Power exhaust and divertor physics issues for DEMO (Slim-CS)

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1. Introduction

- "***SlimCS***" aims P_{fus} ≤3 GW (P_{heat}= 600~700 MW) with A=2.6 and reduced-size CS.
  ⇒ Power exhausting to SOL is 5-6 times larger and R is smaller than ITER.
- Engineering requirement of power handting with W divertor target and pressurized water-cooling is 5-7 MWm^{-2}.
  ⇒ Power handling is the most important issue for the reactor design.

SlimCS

Burning core plasma

P_{fus} = 3 GW

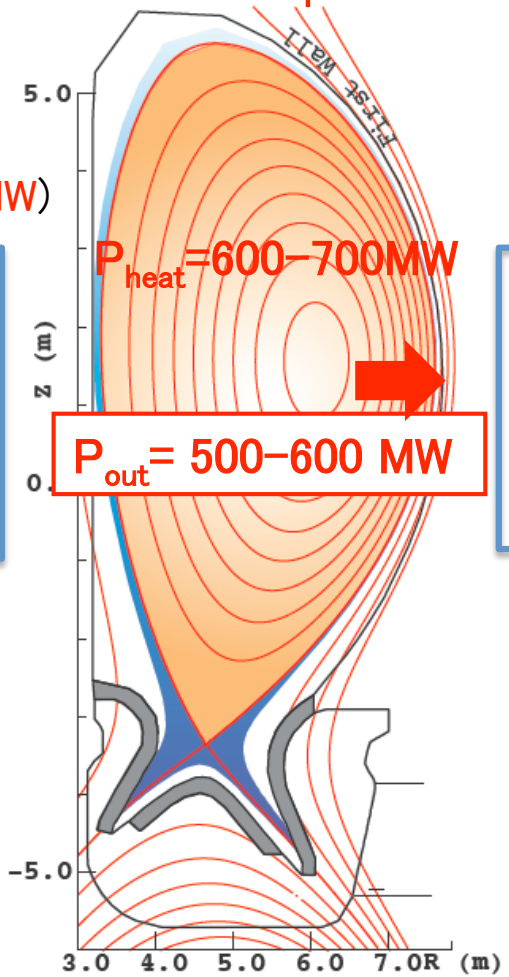
(P_{heat}=P_{α}+P_{ax}=600~700 MW)

P_{out} = 500–600 MW

P_{rad}= 450 ~ 550 MW

2xP_{target} = 50 MW

Major radius : R_p=5.5 m
Minor radius : a_p=2.15 m
Plasma current : I_p=17.5 MA
Toroidal field : B_t=6.0 T
Safety factor : q_{95}=4.7
Ellipticity : κ_{95}=2.0
Triangularity : δ_{95}=0.27
Plasma volume: V_p=924 m^3



ITER

Burning core plasma

P_{fus} = 0.5 GW

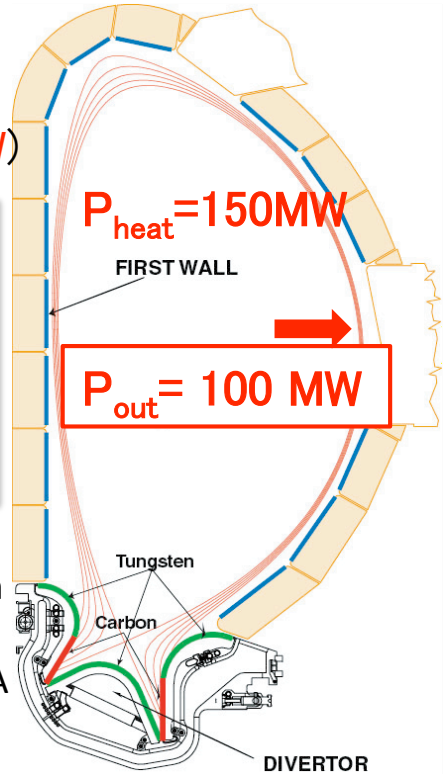
(P_{heat}=P_{α}+P_{ax}=150 MW)

P_{out} = 100 MW

P_{rad} ~50 MW

2xP_{target} = 50 MW

Major radius : R_p=6.2 m
Minor radius : a_p=2.0 m
Plasma current : I_p=15 MA
Toroidal field : B_t=5.3 T
Safety factor : q_{95}=3.0
Ellipticity : κ_{95}=1.7
Triangularity : δ_{95}=0.35
Plasma volume : V_p=840 m^3



Divertor concept for DEMO reactor

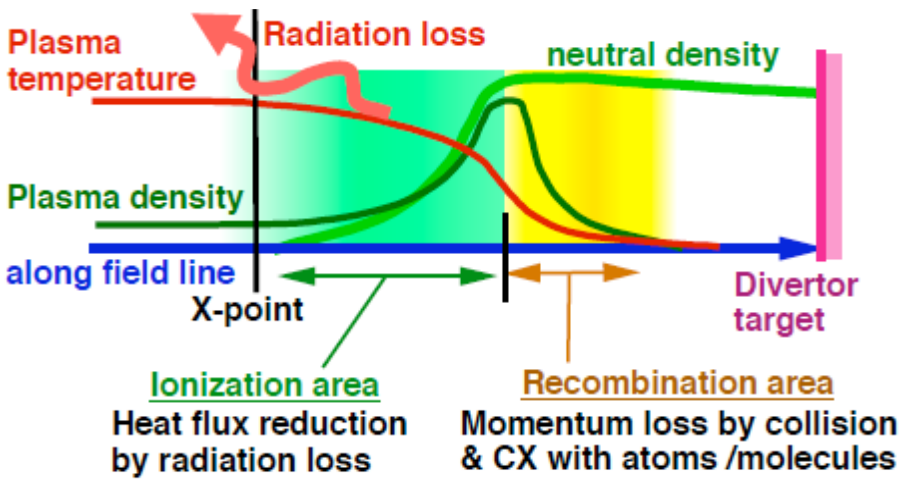
Power handling in *SlimCS* is investigated using SOLDOR/NEUT2D code, where design concept for the ITER divertor is applied/extended to the DEMO divertor: formation of *partially "divertor detachment"* (T_e~ a few eV) is a key for the power handling

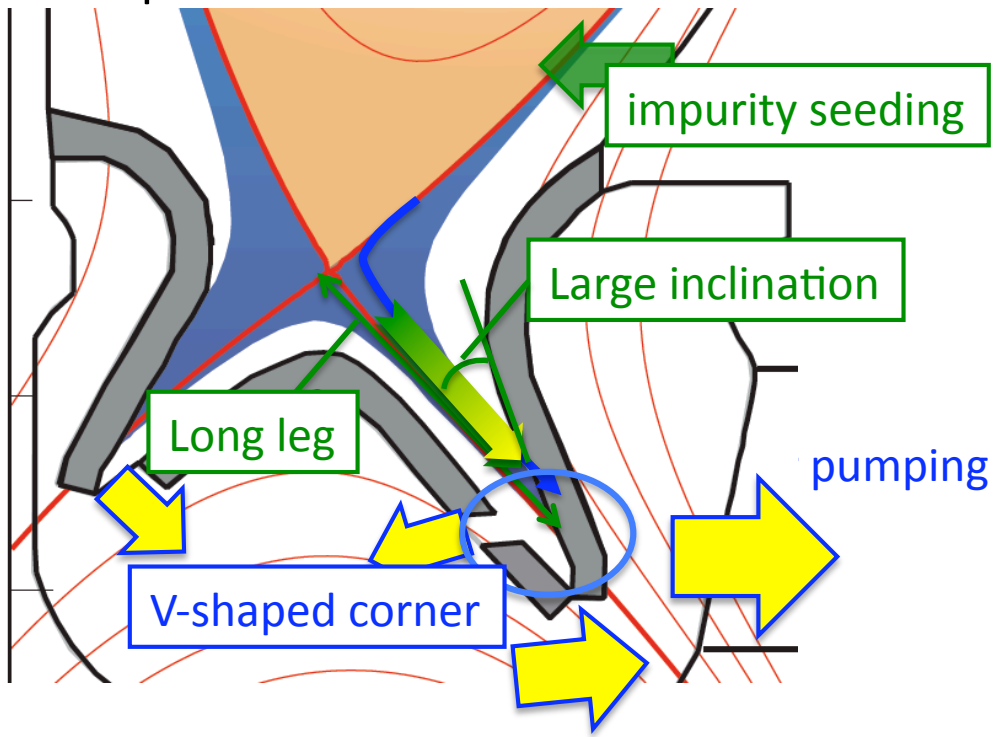
(1) **Divertor leg** and **inclination of the target** are larger than ITER
⇒ increase radiation, CX & volume recombination at upstream, reducing q^{target}.

(2) V-shaped corner ⇒ enhance neutral density at the separatrix

(3) impurity seeding such as Ar (Ne, N_2, Kr, Xe, etc)

(4) Δ_{sep_mid}=3 cm is guided into the divertor for particle control in the divertor





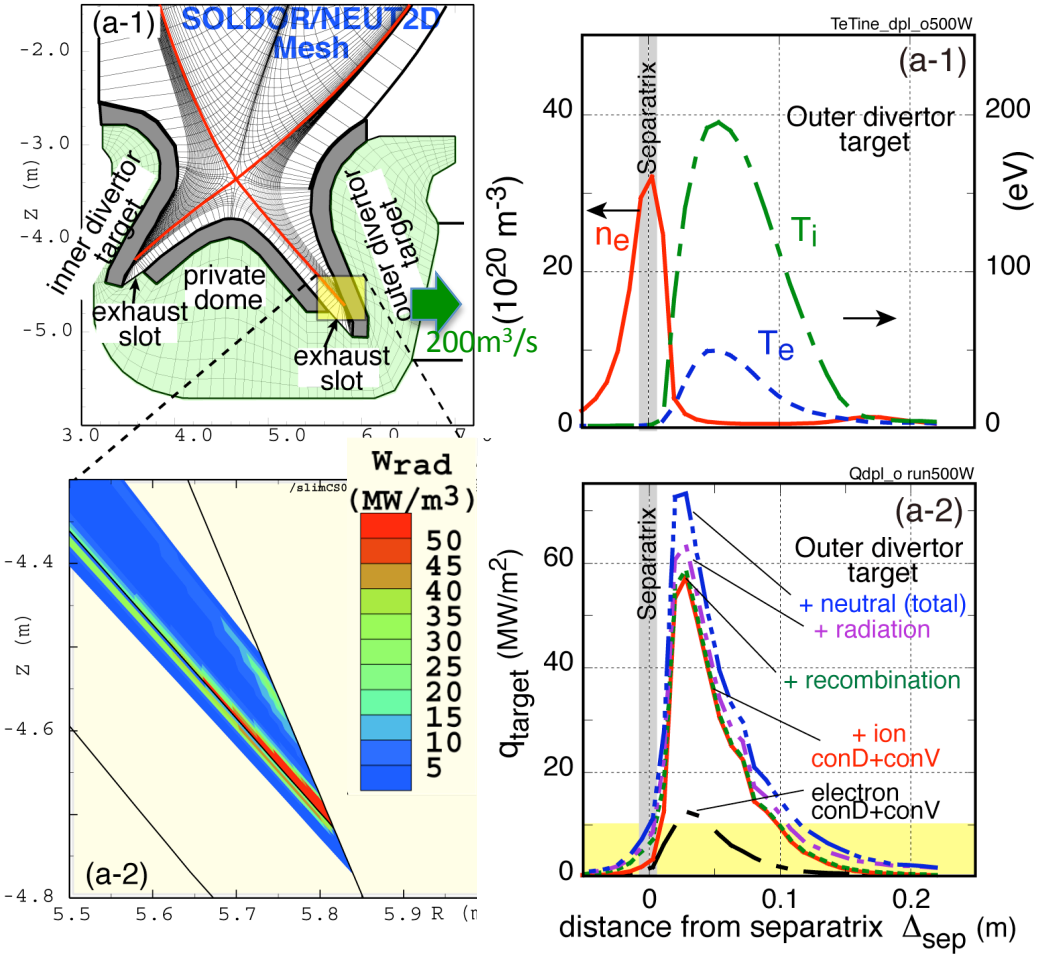
2. Simulation of power handling in the SlimCS divertor

Results from 2008 IAEA FEC, Geneva: FT/P3-10 H. Kawashima, et al. "Divertor Design Study on Compact DEMO Reactor for Handling of Huge Exhaust Power"

Input parameters at edge-SOL

P_{out}= 500 MW, Γ_{ion}=0.5x10^{23} s^{-1}

$\chi_i = \chi_e$ = 1 m^{2}s^{-1}, D = 0.3 m^{2}s^{-1}



Case-1: gas puff rate and impurity fraction similar level to ITER EDA:

D_2/T_2 puff: Γ_{puff}=1x10^{23} s^{-1} (200 Pam^{3}s^{-1})

Ar fraction: $(n_{Ar}/n_i)_{o\text{-}div}$=2% , $(n_{Ar}/n_i)_{edge\text{-}SOL}$=1%

applying *non-coronal model*:$P_{rad} = L(T_e,\tau_r)\, n_z n_e$

Divertor pumping speed at exhaust duct:

S_{pump}= 200 m^{3}s^{-1} is given.

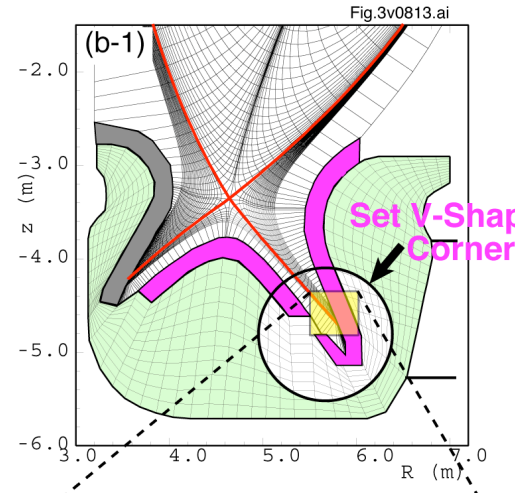
- At the inner divertor target, divertor is detached and q^{target} < 5 MW/m^2.

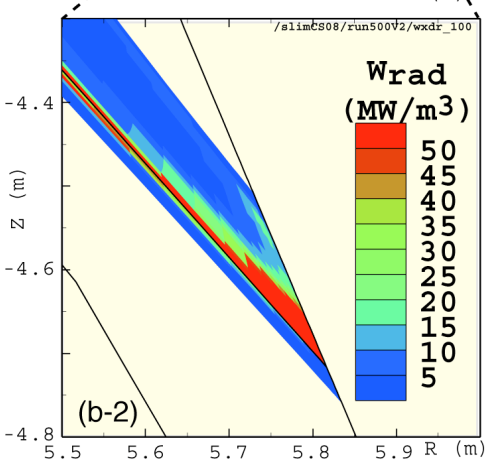
- At the outer target, significantly high temperature at the strike-point:

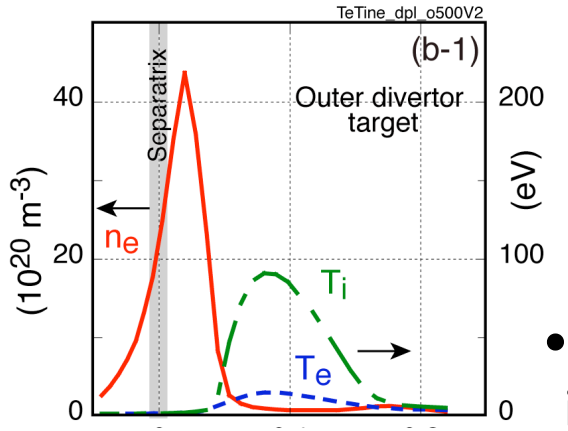
  peak T_e~50 eV and T_i~200 eV, giving **severe peak heat load ~70 MW/m^2**.

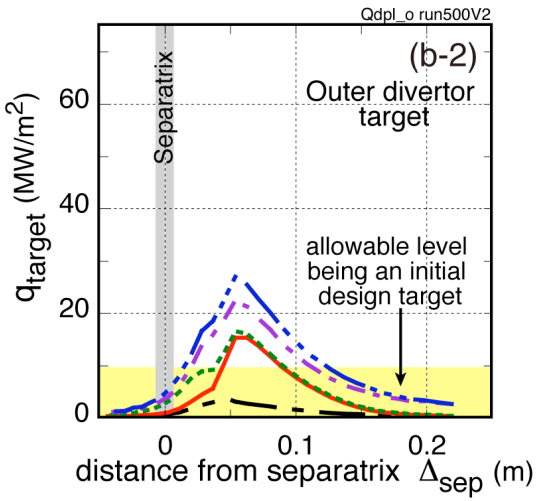
2.1 Effect of the divertor geometry: V-shaped divertor

- **Case 2: Design concept for the ITER divertor**, "***V-shaped divertor***", is investigated: Detachment occurs at the separatrix and Peak heat load is reduced to 27 MW/m^2.









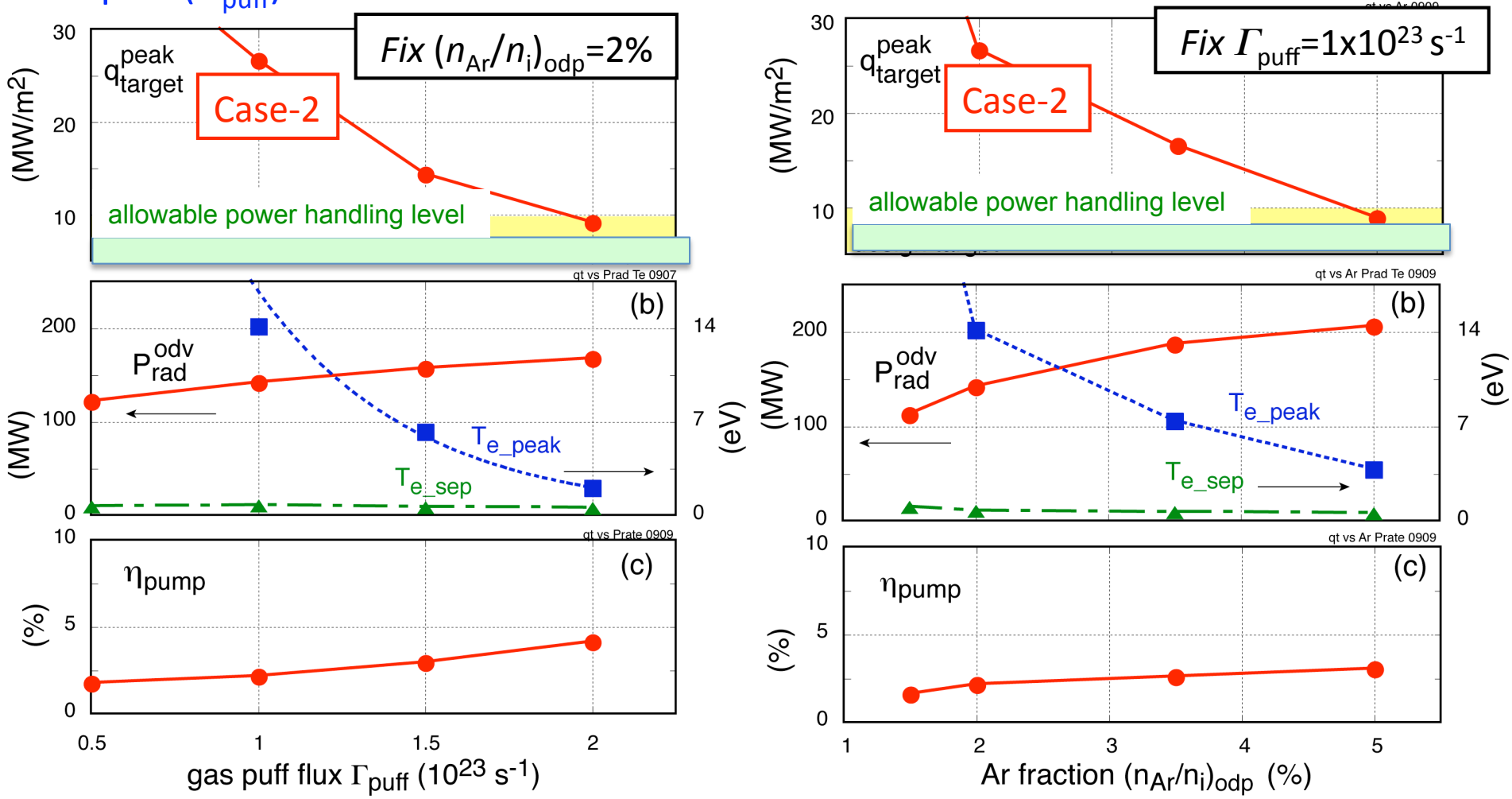
- *Radiation loss at the outer divertor* ($P_{rad}^{o\text{-}div}$) is increased at upstream of the strike point from 85 to 142 MW:
  Total P_{rad}(edge+divertor) = 425 MW
  (P_{rad}^{edge}~160MW, P_{rad}^{div}~260MW)
- *Divertor recycling flux* ($\Gamma_i^{o\text{-}div}$) is increased from 3.7x10^{24} to 4.2x10^{24} s^{-1}
- *Peak T_e~20 eV ,T_i~90 eV* are smaller by the factor of 1/2-2.5.
  ⇒ **Peak heat load can be reduced to 27 MW/m^2.**
- *Conduction/convection heat load is reduced, while* ***recombination and radiation power load increase in the low T_e divertor.***

2.2 Gas puff and Impurity (Ar) seeding in V-shaped divertor

Additional gas puff/ Ar seeding will enhance divertor radiation for q_{target} ~ 9 MWm^{-2}

- Gas puff (Γ_{puff}) is increased to 2x10^{23} s^{-1}

Ar fraction is increased to $(n_{Ar}/n_i)_{odp}$ =5%

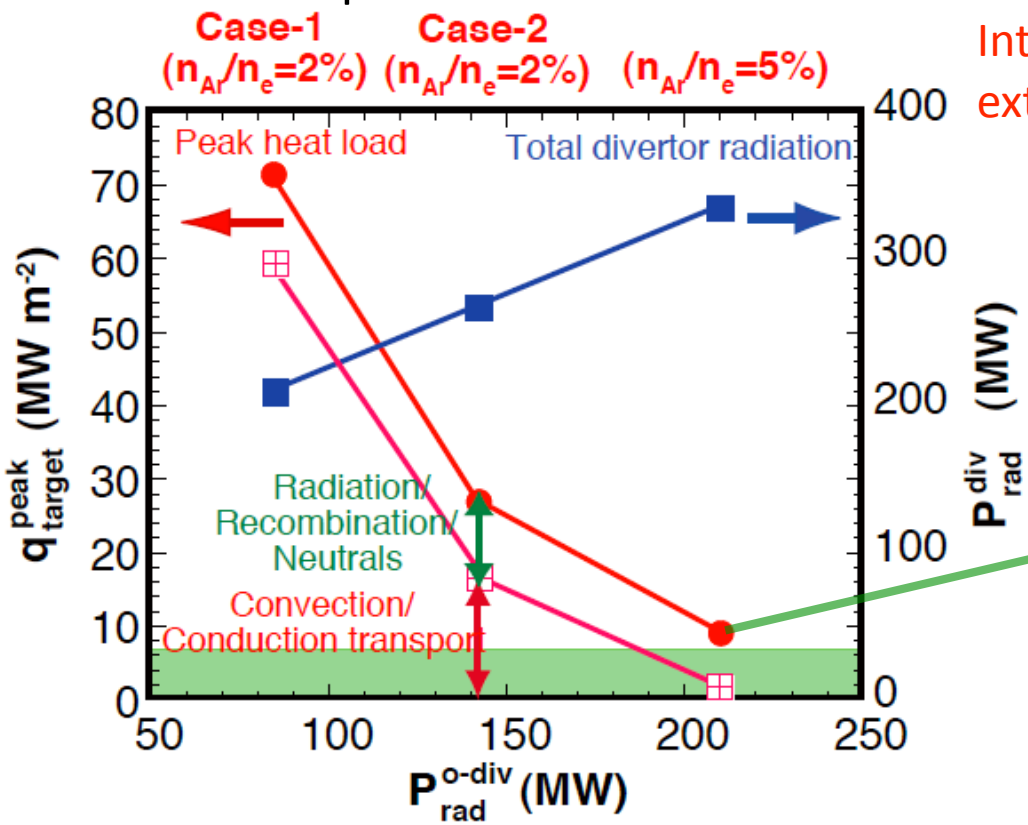


- Pumping rate η_{pump} is increased from 2% to 3-4% by increase the neutral density.

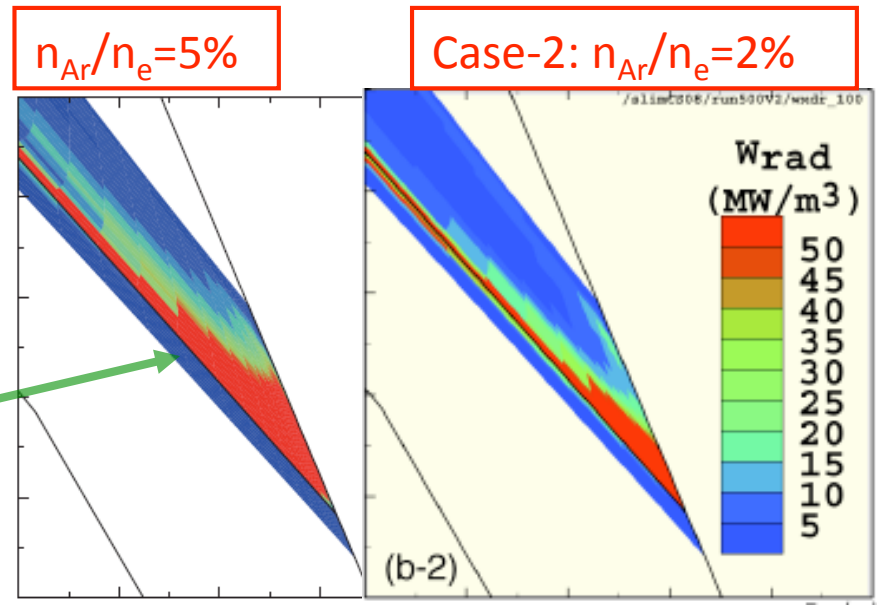
2.3 Power handling in the detached divertor

- **Peak heat load in the detached divertor** with $(n_{Ar}/n_i)_{odp}$ =5% is determined by ***Radiation/Recombination/ Neutral flux*** and ***their distributions in the divertor***.
- Large heat load will extend in a wide divertor area, and P_{rad}/P_{out} increased to >90%
  ⇒ P_{out} = 500 MW would be larger than handling power in this divertor size.

Modelling (understanding) of *atomic/molecule/impurity process* and *their transport* as well as *thermal stability of the divertor plasma under the high recycling condition* should be improved.



Intense radiation area (& plasma detachment) extending to wide and upstream in the divertor



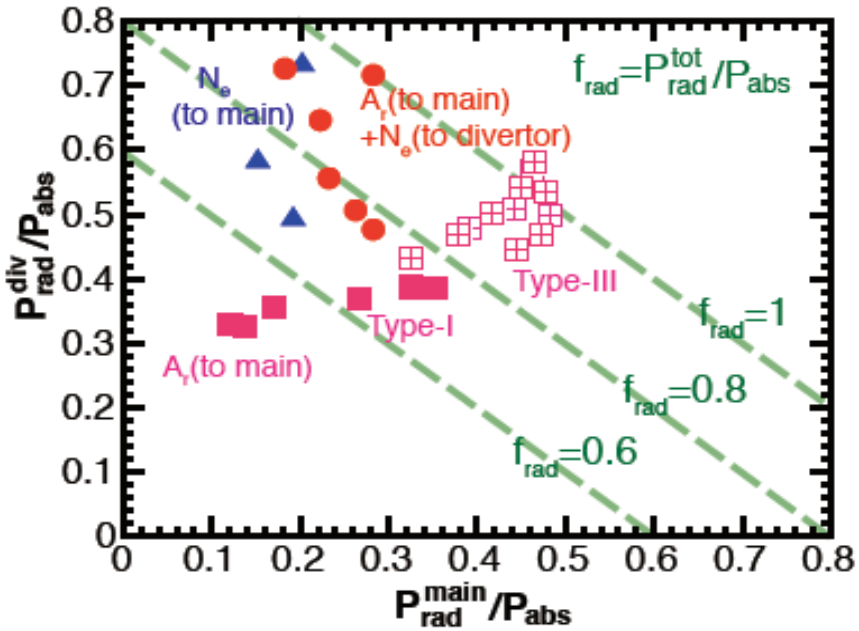
3.1 Issues for divertor design: impurity seeding

Large power handling at the SOL and Edge is required for DEMO operation:

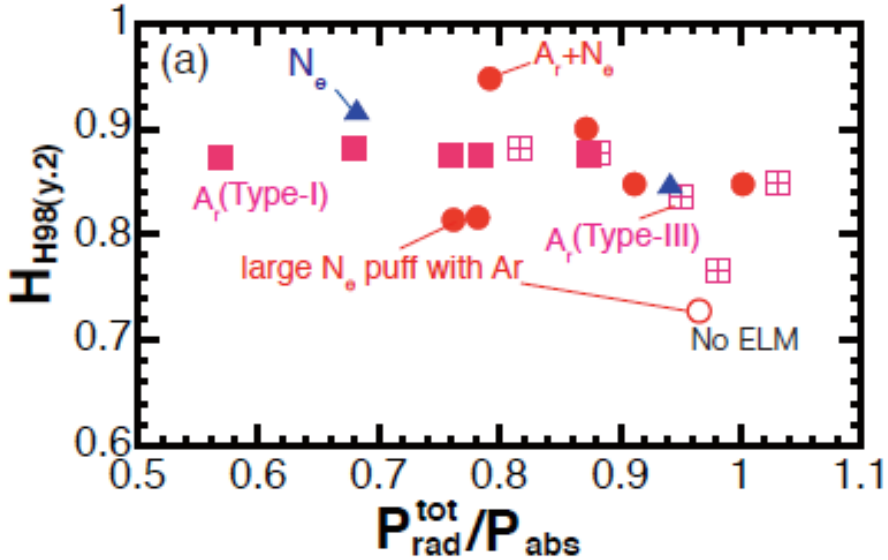
$P_{rad}^{SOL/edge}$ ~160MW ($P_{rad}^{SOL/edge}/P_{out}$~33%) for the simulation of $(n_{Ar}/n_i)_{edge}$=1%

- **Power handling at the SOL and Edge** such as increasing n_{imp}/n_i and $P_{rad}^{SOL/edge}$ (distributions of impurity ions, n_e and T_e), is affected by those in the divertor.
- Operation of the large $P_{rad}^{SOL/edge}$ plasma (and high density) will be restricted also by **compatibility with the core plasma performance**.

Ex.1 Radiation fractions at main and divertor for Ar and/or Ne seeding in JT-60U ELMyH-mode



Ex.2 Confinement factor for Ar and/or Ne seeding in JT-60U ELMyH-mode



Results from 2008 IAEA FEC, Geneva: EX4-4Ra, N. Asakura, et al. "Investigations of impurity seeding and radiation control for long-pulse and high-density H-mode plasmas in JT-60U"

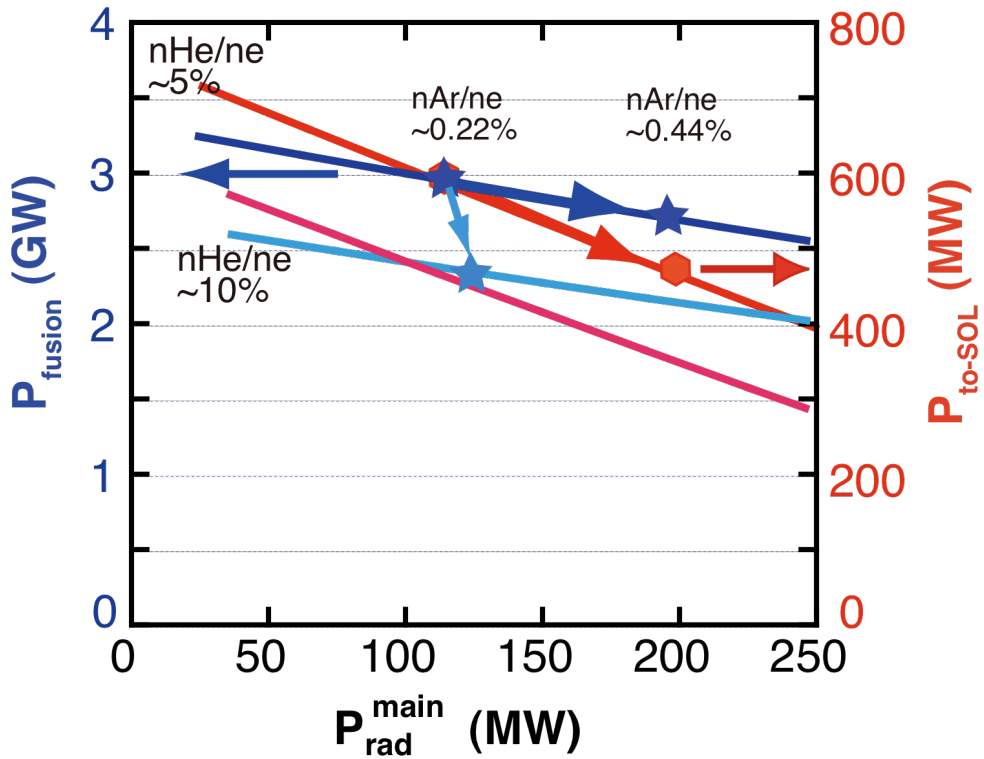
3.2 Issue for divertor design: dilution & He exhaust

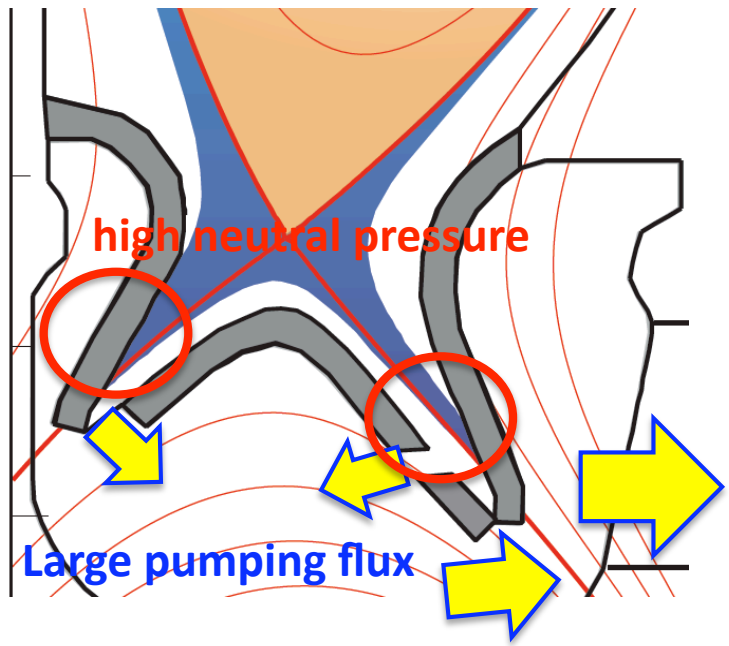
- **Ar seeding (n_{Ar}/n_e=0.22 ⇒0.44%)** reduces ***Power to SOL*** **($P_{to\text{-}SOL}$ = 600 ⇒ 480MW)**, with expense of *lowering Fusion power*: P_{Fusion} = 3.0 ⇒ 2.7GW (-10%).
- **He concentration (n_{He}/n_e=5 ⇒10%) largely reduces P_{fusion} =3.0 ⇒ 2.2GW (-27%)**

  He exhaust is another important issue.
- **Particle control should be determined for optimizing the divertor performance:**

  He exhaust requires divertor pumping

⇔ **Divertor detachment is formed under high neutral pressure and radiation loss.**





4. Summary and issues for the DEMO divertor

- **Huge power handling (P_{out} =500MW) for the SlimCS divertor was investigated:**

Intense Ar seeding (n_{Ar}/n_e~5%) in the Long-leg and V-shaped divertor produces the detached divertor ($P_{rad}^{SOL/edge}/P_{out}$ ~33% and P_{rad}^{tot}/P_{out} > 90%)

⇒ target heat load q_{target}~9MW/m^2, where *radiation and neutral flux are dominant*, while Engineering requirement for W target & pressurized water-cooling: 5-7MWm^{-2}.

***Scenario of the huge power handling (divertor geometry and impurity seeding) for P_{out}~ 500MW* (P_{fusion} ~3GW DEMO) in the ITER-size divertor is difficult.**

Reduction in P_{fusion} is a practical design within existing techniques.

- **Large power handling at the main SOL and Edge is required for the DEMO plasma:**

Restrictions of the power handling (n_{imp}/n_i and $P_{rad}^{SOL/edge}$) such as due to **influence of the divertor** and **degradation of core plasma performance** should be determined.

- **Particle control design should be optimized to satisfy the DEMO divertor functions:**

He exhaust requires the large divertor pumping

⇔**Formation of the detached divertor requires high neutral pressure and radiation.**

Design parameters of SlimCS

Major radius, R_p	5.6 m	He fraction, f_{He}	5%
Minor radius, a	2.1 m	Ar fraction, f_{Ar}	0.23%
Aspect ratio, A	2.6	Total stored energy, W_{tot}	1,170 MJ
Ellipticity, κ_{95}	2.0	Hot ion stored energy, W_h	299 MJ
Triangularity, δ_{95}	~0.35	Total beta, $<\beta>$	5.75%
Safety factor, q_{95}	5.4	Hot ion beta, $<\beta_h>$	1.38%
Toroidal field, B_T	6.0 T	Poloidal beta, β_p	1.56
Maximum field, B_{max}	16.4 T	Normalized beta, β_N	4.3
Plasma current, I_p	16.7 MA	Fusion output, P_{fus}	2,950 MW
Plasma volume, V_p	935 m^3	Current drive power, P_{CD}	60-110
Density, $<n_e>$	1.1×10^{20} m^{-3}	Bootstrap fraction, f_{BS}	70-80%
Peak density, $n_e(0)$	1.6×10^{20} m^{-3}	Enhancement of τ_E, H89P	2.59
Temperature, $<T_e>$	17 keV	Enhancement of τ_E, HHy2	1.3
Peak temp, $T_e(0)$	28 keV	Fusion Gain, Q	26-50
n_e/n_{GW} $(n_{GW} = I_p/\pi a^2)$	1	Wall load, P_n	~3 MW/m^2
Fuel purity	83.5%	Div. Peak heat flux	10 MW/m^2

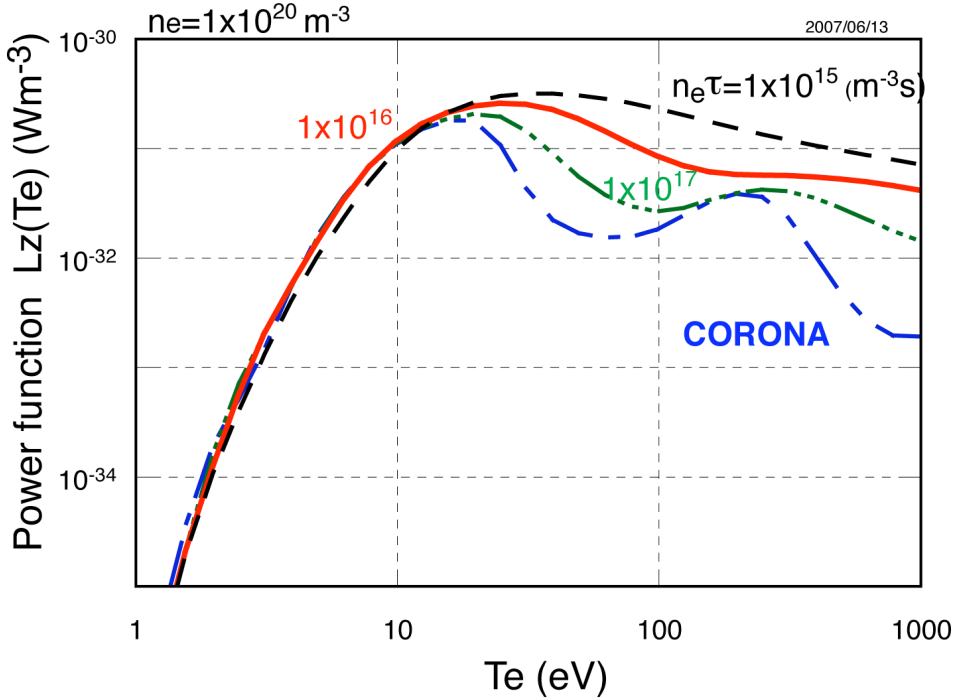
SOLDOR/NEUT2D code simulation

- **Coupling 2D plasma multi-fluid code and neutral MC code.**
- Evaluation of major heat load on the target

$$q_{\text{target}} = \underbrace{\gamma \cdot n_d \cdot C_{sd} \cdot T_d}_{\text{Transport component (electron\&ion-conduction/convection)}} + \underbrace{n_d \cdot C_{sd} \cdot E_{\text{ion}}}_{\text{Surface-recombination loss}} + \underbrace{f(P_{\text{rad}})}_{\text{radiation power load}} + \underbrace{f(\text{neutral})}_{\text{neutral power load}}$$

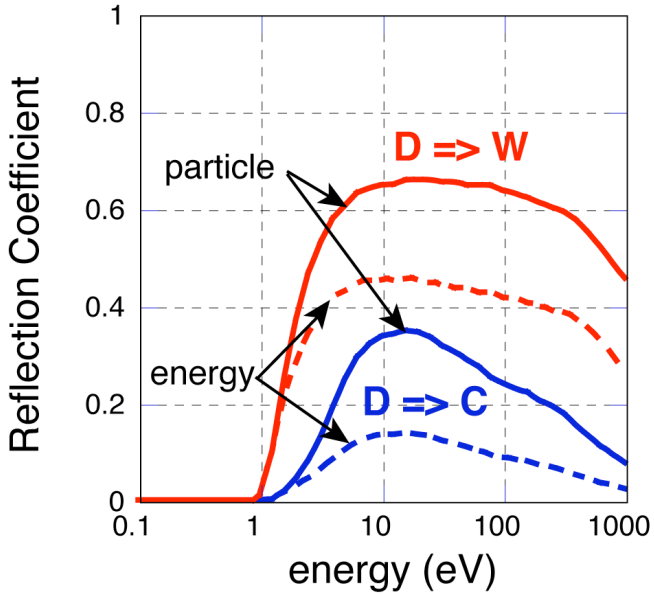
- Plasma boundary : r/a = 0.95
- Input parameters : P_{out}, Γ_{ion}
- Fixed transport parameters: D=0.3 m^2/s, $\chi_{e\perp}=\chi_{i\perp}$=1 m^2/s,
- Fixed particle exhaust parameters: S_{pump}=200 m^3/s, η_{rcycling}=1

Impurity and neutral treatment



Impurity radiation: non-coronal model

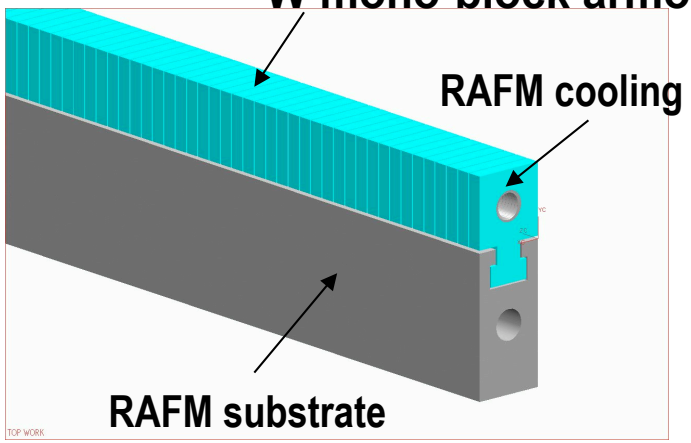
- *Ar* fraction (n_{Ar}/n_i) is given.
- Radiation loss power is evaluated by W_r=-$n_e n_z L_z(T_e)$.
- Loss rate $L_z(T_e)$ is used on $n_e \tau_{res}$=1x10^{16} s/m^3.

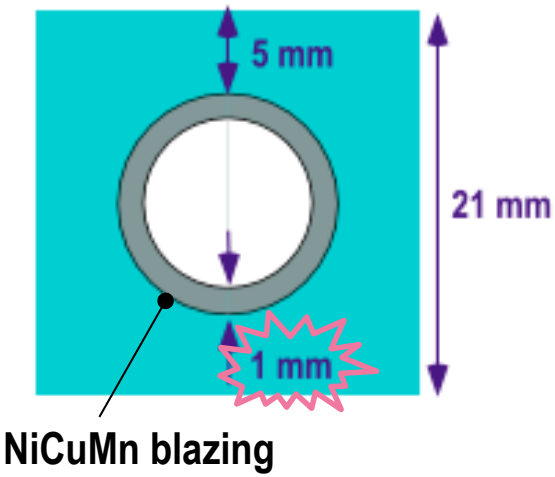


Neutral reflection: W target model

- Assumed *W* wall, which reflection coefficients of particle and energy are almost twice larger than that for carbon.

Original concept of divertor plate





	ITER	DEMO
Cooling tube	Cu-alloy	RAFM
Armor	CFC/Be/W	W
Coolant	water 100°C, 4MPa	water 200°C
Heat load	≤ 20 MW/m^2	≤ 10 MW/m^2

Thermal conductivity 1/10 of Cu

1/2 of ITER
Handle power of x6

Ultimate design w/o margins!

Possible modifications of the concept

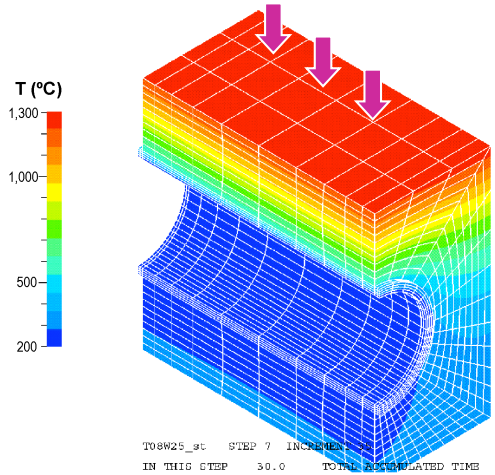
- **Coolant inlet temperature: 200ºC → 290ºC**
  - **Avoid corrosions of F82H by radiation-produced hydrogen peroxide (H_2O_2)**
    H_2O_2 starts to pyrolytically decompose at 240ºC
  - **Avoid irradiation brittleness of F82H (lack of data at < 300ºC)**

Allowable heat flux:

q_{max} ~ 7 MW/m²

($T_{coolant}$ = 290ºC)

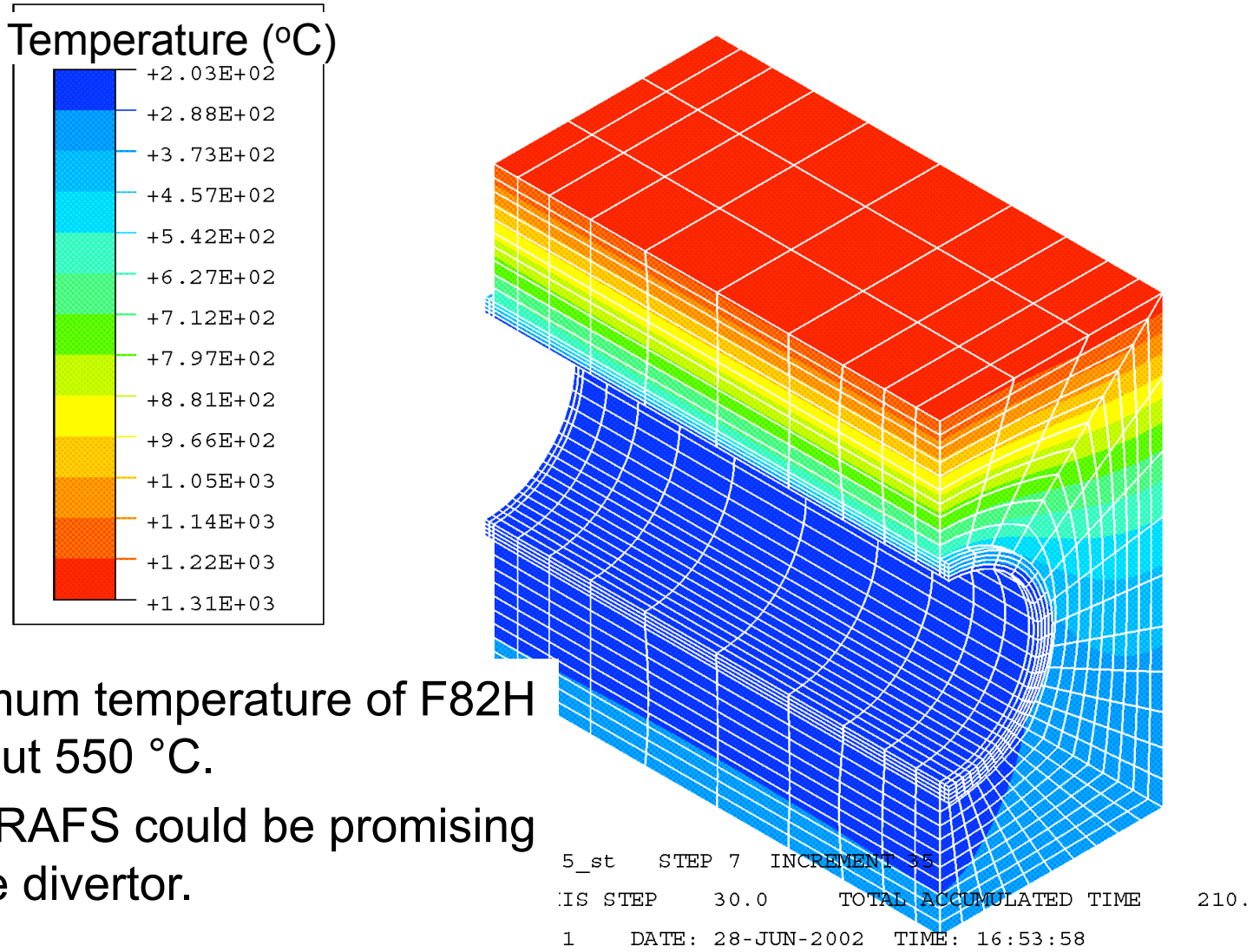
	Requirement
W	T ≤ 1,200ºC
RAFM	290ºC ≤ T ≤ 550ºC



- **Coolant tube thickness: 1 mm → thicker as possible**

Possible to increase the thickness to ~2mm if allowed to use F82H above 550ºC

F82H cooling tube can be applicable for the divertor



- Maximum temperature of F82H is about 550 °C.
- ODS-RAFS could be promising for the divertor.