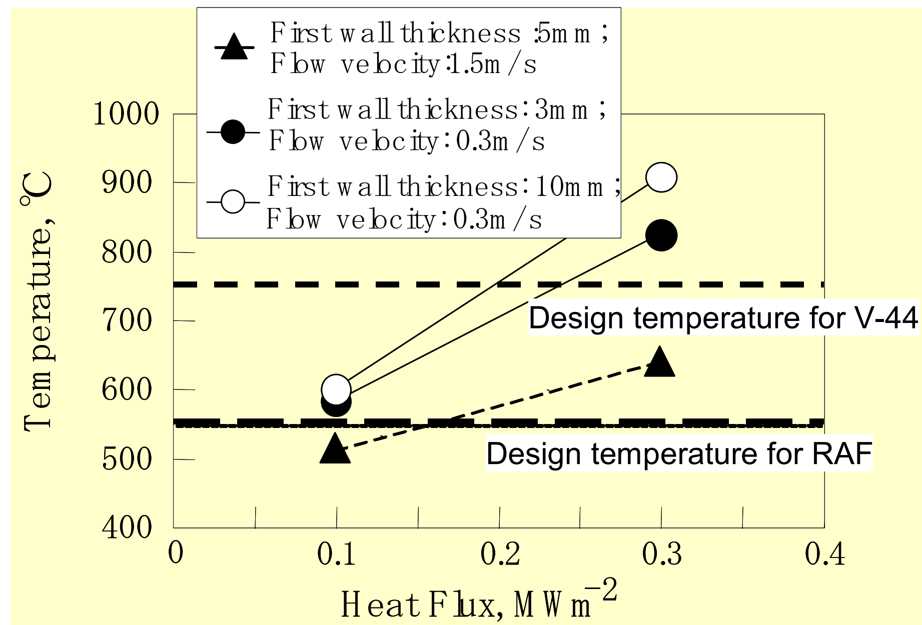
Surface heat flux > 0.1MW/m² is quite severe for the first wall materials



optimum thickness ~ 5mm

@ temperature = 600°C
@ stress < 20MPa
@ heat flux = 0.1MW/m²
@ flow velocity = 0.3m/s.

It is possible to increase the **heat flux up to 0.25MW/m²** with adopting V-4Cr-4Ti



by H.Matsui et al, NIFS Annual Repo. (2000) 175.

Japan-US joint project **JUPITER-II**

From FY2001 to 2006

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

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

Japan

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

UCLA

ORNL

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

ANL

1-2-A: Coatings for MHD Reduction

S A.Sagara

Summary

(1) LHD is exploring new perspective towards attractive fusion reactor by net current-free plasmas.

(2) New findings as well as performance improvement have extended frontiers of plasma physics.

(3) The 5th experimental campaign was successfully completed by the end of February 2002.

(4) NIFS collaboration on reactor design has extended its activities in Fusion Engr. Network in Japan universities.

(5) Optimization studies on FFHR design concept is the next step with the most up-to-date results from LHD.

(6) Worldwide joint works have been partly on going and will deserve to be planned programmatically.