What we have leared from SlimCS design study

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JAEA

Special thanks:

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OUTLINE

1. Background of SlimCS
2. Power Handling in SlimCS

   1) Blanket

   2) Divertor

   Short summary

1. Background of SlimCS

2000 — 2003 — 2010

JAERI | **JAEA**

Warm evaluation system

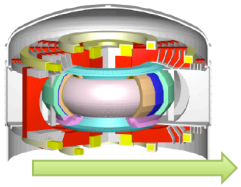
- *Flexibility on reseach issues*

Rigid evaluation system

- *Requested to implement a "midterm plan"*

"Midterm plan":
To conceive a low-A DEMO concept

A-SSTR2



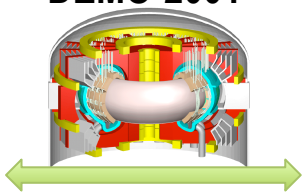
High field (23T)

→ CS-free tokamak

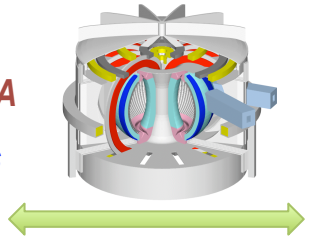
DEMO with SCW cooling

→ BLK concept

DEMO-2001



VECTOR

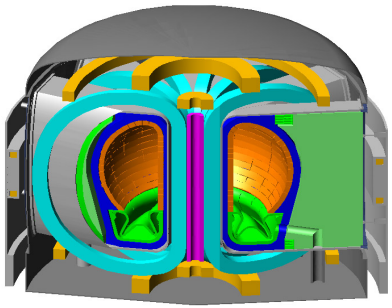


Supercond. low-A

→ Low-A merits

Lessons about CS and low-A

SlimCS



2009

- Final design report
- Next midterm plan

Requirements for DEMO

Definition of DEMO by AEC (2005):

1) Power core, as compact as ITER's

2) Electric output of plant level (~1 GWe)

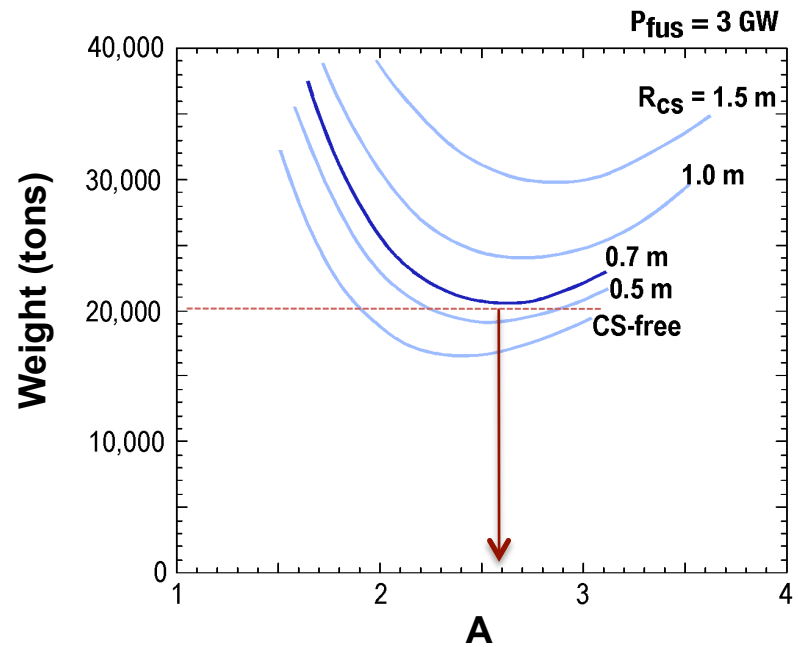
3) Year-long steady-state operation

4) Prospect of economic viability of fusion power

Report seems to demand more economic concept than SSTR

"Reduced size CS"

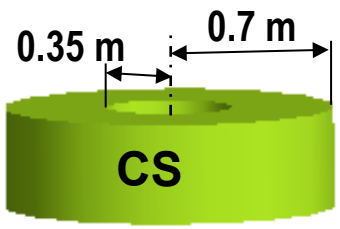
Reactor weight as a function of R_{CS}



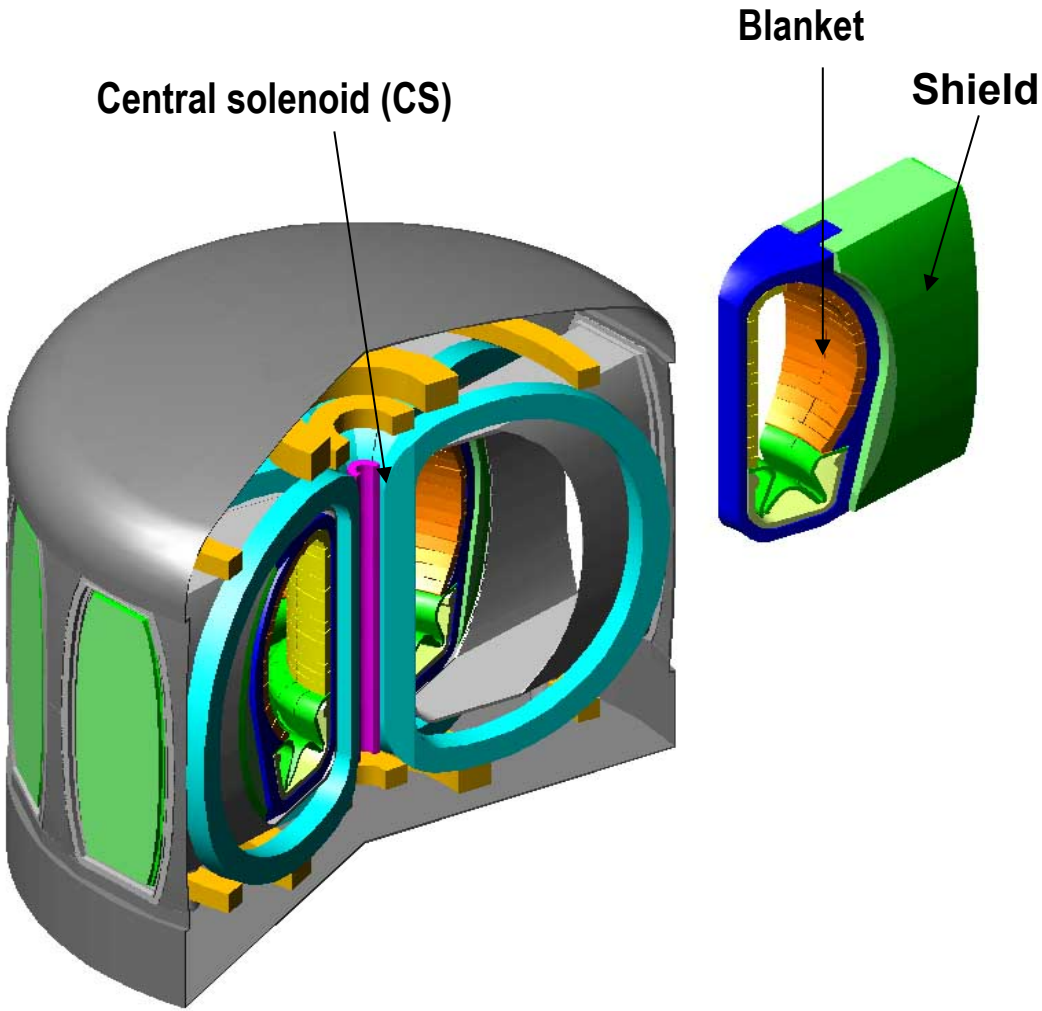
SlimCS — DEMO design

"SlimCS"

Central Solenoid



Major radius R_p	5.5 m
Minor radius a	2.1 m
Aspect ratio A	2.6
maximum field B_{max}	16.4 T
Fusion output P_{fus}	2.95 GW
Neutron wall load P_n	3 MW/m^2



Impact of "Slim CS"

"Small-size CS reduces E_{TFC}"

SlimCS: 50 GJ (B_{max}= 16.4T) ⟷ ***SSTR : 140 GJ (B_{max}= 16.5 T)***
ITER : 41 GJ (B_{max}= 11.8 T)

Virial theorem:

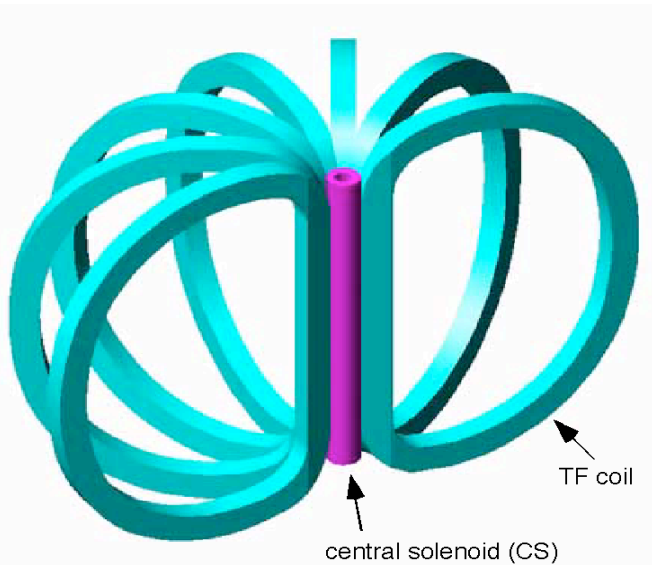
$$M = \frac{Q\rho E}{\sigma}$$

structure weight → M; geometric factor → Q; stored energy → E; design stress → σ

Construction cost ↘

TF coils: 6,200 tons

(cf. SSTR: 11,200 tons)



R_{CS} = 0.7m

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	~7 MW/m^2

2. Power handling in SlimCS

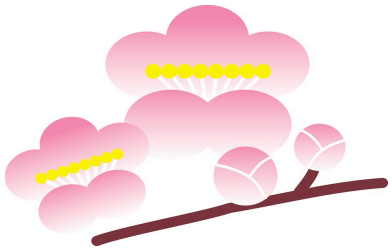
(1)Blanket

Consistency between T self-sufficiency and heat removal

(2) Divertor

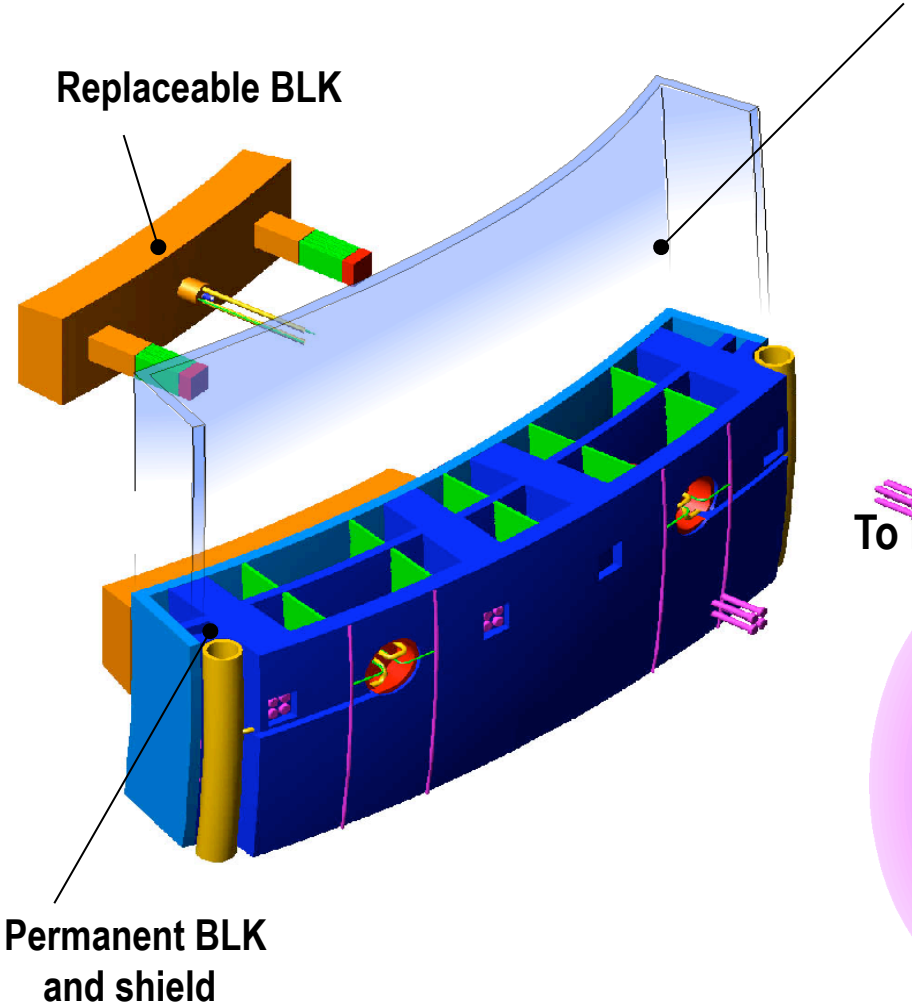
Tradeoff between engineering and physics

→ *Asakura (Physical aspect)*

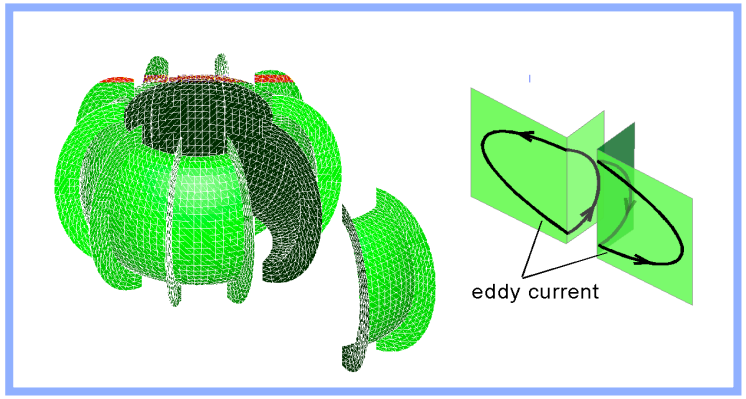


2.1 Power handling on blanket

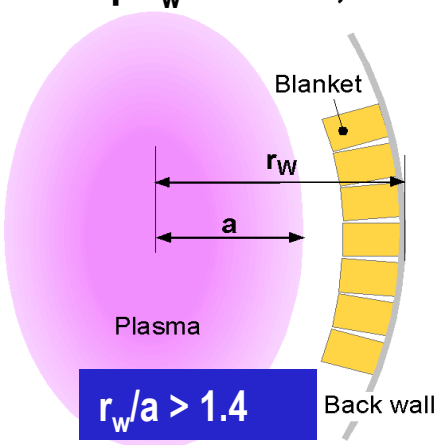
Segmentation of blanket

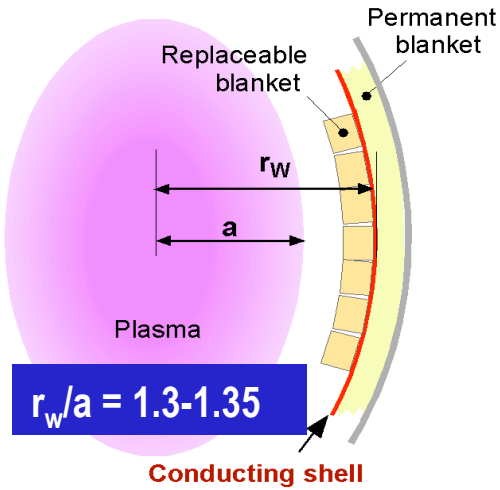


Conducting shell structure → { ✓Vertical stability ✓High beta



To keep $r_w/a < 1.35$, the shell is placed in between BLK

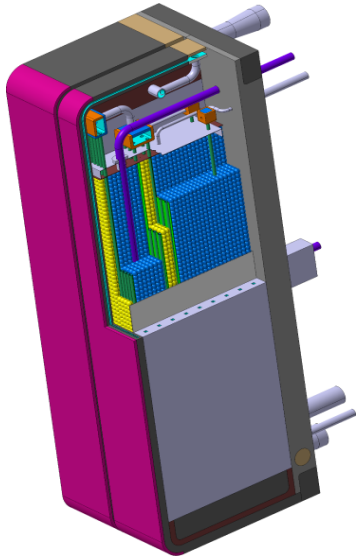




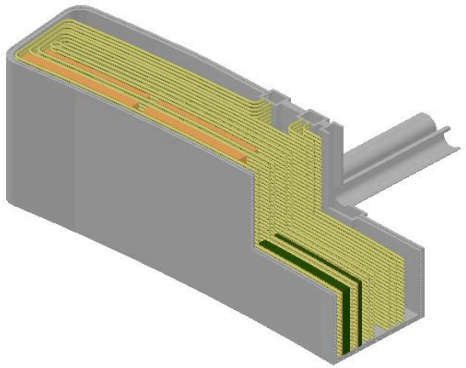
Requirements for blanket

Continuity with ITER-TBM

ITER-TBM



DEMO (SlimCS)

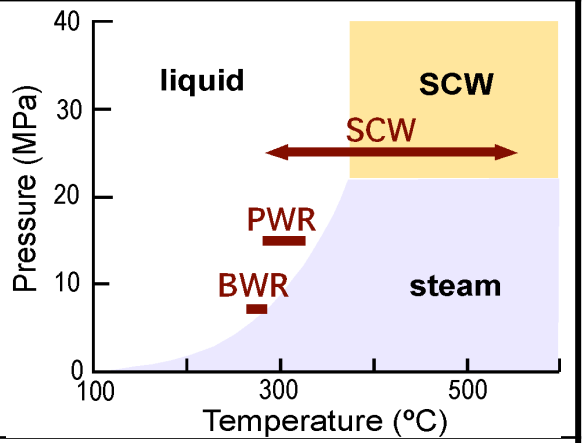


Structural material	F82H
Coolant	Water * 15 MPa, 280-325ºC
Neutron multiplier	Be or $Be_{12}Ti$
Tritium breeder	Li_2TiO_3 or other Li ceramics

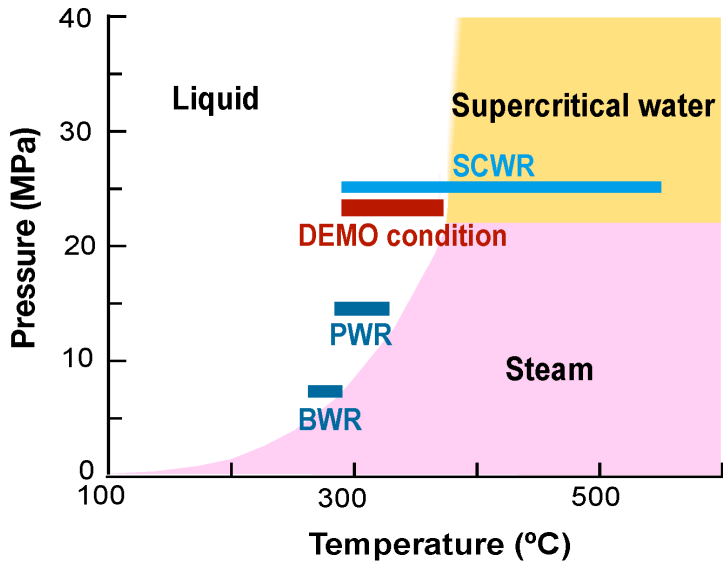
* Another proposal:He gas -- 8 MPa, 300-500ºC

Originally, considered the same coolant and materials

Previous selection of blanket materials (2007)

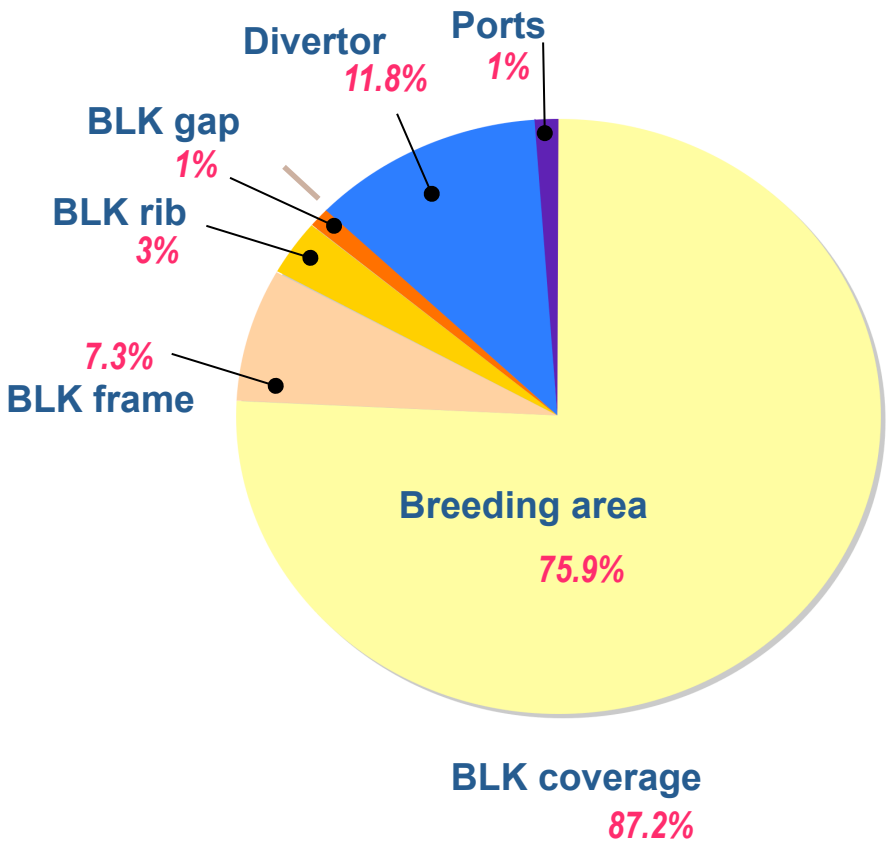
Structure	• **F82H** – Japanese prime option of ITER-TBM • ODS steel – premature
Coolant	• **Pressurized water** **(PWR condition: ~15MPa,285-325ºC)** –compatible with RAFM but low η_{th} • Supercritical water (~25MPa,280-510ºC) – concern about corrosion of RAFM 
Breeder	• **Li_2TiO_3** – good T recovery property, prime option of ITER-TBM 90%-enriched ^6Li required because of reduced TBR • Li_2O – concern about T recovery at high temperature
Multiplier	• **Be** – can meet TBR ~ 1.05, but concern about swelling • $Be_{12}Ti$ – TBR not sufficient when used with Li_2TiO_3

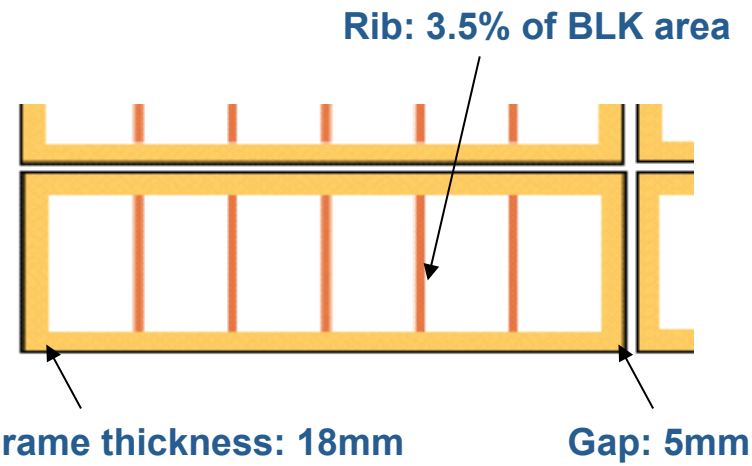
Present selection of blanket materials (2009)

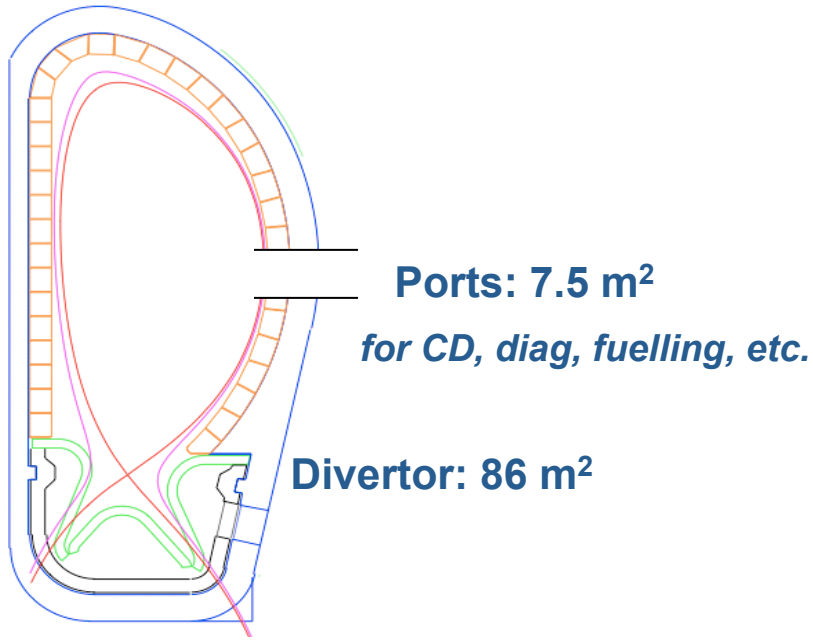
Coolant	**Subcritical water condition:** **290 - 360ºC (ΔT = 70ºC)** **~23 MPa** – Need larger ΔT for cooling the nucl. heating w/o reducing TBR 
Breeder	**1st opt.: Li_4SiO_4 pebbles** **2nd opt.: Li_2TiO_3 pebbles** – ^{6}Li 90%-enrichment (natural abundance 7.5%)
Multiplier	**Be plate and $Be_{12}Ti$ pebbles**

Required "local TBR" ≥ 1.38

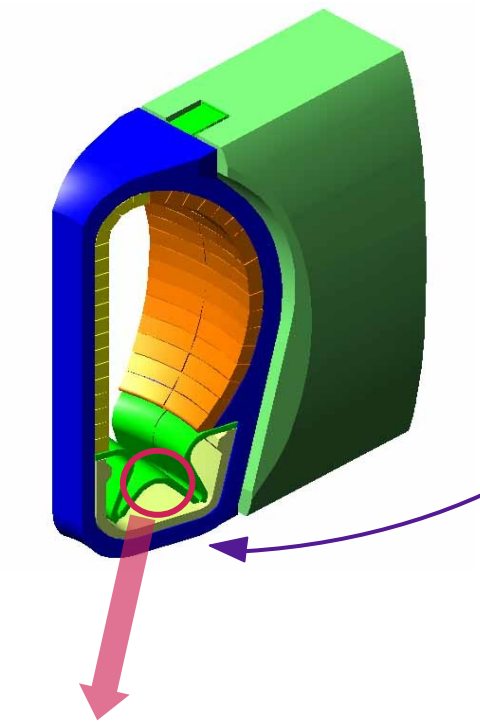
$$\text{Local TBR} = \frac{\text{Gross TBR } (=1.05)}{\text{Effective coverage } (=0.759)} \geq 1.38$$





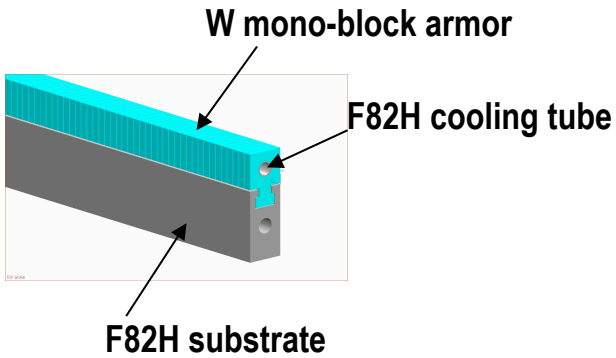


T-breeding behind divertor, not efficient



How is the impact of BLK behind divertor?

Divertor plate concept

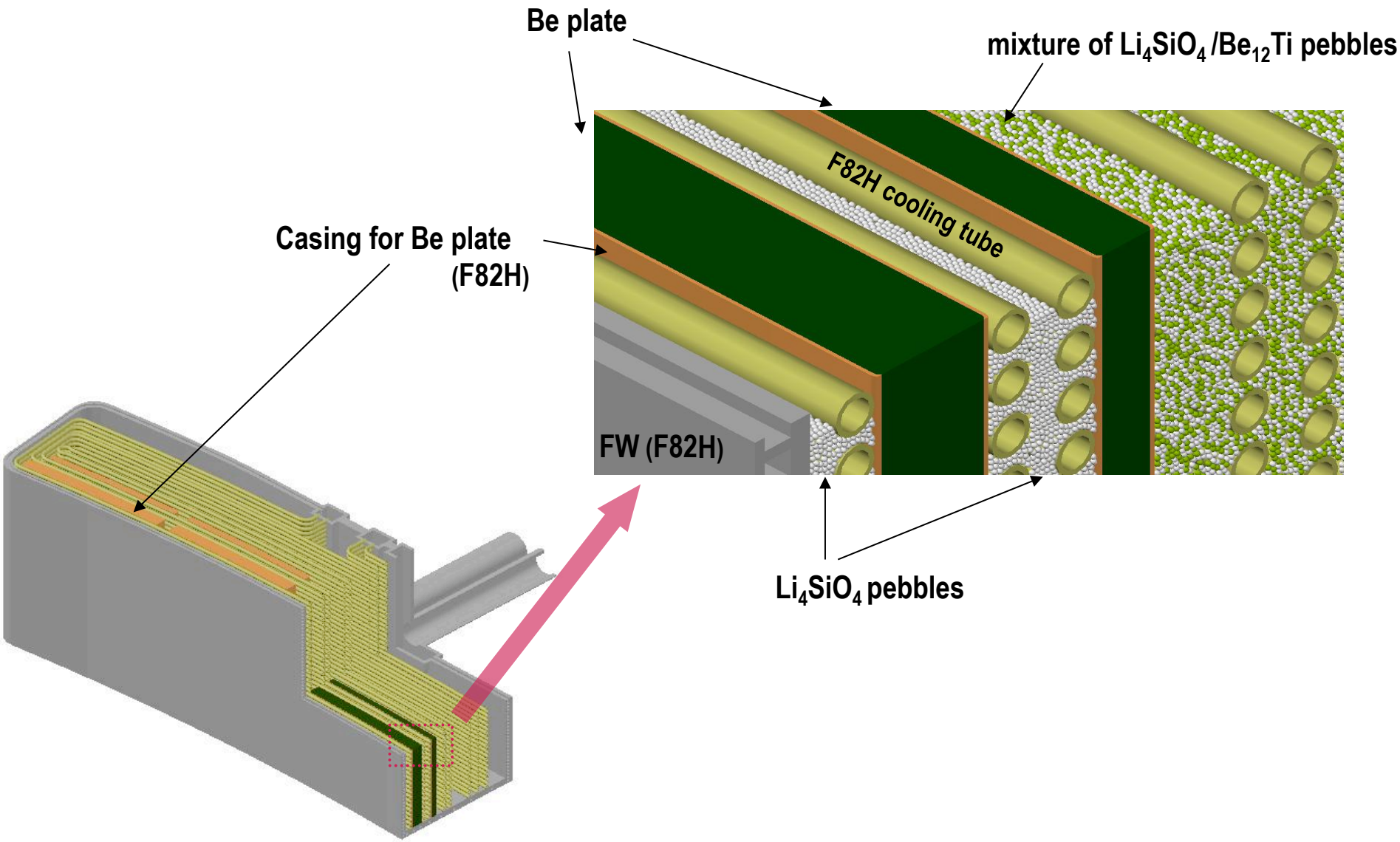


Calculation for a 30cm blanket

Breeder / multiplier	Local TBR	ΔGross TBR
Li_2TiO_3 / Be	0.47	0.018
Li_2TiO_3 / $Be_{12}Ti$	0.45	0.017

Increment, less than 0.02 !

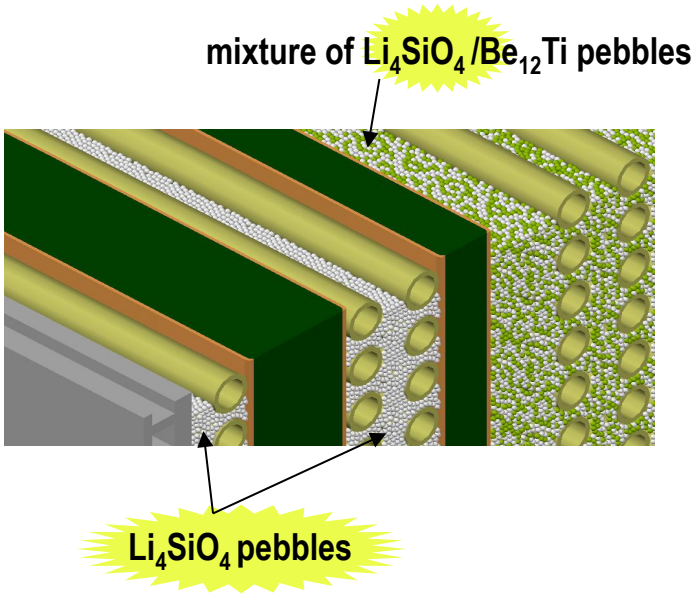
Candidate design of blanket



Why Li_4SiO_4 pebbles?

◆ Why "pebbles"?

- Gaps among pebbles, favorable for tritium recovery *(Packing ~60%)*

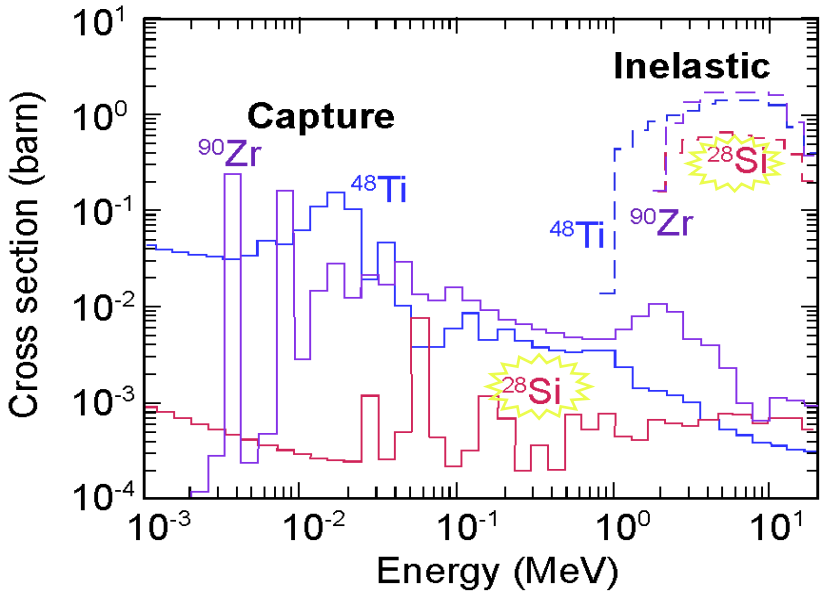


◆ Higher TBR than other Li ceramics (Li_2TiO_3, Li_2ZrO_3)

- Higher Li /cm^3

	Li_4SiO_4	Li_2TiO_3	Li_2ZrO_3
Li density(TD) [g/cm^3]	0.51	0.43	0.38
Thermal cond. [W/mK]	2.4	1.8	0.75
Swelling ΔV/V [%]	1.15	0.8	0.5
T retention time [s]	7.0	2.0	1.1
Reactivity with H_2O	small	none	none

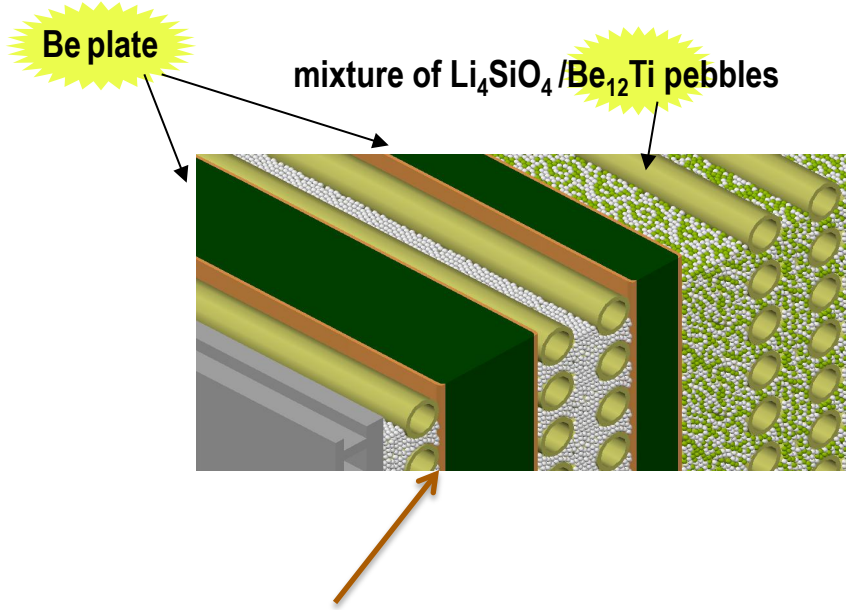
- Low neutron absorption by Si



Why Be plate forward?

- Why Be? (Why not $Be_{12}Ti$?)
  - Ti of $Be_{12}Ti$ absorbs neutrons in MeV range
- Why "plate"? (Why not "pebbles"?)
  - To thicken breeding layer for a given T_{upper}

    κ : plate >> pebbles



Be plate, canned by F82H to avoid reductive decomposition of Li_4SiO_4 by Be

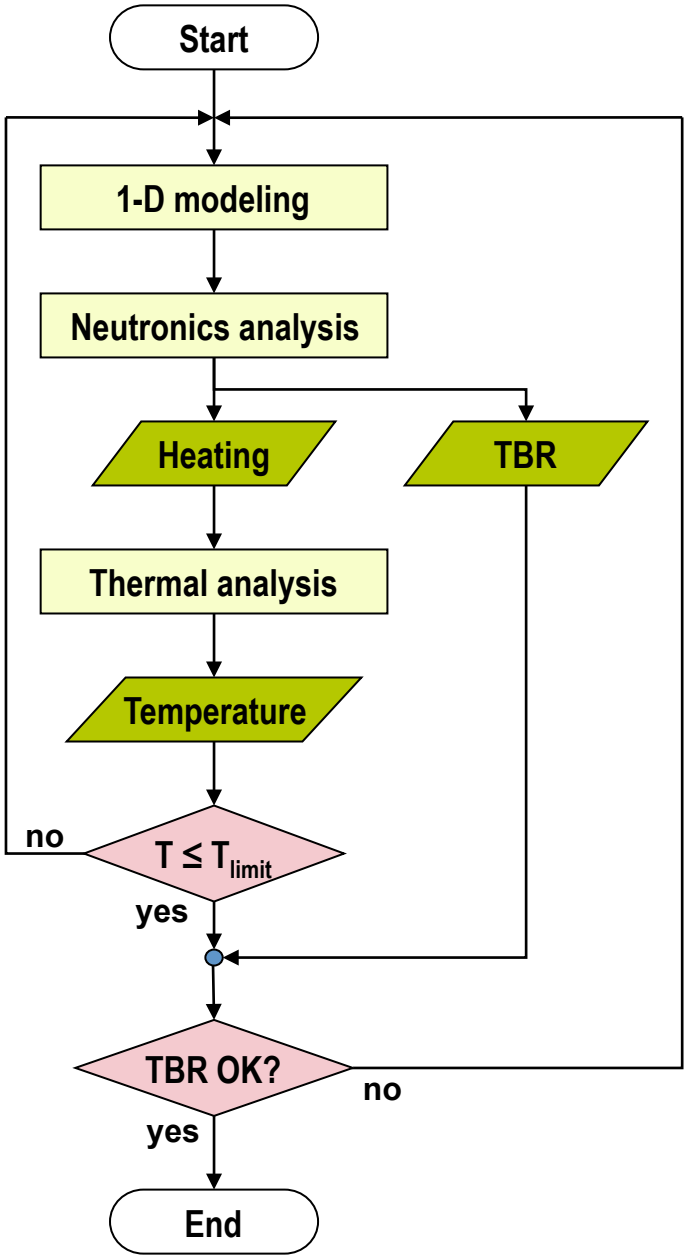
Why $Be_{12}Ti$ pebbles backward?

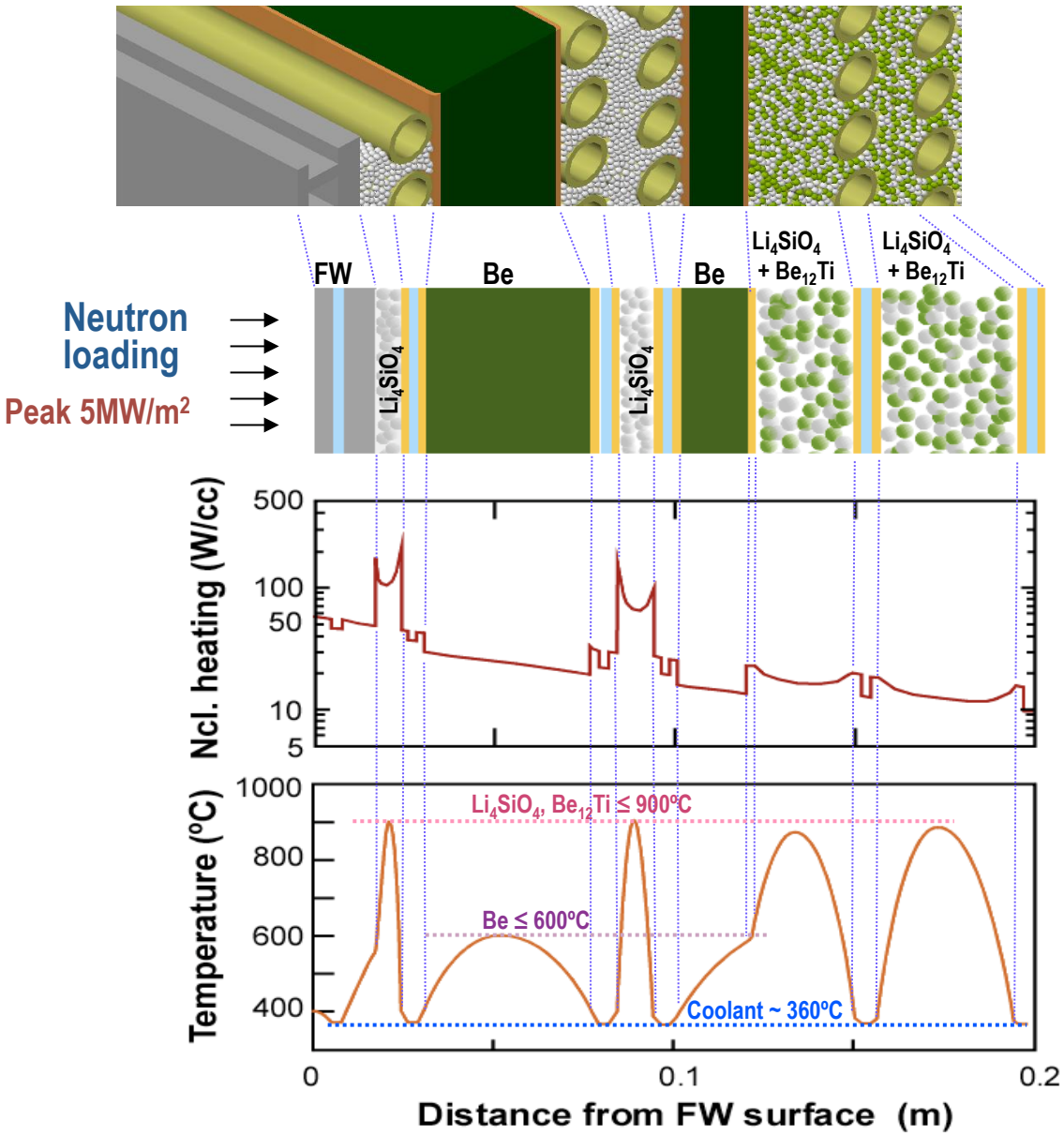
- Why $Be_{12}Ti$?
  - Allow to mix with Li_4SiO_4 ($Be_{12}Ti$ chemically stable)

    ➡ No F82H casing required ➡ reduced n-absorption
  - At E_n < 1 MeV, little n-absorption by $Be_{12}Ti$
- Why "pebbles"?
  - Favorable to mix with Li_4SiO_4

    and to avoid uncertainty in contact heat resistance

Procedure to determine blanket interior



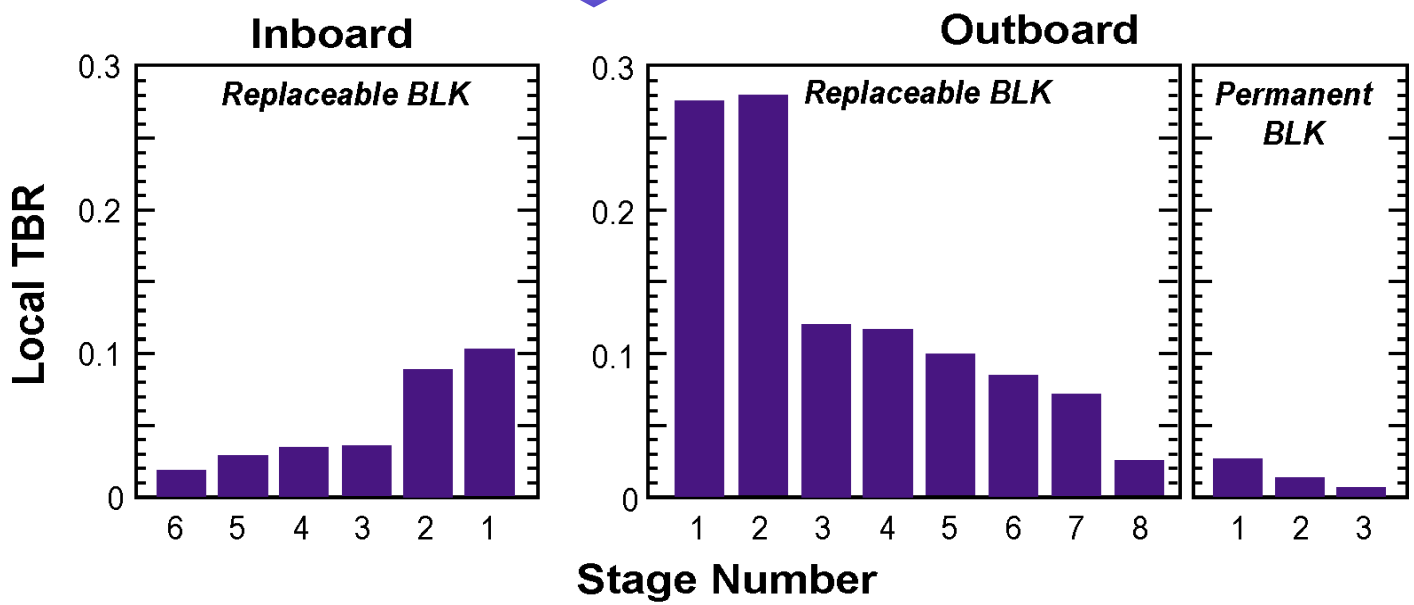


Local TBR (Calculated)

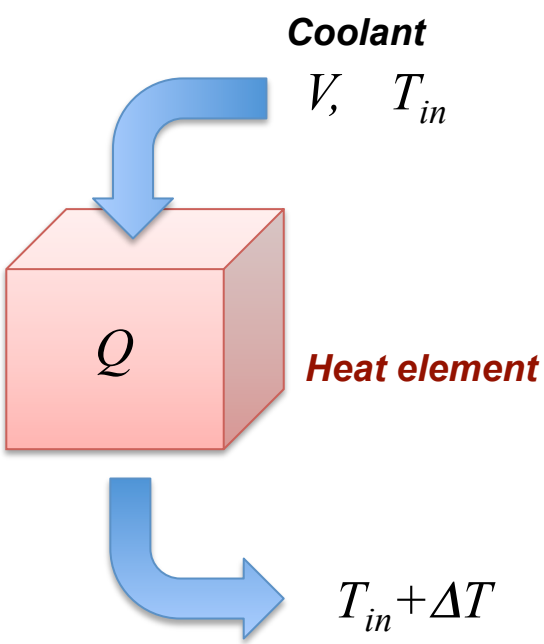
		Li_4SiO_4	Li_2TiO_3
Inboard	Replaceable BLK	0.31	0.30
Outboard	Replaceable BLK	1.07	1.04
	Permanent BLK	0.04	0.04
Total		1.42	1.38

Both satisfies the required local TBR (≥ 1.38)

Local TBR for each stage



Required coolant area for heat removal?



Heat balance: $$\underbrace{Q}_{\text{Generated}} = \int_{T_{in}}^{T_{in}+\Delta T} \underbrace{\rho c_p V dT}_{\text{Removed}}$$

Heat transfer: $$Q = \int_{T_{in}}^{T_{in}+\Delta T} h \underbrace{A}_{\text{Wall area of cooling pipe}} dT$$

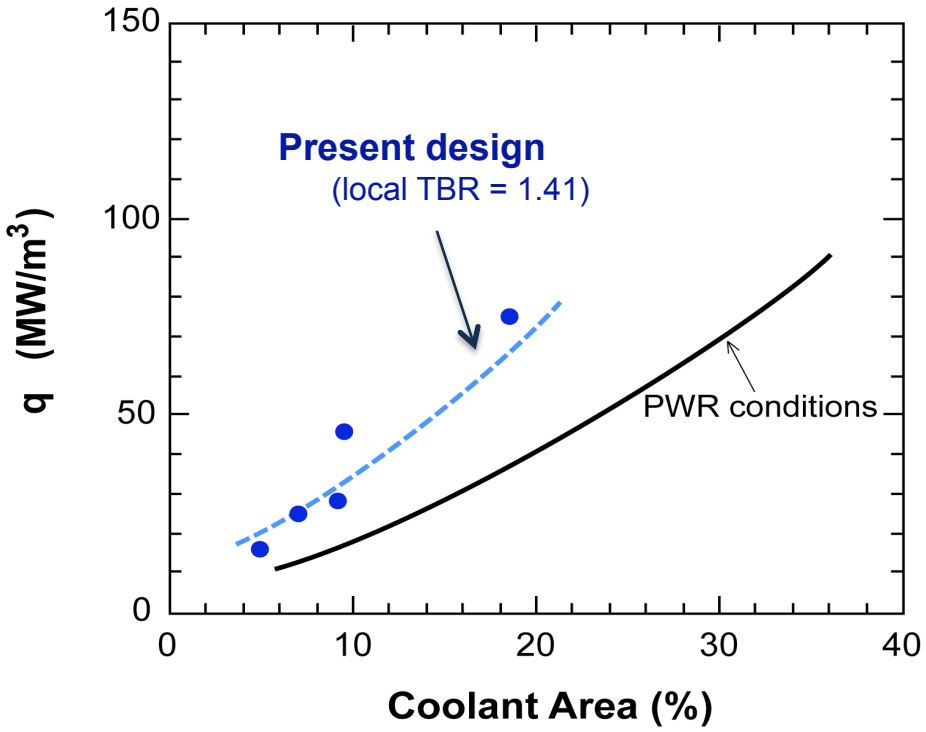
via cooling pipe wall

$$h = \frac{\kappa}{d_0} Nu$$

Assuming

- PWR conditions (15 MPa, 290-330ºC)
- flow speed = 4 m/s, d_0 = 6 mm

PWR water has limitted capacity for coping with P_n^{ave} ~ 3MW/m^2



- The present BLK design requires less coolant area than expected for PWR water to ensure T self-sufficiency.

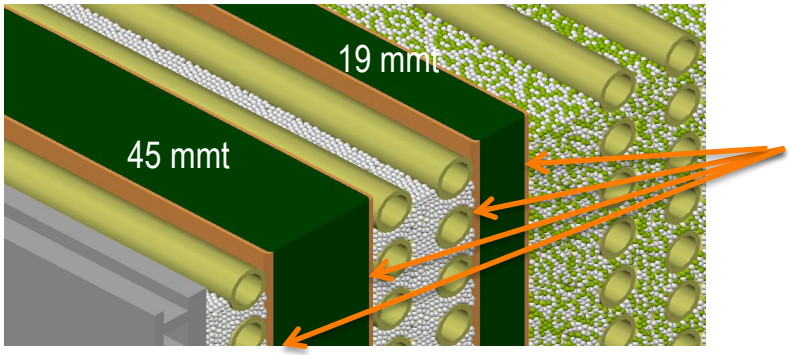
Solution

- PWR condition should be changed to

⇨ ***subcritical water conditions***

(~23 MPa, 290-360ºC)

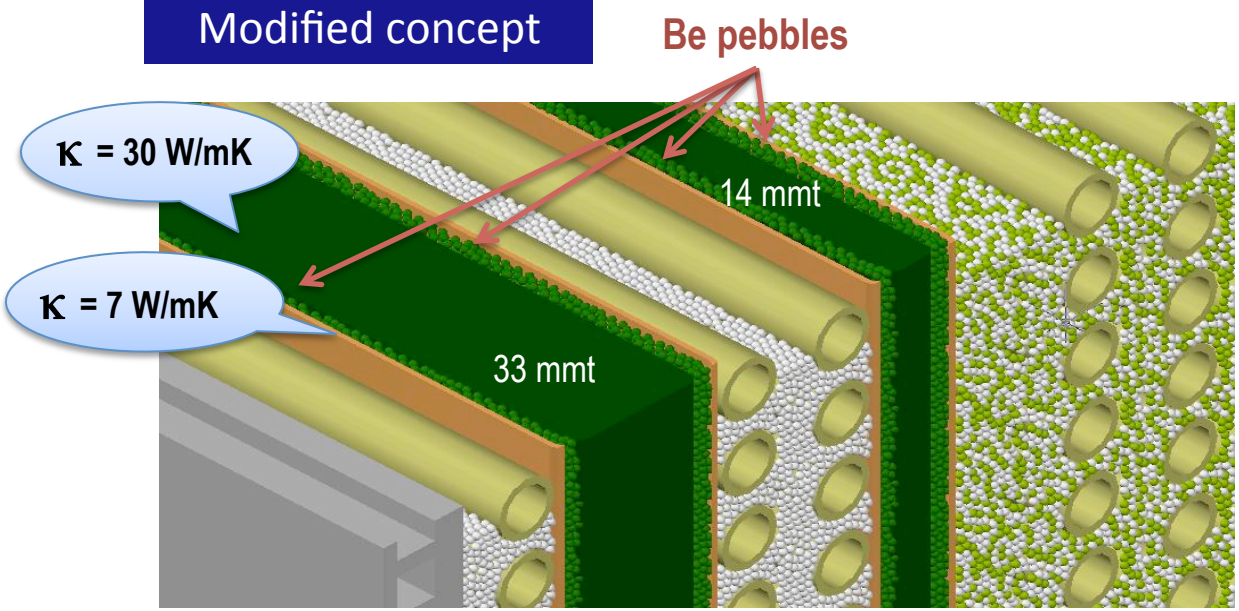
Problem on the concept



contact heat resistance, tricky!

Gross TBR = 1.08
(Local TBR = 1.41)

Modified concept



Has more reality, but
Gross TBR = 0.99
(Local TBR = 1.34)

For water-cooled solid breeding BLK, P_n^{ave} = 3 MW/m^2 seems over-ambitious.

Conclusion on Blanket Design

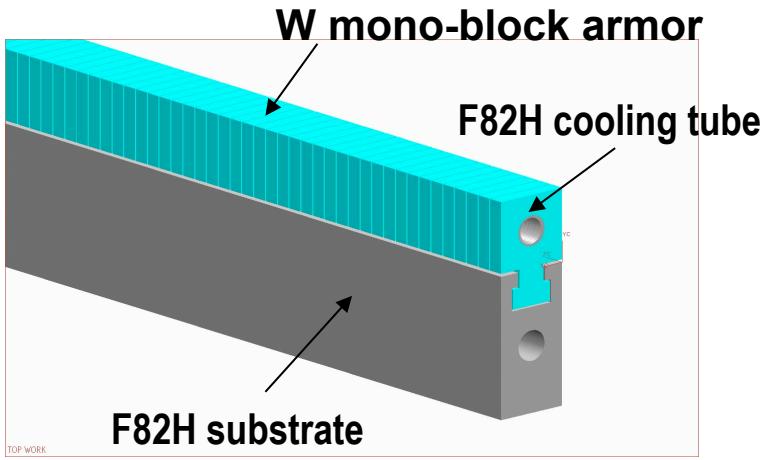
- **For handling P_n^{peak} = 5 MW/m^2 (3 MW/m^2 in average), water conditions with higher p and T than PWR are required.**

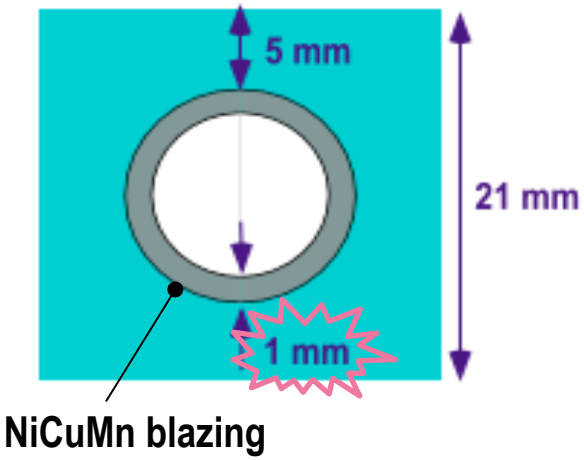
- **Solid breeding blanket has an intrinsic uncertainty in contact heat resistance.**

- **A safety side design seems difficult to meet T self-sufficiancy and heat removal of P_n^{peak} = 5 MW/m^2.**

2.2 Power handling on divertor

Original concept of divertor plate





	ITER	DEMO
Cooling tube	Cu-alloy	F82H
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

Original requirement from us

Possible modifications of the concept

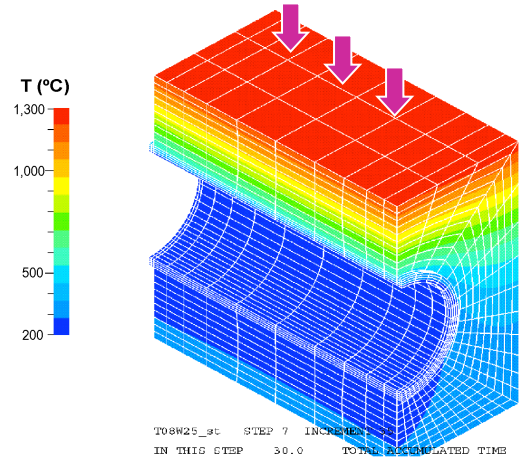
- **Coolant inlet temperature: 200ºC → 290ºC**
  - **Avoid corrosions of F82H by radiation-produced hydrogen peroxide (H_2O_2)**
    H_2O_2 starts to pyrolytically decompose at 240ºC
  - **Avoid irradiation brittleness of F82H (lack of data at < 300ºC)**

<u>Allowable heat flux:</u>

q_{max} ~ 7 MW/m²

($T_{coolant}$ = 290ºC)

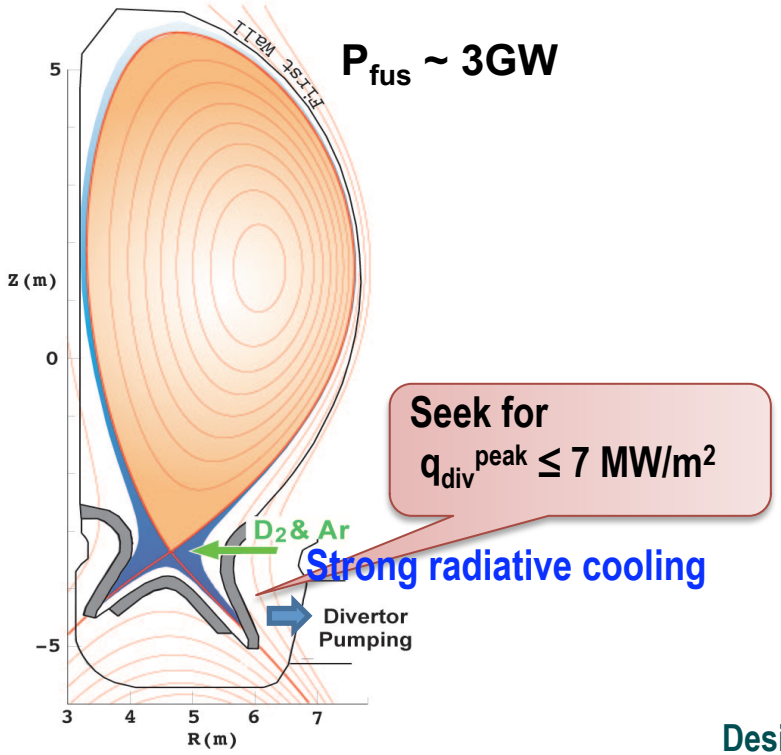
	Requirement
W	T ≤ 1,200ºC
RAFM	290ºC ≤ T ≤ 550ºC



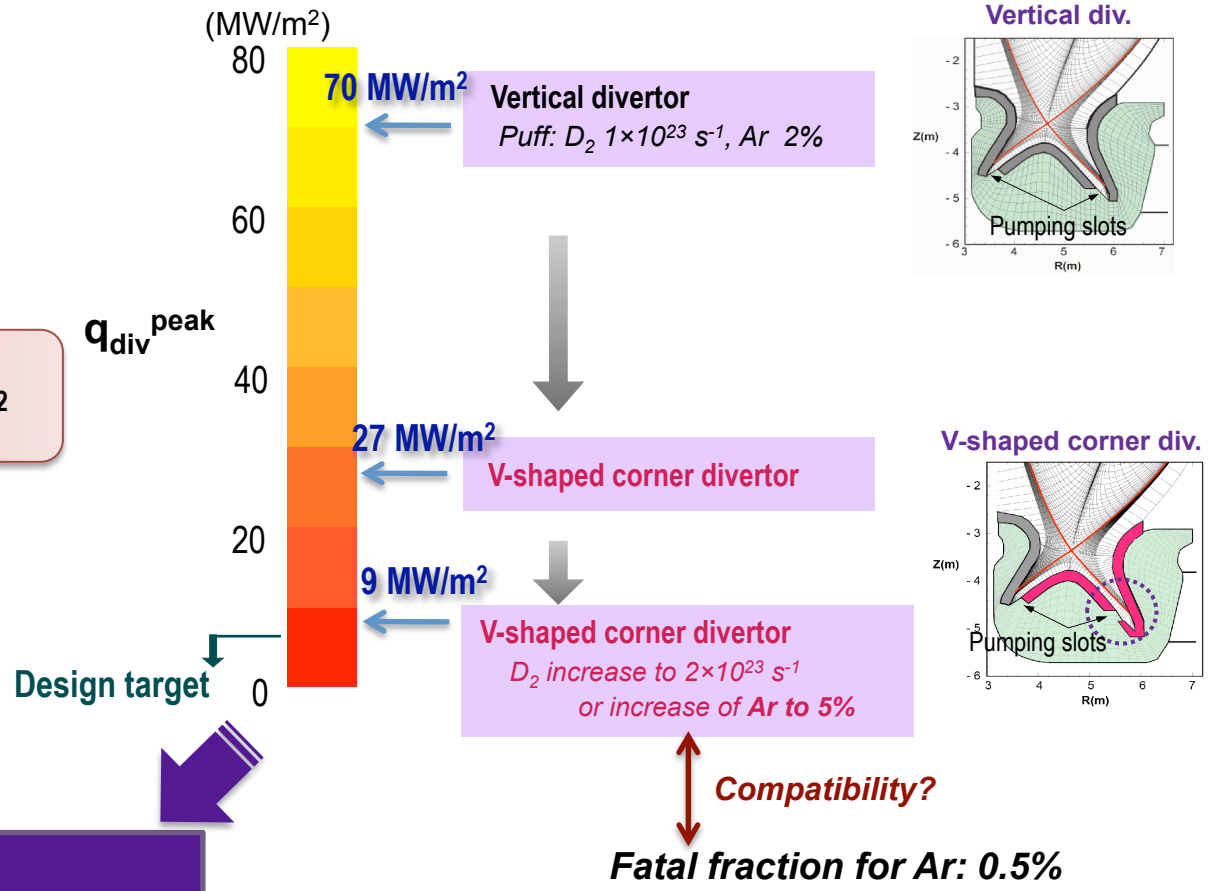
- **Coolant tube thickness: 1 mm → thicker as possible**

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

Approaching, but no solution so far



Summary of simulation result
(P_{fus} ~ 2.7 GW case)



Results:

- Efficiency of V-shaped divertor
- Huge D_2 and Ar puff, required

Considerations on Divertor

- **Numerically a solution satisfying q ~ 7 MW/m^2 seems to be obtained, albeit with an overextended condition (n_{Ar}/n_i ~5%).**

- **Concern: Contamination of main plasma by such a large amount of Ar**

  Confident evidence sweeping away the concern

  at the decision point of DEMO construction (in mid. 2020's)?

Summary

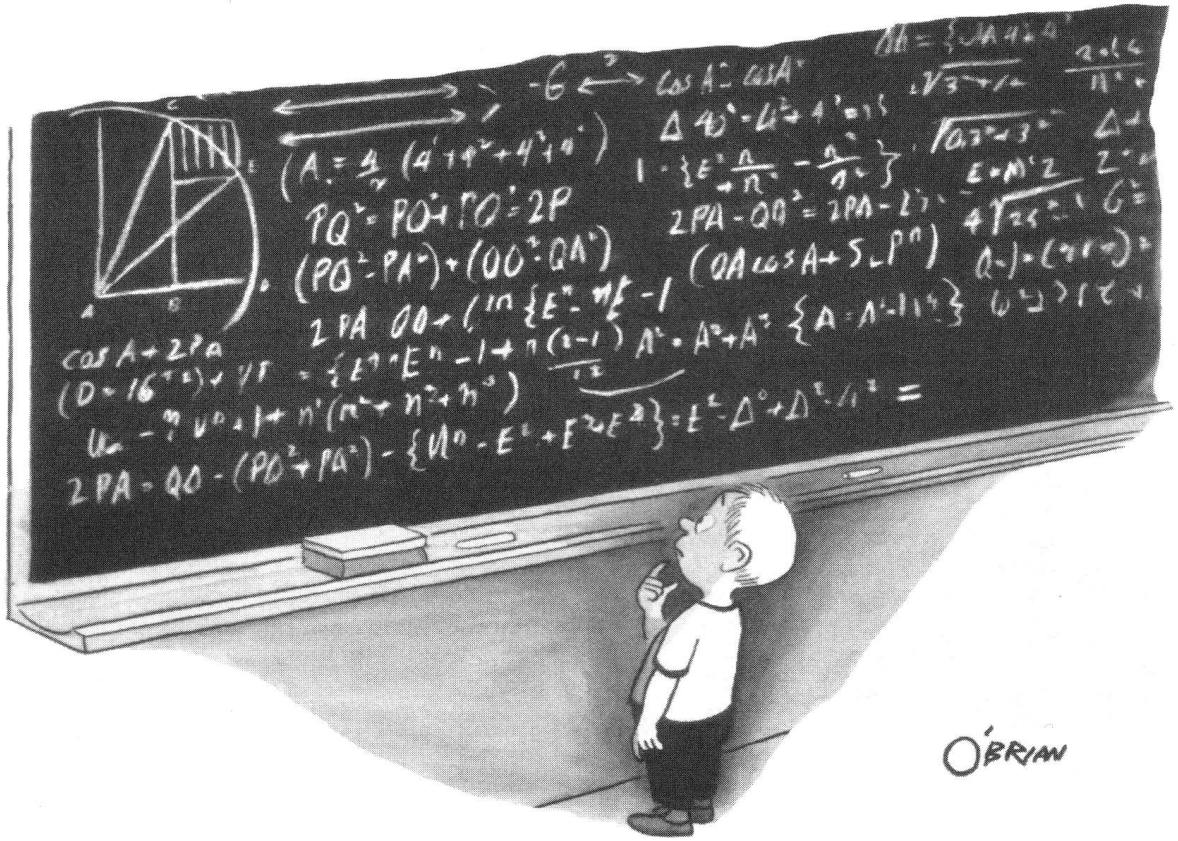
Blanket	• As to water-cooled solid breeder blanket, P_n^{ave} = 3 MW/m^2 seems difficult to be handled using foreseeable technology of DEMO.
Divertor	• Allowable heat flux to the divertor plate should be as low as ~7 MW/m^2 for JAEA's divertor concept. • For P_{fus} ~ 3 GW, a numerical solution meeting the heat flux can be obtained with strong feeding of fuel and impurity (Ar ~ 5%). —— Reality?

My internal arguments:

- *Do we really need P_{fus} = 3 GW (or P_n = 3 MW/m^2) in DEMO?*
- *Is it clever to go back to PWR water conditions?*

Is this challenge reasonable?

老师的提问恰当吗?



"Think!"

The New Yorkers Book of Teacher Cartoons, p.36

Our challenge:

$$P_{fus} = 3\ GW$$

Why not go for a 1-GW DEMO?