Japan-US Workshop on
Fusion Power Plants Related Advanced Technologies with participants from China and Korea
（Kyoto University, Uji, Japan, 26-28 Feb. 2013）

Core Plasma Design for FFHR-d1 and c1

J. Miyazawa, M. Yokoyama, Y. Suzuki, S. Satake, R. Seki,
Y. Masaoka[1], S. Murakami[1], T. Goto, A. Sagara,
and the FFHR Design Group

National Institute for Fusion Science

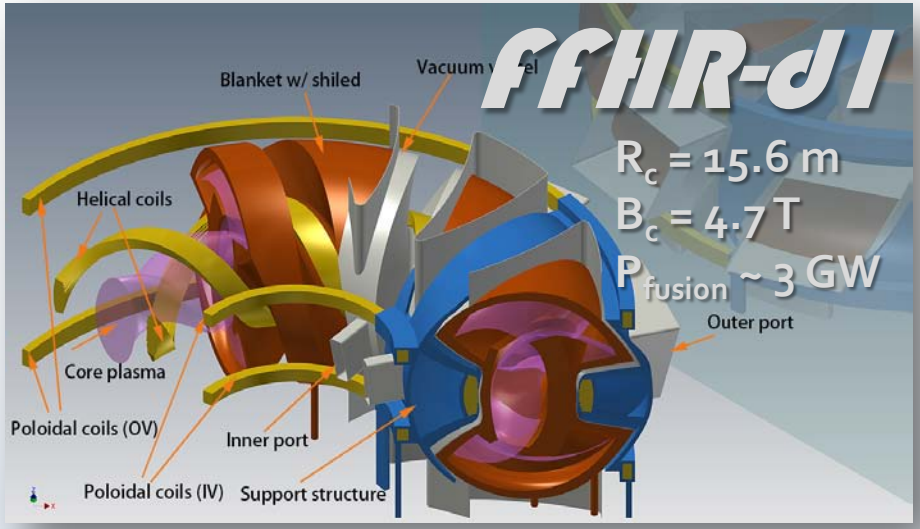
[1)] *Kyoto University*

Outline

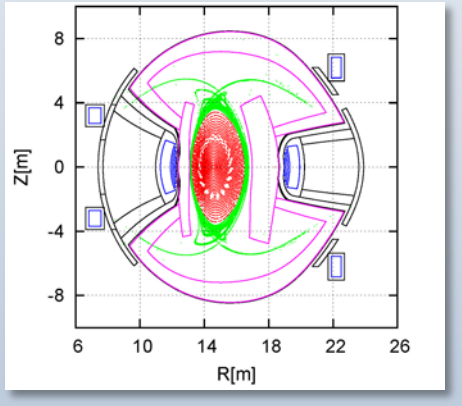
- FFHR-d1
  - concept and specifications
- DPE (Direct Profile Extrapolation) method
  - basic equations
- Detailed physics analysis
  - FFHR-d1 is a semi-optimal Heliotron
  - new reference profiles
- FFHR-c1.0 and c1.1
  - global parameters in FFHR-c1
- Summary

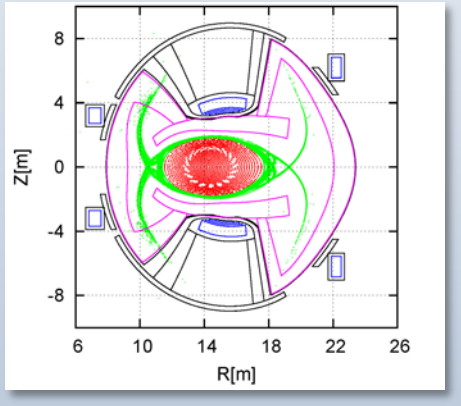
Helical DEMO reactor FFHR-d1

- Conceptual design study of FFHR-d1 has been started in 2010



Schematic view of FFHR-d1 (by H. Tamura)





Vertical slices of FFHR-d1 (by T. Goto)

- The newest version of the FFHR series (FFHR1, FFHR2, FFHR2m1, FFHR2m2)
  - ✓ Heliotron type steady-state reactor
  - ✓ Self-ignition (**no auxiliary heating**)
  - ✓ 3 GW of fusion output
  - ✓ **1.5 MW/m²** of neutron wall load
- What's new in FFHR-d1
  - ✓ **Coil arrangement is similar to LHD**
  - ✓ **Based on LHD normal confinement**
  - ✓ R_c = 15.6 m (LHD x 4), B_c = 4.7 T
  - ✓ 6 poloidal coils in LHD are reduced to 4 in FFHR-d1, to secure **large ports for maintenance**
  - ✓ **Divertors are placed on the backside of blankets**

J. Miyazawa, "Core Plasma Design for FFHR-d1 and c1", Japan-US WS on Fusion Power Plants Related Advanced Technologies (Kyoto Univ., 26-28 Feb. 2013)

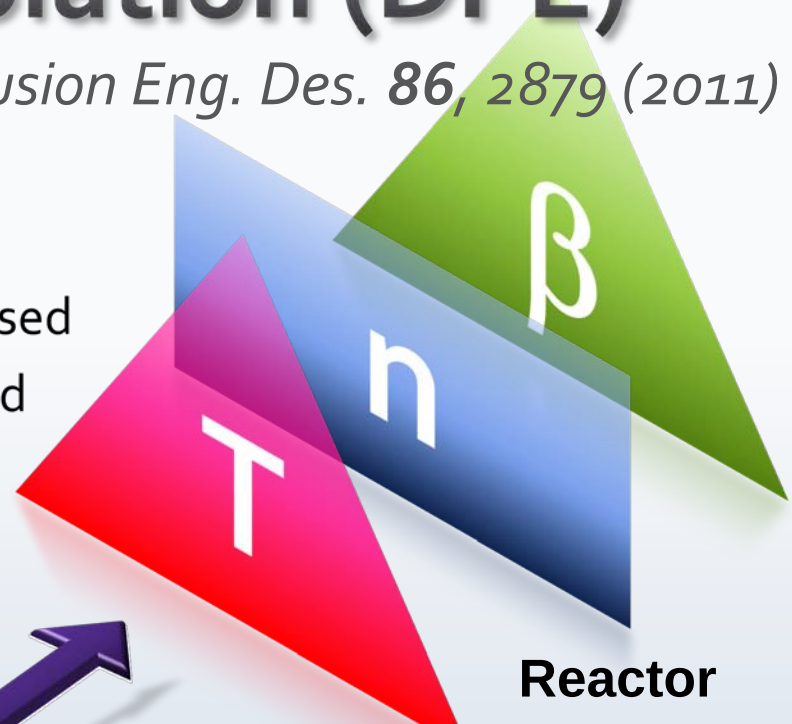
Direct Profile Extrapolation (DPE)

J. Miyazawa et al., Fusion Eng. Des. ***86****, 2879 (2011)*

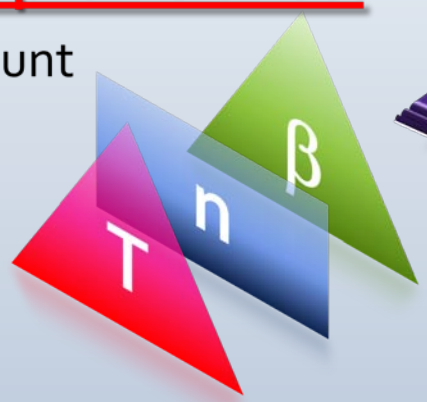
- ✓ Extrapolation without assumed profiles
- ✓ Directly extrapolates the profile data to reactor
- ✓ **MHD equilibrium** similar to the experiment is used
- ✓ **Gyro-Bohm normalized pressure profile** is fixed

$$p_{GB\text{-}BFM}(\rho) = \alpha_0\, n_e(\rho)^{0.6}\, P^{0.4}\, B^{0.8}\, J_0(2.4\rho/\alpha_1)$$

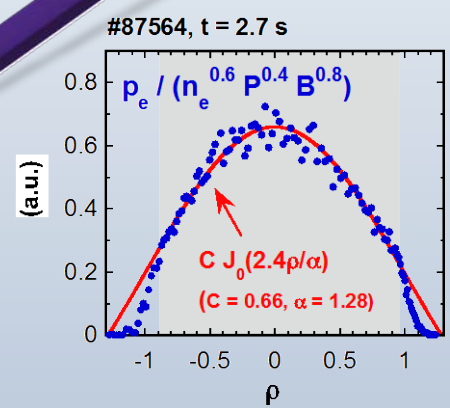
- ✓ Enhancement factors of beta and density are determined to achieve **self-ignition**
- ✓ **The heating profile effect** is also taken into account

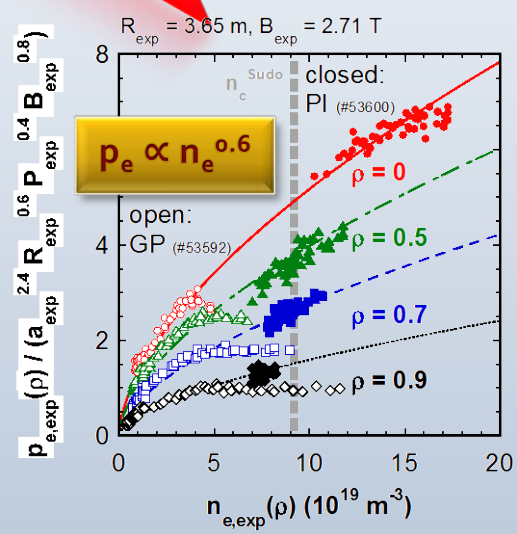


Reactor



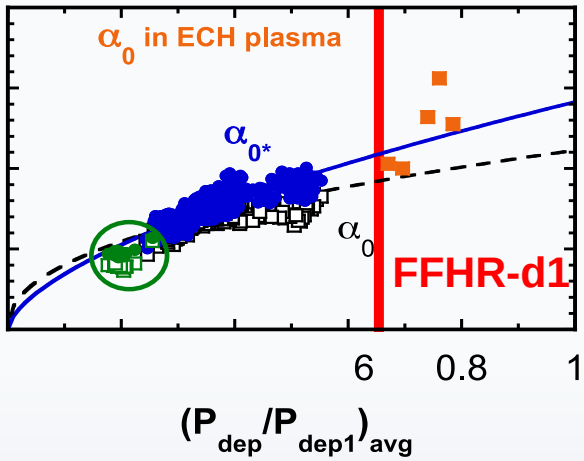
Experiment

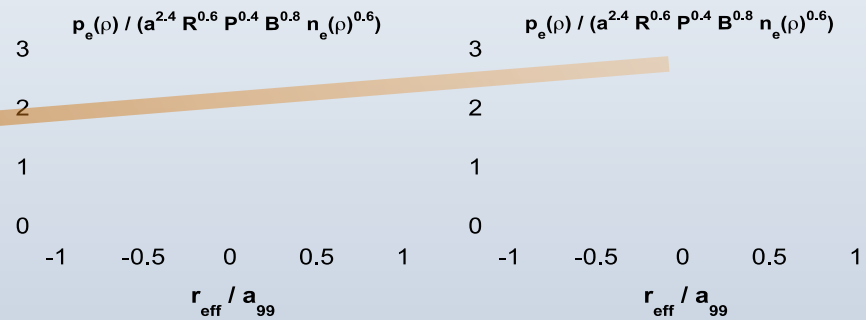




The confinement enhancement factor is expressed as a function of the heating profile

J. Miyazawa et al., Nucl. Fusion ***52*** *(2012) 123007.*





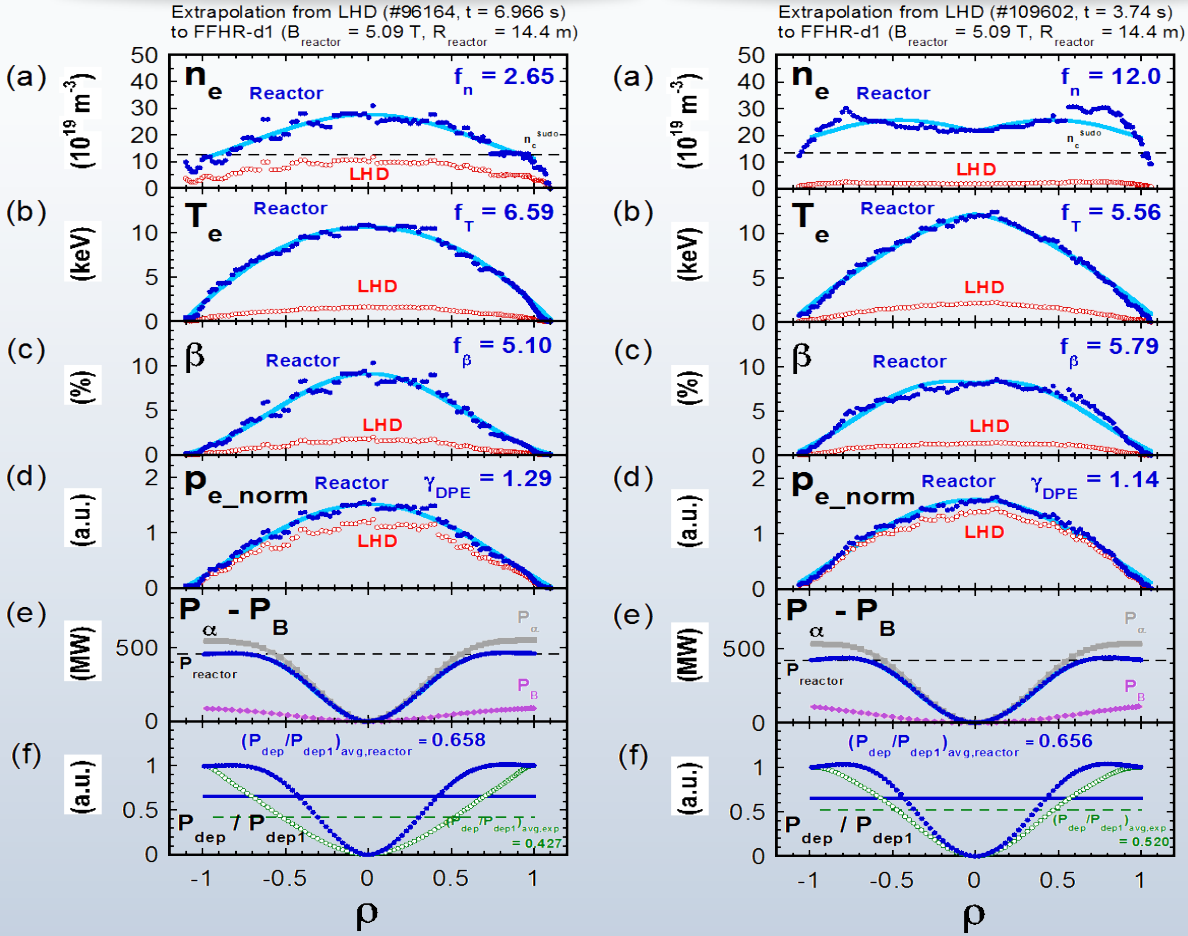
$$\gamma_{DPE} \propto ((P_{dep}/P_{dep1})_{avg,reactor}) / (P_{dep}/P_{dep1})_{avg,exp})^{0.6} = (0.65 / (P_{dep}/P_{dep1})_{avg})^{0.6}$$

...where 0.65 is the assumed peaking factor of the alpha heating in the reactor

Two typical profiles have been chosen

Case A: $\gamma_c = 1.25$

Case B: $\gamma_c = 1.20$



- ✓ One from the standard configuration of R_{ax} = 3.60 m, **$\gamma_c = 1.25$**
- ✓ Another from the high aspect ratio configuration of R_{ax} = 3.60 m, **$\gamma_c = 1.20$**
- ✓ The high-aspect ratio configuration is effective for Shafranov shift mitigation

$\gamma_c = (m\, a_c) / (l\, R_c)$ is the pitch of helical coils, where m = 10, l = 2, a_c ~ 0.9 – 1.0 m, and R_c = 3.9 m in LHD

Detailed physics analyses are being carried out

FFHR-d1 Device Parameters

- device size $R_c = 15.6$ m
- mag. field $B_c = 4.7$ T
- fusion output $P_{DT} \sim 3$ GW
- radial profiles given by DPE

MHD Equilibrium and Stability

- HINT2
- VMEC
- TERPSICHORE

Neoclassical Transport

- GSRAKE/DGN
- DCOM
- FORTEC-3D

α Particle Transport

- GNET
- MORH

Anomalous Transport

- GKV-X

✓ Detailed physics analyses on MHD, neoclassical transport, and alpha particle transport using profiles given by the DPE method have been started

MHD Equilibrium [HINT2]

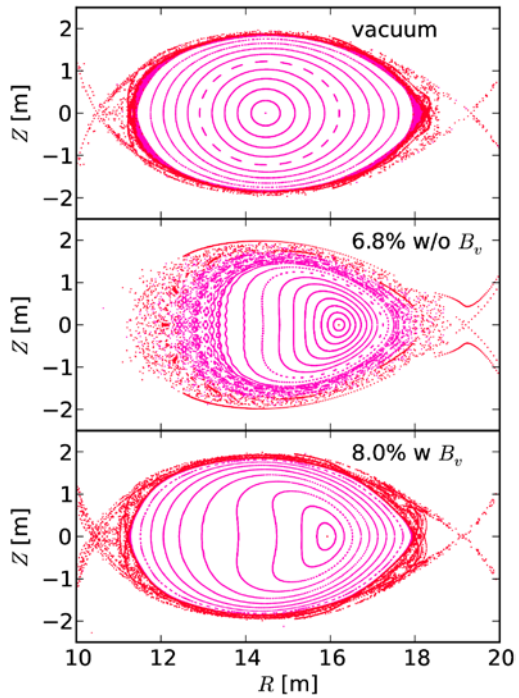
By Y. Suzuki

Case A: γ_c = 1.25

R_{ax} = 14.4 m
"Vacuum"

R_{ax} = 14.4 m
"w/o B_v"

R_{ax} = 14.0 m
"w/ B_v"

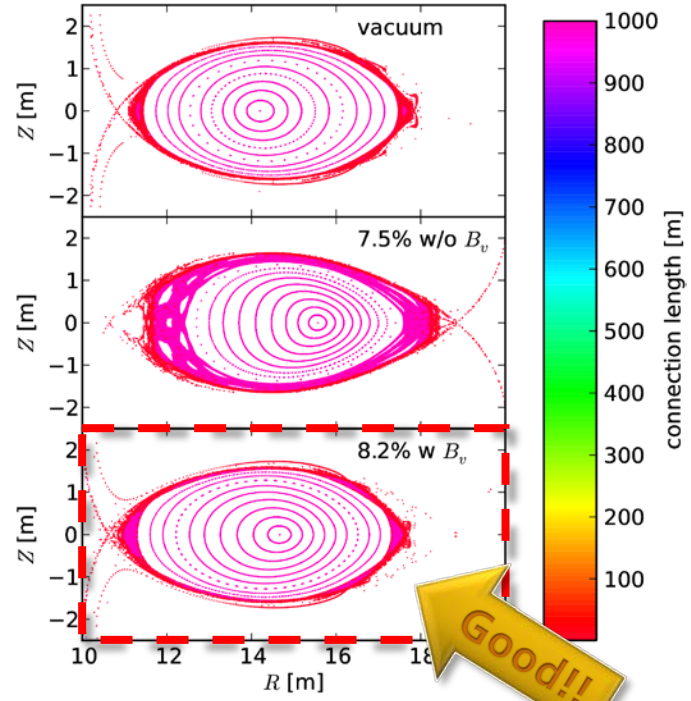


Case B: γ_c = 1.20

R_{ax} = 14.4 m
"Vacuum"

R_{ax} = 14.4 m
"w/o B_v"

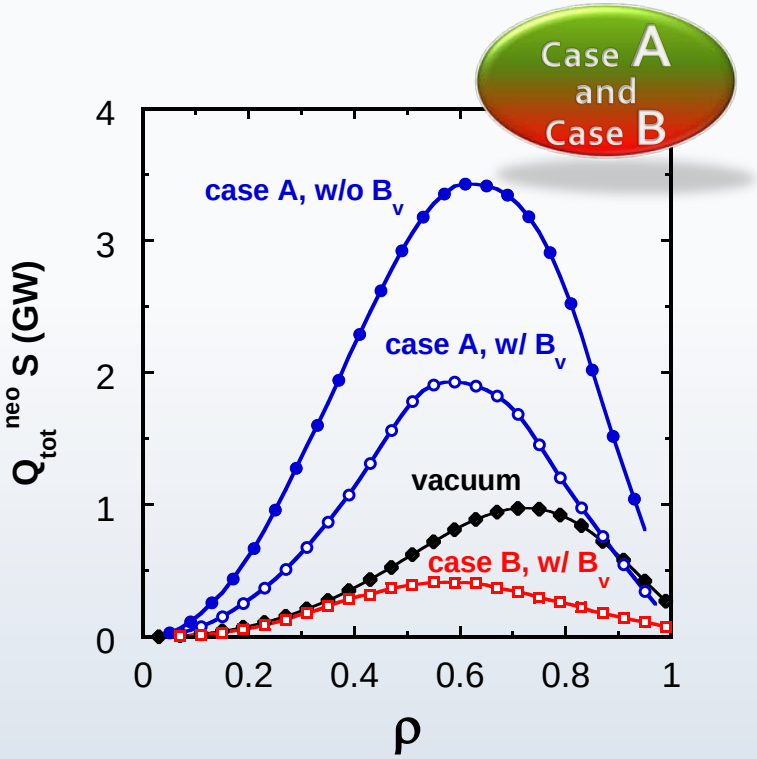
R_{ax} = 14.0 m
"w/ B_v"



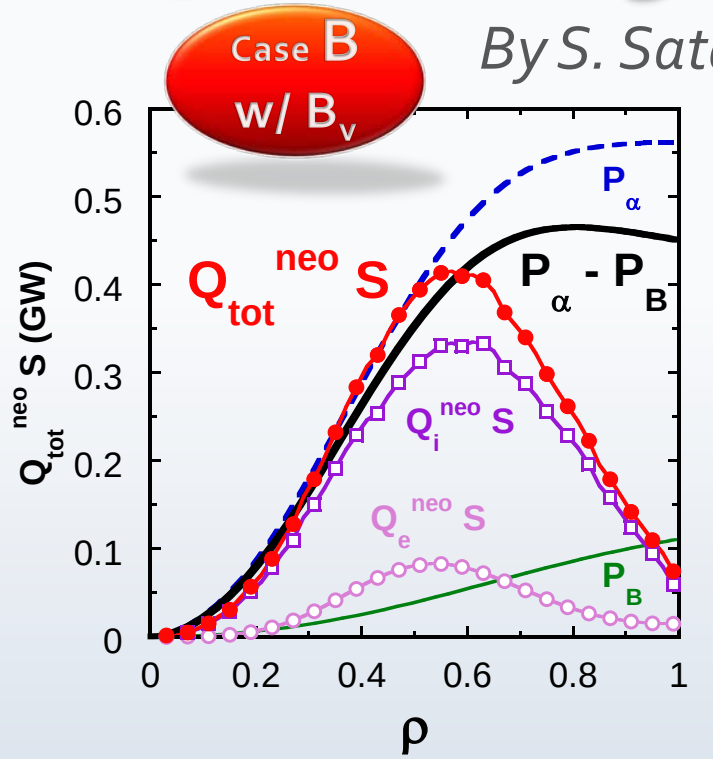
- ✓ Shafranov shift can be mitigated and destructed magnetic surfaces are reformed by plasma position control using vertical magnetic field
- ✓ Especially, magnetic surfaces similar to those in vacuum are formed with finite beta in the high aspect ratio configuration

Neoclassical Transport [FORTEC3D]

By S. Satake



Neoclassical heat flow in various cases



Comparison with P_α in Case B w/ B_v

- ✓ Large neoclassical transport is expected in the case A w/o B_v
- ✓ Plasma position control with B_v is effective and the heat flux is halved
- ✓ Neoclassical heat flux in the Case B with B_v is smaller than the alpha heating power minus the radiation loss of ~ 400 MW at $\rho > 0.6$

➔ Both high aspect ratio config. and plasma position control are effective

Alpha Particle Confinement [GNET]

By Y. Masaoka and S. Murakami (Kyoto Univ.)

- ✓ Heat deposition profile is peaked

  Case A w/ B_v: $(P_{dep}/P_{dep1})_{avg}$ ~ 0.65

  (B_v effect is unclear)

  Case B w/ B_v: $(P_{dep}/P_{dep1})_{avg}$ ~ 0.51

- ✓ Significant loss of alpha particle due to the large Shafranov shit

  Energy loss ratio:

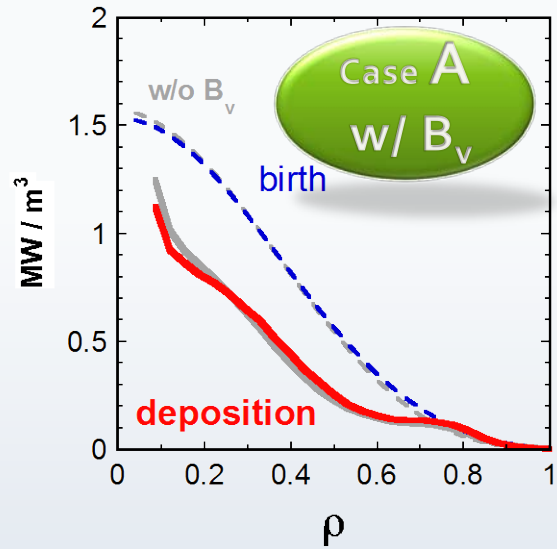
  Case A w/ B_v: 41 % ➔ Case B w/ B_v: 11 %

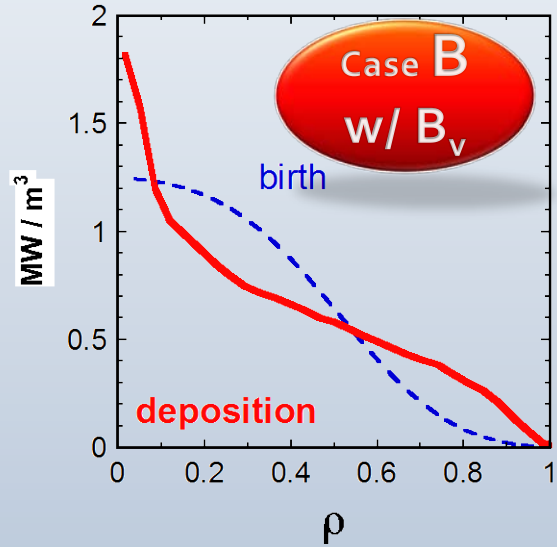
  Particle loss ratio:

  Case A w/ B_v: 69 % ➔ Case B w/ B_v: 20 %

Note: *Reentering alpha is not considered*

(α particles are lost at LCFS in GNET)

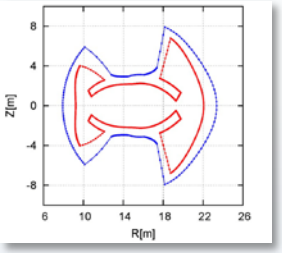


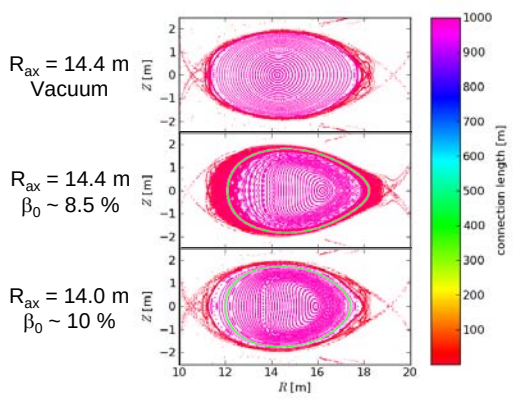


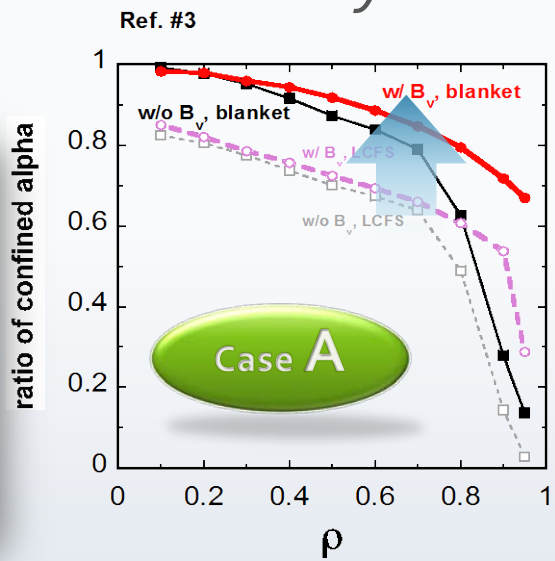
3D Tracking of Alpha Particles [MORH]

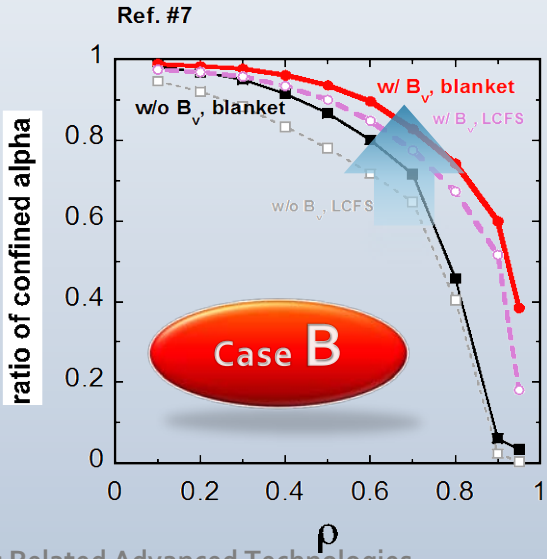
By R. Seki

- ✓ Three dimensional orbit tracing by MORH using MHD equilibriums given by HINT2
  - E_o = 3.5 MeV
  - ρ = 0.1 – 0.9
  - ϕ = 0° – 18°
  - pitch angle = (1 – 19) π/20
  - w/o CX loss
  - Vacuum vessel or blanket is used as the loss boundary (**no significant difference is recognized**)









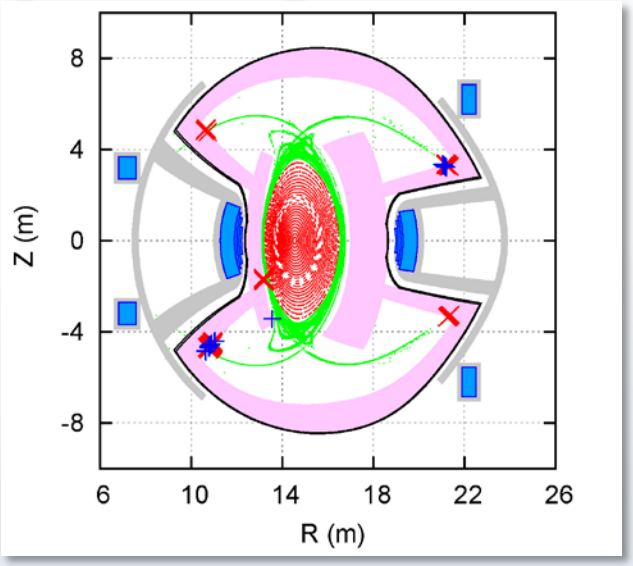
- ✓ **Plasma position control is effective**
- ✓ **Reentering effect is also effective**
- ✓ **Direct loss of alpha particles generated at ρ < 0.7 is less than 20 %**

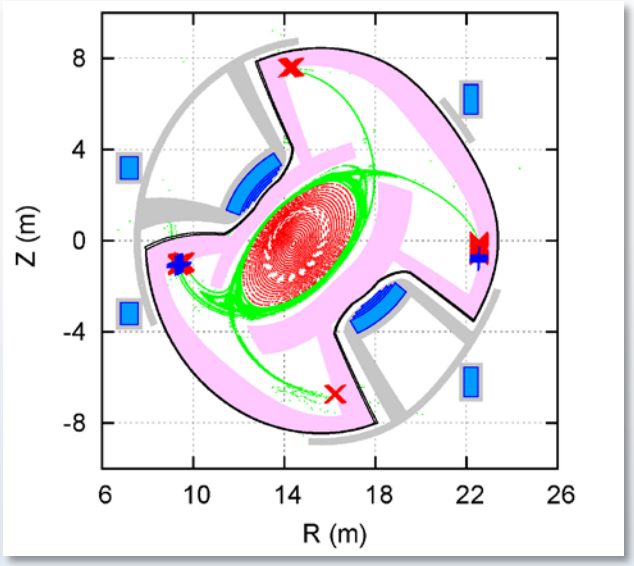
J. Miyazawa, "Core Plasma Design for FFHR-d1 and c1", Japan-US WS on Fusion Power Plants Related Advanced Technologies (Kyoto Univ., 26-28 Feb. 2013)

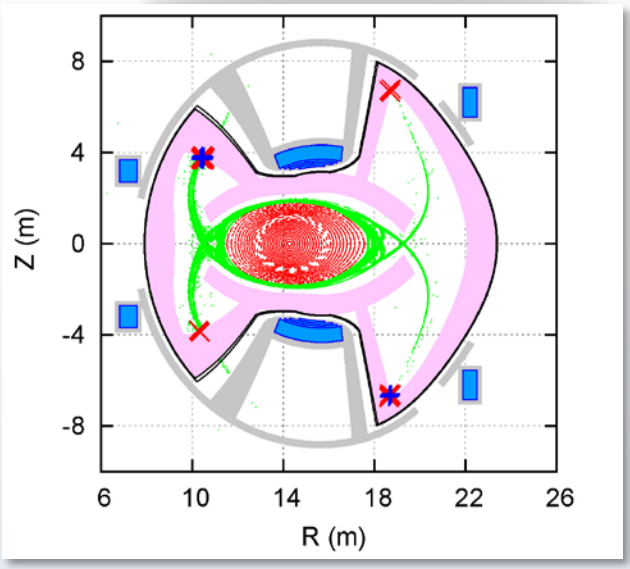
Lost Alpha Particles Hit the Divertor [MORH]

By R. Seki and T. Goto

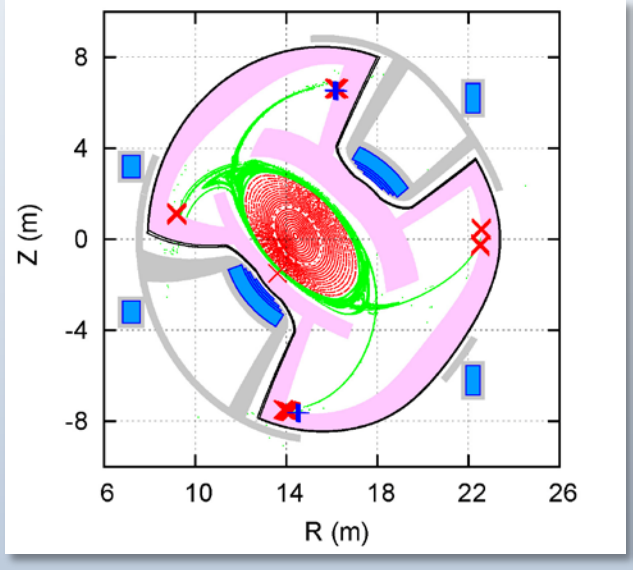
Case A: $\gamma_c = 1.25$
w/o or w/ B_v







- ✓ Lost α particles go to the divertor region
- ✓ Negligibly small number of α particles starting from ρ = 0.95 hit the blanket



Case A
R_{ax} = 3.60 m
γ_c = 1.254
n_{eo}/n_{ebar} ~ 1.5

→ MHD equilibrium: Shafranov shift is too large

→ Neoclassical → ✖; α confinement → ✖

Both NC and α confinement are **bad** in the Case A, due to the large Shafranov shift.

Case B
R_{ax} = 3.60 m
γ_c = 1.20
n_{eo}/n_{ebar} ~ 0.9

→ MHD equilibrium: Shafranov shift is mitigated by applying B_v

→ Neoclassical → ○; α confinement → ○

→ α heating profile → ✖ (hollow)

Both NC and α confinement are **good** in the Case B where Shafranov shift is mitigated. However, **α heating profile is hollow** (not consistent with the assumption for γ_{DPE}).

Case C
R_{ax} = 3.55 m
γ_c = 1.20
n_{eo}/n_{ebar} ~ 1.5

→ MHD equilibrium: ?

→ Neoclassical → ?; α confinement → ?

→ α heating profile → ?

A new profile "**Case C**" has been chosen.
Peaked density profile as the Case A.
High-aspect ratio as the Case B but **more inward-shifted.**

Case C′
R_{ax} = 3.55 m
γ_c = 1.20
n_{eo}/n_{ebar} ~ 1.5

→ MHD equilibrium: ?

→ Neoclassical → ?; α confinement → ?

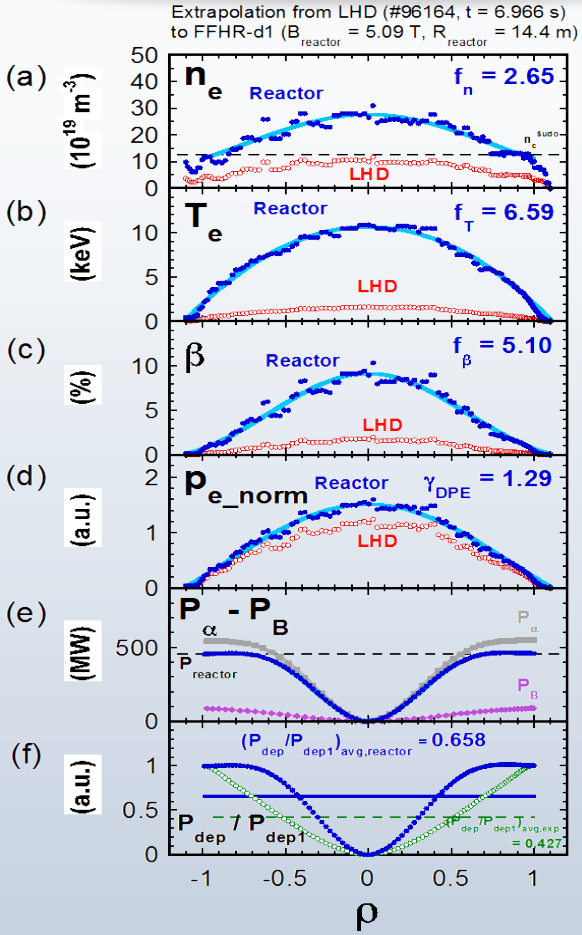
→ α heating profile → ?

→ α heating efficiency → ?

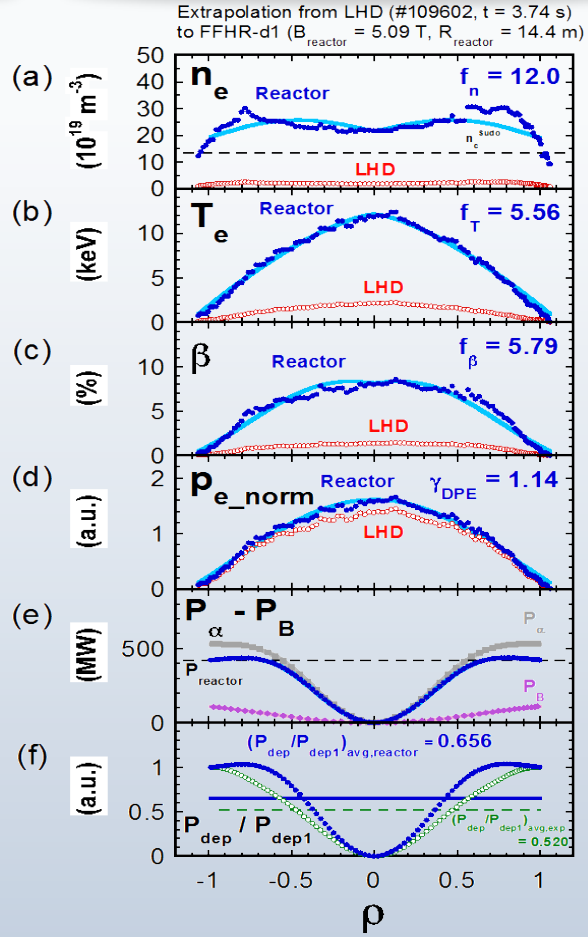
In the Case C′, α heating efficiency, η_α = 0.85 is assumed (η_α = 1.0 in Cases A, B, and C).

Profiles in the three Cases

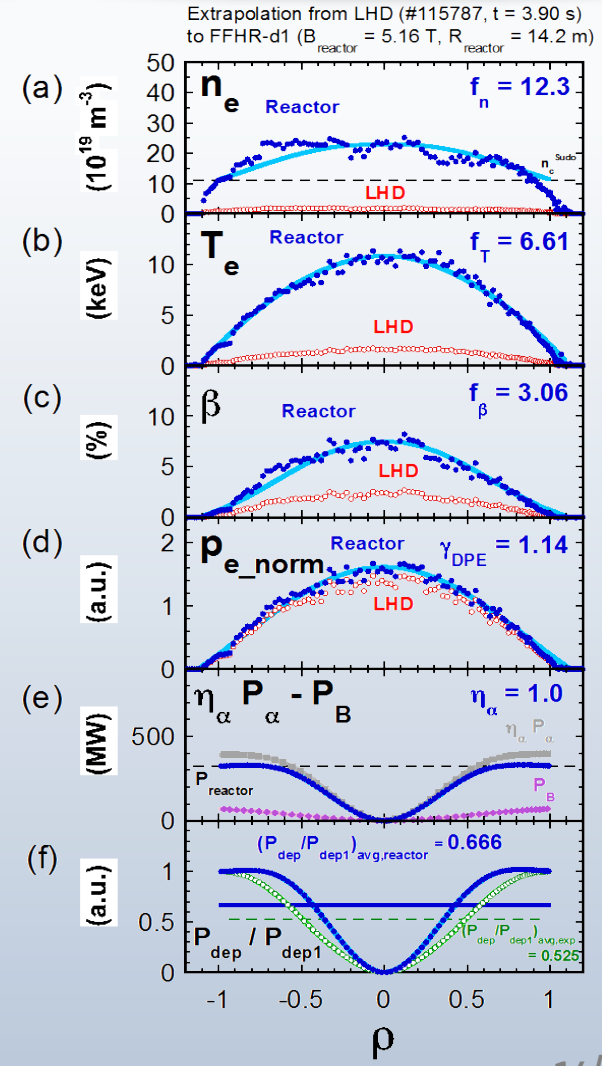
Case A: $\gamma_c = 1.25$



Case B: $\gamma_c = 1.20$



Case C: $\gamma_c = 1.20$

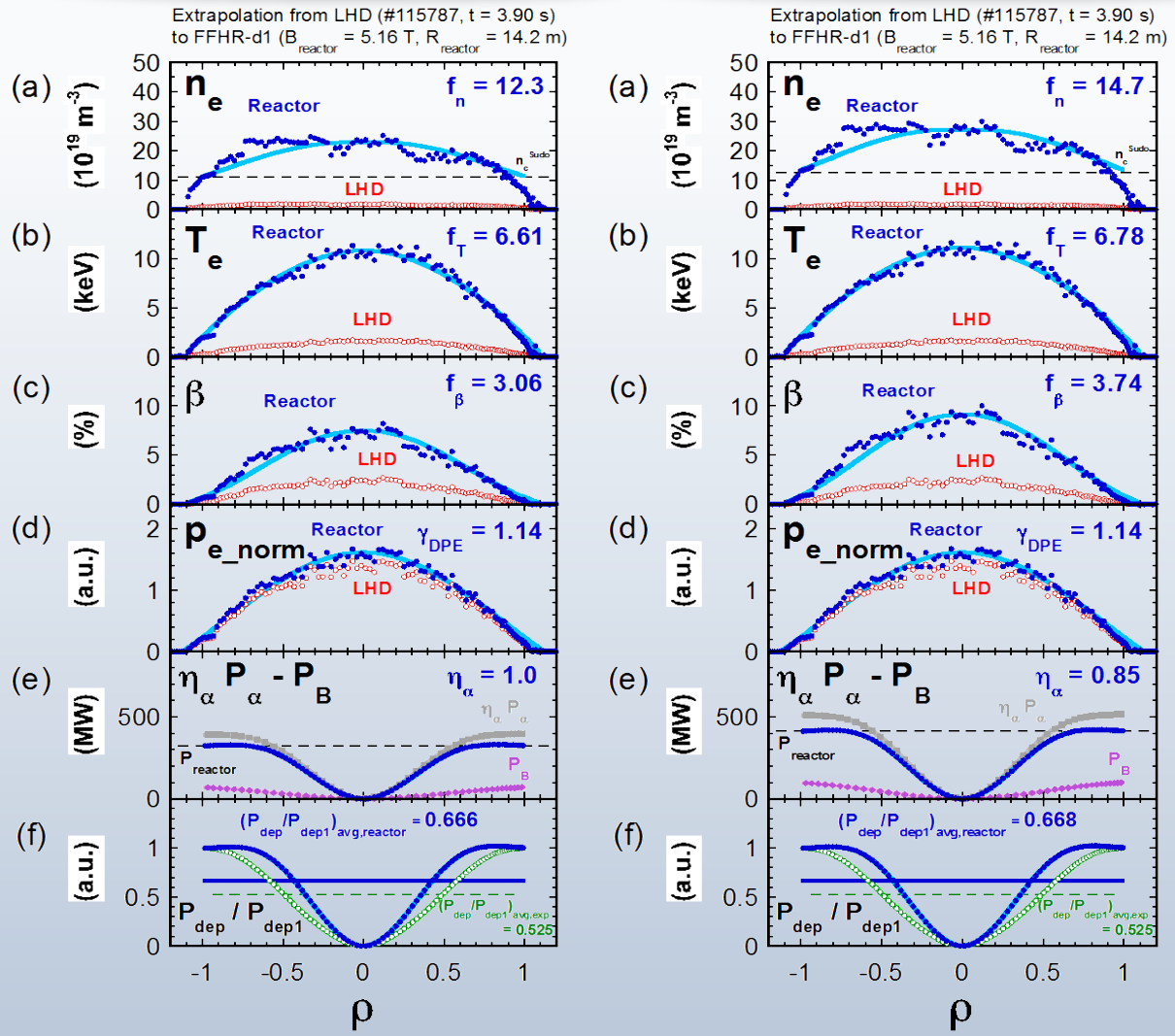


Profiles in the Cases C and C′

Case C: η_α = 1.0

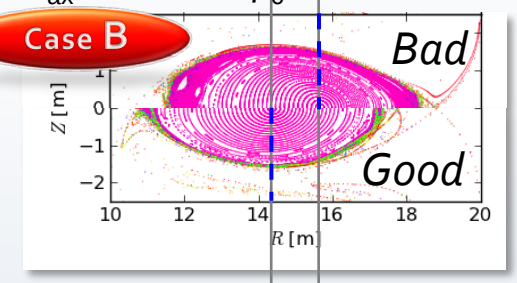
Case C′: η_α = 0.85

MHD equilibrium is ready



R_{ax} = 14.4 m, β_0 ~ 7.5 %, **"w/o B_v"**

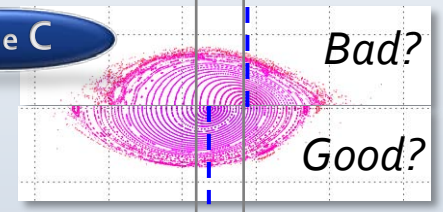
Case B



R_{ax} = 14.0 m, β_0 ~ 8.2 %, **"w/ B_v"**

R_{ax} = 14.2 m, β_0 ~ 7.6 %, **"w/o B_v"**

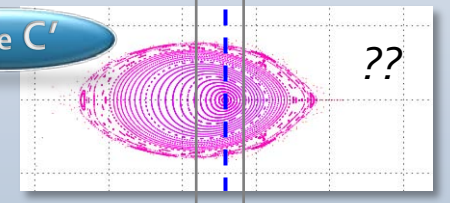
Case C



R_{ax} = 14.0 m, β_0 ~ 7.6 %, **"w/ B_v"**

R_{ax} = 14.0 m, β_0 ~ 9.1 %, **"w/ B_v"**

Case C′



The basic concept of a helical experiment reactor FFHR-c1

- ✓ FFHR-c1 is a small copy of FFHR-d1 ("c" means "before d" and "compact")

Objectives of FFHR-c1:

1. "Q > 5" (c1.0) or "Q = ∞" (c1.1)

   $Q = P_{fusion} / P_{aux} = 5\, P_{\alpha} / P_{aux}$
2. More than 200 MW of P_{fusion}
3. Safe and steady-state operation for one year
4. Tritium self-sufficiency (local TBR > 1.3)
5. Neutron irradiation test of large components

2 types of FFHR-c1 (conservative (c1.0) or challenging (c1.1))

Device Name	LHD	FFHR-c1.0	FFHR-c1.1	FFHR-d1
Target	Experiment	Q ~ 7	Self-ignition	Self-ignition
R_c helical coil major radius	3.9 m	13.0 m (10/3 times LHD)	←	15.6 m (4 times LHD)
V_p plasma volume	~30 m³	~1,000 m³ (similar to ITER)	←	~2,000 m³
B_c magnetic field strength at the helical coil center	2.5 T	4.0 T (design value of LHD) NbTiTa (He II) / HTS	5.3 T (similar to ITER) Nb_3Sn / Nb_3Al / HTS	4.7 T Nb_3Sn / Nb_3Al / HTS
W_{mag} stored magnetic energy	1.6 GJ (at 4 T)	67 GJ	113 GJ	160 GJ
P_{aux} auxiliary heating power	25 MW (short pulse)	?	?	50 MW - 1 hour (for start-up)
P_{fusion} fusion power	–	?	?	~3 GW (Q = ∞)
$\tau_{duration}$ duration time of a shot	~1 hour	?	?	~1 year
Φ_n dpa per shot	–	?	?	~15 dpa (1.5 MW/m² x 1 year)
Issues	*DD exp.*	?	?	*large HC winding* *react & wind* *nuclear heating on SC* *lifetime of SC/insulator*

Effect of central heating on γ_{DPE*}

- The confinement enhancement factor is proportional to the 0.6 power of the heating profile peaking factor:

$$\gamma_{DPE} = ((P_{dep}/P_{dep1})_{avg,reactor}) / (P_{dep}/P_{dep1})_{avg,exp})^{0.6}$$

$$= (0.65 / (P_{dep}/P_{dep1})_{avg})^{0.6}$$

- The heating profile peaking factor with the central auxiliary heating can be larger than 0.65

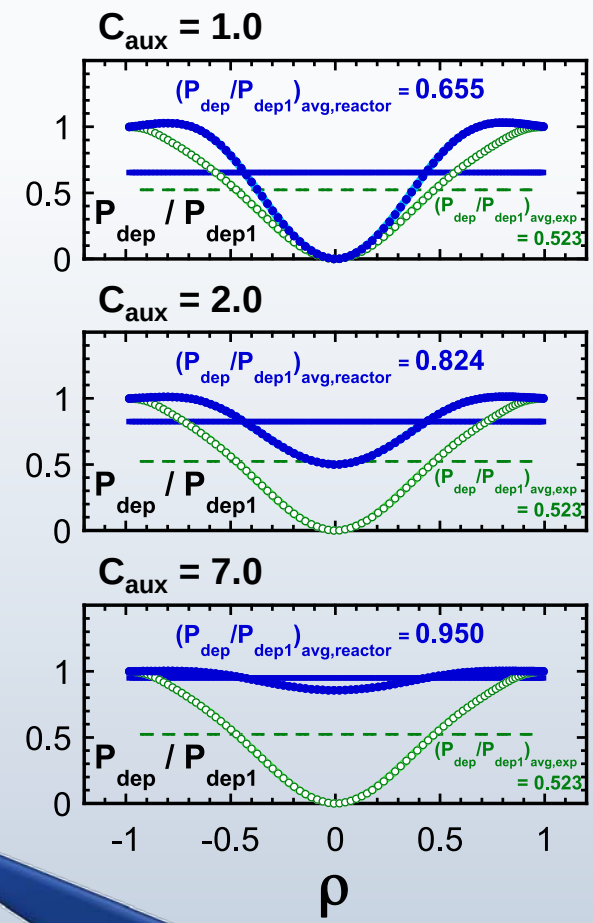
- Assume that the auxiliary heating is focused on the center and expressed by the delta function, i.e.,

$$(P_{dep}/P_{dep1})_{avg,reactor} = 0.65\,(1/C_{aux}) + (1.0 - 1/C_{aux})$$

$$= 1.0 - 0.35/C_{aux}$$

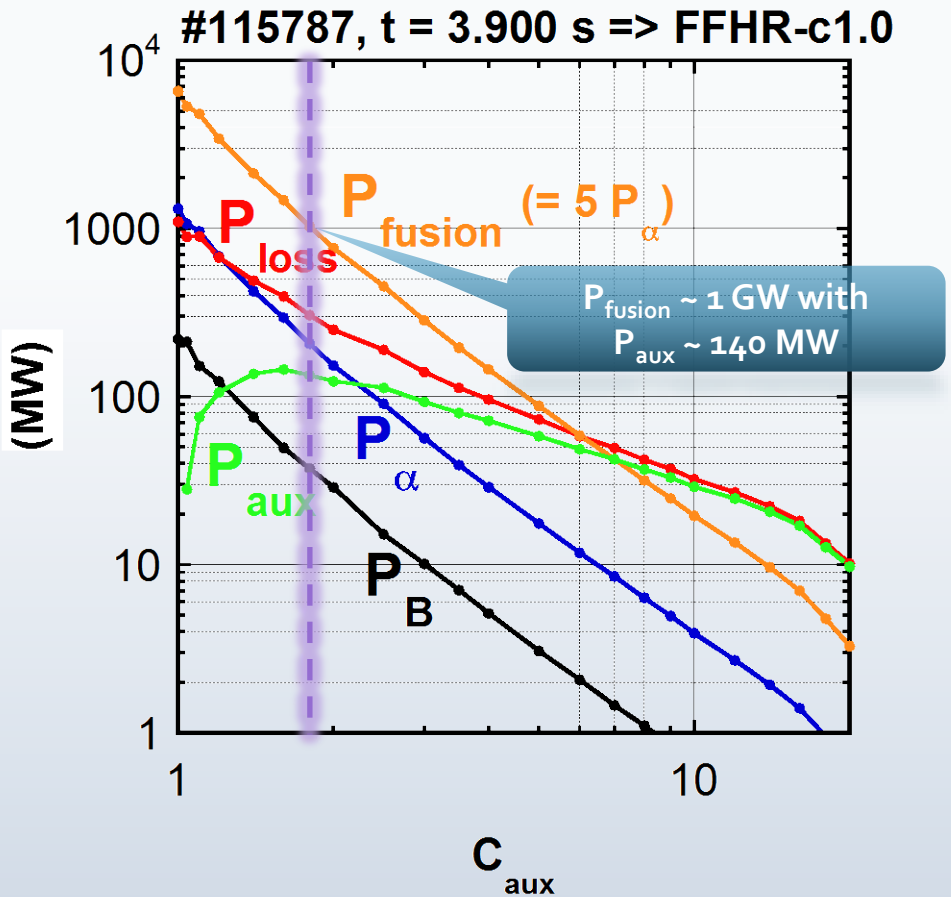
- The confinement enhancement factor is given by

$$\gamma_{DPE*} = ((1.0 - 0.35/C_{aux}) / (P_{dep}/P_{dep1})_{avg})^{0.6}$$

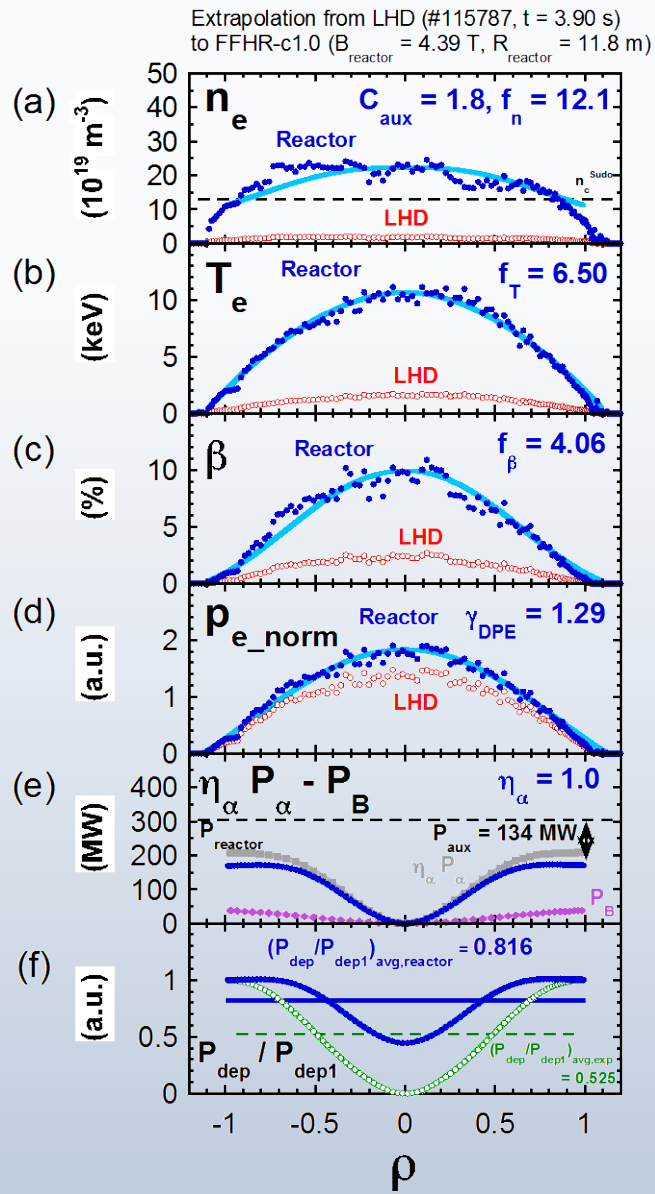


$P_\alpha - P_B = (1 / C_{aux})\, P_{reactor}$
$P_{aux} = (1 - 1 / C_{aux})\, P_{reactor}$

Powers in FFHR-c1.0



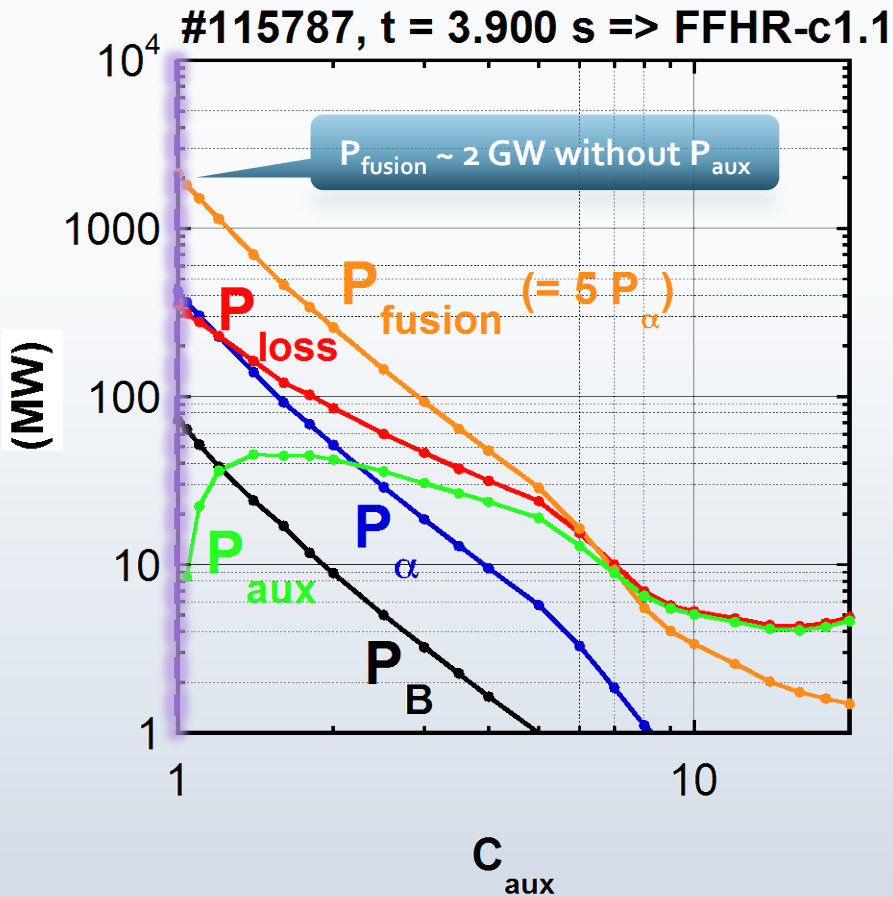
FFHR-c1.0, C_{aux} = 1.8



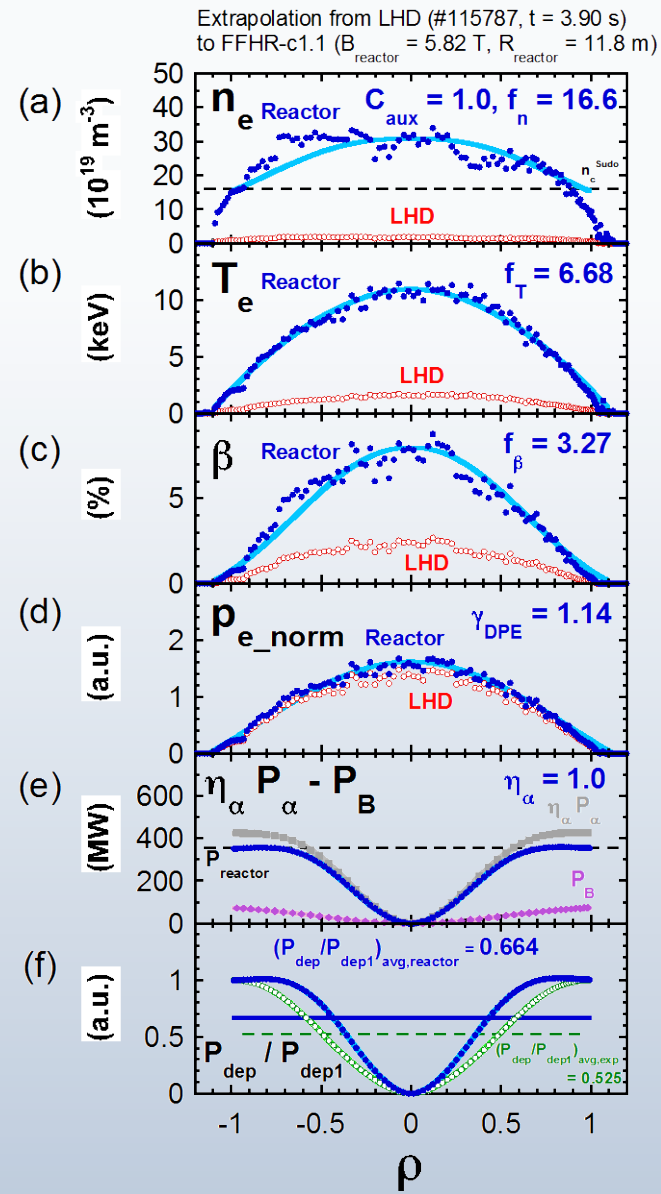
FFHR-c1.0 (R_c = 13.0 m, B_c = 4.0 T)

✓ **"Q ~ 7"** (at C_{aux} ~ 1.8) with P_{aux} ~ 140 MW and P_{fusion} ~ 1 GW

Powers in FFHR-c1.1



FFHR-c1.1, C_{aux} = 1.0



FFHR-c1.1 (R_c = 13.0 m, B_c = 5.3 T)

- ✓ **"Self-ignition"** (P_{aux} = 0 (Q = ∞) at C_{aux} = 1) with P_{fusion} ~ 2 GW
- ✓ P_{aux} ~ 50 MW is enough

2 types of FFHR-c1 (conservative (c1.0) or challenging (c1.1))

Device Name	LHD	FFHR-c1.0	FFHR-c1.1	FFHR-d1
Target	Experiment	Q ~ 7	Self-ignition	Self-ignition
R_c helical coil major radius	3.9 m	13.0 m (10/3 times LHD)	←	15.6 m (4 times LHD)
V_p plasma volume	~30 m^3	~1,000 m^3 (similar to ITER)	←	~2,000 m^3
B_c magnetic field strength at the helical coil center	2.5 T	4.0 T (design value of LHD) NbTiTa (He II) / HTS	5.3 T (similar to ITER) Nb_3Sn / Nb_3Al / HTS	4.7 T Nb_3Sn / Nb_3Al / HTS
W_{mag} stored magnetic energy	1.6 GJ (at 4 T)	68 GJ	113 GJ	160 GJ
P_{aux} auxiliary heating power	25 MW (short pulse)	140 MW - CW (for sustainment)	50 MW - 1 hour (for start-up)	50 MW - 1 hour (for start-up)
P_{fusion} fusion power	–	~1 GW (Q > 7)	~2 GW (Q = ∞)	~3 GW (Q = ∞)
$\tau_{duration}$ duration time of a shot	~1 hour	~1 year	~5 month	~1 year
Φ_n dpa per shot	–	~8 dpa (0.8 MW/m^2 x 1 year)	~7 dpa (10 dpa at peak) (1.7 MW/m^2 x 5 month)	~15 dpa (1.5 MW/m^2 x 1 year)
Issues	*DD exp.*	***large HC winding*** *R&D of (NbTiTa & He II cooling) or HTS*	***large HC winding*** *react & wind* ***nuclear heating on SC*** ***lifetime of SC/insulator***	***large HC winding*** *react & wind* ***nuclear heating on SC*** ***lifetime of SC/insulator***

Summary

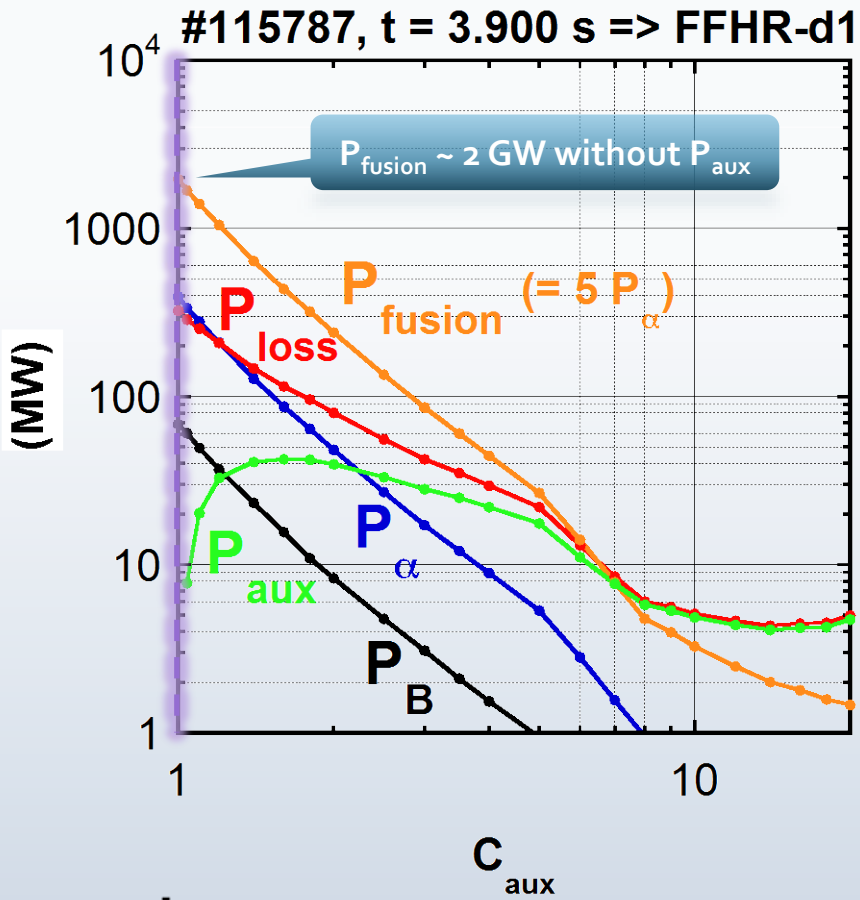
✓FFHR-d1 is a "sub-optimal" heliotron

- R_c = 14.6 m (4 times LHD), B_c = 4.7 T
- Shafranov shift is effectively suppressed
- neoclassical transport compatible with the alpha heating
- small alpha energy loss of ~10 %
- a new profiles (Cases C and C') are being analyzed

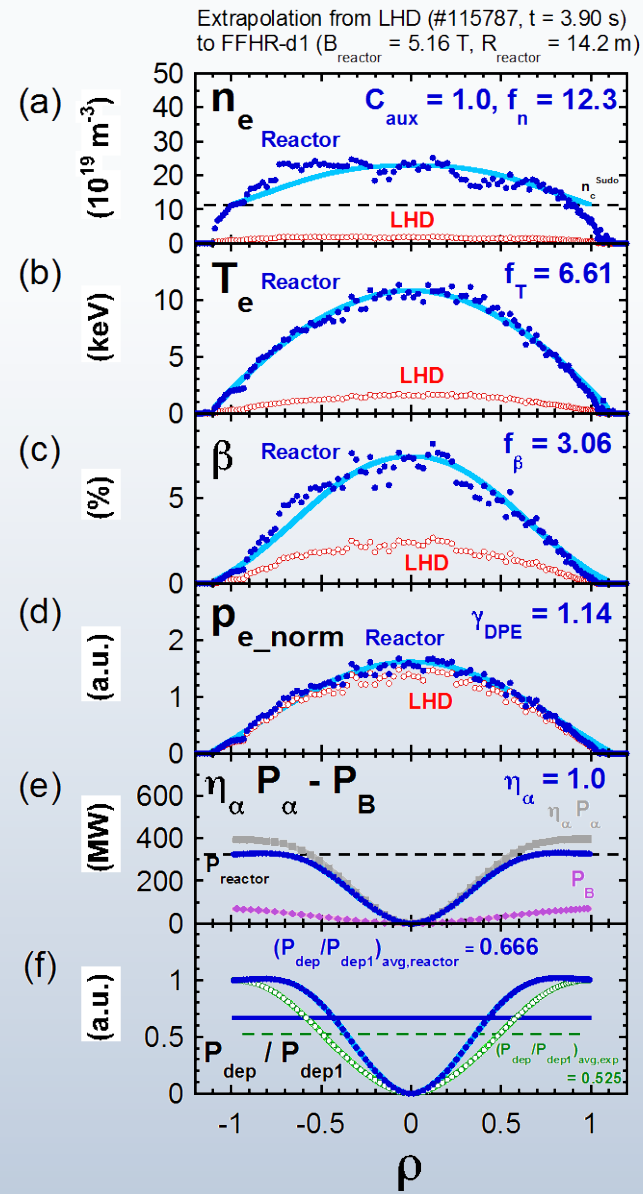
✓FFHR-c1 is as attractive as FFHR-d1

- R_c = 13.0 m (10/3 times LHD)
- "Q > 7" with B_c = 4.0 T, P_{aux} ~140 MW. and P_{fusion} ~ 1 GW
- "self-ignition" with B_c = 5.3 T and P_{fusion} ~ 2 GW
- component test up to ~10 dpa will be possible

Powers in FFHR-d1



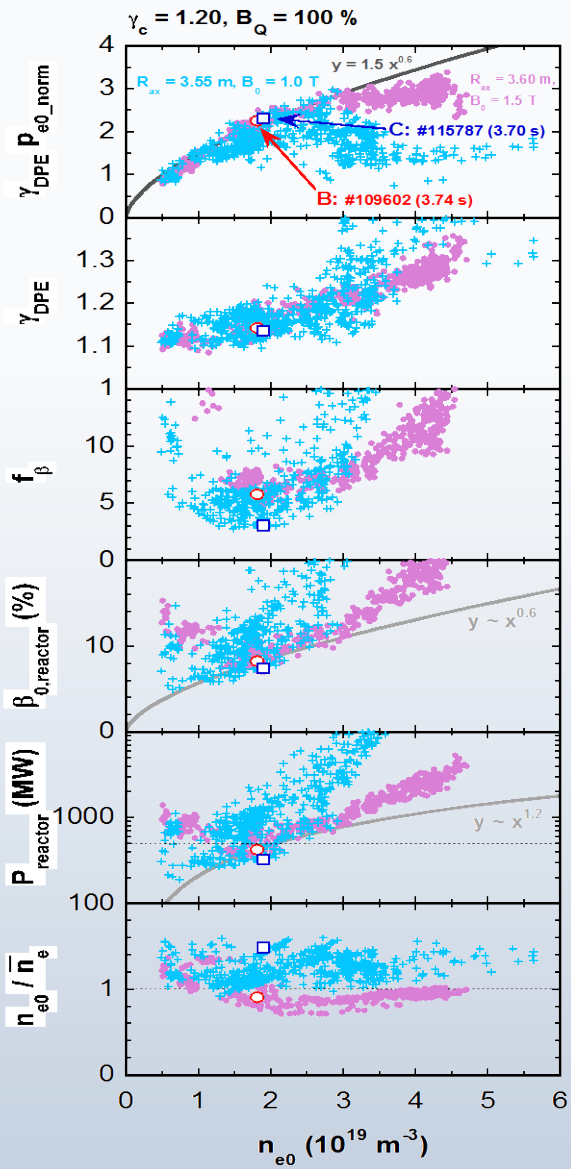
FFHR-d1, C_{aux} = 1.0



FFHR-d1 (R_c = 15.6 m, B_c = 4.7 T)

- ✓ **"Self-ignition"** (P_{aux} = 0 (Q = ∞) at C_{aux} = 1) with P_{fusion} ~ 2 GW (P_{fusion} ~ 3 GW is also achievable)
- ✓ P_{aux} ~ 50 MW is enough

f_β is low at R_{ax} = 3.55 m and γ_c = 1.20



- ✓ The central pressures similar to those in the γ_c = 1.254 configuration have also been achieved in the γ_c = 1.20 configuration, in spite of smaller plasma volume (i.e., better confinement in γ_c = 1.20)
- ✓ f_β as low as ~3 has been achieved
- ✓ $P_{reactor}$ can be as low as ~200 MW
- ✓ Density peaking factor is high (density peaking was observed after NB#1 break down)