Progress of power and He exhaust simulation study for JA DEMO

Nobuyuki Asakura[1], K. Hoshino[2], Y. Homma[3], Y. Sakamoto[3], and Joint Special Design Team for Fusion DEMO

1) National Institutes for Quantum and Radiological Science and Technology (QST), Naka
2) Graduate School of Science and Technology, Keio University, Hiyoshi, Yokohama
3) QST, Rokkasho

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1. JA-DEMO design and power exhaust concept

Large power exhaust: P_{sep}/R = 30-35 MW/m, is required

Power exhaust concept of primary JA-DEMO design (JA-DEMO 2014)[1,2]:

System code predicted *Greenwald density* (n^{GW}: $0.67 \times 10^{20} m^{-3}$) is lower than ITER

⇒ Impurity seeding is restricted up to n_{Ar}/n_e= 0.25% due to fuel dilution:

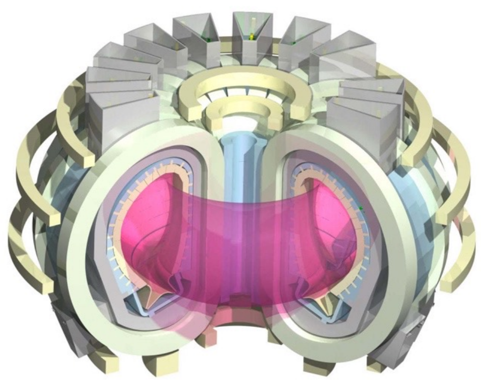
to obtain Fusion power (P_{fusion} = 1.5GW) and Net electricity output ($P_{e\text{-}net}$ ~0.25GW), and β_N (3.5) and Bootstrap-fraction (0.6) with relatively high HH_{98y2} (~1.3).

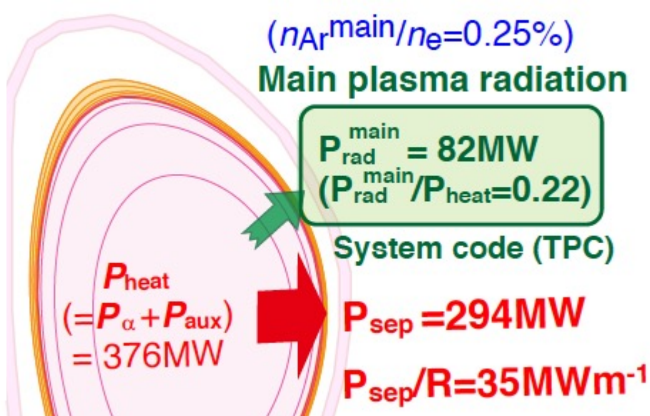
Revised proposal (JA-DEMO higher-κ)[3]: κ_{95} is increased from 1.65 to 1.75 for the same R_p, a_p, B_t and q_{eff}, which increases I_p (12.3 ⇒ 13.5MA) and n_{GW} (0.67⇒ 0.73×10^{20}).

⇒ n_{Ar}/n_e and Radiation loss fraction ($f_{rad}^{main}=P_{rad}^{main}/P_{heat}$) are increased.

JA-DEMO 2014 B_t=5.9T, R/a=8.5/2.42m

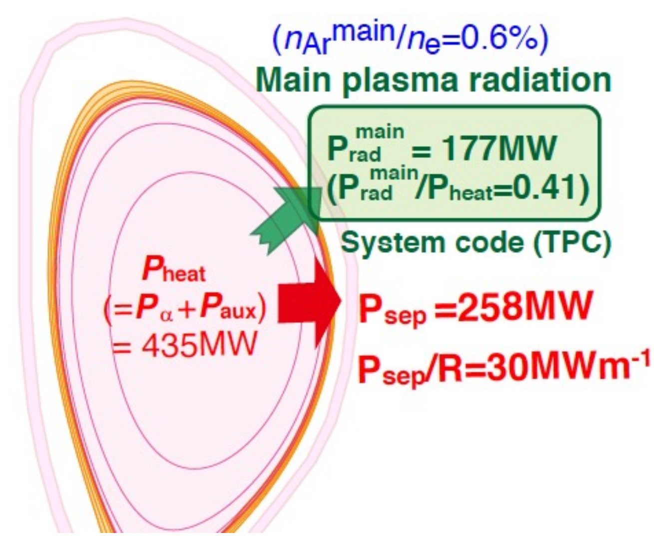
I_p=12.3MA, κ_{95}=1.65, q_{95}=4.1, P_{fusion}~1.5GW





JA-DEMO higher-κ proposal

I_p=13.5MA, κ_{95}=1.75, P_{fusion}~1.7GW



[1]Sakamoto, et al. IAEA FEC 2014,
[2]Tobita, et al. Fusion Sci. Technol. 72 (2018) 537
[3]Asakura, et al. Nucl. Fusion 57 (2017) 126050

Development of SONIC V4 and recent progresses

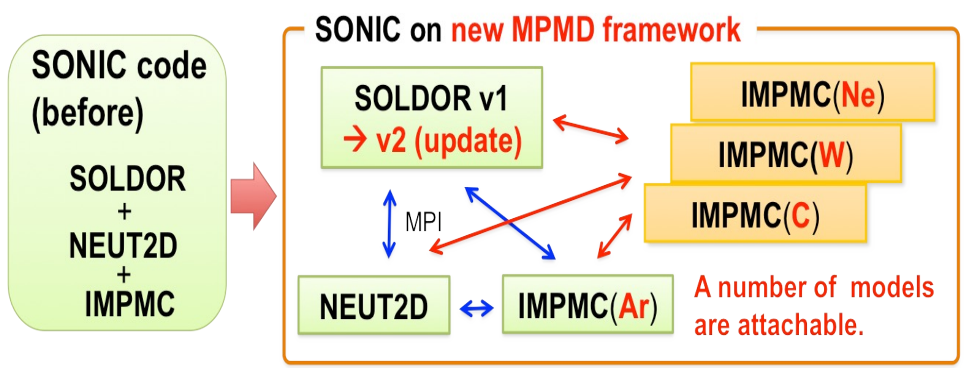
- **Modeling framework using MPMD (Multiple-Program Multiple-Data) approach and MPI (Message Passing Interface) data exchange scheme has been developed for**

(1) Each code can be independently developed, added and replaced.

(2) Improved numerical efficiency: e.g. number of CPUs used for each code can be arbitrarily adjusted to optimize performance.

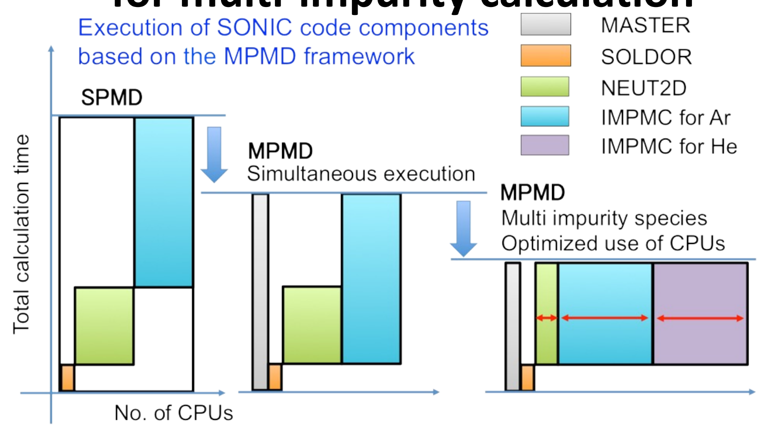
⇒ Power and particle exhaust of DEMO divertor, consistent with Ar and He transports, has been recently simulated. Introduction of drifts (SOLDOR v2) is considered.

(1) Restructured SONIC code with MPMD framework



Each code can be independently developed.

(2) Improved numerical efficiency for multi-impurity calculation



Recent progresses of modelling to evaluate influences under the DEMO condition:

- **Kinetic models** (**thermal force on impurity transport** and **flux limiter for ion conduction**) for low collisionality SOL in DEMO were developed [4, 5].
- **Elastic collision model** of D-D, D-D2, D2-D2, D-He was incorporated, and during improvement[6]
- **Self-consistent photon transport simulation** was performed for SlimCS [7] and JA DEMO.

[4] Y. Homma, et al, Nucl. Fus.60 (2020) 046031, [5] Y. Homma, et al, Nucl. Fus.62 (2022) 045020.
[6] K. Hoshino, et al., PET-18 (2021) [7] K. Hoshino, et al., Contrib. Plasma Phys., 56 (2016) 657.

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Divertor simulation for JA DEMO

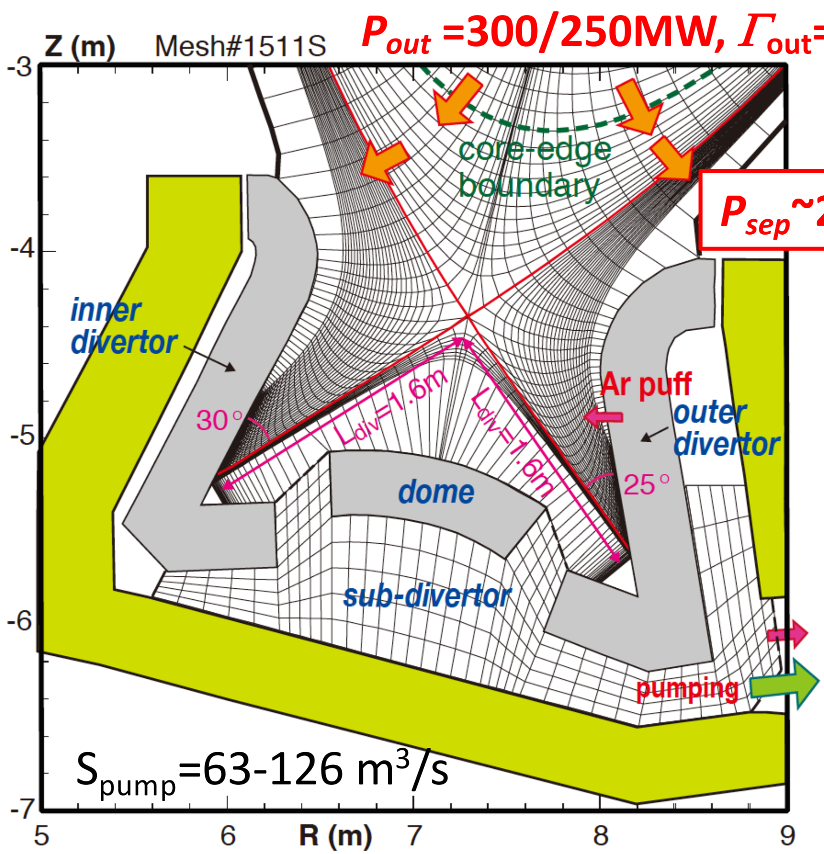
Divertor leg length: L_{div}=1.6m is proposed (x1.6 longer than ITER)

- P_{out}= 300 MW(DEMO 2014), 250MW(DEMO higher-κ) at core-edge boundary(r/a=0.95)
- SOL width 3.2 cm: covering connecting SOL between inner and outer divertors.

Operation window for $q_{target} \leq 10$ MWm^{-2} is determined in severe power exhaust params.:

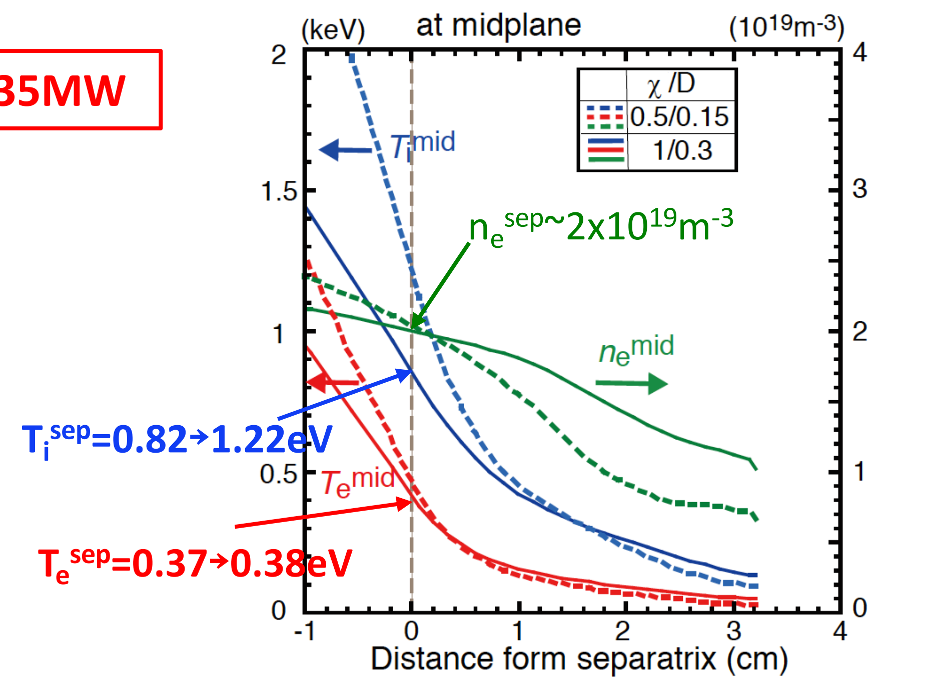
(1) Total P_{rad}/P_{out} is reduced from 0.8 ($f^{*\,div}_{rad}=P^{div}_{rad}/P_{sep}$~0.78) to 0.7 ($f^{*\,div}_{rad}$~0.68).

(2) Diffusion coefficients are reduced from χ=1m^2/s & D=0.3m^2/s to half values.



DEMO higher-κ case: P_{out} =250MW

Influence of reducing χ and D to 1/2



Note: T_e^{sep} & T_i^{sep} are 2-3 times larger than ITER

2. Power exhaust for JA DEMO higher-κ case [3]

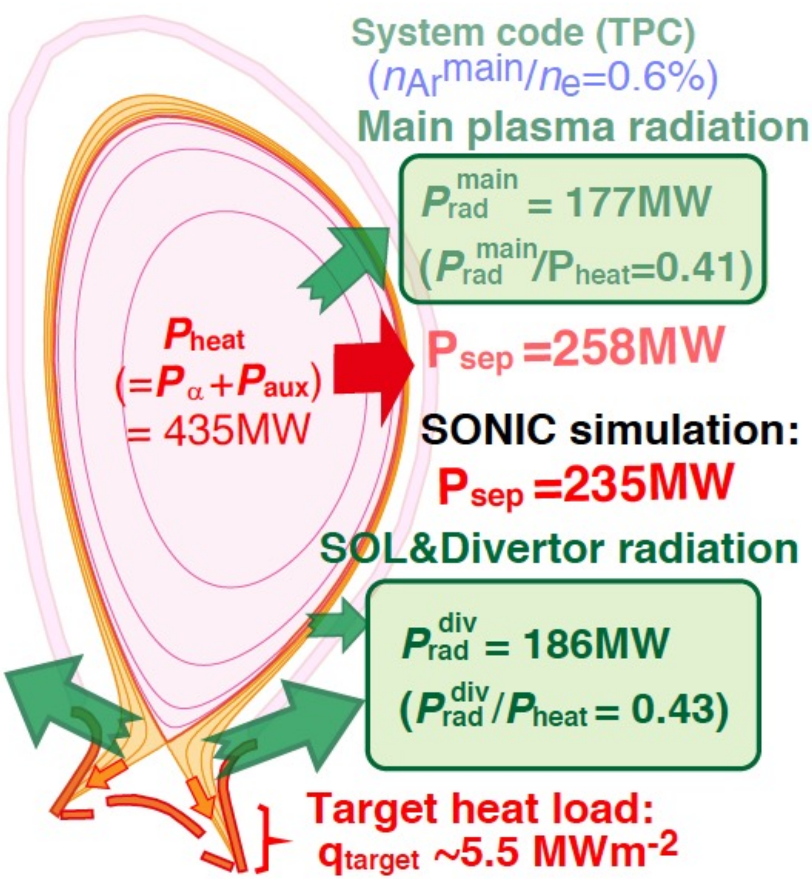
Divertor radiation in P_{sep}~235MW: $f^*{}_{rad}^{div}=P_{rad}^{div}/P_{sep}$ = 0.78

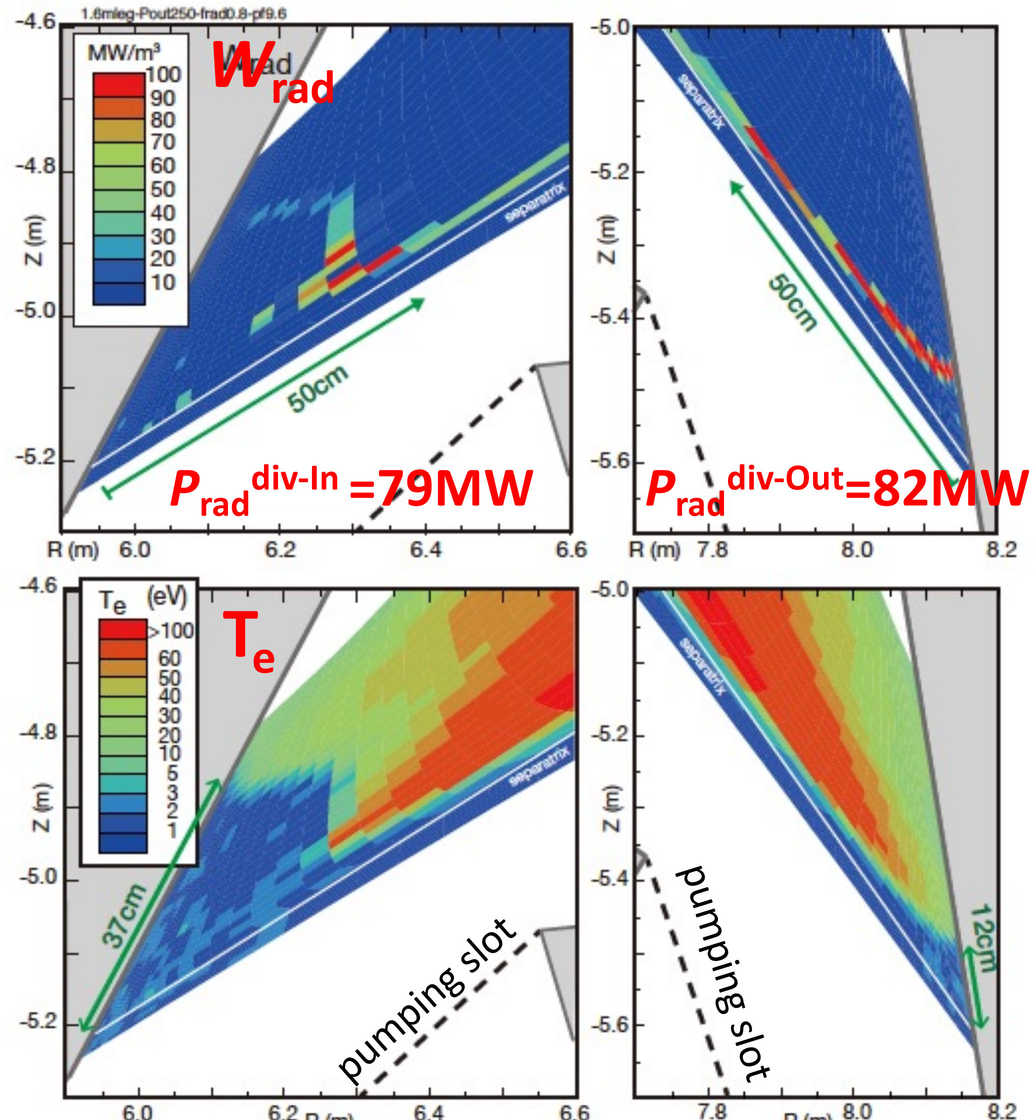
Inner target: radiation peak far above target ⇒ Full detachment ($T_{e,i}^{div}$~1eV in all r^{div})

Outer target: radiation peak becomes near target ⇒Partial detachment (r^{div} < 12cm)

Total radiation fraction in P_{heat} :

$f_{rad}^{tot}=(P_{rad}^{main}+P_{rad}^{div})/P_{heat}$ = 0.84





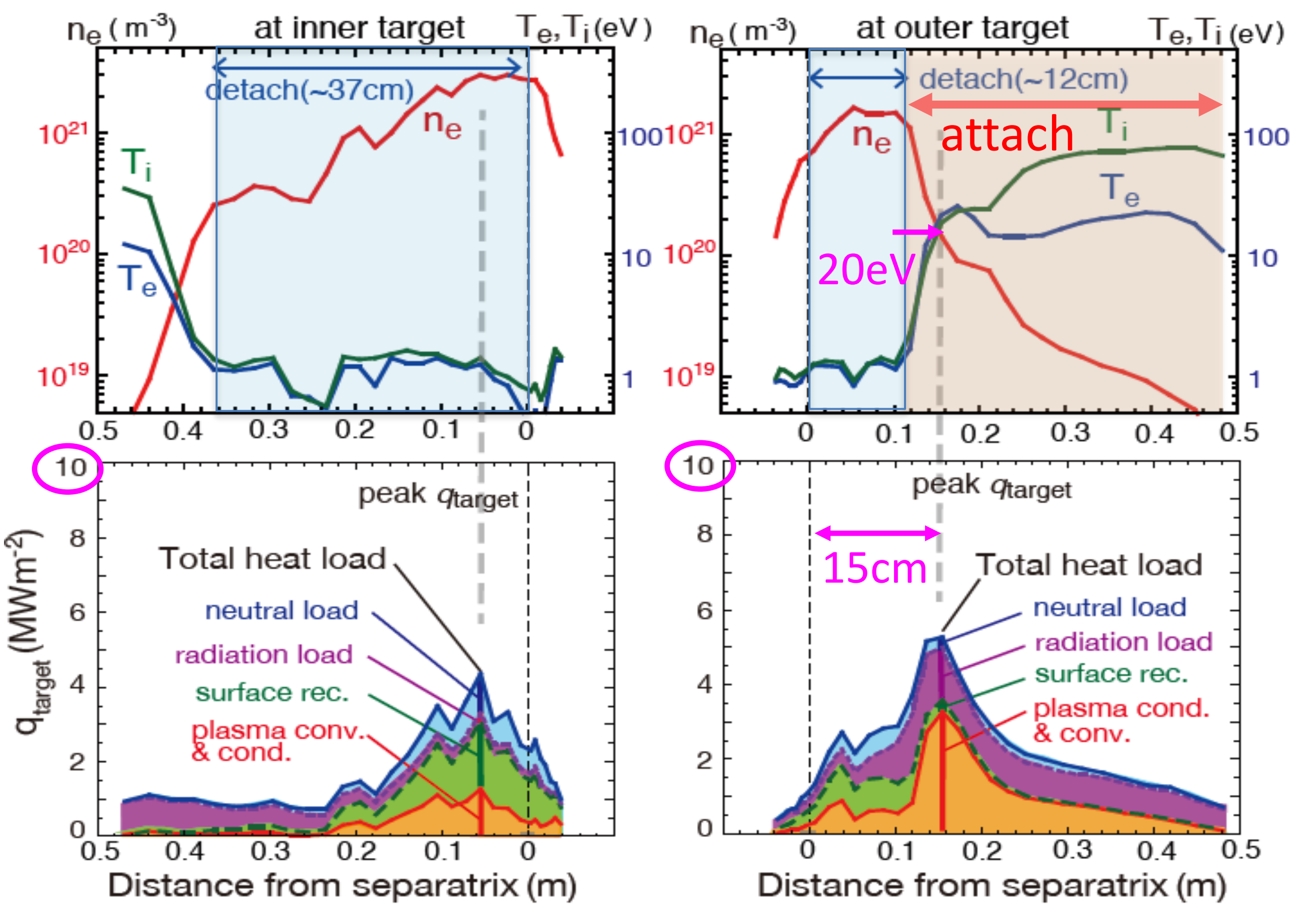
Detachment is produced: q_{target} is less than 10 MWm^{-2}
Outer peak-q_{target} appears in "partially attached" region

Inner target: peak q_{target}~4 MWm^{-2}, where ionization still occurs at $T_e^{div}=T_i^{div}$~1 eV.

⇒ Surface recombination is a dominant Volume-recombination is not significant.
Significant reduction in ion flux (seen in experiments) is *not* simulated.

Outer target: peak q_{target} ~5 MWm^{-2} is seen at "*attached*" region (r^{div} ~ 15cm).

⇒ Plasma heat load is dominant, and Radiation load is also large.



SOL heat flux and Target heat load mapping to r^{mid}

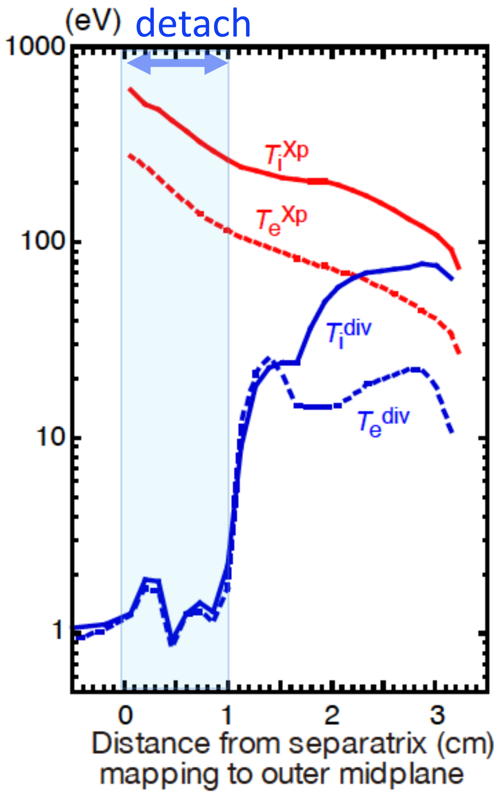
Large $q_{//}^{Xp}$ near the separatrix is reduced in *partial detachment*

- Detachment (T_e^{div}, T_i^{div}=1-2 eV) is produced near separatrix; $r^{mid} \leq$ 1cm ($r^{div} \leq$ 12cm).
- Reduction of total plasma pressure ($p_e^{div}+p_i^{div}+m_i n_i v_{i//}^2$) is not significant (~1/2).
- Large $q_{//}^{Xp}$ in "near-SOL" (short $\lambda_{q//i+e}^{mid}$ = 2.9 mm) is significantly reduced at the target

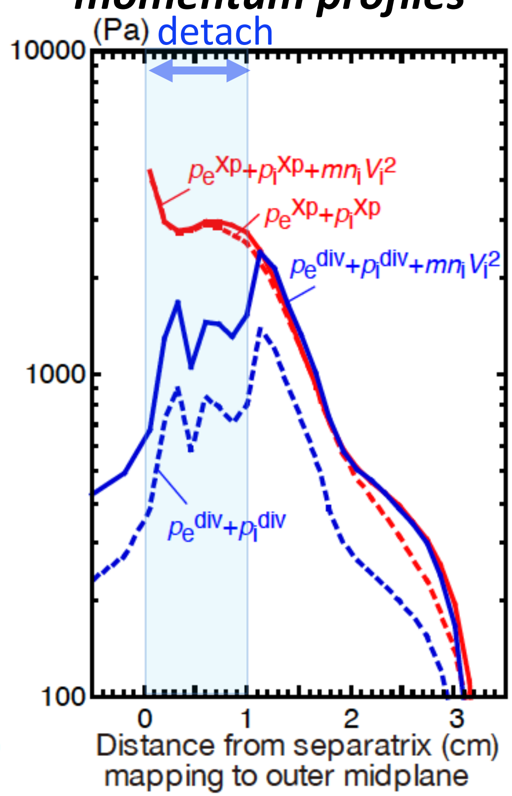
  ⇒ peak q_{target} is produced in "far-SOL" region.

JA DEMO higher-κ with $f^*_{rad}{}^{div}$~0.8:

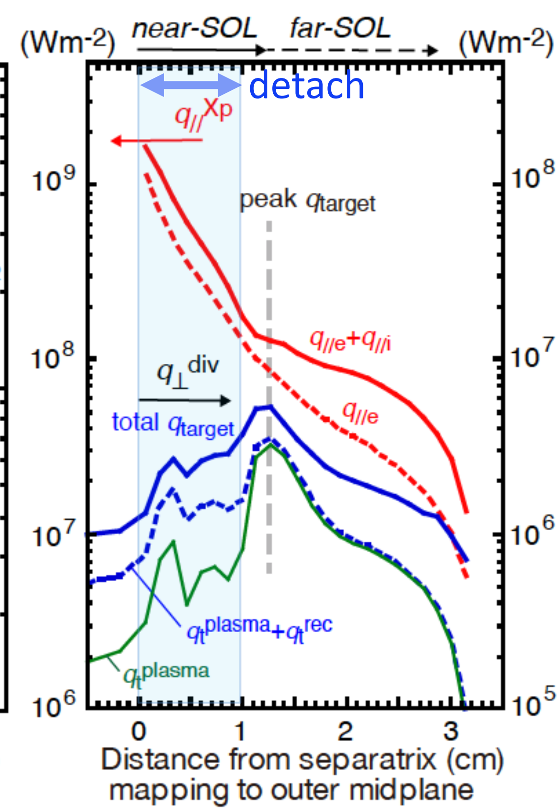
T_e & T_i profiles



Static and Total momentum profiles



$q_{//}$ near X_p vs $q_\perp$ at outer target



[8] N. Asakura, et al., Nucl. Mater. Energy 26 (2021) 100864

Effects of ion transport and diffusion coefficient on $q_{//}$ profile

Flow (ion convection) reversal is seen in "near-SOL", which is produced above target

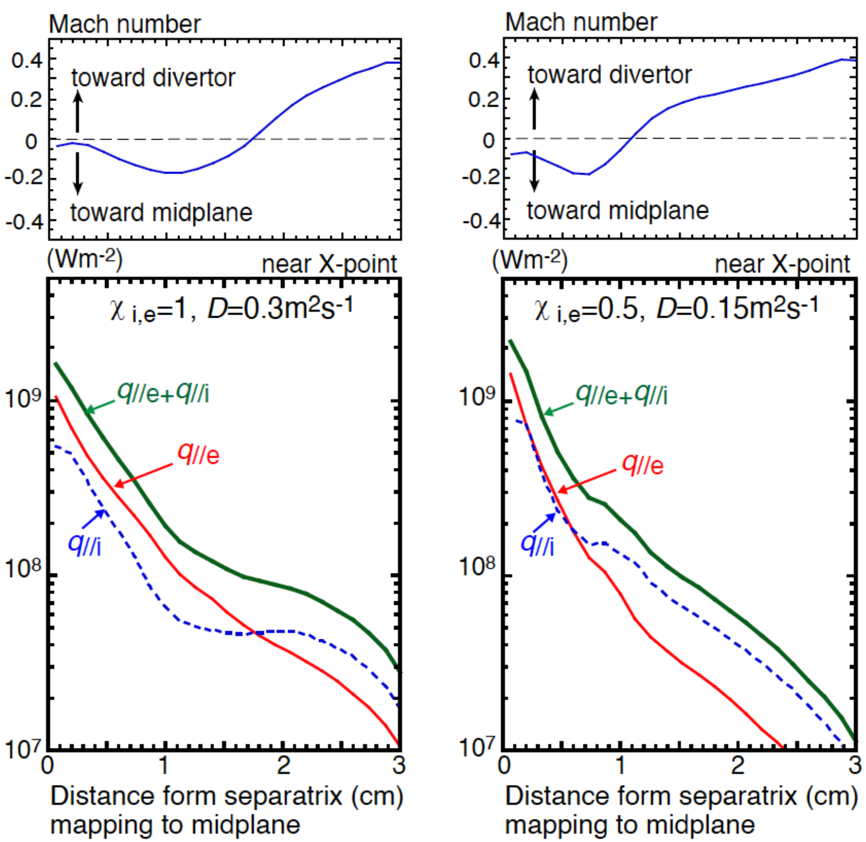
⇒ "shoulder" is formed in Ion heat flux profile: $\lambda_{q//i+e}$ in "near-SOL" is larger than $\lambda_{q//e}$

JA DEMO: T_e^{sep} & T_i^{sep} are 2-3 times larger than ITER

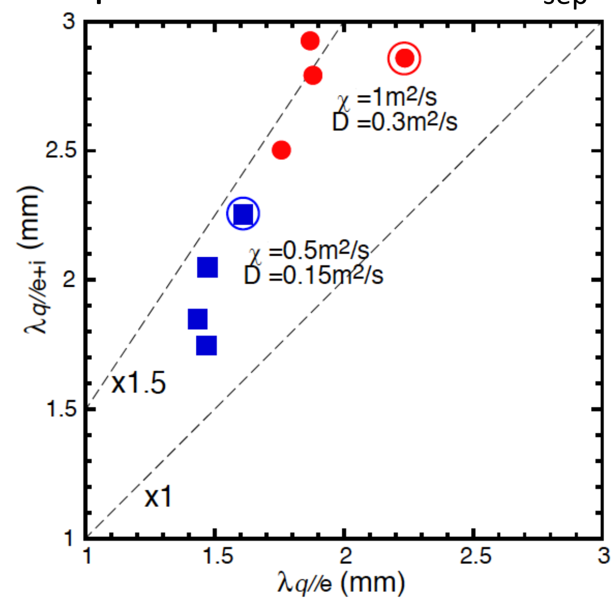
⇒ $\lambda_{q//e}$=2.2, $\lambda_{q//e+i}$=2.9 mm for standard ITER χ/D=1/0.3 m^2s^{-1}: $\lambda_{q//}$= 3.4 mm in ITER[9].

Reducing to half values (χ/D = 0.5/0.15 m^2s^{-1}) ⇒ $\lambda_{q//e}$&$\lambda_{q//e+i}$ are reduced to 1.6, 2.3 mm.

DEMO $q_{//e+i}$ profiles are *still wider than Eich's scaling*[10] (~1mm) & *GS model*[11] (~1.4mm).



"near-SOL" e-folding lengths of el. & total heat flux profiles for 4 cases of P_{sep}&$f^{*}_{rad}{}^{div}$



[9] Kukushkin, et al. J. Nucl. Mater. (2013).
[10] Eich, et al. Nucl. Fusion (2013).
[11] R. Goldston, Nucl. Fusion (2012).

3. Divertor operation in low density (n_e^{sep} ~1/3-1/2*n^{GW})

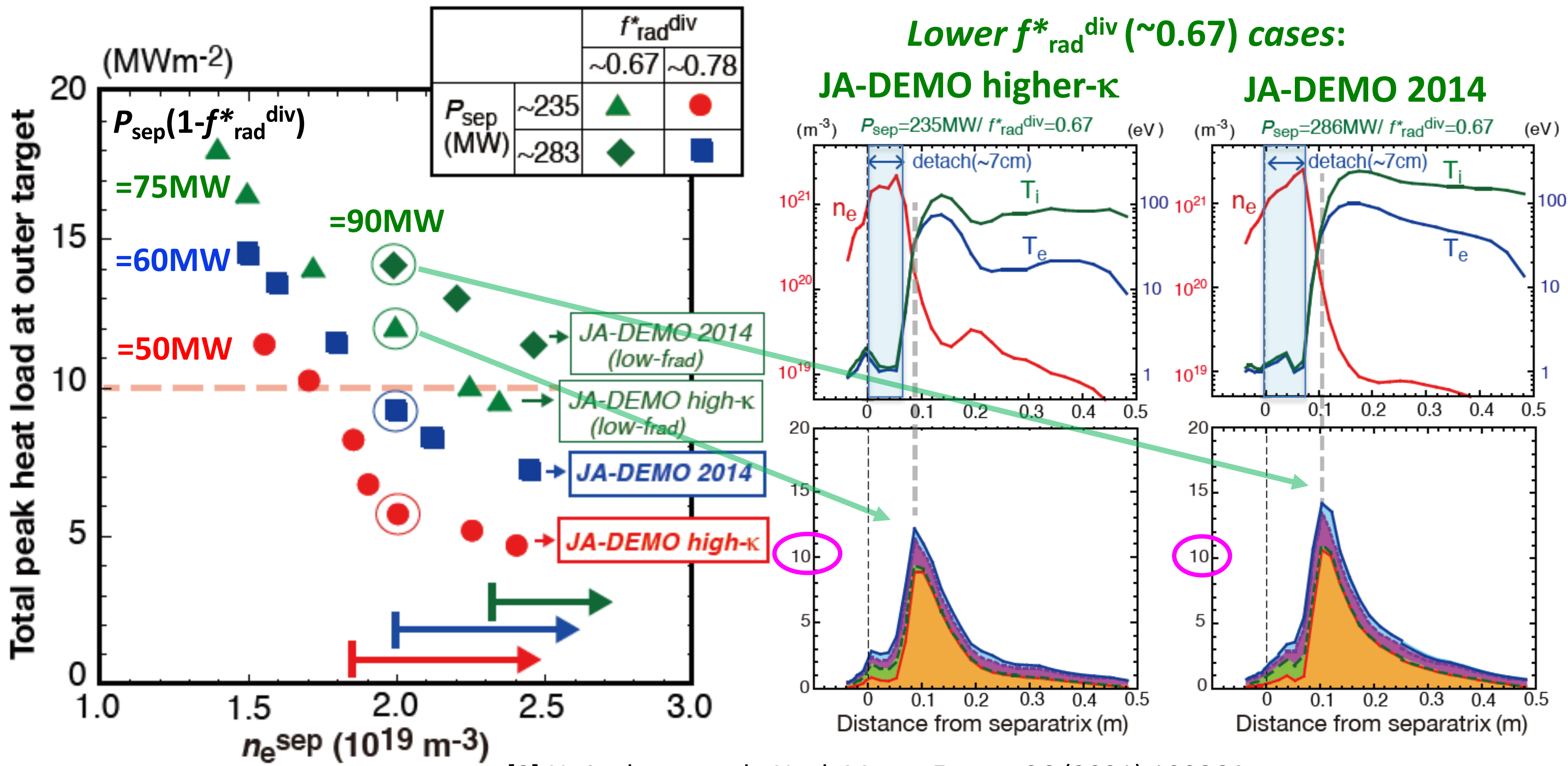
q_{target} is reduced (≤10MWm^{-2}) in *Both reference cases* (f^{*div}_{rad}~0.78)

Higher-κ case can furthermore reduce peak-q_{target} and allow enough operation margin.

Lower f^{*div}_{rad} ~0.67 (P_{sep}~235 and 283 MW) cases:

narrower *detachment*, and increasing T_i^{div}, T_e^{div} *at attached region* ⇒ increasing q_{target}:

higher n_e^{sep} >2.3x10^{19}m^{-3} (DEMO *higher-κ*) or >2.6x10^{19}m^{-3}(DEMO 2014) is required.



		f^{*div}_{rad} ~0.67	f^{*div}_{rad} ~0.78
P_{sep} (MW)	~235	▲	●
	~283	◆	■

[8] N. Asakura, et al., Nucl. Mater. Energy 26 (2021) 100864

Divertor operation: smaller diffusion coefficients

Influences of χ and D become large for *lower* radiation fraction

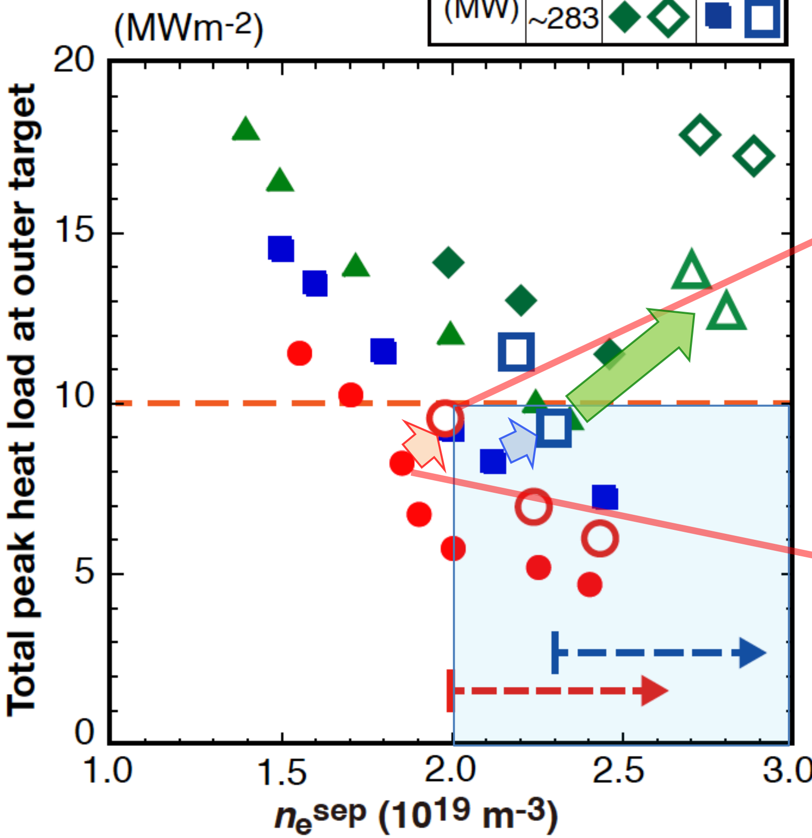
Peak-q_{target} for DEMO higher-κ and DEMO 2014 cases (**$f^*{}_{rad}^{div}$ ~0.78**) :

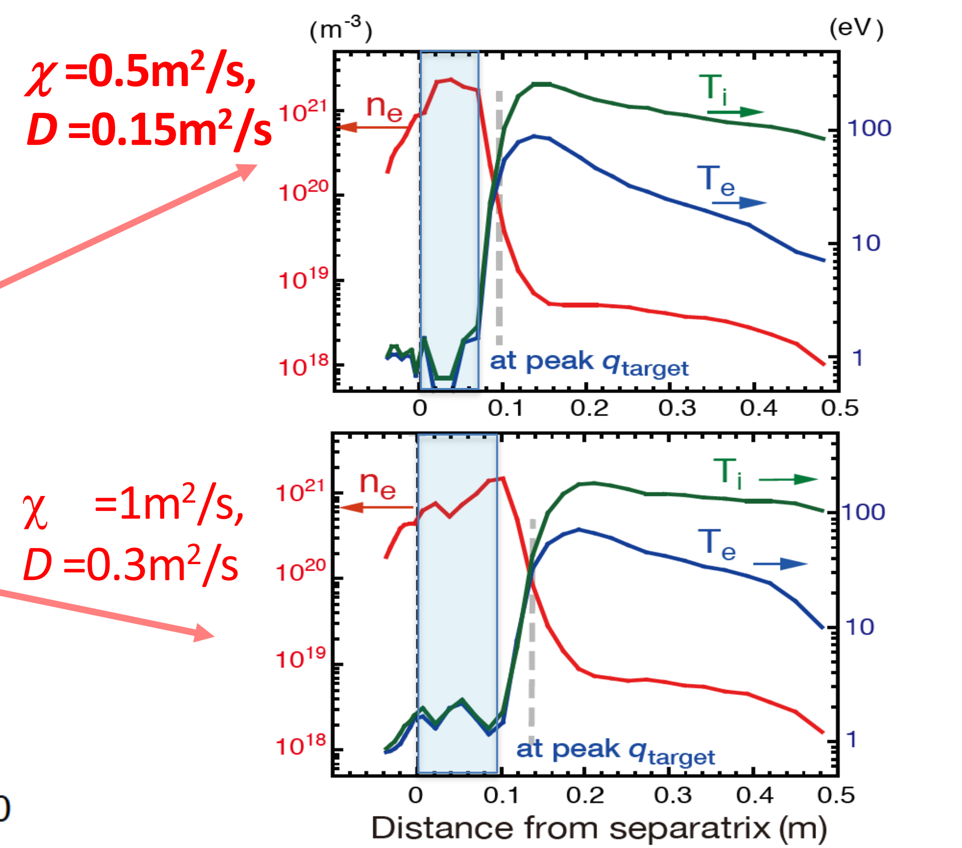
- Detachment region is reduced from 10 to 7 cm, and T_i^{div}, T_e^{div} *at attached region* increased

  ⇒ peak-q_{target} is increased, but acceptable for *higher-κ*, DEMO 2014: n_e^{sep} >2.3x10^{19} m^{-3}.

For low f_{rad}^{div} ~0.67 cases, divertor operation is difficult in the Low n_e^{sep} (2-3x10^{19} m^{-3}).

		$f^*{}_{rad}^{div}$	
		~0.67	~0.78
P_{sep} (MW)	~235	▲ △	● ○
	~283	◆ ◇	■ □





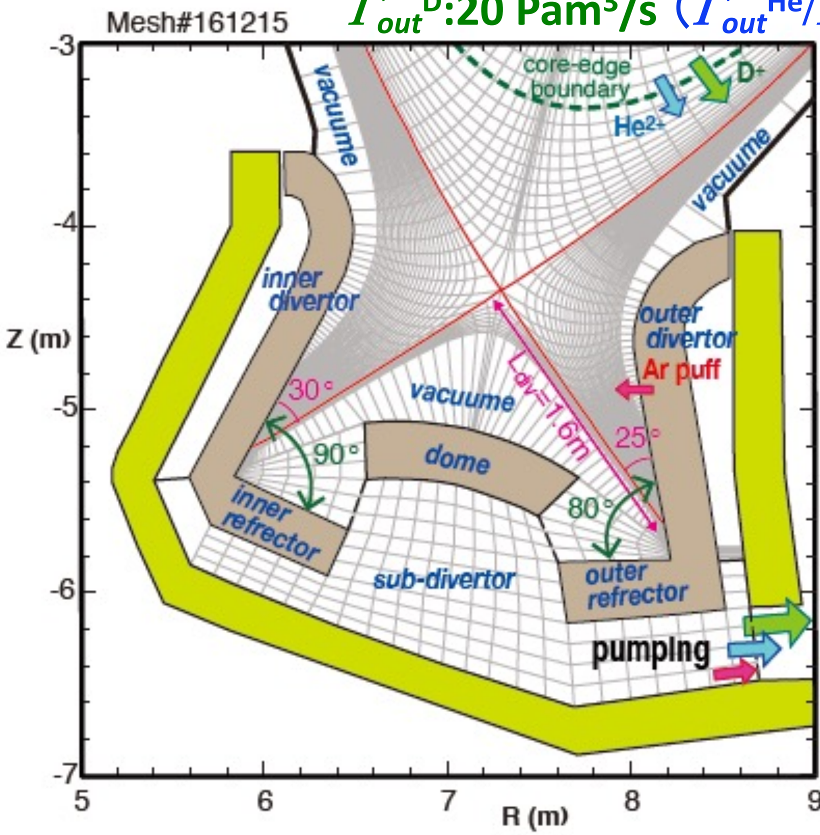
4. He exhaust study in Plasma edge and Divertor
He ion flux equivalent to P_{fusion}: 1.5GW is exhausted from *core-edge*

Simulation parameters for *He exhaust study*:

- He flux (Γ_{out}^{He}=5.3x10^{20} s^{-1}) is exhausted, corresponding to P_{fusion} =1.5GW (Γ_{out}^{D}=1x10^{22} s^{-1}).
- Diffusion coefficient (D_i, D_{imp} = 0.3m^2/s) is the same for D, He and Ar (the same as ITER calc.).
- Full MC cal. of He and Ar including Sub-divertor is performed once in every 10 IMPMC runs, and the pump and backflow fluxes are assumed in following 9 runs: **"Backflow model".**

Note: reflector was (2019-) opened from 60° to (in)90°/(out)80° (n-protection)

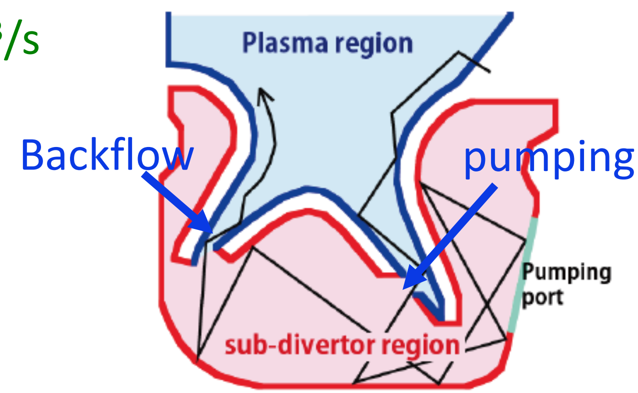
Γ_{out}^{D}:20 Pam^3/s (Γ_{out}^{He}/Γ_{out}^{D}~5%)



Gas puff from midplane
Γ_{puff} = 25-150 Pam^3/s

Same absorption probability is given for D, He, Ar.

"Backflow model": pumping and backflow fluxes are assumed.



Full IMPMC in Sub-divertor is calculated in the solution.

Plasma detachment and D^0/D_2 pressure in divertor

Wider reflector angle: plasma detachment and neutral pressure were similar

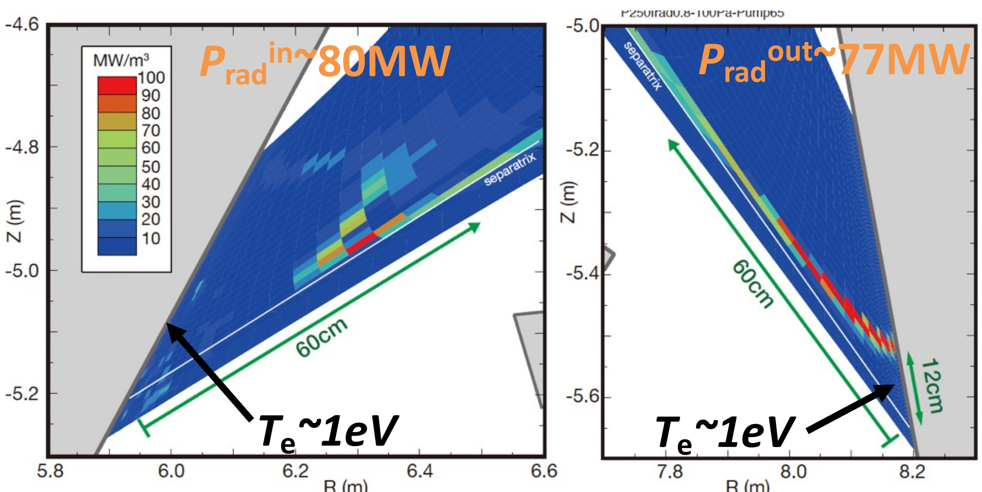
Gas puff 5.3×10^{22}D/s and Ar seeding : 3.9×10^{20}Ar/s,

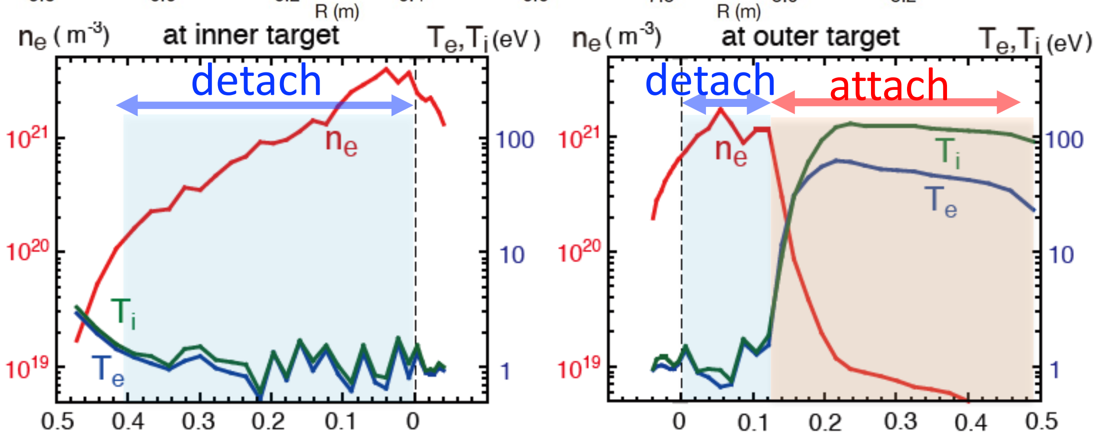
$f^{*\,div}_{rad}$ = 0.78 (He radiation loss is small fraction: 0.04)

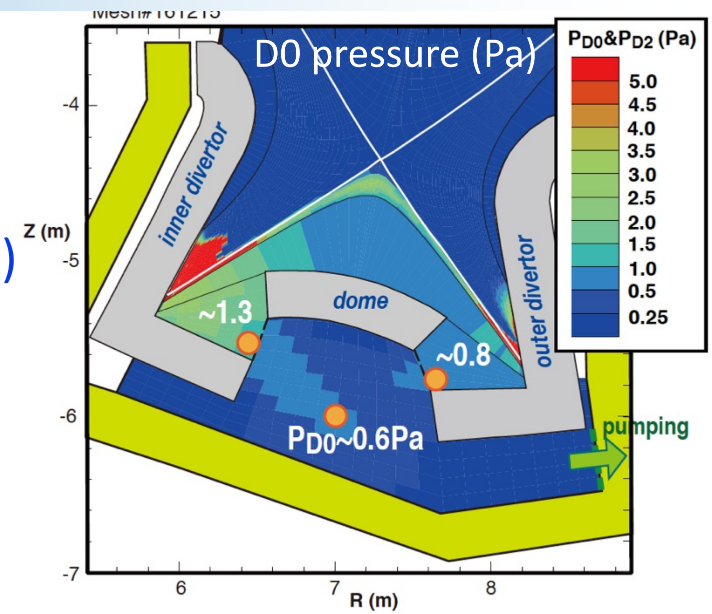
Inner target: Full detachment ($T_{e,i}$~1eV)

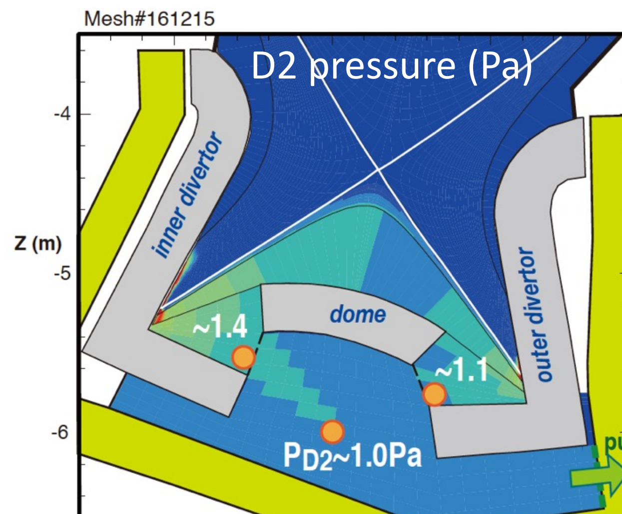
Outer target: Partial detachment ($T_{e,i}$~1eV in r^{div}<12 cm)

P_{D2} and P_{D0} are comparable at exhaust slots.







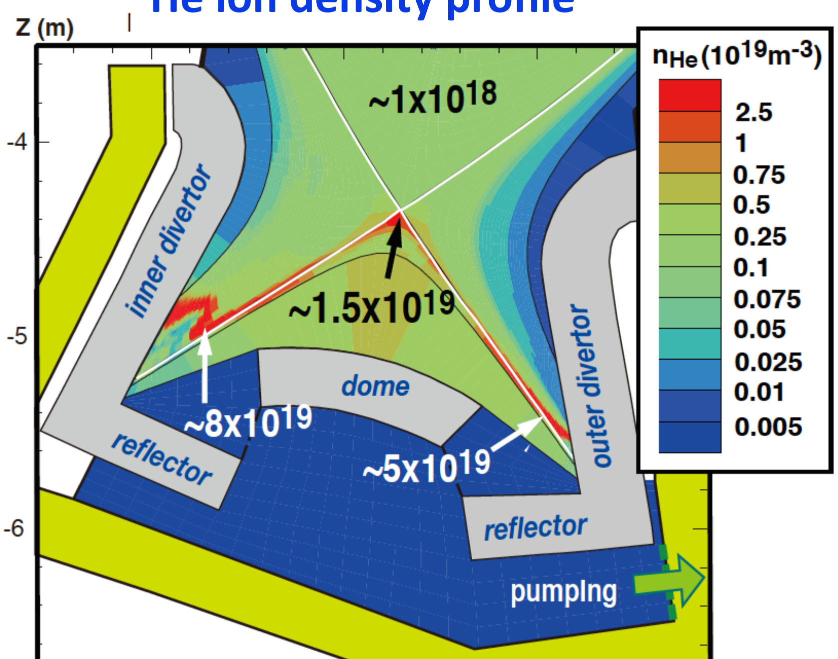


Elastic collisions of D0-D0, D0-D2, D2-D2 etc. are not considered.

He ion density in divertor and plasma edge

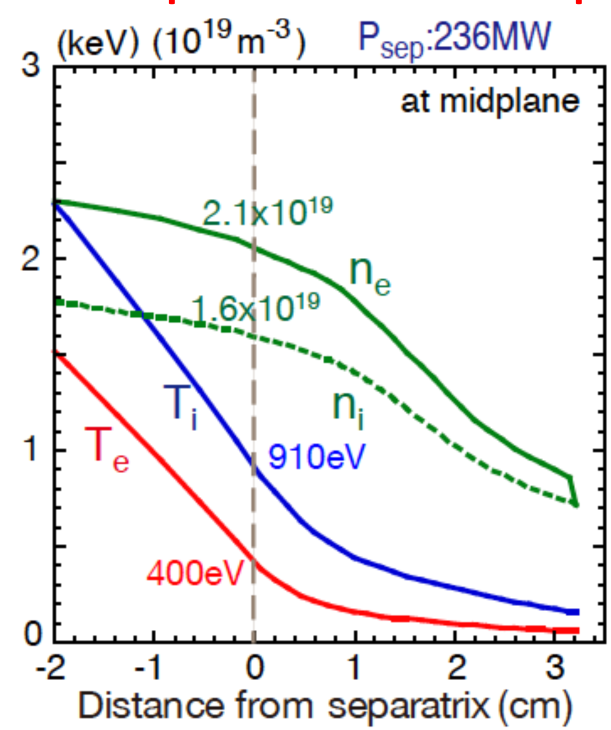
- He ion density (n_{He}) is significantly increased near the detachment front (between Ar radiation peak and D ionization front) due to recycling in the divertor.
- n_{He} is increased also near X-point (similar to D^+ density).
- n_{He}~1x10^{18} m^{-3} *inside the separatrix* (r^{mid}/a=0.96-0.98): n_e^{mid} is 25% larger than n_i^{mid} due to Ar and He ions (similar contributions to Δn_e^{mid}).

He ion density profile



Elastic collision of He^+-D^+, He^{++}-D^+ are not included

Plasma profiles at outer midplane



He concentration in detached divertor

C_{He}^{edge} = 4-7% similar to exhausting Γ_{He}/Γ_D: Accumulation of He is NOT seen.

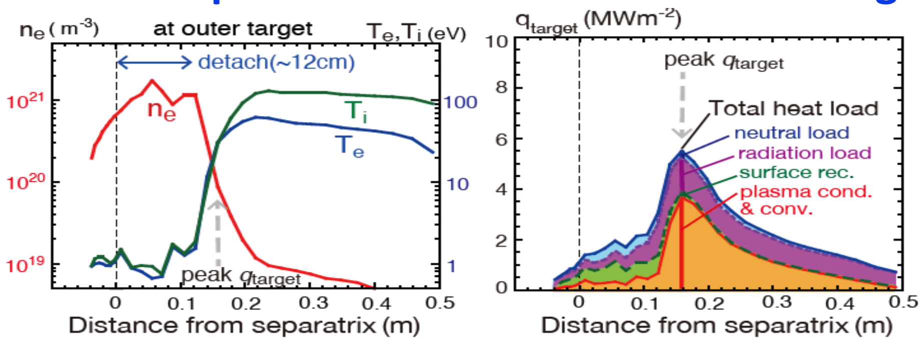
With increasing gas puff rate, *detachment width* increases and peak q_{target} is reduced.

He concentrations at SOL and plasma edge（$C_{He} = n_{He}/n_i$）:

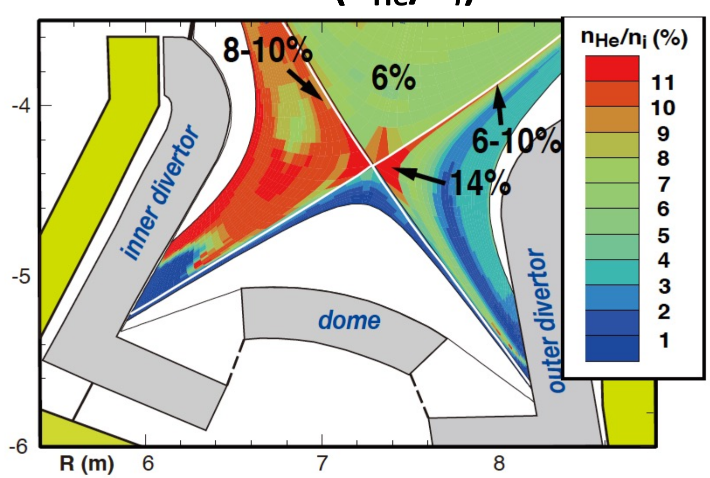
- In-out asymmetry of C_{He} in SOL/divertor is 2-3 times, but decreasing near separatrix.
- C_{He} = 4-7% at plasma edge（smaller than SOL）⇒ Accumulation of He is NOT seen.

Note: χ and D were reduced to half values⇒ C_{He} at plasma edge is increased to 7-9%.

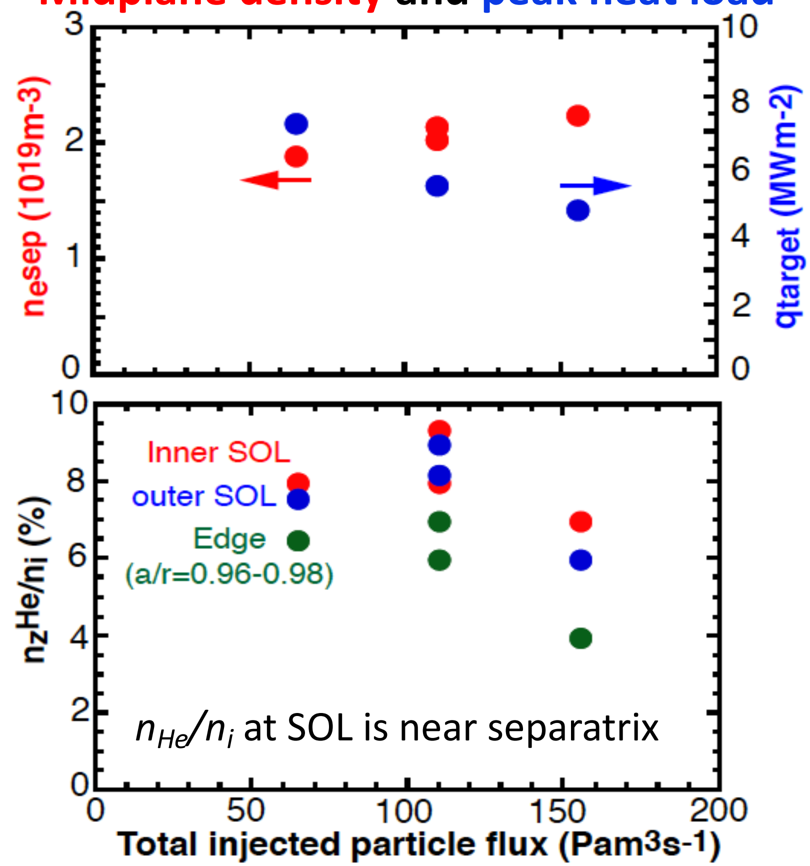
Profiles of plasma and heat load at outer target:



He concentration (n_{He}/n_i) in divertor



Midplane density and peak heat load



5. Summary: Simulation of JA DEMO divertor performance

Heat load and plasma detachment in a long-leg divertor (L_{leg}=1.6m) were evaluated for JA-DEMO 2014 and higher-κ (P_{sep}= 283/ 235 MW) in the low SOL n_e^{sep}= 2-3x10^{19}m^{-3}.

Divertor operation (≤10 MWm^{-2}) was determined with reducing $f^*_{rad}{}^{div}$ or/and χ & D ;

- Peak-q_{target} at *partially detached* outer divertor was increased with decreasing detachment width and increasing peak-T_e^{div} and T_i^{div} at the attached region.
- *Two references* ($f^*_{rad}{}^{div}$ ~0.78) was acceptable; higher-κ case allows larger operation margin.
- Severe cases of reducing $f^*_{rad}{}^{div}$ to 0.67 *or/and* χ & D to half values; higher n_e^{sep} was required. Particularly, impact of reducing χ and $f^*_{rad}{}^{div}$ was serious.

He exhaust for JA-DEMO (higher-κ) was simulated;

He densities in the divertor and edge were evaluated with enhancing the detachment.

- With increasing detachment width by increasing gas puff rate (but same f_{rad}^{div}~0.8), accumulation of He ion was not seen in the plasma edge: $(n_{He}/n_i)^{edge}$~4-7%.
- For the case with improving confinement, $(n_{He}/n_i)^{edge}$ to 7-9% is still acceptable.

Some other activities :

- **Benchmark of SONIC and SOLPS-ITER codes** both for EU- and JA-DEMOs (in BA-DDA).
- **Integration of transport codes, SONIC and TOPICS (main plasma)**, is in progress.
- **Renewing SOLDOR to incorporate drifts** is considered; now debugging in slab-model.

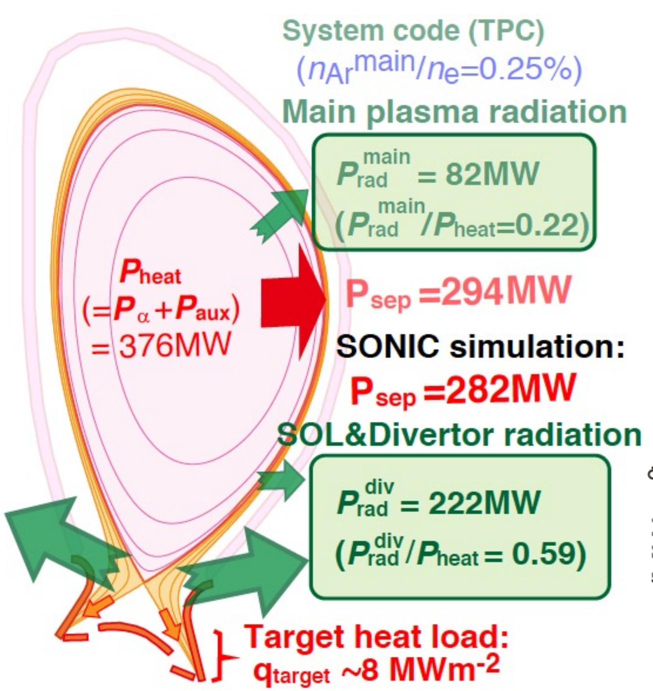
Power exhaust for DEMO 2014:
peak-q_{target} and $T_{e,i}^{div}$ increased with reducing detachment region

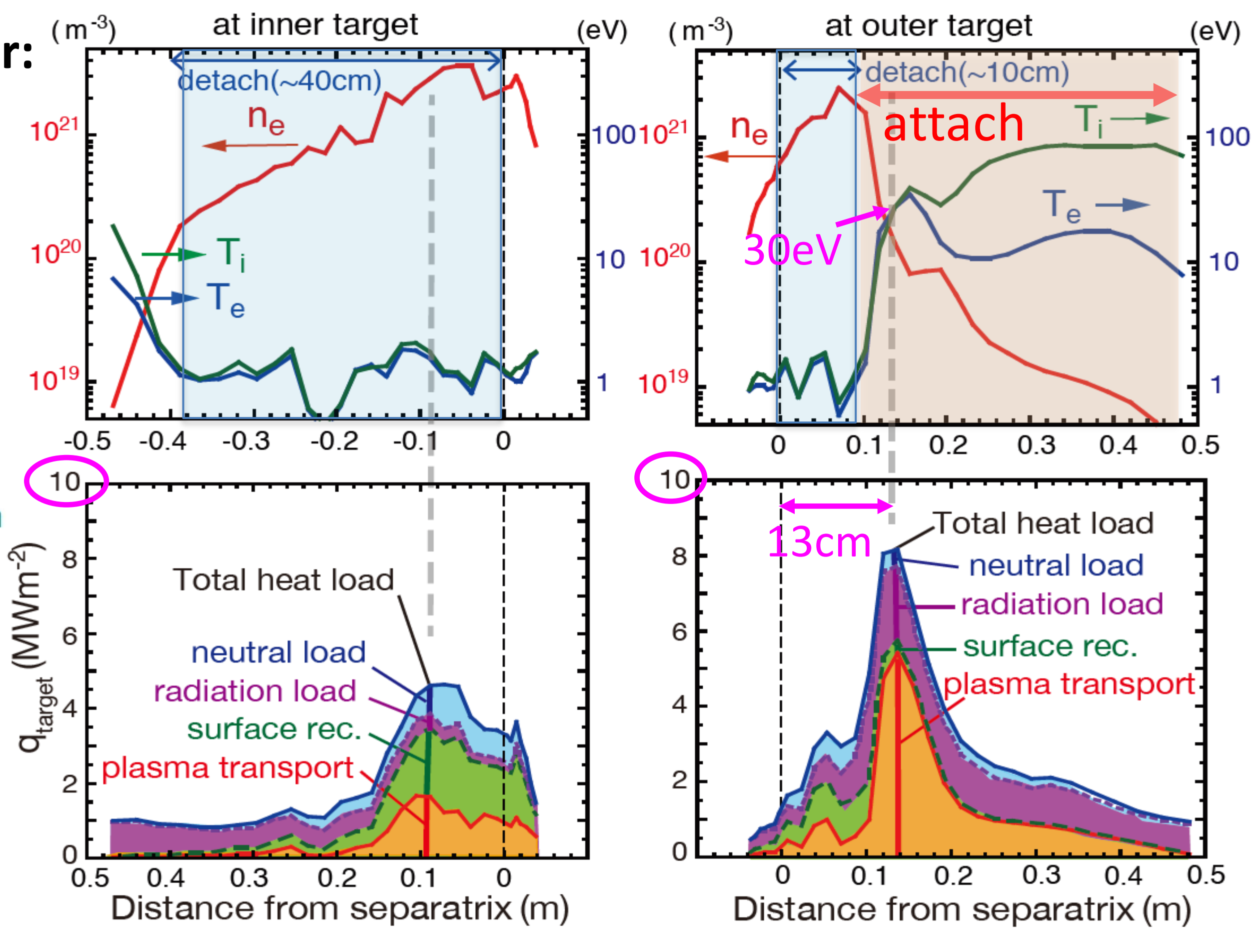
Divertor radiation fraction is the same: $f^*{}_{rad}^{div}=P_{rad}^{div}/P_{sep}$ = 0.78, where P_{sep} is larger.

Outer target: T_e^{div} and T_i^{div} become larger (~30eV) in the attached region (r^{div} ~13cm)

⇒ peak q_{target} ~8 MWm^{-2} and become closer to separatrix.

f_{rad}^{tot} = 0.81 is slightly smaller:





Other issues: Reduction in T_e^{div} and T_i^{div} at attached area is required such as "pronounced detachment: AUG"[12]

<u>Partially detachment for both ref. cases</u>: Low T_e^{div} = 20-30 eV is expected in low n_e^{sep}

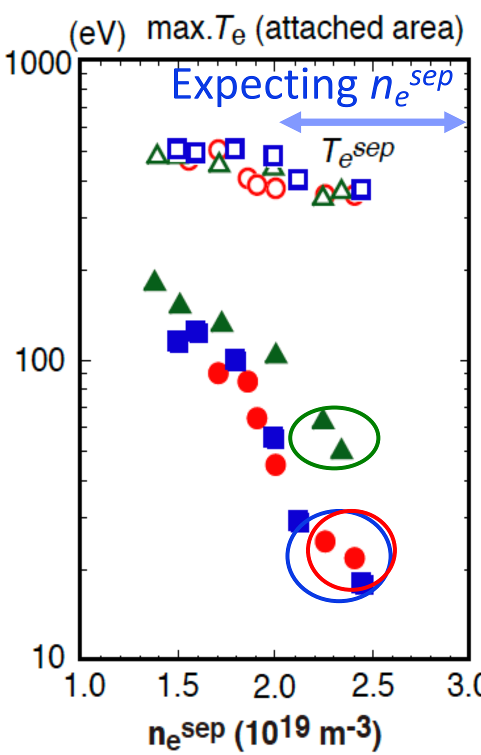
⇒ Evaluation of net-erosion rate and improvement of its accuracy are required.

For the low f_{rad}^{div} case, decreasing detachment width, and reduction in T_e^{div} is small.

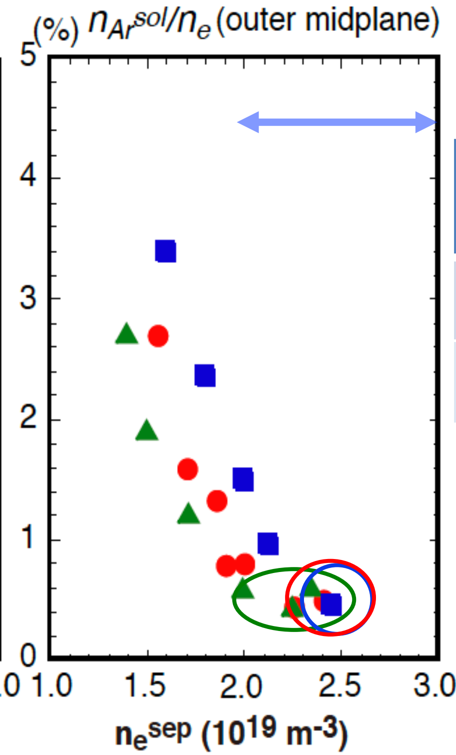
Exp. data and Modeling of erosion & transport (finite-Larmor effect[13]) must be improved.

<u>Impurity concentration in SOL</u> ($c_{Ar}^{SOL}=n_{Ar}^{SOL}/n_e^{SOL}$) : Increasing P_{rad}^{div} and Controlling core dilution are required ⇒ c_{Ar}^{SOL} (0.4-0.6%) is comparable to c_{Ar}^{main} in system code.

<u>T_e at attach plasma</u>



<u>Ar conc. in SOL</u>



<u>Estimation of net-erosion with 90% re-deposition</u>

Net erosion (Δd) becomes a half of W-width (d:5mm), if T_e^{div}~20eV at attached area.

Net erosion/yr(mm)	T_e=5eV	10eV	20eV
DEMO (steady state)	0.15	1	**2.5**
ITER(400s,2000shot)	0.004	0.026	**0.064**

attach plasma Γ_i~10^{23} m^{-2}s^{-1}, ~20eV <Z>=4,

n_{Ar}/n_i =0.2%, **<u>assuming net erosion: R_{net}=0.1</u>**

Sputtering yield with Ar Y_iC_i ~4×10^{-4} (at 20eV) [13]

$$\Delta d \text{ (mm)} = 4.95\times10^{-19} R_{net} * Y_iC_i * \Gamma_i * t\text{(year)}$$

[12] A. Kallenbach, et al., J. Nucl. Mat. (2011). [13] Y. Homma, et al., Nucl. Mater. Energy. (2017).

Distribution of He atom density in detachment

n_{He0}/n_{D2} in the divertor is also 4-6%, similar to that at plasma edge

- Neutral pressure ($P_{D2}+P_{D0}$) in the divertor is increased with gas puff rate.

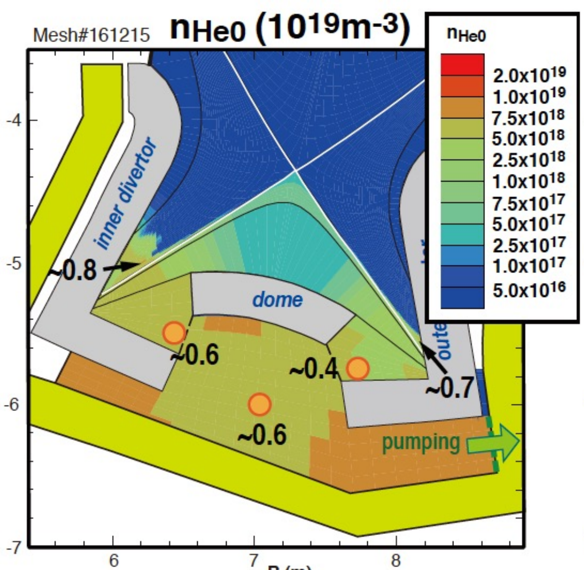
  Note: for large throughput cases, exhaust flux is smaller than total injected D flux.

He atom density (n_{He}) in the divertor:

- n_{He0} increases downstream of the ionization front.
- n_{He0} and n_{D2} are relatively uniform in the divertor

⇒ n_{He0}/n_{D2} is 4-6%: similar to that at the plasma edge.

He atom density profile



D gas density profile