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Activities of National Institute for Fusion Science towards Realization of Helical Fusion Reactor

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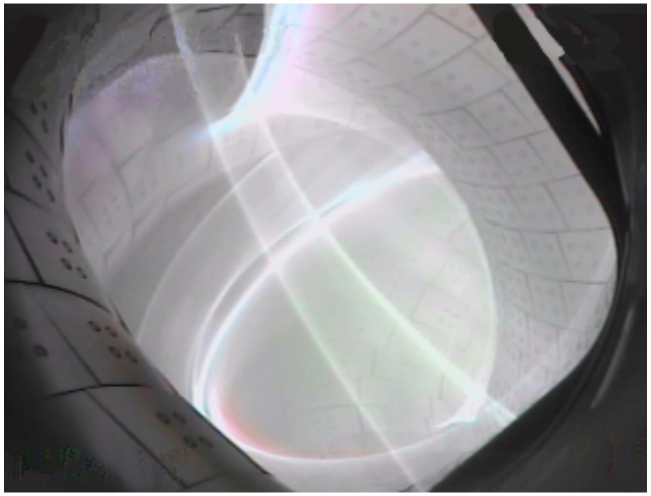
Outline



1. Introduction
- Roles of NIFS

2. Tokamak vs LHD – similarities and differences
- Magnetic configuration, Plasma characteristics
- Unique features of LHD

3. Current activities in NIFS towards helical DEMO
- LHD
- Numerical Experiment research
- Fusion Engineering research

4. Summary

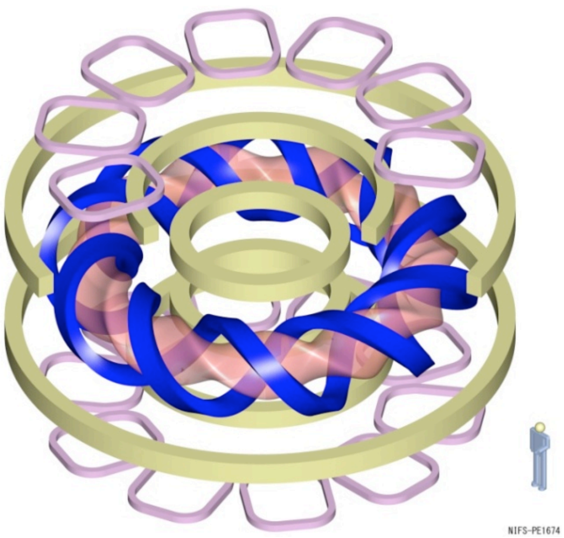
Roles of NIFS

National Institute for Fusion Science
established on 29 May, 1989 by MOE
as an inter-university research organization

- **Aiming at realization of fusion energy no later than in the mid of the 21st century, NIFS pursues academic approach towards**
  - **Exploration of fusion science for steady-state helical reactor**
  - **Establishment of comprehensive understanding of toroidal plasmas**

- **Major Research Categories;**

Large Helical Device
Numerical Simulation Science
Fusion Engineering
+ *Education of graduate students*



In order to understand what LHD is…

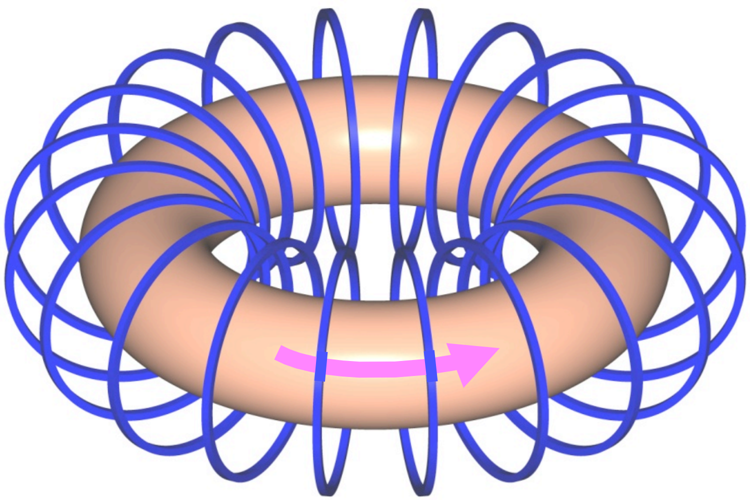
Tokamak vs LHD(heliotron)
- similarities and differences -

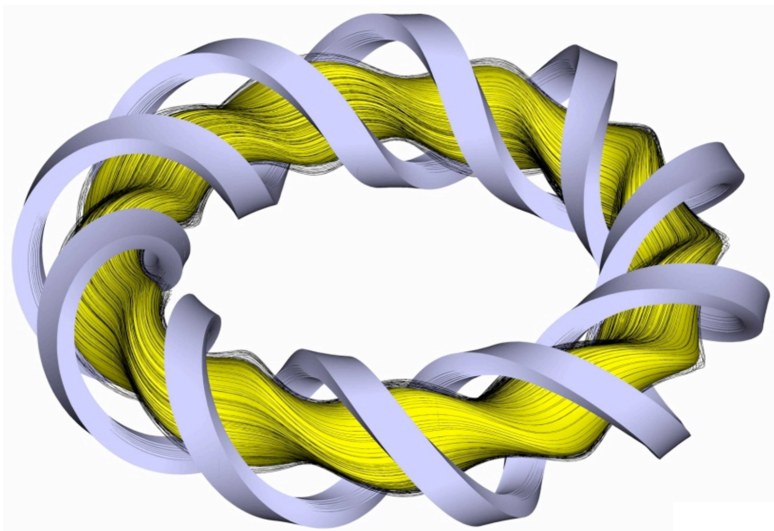
Magnetic coil system

Both have toroidal geometry

Tokamak: Set of toroidal coils and poloidal shaping coils + plasma current

LHD: Pair of helical coils and poloidal shaping coils



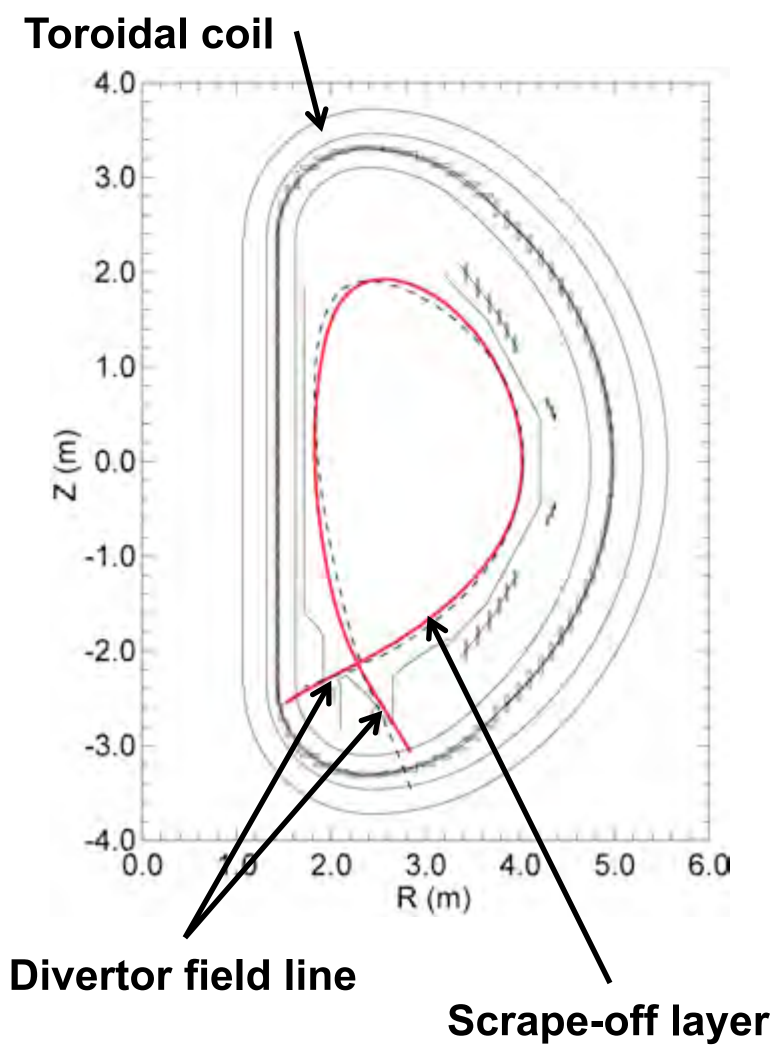


- Approximately 2-D structure
- Needs large toroidal current in plasma to confine plasma
- Low aspect ratio (R/a)
- Needs Ohmic coils to flow toroidal current and additional sets of coils to make divertor configuration

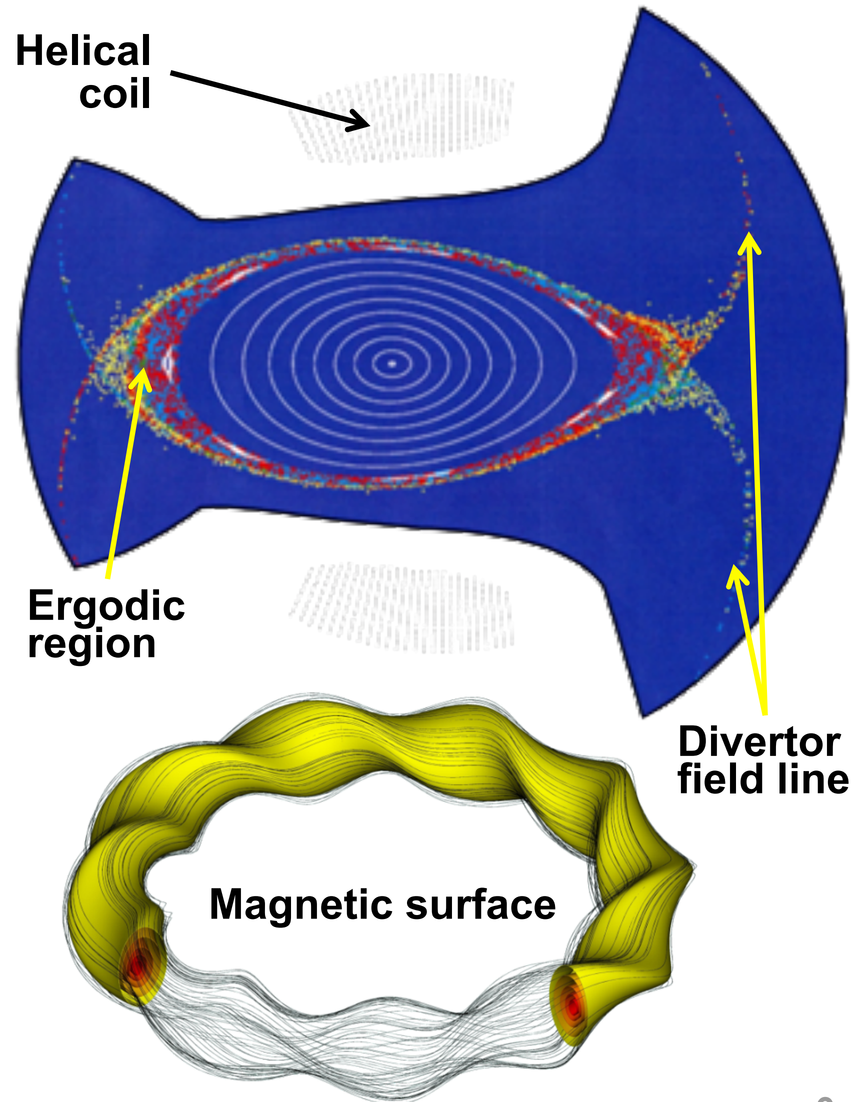
- Intrinsically 3-D structure
- Capable of Steady state operation by external helical coils
- High aspect ratio (R/a)
- Has natural divertor configuration
- Makes large ergodic region outside of closed magnetic surfaces

Magnetic configuration (plasma cross section)

Tokamak: D-shape + X-point

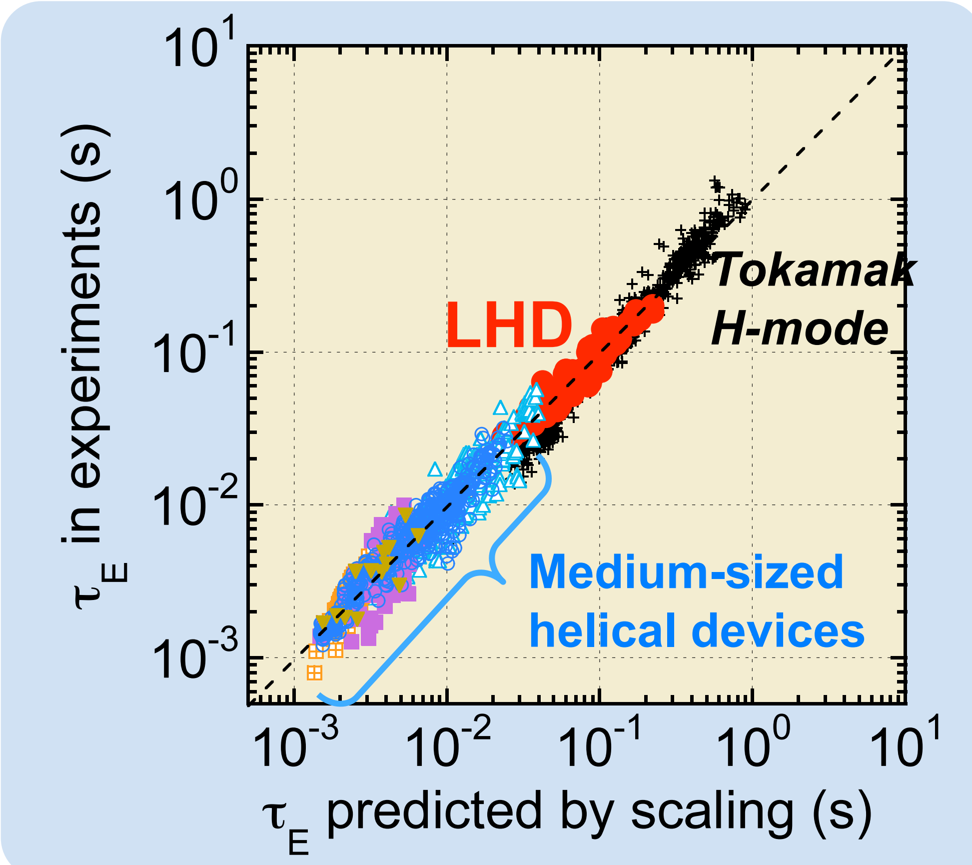


LHD: Eliptical shape + X-points



NIFS-PE413

Global plasma characteristics is similar



International Stellarator/ Heliotron Confinement Database

Expression by operational parameters

a: minor radius
R: major radius
P: heating power
n: density
B: magnetic field
ι : rotational transform
etc.

$$\tau_E^{ISS04} = 0.134 \times a^{2.28} R^{0.64} P^{-0.61} \overline{n}_e^{0.54} B^{0.84} \iota^{0.41}$$

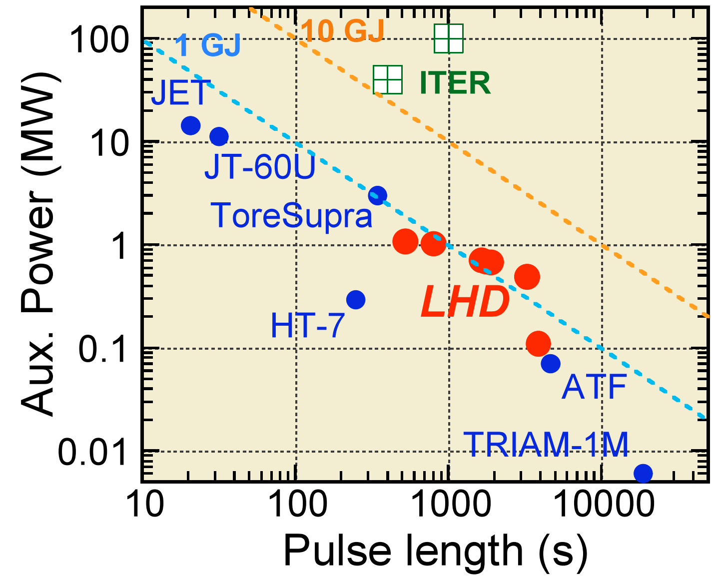
$$\propto \rho^{*-0.79} \beta^{-0.18} \nu^{*0.00} \iota^{1.06} \varepsilon^{-0.08}$$

Reliability of extrapolation much depends on clarification of underlying physics

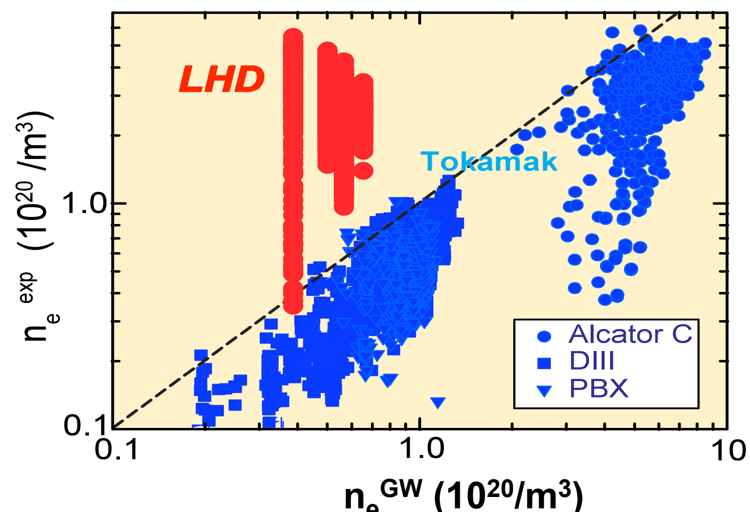
Characteristic in operation is different

In LHD operation,

- Disruption free
  - No net large toroidal current
  - Only mild collapse occurs
- Capable of steady state operation
  - Plasma is confined by external coils only and then stable



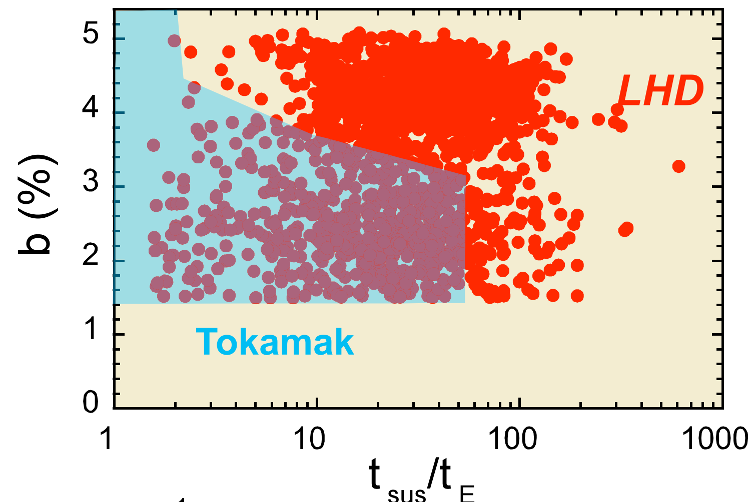
- Capable of very high density operation
  - Maximum density is limited by the condition at the periphery of plasma (radiation balance)



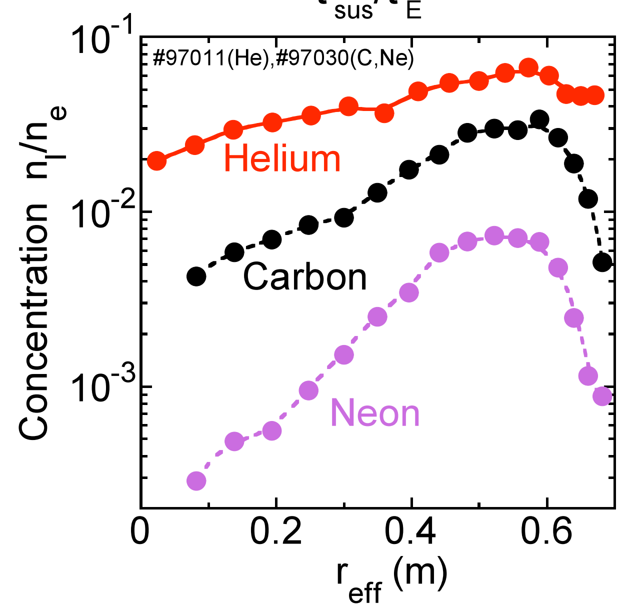
Characteristics of plasma is different

In LHD plasma, we found

- Major MHD activity is pressure driven mode, not current driven mode
  - Quasi-steady state high beta operation is possible
  - Nonlinearity stabilizes plasma



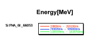
- Ion transport is different from that of tokamaks
  - Impurities do not accumulate in the plasma core but are expelled from the core when the ion temperature gradient becomes large



High energy ions can be confined in LHD

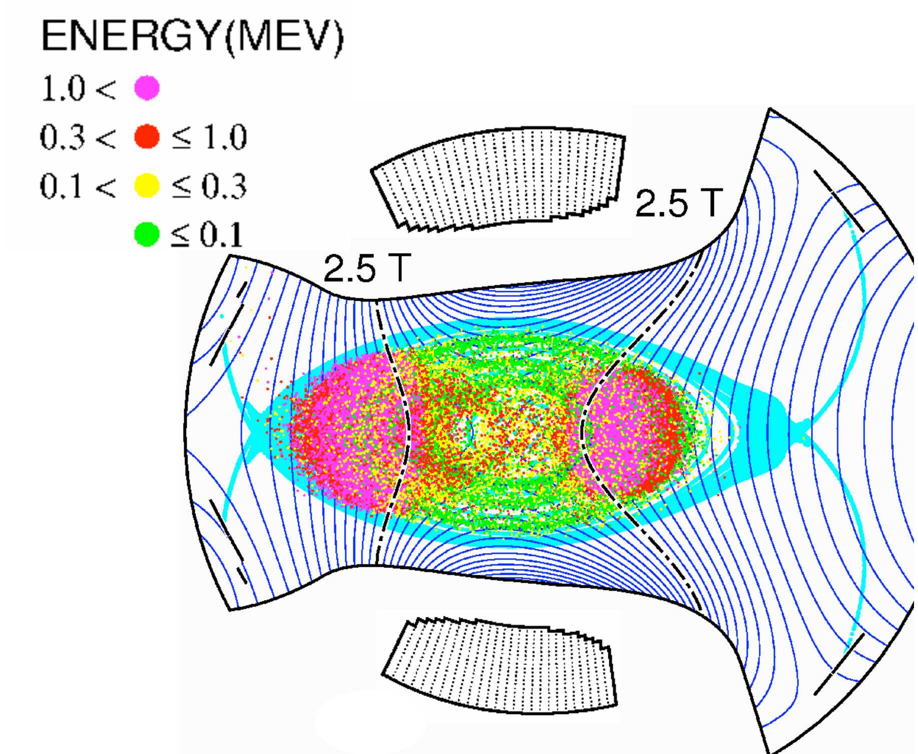
Some people worry about the α-particle loss , but…

- ***The loss of high energy ions due to trapping in helical ripples can be reduced by optimizing magnetic field configuration.***
  - *An evidence is formation of high energy ion tail in minority heating experiment of ICH.*
  - *Another evidence is that the perpendicular NBI heating is successful*



54 min. Operation (#66053)

NPA

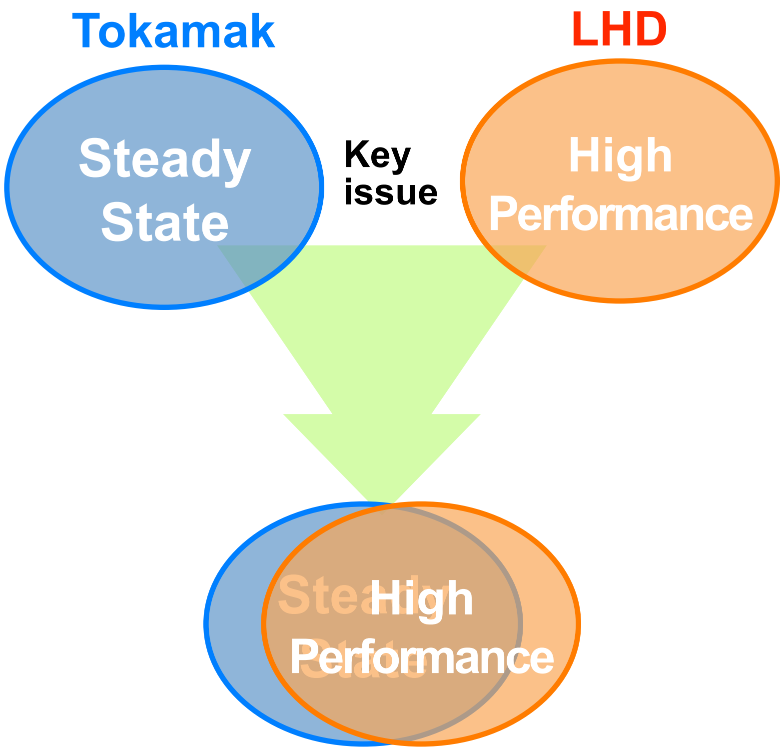


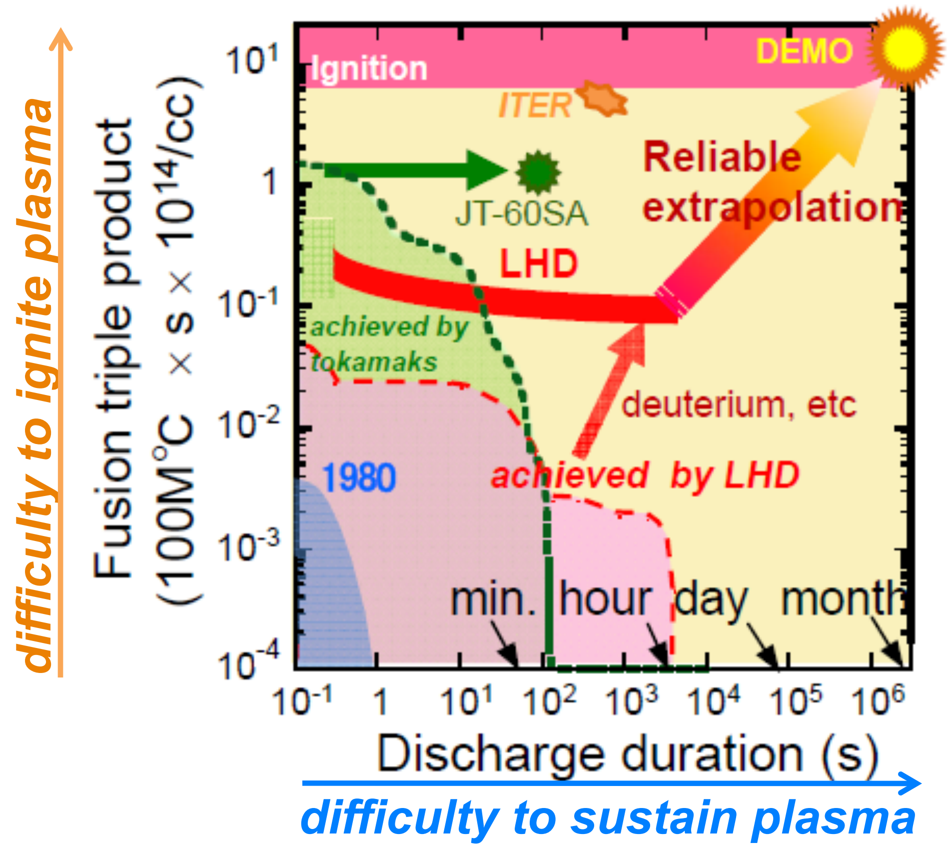
Particle Orbit Calculation

LHD has shown its favorable characteristics as a reactor

– why don't we think about helical DEMO ?

Strategies to design DEMO (tokamak and LHD)





Keys to DEMO

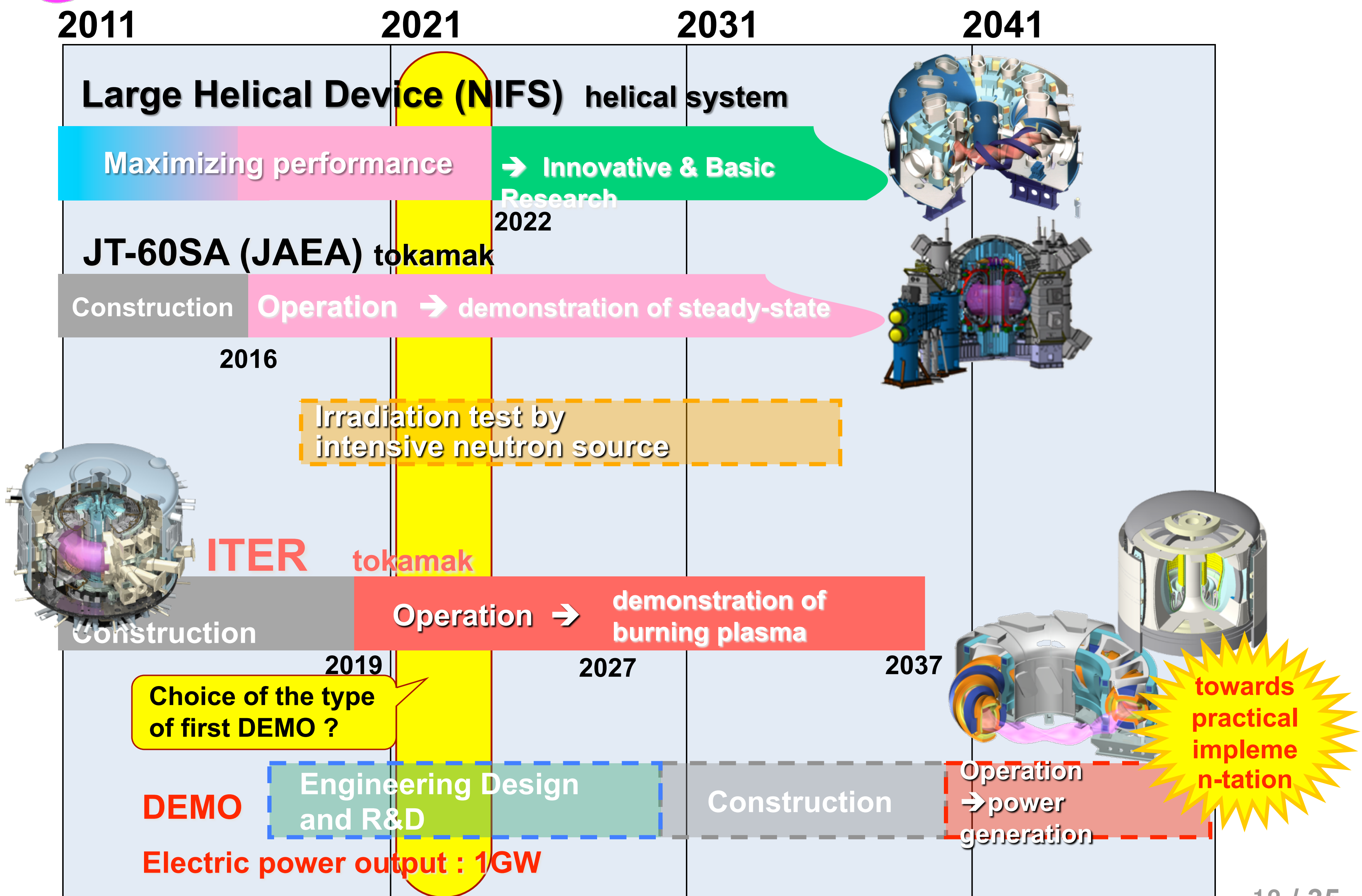
(1) Demonstration and control of burning plasma ➔ ITER

(2) Steady state operation ➔ LHD : high performance plasma to convince us of burning

➔ JT-60SA : non-inductive current drive with minimizing circulation power

Note: Japanese fusion research is shared by
NIFS (academic, integration of commitment from universities) and
JAEA (programmatic under governmental decision)

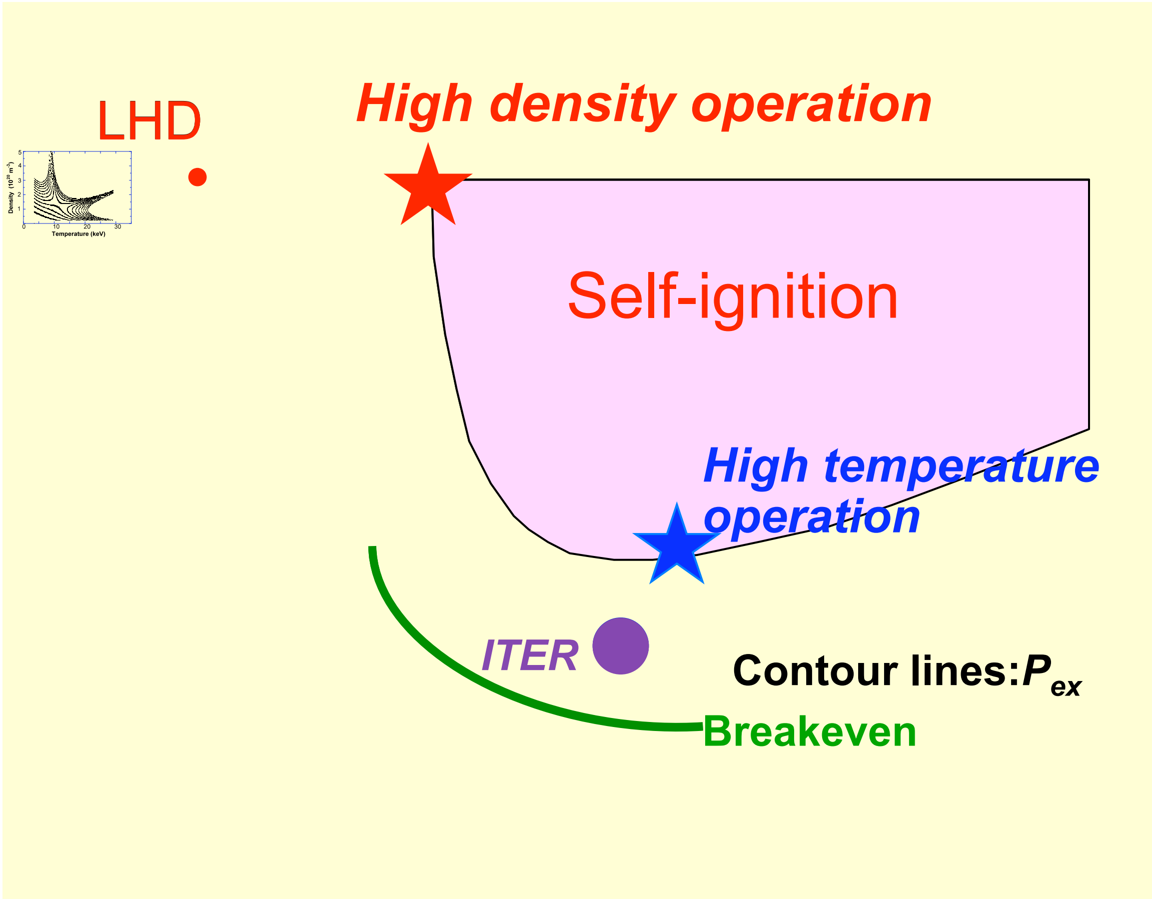
A road map to DEMO (tokamak and LHD)



Strategy towards helical DEMO (NIFS)

- Design is based on the experimental database
  - So far only from LHD for the plasma performance
  - Results of W7-X will be welcome for optimization
- "Helical ITER" can be skipped using numerical simulation
  - Integration of phenomena in various space and time scale is necessary and is being carried out in 3D geometry
  - Burning physics from ITER results is essential
- Engineering capability of some specific components for helical DEMO should be studied by ourselves
  - Large helical coils, unique shaped blanket, and so on
- These three activities; LHD, Numerical Test Reactor, Fusion Engineering, are enforced recently in NIFS

Super Dense Core Reactor
– a unique option of helical reactor -



Note: Two operation scenarios exit in LHD type reactor

- Novel scenario to super-dense-core reactor (SDCR)
    - Ignition at lower temperature (∼7 keV) and high density (> 6 $\times 10^{20} m^{-3}$)
    - Thermally unstable but can be controlled
- SDC scenario reduced engineering demand and neoclassical ripple transport

Current Activities in NIFS

- three projects have been re-organized
to promote studies towards helical demo –

1. LHD Project

- to make reactor relevant plasma

LHD, the worlds' largest fully superconducting fusion experimental device, has worked very well for 13 years

< LHD basic dimension >

- Outer diameter 13.5 m
- Cold mass 820 ton
- Total weight 1500 ton
- Magnetic field 3 T
- Magnetic energy 0.77 GJ

Several-month-long operation,
14 times since 1998

- Operational time of He compressor:
  63,751 hours ➔ Duty = 99.1 %
- Coil excitation number: 1,394 times
- Plasma discharges: 104,824 shots

The 100,000th shot on Nov.10th, 2010

Targets of LHD in FY2010 campaign and the results

Targets of FY2010

1. Get high electron and ion temperature by increase in heating power (ECH, NBI ICH)
2. Study the impurity ion transport in helical plasma
3. Study of Super Dense Core plasma
4. Confirm the effects of structure of divertor plates to design the closed divertor system

Achievements in FY2010 [Final Goal]

Central ion temperature
6.5 keV [10 keV]
@ $1.6\times10^{19}m^{-3}$ [$2\times10^{19}m^{-3}$]

Central electron temperature
15 keV [10 keV]
@ $3\times10^{18}m^{-3}$ [$2\times10^{19}m^{-3}$]

Central density
$1.2\times10^{21}m^{-3}$ [$4\times10^{20}m^{-3}$]
@ 0.3keV

Volume averaged beta
5.1 % (0.425T) [>5 % (1-2T)]

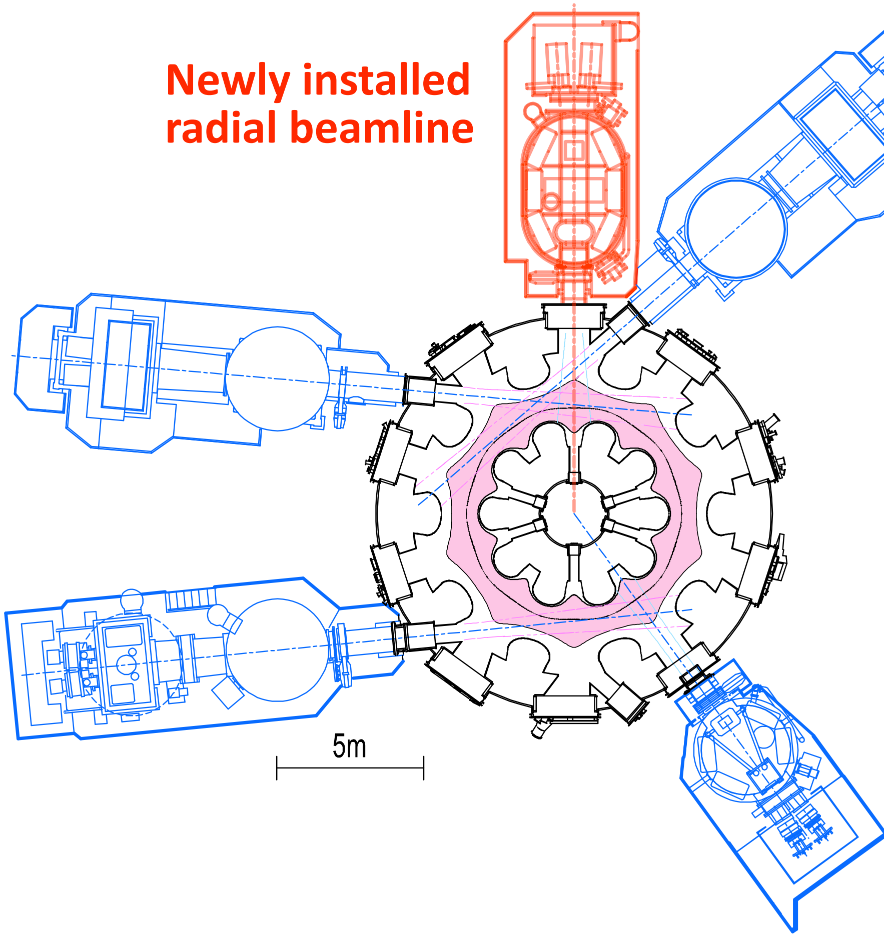
Steady state operation
54 min. 28 sec. (500kW)
31 min. 45 sec. (700kW)
13 min. 20 sec. (1 MW)
[1 hour (3 MW)]

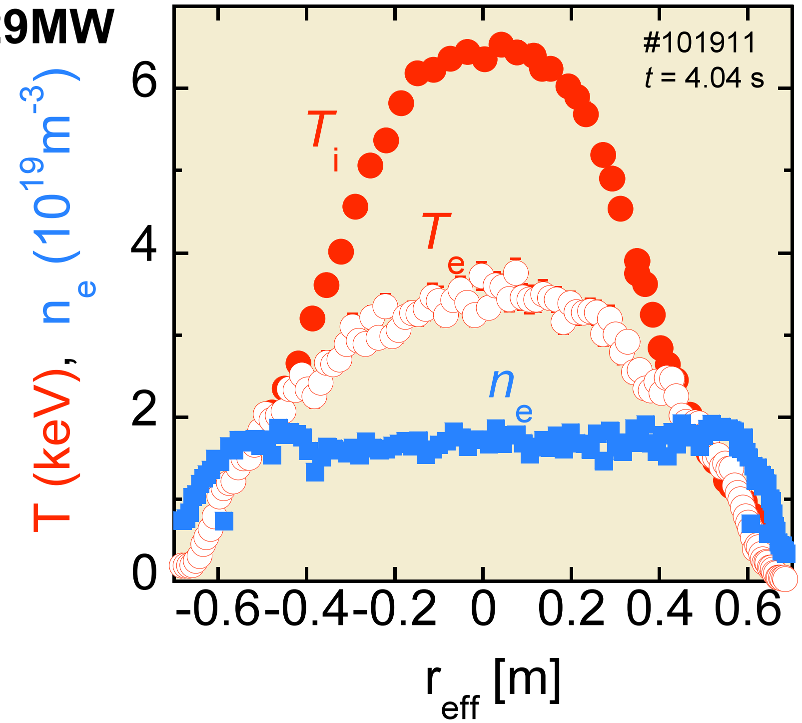
Confinement energetic particles
Ion tail up to 1.6 MeV

1. High ion temperature was obtained with detailed spatial profile

- **Radial NB Injector of 6MW (60keV H^0) was installed**

➔ Total NB power increased from 23MW to 29MW





(1) Max. T_{i0} = 6.5 keV

(2) Obtained plasma with T_{i0} = T_{e0} = 5 keV

- **Detailed temperature profile can be obtained due to the fast sampling of 2 kHz in Charge Recombination Spectroscopy measurement CXRS(T_i, V, n_{imp})**

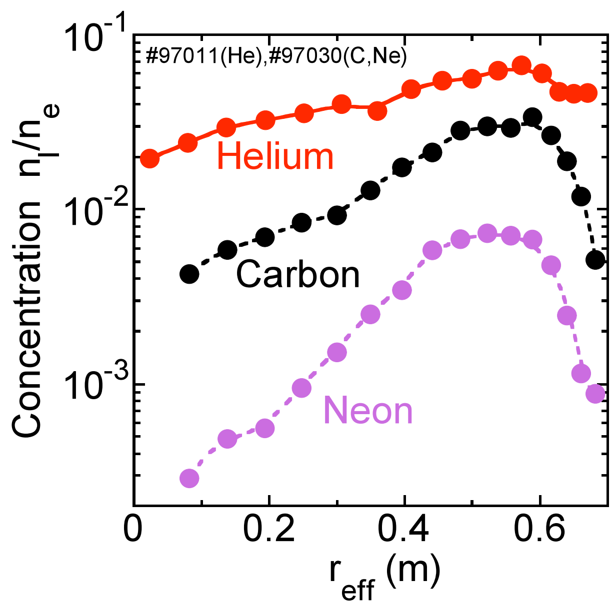
2. Characterization of *Impurity Hole* has progressed

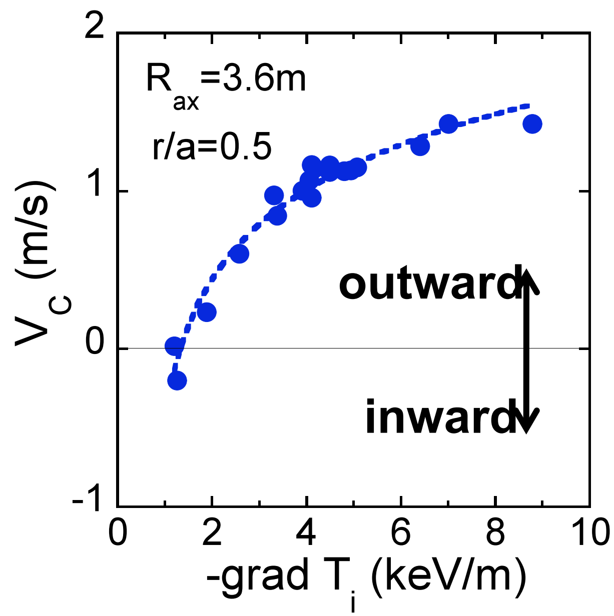
Extremely hollow impurity profiles observed in the plasma with a steep gradient of ion temperature are referred to *Impurity Hole*

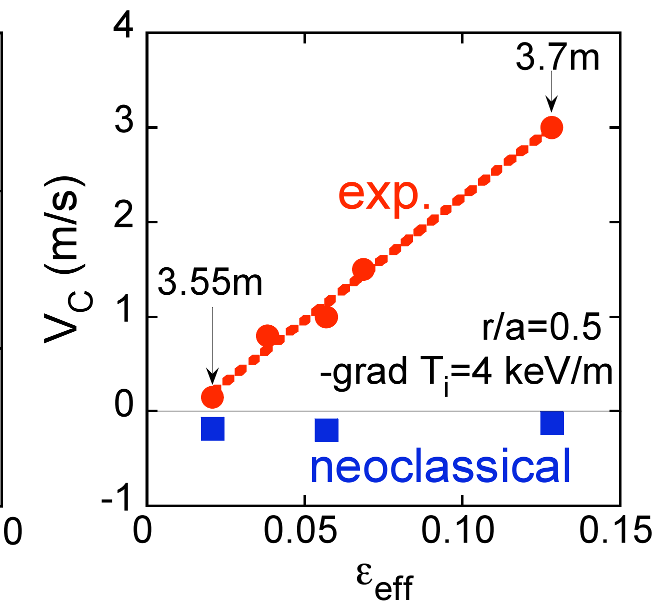
Key: Outward convection in spite of negative E_r, which is not predicted by neoclassical theory

Outward convection, which causes impurity hole, is enhanced more with

Mass of impurity | **Ion temperature gradient** | **Effective helical ripple**



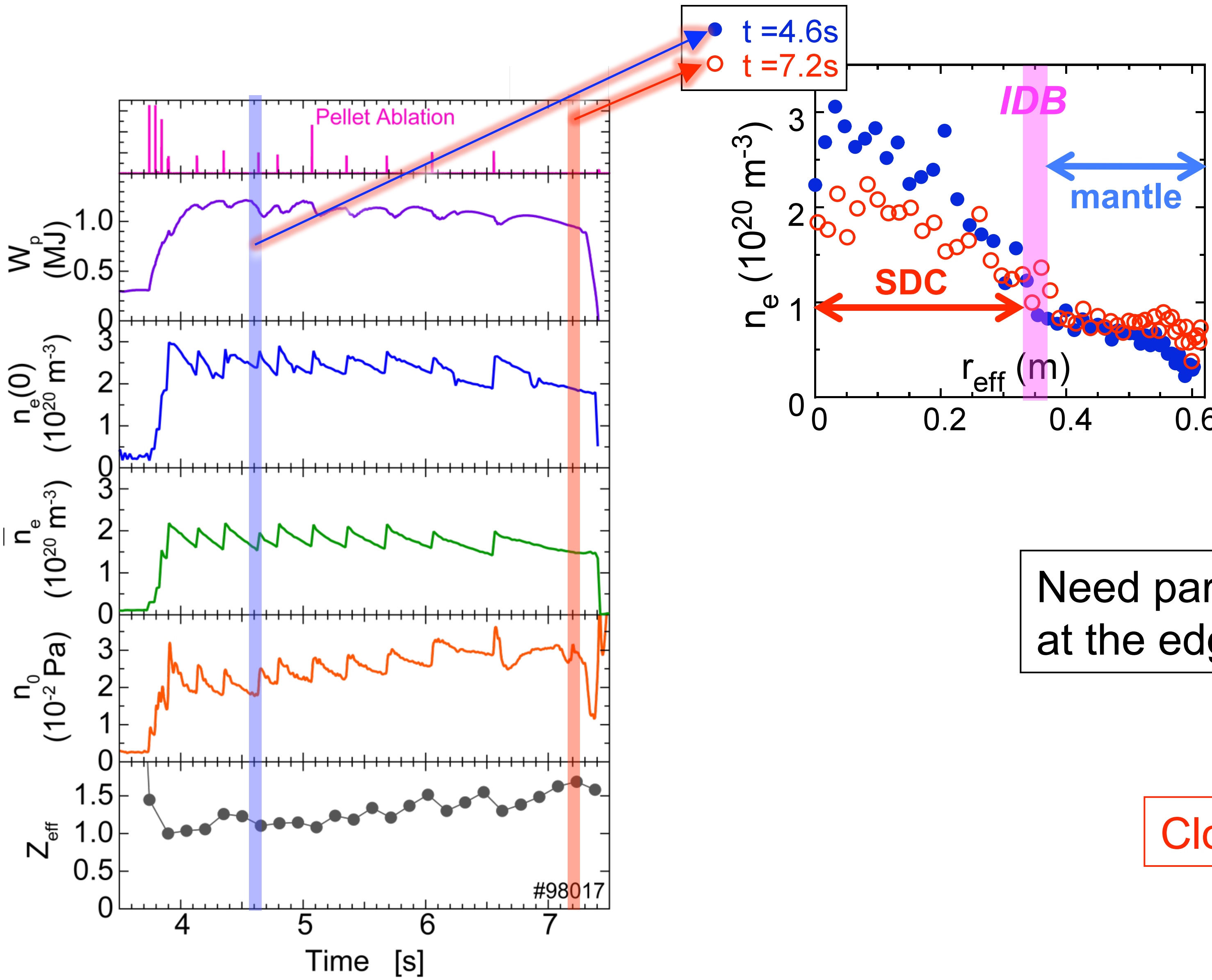




Impurity hole is driven by anomalous convection : off-diagonal transport

3. *Super-Dense Core* (SDC) due to *Internal Diffusion Barrier* (IDB) was sustained over 3 s

SDC plasma is sustained by feed-back controlled repetitive pellet injection



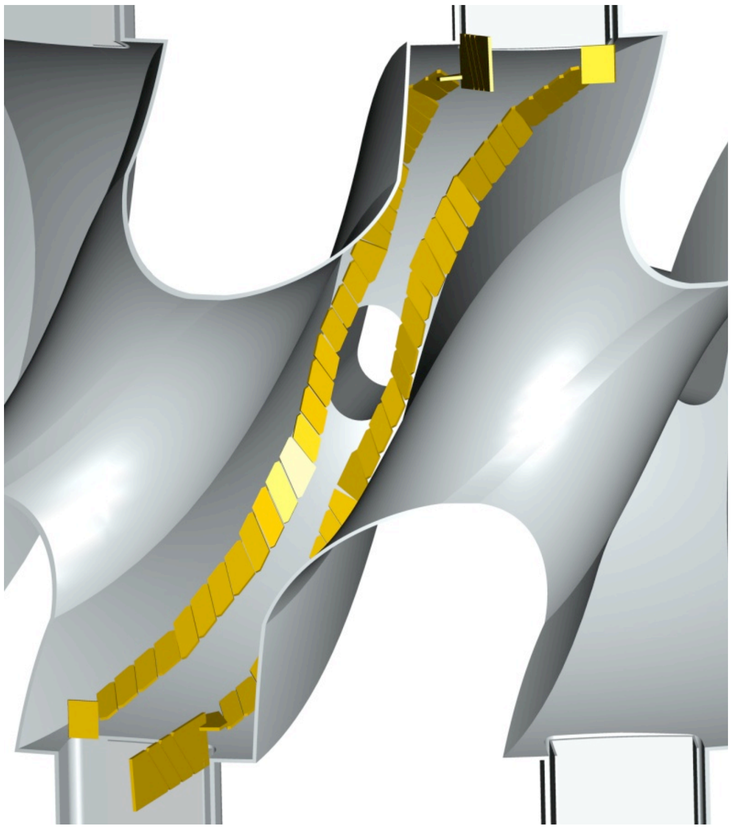
Discharge degrades Gradually

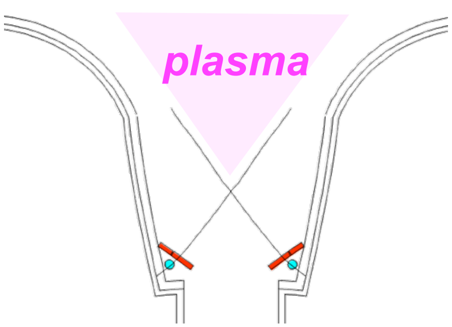
← enhancement of recycling

Need particle control at the edge of plasma

Closed divertor

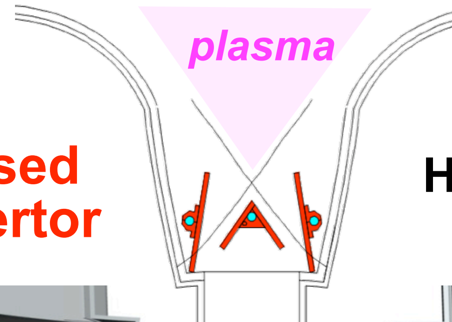
4. Partial Modification of Divertor Configuration towards Closed Divertor



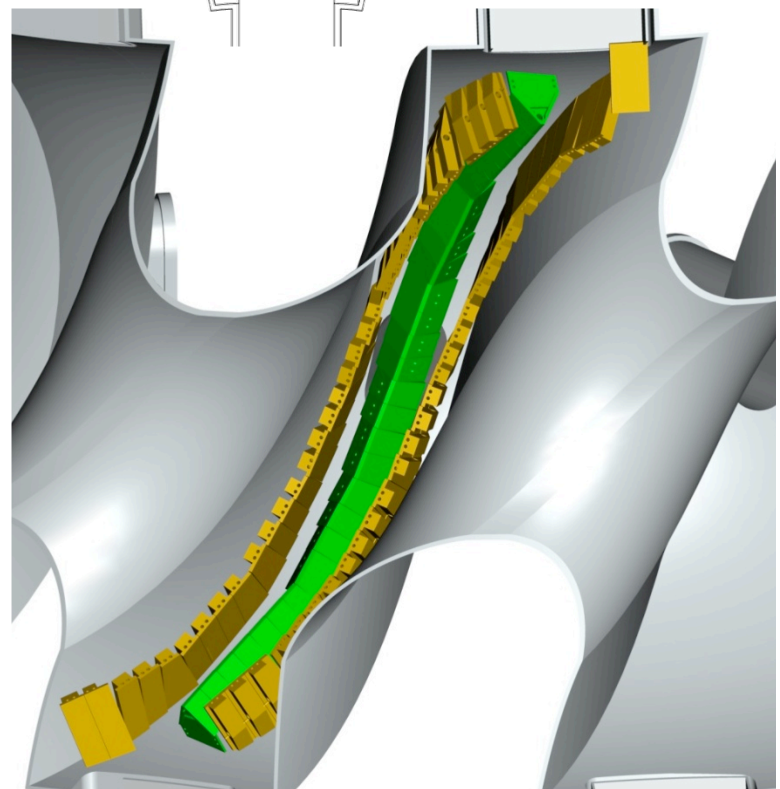


Open divertor

Closed divertor



Heat capability 1.5 MW/m^2



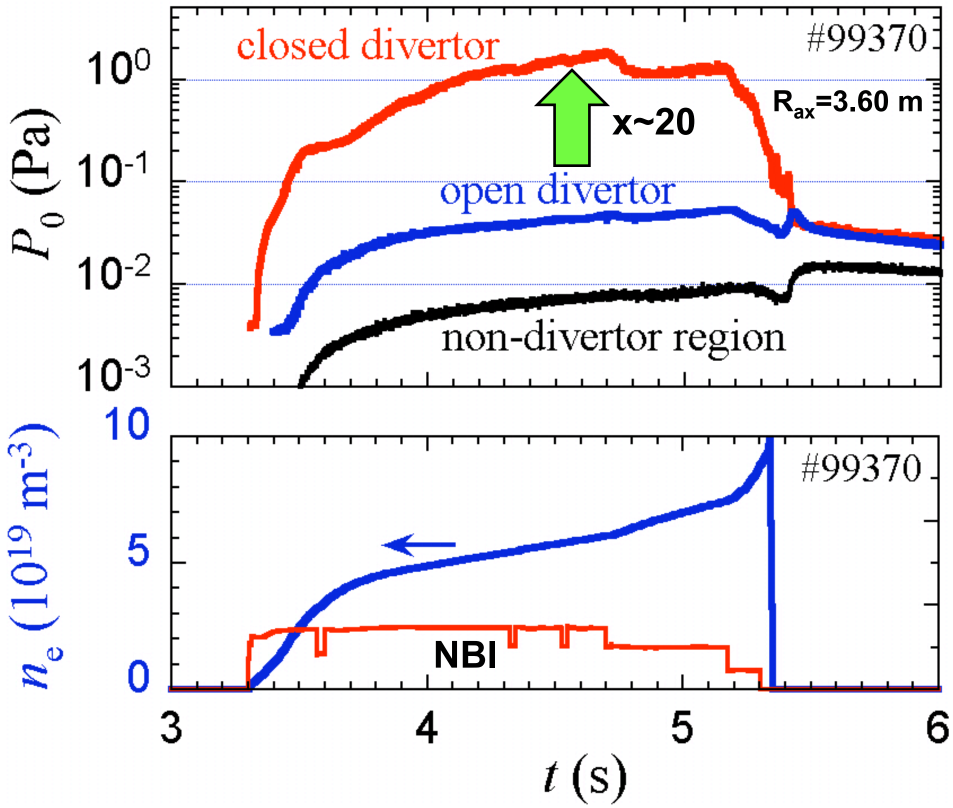
- Before the campaign, 2 toroidal sections of inboard side are modified
- Note: vacuum pump is not installed yet.
  ➔ Confirmed the capability of neutral compression

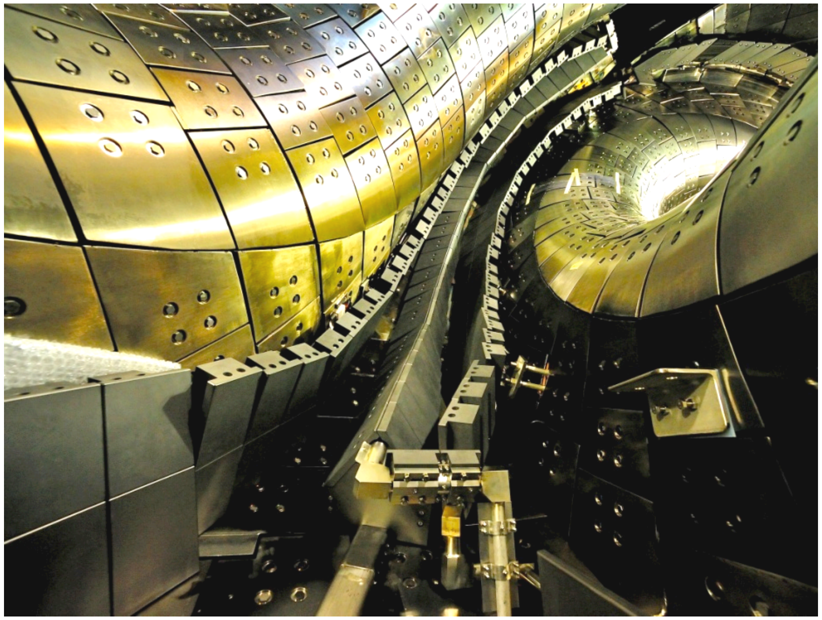
4. Demonstration of Neutral Compression by Closed Divertor Configuration

Sufficient neutral compression has been confirmed as expected

- more than **1 Pa** in closed divertor section
- about **20 times higher pressure** in closed divertor, compared to open divertor
- more than 100 times, compared to non-divertor region

➔ Design has been validated





Inboard side of 2 toroidal sections among 10 periods have been modified

Future direction of LHD for higher performance

- **Closed divertor with cryopump will be fully installed in inboard side after the next 15th experimental campaign**

- **Prepare deuterium plasma operation**
  - Experiment may start in FY2013 hopefully

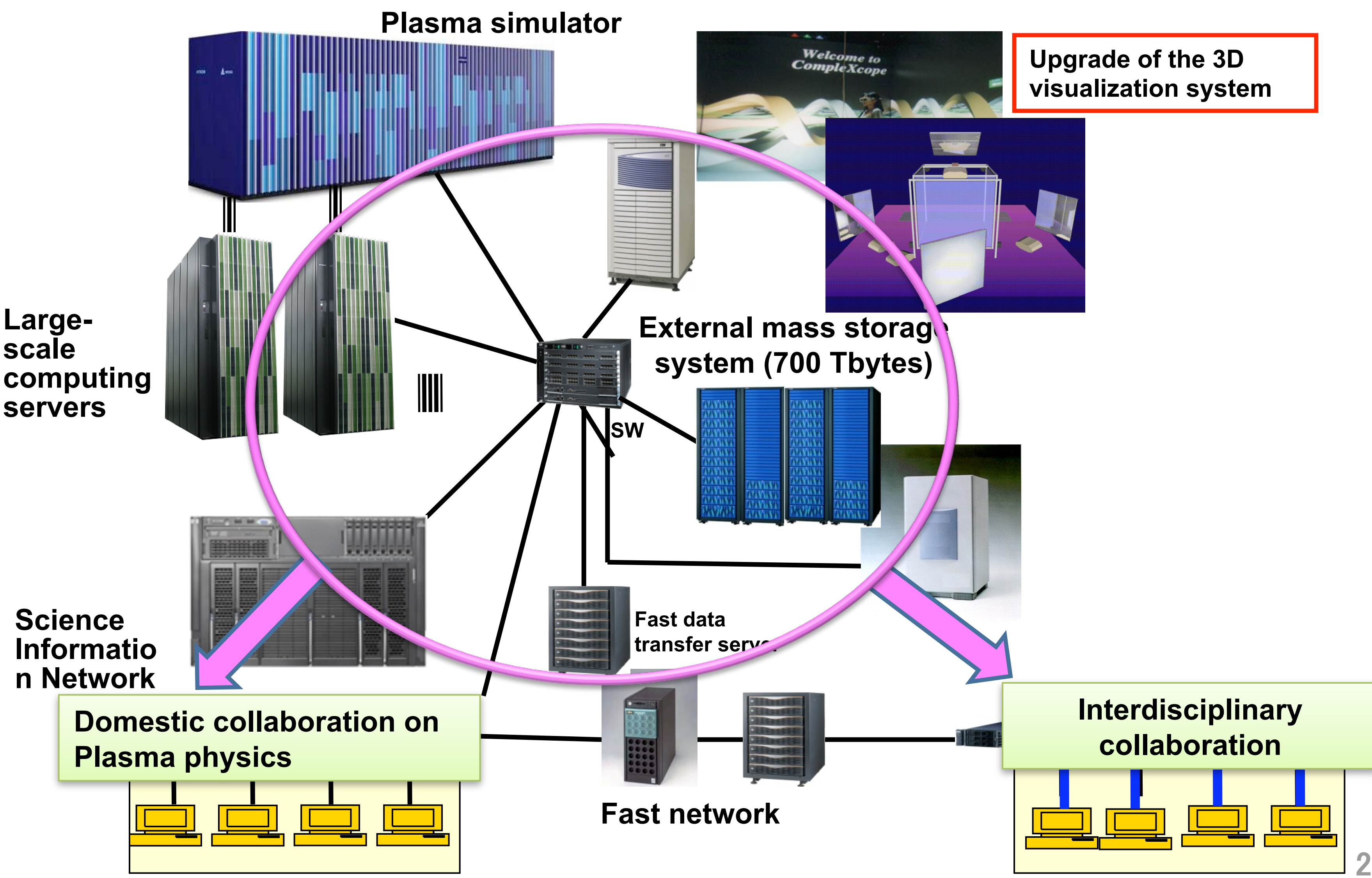
- **Increase in heating power**
  - **Perp.NBI (D beam with doubled energy)**
  - **ECH**
  - **ICH**

Current Activities in NIFS

2. Numerical experiment research Project

- theoretical forecast of fusion plasma behavior from edge to core

NIFS has the largest human and infrastructural resources for plasma numerical simulation in Japan

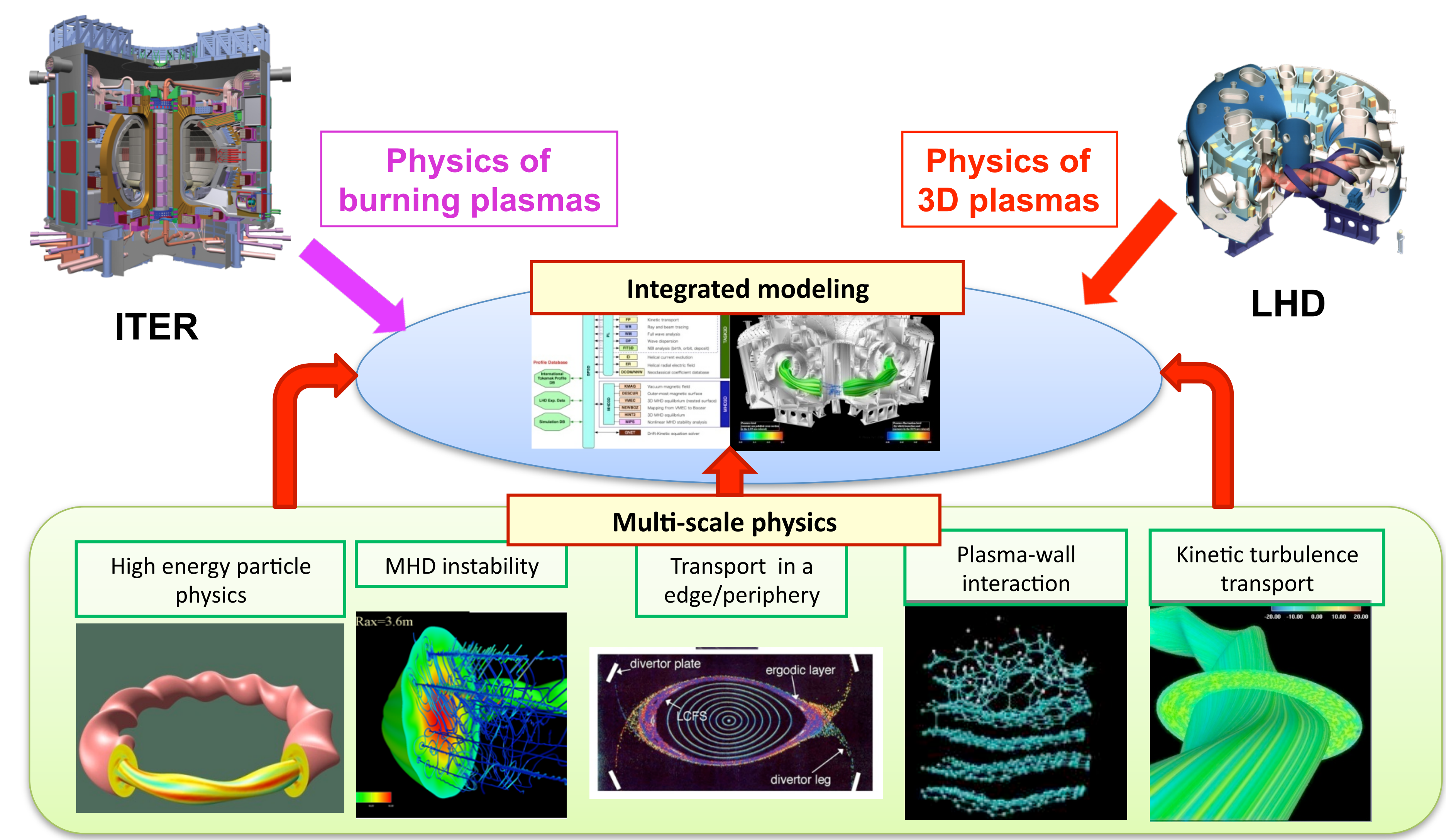


Project consists of nine research groups: - from element physics to their integration

- Equilibrium and stability
  Multiscale MHD simulation of beta-increasing LHD plasma, Pellet injection
- High energy particle physics
  Hybrid simulation for MHD and energetic particles, TAE
- Fluid turbulence
  Interaction between a macro-MHD mode, micro-turbulence and zonal flow
  Simulation of drift wave turbulence, Radial electric fields in LHD
- Kinetic transport
  Gyrokinetic simulation of ITG turbulence and ZF in LHD, ETG turbulence
  Monte Carlo simulation of neoclassical transport in LHD and stochastic fields
- Periphery plasmas
  Impurity transport in SOL/divertor plasma and redeposition near PFW in LHD
- Plasma-wall interaction
  Chemical reactions between H and C, Mechanisms of yielding hydrocarbon
- Integrated transport code
  Modules for neoclassical diffusion and bootstrap current, Application to LHD
- Multiscale physics
  Success of Micro-Macro multi-hierarchy magnetic reconnection simulation
- Simulation methodology
  Simultaneous visualization of simulation and experimental device data in VR

Collaborators from the universities also join each group

Integration of elementary researches to construct numerical test reactor



Current Activities in NIFS

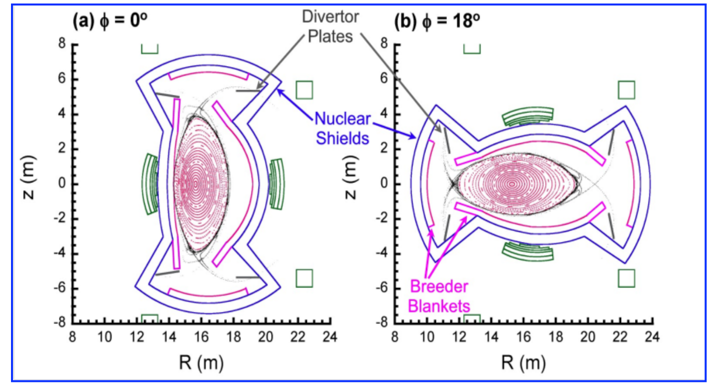
3. Fusion engineering research Project

- make technical data base to design reliable DEMO structure

NIFS has been carried out reactor study

Design parameters			LHD	FFHR2	FFHR2m1
Polarity	l		2	2	2
Field periods	m		10	10	10
Coil pitch parameter	γ		1.25	1.15	1.15
Coil major Radius	R_c	m	3.9	10	14.0
Coil minor radius	a_c	m	0.98	2.3	3.22
Plasma major radius	R_p	m	3.75	10	14.0
Plasma radius	$\langle a_p \rangle$	m	0.61	1.24	1.73
Plasma volume	Vp	m^3	30	303	827
Blanket space	Δ	m	0.12	0.7	1.1
Magnetic field	B_0	T	4	10	6.18
Max. field on coils	B_{max}	T	9.2	14.8	13.3
Coil current density	j	MA/m^2	53	25	26.6
Magnetic energy		GJ	1.64	147	133
Fusion power	P_F	GW		1	1.9
Neutron wall load	Γ_n	MW/m^2		1.5	1.5
External heating power	P_{ext}	MW		70	80
α heating efficiency	η_α			0.7	0.9
Density lim.improvement				1	1.5
H factor of ISS95				2.40	1.92
Confinement time	τ_E	s			
He ash confinement time ratio	τ_α^*/τ_E				
Effective ion charge	Z_{eff}			1.40	1.34
Electron density	$n_e(0)$	$10^{19}\ m^{-3}$		27.4	26.7
Temperature	$T_i(0)$	keV		21	15.8
Plasma beta	$\langle\beta\rangle$	%		1.6	3.0
Radiation loss	P_r	MW			
Plasma conduction loss	P_L	MW			290
Heat flux to first wall	P_W	MW/m^2			
Diverter heat load (1m wet width)	Γ_{div}	MW/m^2			1.6
Total capital cost		G$(2003)		4.6	5.6
COE		mill/kWh		155	106

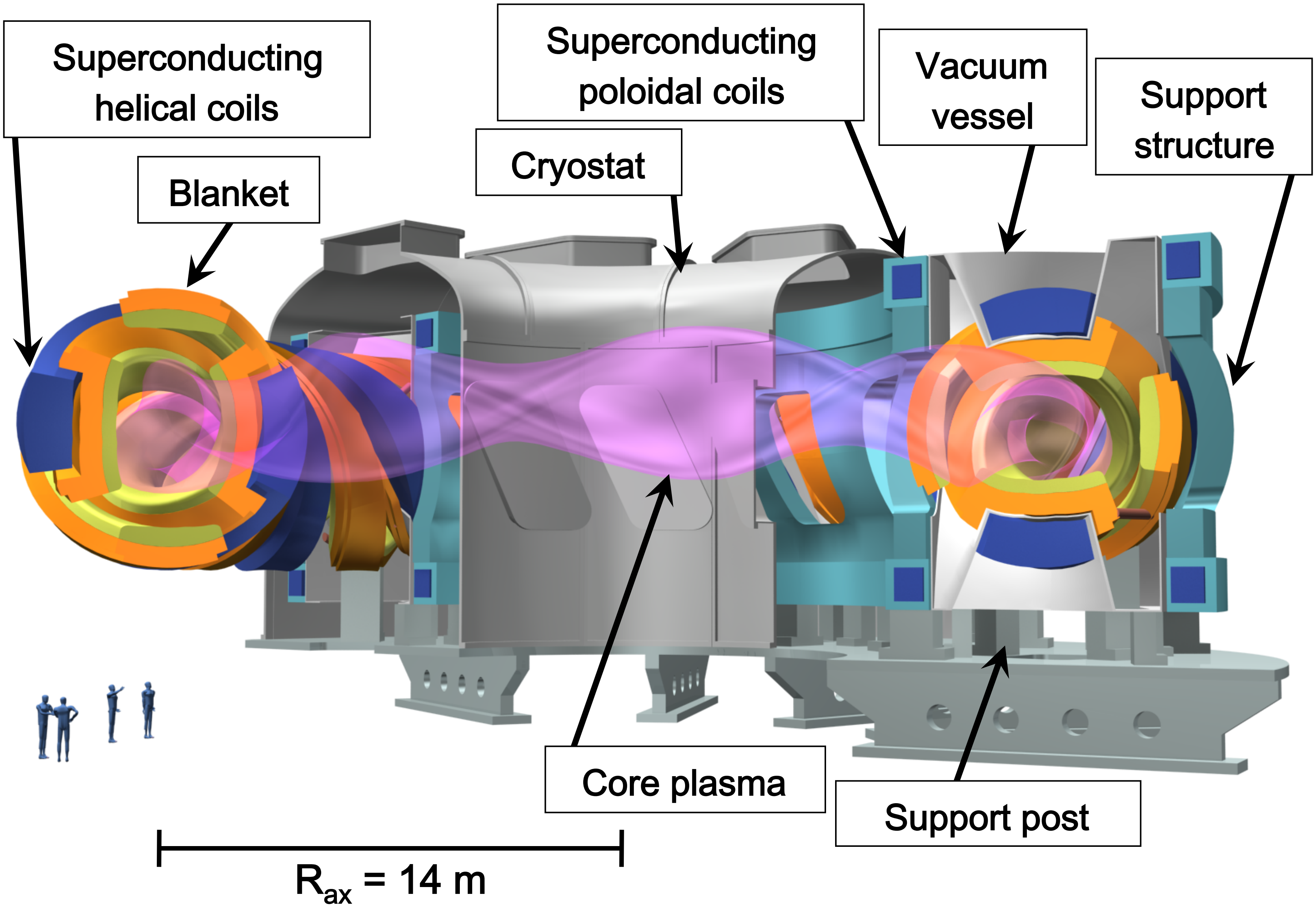
FFHR



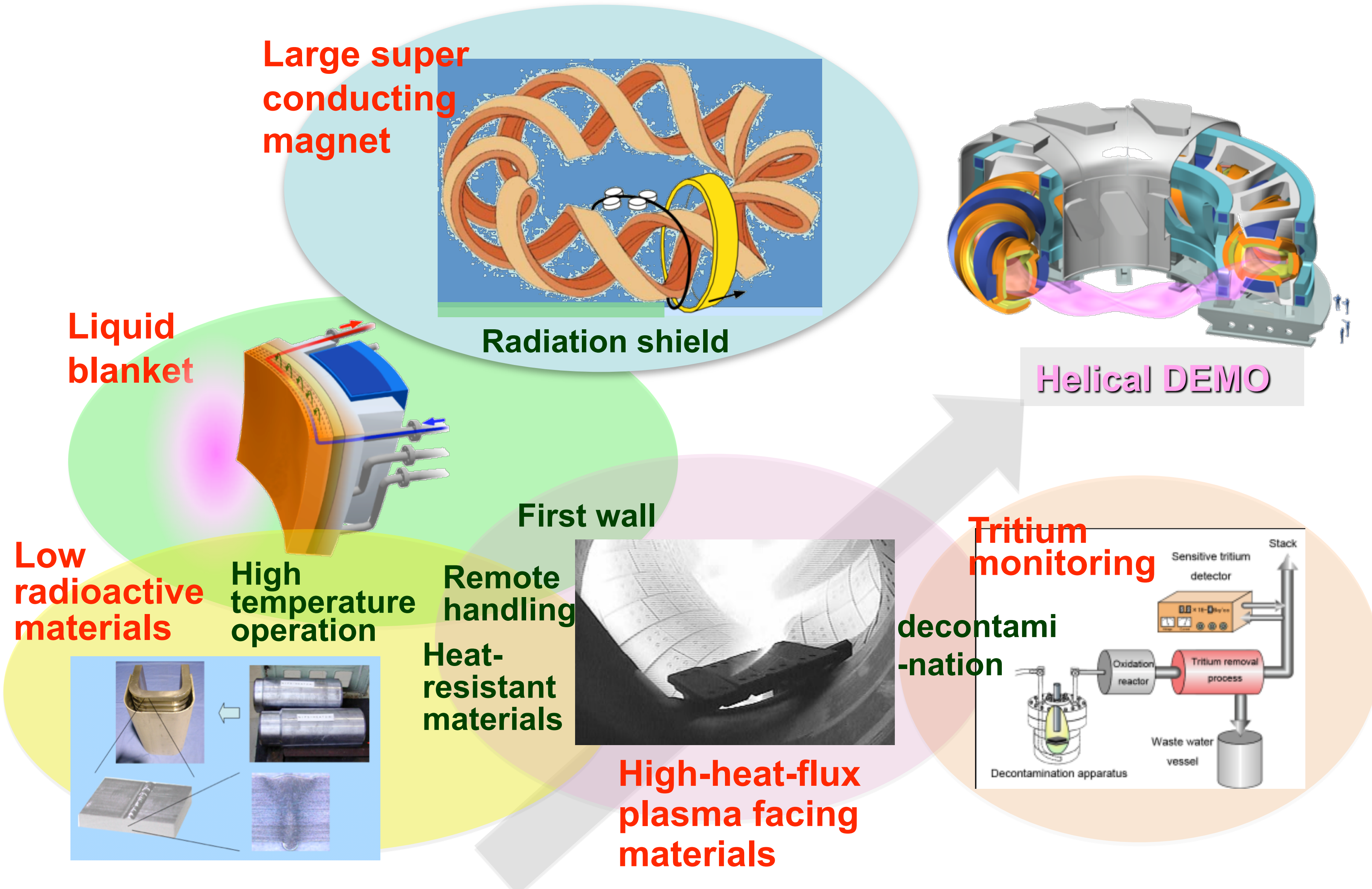
Latest design "FFHR-2m2 Type-D" with maximized magnetic surfaces by optimization of poloidal coil positions and currents. R_c = 17 m, R_p = 15.7 m, $\langle a_p \rangle$=2.5m, g = 1.2, a = +0.1, $B_{t,c}$ = 4.45 T, j_{HC} =25 A/mm^2, W_{mag}=160Gj. <b>=5.6% by Finite-beta equilibrium analysis using VMEC code

In SDC reactor, divertor heat load is drastically reduced (1/3~1/4).

Schematic view of FFHR



Engineering issues to be solved for designing helical DEMO

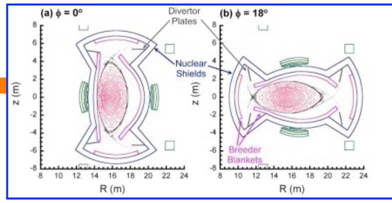


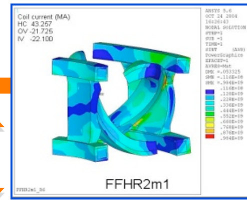
Fusion Eng. Research Project has started towards steady-state helical DEMO

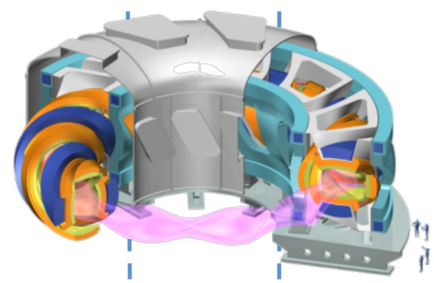
2010 | 2016 | 2022 | 2027 | 2036FY

Step by step advancement of reactor design

Conceptual design → Basic design → Improved basic design





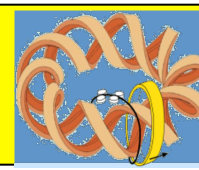


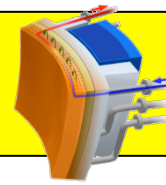
Establishment of engineering base

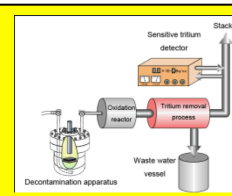
Full-scale, full-condition testing

Collaborate with universities

- Large-scale high-field superconducting magnet
- Long-life liquid blanket
- Low activation structural materials
- High heat flux plasma facing wall
- Tritium control







Engineering design

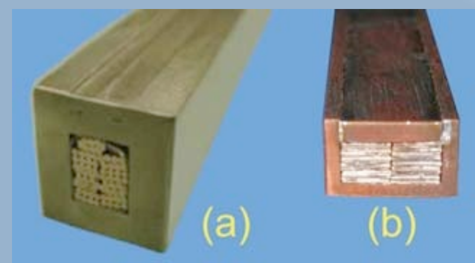
Construction

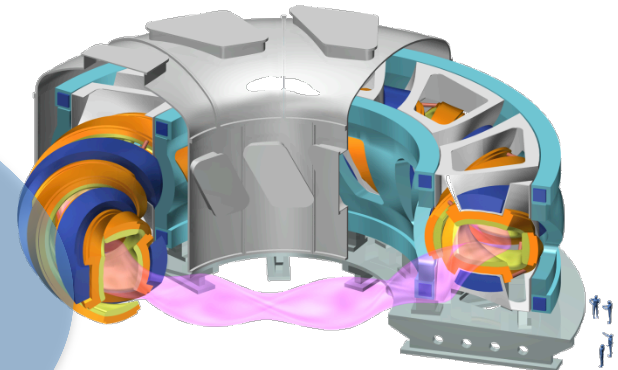
Licensing

Operation

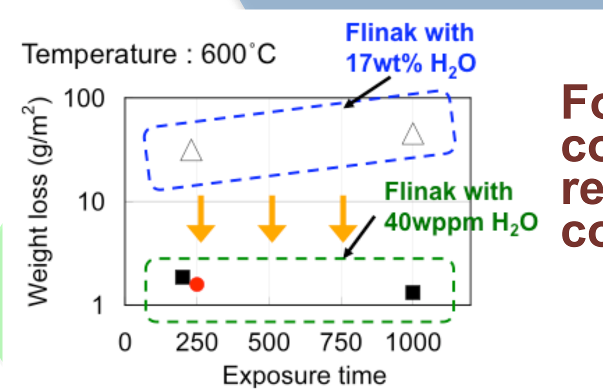
Basic research has already started in collaboration research

Developments of
(a) Nb_3Sn superconductor with Al-alloy jacket,
(b) 10 kA-class high-temp. superconductor.





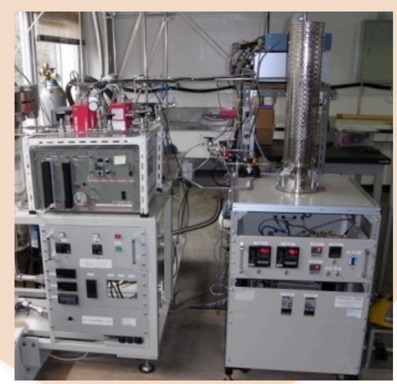
Steady-state Reactor design, 3 GWth, 30,000 ton



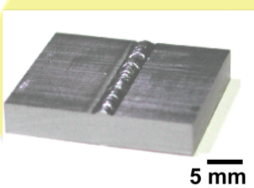
For liquid blanket, corrosion can be reduced by impurity control.

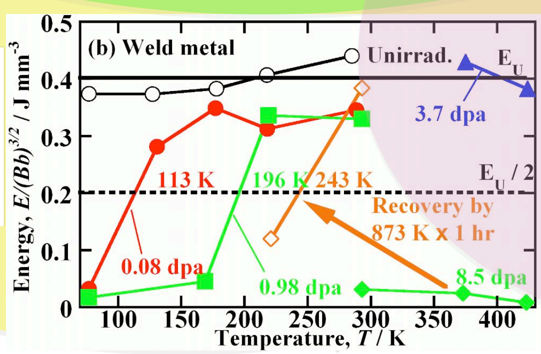
Basements for full-scale or full-condition testing

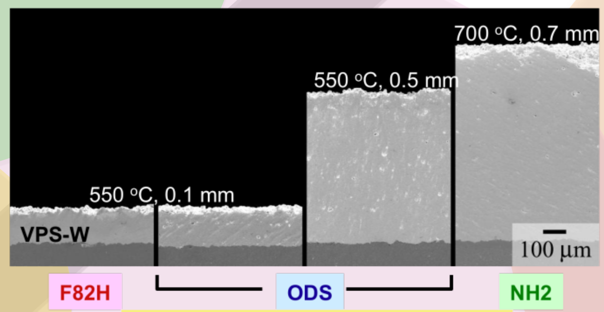
High-sensitive tritium Monitor (with Nagoya Univ.)



Welded V-alloy can recover from neutron damage by heating.







Cross section of VPS-W coating

For plasma facing wall, W coating was succeeded by plasma spray.

Summary

- Since 1989, NIFS has been pursuing the research on fusion science and technology by experimental research and technological development on LHD device, large-scale numerical simulation, and basic fusion engineering R&D's through collaboration with Japanese universities.
- Among them LHD has verified intrinsic ability of Heliotron as fusion reactor.
  - ➢The advantages are mainly in plasma characteristics, that is, disruption-free, stable steady-state, high-density, high-beta and small impurity contents.
  - ➢On the other hand, difficulties lies in engineering issues which come from large-scale three-dimensional structure. SDC scenario may mitigate some conditions
- Activities in NIFS have addressed towards realization of helical DEMO more clearly now.
  - ➢Newly re-organized three project: LHD, numerical experiment and fusion engineering, will cooperate one another and also with universities to make design basis of helical DEMO in next decade.