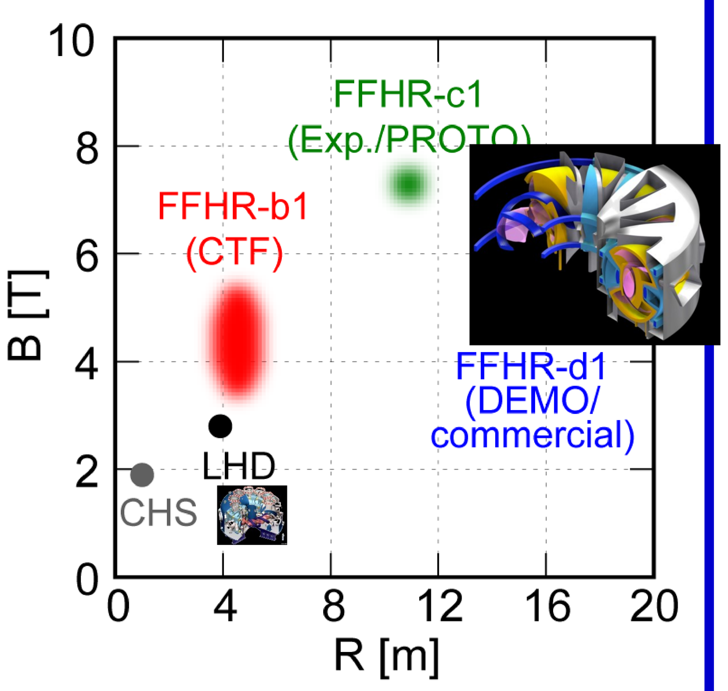
Importance of divertor physics modeling in system design of helical reactor

Takuya Goto, Masahiro Kobayashi, Nagato Yanagi

National Institutes of Natural Sciences, National Institute for Fusion Science, Toki, Gifu, Japan

Introduction

- Step-by-step development strategy[1] has been proposed towards LHD-type helical fusion power plants and 3 designs have been proposed.
  - **FFHR-b1** (CTF): compact reactor (R = 3–5 m) which aims to breakeven condition (Q > 1) with beam-driven fusion
  - **FFHR-c1** (Experimental/Prototype) : medium size reactor ($R \sim 10$ m) which aims to electric breakeven condition ($P_{\mathrm{e,net}} > 0$)
  - **FFHR-d1** (DEMO): commercial-scale reactor ($P_{\mathrm{e,net}} \sim 1$ GW) with a self-ignition plasma

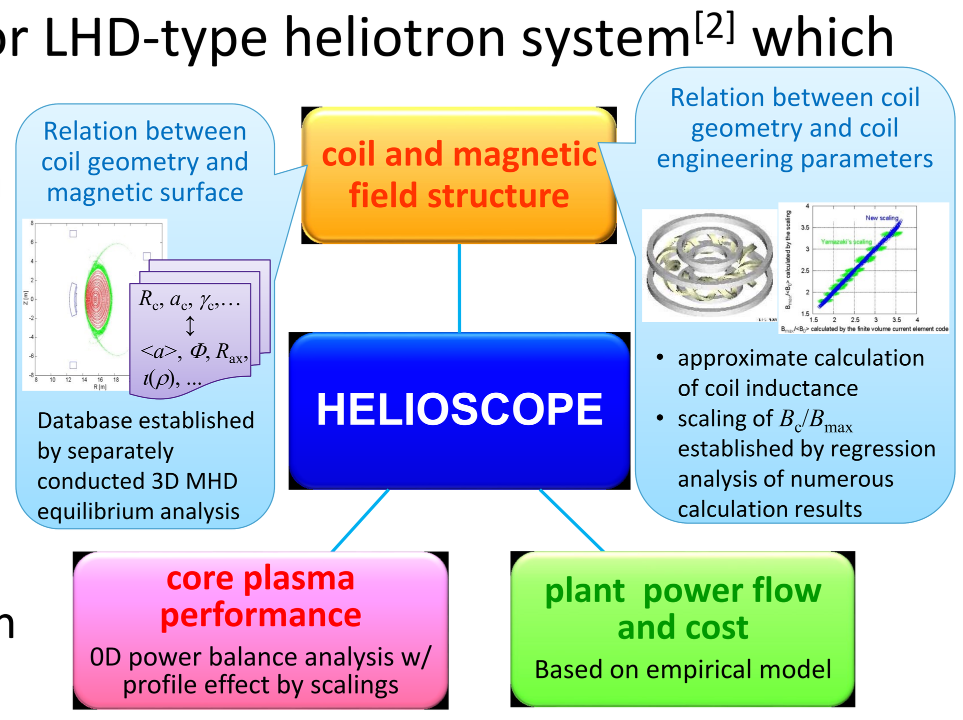


- Design parameters of these designs was selected by considering the following 3 factors:
  - Core plasma performance (fusion power)
  - Space for blanket (TBR, neutron shielding performance)
  - Stored magnetic energy (engineering difficulty, construction cost)

Design window analysis for the LHD-type helical reactor including the design points of the proposed designs with a consideration on the divertor heat load has been conducted.

Systems code HELIOSCOPE

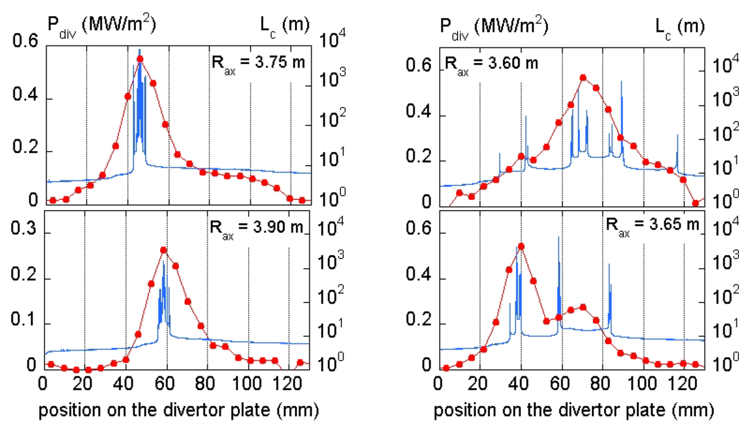
- Systems code is an assembly of simple algebraic models for all components in fusion power plant system.
- HELIOSCOPE (Heliotron Systems Code for Plant Performance Evaluation) is a systems code for LHD-type heliotron system[2] which enables:
  - Plasma performance analysis with an arbitrary radial profile and 3D magnetic geometry
  - SC coil design considering actual shape of 3D helical coils
  - Cost and plant power balance analysis considering all components/facilities in the fusion power plant



- Divertor heat load Γ_{div} is estimated using a simple model:

$$\Gamma_{\mathrm{div}} = P_{\mathrm{div}}/S_{\mathrm{div}},\quad S_{\mathrm{div}} = S_{\mathrm{div,LHD}}(R_c/R_{\mathrm{c,LHD}})^2.$$

- This model is based on the following facts:
  - LHD-type system has a rigid divertor field structure that is less sensitive to the core plasma state
  - In the LHD experiment, strong correlation is observed between the particle flux (ion saturation current) and the connection length of the lines of the magnetic force[3]
  - If the shapes of two reactors are similar to each other, the magnetic field structures including the divertor region are also similar to each other

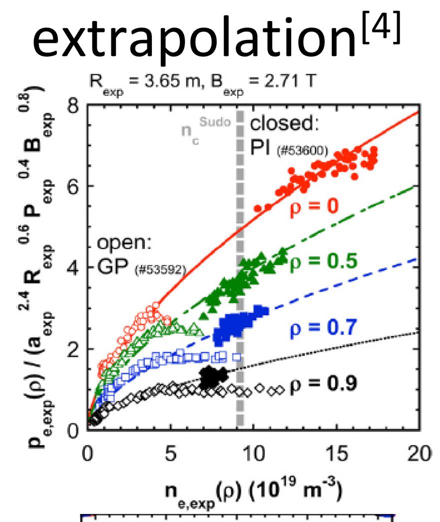


- The total power to divertor region is estimated as follows:

$$P_{\mathrm{div}} = P_\alpha - P_{\mathrm{brem}} + P_{\mathrm{aux}}.$$

Design window analysis

- Reactor size (R_c) and the magnetic field (B_c) were scanned.
- Plasma pressure is estimated by direct profile extrapolation[4] from the reference LHD experimental data



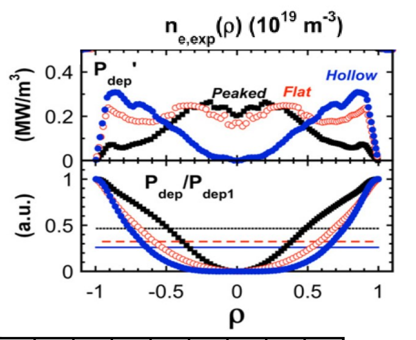
$$p_e(r) = \gamma_{\mathrm{DPE}}\hat{p}(r)P_{\mathrm{abs}}^{0.4}B^{0.8}n_e(r)^{0.6},$$

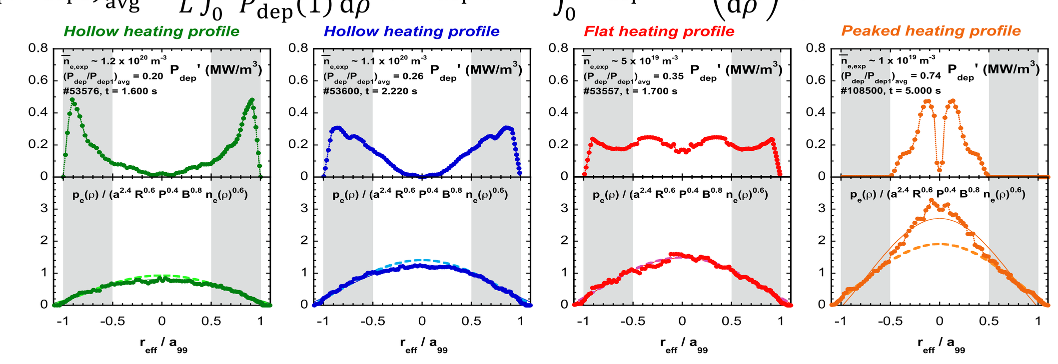
$$\hat{p}(r) = \frac{p_{e,\mathrm{exp}}(r)}{P_{\mathrm{abs,exp}}^{0.4}B_{\mathrm{exp}}^{0.8}n_{e,\mathrm{exp}}(r)^{0.6}},$$

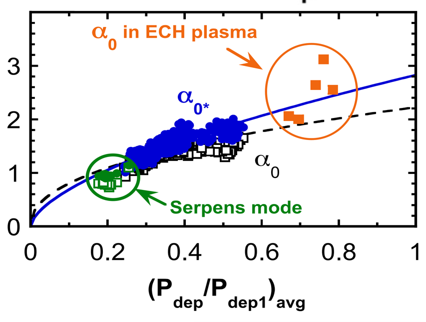
with confinement improvement effect by the peakedness of the heating profile

$$\gamma_{\mathrm{DPE}} = \left\{\frac{(P_{\mathrm{dep}}/P_{\mathrm{dep1}})_{\mathrm{avg}}}{(P_{\mathrm{dep}}/P_{\mathrm{dep1}})_{\mathrm{avg,exp}}}\right\}^{0.6}.$$

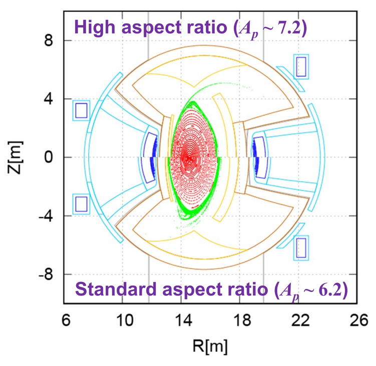
$$(P_{\mathrm{dep}}/P_{\mathrm{dep1}})_{\mathrm{avg}} = \frac{1}{L}\int_0^1 \frac{P_{\mathrm{dep}}(\rho)}{P_{\mathrm{dep}}(1)}\frac{\mathrm{d}L}{\mathrm{d}\rho}\mathrm{d}\rho, P_{\mathrm{dep}}(\rho) = \int_0^\rho P_{\mathrm{dep}}'(\rho')\left(\frac{\mathrm{d}V}{\mathrm{d}\rho'}\right)\mathrm{d}\rho'$$

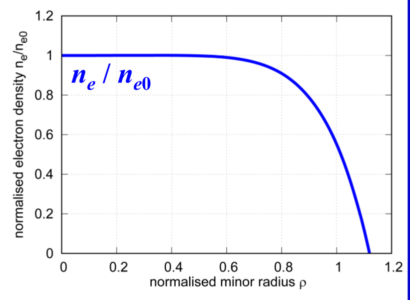


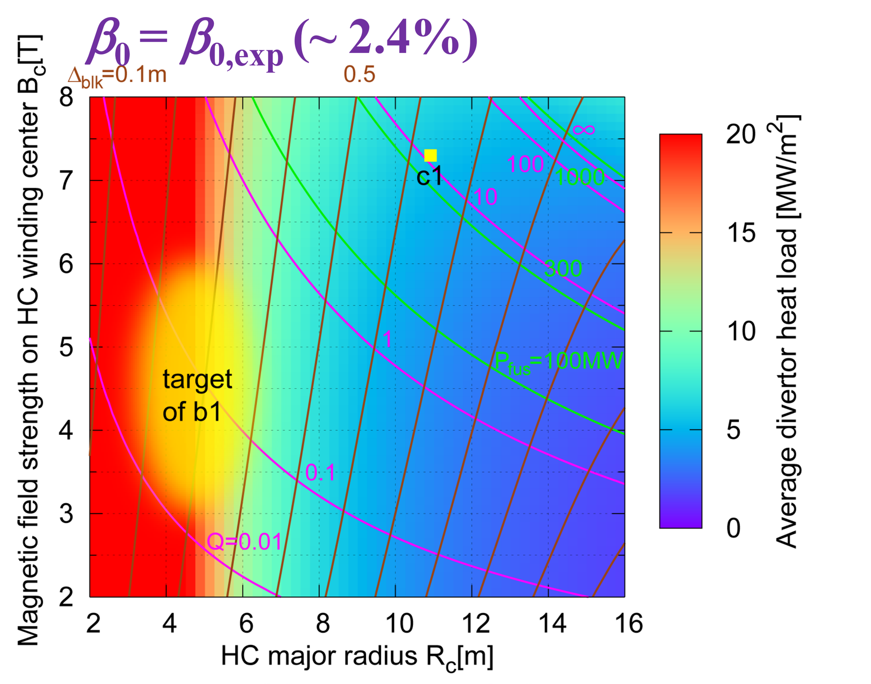


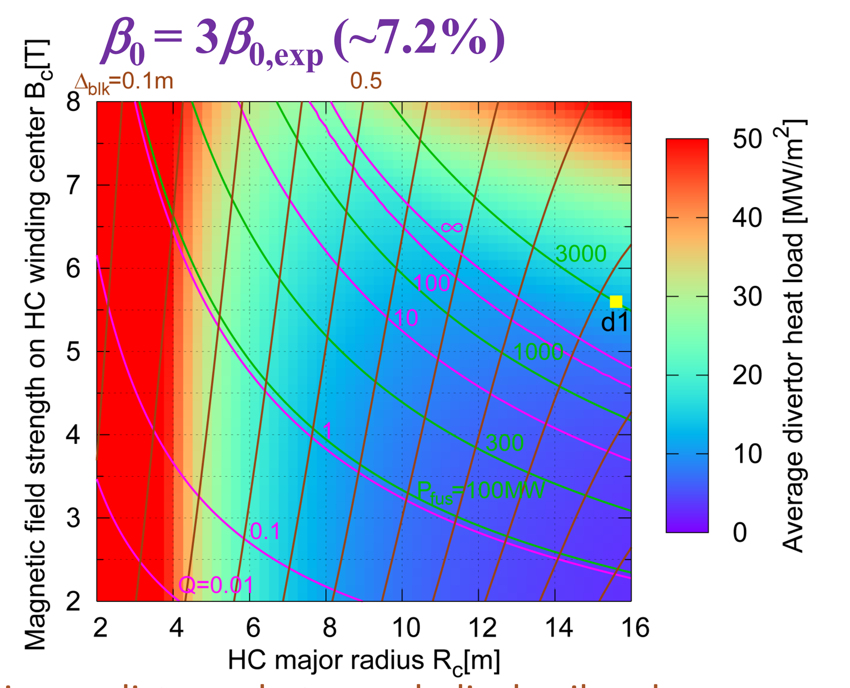


- Following conditions are assumed.
  - High aspect ratio/inward-shifted configuration: ($A_p \sim 7.2$, $R_{\mathrm{ax,vac}}/R_c$ = 3.5/3.9)
  - Constant temperature: T_{e0} = 10 keV
  - Flat density profile: $n_e = \frac{n_{e0}}{1-\alpha}\left[1-\left(\frac{r}{a}\right)^\beta - \alpha\left\{1-\left(\frac{r}{a}\right)^2\right\}\right]$
  - Absorption coefficient alpha heating power : 85%
  - 100% absorption of external heating power in core region (ρ < 0.2)
  - Helium ash fraction: 5%









Contours of fusion gain Q, fusion power P_{fus}, minimum distance between helical coil and plasma Δ_{blk} (space for the blanket module) and average divertor heat load (color contour).

	FFHR-b1 (CTF) *one example	FFHR-c1 (Exp./PROTO)	FFHR-d1 (DEMO)	ITER	JA Tokamak DEMO[5] ** w/ increasing elongation and seeding
R / a [m]	3.9 / 0.58	10.92 / 1.6	15.6 / 2.5	6.2 / 2.0	8.5 / 2.42
B_c [T]	5.7	7.3	~5	5.3	5.94
P_{fus} [MW]	~15 (w/ beam fusion)	380	3000	500	1462 (1694)**
f_{He} / f_{Ar} [%]	5 / 0	5 / 0	5 / 0	<5 / NA	7 / 0.25 (0.6)**
P_{rad} [MW]	~0.1	45	200	~70	82 (177)**
P_{aux} [MW]	~20	30	0	73	84 (96)**
P_{div} (= $P_\alpha + P_{\mathrm{aux}} - P_{\mathrm{rad}}$) [MW]	~23	60	400	~100	294 (258)**
P_{div}/R [MW/m]	~6.0	5.5	35	16	35 (30)**
Γ_{div} [MW/m²]	~12	3.8	13		

Simple linear scaling of divertor wetted area with machine size predicts modest AVERAGED divertor heat load in FFHR (helical DEMO), while non-uniformity of heat load distribution in helical direction is still under investigation.

Summary

- Heliotron configuration may benefit from the wide divertor surface area distributed in helical direction, i.e., in both poloidal and toroidal directions.
- The PEAK divertor heat load, however, can become high, > 20 MW/m² due to the non-uniformity of the magnetic footprint in helical direction.
  - Detached operation or radiation cooling in SOL/div. region with a large fraction (~80%) is necessary if the divertor with a solid target is used.
- Establishment of simple calculation models (a function of primary design parameters) for the following factors is desired to conduct design window analysis from the viewpoint of a consistent divertor design.
  - Divertor heat load profile (or wetted area)
  - Achievable radiation cooling fraction in SOL/div. region compatible with core plasma

References

[1] J. Miyazawa et al., Fusion Eng. Des. **146** (2019) 2233.

[2] T. Goto et al., Nucl. Fusion **51** (2011) 083045.

[3] S. Masuzaki et al.,. Contrib. Plasma Phys. **50** (2010) 629.

[4] J. Miyazawa et al., Nucl. Fusion **54** (2014) 043010

[5] N. Asakura et al., Nucl. Fusion **57** (2017) 126050.