# Current status of helical fusion reactor design and study on operation control scenario


National Institute for Fusion Science

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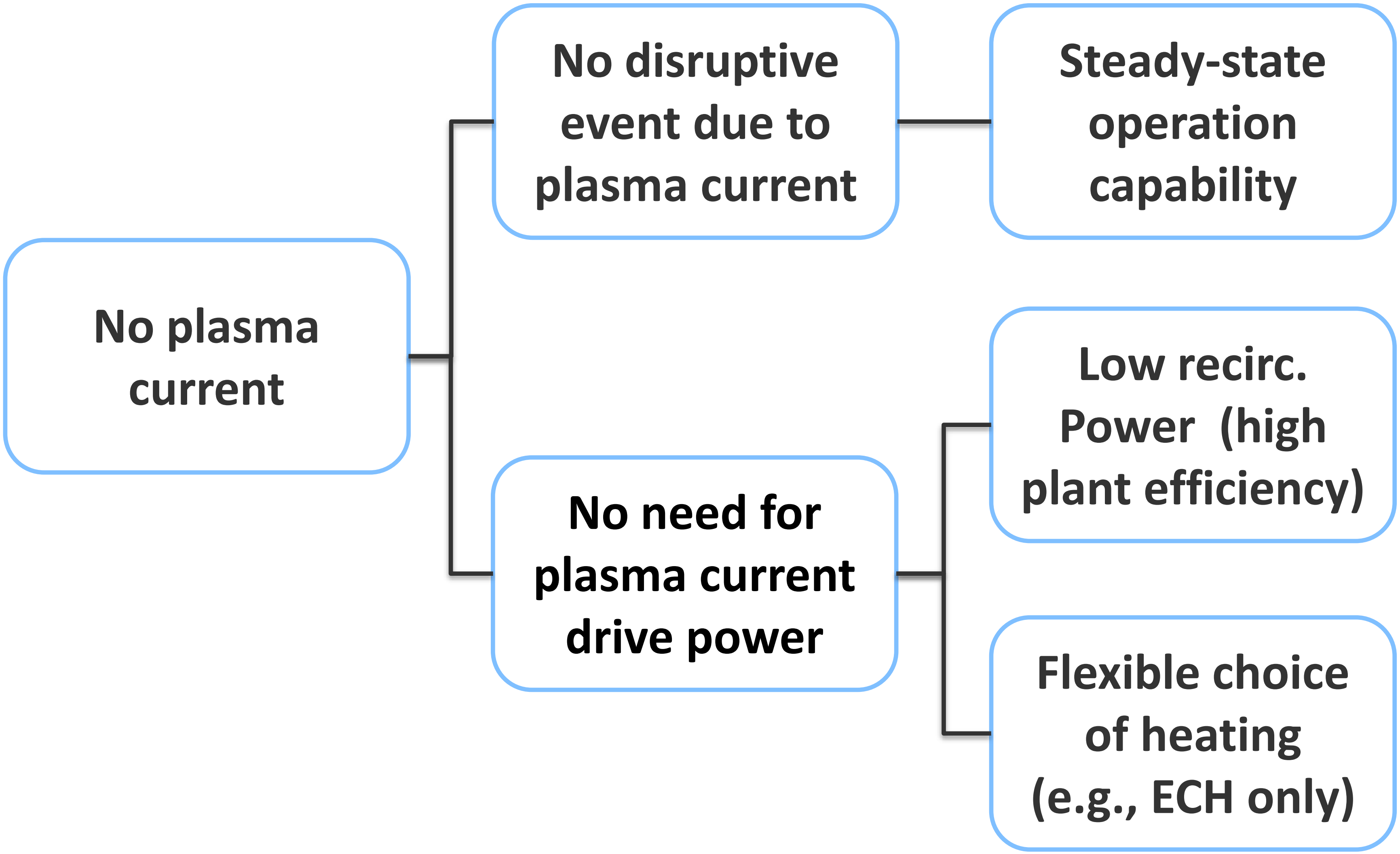
# Advantages of helical reactor (heliotron/stellarator)



# Advantages of heliotron-type fusion reactor

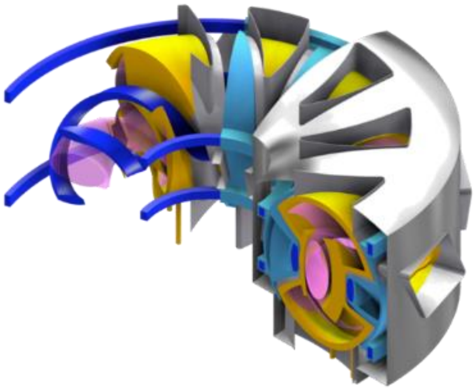
- LHD-type helical fusion reactor has several advantages in addition to steady-state operation capability

**Highly reliable core plasma design**

Plenty of LHD exp. data & numerical tools verified by LHD exp.

**Robust divertor field structure**

Flexible design/placement of divertor components

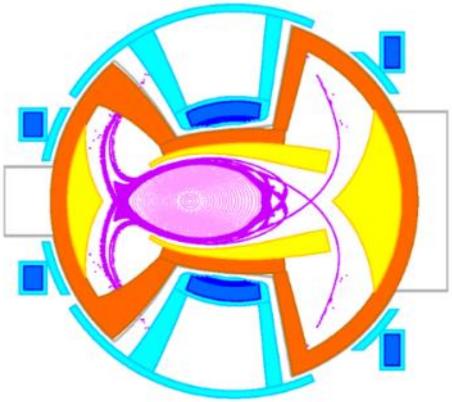


**Coil with a small curvature variation**

Relatively easy coil fabrication

**Large aperture b/w coils**

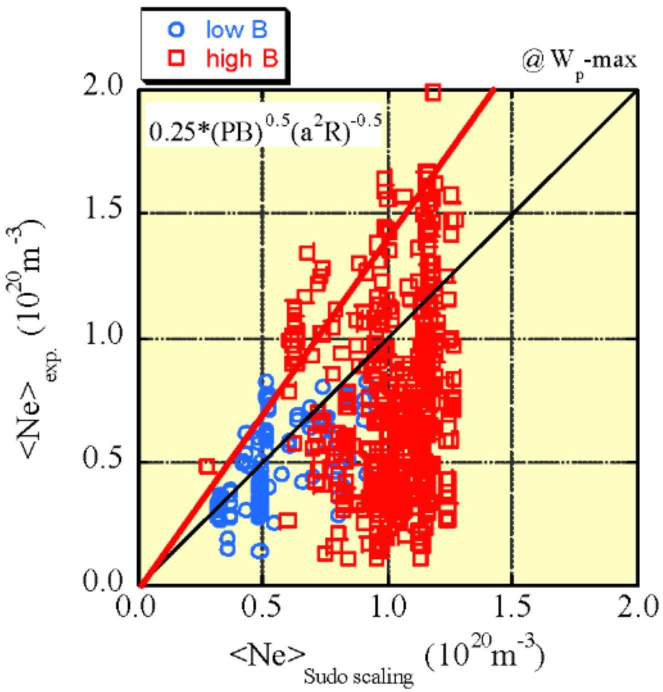
Flexible maintenance of in-vessel components



# Two major transients in heliotron/stellarator devices

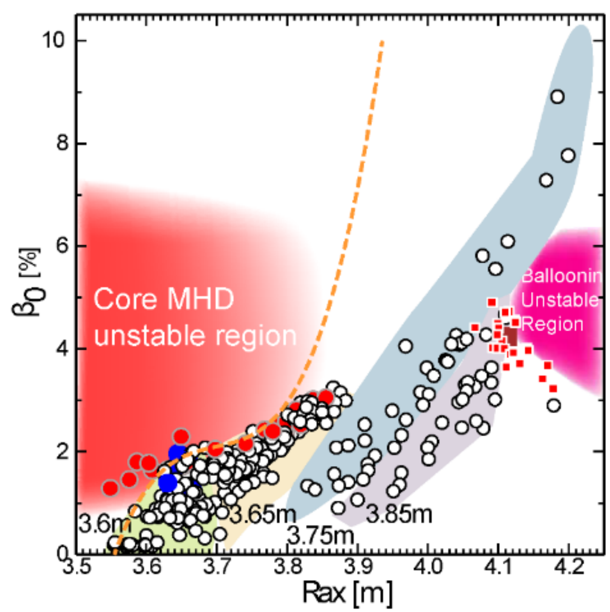
- **Density limit**
  - Radiating collapse
  - Sudo(-like) scaling

$$n_{\mathrm{Sudo}} \propto \sqrt{PB/a^2R}$$
$$n^* \propto P^{0.6}/f_{\mathrm{imp}}^{0.4}$$



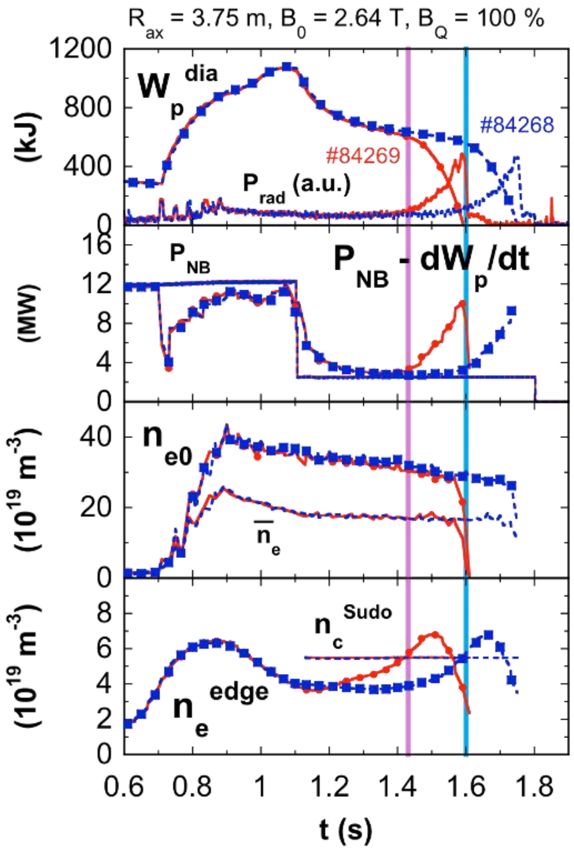

B.J. Peterson, Proc. IAEA-FEC20 EX/6-2.

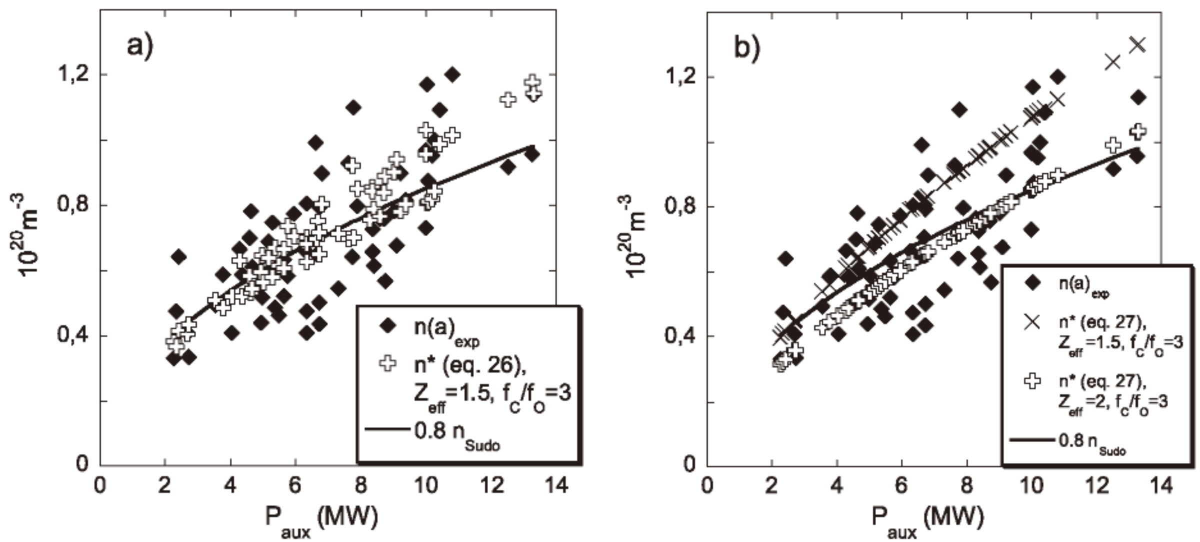
- **Beta limit**
  - Equilibrium limit
    - ✓By Shafranov shift and stochastisation
  - Stability limit
    - ✓By low-n MHD mode




By S. Ohdachi

# New interpretation of density limit




$$n^* \propto P_{\mathrm{aux}}^{0.57} B^{0.33} R^{-0.54} a^{-0.72} \iota_{2/3}^{0.16} f_{\mathrm{imp}}^{-0.4}$$

$$n^* \propto P_{\mathrm{aux}}^{2/3} B^{0.4} R^{-2/3} a^{-0.5} f_{\mathrm{imp}}^{-1/3}$$

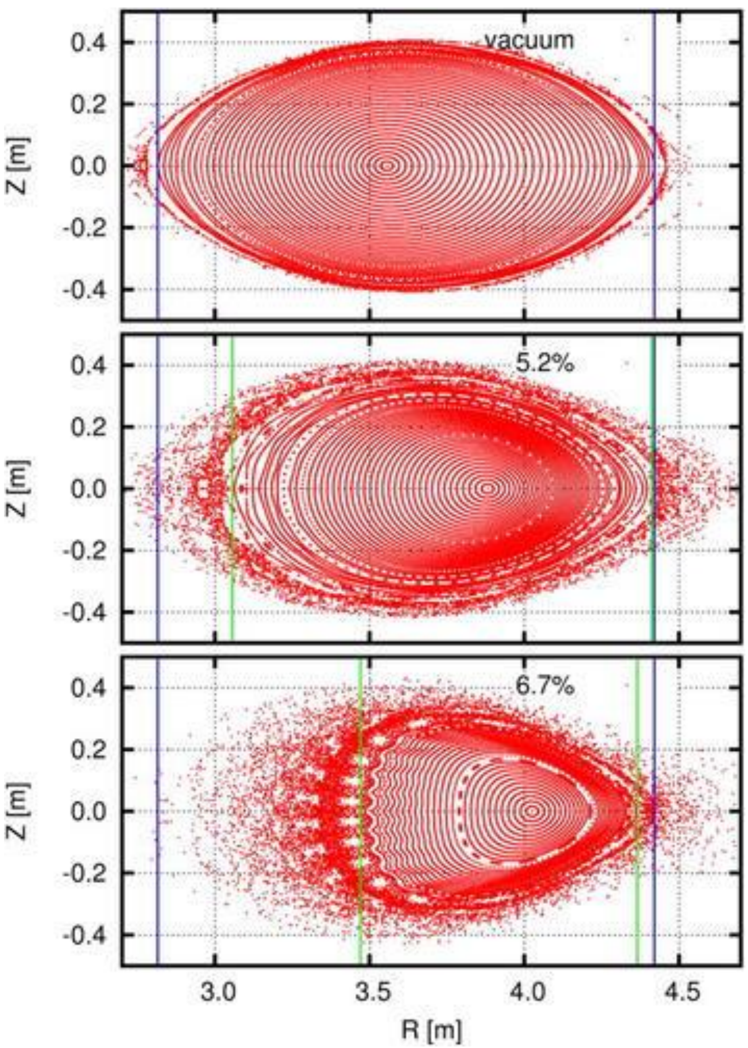
$$n_e^{\mathrm{Sudo}} \propto \sqrt{P_{\mathrm{aux}} B / a^2 R}$$

by J. Miyazawa

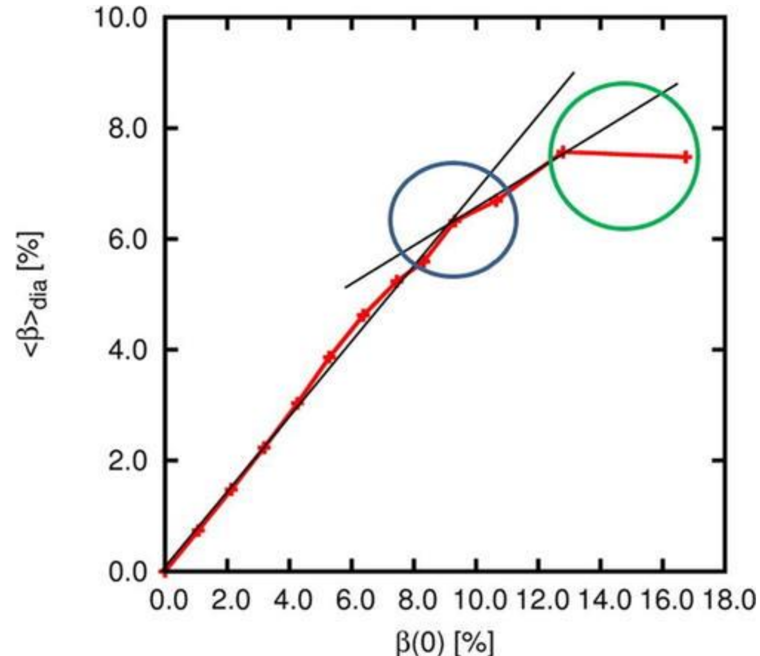
P. Zanca *et al.*, NF **57** (2017) 056010.

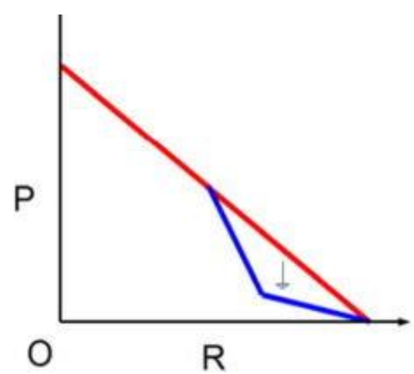
- Sudo(-like) scaling has reinterpreted as the ‘edge’ density limit scaling.

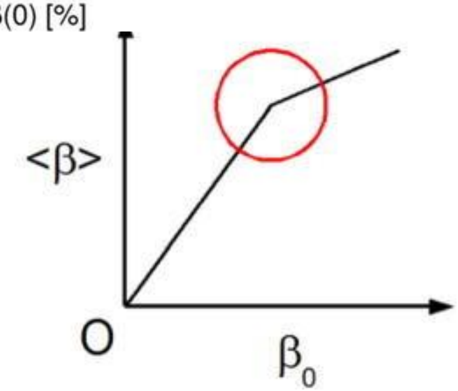
# Equilibrium beta limit by Shafranov shift and stochastisation




Y. Suzuki et al., PoP **27** (2020) 102502.



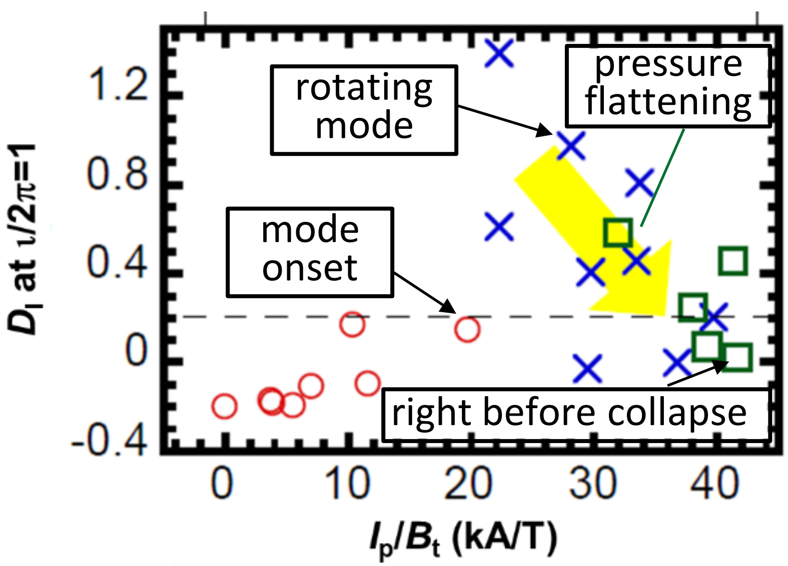



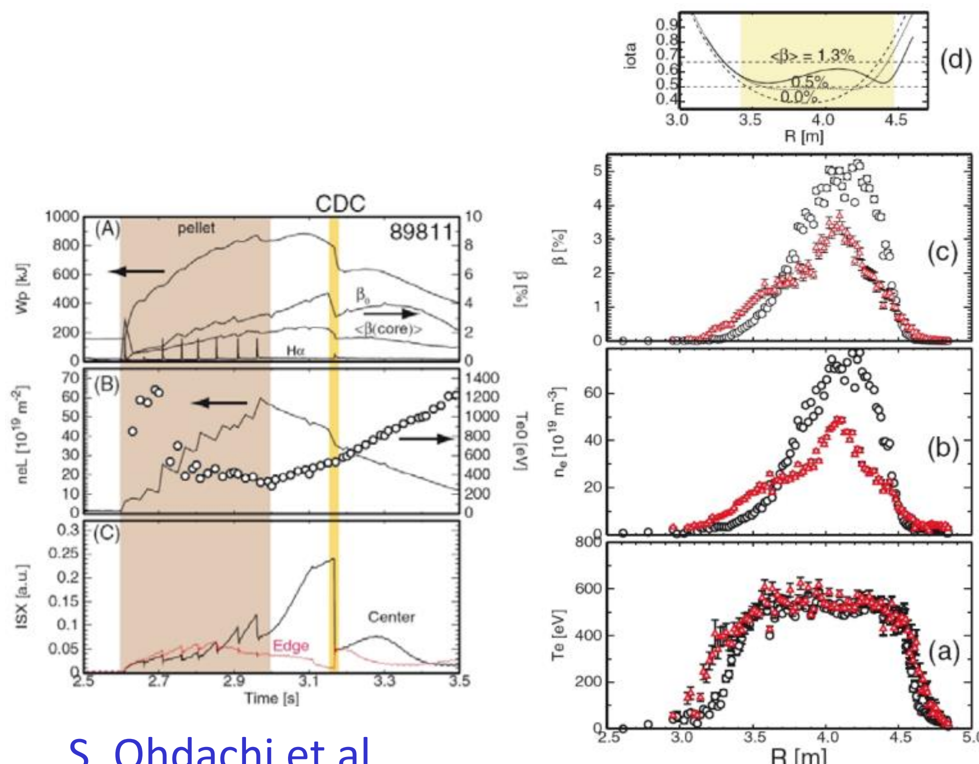
# Beta limit in LHD experiment

- **Core MHD instability**
  - Low-*n* MHD mode which causes pressure collapse generates when Mercier parameter $D_I$ exceeds 0.2–0.3 at rational surface of $\iota/2\pi = 1$.

- **Core density collapse (CDC)**
  - High-*n* balloning mode at the edge is destabilised
  - Does not occur in the case of inward-shifted configuration




by S. Sakakibara

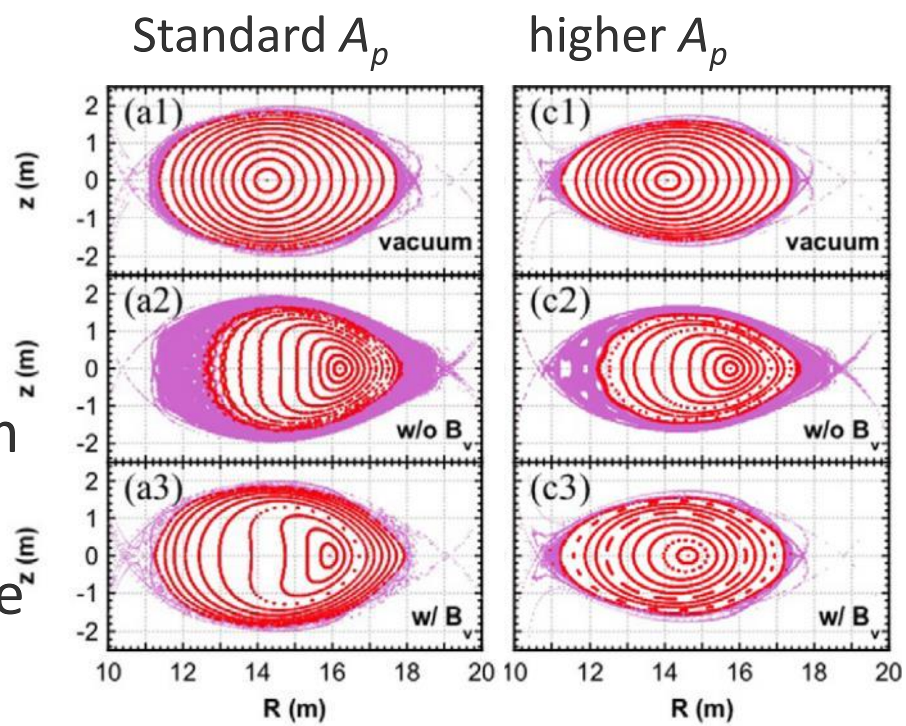



S. Ohdachi et al.,
Contrib. Plasma Phys. **50** (2010) 552.

# Consideration on plasma operation control

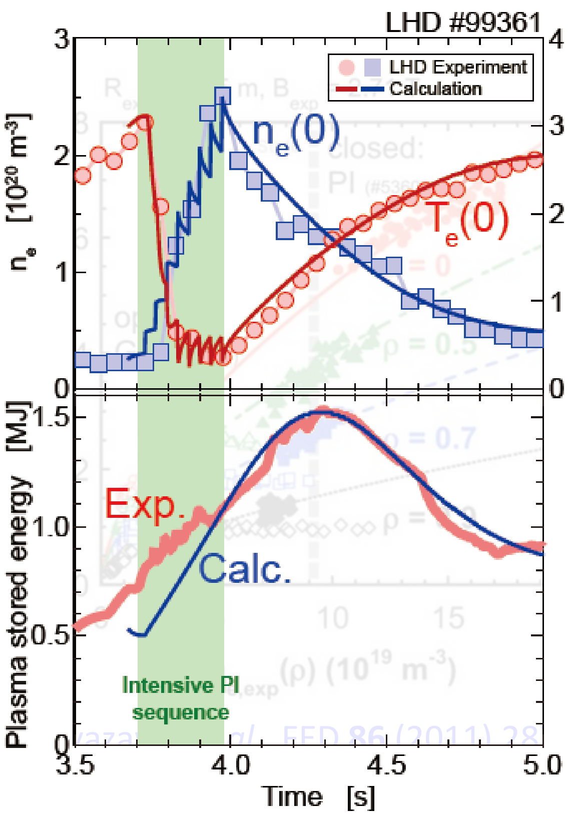
- There are no disruptive events and no limit on plasma ramp-up time in heliotron/stellarator.
  - Plasma ramp-up can be achieved by successive transitions of quasi-steady state
  
  ➔ Possibility of operation control with a few simple measurements

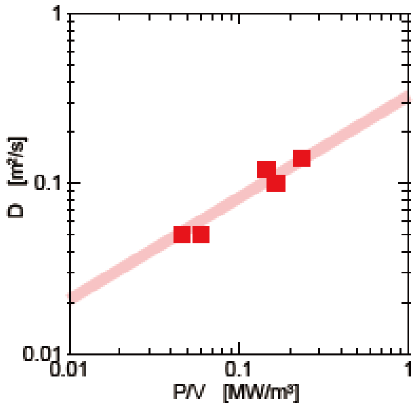
- Density limit can be avoided by increasing heating power
- Beta limit can be avoided by:
  - Compensate Shafranov shift
    
    ➔ Control the vertical field
  - Control the rotational transform
    
    ➔ Current drive
  - Avoid operation regime with the limit in the first place
    
    ➔ Configuration optimisation

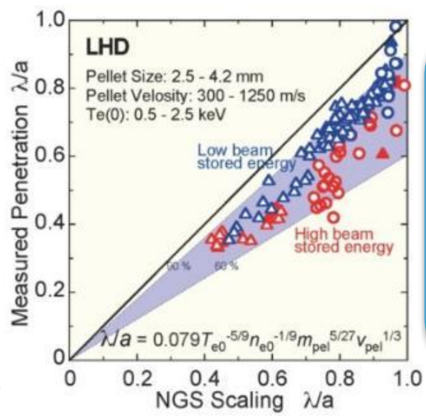



J. Miyazawa et al., NF **54** (2014) 043010.

# Model for examination of operation control scenario




Thermal transport analysis by the equation:

$$\frac{\partial p(r)}{\partial t} = \frac{1}{\tau_E}\{\hat{p}_{\mathrm{exp}}(r) n(r)^{0.6} P_{\mathrm{abs}}^{0.4} B^{0.8} - p(r)\}$$

with $\hat{p}_{\mathrm{exp}}(r) = \dfrac{p_{\mathrm{exp}}(r)}{n_{\mathrm{exp}}(r)^{0.6} P_{\mathrm{exp}}^{0.4} B_{\mathrm{exp}}^{0.8}}$

Particle transport is calculated by solving diffusion equation:

$$\frac{\partial n(r)}{\partial t} = \frac{1}{r}\frac{\partial n}{\partial r}\left\{r\left(D\frac{\partial n}{\partial r} - nV\right)\right\} + S$$

with $D(r) = D \propto (P_{\mathrm{abs}}/n)^{0.6} B^{-0.8}, V = 0$

- Strong gyro-Bohm-type parameter dependence is observed between local electron density and pressure in LHD plasma.
- Radial profile of electron pressure is estimated by assuming gyro-Bohm normalized pressure profile is conserved.

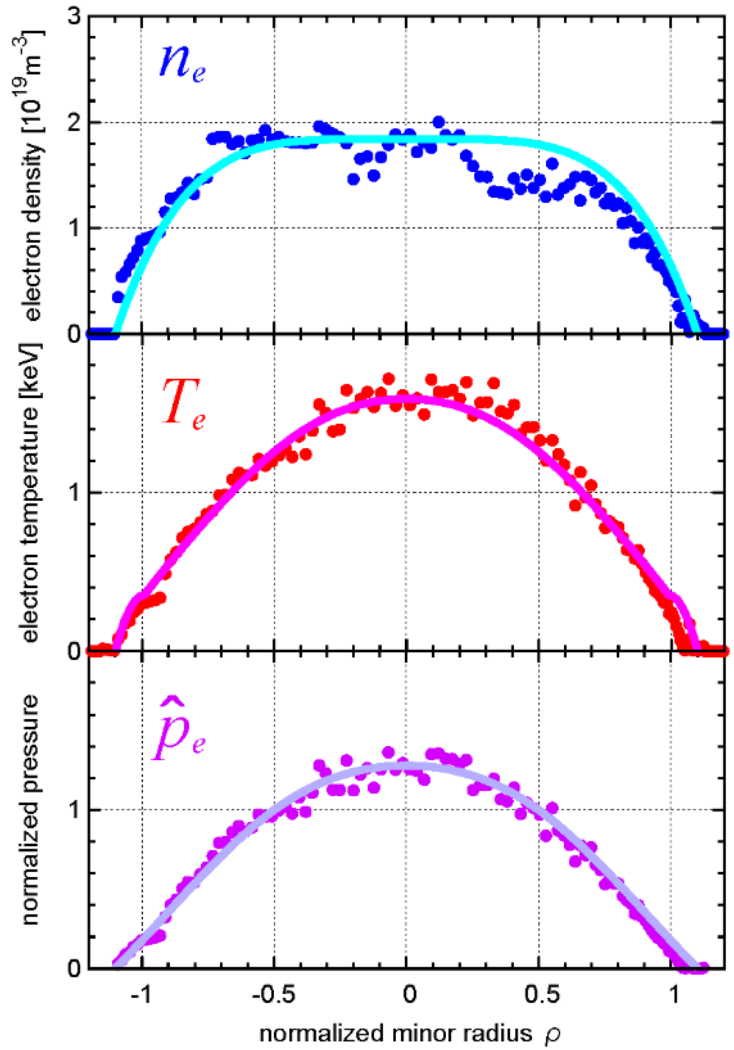
# Prerequisites of the calculation

- **Pressure profile**
  - Extrapolated from LHD experiment with high aspect ratio, inward-shifted configuration
- **Fueling**
  - Assuming $2 \times 10^{22}$ particles/pellet (~10% of main plasma) and injection velocity of 1.5 km/s

➔ Assuming that the flat density profile is maintained

- **MHD equilibrium**
  - Using VMEC calculation result with the fixed boundary shape as that in vacuum condition

➔ Assuming magnetic axis position control according to beta value

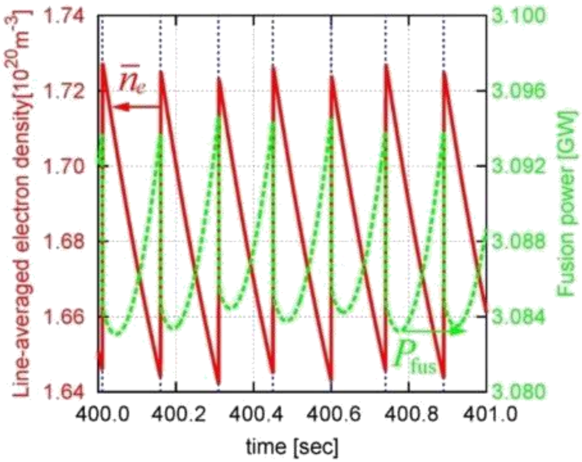


# Control method

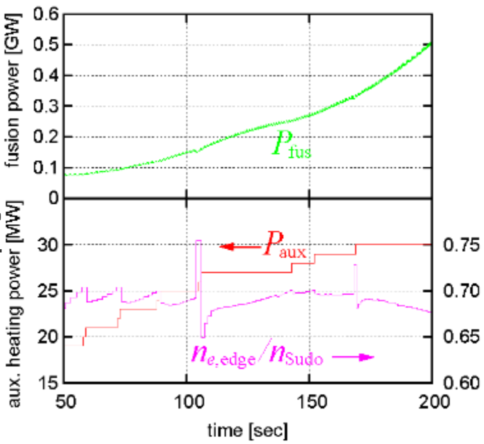
- Feedback control of **pellet fueling** based on the measurement of the **line-averaged electron density**
  - Stable control wo/ delayed response
  - Plasma radial profile is almost unchanged at the reactor condition
  - Required resolution: ~$10^{17}$ m$^{-3}$, <100 ms

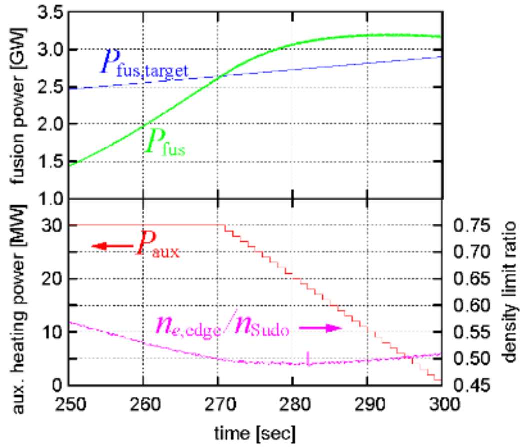
    Can be achieved by the combination of dispersion interferometer and polarimeter : (T. Akiyama *et al.*, NF **55** (2015) 093032)

T. Goto *et al.*, FED **89** (2014) 2451.

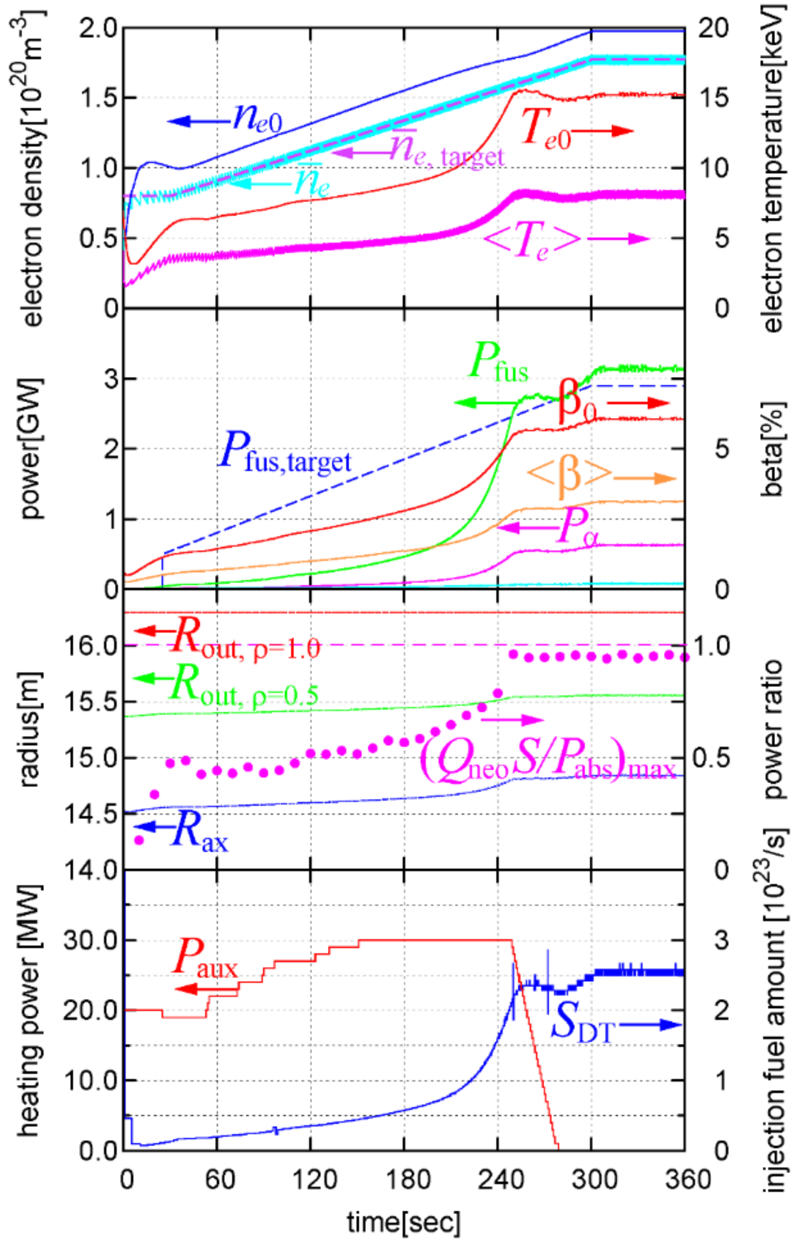



- Staged variation of the **aux. heating power** based on the measurement of **the edge density and the fusion power**
  - Required resolution: ~10%
  - Variation range of 1 MW at an interval of 1 sec is sufficient

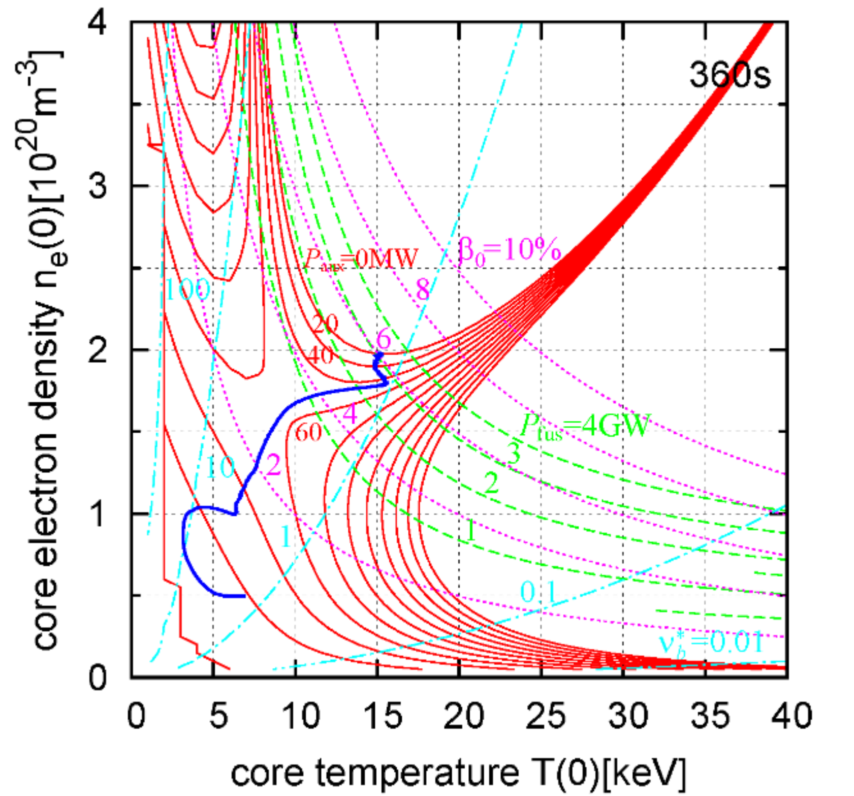


# Examination of operation control scenario

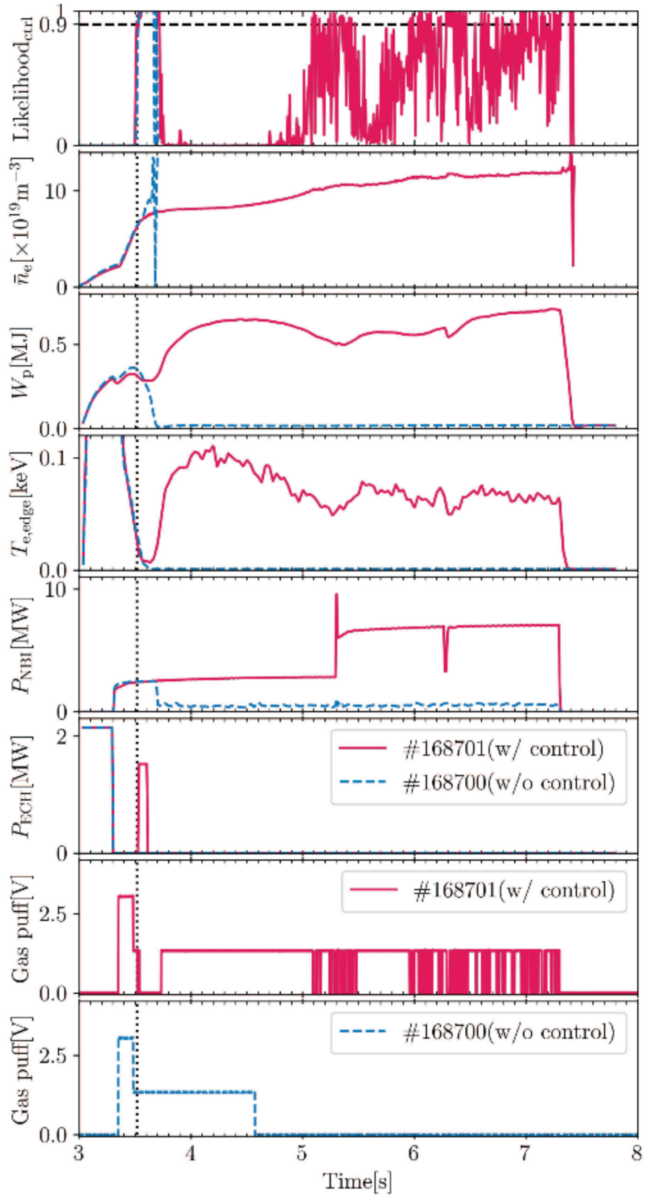



T. Goto *et al.*, NF **55** (2015) 063040.

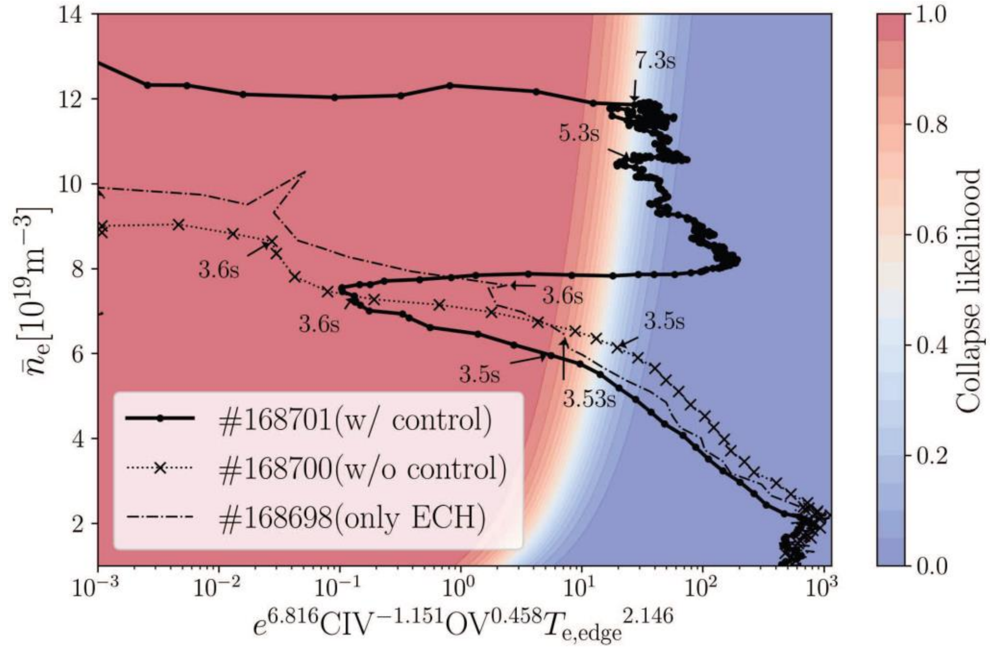



- Smooth variation and steady-state sustainment of the fusion power are demonstrated.

# Collapse avoidance experiment in LHD




T. Yokoyama et al.,
Plasma Fusion Res. **17** (2022) 2402042



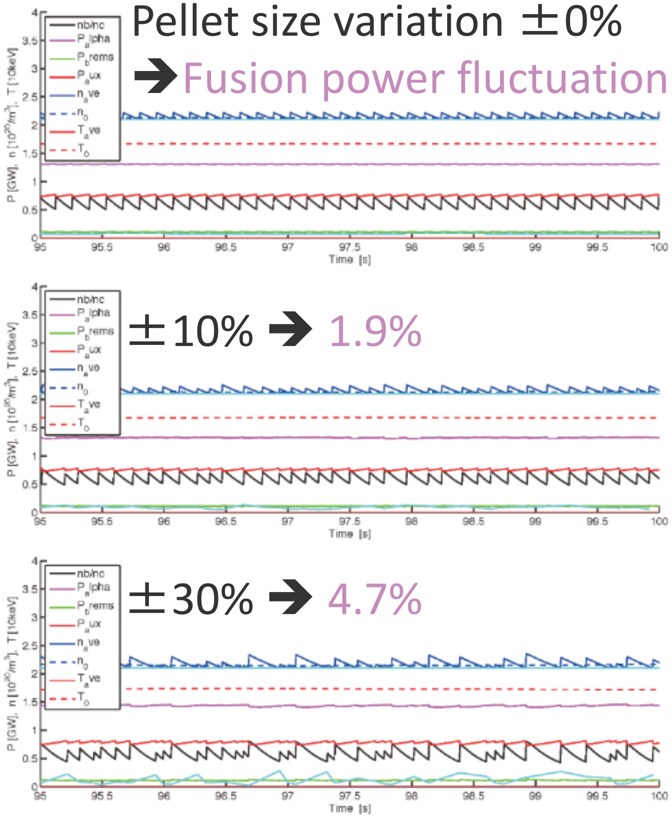

- Collapse likelihood is evaluated by data-driven approach:
  $\bar{n}_e(\mathrm{Likelihood}) \propto \mathrm{CIV}^{-1.151}\mathrm{OV}^{0.458}T_{e,\mathrm{edge}}^{2.146}$
- Collapse can be avoided by boosting ECH when collapse likelihood exceeds 0.9

# Controllability against the variation in pellet fueling

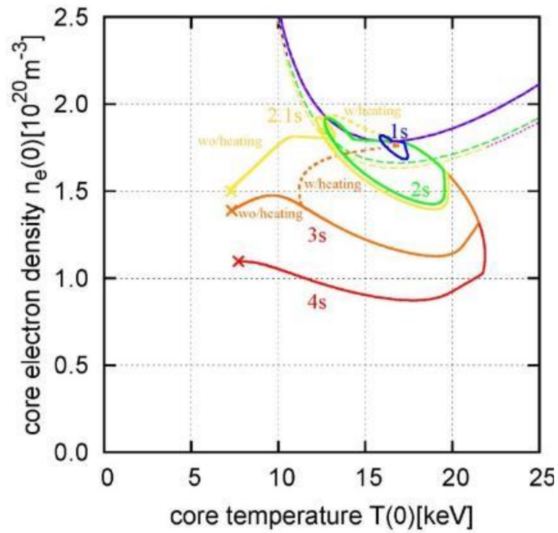
Pellet size variation ±0%
➔ Fusion power fluctuation 0.8%

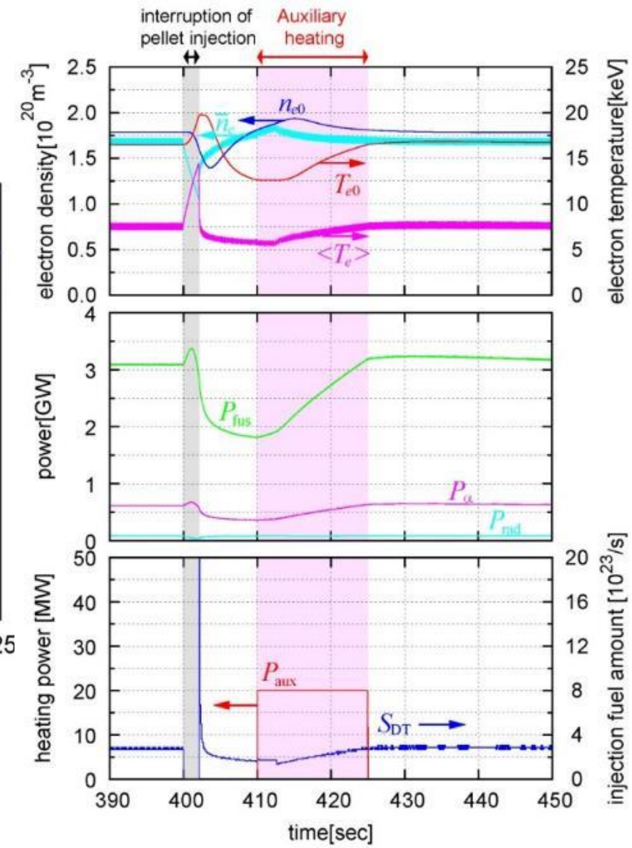
±10% ➔ 1.9%

±30% ➔ 4.7%



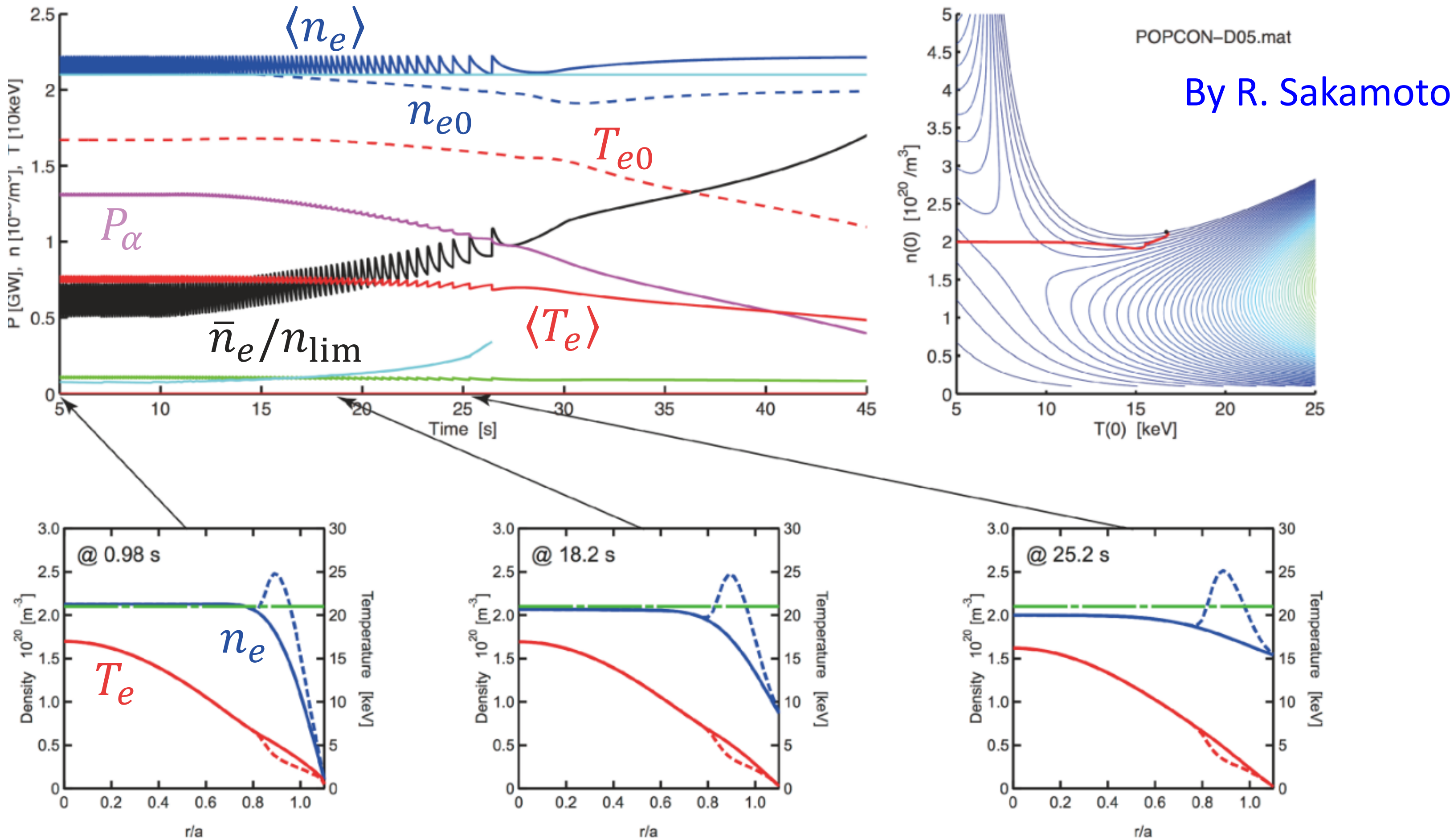

By R. Sakamoto




- Variation in the pellet size with 30% is acceptable.
- Fusion power can be recovered after the pellet injection failure for ≲2 s (additional heating can improve the recoverability).

# Effect of the edge density

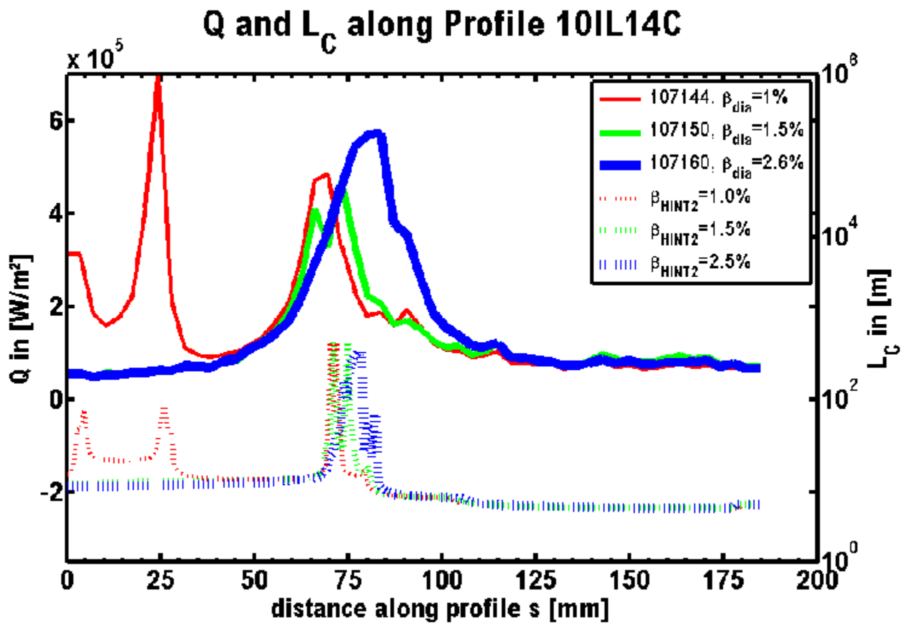



- If the edge density increases, the line averaged electron density does not decrease and the pellet injection stops, resulting in the decrease of the fusion power and radiating collapse.

➔ Edge density control is necessary for a stable operation.

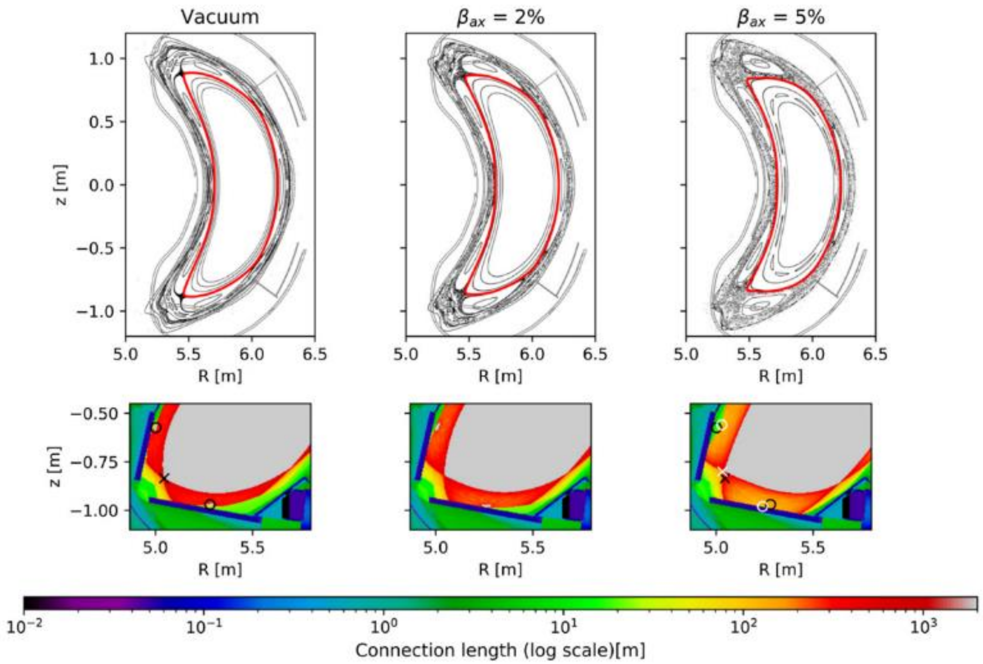
# Beta effect on divertor heat load profile

## LHD




P. Drewelow et al., 39th EPS

## W7-X (calc.)



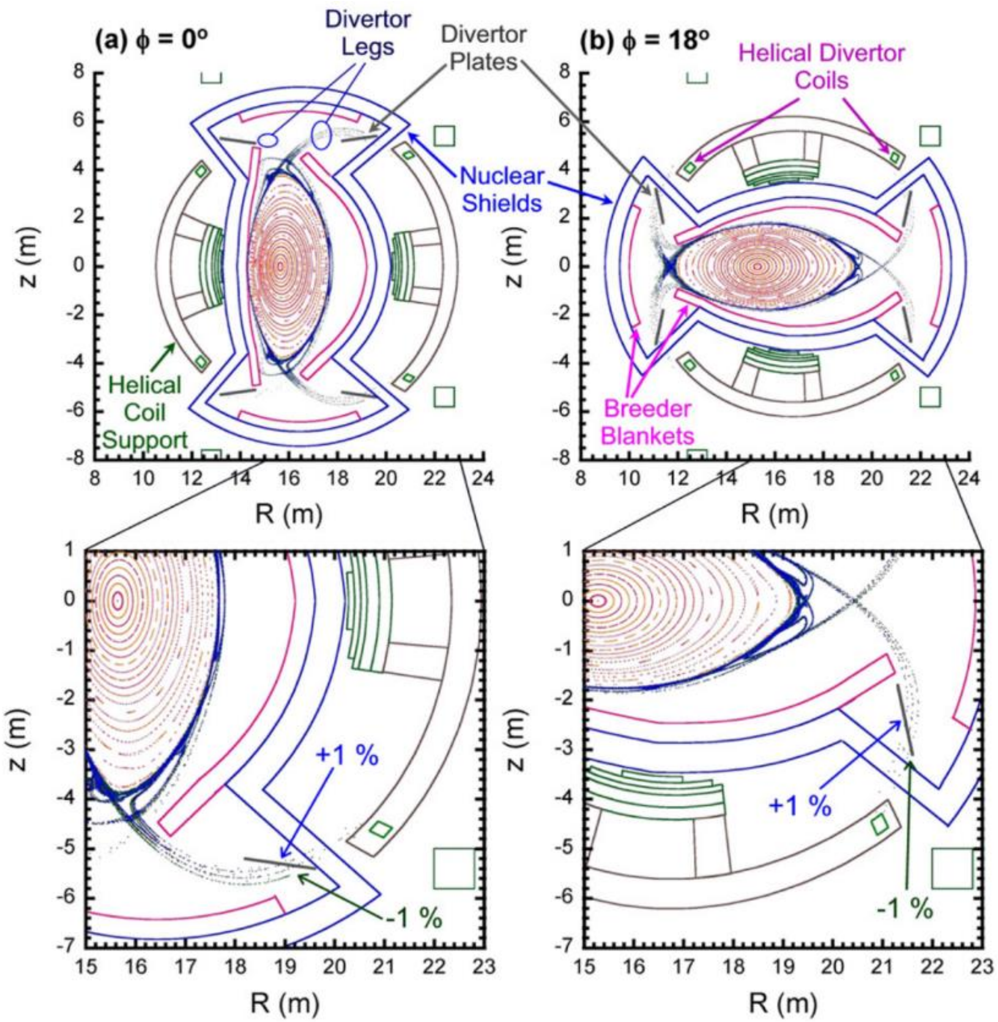

A. Knieps et al., NF **62** (2022) 026011.

- Divertor heat load profile is strongly correlated to the pattern of field lines with long connection lengths both in LHD and W7-X.
- Though the peak position of the heat load changes with the increasing beta, the divertor magnetic field lines are properly hitting somewhere on the divertor plate, and the peak heat load factor does not change much.
- Divertor detachment is needed as long as the solid target is used.

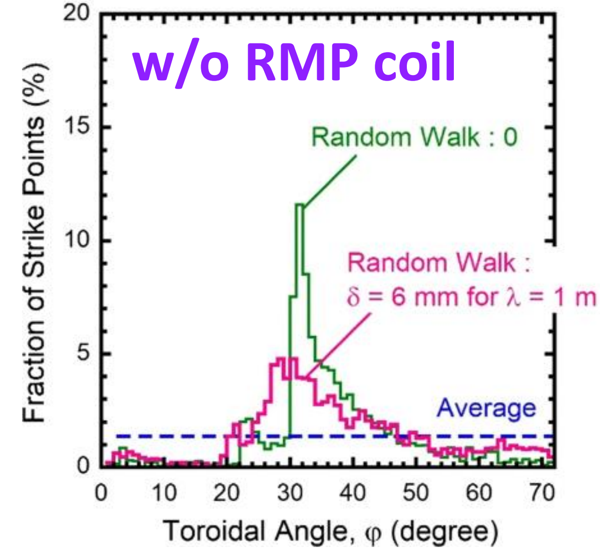
# Sweeping / averaging of divertor footprints

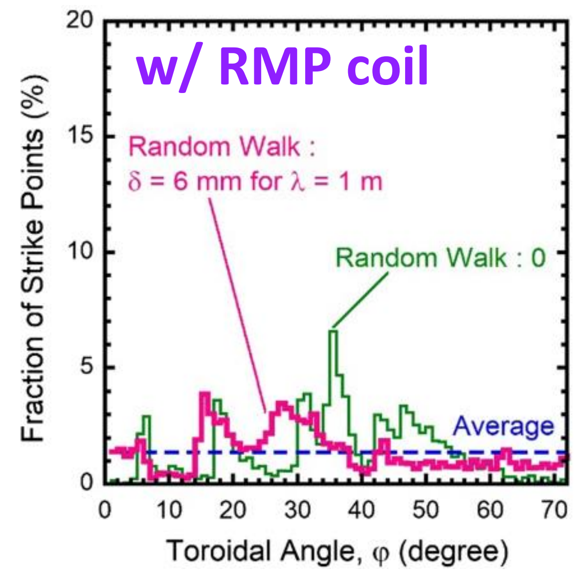
N. Yanagi et al., NF **51** (2011) 103017.

Current sweeping by helical divertor coils at ±2% of the main helical coil at 0.5 Hz

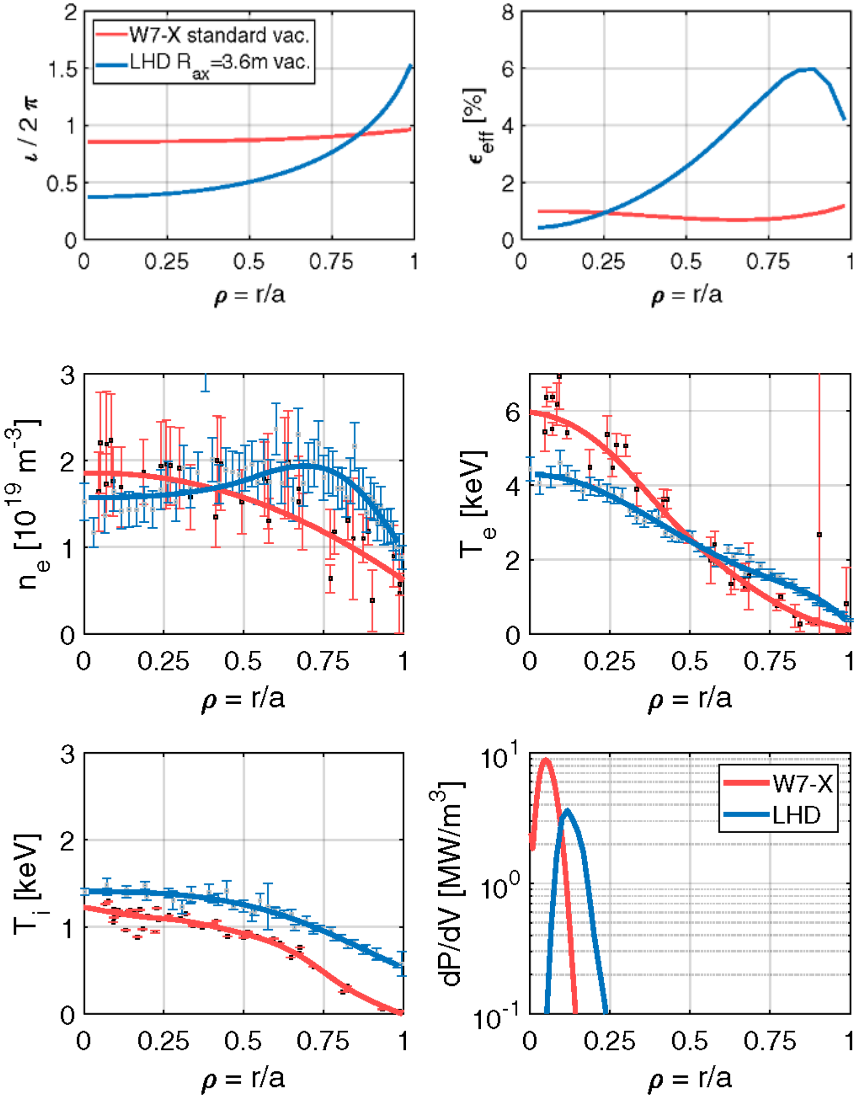


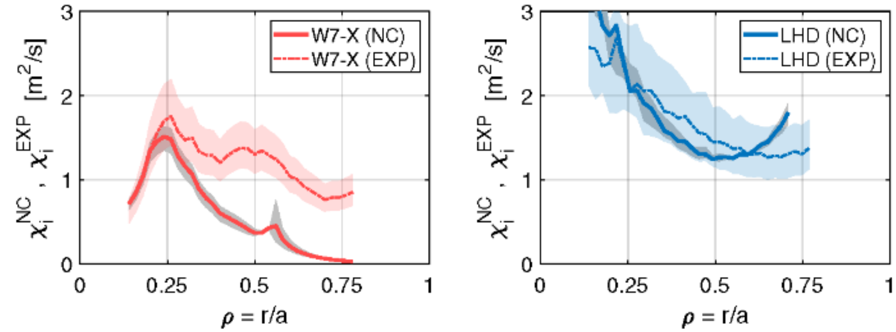

N. Yanagi et al., Journal of Fusion Energy **38** (2019) 147

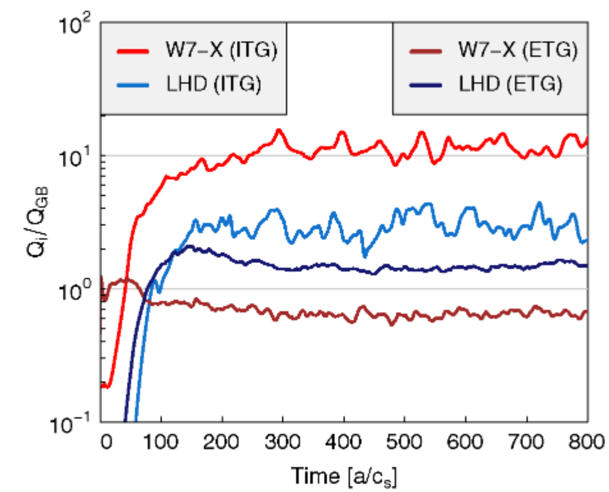


# Difference in heat transport btw. W7-X and LHD



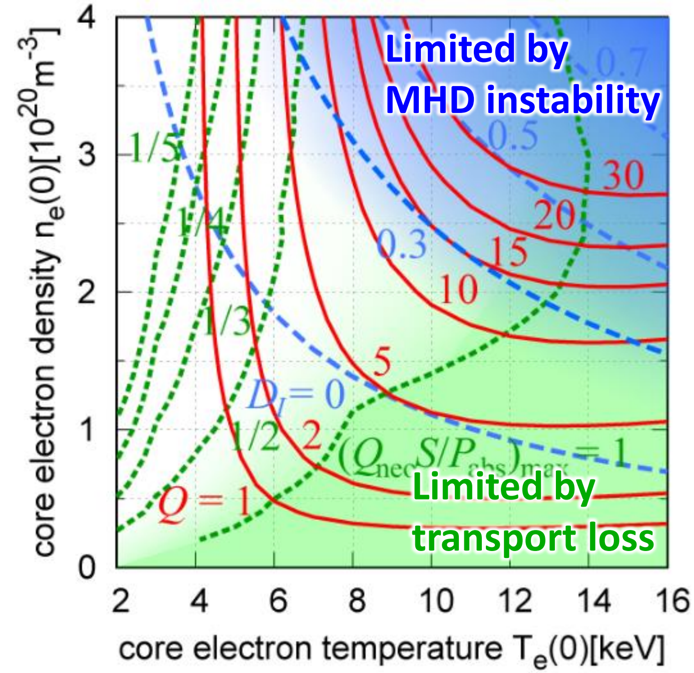



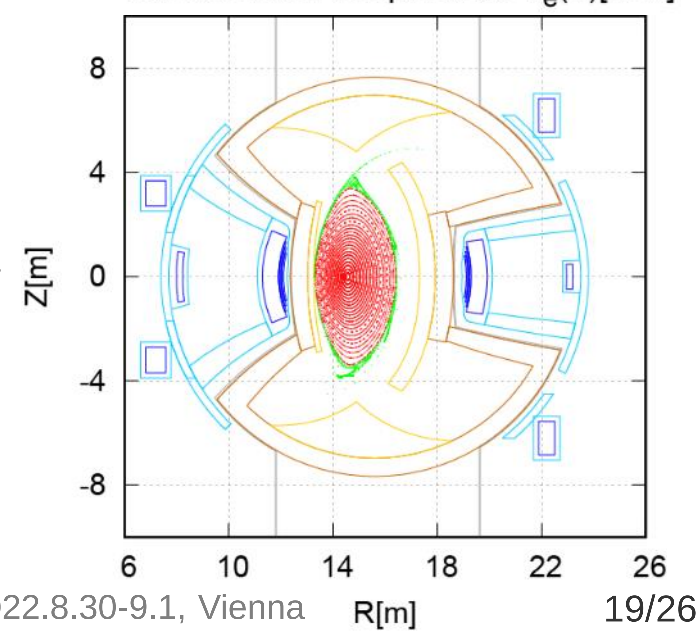




F. Warmer et al., PRL **127** (2021) 225001.

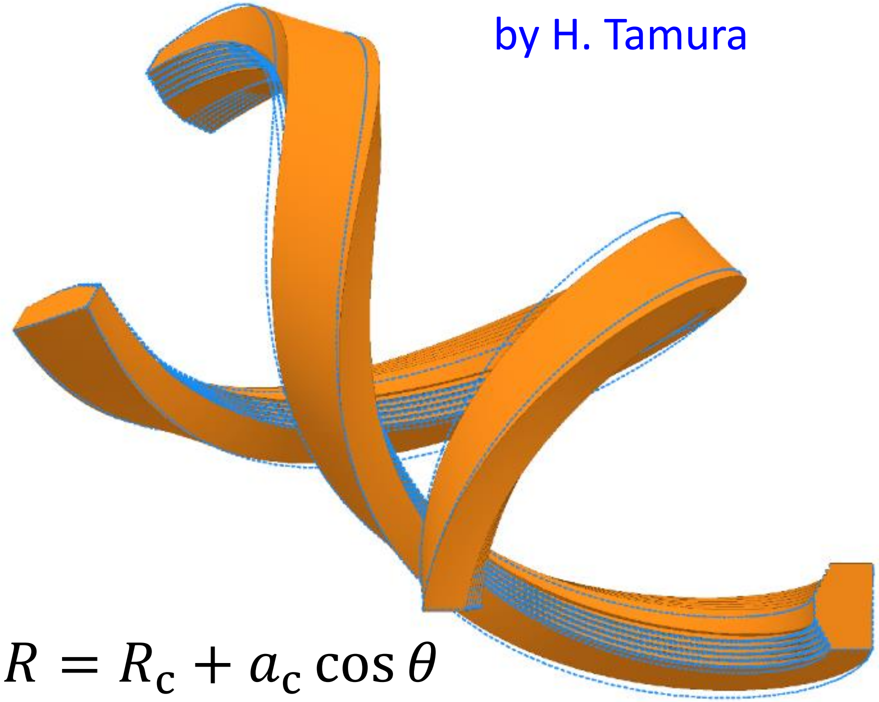
# Two major issues towards LHD-type fusion power plant

- **Physics issue**
  - Trade-off btw. MHD stability and energy confinement property
  - Achievable fusion gain is limited to ~10 with a feasible magnet design if there is no improvement in plasma performance from present LHD experimental results
- **Engineering issue**
  - Limited space btw. the plasma and the helical coil
  - Reduction of the reactor size is difficult due to the insufficient neutron shielding performance and tritium breeding ratio
- Reactor design with a compact size and high power density is difficult.



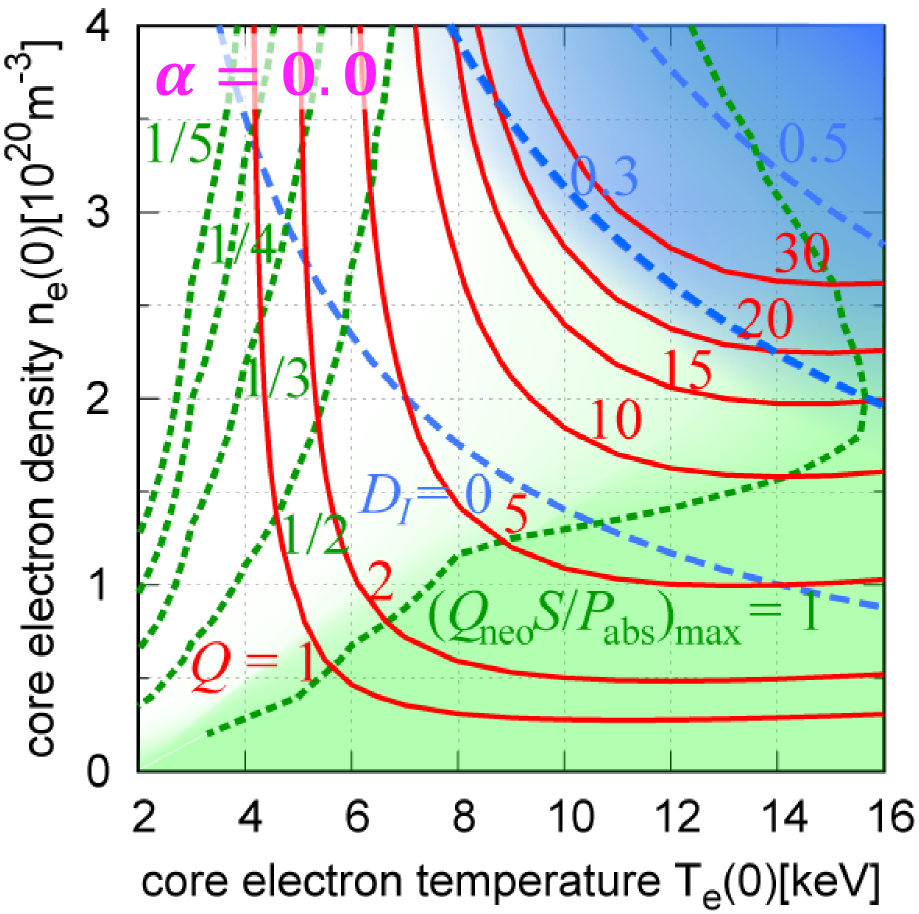
# Room for optimization of the shape of helical coils




$$R = R_c + a_c \cos\theta$$
$$Z = a_c \sin\theta$$
$$\theta = -\frac{m}{\ell}\phi - \alpha \sin\left(\frac{m}{\ell}\phi\right)$$

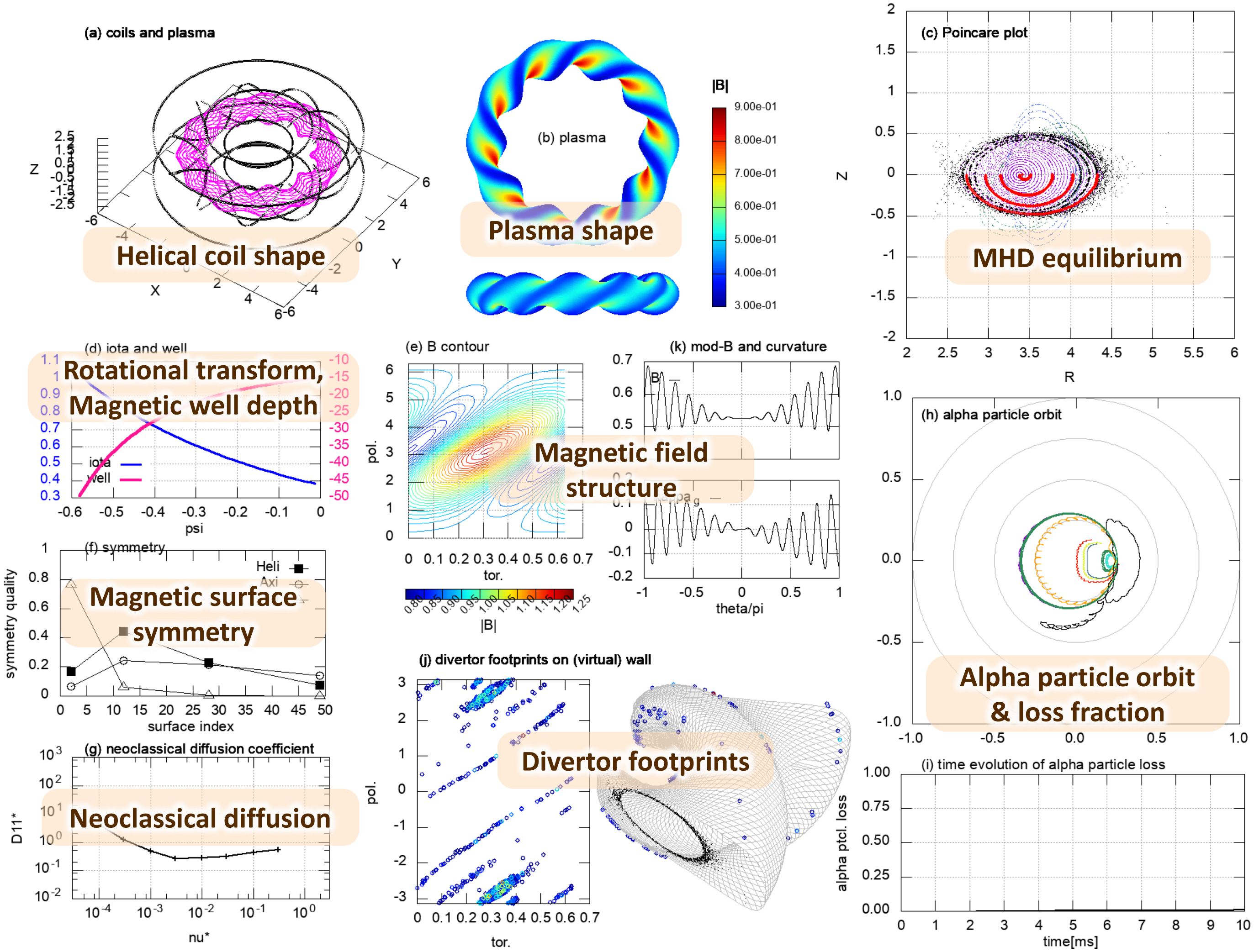
solid: $\alpha = 0.0$, broken: $\alpha = 0.1$ (LHD)

T. Goto et al., Plasma Fusion Res. **16** (2021) 1045085.




- Slight change in the pitch modulation $\alpha$ $(0.1 \rightarrow 0.0)$ enables simultaneous improvement of MHD stability and energy confinement. However, the blanket space decreases.

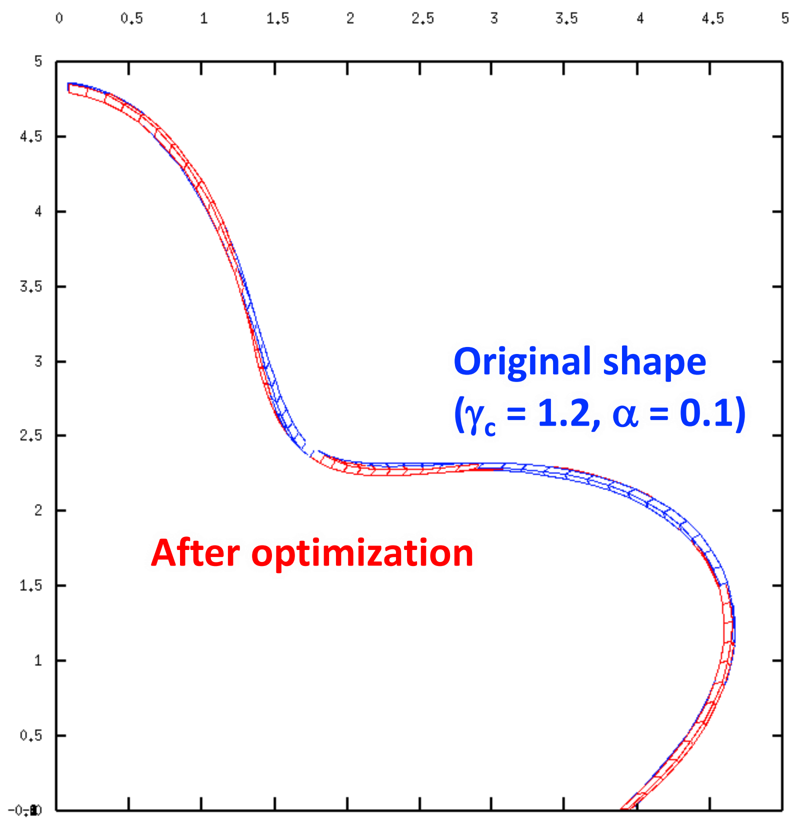
# Helical coil optimization code "OPTHECS"

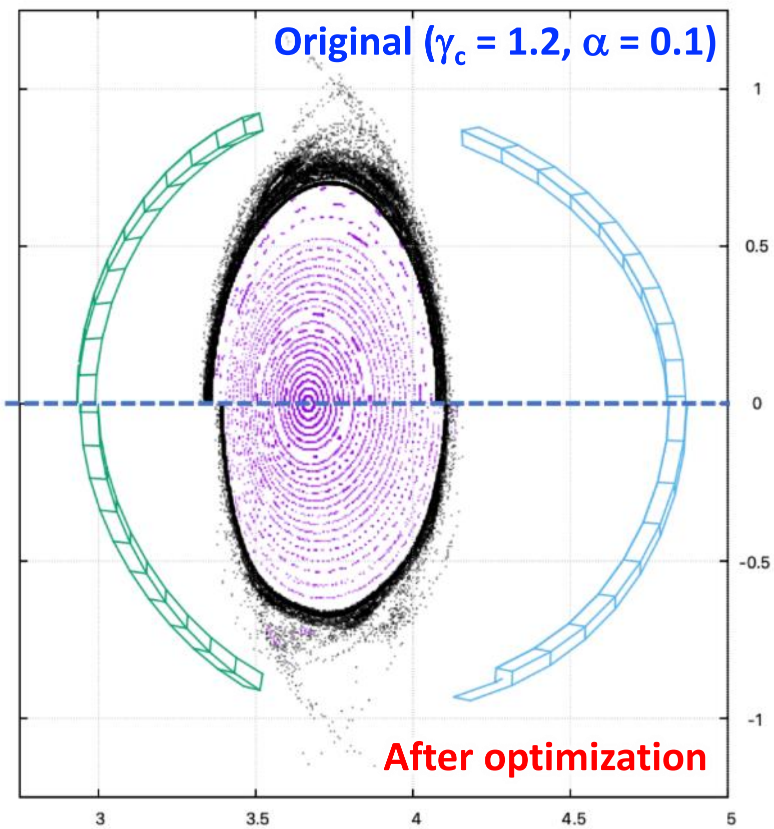



H. Yamaguchi,
ITC-28, 2019, O1-4

- Optimization of the coil shape and current by considering overall plasma performance has become possible.

# Optimization result – coil shape and blanket space




- ~10% increase in the blanket space is achieved with a comparable plasma performance to the original configuration

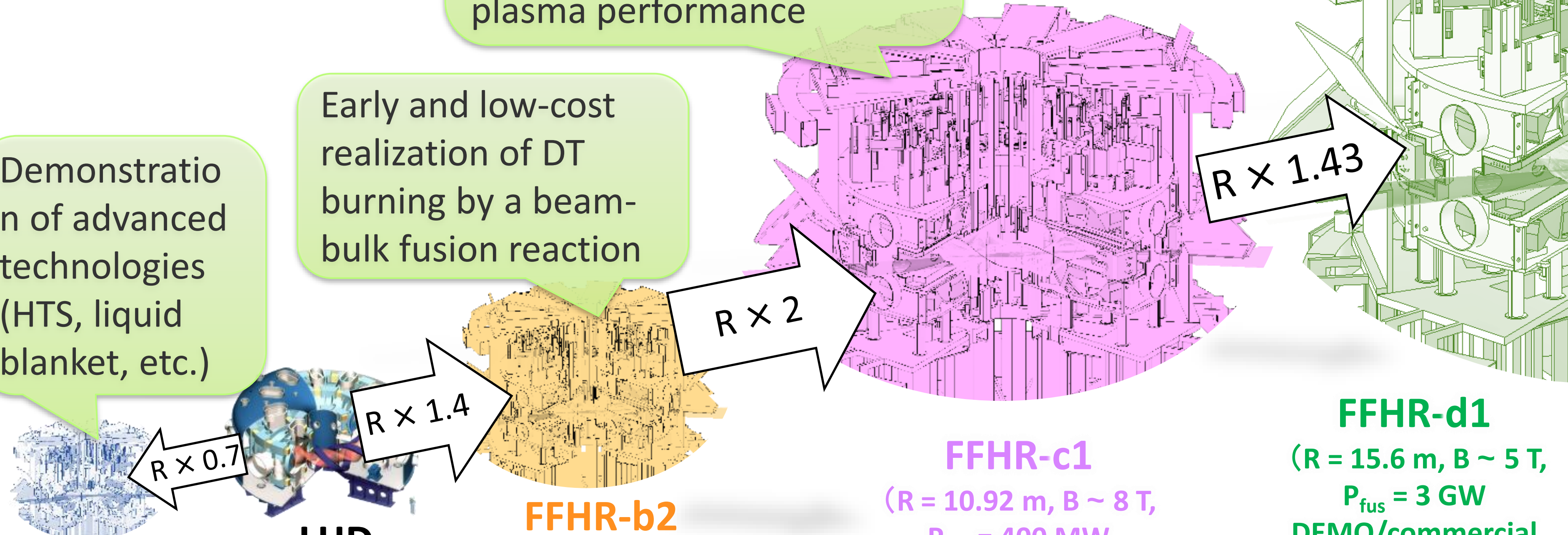
# "Original" development strategy of helical fusion reactor

Can be built with an extension of the ITER technology, but improvement in plasma performance is required

The smallest size device that enables self-sufficiency of electricity and tritium fuel without any improvement in plasma performance

Early and low-cost realization of DT burning by a beam-bulk fusion reaction

Demonstration of advanced technologies (HTS, liquid blanket, etc.)




**FFHR-a1**
**(R = 2.73 m, B ~ 4 T**
**Primary reactor)**
Demonstration of advanced engineering concepts

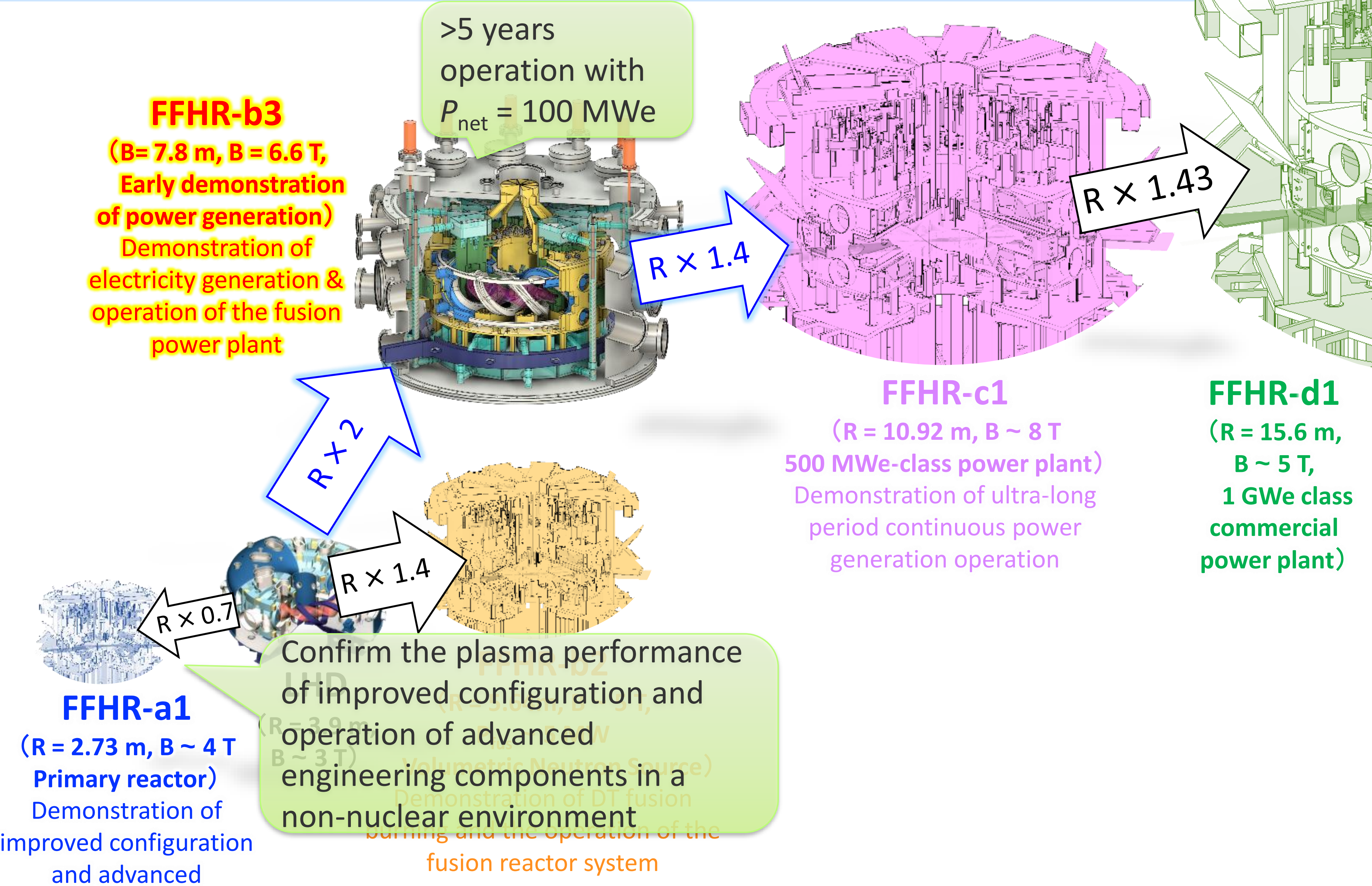
**LHD**
**(R = 3.9 m, B ~ 3 T)**

**FFHR-b2**
**(R = 5.64m, B ~ 5 T, $P_{fus}$ ~ 5 MW**
**Volumetric Neutron Source)**
Demonstration of DT fusion burning and the operation of the fusion reactor system

**FFHR-c1**
**(R = 10.92 m, B ~ 8 T, $P_{fus}$ = 400 MW**
**Exp./Prototype)**
Demonstration of year-order steady-state operation of the fusion power plant

**FFHR-d1**
**(R = 15.6 m, B ~ 5 T, $P_{fus}$ = 3 GW**
**DEMO/commercial power plant)**
Demonstration of plant economy and safety

# “New” development strategy of helical fusion reactor



# Design requirements to realize new design “FFHR-b3”

| | Previous designs | FFHR-b3 |
|---|---|---|
| **Plasma temperature** | ≤ 9 keV (neoclassical transport calculation) | ≤ 11.7 keV (optimum value from the viewpoint of plasma power balance ) |
| **Beta value** | ≤ 3.0% (linear MHD stability analysis) | ≤ 5.0% (expected value by configuration optimization) |
| **Confinement improvement** | 1.0 (direct extrapolation from LHD) | 1.3 (deterioration due to the increase of plasma beta is considered) |
| **Helium ash fraction** | 5% | 3% (configuration optimization) |
| **Alpha particle loss** | 15% (orbit calculation) | 5% (configuration optimization) |
| **HC current density** | ≤ 48 A/mm$^2$ | ≤ 80 A/mm$^2$ (development target) |
| **Enlargement of the space between helical coil and plasma** | ~15% (supplemental coils) | ~25% (supplemental coil + optimization of HC winding law) |
| **Attenuation of fast neutron flux in breeding zone** | 1 order atten. by 30 cm | 1 order atten. by 20 cm (optimization of material selection and layout) |
| **Divertor heat recovery** | 20% | 30% (by design optimization) |
| **Thermal efficiency** | 42% | 50% (S-$CO_2$ gas turbine) |
| **Total efficiency of heating system** | 50% | 66% (target of JA-DEMO) |
| **Cryogenic efficiency** | 1.5% (20 K operation) | 2.0% (by design optimization) |

# Summary

- **Two major transients in heliotron/stellarator**
  - Density limit by the radiating collapse
  - Beta limit due to MHD instability with low-n mode
- **Avoidance of radiating collapse is a key control target**
  - Stable control of fusion power can be achieved with feedback control of pellet fuelling with few simple diagnostics
  - MHD instability can be avoided by selecting an adequate magnetic configuration and by controlling the vertical field
- **Further optimization is anticipated to realise a compact fusion power plant**
  - Simultaneous suppression of MHD instability and energy/particle transport in high beta condition
  - Ensuring blanket space at the inboard side of the torus
  - Numerical optimization of the shape of helical coils by OPTHECS is ongoing