Development of power handling and plasma modelling for SlimCS DEMO divertor

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Contents

1. Introduction: power handling in DEMO reactor

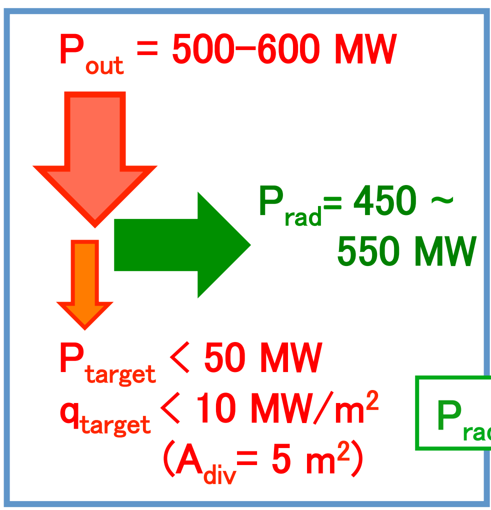
- **Power handling by plasma operation, divertor design and target engineering is the most important issue for the reactor design.**

"***SlimCS***" aims P_{fus} ≤3 GW (P_{heat}= 600~700 MW) with reduced-size CS (R = 5.5m and A=2.6) ⇒ Power exhausting to SOL is 5-6 times larger and R is smaller than ITER.

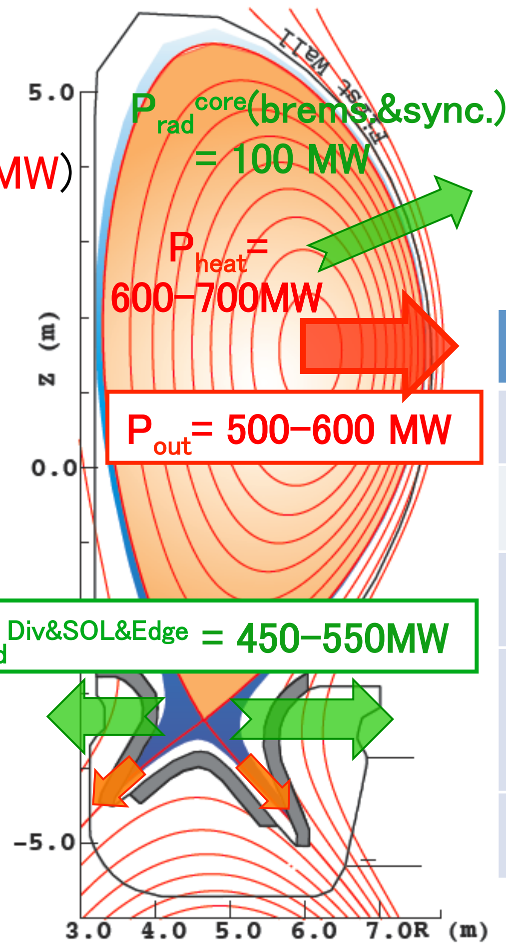
SlimCS

P_{fusion} = 2.95 GW

(P_{heat}=P_{α}+P_{ax}=600~700 MW)



Major radius : R_p=5.5 m
Minor radius : a_p=2.1 m
Plasma current : I_p=16.7 MA
Toroidal field : B_t=6.0 T
Plasma volume: V_p=941 m^3



Power handling factor (P/R) is 6-7 times larger than ITER

⇒ **Radiation loss in edge/divertor is required 10 times larger than ITER**

Power handle	SlimCS	ITER
P_{heat} (α + add.) [MW]	650	150
P_{out}= P_{heat} –P_{rad}^{core} [MW]	550	100
P_{out}/R_p [MW/m]	100	16
P_{div} (=P_{out}–$P_{rad}^{Div/SOL/Edge}$) [MW]	< 50	~50
⇒$P_{rad}^{Div/SOL/Edge}$ [MW]	~500	~50

Extension of ITER divertor concept for the DEMO divertor

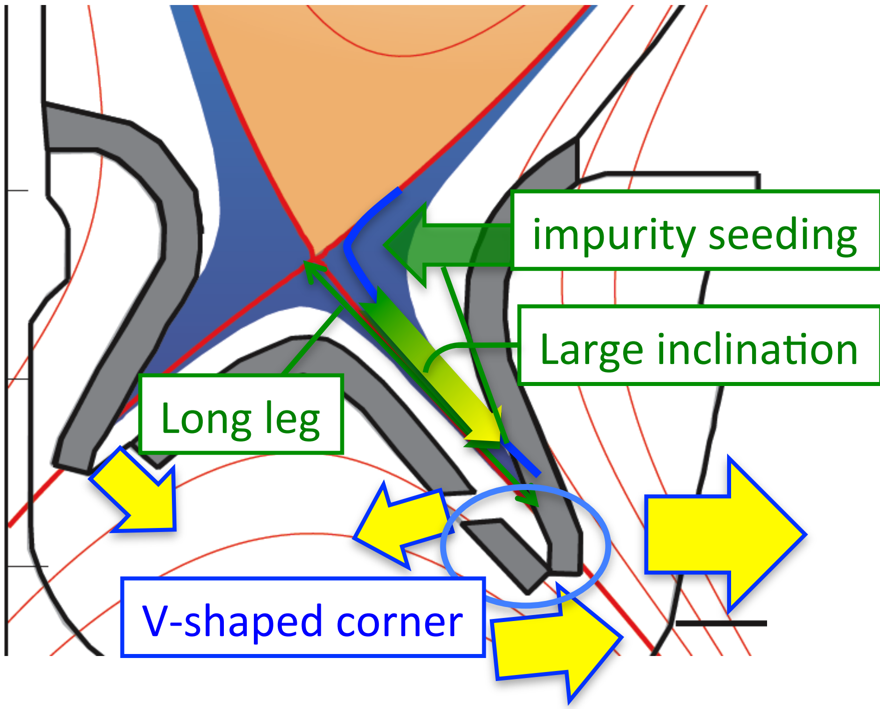
Design concept for ITER divertor is applied and extended to SlimCS divertor:
"divertor detachment" **(T_e~ a few eV) is a key for the power handling.**

Flux expansion is smaller than ITER due to large separation of Div-coil from plasma ⇒

(1) Divertor leg and **inclination of the target** are larger than ITER
⇒increase radiation and recombination upstream of the target ⇒ reducing q^{target}.

(2) V-shaped corner ⇒ enhance recycling near the strike-point.

(3) Impurity seeding such as Ne, N_2, Ar, Kr, Xe ⇒ enhance edge & divertor radiation



For the first design, effective wet area is increased comparable to ITER:

geometry parameters	SlimCS	ITER
Flux expansion (outer)	3	6
L_{sp} (outer leg length)	1.83m	1.14m
θ_{sp} (outer inclination angle)	18°	25°
V-shaped corner	Out*	In & Out
Outer wet area, A_{wet}, for λ_q^{mid} = 5mm	2.2 m^2	2.1 m^2

* Inner divertor is detached without V-corner

2. Recent studies of power handling for demo divertor

Divertor concepts of the power handling for DEMO reactor have been investigated:

(1) Power exhaust have been studied in the "conventional divertor" , extending from the ITER divertor design, i.e. increase radiation power and detachment:
modelling and divertor design, has been progressed using SONIC
⇒ Key parameters for plasma transport modelling such as diffusion coefficient
Effects of the divertor geometry such as "Long leg divertor"

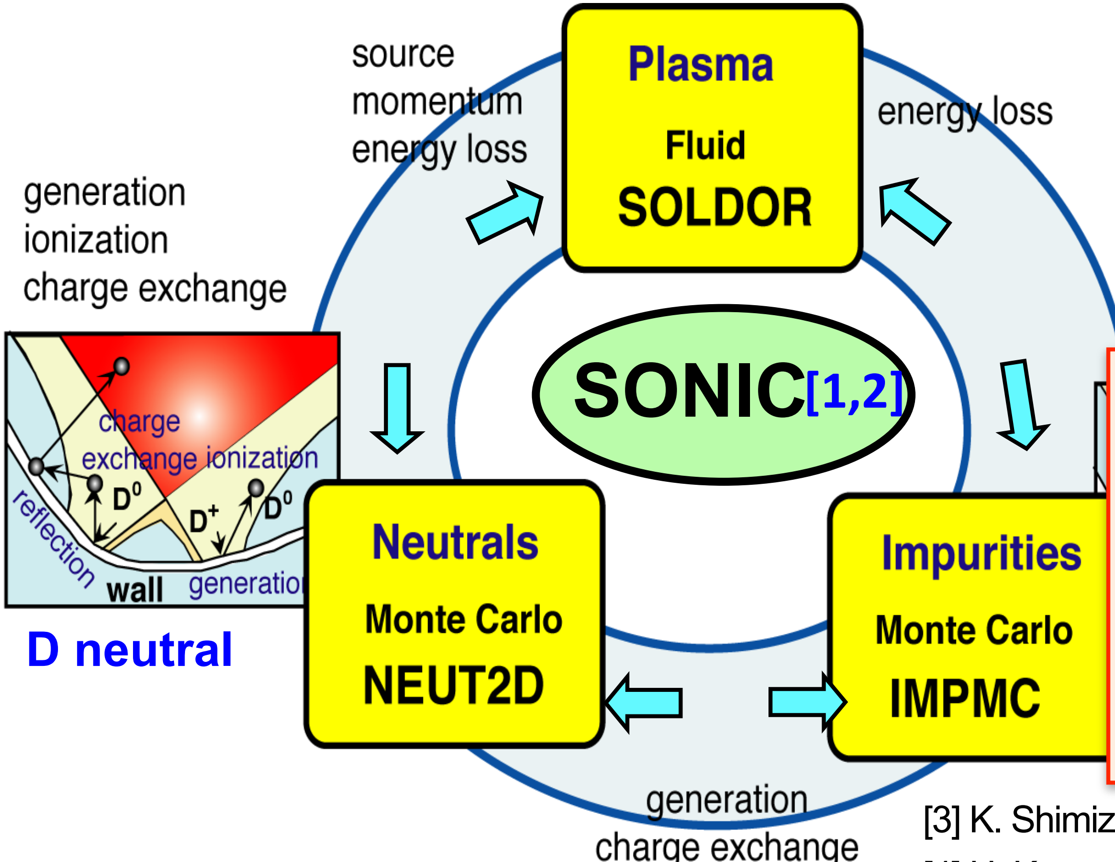
(2) Studies of the "advanced" DEMO divertor started:
Connection $L_{//}$ ($L_{sp}B_{//}/B_p$) and Wetted area (A_{wet}) are increased:
⇒ Magnetic configurations and coil currents of "Super-X divertor" (L_{sp} & A_{wet})
and **"Snow-flake divertor" ($L_{//}$ & flux expansion) are investigated.**
Detachment control ("ITER divertor concept") will be combined with the approach.

(3) High radiation loss in edge plasma (f_{rad}>50%) will be investigated (furture work):
⇔ restricting core-plasma performance and extra radiation power load to blanket.

Development of divertor simulation: SONIC

- **Transport modelling of plasma and impurity (Ar seeding)** changes formation of detachment and radiation distribution in the divertor
- **Effects of the divertor geometry** (divertor leg, exhaust location, V-shape corner, etc.) has been studied in SONIC simulation in order to reduce the target heat load.

[1] Asakura, et al. J. Plasma Fusion Res. SERIES 9 (2010) 136. [2] Hoshino, et al. PET 2011.



Ar impurity transport

Ionization processes:

ionization, charge exchange

surface/volume recombination

Ion transport processes:

thermal force – friction force

Ar^0 atom dynamics

[3] K. Shimizu, et al., J. Nucl. Mater. 313-316 (2003) 1277

[4] H. Kawashima, et al. J. Plasma Fusion Res. 1 (2006) 31

[5] H. Kawashima, et al. Nucl. Fusion 49 (2009) 065007

Recent improvement of SONIC for DEMO divertor simulation

Input parameters: P_{out} = **500 MW**, Γ_{out}=**0.5x10^{23} s^{-1} (r/a=0.95)**, $\chi_i = \chi_e$ = **1 m^{2}s^{-1}**, D = **0.3 m^{2}s^{-1}**

SONIC was calculated in a time scale of impurity transport in the divertor (50-100 ms).

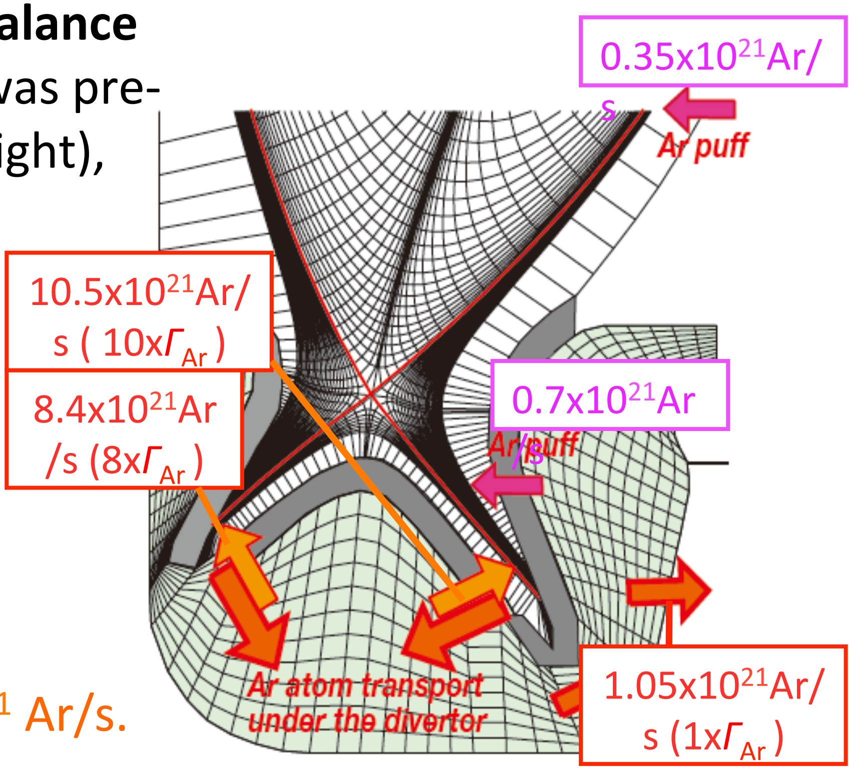
- **Conversion to self-consistent coupling solution of the fluid plasma, MC neutral and MC impurity (Ar) became more stable *under the Demo divertor condition:***
  $q_{//}$ in SOL is 5 times larger than ITER, λ_q^{SOL} of 2-3 mm is smaller due to T_e, T_i higher.

- **Applying techniques and corrections are such as**

(1) using **distribution of Ar atom and particle balance under the divertor (exhaust route)**, which was pre-calculated to a steady-state condition (see right),

(2) smoothing **source terms just above the target** to reduce MC noise/perturbation,

(3) correcting **thermal force term to include effect for long mean-free-path, i.e. reduction at high T_i**, etc.



For the detached divertor, Ar backflow from exhaust slots was handled as gas puff:
Γ_{Ar}^{dom}(in) = 8.4x10^{21} and Γ_{Ar}^{dom}(out) = 10.5x10^{21} Ar/s.

Impurity transport is simulated in the detached divertor

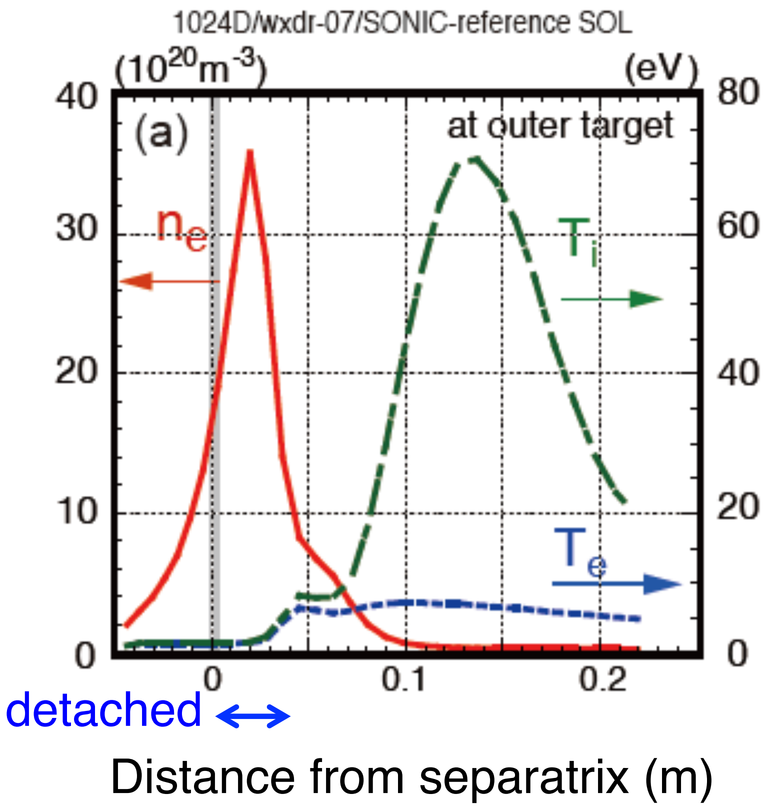
P_{rad}^{tot}/P_{out} increased to ~92% (P_{rad}^{tot} = 460MW),

radiation is distributed at edge and divertor: $P_{rad}^{Edge\&SOL}/P_{out}$ = 39%, P_{rad}^{div}/P_{out} = 53%.

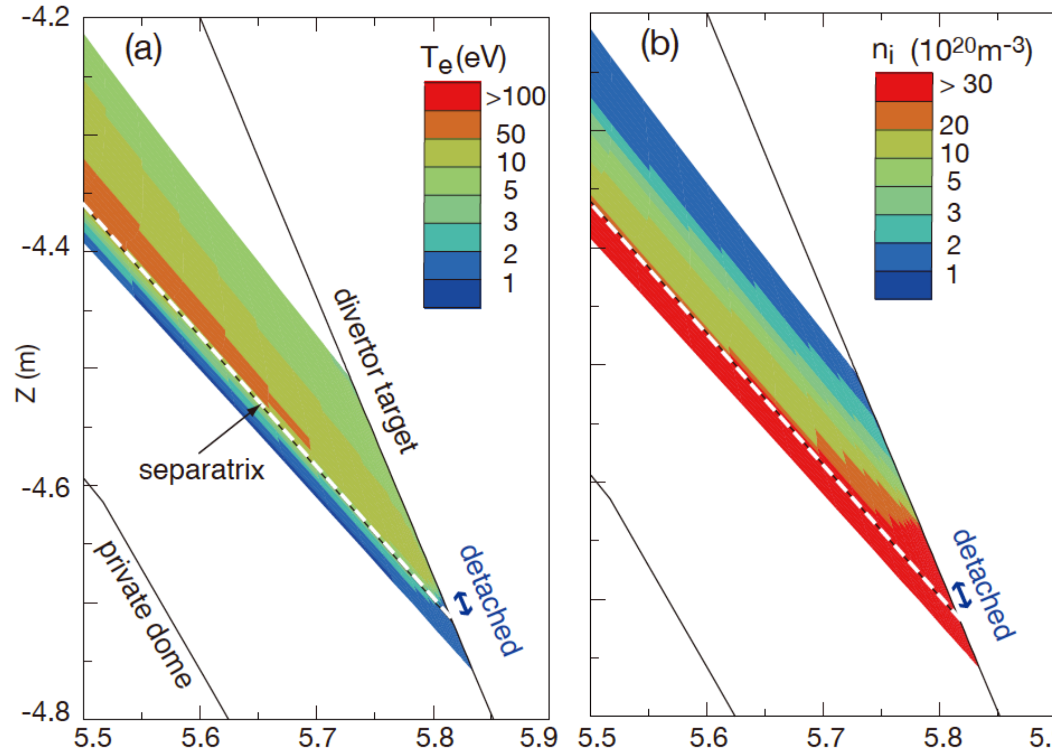
Partial detachment (T_e < 1-2 eV) is seen near the strike-point (< 3cm).

T_i is high 70 eV at outer flux surfaces, due to low density (low collisionality)

T_e, T_i and n_e profiles at outer target



T_e and n_e distributions in outer divertor



Power load profile at the outer target in detached divertor

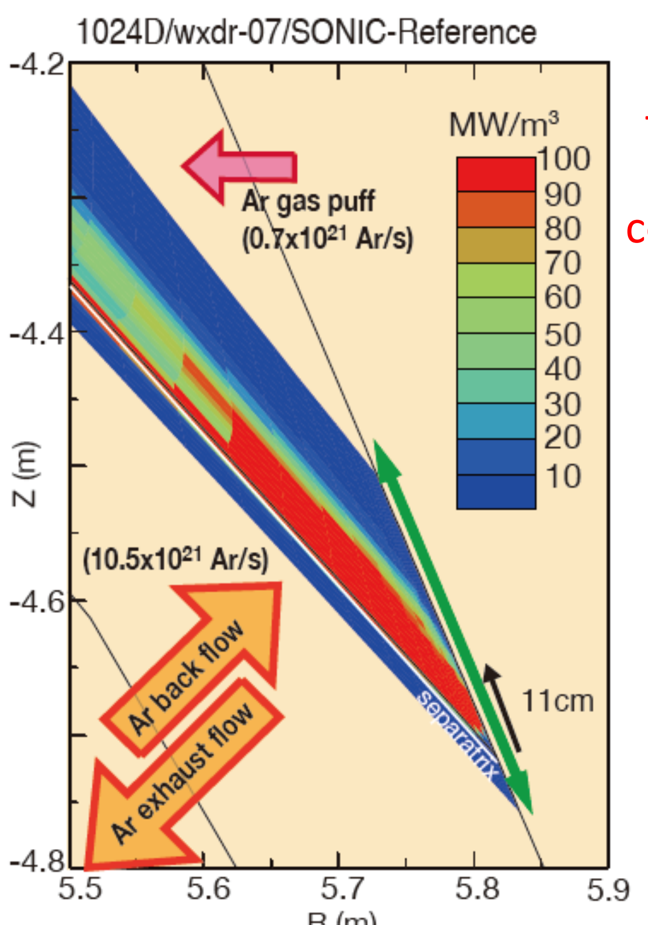
heat load of the plasma transport (conduction and convection) is reduced to ~8MWm^{-2}

Surface recombination of low-temperature ions contributes near the strike-point.

- **Radiation load is large (4-7 MWm^{-2})** ***over a wide area in the outer divertor***

⇒ peak heat load (q_{target}) is ~18MWm^{-2} due to radiation source near above target.

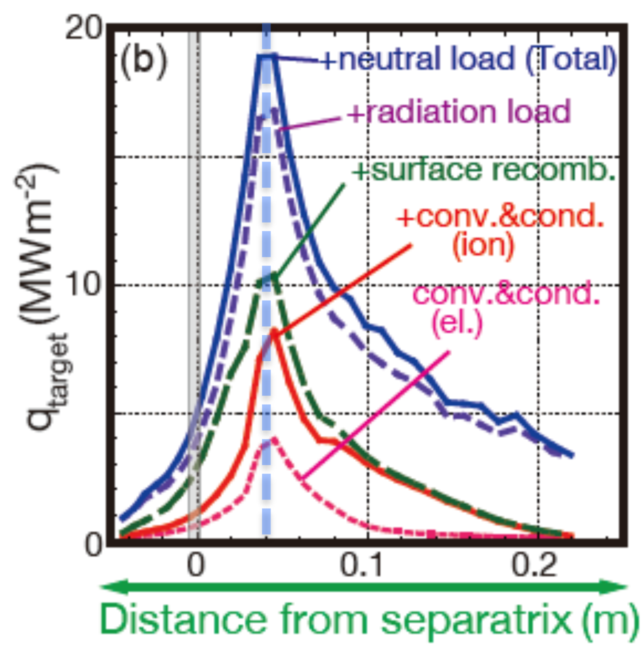
Radiation profile



- **Evaluation of major heat load on the target**

$$q_{target} = \underbrace{\gamma \cdot n_d \cdot C_{sd} \cdot T_d}_{} + \underbrace{n_d \cdot C_{sd} \cdot E_{ion}}_{} + \underbrace{f_1(P_{rad})}_{} + \underbrace{f_2\,(1/2 m v_0^2 n_0 v_0)}_{}$$

Transport component (incl. electron&ion-conduction/convection) | Surface-recombination | radiation power load | neutral load

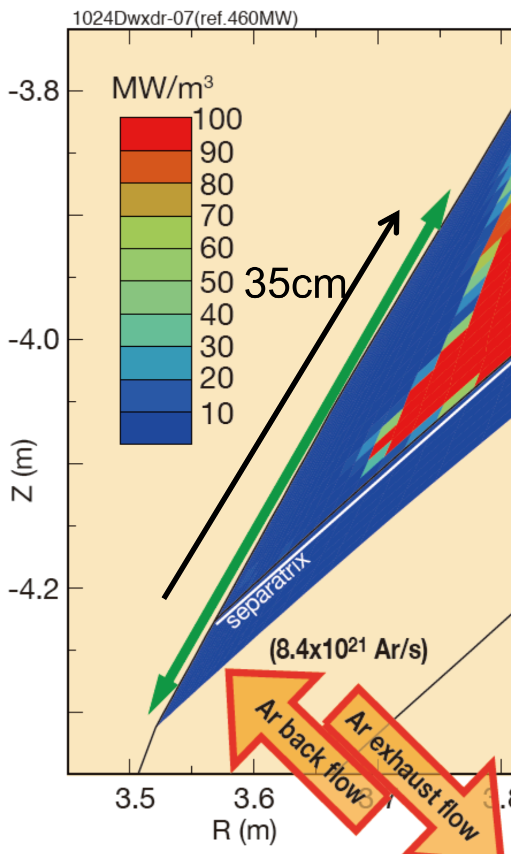


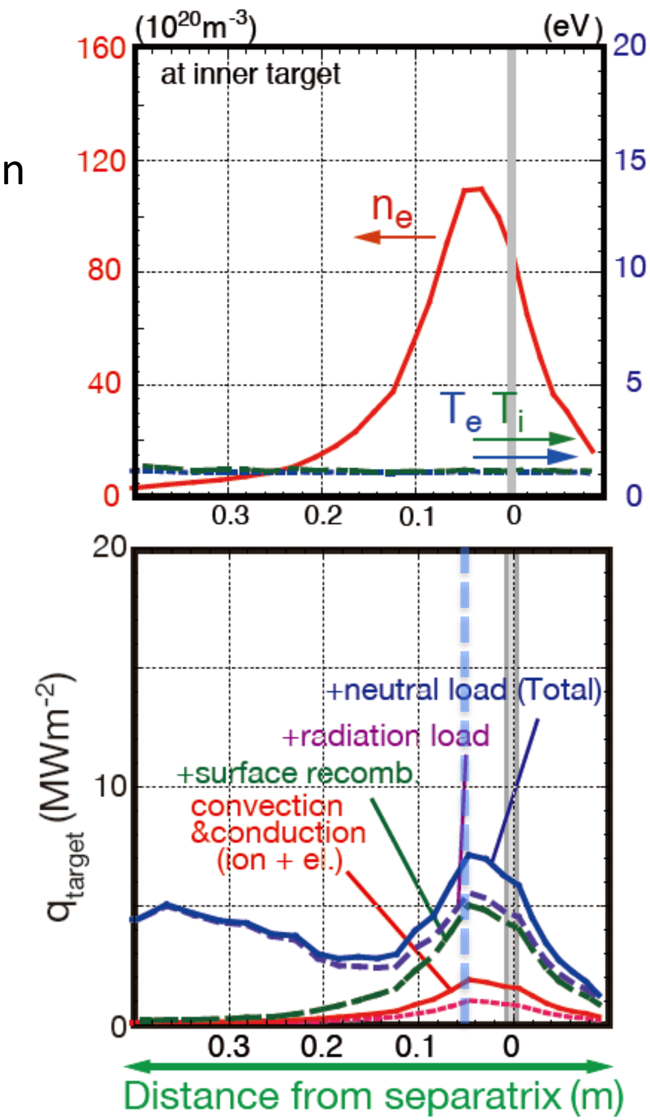
Location of large radiation volume affects heat load profile

Peak q_{target} is ~7 MWm^{-2} near the inner strike-point :
due to surface recombination near the strike-point. Plasma load is small (< 2MWm^{-2}).

- **Radiation moves upstream in the full detachment**
  ⇒ q_{target} is 3-5 MWm^{-2} mostly due to radiation load.

Note: $P_{rad}^{div\text{-}IN}$ = 114 MW ($P_{rad}^{div\text{-}IN}/P_{out}$ = 23%) is smaller than $P_{rad}^{div\text{-}OUT}$ = 151 MW ($P_{rad}^{div\text{-}OUT}/P_{out}$ = 30%).





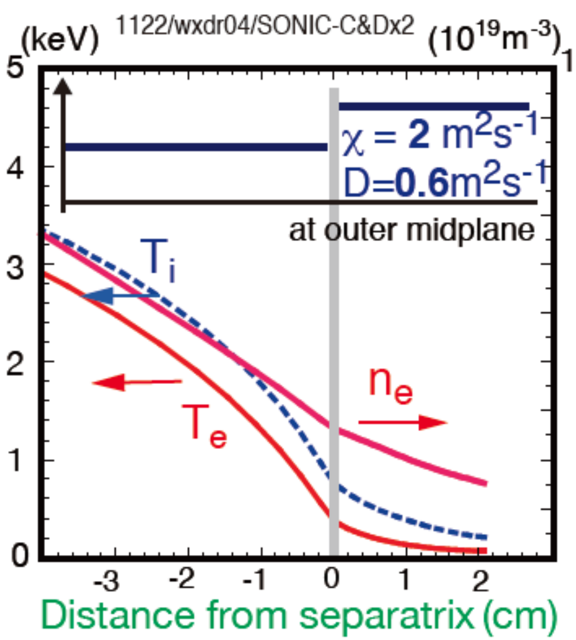
3.1 SOL diffusion is a key to determine detachment

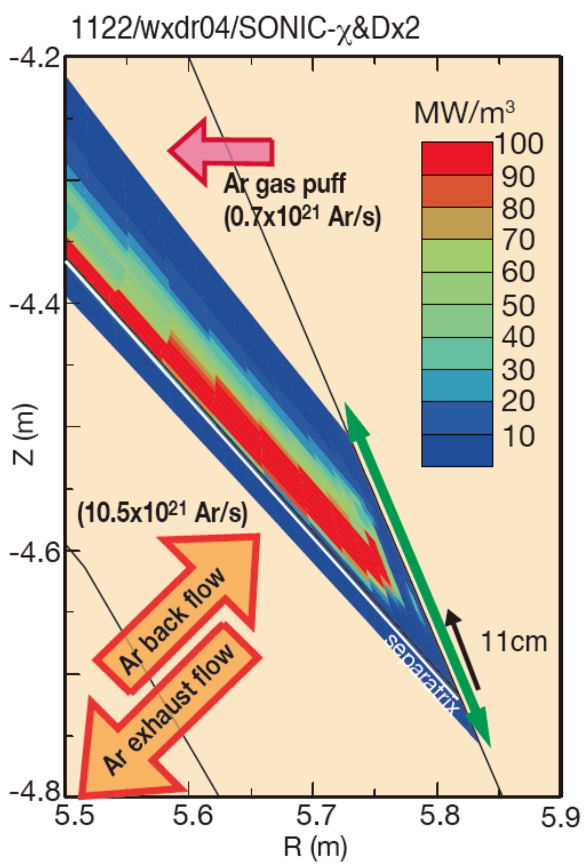
Influence of radial diffusion on detachment and radiation profiles is investigate:

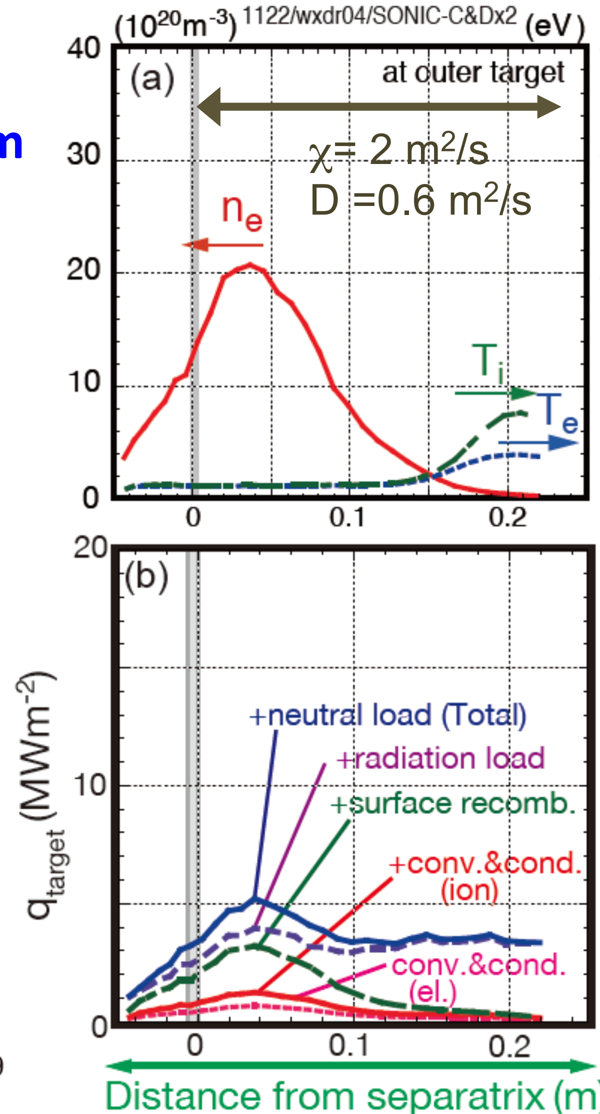
SOL diffusion is enhanced: χ = 2 m^2/s & D =0.6 m^2/s (ref. χ = 1 m^2/s & D =0.3 m^2/s)

⇒ λ_q^{SOL} is increased slightly from 2.2 to 2.7 mm.

- **detachment and heat load are significantly affected:**
  T_i, T_e and n_e decrease, radiation region moves upstream
  ⇒ **both radiation load and plasma load decrease**
  ⇒ **peak q_{target} is ~5MWm^{-2}**







Diffusion at the outer SOL affects detachment and q-profile

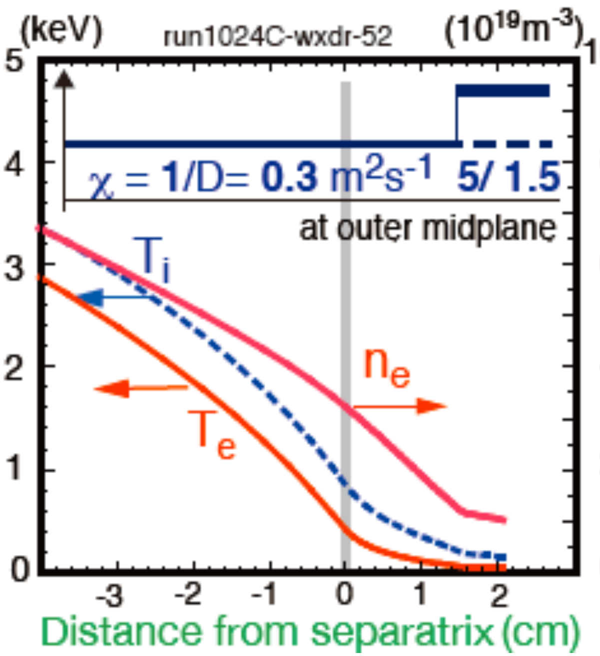
Non-diffusive transport such as "blob" has been widely determined in experiments:

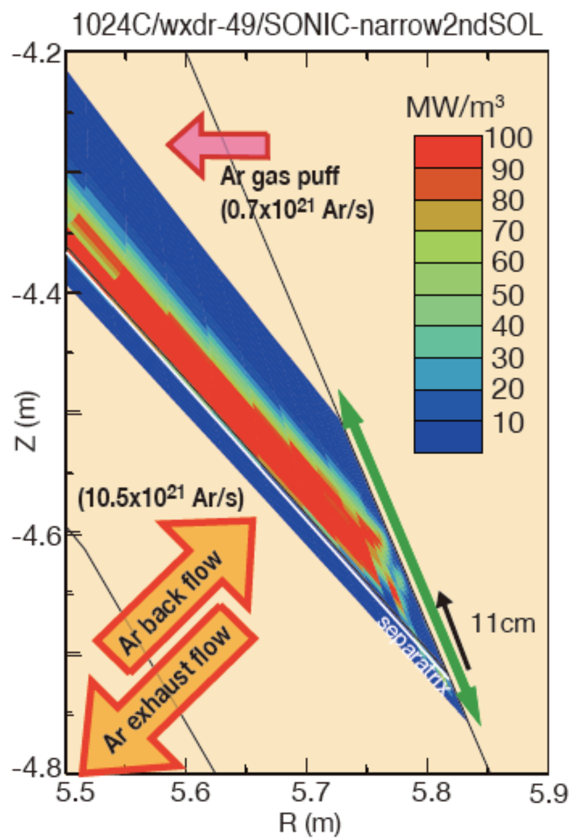
At the outer SOL region ($r^{mid} \geq$ 1.5cm: 15 cm at divertor), χ = 5 m^2/s & D =1.5 m^2/s

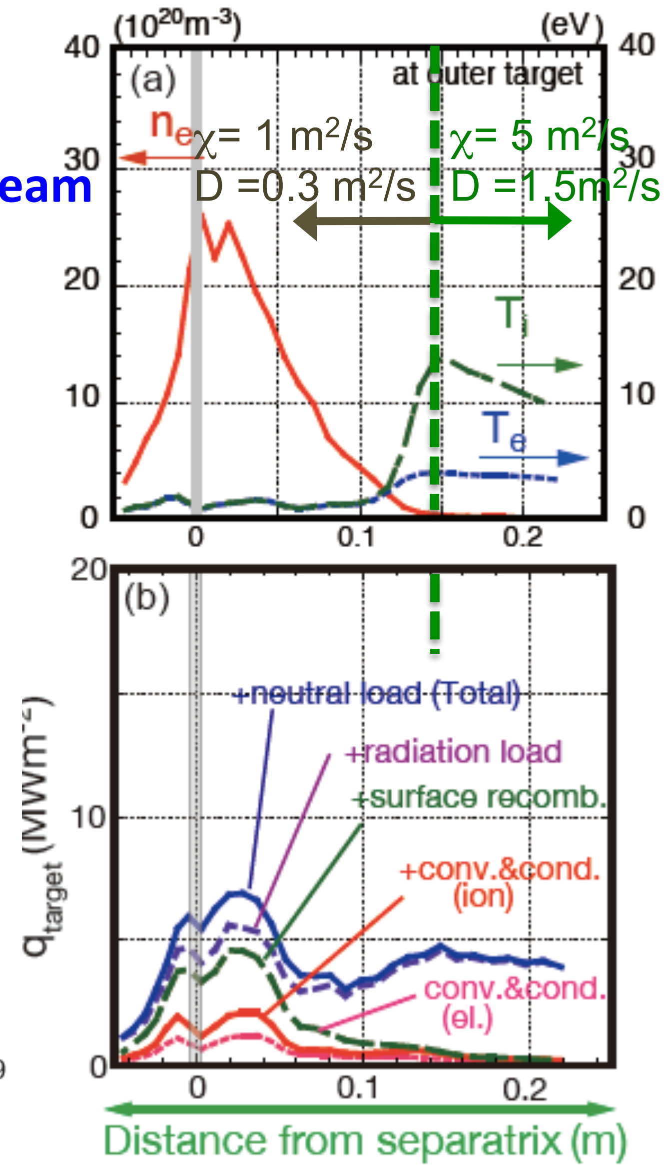
- **SOL plasma near the separatrix is affected by enhancement of diffusion in the outer SOL**

⇒ **T_i, T_e and n_e decrease, radiation region moves upstream**

⇒ **peak q_{target} is ~7MWm^{-2}**







3.2 Effects of the divertor geometry on detachment

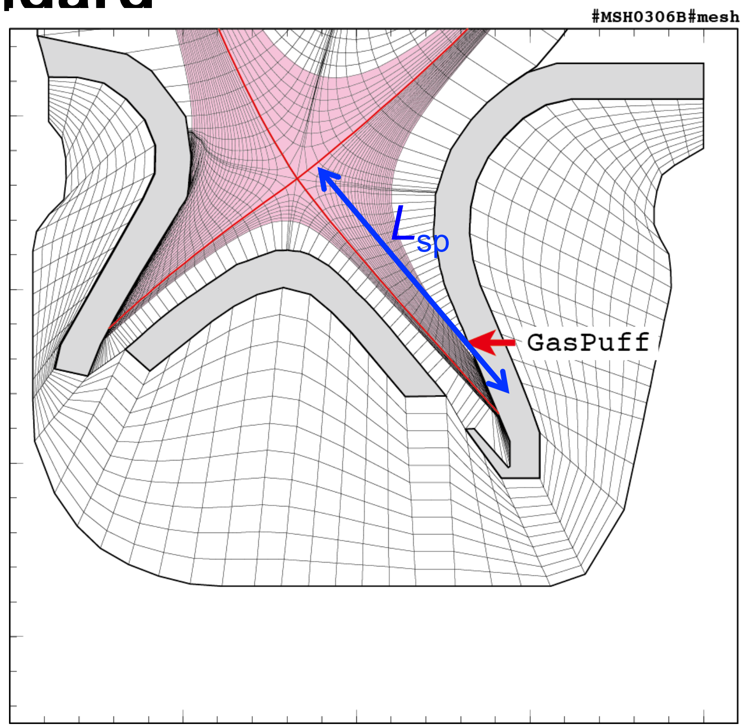
In the previous study [1], effect of V-shaped corner was demonstrated

Divertor leg (L_{sp}) is extended from 1.7m to 2.6m, while flux expansion is decreased.

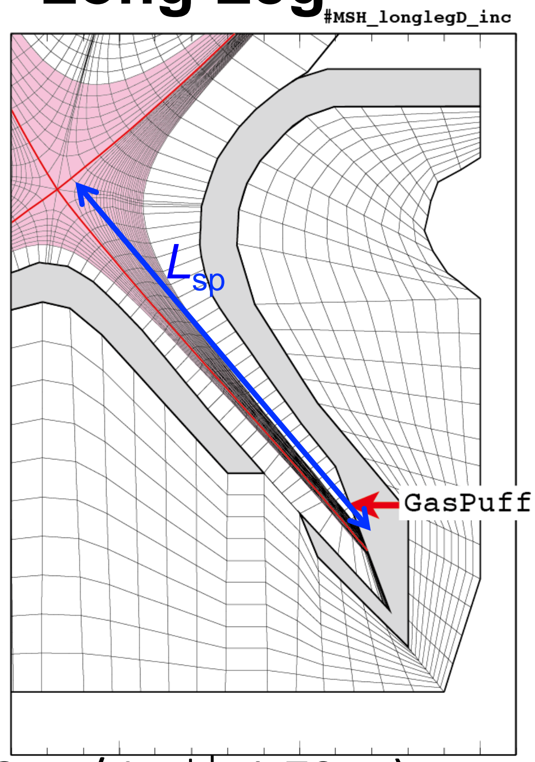
"longer leg divertor" ($L_{//}=L_{sp}B_{//}/B_p$) is expected to decrease target temperature:

$T_{div} \propto q_{//}^{10/7}/n_u^2 L_{//}^{4/7}$ (from 2-point model)

Standard

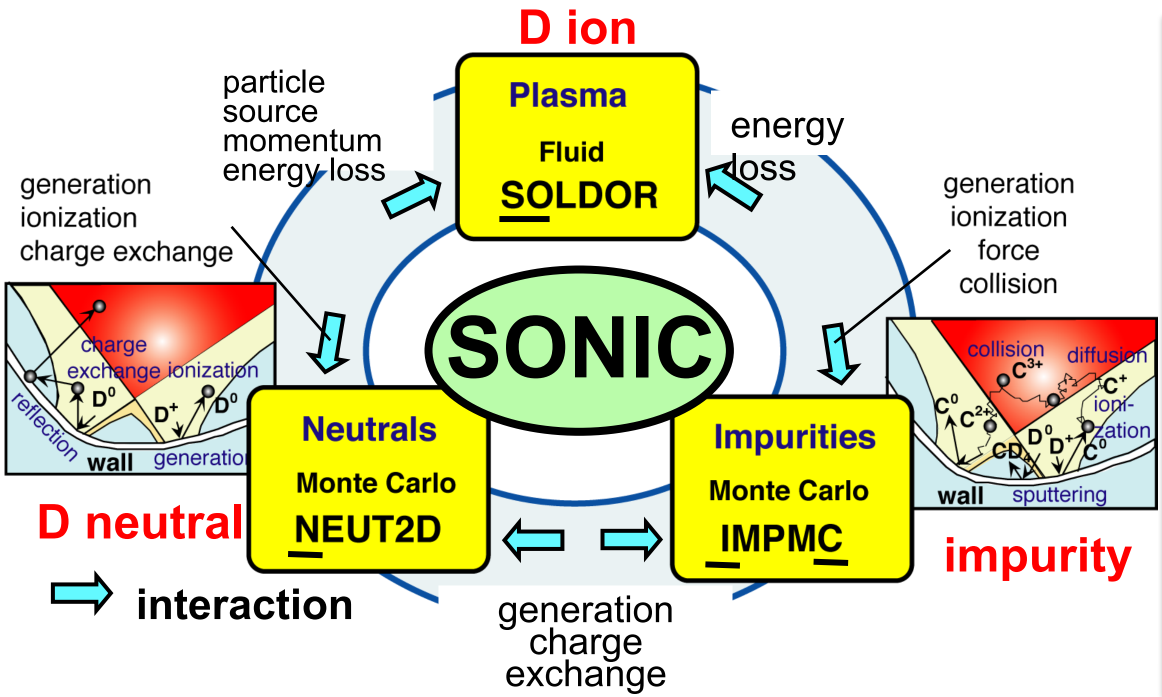


Long Leg



- Outer divertor leg is 1.5 times longer (L_{sp}^{long}=2.58 m / L_{sp}^{std}=1.72 m)
- **In the parallel , $L_{//}^{long}/L_{//}^{std}$=1.2 near the separatix**
- Flux expansion d_{odiv}/d_{mid}=**11.4** in the standard case, and **6.0** in the Long-Leg case

Simple radiation model (in SONIC) was used to survey the geometry parameters for the first & fast investigation



Input parameters

Core boundary at r/a=0.95

$F_i = 6\times10^{22}$ s^{-1}

$Q_i = Q_e$=250 MW

Gas Puff

6×10^{22} s^{-1} from mid-plane

4×10^{22} s^{-1} from outer div.

S_{Pump} = 200 m^3/s

D=0.3 m^2/s

χ =1.0 m^2/s

In the present study, Ar impurity is treated by a non-coronal model instead of the IMPMC calculation:

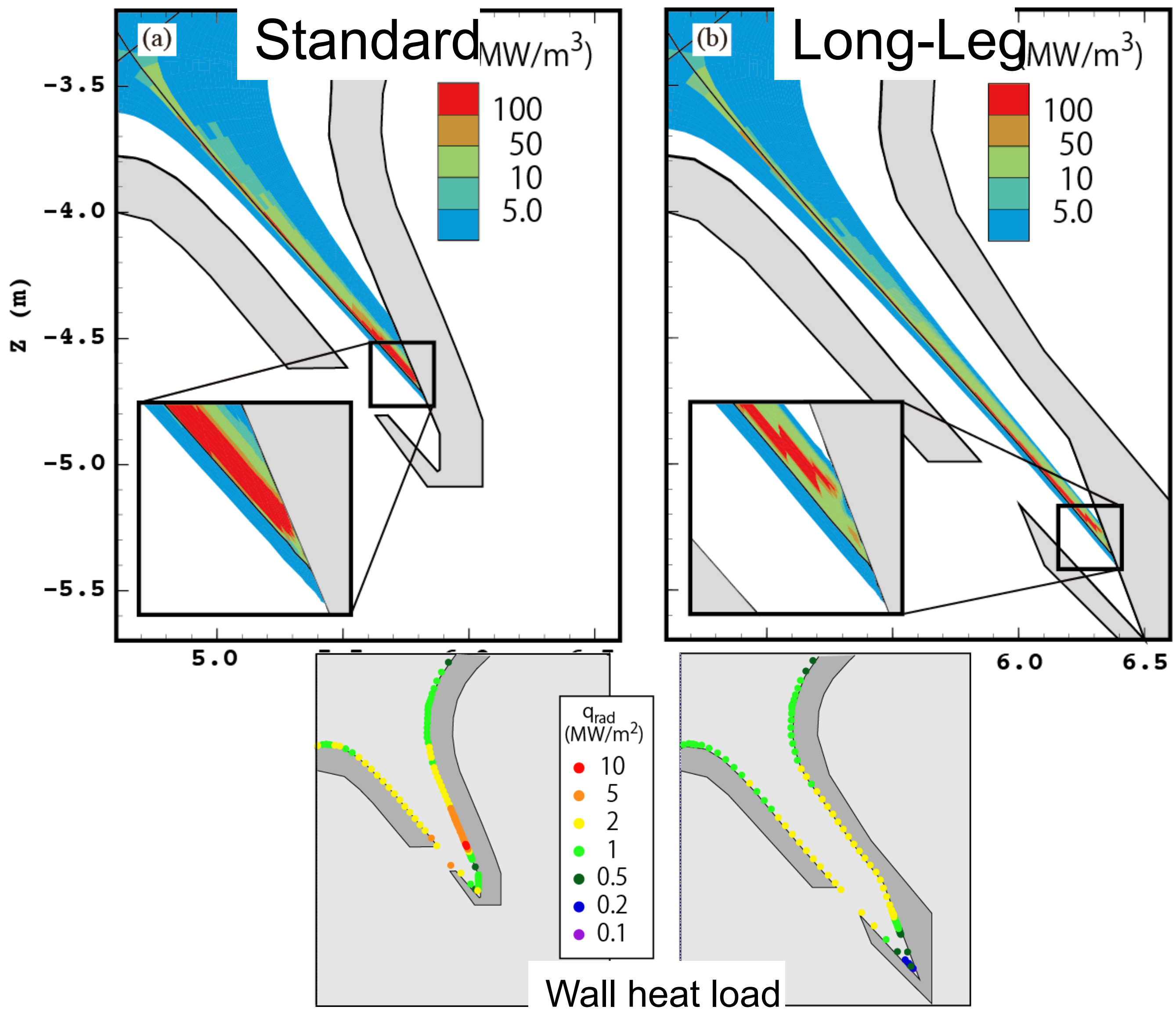
$$P_{rad} = L(T_e, t_r)\, n_{Ar} n_e$$

$n_{Ar} = 0.04\, g_{imp} n_i$ in the outer divertor region, $n_{Ar} = 0.02\, g_{imp} n_i$ in other region

g_{imp} : a control parameter to achieve **P_{rad}^{tot} = 460MW**.

*H. Kawashima, Plasma Fusion Res. 1(2006)031, K. Shimizu, Nucl. Fusion 49(2009)065028.

Radiation load to the divertor is decreased for the long leg



Radiation Power(MW)	Stan-dard	Long Leg
Edge	116.8	97.3
SOL	41.2	33.3
I-Div	93.0	99.2
O-Div	208.6	**230.1**
g_{imp}	0.78	**0.59**

The radiation peak moves toward the upstream.

The radiation heat load on the wall is reduced.

Detachment area is extended outboard (≥10cm) for long-leg

- both T_i, T_e and n_e decrease, and radiation region moves upstream

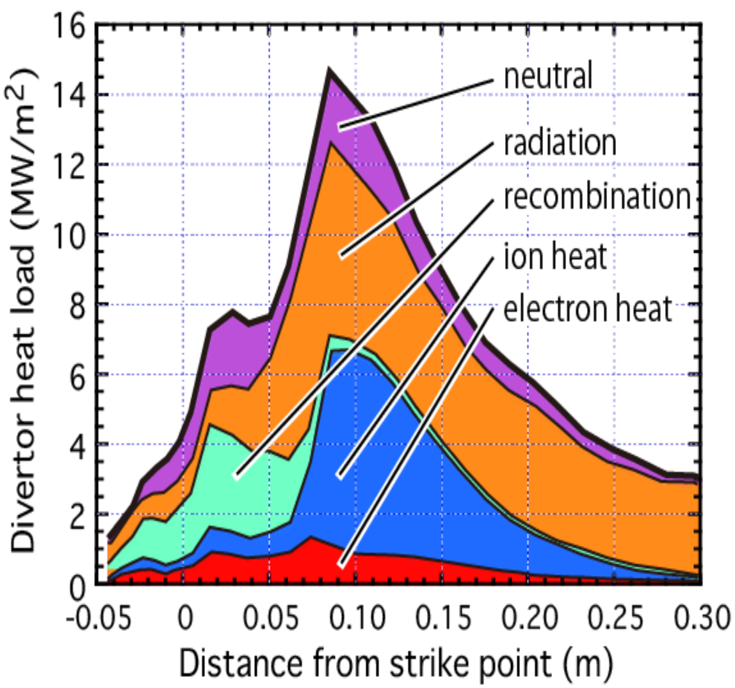
  ⇒ radiation load decrease ⇒ peak heat load decreased from 15 to 10 MWm^{-2}

  recombination and neutral load are still large since recycling is still dominant.

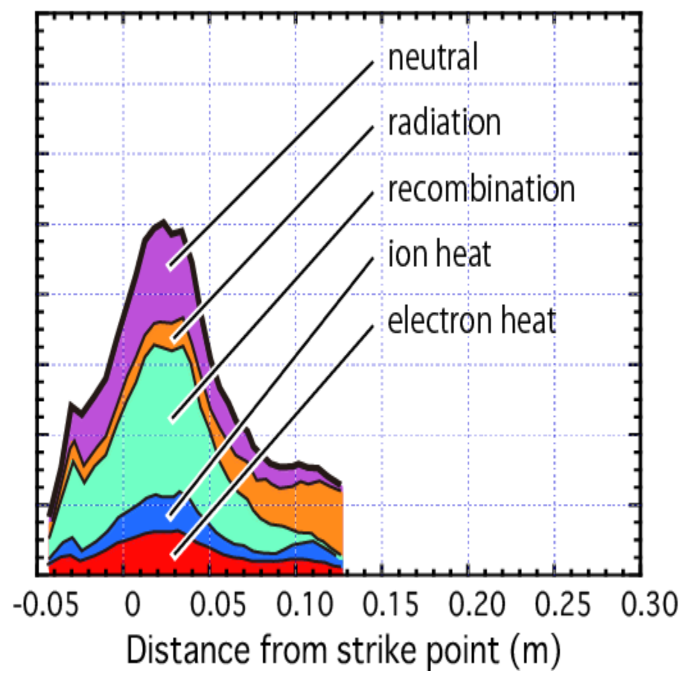
Location of the exhaust slot also affects enhancement of recycling:

⇒ Divertor design of the geometry and exhaust location will be improved to satisfy radiation control and particle exhaust.

Standard



Long-Leg



	Standard	Long Leg
Tot.	**88.0**	**36.8**
Ele.	5.8 (7%)	4.0 (11%)
Ion	**21.5 (24%)**	4.0 (11%)
Rec.	8.9 (10%)	**13.6 (37%)**
Rad.	**39.0 (44%)**	6.3 (17%)
Ntl.	12.8 (15%)	**9.0 (24%)**

4.1 Magnetic structure study: Super-X divertor

Longer leg and wide wetted area (A_{wet}): "Super-X divertor"

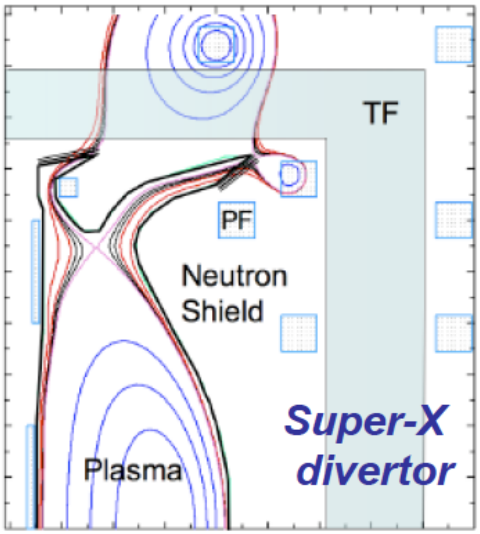
$A_{wet} = [B_p/B_t]_{sol}\ [R_{div}/R_{sol}]\ A_{sol}/\sin\theta$ ⇒ T_e reduction is enhanced ! [6, 7]

At the first step, magnetic configuration for SlimCS is investigated with *minimal number of E-coils outside TFC* ⇒ divertor volume and coil currents are determined

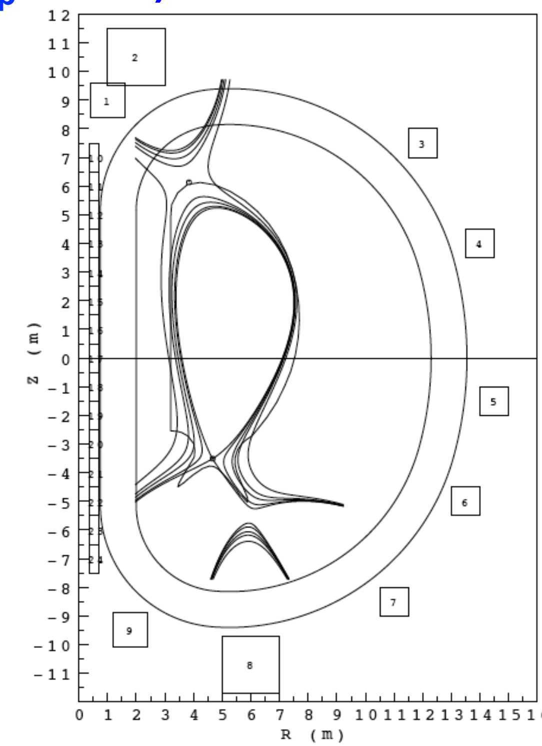
⇒ detachment and head load reduction will be studied.

Super-X divertor

HPDX (A=2.5, R_p=2.1m)
(High Power Density Experiment)



Example of Super-X divertor for SlimCS (A=2.6, R_p=5.5m) in vertical maintenance option



[6] M. Kotschenreuther,et al., Nucl. Fusion 50 (2010) 035003.
[7] P.M.Valanju, et al., Fus. Eng. Design 85 (2010) 46.

Divertor geometry is determined mostly by SX-null location

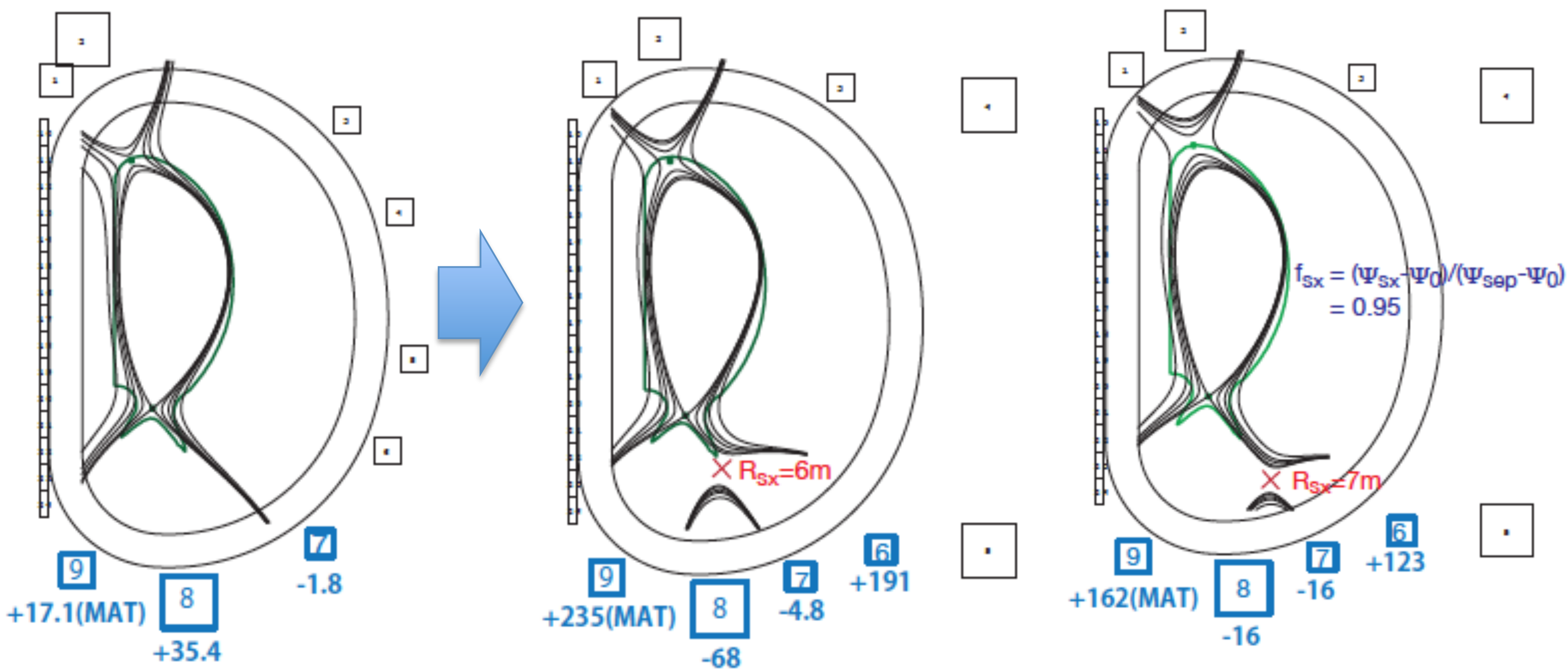
(1) Main divertor coil is shifted from #8 to #9,

(2) SX divertor coils (reversed currents of #7&8) are necessary to obtain the SX config.

(3) Large current of #6 is necessary to extend the field lines towards the outboard

Large divertor coil current (~200MAT) and SXD coil current (-70MAT) are required to have a compact (smaller R_{sx}) SX divertor.

Super-X for SlimCS in Horizontal maintenance option



Note: size of coil do not correspond to the current

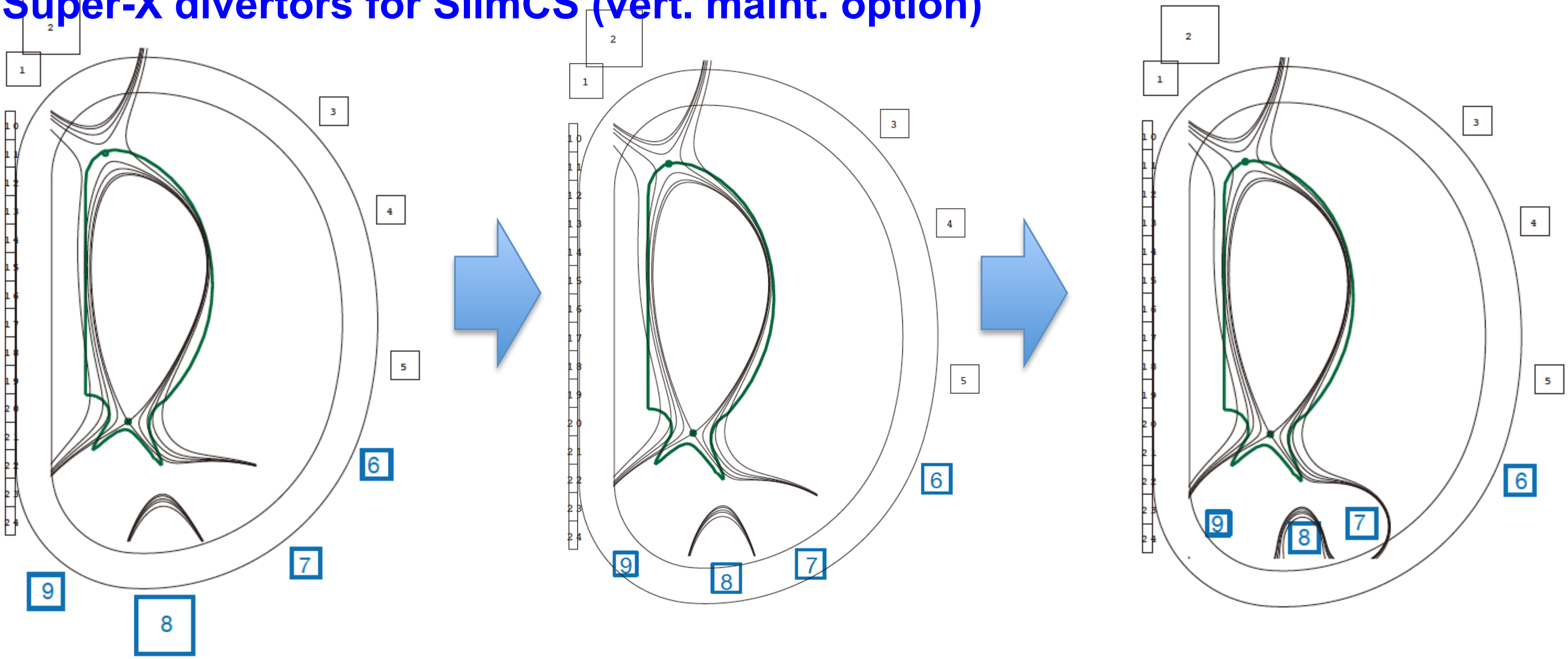
Internal divertor coils are preferable for currents

Internal diver coils are required to obtain the SXD configuration with comparable currents of the std. divertor.

⇔ engineering work (neutron & thermal shields, feedthrough, VV design etc.) is necessary.

Coil#	6	7	8	9
STD.	-8	-2	+36	+17
External	-67	+107	-99	+241
Intermid.	-22	+37	-31	+72
Internal	-5	+10	-44	+18

Super-X divertors for SlimCS (vert. maint. option)



4.2 Magnetic structure study: Snow-flake divertor

"Snow-Flake divertor": Larger flux expansion (F), i.e. longer connection ($L_{//}$), is obtained in a given divertor volume [8,9].

enhancement of P_{rad} and reduction of peak $q_{//}$ are expected near the null-point.

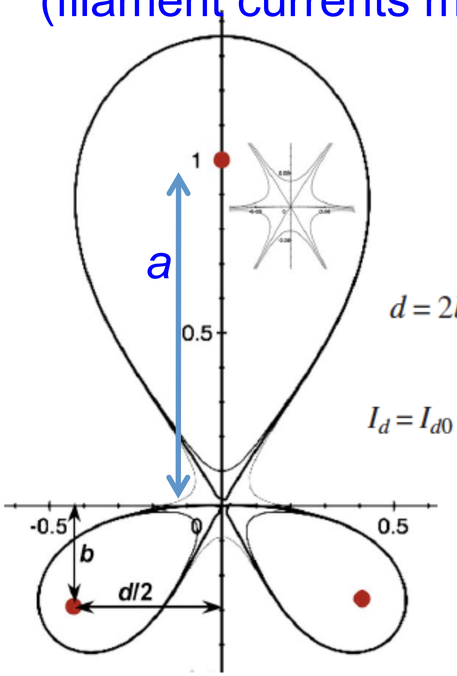
Initial survey of the divertor coil currents shows

⇒ large currents are required for lower CS coils as well as Div. coils, and influence of the current distribution on the plasma shaping control.

D-coils will be installed inside and CS coil design ⇒ Coil design issues are increased

Separatirix for snow-flake

(filament currents model)

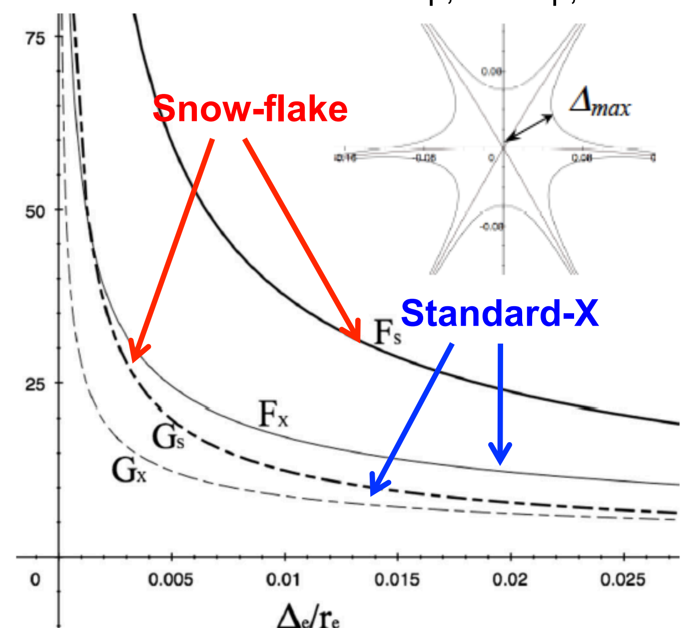


$$d = 2b\sqrt{\frac{a+b}{a-b}}$$

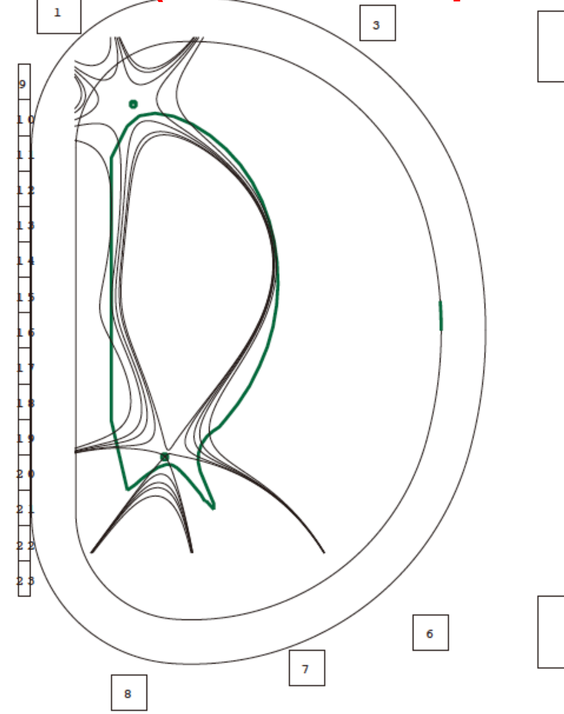
$$I_d = I_{d0} \equiv I\frac{2b}{a-b}.$$

Flux expansion & B_p ratio

$F = \Delta_{max}/\Delta_{mid}$ $G = B_{p,mid}/B_{p,max}$



Snowflake divertor for SlimCS (hor. maint. option



[8] Ryutov, Phys. Plas. 14 (2007) 064502

Summary: Progress in DEMO divertor design

- Power handling scenario is the critical issue for DEMO divertor.
  Simulation study of the power handling (P_{out} = 500 MW) have been progressed:

(1) Power handling simulation with impurity (Ar) seeding:

Conversion of solution became more stable by recent development of simulation.

For P_{rad}^{tot} = 460MW (P_{rad}^{tot}/P_{out}~92%),

plasma detachment ($T_e \sim T_i \leq$ 2 eV) obtained near the outer strike-point (≤ 3 cm)

⇒ peak q_{target} = 18 MWm^{-2}: radiation power load is dominant (~7MWm^{-2}).

Radiation load (3- 7 MWm^{-2}) is dominant over a wide area at both targets.

⇒ **Control of radiation distribution is the important issue.**

(2) Investigation of key physics issues and divertor geometry effects:

Diffusivities and their profiles are important key

⇒ peak q_{target} varies from 5 to 18 MWm^{-2}

Longer divertor-leg (1.8->2.7m) is efficient rather than flux expansion

⇒ peak q_{target} decreases to 10 MWm^{-2}

(3) Investigation of the magnetic structure (Super-X, Snow-flake) may help or improve the divertor design:

studies of coil arrangement and magnetic configuration started:

large divertor coil currents are required for the case of external arrangement, while severe coil design issues are increased.

Future plans for power exhaust simulation

Radiation power load is dominant, and control of the radiation power and distribution is important.

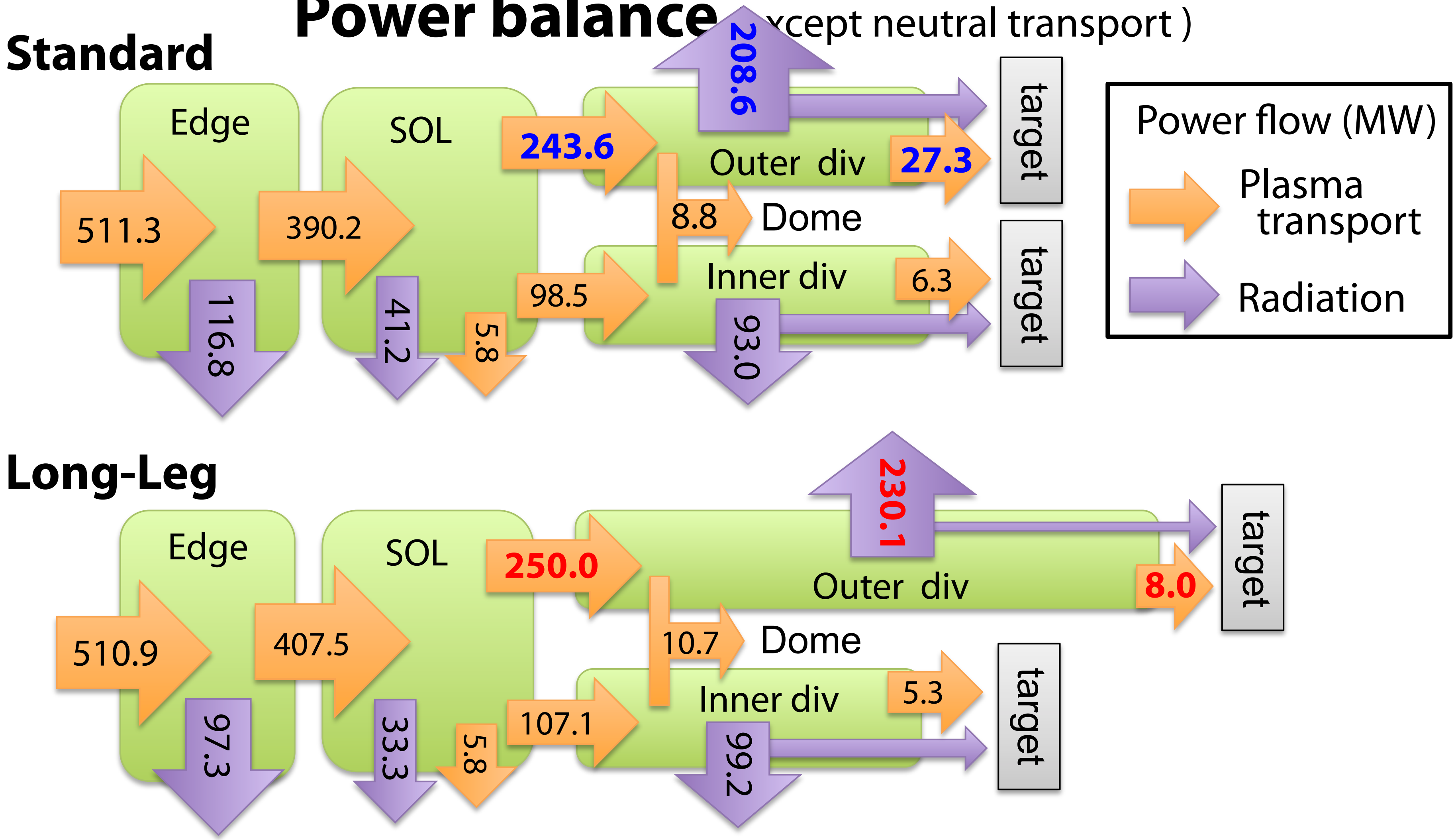
- **Radiation region/distribution would be control by**
  (1) different impurity species such as Ne, Kr, Xe, etc. and puff locations,
  (2) geometry of target inclination and pumping duct,
  (3) impurity transport (pinch/diffusion) in edge will be considered in core-edge boundary region (now r/a >0.95 is treated).

- **Photon absorption by dense and large-size divertor plasma will affect distribution of the detachment → evaluation of absorption incl. atomic model.**

- **Magnetic structure and effect on the plasma will be investigated in SXD and SFD.**

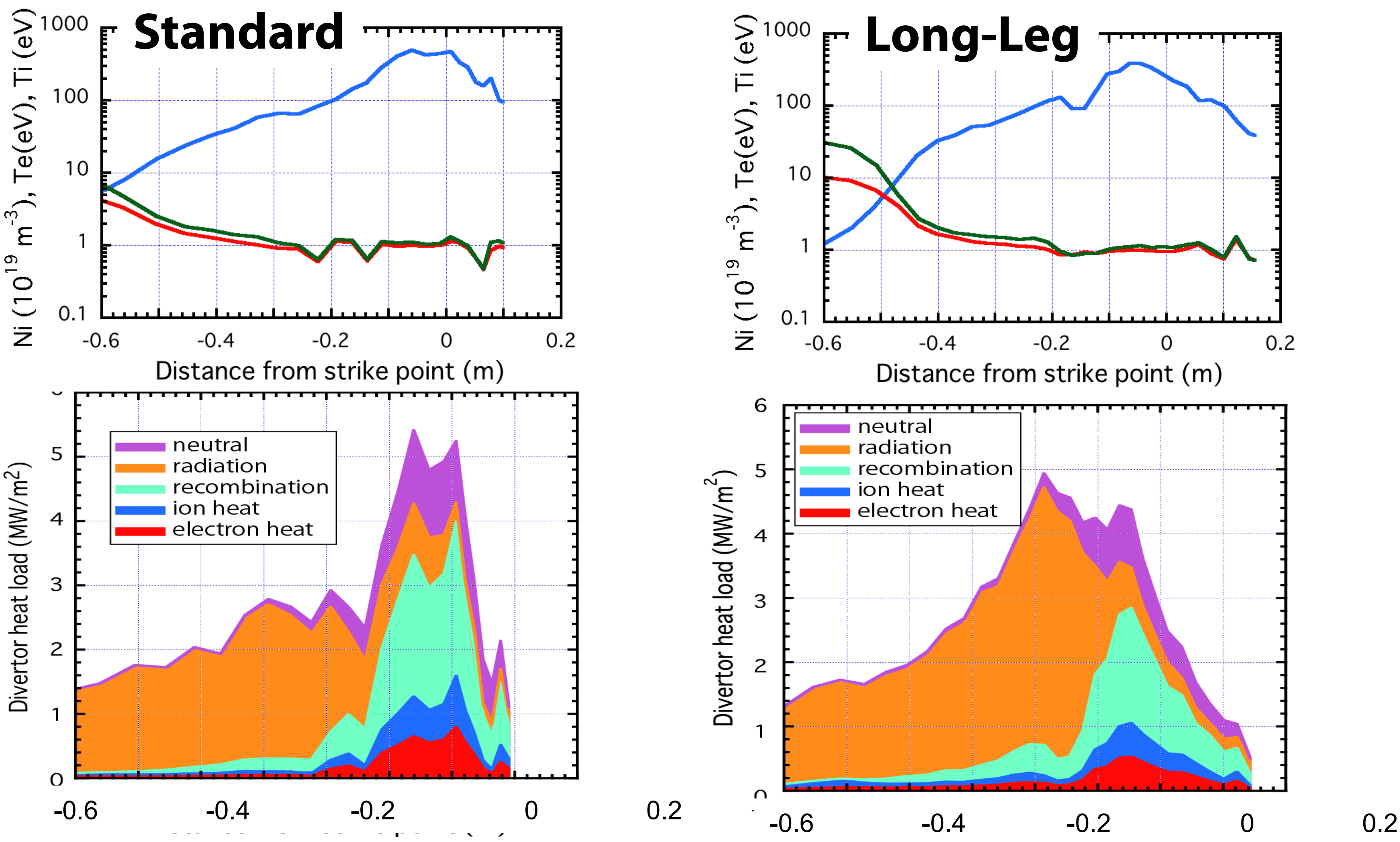
Simulation result

Power balance (except neutral transport)



The power flow is almost the same except the heat to the outer divertor.
In the outer divertor, about 20MW moves from transport to radiation

Heat load on the inner divertor



In the both cases, the peak heat load is less than ~6MW/m^2.