Divertor study for the DEMO reactor, and overview of JAEA Aomori Fusion R&D center

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Requirements for Tokamak DEMO Reactor in Japan (1/2)

Recent DEMO reactor concepts in Japan are based on
Atomic Energy Commission: Check and Review of 3rd Phase Basic Program 2005
Report on National Policy of Future Nuclear Fusion Research and Development

http://www.aec.go.jp/jicst/NC/senmon/kakuyugo2/siryo/kettei/houkoku051026_e/index.htm

1. Basic guideline

Core dimension similar to that of ITER
Power generation capability of GW level
DEMO needs to **operate continuously** for about one year
Overall tritium breeding ratio **(TBR) exceeding unity**

2. Plasma performance

High plasma pressure operation is required to increase fusion power density to realize 3-4 GW level
Non-inductive steady-state operation

3. Structure material

Blanket structural material with strong candidates such as **reduced activation ferritic steel: withstand about 3-6 years neutrons** (as the **neutron fluence of about 10-20 MW year/m^2**) and **heat flux (about 1MW/m^2)**

Requirements for Tokamak DEMO Reactor (2/2)

4. Blanket

Breeding and power generation blanket **must realize the breeding** and **recovery of the tritium** with high reliability securing tolerance against the disruption**.**

5. Divertor

Divertor components have **tolerance to neutron irradiation and high particle flux for several years level.**

6. Maintenance

Maintenance period of the first wall and divertor is scheduled in several years:
it should be sufficiently short not to affect plant availability.
Reliability of **continuous operation of the heating and current drive system up to one year** should be established as well.

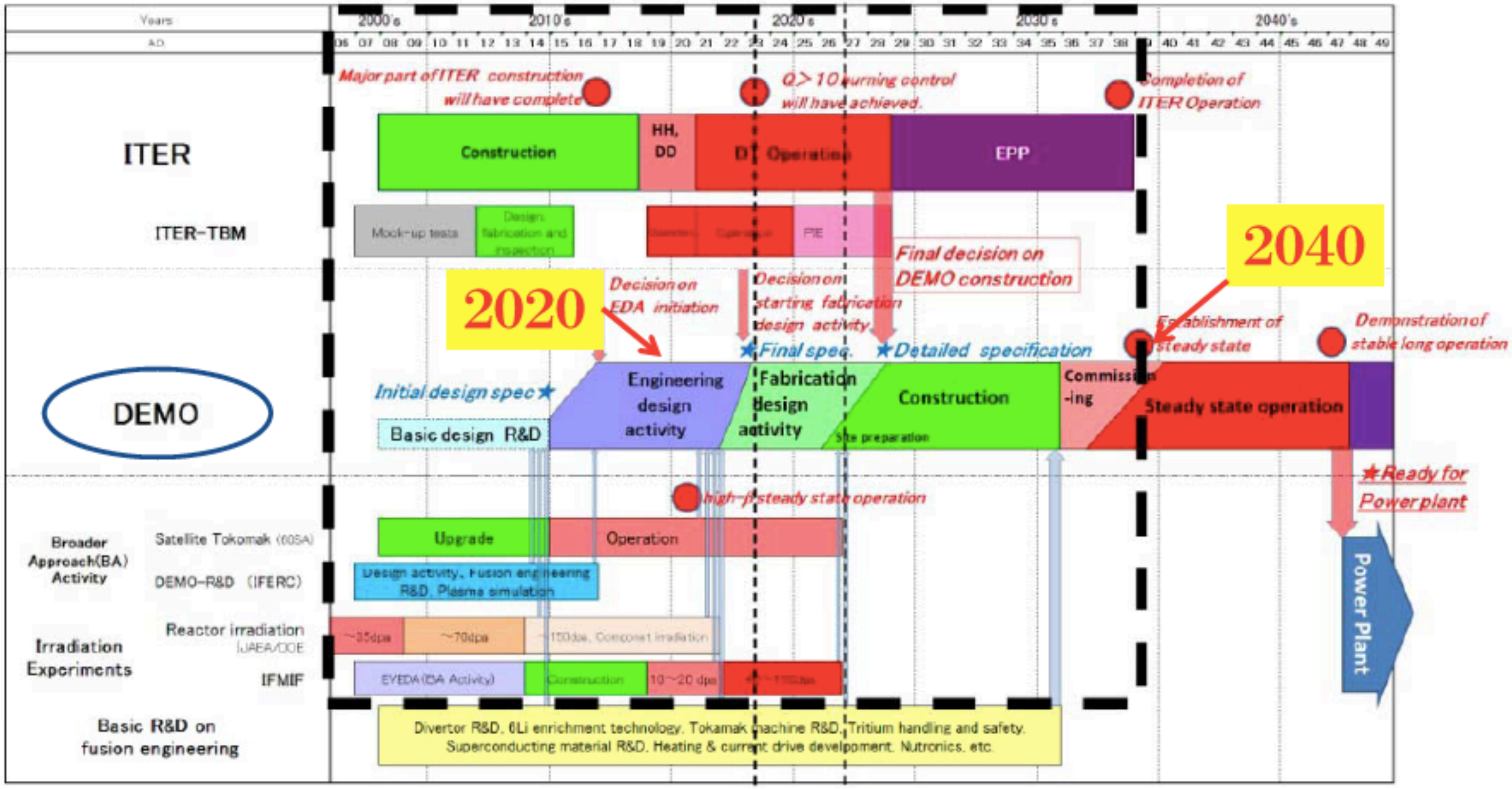
7. Cost

Construction cost of DEMO should be controlled to an acceptable level taking the future commercialization into account.

Fusion DEMO reactor roadmap plan in Japan (2008)

(proposed by Fusion Energy Forum of Japan)

Roadmap is based on sophisticated analysis with Work Breakdown Structure

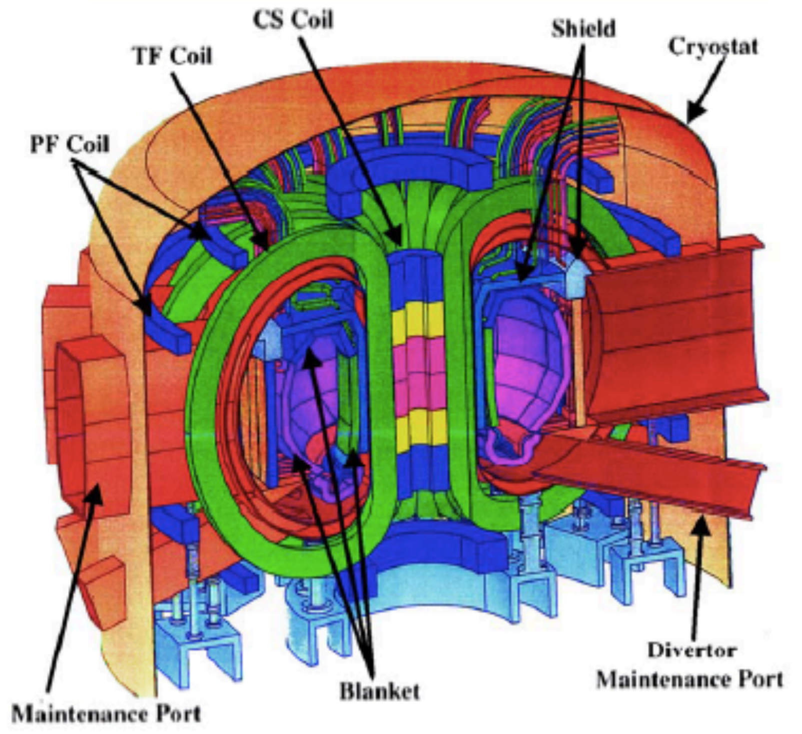


This roadmap has been studied and presented by Working Group on the Roadmap Development and Related Issues organized under ITER・BA Technical Promotion Committee of Fusion energy forum of Japan, as a case study of Tokamak type DEMO reactor R&D. It have to be noted that this roadmap is A case study and NOT the roadmap which is authorized by Japanese government.

Note) In 2010,
Schedule of the ITER DT Experiment will change along the ITER Scenario-1.
The modification of roadmap is under examination.

Tokamak DEMO concept designs in Japan

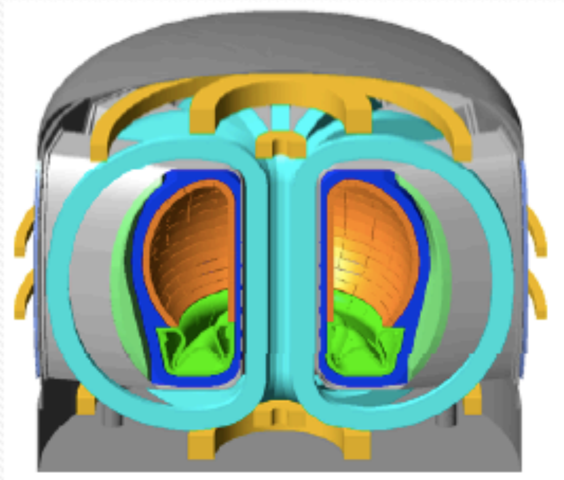
Demo-CREST(CRIEPI)#



R_p = 7.3 m, A = 3.4, a=2.1m
V*S~0.7 LpIp
B_{max} = 16 T, β_N = 1.9 min, 4.0 max w/ rs

Based on ITER physics.
Similar plasma configuration to ITER
Moderate size (but larger than SSTR)

SlimCS(JAEA)



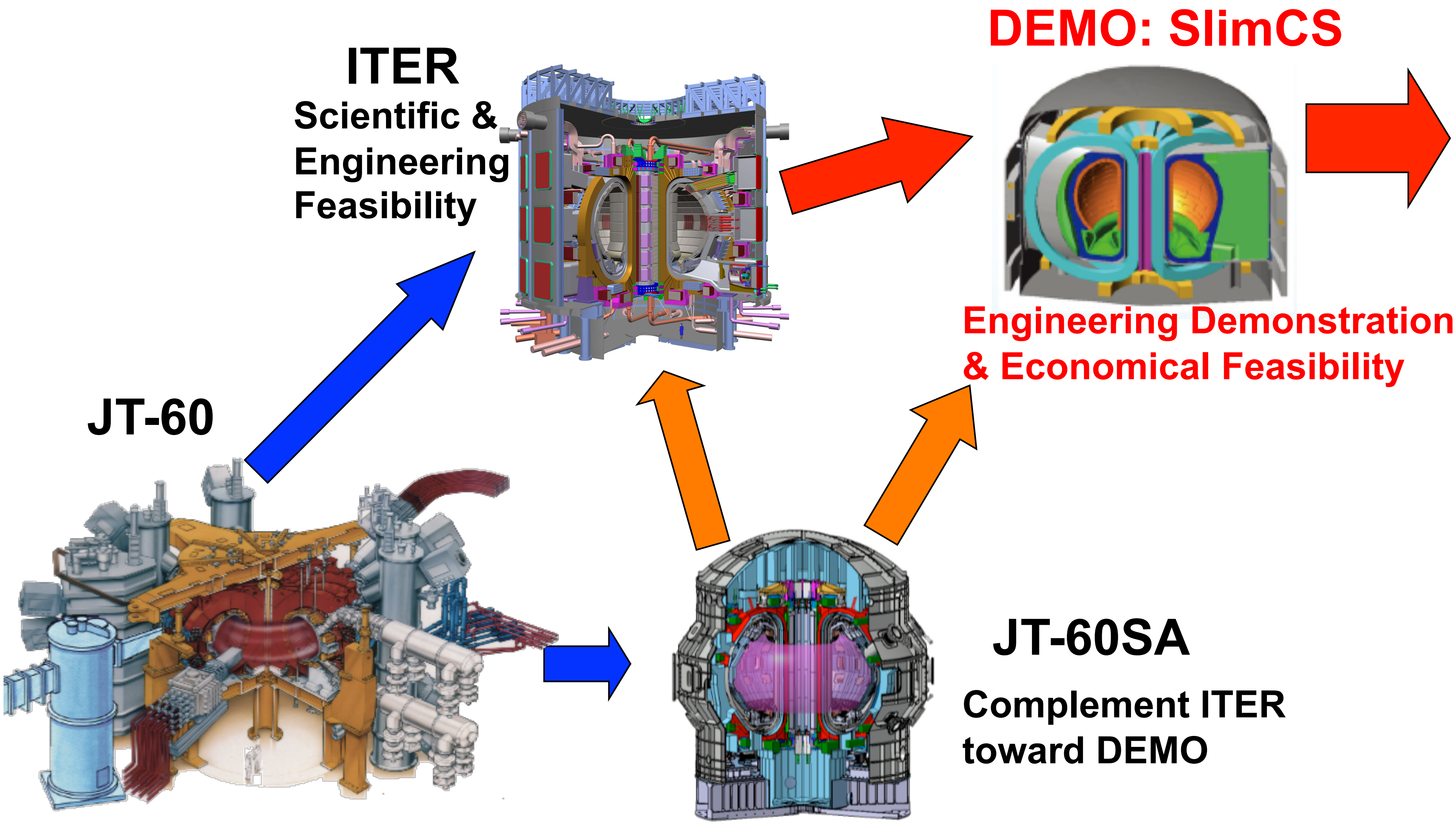
R_p = 5.5 m, A = 2.6, a=2.1m
V*S~0.3LpIp
B_{max} = 16.4 T, β_N = ~4.3

Naturally high beta by low A design
Similar to JT-60SA plasma
Compact, Flexibility in blanket design

#) Central Research Institute of Electric Power Industry

Scenario for Fusion Reactor Development in JAEA

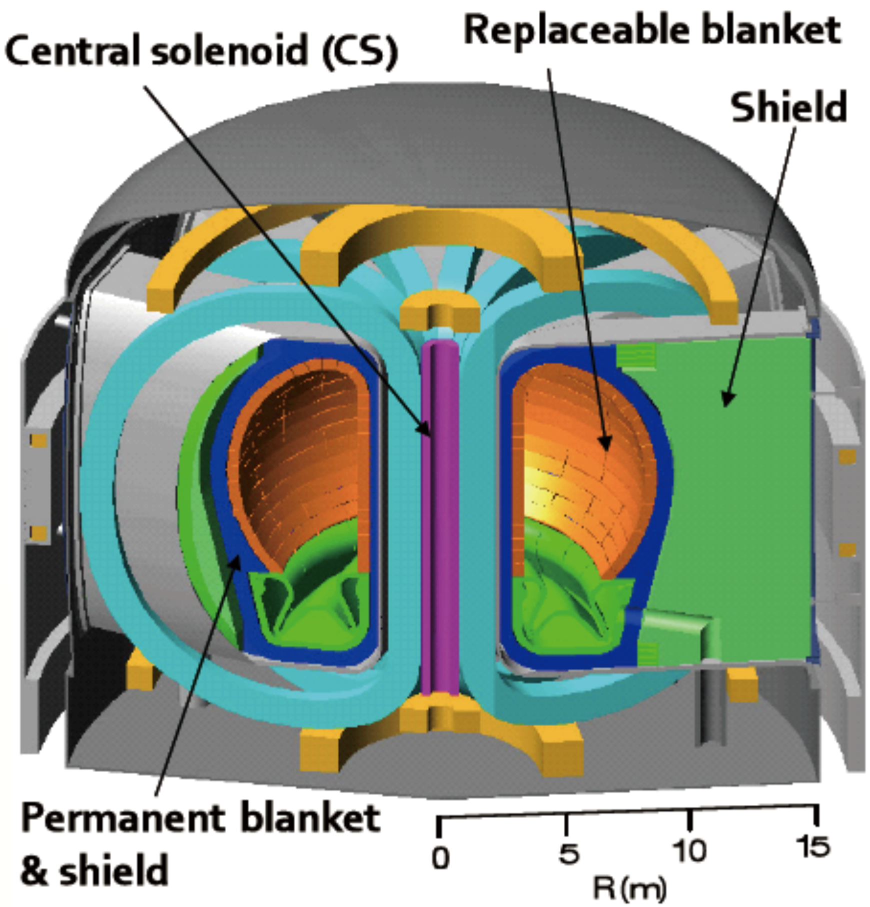
DEMO would be the last integrated R&D device just after experimental reactor ITER, and before the 1st generation commercial reactors.



Compact DEMO reactor concept: SlimCS (JAEA)

- As compact as ITER's
- Electric output of 1 GWe level
- Economic viability of fusion power
- Steady-state operation
- High beta plasma

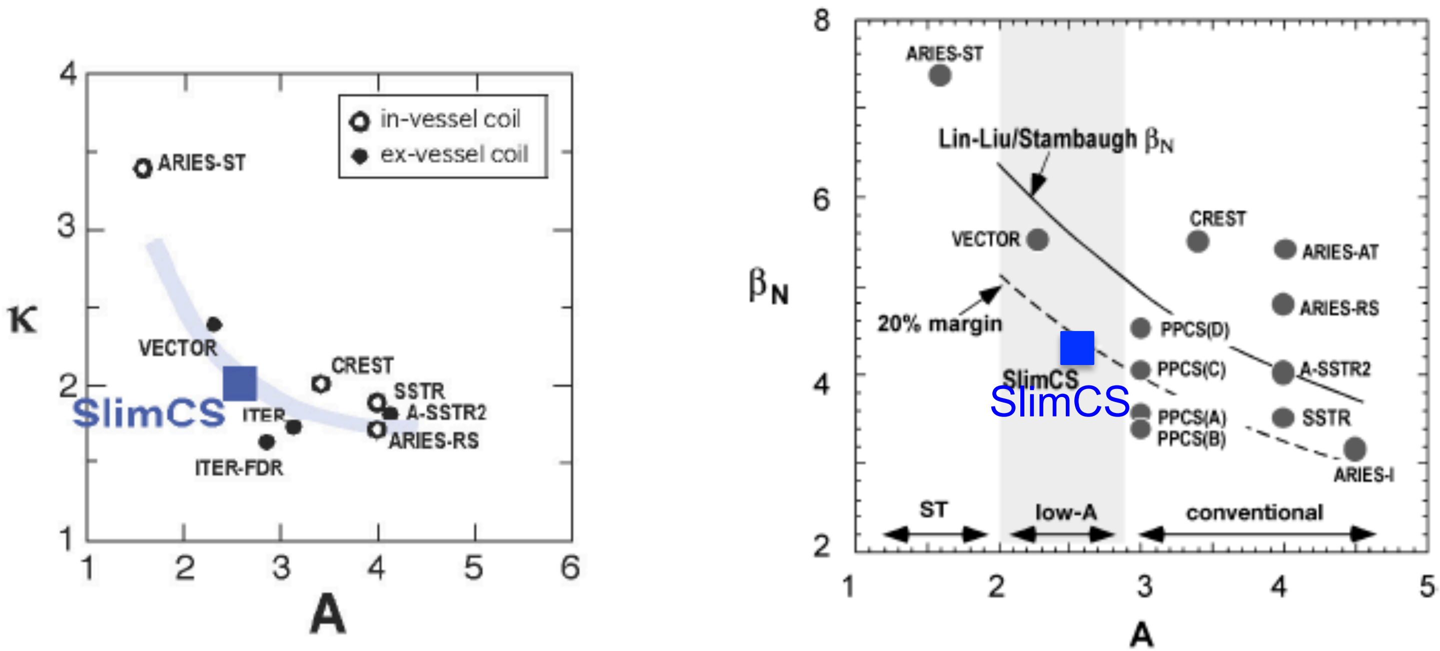
Major radius, R_p	5.5 m
Minor radius, a	2.1 m
Aspect ratio, A	2.6
Plasma current, I_p	16.7 MA
Toroidal field, B_T	6.0 T
Maximum field, B_{max}	16.4 T
Elongation, κ_{95}	2.0
Safety factor, q_{95}	5.4
Normalized beta, β_N	4.3
Density, $\langle n_e \rangle$	1.15×10^{20} m^{-3}
Normalized density, n_e/n_{GW}	1.0
Confinement enhancement, HHy2	1.3
Bootstrap current fraction, f_{BS}	~0.75
Current drive power, P_{CD}	60-100 MW
Fusion output, P_{fus}	2,950 MW
Neutron wall load, P_n	~3 MW/m^2



K. Tobita, et al. Nucl. Fusion 49 (2009) 075029

Compact DEMO reactor concept: SlimCS (JAEA)

Low-A facilitates high κ and high β_N access with reasonable design margins



High κ increases n_{GW} (because of an increase in I_p), which allows efficient use of the capacity of high β_N

Neutron flux ($3MWm^{-2}$), Bmax(16.5T), Blanket design are based on VECTOR design (A=2.3)
Engineering components such as RAFM, Water cooling are applied from SSTR design (A=4.1)

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2.1 Power handling in DEMO divertor

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

SlimCS

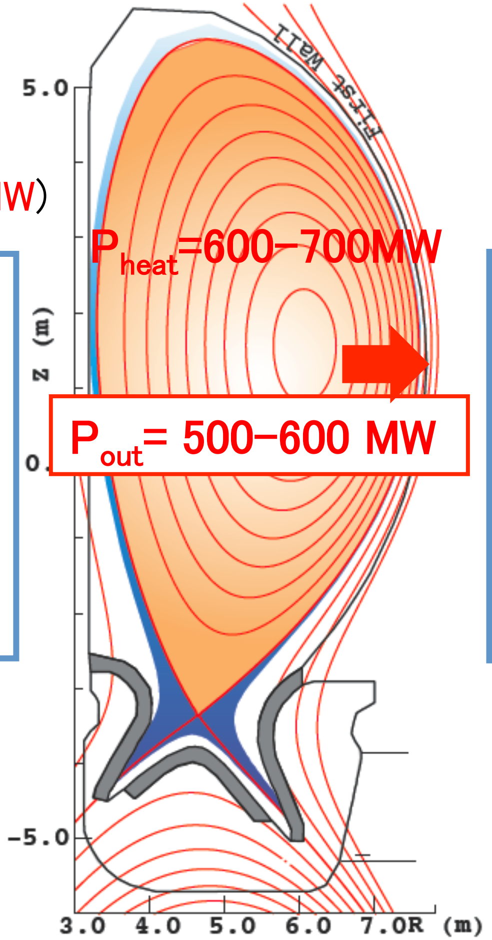
P_{fusion} = 2.95 GW
($P_{heat}=P_{\alpha}+P_{ax}$=600~700 MW)

P_{out} = 500–600 MW

P_{rad}= 450 ~ 550 MW

P_{target} < 50 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



ITER

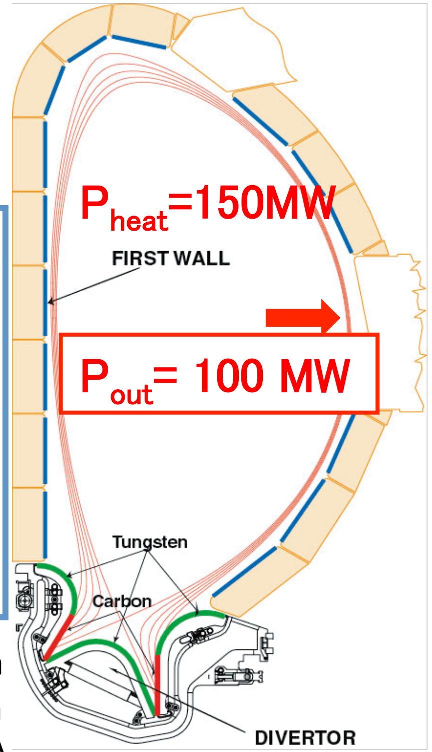
P_{fusion} = 0.5 GW
($P_{heat}=P_{\alpha}+P_{ax}$=150 MW)

P_{out} = 100 MW

P_{rad} ~50 MW

P_{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
Plasma volume : V_p=830 m^3



Extension of ITER divertor concept to DEMO divertor ?

Design concept for ITER divertor is applied/extended to the DEMO (SlimCS) divertor:

"divertor detachment" (T_e~ a few eV) is a key for the power handling

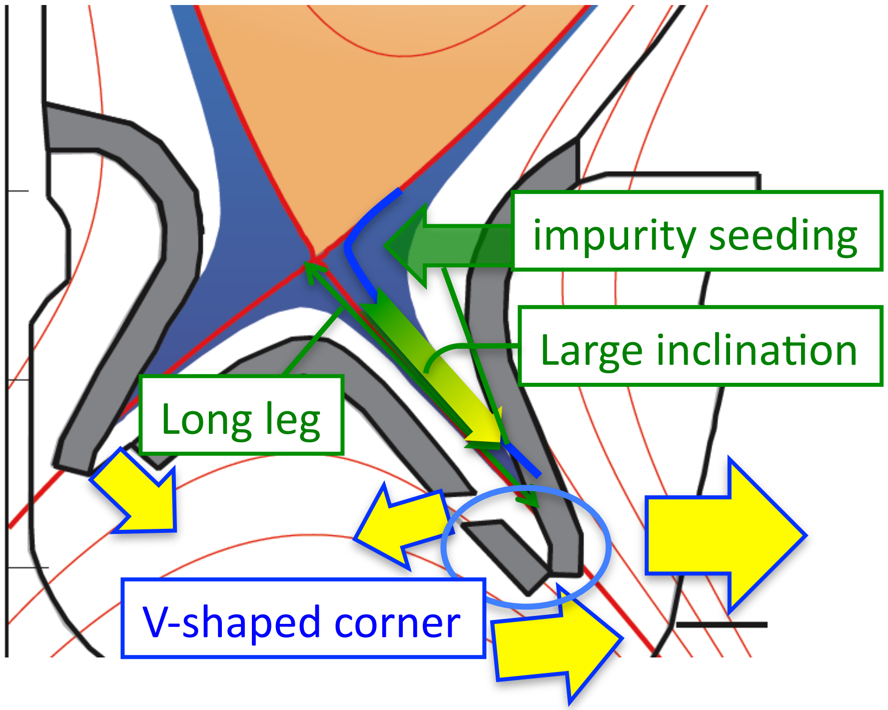
(1) Divertor leg and **inclination of the target** are larger than ITER

⇒ increase radiation, CX & volume recombination at the upstream, 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.

⇔ **Flux expansion** may be smaller than ITER due to D-coil distance from the plasma



geometry factors	SlimCS (2008)	ITER (2009)
leg length, L_{sp} (in/out)	1.37m/1.83m	0.97m/1.14m
incl. angle, θ_{sp} (in/out)	21°/18°	38°/25°
Dome top below Xp	~0.5m	~0.55m*
V-shaped corner	out **	in & out
Flux expansion (in)/(out)	7/3	7/6
Wet area for λ_q^{mid} = 5mm (in/out)	2.2/1.9m^2	1.4/1.9m^2

* Lower dome design (2009)

** Inner divertor is detached without V-corner

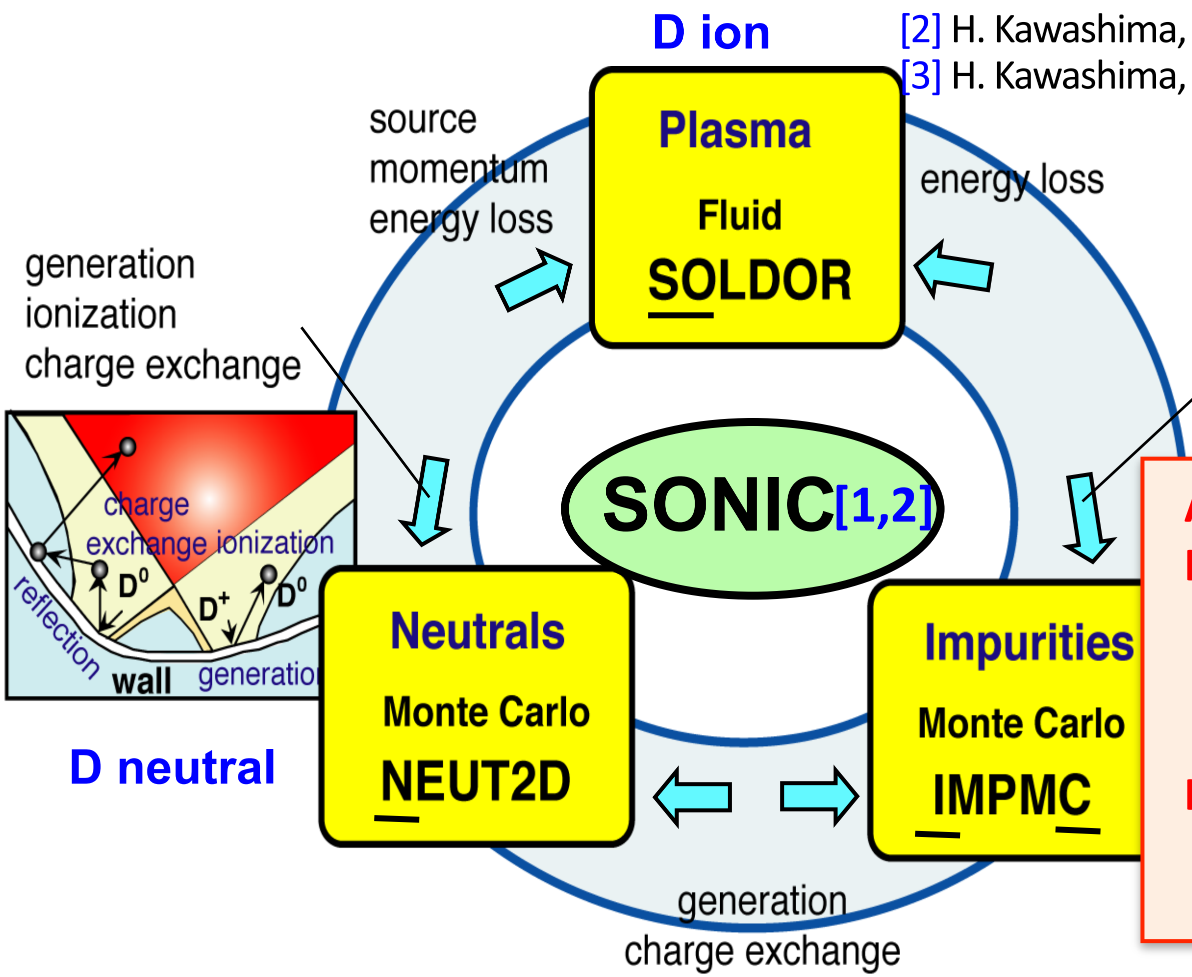
SONIC: self-consistent coupling with Ar impurity Monte Carlo has been developed for Ar seeding and transport

- **SOLDOR/NEUT2D** were used for DEMO divertor design, where Ar impurity radiation with *non-coronal model*: $P_{rad} = L(T_e, \tau_r)\, n_z n_e$, and *constant* n_{Ar}/n_i was applied. **[3]**

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

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

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



Ar impurity transport

Ionization processes:

ionization

charge exchange

surface/volume recombination

Ion transport processes:

thermal force – friction force

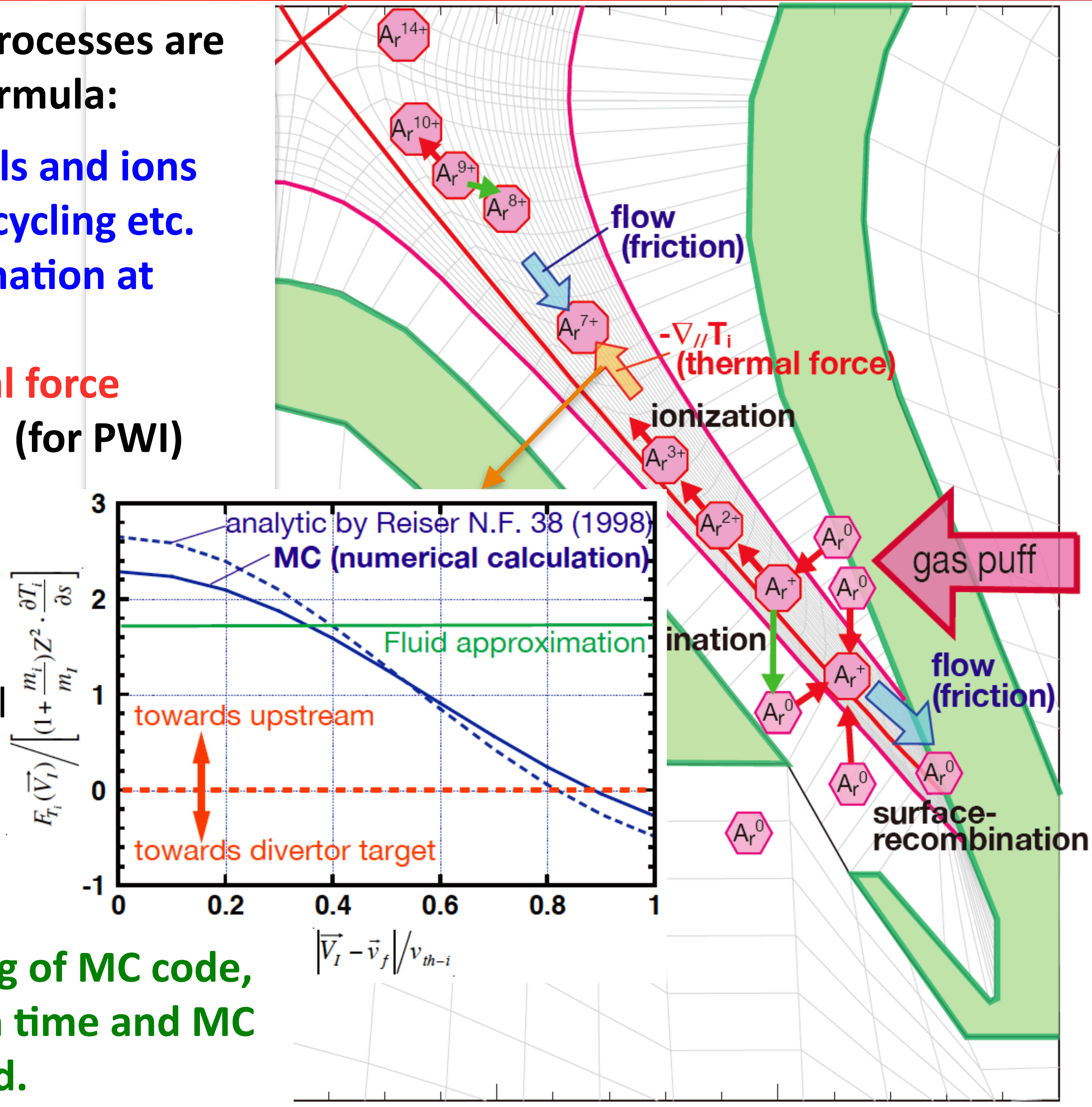
Ar^0 atom dynamics

MC approach has advantages to impurity modelling

Most impurity transport processes are incorporated in original formula:

- **Tracking impurity neutrals and ions**
  ⇒ CX-loss, n-collision, recycling etc.
  Radiation & Recombination at multi-charge states
- **Kinetic effect ⇒ Thermal force**
- **Gyro-motion ⇒ Erosion (for PWI)**

Kinetic thermal force (F_{Ti}) decreases with impurity ion speed (V_I) approaching to ion thermal velocity ($v_{th\text{-}i}$).



For self-consistent coupling of MC code, problems (long calculation time and MC noise) have been improved.

2.2 Simulation of power handling in the SlimCS divertor

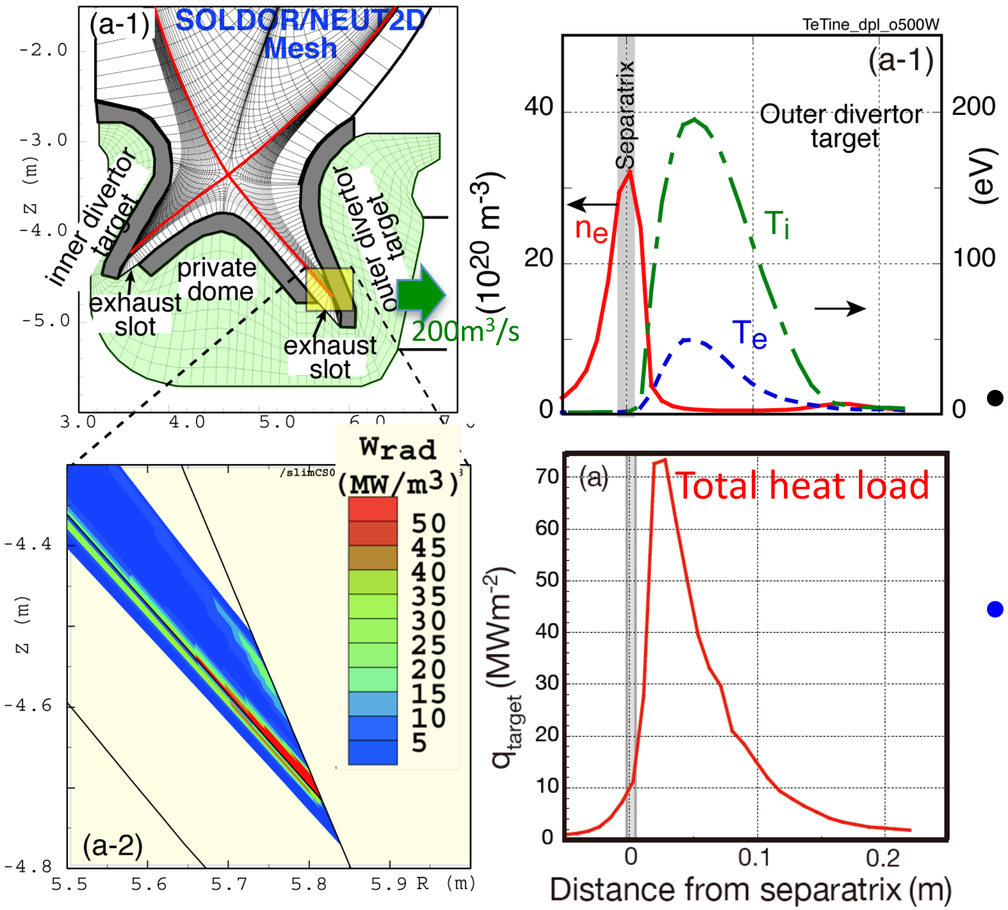
SOLDOR/NEUT2D was used for the DEMO divertor design with Ar impurity radiation.

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

Input parameters at edge-SOL

***P*$_{out}$= 500 MW, Γ_{out}=0.5x10^{23} s^{-1} (r/a=0.95)**

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



Case-1:pumping from bottom corner with gas puff and impurity seeding:

D_2/T_2 gas 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:

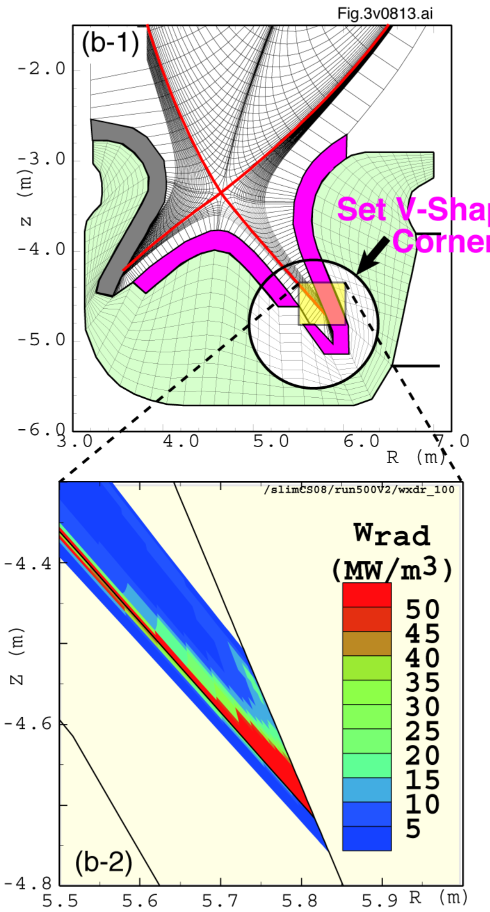
S_{pump}= 200 m^{3}s^{-1} is given.

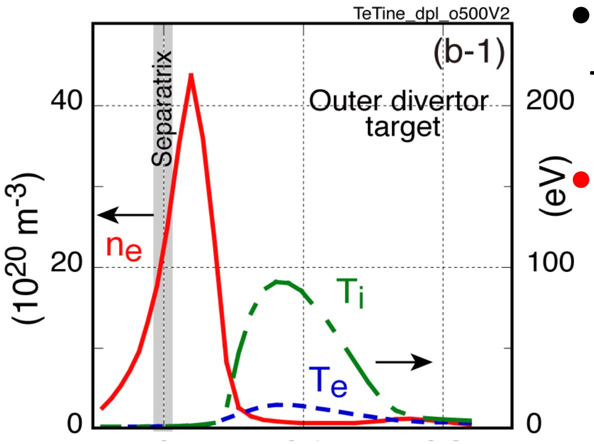
- At the inner target, divertor is detached and q^{target} < 5 MW/m^2.
- At the outer target, high temperature at the strike-point:

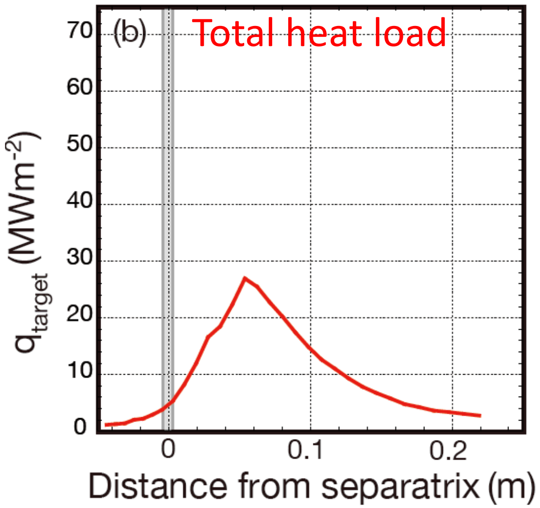
peak T_e~50 eV and *T_i~200 eV*, giving **severe peak heat load ~70 MW/m^2 !**

Power handling in divertor: divertor geometry

Case-2: Concept for the ITER divertor, *V-shaped corner*, was investigated







- *Divertor recycling* is increased from 3.7×10^{24} to 4.2×10^{24} s^{-1}.
- *Radiation loss at the outer divertor* is increased at upstream of the strike point **from 85 to 142 MW**:
  Total P_{rad}*(edge+div.)* = 390 MW
  (P_{rad}^{edge}~130MW, P_{rad}^{div}~260MW)
- *Peak* T_e~20 eV , T_i~90 eV are smaller by the factor of 1/2~1/2.5.

Peak heat load is reduced from 70 MW/m^2 to 27 MW/m^2.

"Full detachment" is necessary to decrease power loading

Radiation power load becomes large ⇒ Impurity transport is important

Ar impurity radiation loss is calculated with *non-coronal model*: $P_{rad} = L(T_e, \tau_r)\, n_z n_e$, and *constant* n_{Ar}/n_i at the outer divertor was increased from 2% to 5%.

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

$$q_{\mathrm{target}} = \underbrace{\gamma \cdot n_{\mathrm{d}} \cdot C_{\mathrm{sd}} \cdot T_{\mathrm{d}}}_{\text{Transport component}} + \underbrace{n_{\mathrm{d}} \cdot C_{\mathrm{sd}} \cdot E_{\mathrm{ion}}}_{\text{Surface-recombination loss}} + \underbrace{\mathrm{f}_1(P_{\mathrm{rad}})}_{\text{radiation power load}} + \underbrace{\mathrm{f}_2\,(1/2\mathrm{m}\mathrm{v}_0^{\,2}\mathrm{n}_0\mathrm{v}_0)}_{\text{neutral power load}}$$

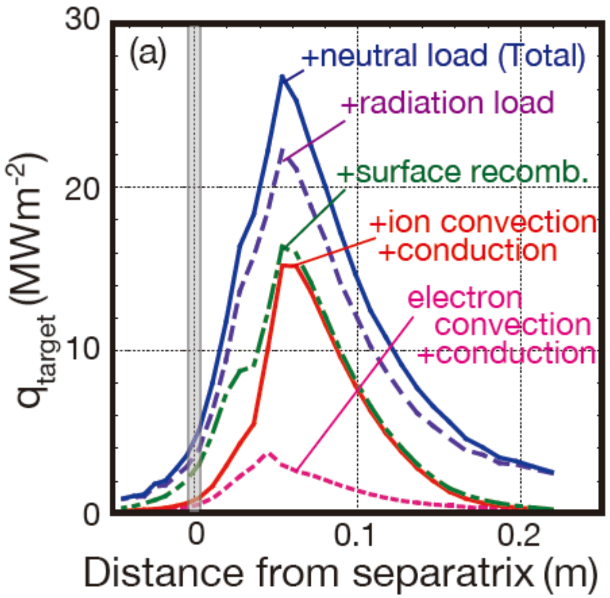
Transport component (incl. electron&ion-conduction/convection)

Surface-recombination loss

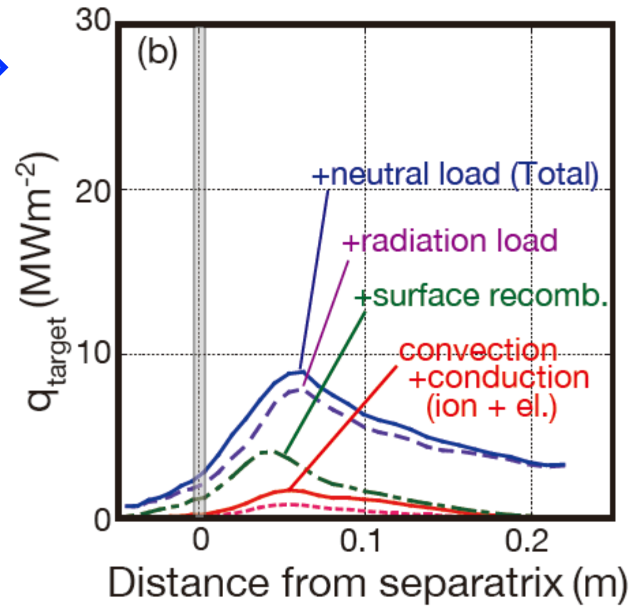
radiation power load

neutral power load

Case-2: "*V-shaped divertor*" (n_{Ai}/n_i=2%)



Case-3: "*V-shaped divertor*" (n_{Ai}/n_i=5%)

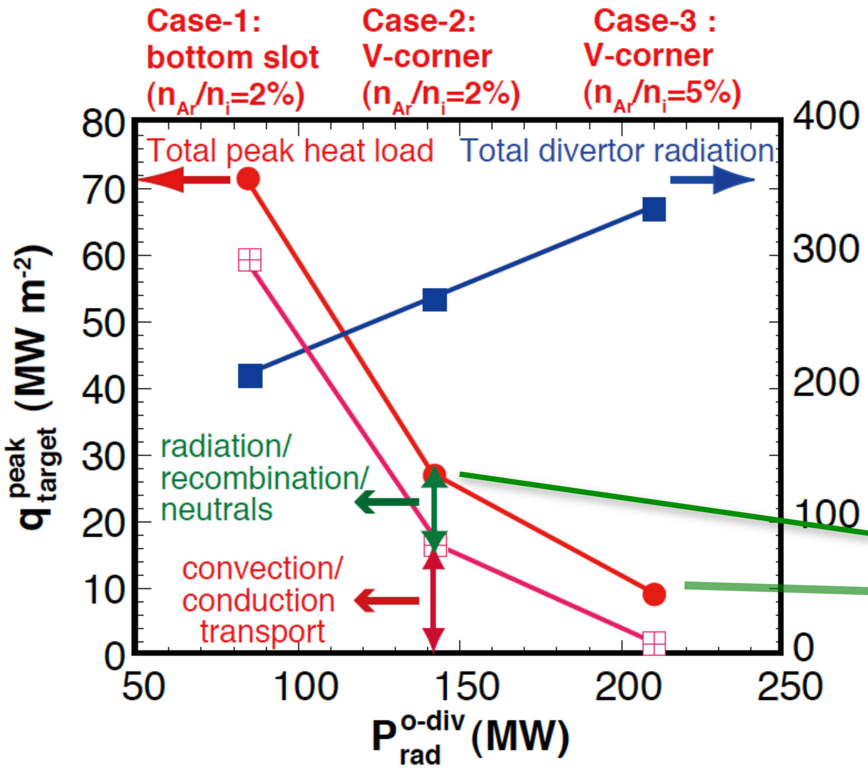


Peak heat load is sensitive to radiation region in the divertor

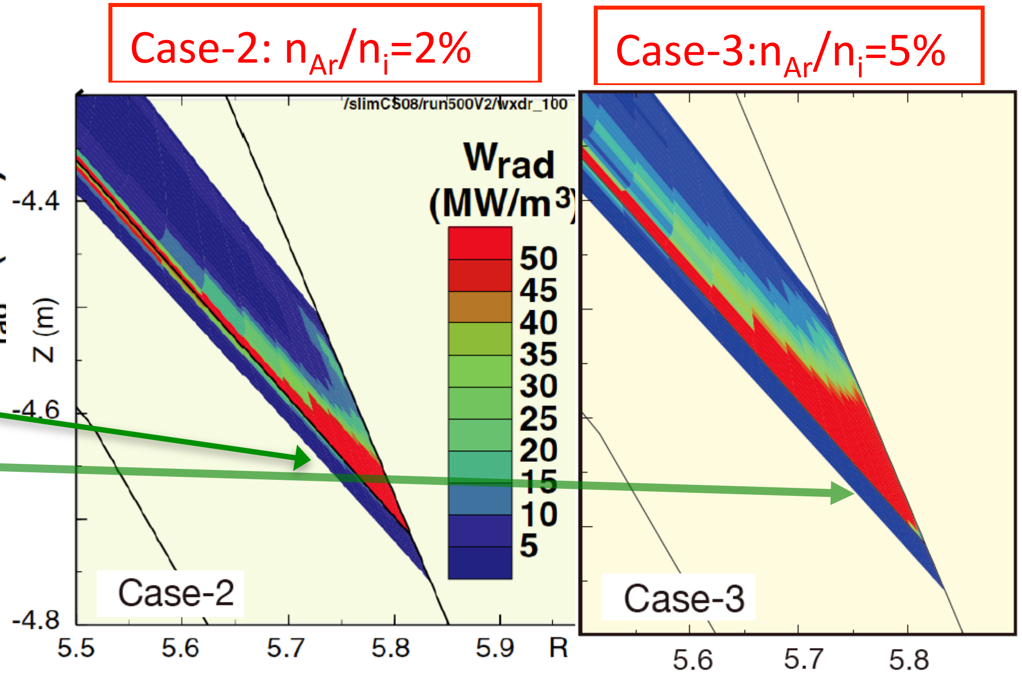
"Full detachment" is produced at radiation fraction: P_{rad}^{tot}/P_{out} ~92% (P_{rad}^{div}/P_{out} ~67%)

⇒ Radiation region extends in a wide divertor area, and peak heat loading is reduced to lower than 10MWm^{-2} (q^{peak} ~ 9 MWm^{-2})

Transport of impurities and control of radiation distribution are key to reduce q^{peak}



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

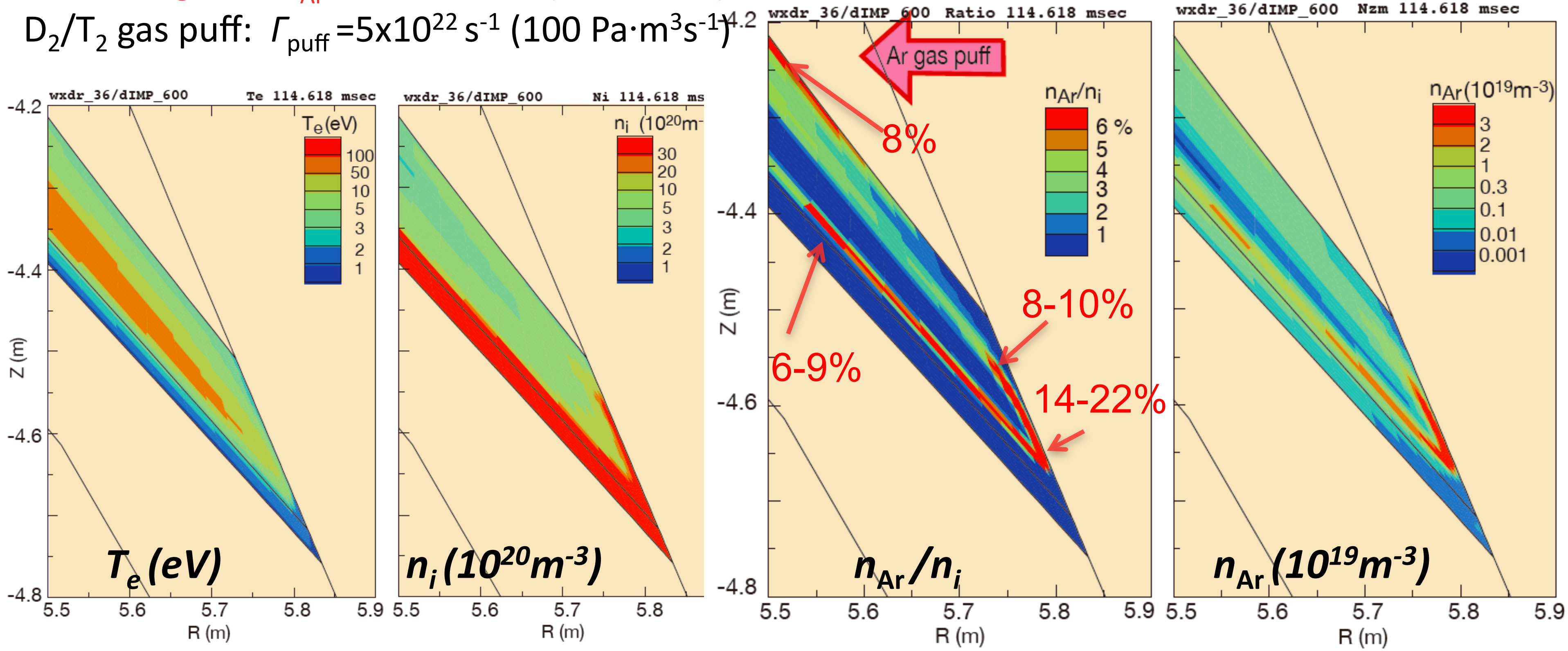


Development of MC modelling for Ar seeding

MC modelling for Ar seeding was investigated in the high recycling divertor:
Self-consistent coupling of the fluid plasma, MC neutral and impurity has been developed for the reactor divertor.

Ar transport was simulated till t ~ 100 ms (time scale of particle transport in divertor):
friction force by the plasma flow is dominant ⇒ Ar recycling is enhanced near target
⇔ Influence of ***thermal force*** on impurity transport is dominant **near separatrix**

Ar seeding rate : Γ_{Ar} = 2×10^{21} Ar/s (4 Pa·m^{3}s^{-1})
D_2/T_2 gas puff: Γ_{puff} = 5×10^{22} s^{-1} (100 Pa·m^{3}s^{-1})



Detachment was different depends on initial condition of the divertor

Self-consistent solutions for P_{out} =500 MW, Γ_{Ar} =2x10^{21} Ar/s were obtained using **the different initial conditions**:

IC-1: background plasma in Case-2 (n_{Ar}/n_i = 2%), partially detached divertor, was used.

IC-2: background plasma of full detached divertor was used.

- Different divertor plasma profiles were sustained *after the time scale of particle transport in the divertor (100 ms), while they were still transient*.

IC-1: plasma from Case-2

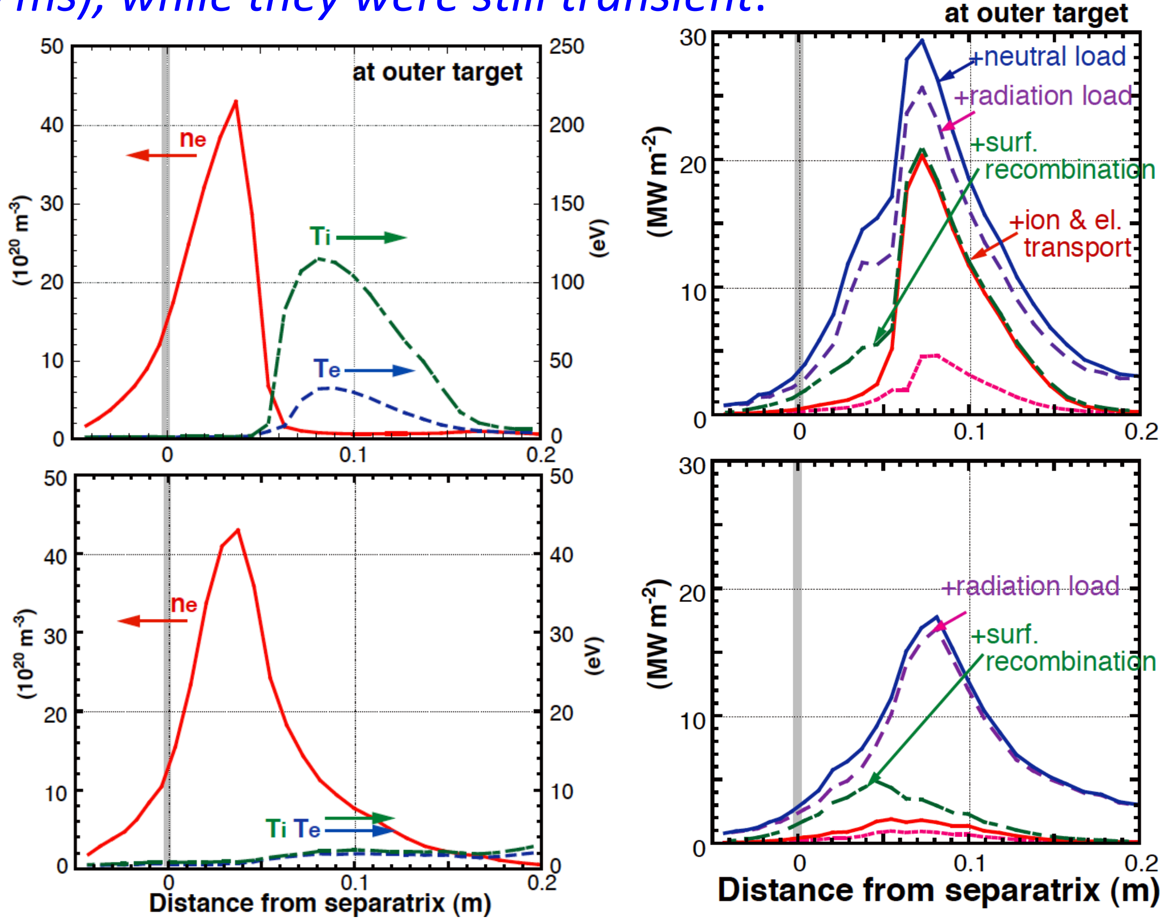
Detached near separatrix, and attached at the outer flux surfaces.

➔ max. q_{div} ~28 MWm^{-2}, where transport heat flux is dominant.

IC-2: full detached divertor

Full detached divertor is sustained, while radiation loss near the target becomes significant.

➔ max. q_{div} ~18 MWm^{-2}, where radiation power flux is dominant.

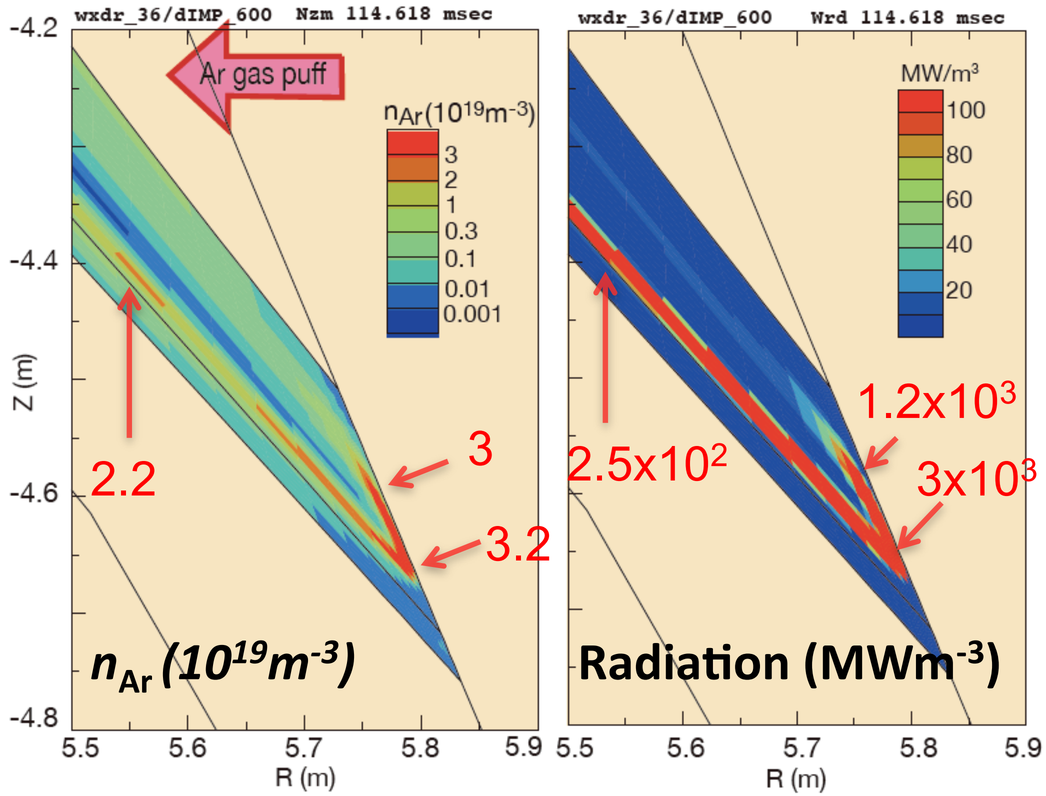


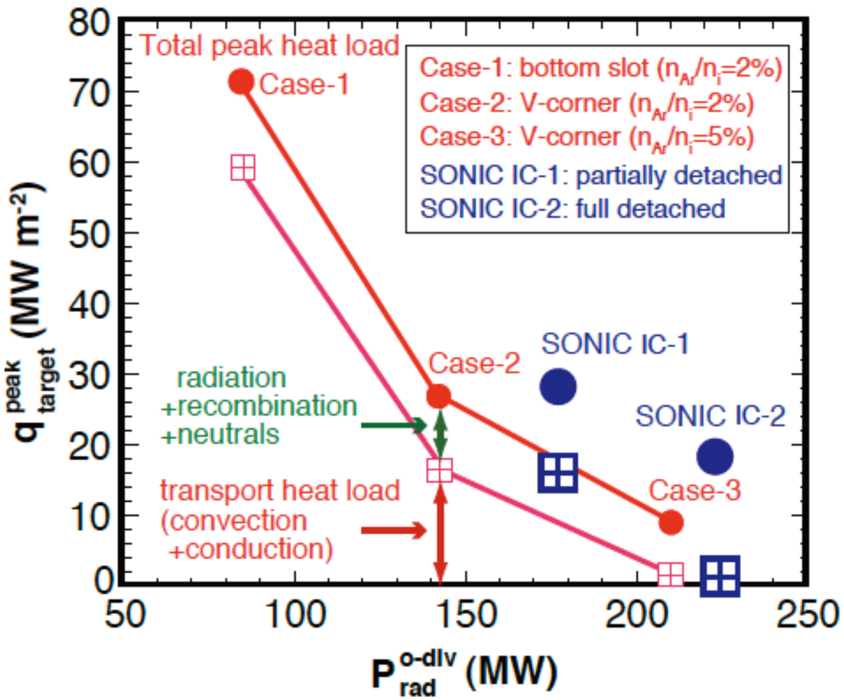
Radiation power load from MC-sim. is larger than *constant n_{Ar}/n_i model*

- Region with large radiation loss (> 100 MWm^{-3}) is localized just above the target, while the full detachment is sustained in the transport time scale (IC-2 case).
- Ar transport to the upstream SOL/edge is still transient:
  radiation at SOL/edge, P_{rad}^{edge} ~ 80MW, is smaller than P_{rad}^{edge}~130MW (n_{Ar}/n_i ~1%)

➔ **Investigation of Ar transport and radiation power at upstream SOL/edge is necessary to determine appropriate or combination of the radiators.**

SONIC IC-2: full detached divertor case





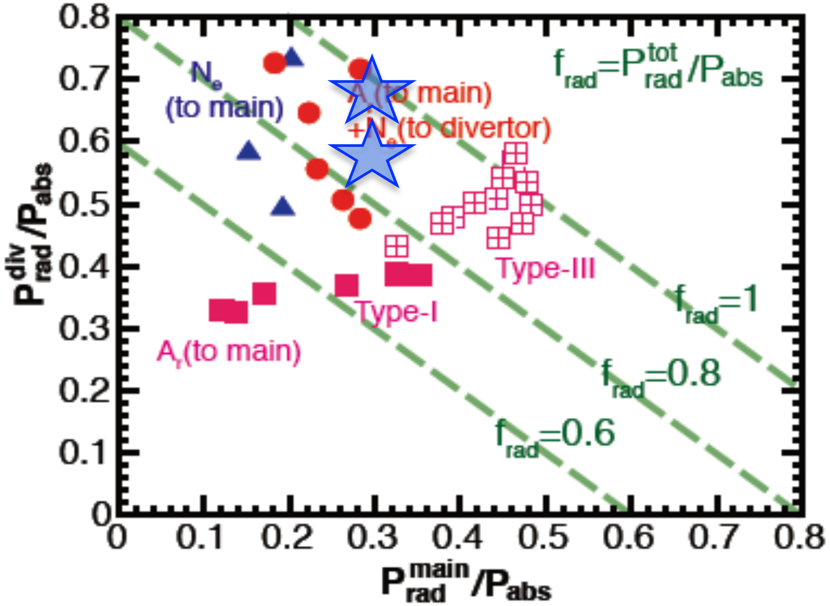
2.3 Issues of power and particle handling

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

***$P_{rad}^{SOL\&edge}$* ~150MW (*$P_{rad}^{SOL\&edge}/P_{out}$*~30%)** 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) can be controlled by multi-impurity seeding.
- Operation of the large $P_{rad}^{sol/dge}$ plasma (and high density) will be restricted also by degradation in the core plasma performance.

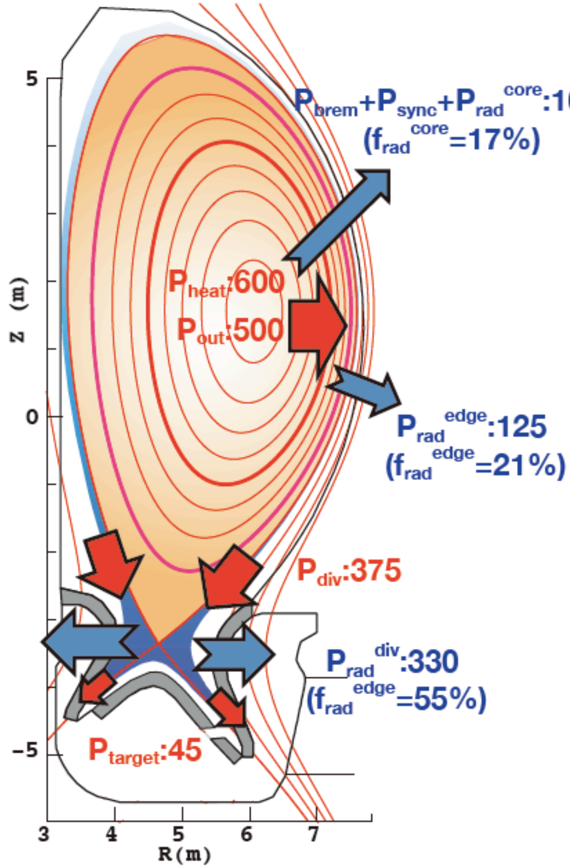
Ex.1 Radiation fractions at main and divertor for Ar and/or Ne seeding in JT-60U ELMyH-mode



N. Asakura, et al. Nucl. Fusion 49 (2009) 115010

P_{rad}^{tot}/P_{out} ~92%（P_{rad}^{div}/P_{out} ~67%)

Power flow in Slim CS (normalized to P_{heat})



Divertor pumping (He exhaust in the detach divertor)

He exhaust (α-particle production rate ~4×10^{21} s^{-1} for 3GW) is crucial,

⇔ pumping rate is low for formation and sustainment of “full detached divertor”

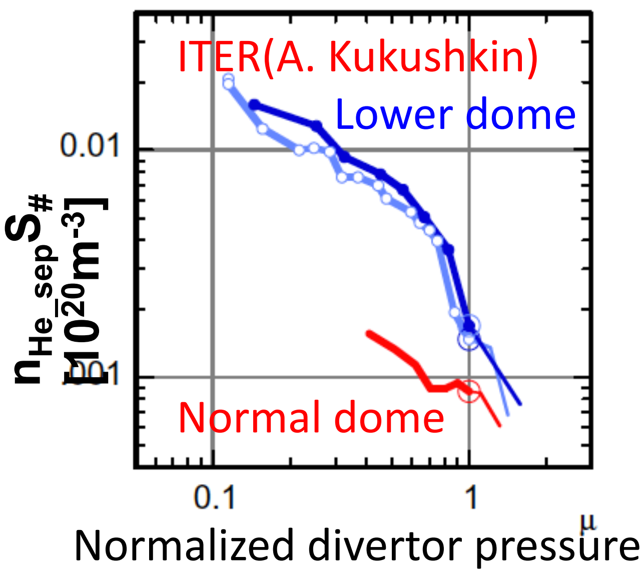
τ_{He}^{*}/τ_{E} <5-10, $\eta_{He} = (p_{He}/2p_{H2})^{div}/(n_{He}/n_{H})^{main} > 0.2$

- He density at the main edge depends on the divertor (Dome) structure :
- He exhaust efficiency is sensitive more than that for Fuelling gas

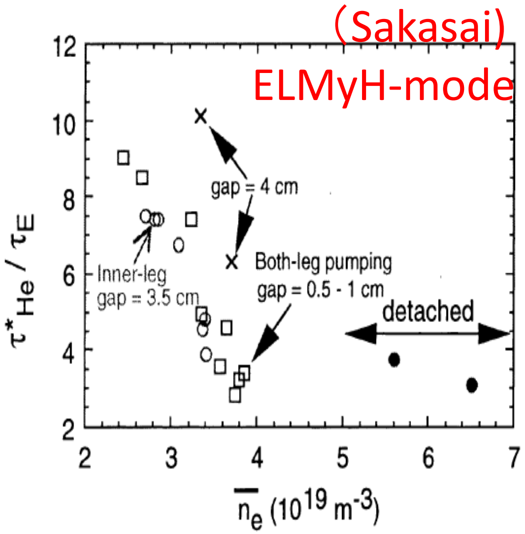
⇔ ITER and DEMO divertor size is larger than MFP of He

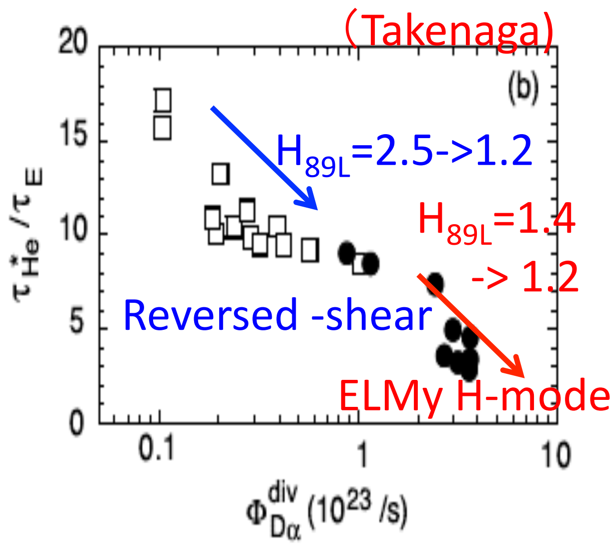
⇒ Minimum pumping rate and port are required for Tritium handling/retention and neutron shield.

He density at separatrix



He-NB& Ar-frost cryopump experiment (JT-60U)





Extension of ITER divertor concept to DEMO divertor ?

"Full detachment" is necessary for DEMO divertor, extending from ITER divertor

⇒ operation scenario of the divertor and main plasmas will be restricted by requirement of the high radiation loss and high edge density.

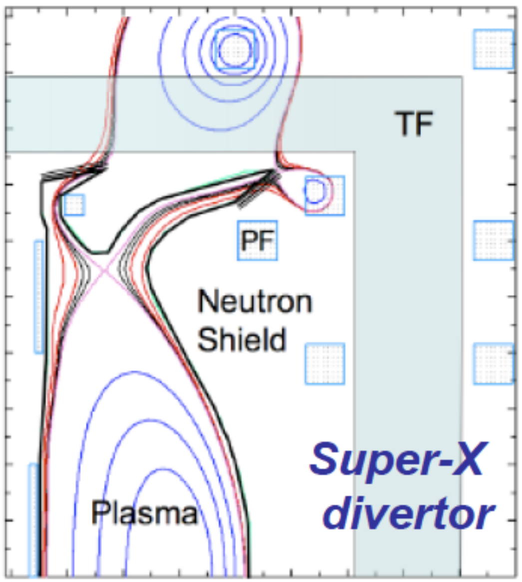
Design concept for DEMO divertor may be investigated from different viewpoints

(1) Super-X divertor ⇒ Divertor leg and target area are increased
long field-line and extending area to reduce T_e^{div} and q_{target}.

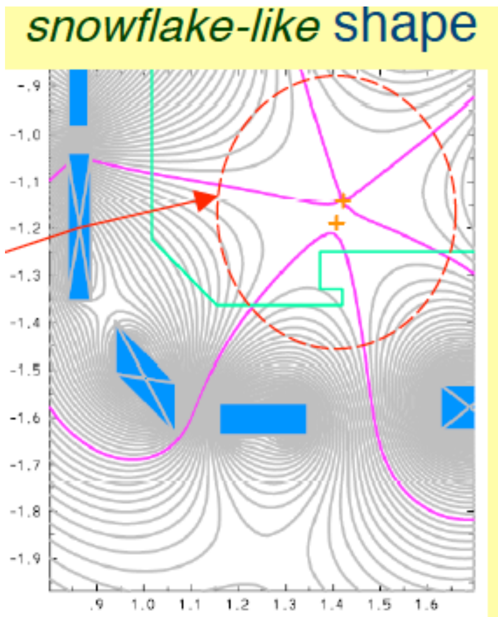
(2) Snowflake-like divertor ⇒ Flux expansion and effective field-line length

(3) Helical field ⇒ Enhancing diffusion by magnetic perturbation

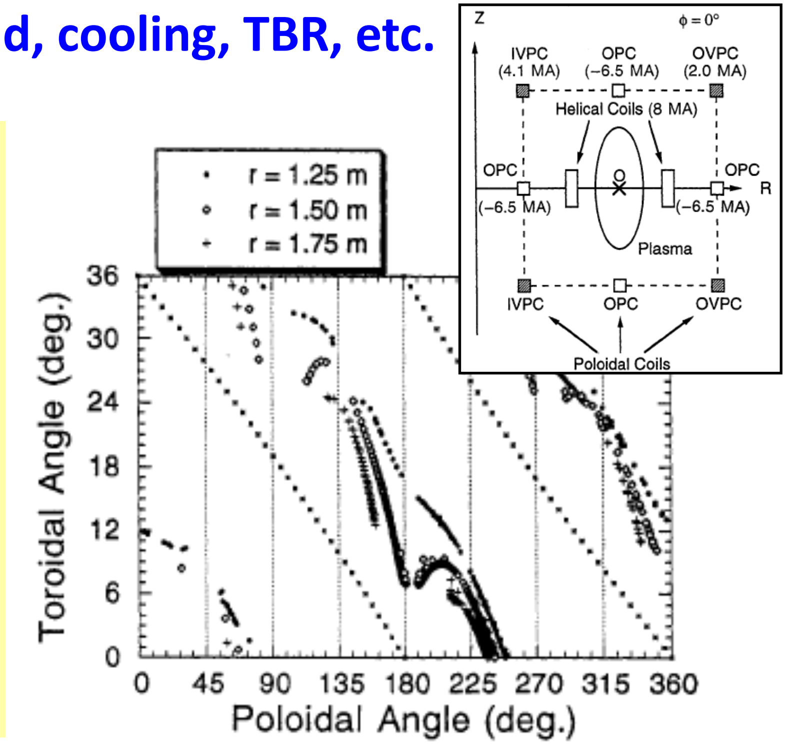
⇔ **Coil design issues are remained: neutron shield, cooling, TBR, etc.**



Kotschenreuther



Ryutov, Phys. Plas. 14 (2007) 064502



Takase, et al, Nucl. Fusion 35 (1995) 123

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   Tokamak DEMO strategy in Japan/JAEA
   SlimCS design concept

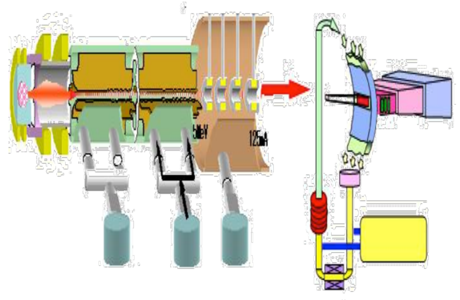
2. Divertor study for the DEMO reactor
   Power handling in divertor
   Divertor simulation
   Issues of power and particle handling

3. JAEA Aomori Fusion R&D center
   and BA Demo Design Activity

4. Summary

Broad Approaches comprises three Projects (JA-EU)

1) Engineering Validation and Engineering Design Activities for the International Fusion Materials Irradiation Facility (IFMIF/EVEDA)

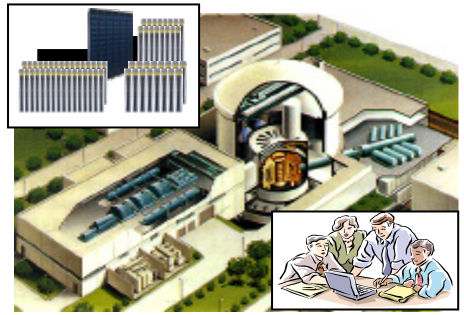


2) International Fusion Energy Research Center (IFERC),

a) DEMO Design and R&D coordination Center

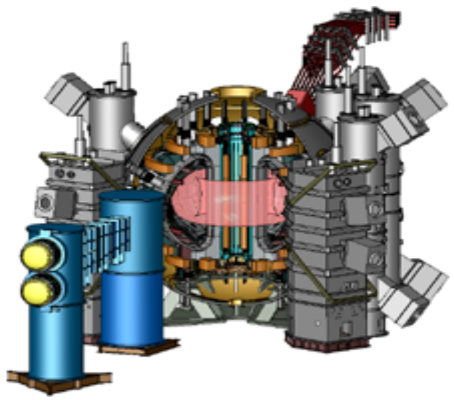
b) Computational Simulation Center

c) ITER Remote Experimentation Center



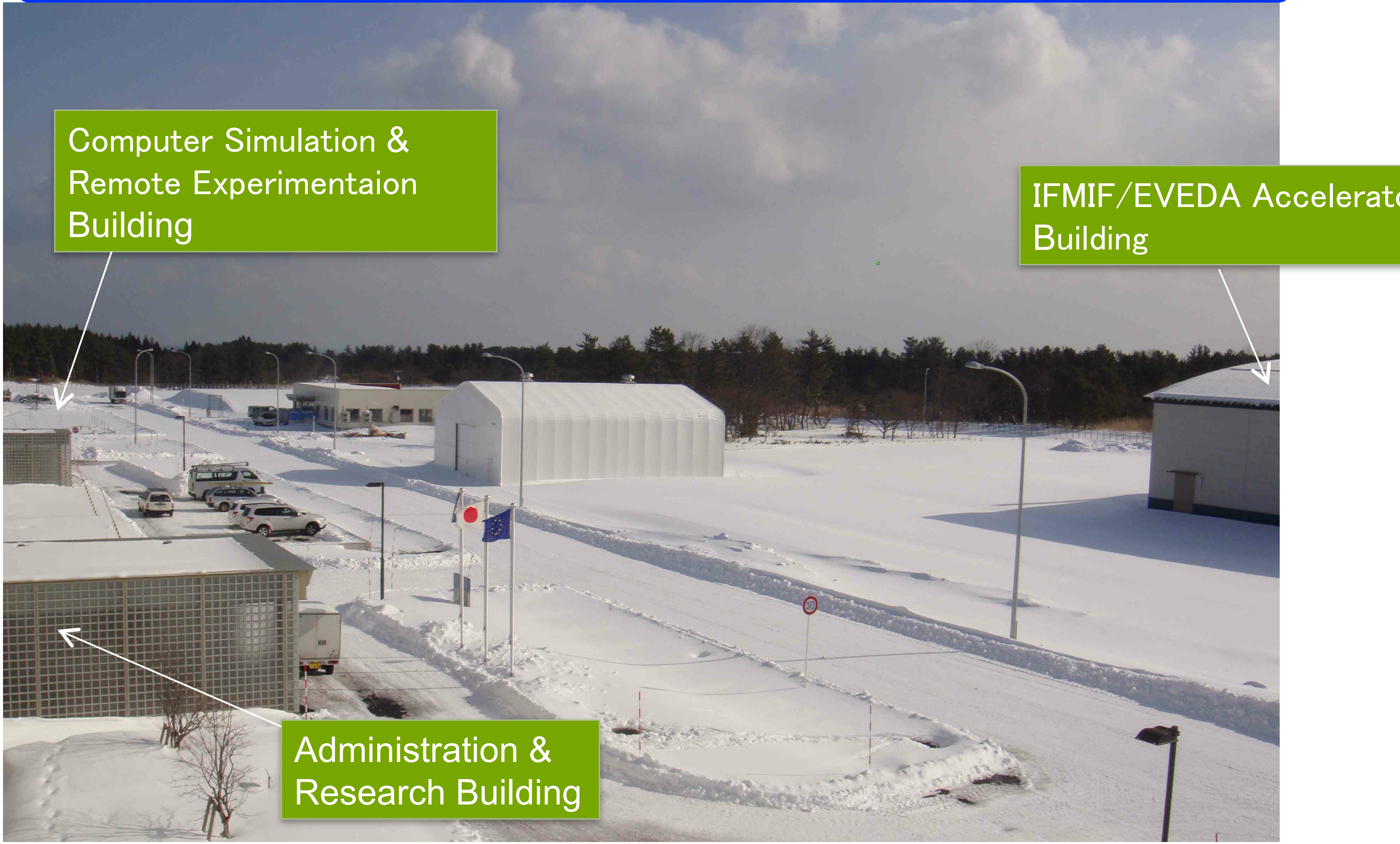
3) Satellite Tokamak Programme

Participation to upgrade of JT-60 tokamak to JT-60SA and its exploitation.

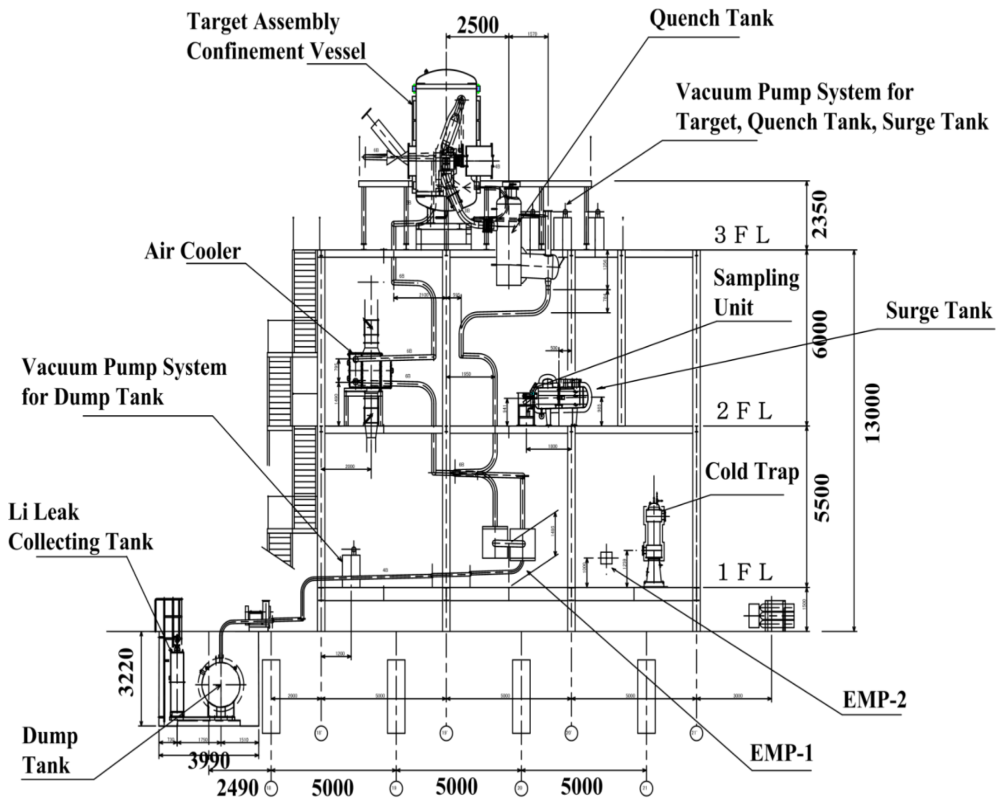


International Fusion Energy Research Center (IFERC)

International Fusion Energy Res. Center (Last week)



Present status of IFMIF/EVEDA Li Test loop

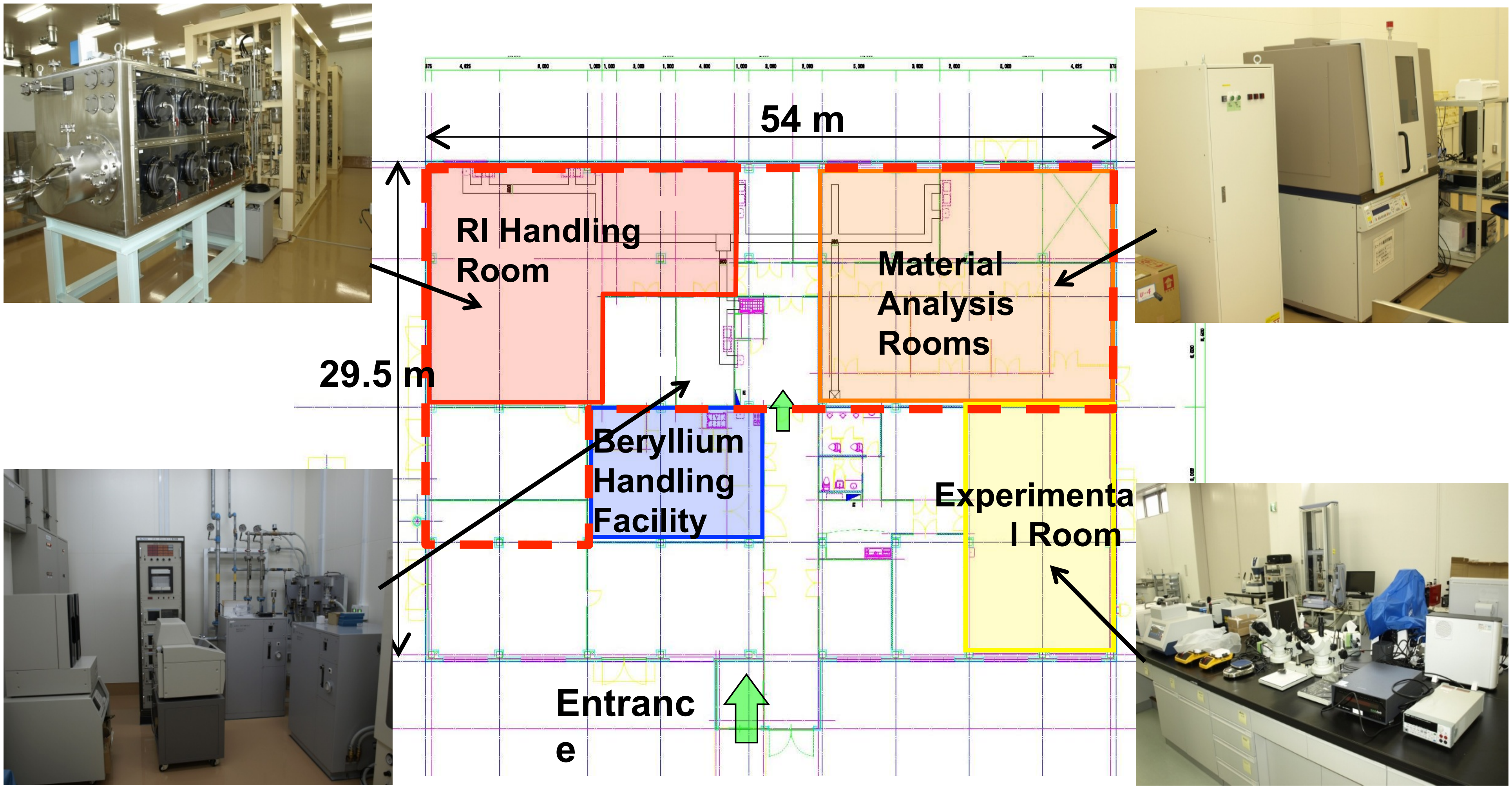


Construction of the Li Test Loop has almost been completed, and the acceptance test will be completed early in 2011.

DEMO R&D Building

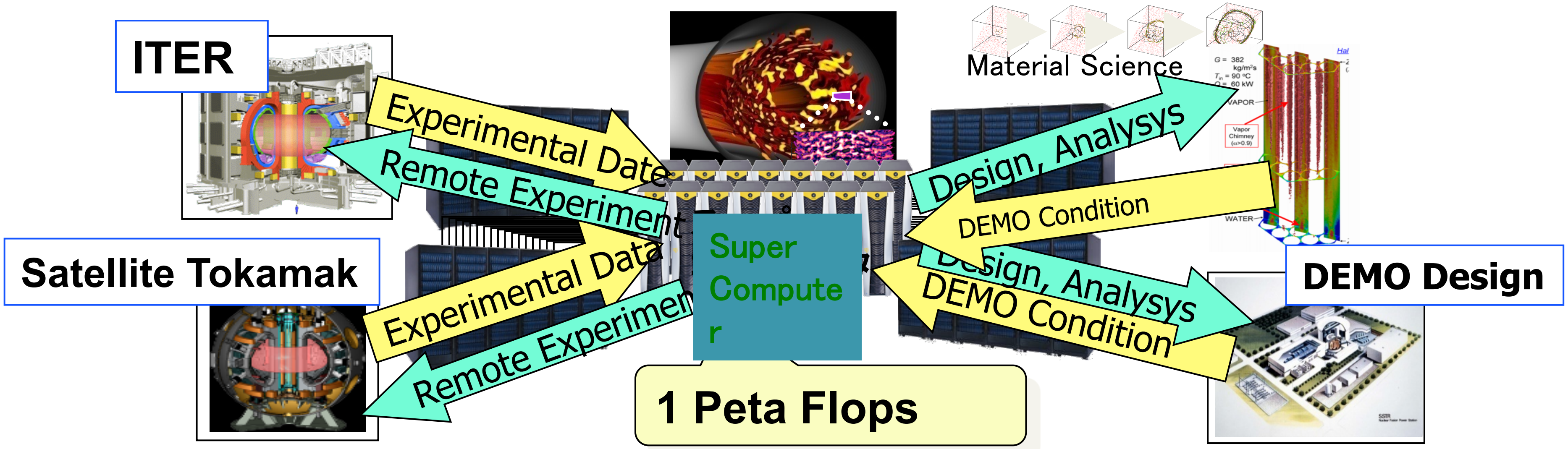
Technological R&D on key issues for the future DEMO reactor

a) SiCf/SiC Composites, b) Tritium Technology, c) Materials Engineering for DEMO Blanket,
d) Advanced Neutron Multiplier for DEMO Blanket, e) Advanced Tritium Breeders for DEMO Blanket



Computer Simulation Center (CSC)

Super computer (1Peta-flops) will be operational in January 2012 for fusion simulation.

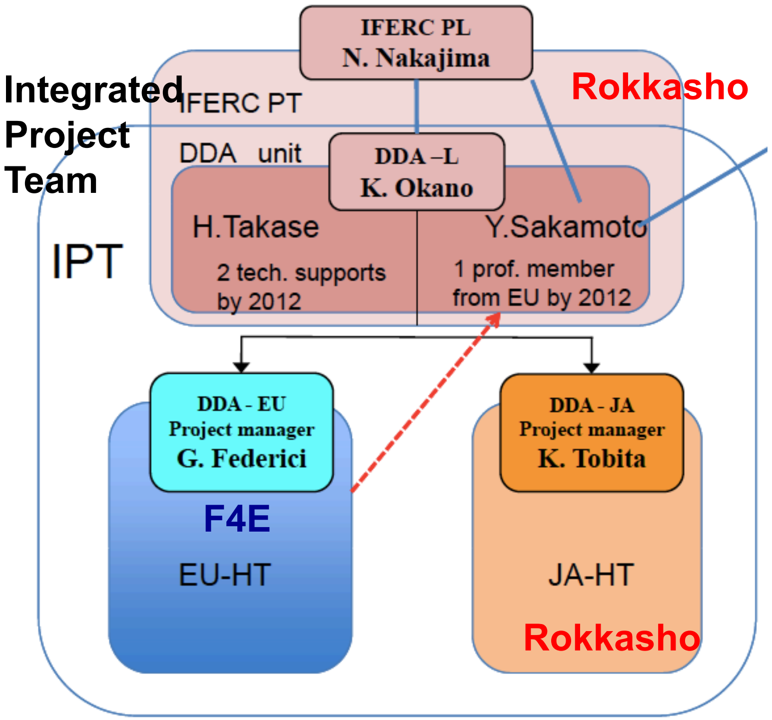


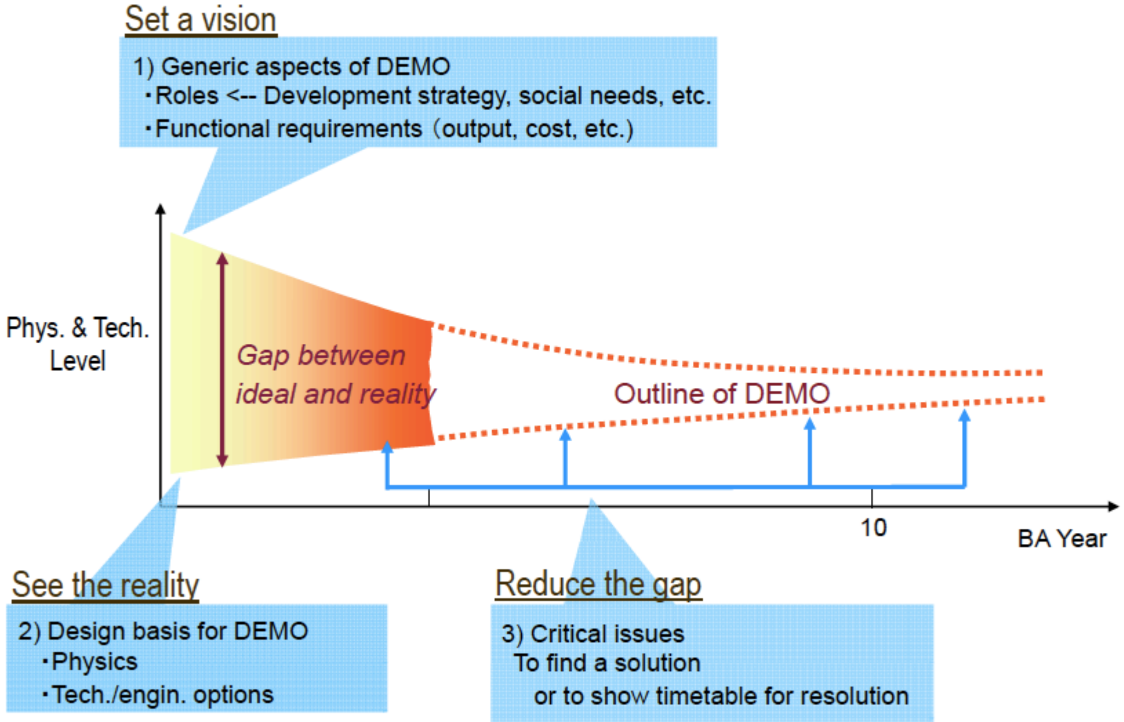
	200 7	2008	2009	2010	2011	2012	2013	2014	2015	2016
Computer Simulation Center			Preparation 1st Stage					Operation 2nd Stage		
Procurement				Preparation	Installation			Operation		
Special Working Group（SWG）	SWG-1		Selection							
			SWG-2	Operation Rule						

Demo Design Activity (DDA) entering Phase Two (2011)

BA DEMO Design Activity in IFERC Project

Objective: to establish a common basis for a DEMO design, including:
(i) provision and exchange of scientific and technical information;
(ii) DEMO conceptual design activities.



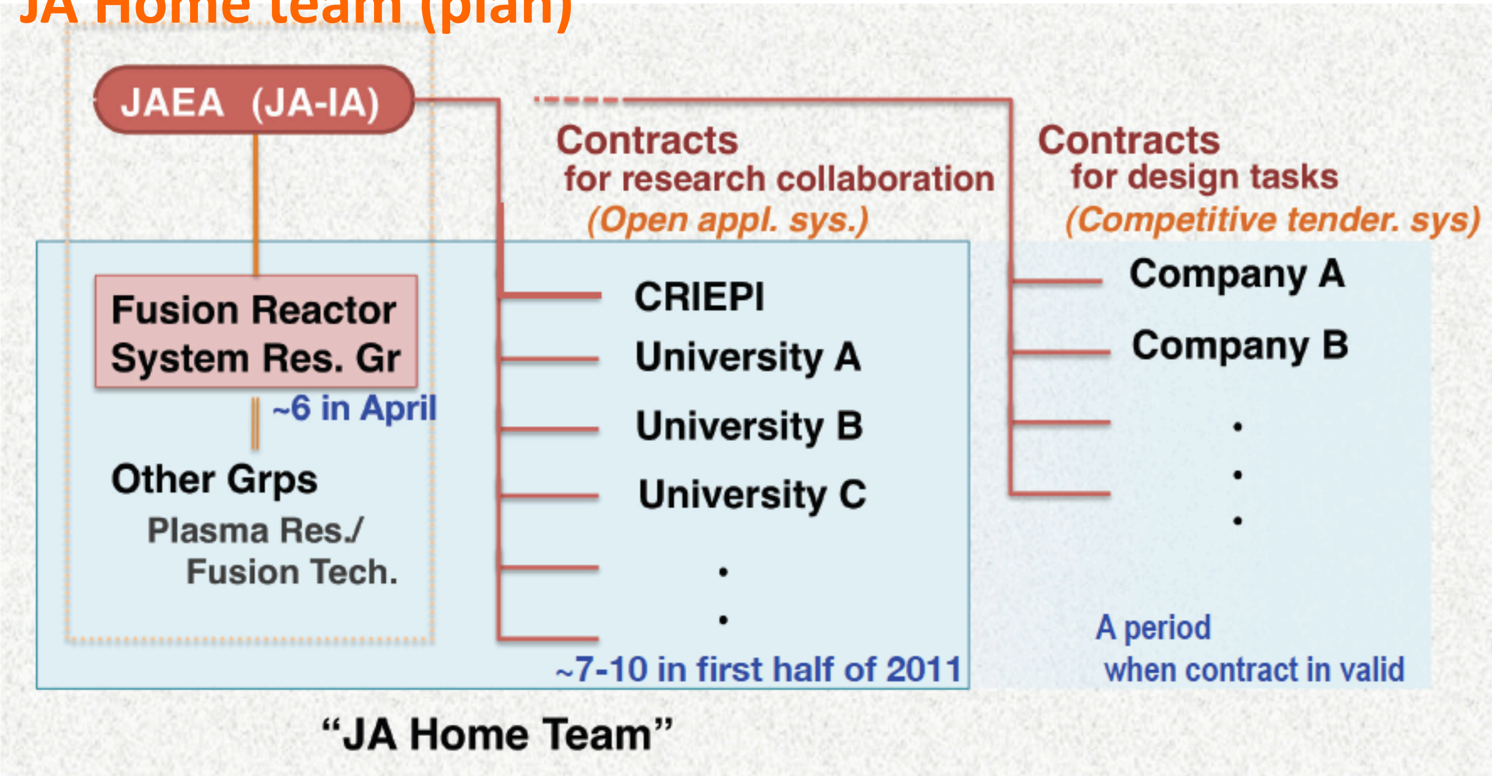


Phase Two: Development of pre-conceptual design options for DEMO

IFERC Project	2007	2008	2009	2010	2011	2012	2013	2014	2015	2016	2017
DEMO Design Coordination	Phase One Workshop: 1-2/year				Phase Two Joint work at Rokkasho to develop conceptual DEMO design						

Research collaboration starts from 2011

JA Home team (plan)



Plan of EU Home Team

F4E directly contacts to IPT and JA-HT, and conducts

Garching: EFDA (Core team), and

distributed Project Teams (associations)

Categories of working issues in Phase Two

Generic issues

i) Role of DEMO, ii) Requirements for DEMO, iii) Development strategy

Design issues on "Plasma Physics"

i) Required physics parameter, ii) MHD equilibrium and shaping

Design issues on "Plasma Engineering"

i) Plasma control, ii) Current drive, iii) Divertor and plasma wall interaction

Design issues on "Engineering"

i) Blanket, ii) Magnet

"System issues" on DEMO Design

i) Maintenance, ii) Safety, iii) Systemic issues

System Code

Working Plan from Japan in Phase Two-A

Activities in Phase Two-A (2011-12)

a: Design criteria and cost model

- To discuss and agree on
  DEMO objectives, requirements, standards, design rules, etc.

b: Analysis key design issues and options and launch preliminary design work

- To review and launch design work on
  key design issues such as power exhaust (divertor), extrantion and breeding (blanket), remote maintenance, …

c: Preparation and start implementation of system design code

- To develop a common system desig code
  — *Evaluation of existing codes, revisit modelings on physics, engineering and cost*

Divertor armor material is investigate from viewpoints more than ITER

Tungsten is foreseen as PFCs (divertor and first wall) in DEMO reactor,

Engineering properties:

(1) High thermal conductivity, (2) High melting temperature, (3) Low activation,

Performance of high-temperature fusion plasma:

(4) Compatible with high temp. fusion plasma --- low accumulation/reliable control

Divertor operation/ PWI performance:

(5) **Life time:** Low erosion rate/high threshold energy, Low surface damage (blistering, crack, bubble, etc.), Melting dynamics (influence on structure materials),

(6) **Safety:** T-retention, Dust generation, Activation

Tritium-breading/fuel-circulation (incl. First wall):

T-retention, Dust production, Neutron-energy spectrum (reflection, deflection)

⇒ Development of armor materials (W-alloy, fine-grain-W, etc.)

	ITER (1 shot)	DEMO (continuous)
T_w at SS (°C) water-cool	~1000 [base 100-200]	<1200 [base 290]
T_e near strike-point (eV)	1-30	1-20
Fuel ion fluence (m^{-2})	5×10^{25} - 5×10^{26} (400s)	10^{30}-10^{31} (~year)
He ion fluence (m^{-2})	10^{24} - 10^{25} (400s)	10^{29}-10^{30} (~year)
Neutron fluence (dpa)	~0.5 (~5 year)	20-100 (1-3 year)

Divertor system issues for DEMO

Divertor Technology and System has been developed in different concept/materials

Divertor cooling techniques and (armor/structure) Material selection:

- Water cooling (T_{base}~ 290°C, 4MPa PWR, 4-8 m/s in SlimCS) : W & RAFMS
  ⇒ better heat transfer coefficient in conventional simple structure
- He gas cooling (T_{base}~ 600°C, 10MPa, in HEMJ & T-tube) : W & W-arroy & ODS-FM
  ⇒ non-active nuclear & chemical, safety, Jet impingement to increase heat transfer

Feasible divertor system should be investigated, from viewpoints:

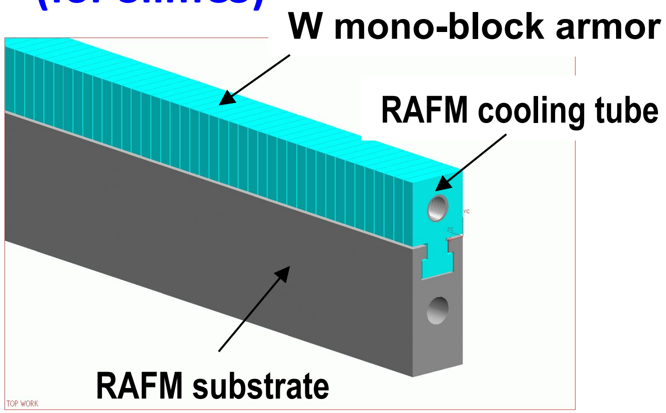
(1) Heat removal efficiency --- potential improvement larger than 10 MWm^{-2}

(2) Safety such as abnormal events (leak, crack, etc.) and detection

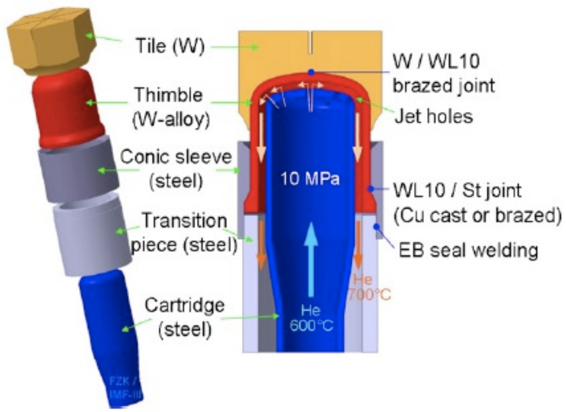
(3) Material combination (armor, structural, joint) at different operation temperature

(4) Joint material and technology

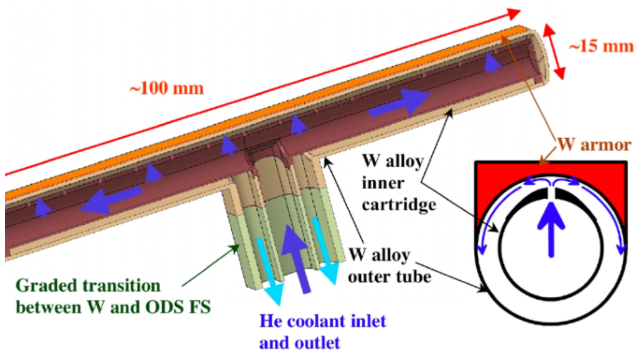
Water cooling component (for SlimCS)



He/W component (EU-Finger)



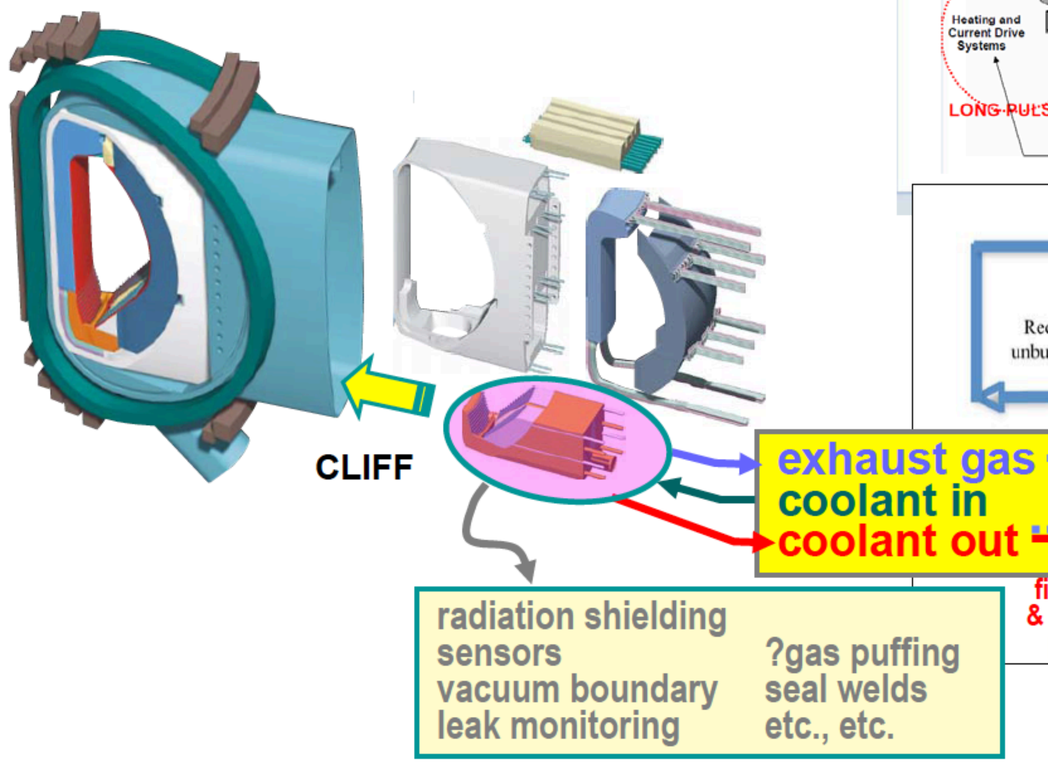
He/W component (ARIES: T-tube)

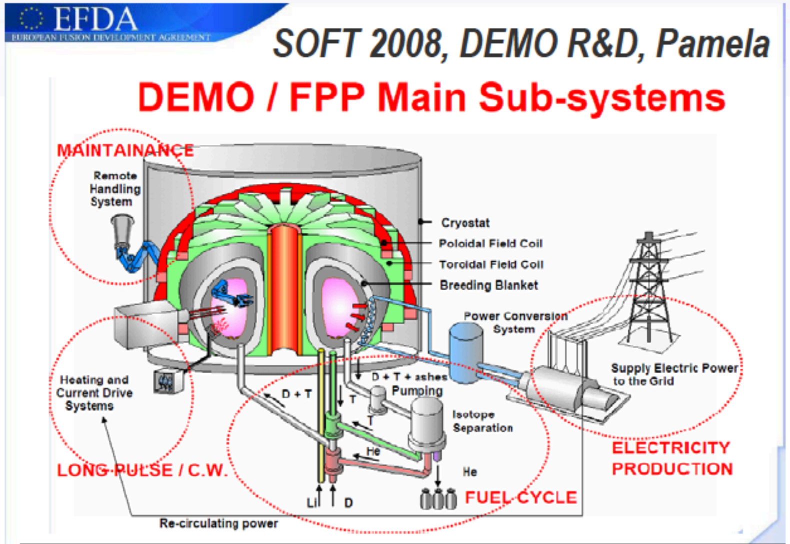


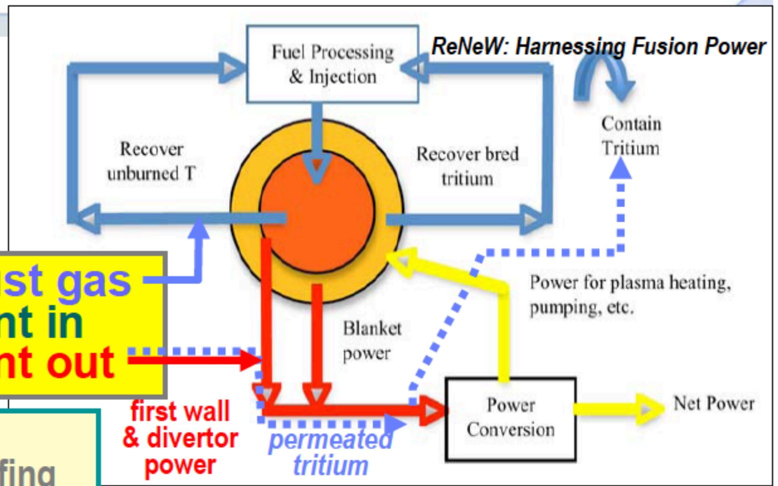
Power plant Divertor: heat removal⇒ Electricity Generation

The Divertor - a complex heat removal system

To produce energy efficiently, we must use the (20%) power going to the divertor; so the **<u>divertor must be integrated with heat removal from the first wall and blanket.</u>**







Sandia National Laboratories

4. Summary : DEMO divertor simulation

- **Power handling scenario such as P_{out} =500MW for DEMO divertor was critical issue: Design of the huge power handling must be accomplished at least by simulation.**

⇒ *Intense Ar seeding (such as n_{Ar}/n_e~5%) in the divertor extending from ITER will* produces **the full detached divertor** ($P_{rad}^{SOL/edge}/P_{out}$ ~30% and P_{rad}^{tot}/P_{out} > 95%)

- SONIC with impurity Monte-Carlo has been developed for Ar impurity seeding:

Self-consistent coupling of the fluid plasma, MC neutral and impurity has been developed for the reactor divertor (*but still transient at the upstream SOL/edge*)
⇒ Region with large radiation loss (>100 MWm^{-3}) is localized just above the target, while the full detachment is sustained in transport time scale in the divertor.

- Radiator/the combination and divertor geometries appropriate for controlling the full divertor detachment will be investigated.
- He exhaust (pumping) consistent with formation of detached divertor is a crucial.

⇒ Operation of the edge and core plasma would be restricted, and development of handling target load q_{target}>10MW/m^2 will be necessary.

Advanced DEMO divertor scenarios need to investigate coil design issues.

Summary : IFERC DEMO Design Activity

- **"Divertor and PWI" is important key design issue of Plasma Engineering in BA DDA:**

  Other than the divertor physics design incl. the advanced DEMO divertor, investigation of the following issues is proposed:

  Divertor armor material should have appropriate properties for various viewpoints

  Engineering

  Performance of high-temperature fusion plasma:

  Divertor operation/ PWI performance (Life time, Safety)

  Tritium-breading/Fuel-circulation (incl. First wall)

  Feasible divertor system should be investigated/developed, from viewpoints:

  (1) Heat removal efficiency --- potential improvement larger than 10 MWm^{-2}

  (2) Safety such as transient/abnormal events (leak, crack, etc.) and detection

  (3) Material combination at different operation temperature

  (4) Joint material and technology

  Power plant Divertor: heat removal⇒ Electricity Generation

In addition

- Steady-state distribution (t~1s) will be investigated in **IFARC parallel computer (1PTlops)** with effective speed of 100TFlops (x5000 faster than JAEA: 20GFlops). Now, 9 hours are required for SONIC calculation of 10ms (need 100 times more).

  ⇒ SONIC calculation in steady-state (1s) will be 0.2 hour !

5. Summary and issues for the DEMO divertor (2/2)

- **Divertor armor material should have appropriate properties for various viewpoints**

  Engineering:

  (1) high thermal conductivity, (2) high melting temperature, (3) low activation,

  Performance of high-temperature fusion plasma:

  (4) compatible with fusion plasma --- low accumulation/reliable control

  Divertor operation/ PWI performance:

  (5) Life time: low erosion rate, Low surface damage, Melting dynamics,

  (6) Safety: T-retention, dust generation, activation

  Tritium-breading/Fuel-circulation (incl. First wall)

- **Feasible divertor system should be investigated, from viewpoints:**

  (1) Heat removal efficiency --- potential improvement larger than 10 MWm^{-2}

  (2) Safety such as transient/abnormal events (leak, crack, etc.) and detection

  (3) Material combination at different operation temperature

  (4) Joint material and technology

Power plant Divertor: heat removal⇒ Electricity Generation

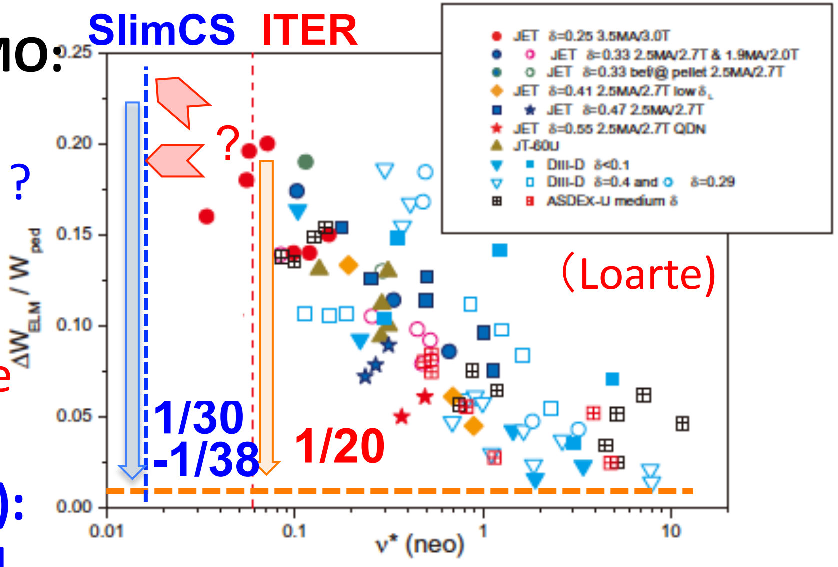
Prediction and control of transient (ELM) heat load

<u>ELM mitigation</u> **is the most important for ITER&DEMO:**

- T_e^{ped}, W_{ped} will be increased 1.6-2 times than ITER
- ν^*(neo) is increased (0.06 → 0.016) ⇒ $\Delta W_{ELM}/W_{ped}$?
- Acceptable ΔW_{ELM} is increased from 1MJ (ITER) to 1.6MJ (SlimCS) due to increase of wetted area.

⇒ ELM mitigation should be reinforced to acceptable $\Delta W_{ELM}/W_{ped}$ =1/30-1/38 (SlimCS)

Physics and Engineering issues (incl. ITER/ITPA R&D): Grassy ELM, QH-mod, pellet pace-making, RMP-coil, edge-ECH, Type-III, etc.



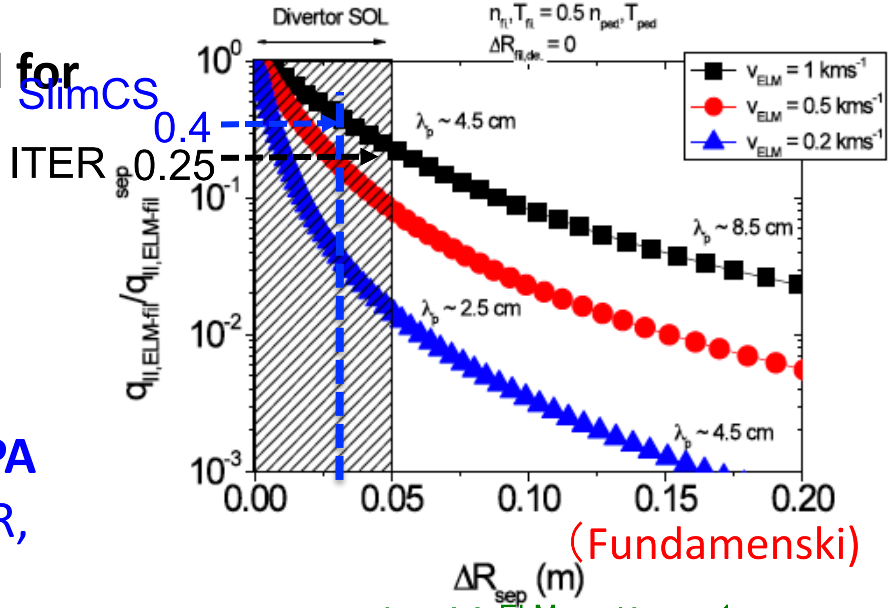
ν^*(neo) = $Rq_{95}A^{1.5}/\lambda_{ee}$

<u>**Heat load to FW (near upper null)**</u> **is increased for high shaping configuration (δ_{95}, κ_{95}):**

- Design of SOL configuration, such as 2^{nd} separatrix distance Δr_{2sep}= 3cm (SlimCS), first wall distance Δr_{wall}< 20cm (SlimCS),

should be consistent with

Physics and Engineering issues (incl. ITER/ITPA R&D): ELM propagation, Blanket width for TBR, Conducting shell design(r_s/a=1.35), etc.



assuming $V_\perp^{ELM}$ = 1kms^{-1}

Plasma Facing Component

Divertor armor material should have appropriate properties for various viewpoints,

Engineering properties:

(1) High thermal conductivity, (2) High melting temperature, (3) Low activation,

Performance of high-temperature fusion plasma:

(4) Compatible with high temp. fusion plasma --- low accumulation/reliable control

Divertor operation/ PWI performance:

(5) **Life time:** Low erosion rate/high threshold energy, Low surface damage (blistering, crack, bubble, etc.), Melting dynamics (influence on structure materials),

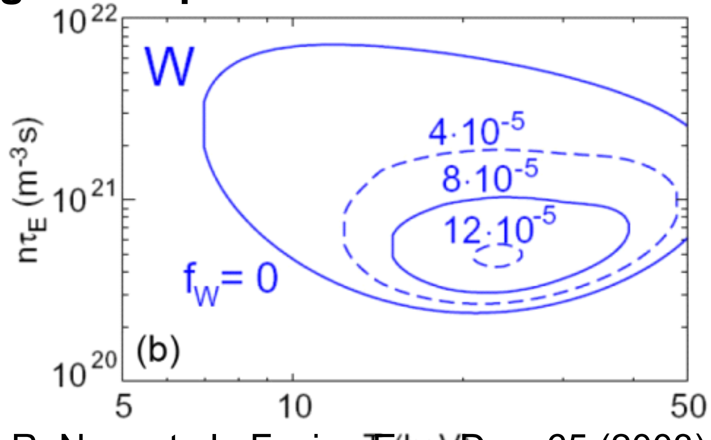
(6) **Safety:** T-retention, Dust generation, Activation

Tritium-breading/fuel-circulation (incl. First wall):

T-retention, Dust production, Neutron-energy spectrum (reflection, deflection)

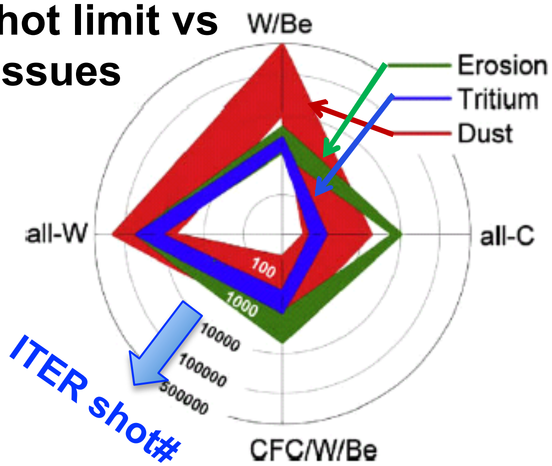
⇒ Development of armor materials (W-alloy, fine-grain-W, etc.)

Ignition operation vs W-concentration



R. Neu, et al., Fusion Eng. Des. 65 (2003) 367.

ITER shot limit vs PWI issues

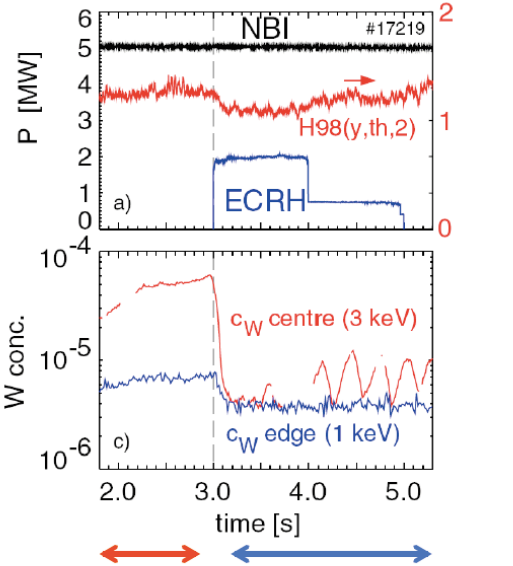


J. Roth, et al., J. Nucl. Mater. 390–391 (2009) 1

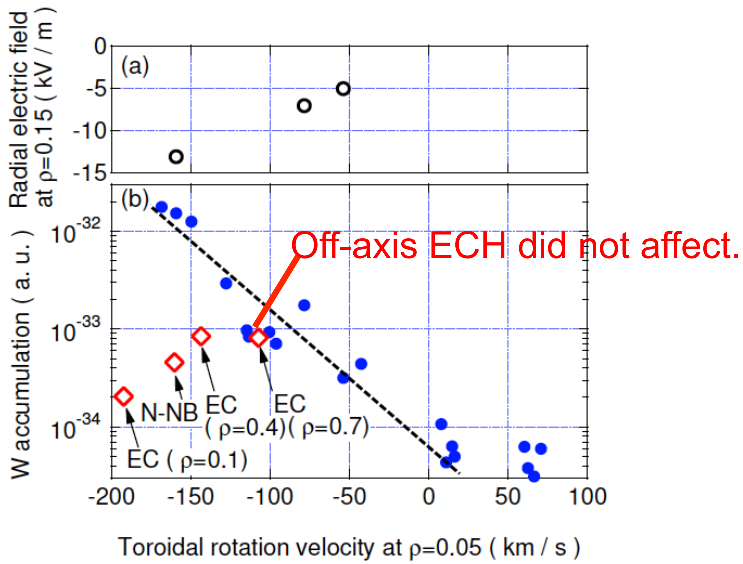
Accumulation and Control of high-Z impurity

Research of high-Z (W) impurity accumulation and its control have been developed:

- **Accumulation of W was mostly associated with density peaking (AUG)/ Counter-rotation(JT-60U)** ⇒ **Determination of W impurity transport model is required:**
  Some mechanisms of Internal E_r and rotation (acceleration of W) were proposed.
- **Control (exhaust) techniques of W-transport should be determined for high temperature DEMO plasma** (higher charge-state and lower collisionality)
  ⇒ **ECH center heating (producing peak temperature profile) and gas puff (producing flat density profile) were reproduced in some devices.**
  Other techniques (rotation, ST-control, α-heating effect, etc) are investigated.
- **W at RF limiter and first wall penetrates into core plasma rather than divertor.**



R. Neu, et al., PPCF 44 (2002) 811



(Nakano, et al IAEA 2008)

ELM plasma pulse and high heat flux to W-armor

Divertor target (mono-block) structure and Melting layer :

Plasma gun exposure (QSPA) and TEXTOR W-limiter experiment showed “melting layer” dynamics (by plasma pressure and JxB) ⇒ formation of “bridge” between W-blocks

⇒ extra thermal stress on cooling-pipe/joint ⇒ damage on cooling-pile in worst case

Divertor operation in Reactor plasma and Life time :

Combination of repeated plasma pulses (0.5MJ/m^2, 0.5ms) & heat flux (10-20MW/m^2)

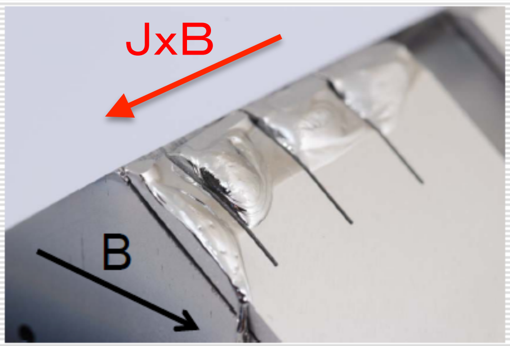
⇒ primary (grain boundary) and secondary (surface) cracks :

Acceleration? Plasma pulse affect melting/large thermal stress on the surface ⇔ High heat flux (SS) affects thermal stress into cooling pipe

W-block tile test (QSPA)

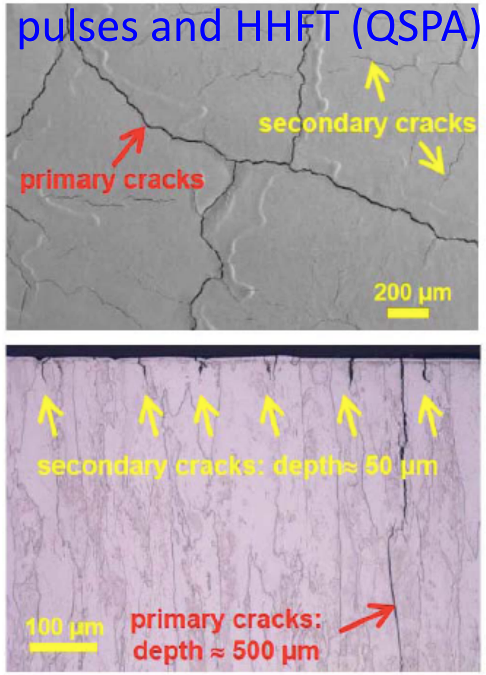
E = 1.0 MJm^{-2} Δt = 500 μs (100 pulses)

W test limiter in TEXTOR after one discharge



角を丸めていないスリット

W-block tile test by plasma pulses and HHFT (QSPA)



Deterioration of material properties: neutron irradiation

Enhancement of defects and increasing DBTT due to neutron irradiation

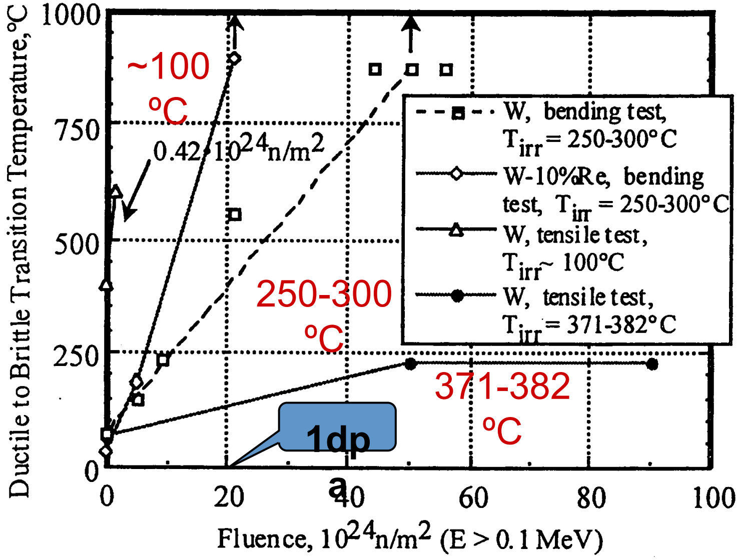
⇒ deterioration of thermal conduction and stress at interlayer is anticipated.

- **DBTT increase with neutron flux**: saturated at 250°C for T_{wall}=370°C
  but effect of high energy fusion neutron flux (14MeV) is also concerned.

- **Change in material property by transmutation: W -> Re, Os**
  Thermal diffusivity is decreases with increase Re impurity
  Mechanical property will be degradated with contamination of Re and Os

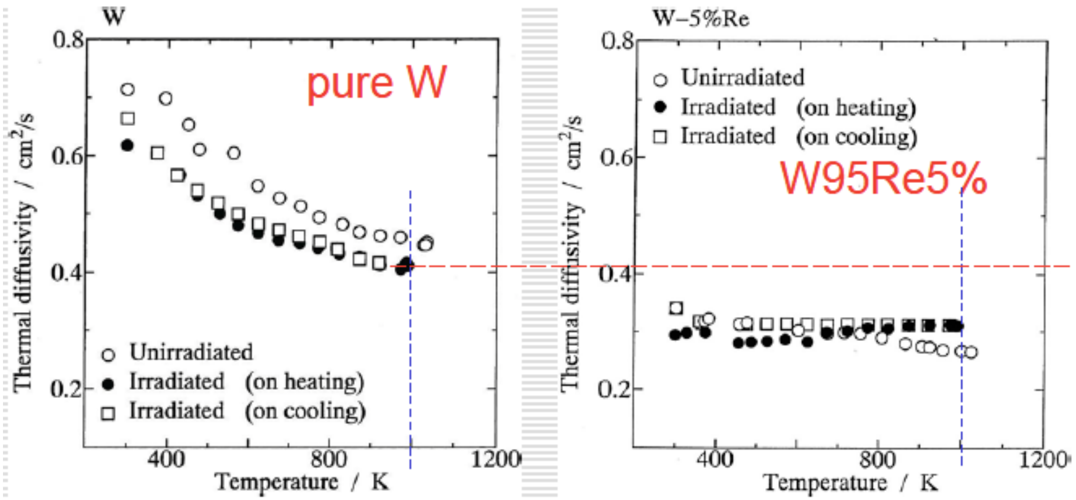
- **Change in PWI properties (T-retention, blistering, He-bubble/nano-structure, etc.)**
  Database is restricted by neutron fluence/dpa.

DBTT for W with n-irradiation



J.W. Davis et al., J. Nucl. Mater. 258-263 (1998) 308

Thermal diffusivity of W and W-Re alloy



Fujitsuka et al., J. Nucl. Mater. 283-287 (2000) 1148

W-PSI issues expected from ITPA/ITER R&D

Tungsten is foreseen as PFCs (divertor and first wall) in DEMO reactor.

PSI properties have been investigated for application of the ITER divertor.

Following W-PWI issues/database should be focused under the high fluence:

(1) “bubbles”, “holes”, “nano-structure” formation by He ion irradiation at T_W >700°C

(2) Neutron irradiation effects : defect, blistering, increasing DBTT and T-retention.

(3) Target design/arrange of mono-block armors and melt-layer dynamics.

Their dependence/threshold on temperature and fluence and energy are investigated in recent experiments under the ITER-level condition.

On the other hand, **fluences of D/T/He ions and neutrons in DEMO reactors are far beyond existing database.**

	ITER (1 shot)	DEMO (continuous)
T_w at SS (°C) water-cool	~1000 [base 100-200]	<1200 [base 290]
T_e near strike-point (eV)	1-30	1-20
Fuel ion fluence (m^{-2})	5×10^{25} - 5×10^{26} (400s)	10^{30}-10^{31} (~year)
He ion fluence (m^{-2})	10^{24} - 10^{25} (400s)	10^{29}-10^{30} (~year)
Neutron fluence (dpa)	~0.5 (~5 year)	20-100 (1-3 year)

Generic Issues

i) Role of DEMO

It was confirmed that EU and Japan had a common view on the role of DEMO. Toward early realization of fusion energy, DEMO is considered as a single step between ITER and a first commercial reactor, which means that DEMO would be: i) a first integrated machine both in plasma performance and in nuclear reactor technology; (ii) a last integrated R&D device before the first fusion power plant. In this sense, DEMO needs to meet wide-ranging requirements such as engineering feasibility, operation reliability and economic prospect.

ii) Requirements for DEMO

Regarding requirements for DEMO, there was a divergence in opinion between EU and Japan. Japan conceives a DEMO having core dimension similar to that of ITER, producing a Giga-watt level of power, being capable of continuous operation and self-sufficient tritium supply. In contrast, EU does not think that the dimension, power level and continuous operation are the important requirements. On the other hand, both Parties agreed on the importance of high plant availability.

iii) Development strategy

The roadmap toward DEMO will be subject to change in accordance with various situations such as the development program of each country and budgetary situation. Therefore, discussion on roadmap and development strategy was carried out regarding rough timeline. For the demonstration of fusion power generation in the middle of this century, EU and Japan plan to conduct a conceptual design of DEMO in the BA period, and move on to the engineering design phase (~10 years) and the construction one (~10 years) successively.

Design Issues on "Plasma Physics"

i) Required physics parameter

Although fusion power plant designs have a wide diversity of design parameters, there is a common requirement of high density operation ($n/n_{GW} \geq 1$) with high confinement ($HHy2 \geq 1$) beyond the present target of ITER. In addition, for production of plant-level electricity ($P_{elec} \geq$ several hundreds of MWe), access to high beta regime seems to be necessary. According to the calculation in Demo-CREST, high β_N and high density operation beyond ITER allows electricity generation of 1 GWe at the sending end when thermal efficiency is 30%. Furthermore, DEMO plasma requires the simultaneous achievement of high performance parameters. Normalized plasma parameters foreseen in DEMO have been individually achieved experimentally worldwide, but the integrated performance has not yet been achieved in present devices. Such integrated plasma performance should be exploited by advanced tokamak experiments in satellite machines such as JT-60SA and by an extended operation of ITER.

ii) MHD equilibrium and shaping

Control of plasma equilibrium and shape is essential for high plasma and fusion performance of power core. Although a highly shaped plasma regarding elongation and triangularity is favorable for high β and high density operation, the design parameters on shaping need to be determined in consideration of trade-off relations with system design. For example, considering engineering difficulties of using in-vessel coils in DEMO, intermediate elongation without the coils would be favorable in DEMO, rather than higher elongation with them.

Design Issues on “Plasma Engineering”

i) Plasma control

Variety of actuators and sensors available on DEMO may be limited due to several design constraints. Therefore, it is important to identify the control strategy including controls of shape, profiles, MHD modes and disruption mitigation, and then to examine its control method on Satellite devices and ITER in blindfold way.

ii) Current drive

Current drive (CD) can play an important role in determining the overall power balance of the plant, and NBCD appears to be the best choice in terms of CD efficiency at the moment. But taking account of other aspects like readiness of maintenance and controllability of current profile, further study on alternative CD (e.g. with ECCD) should be encouraged.

iii) Divertor and plasma wall interaction

Physics requirements for divertor is determined by engineering constraints in that the lifetime of the divertor plate is dependent on divertor plasma conditions. The operation temperature of materials used in the divertor plate constrain the allowable heat flux and erosions of the plate provide a temperature constraint of the divertor plasma. Because of material constraint due to severe neutron irradiation and high operation availability, DEMO will face more difficult challenges on divertor technology than ITER.

Design Issues on "Engineering"

i) Blanket

The prime option of Japan is water-cooled pebble bed (WCPB) blanket. In contrast, Helium-Cooled Lithium Lead (HCLL) and Helium-Cooled Pebble Bed (HCPB) blanket are reference concepts in the European breeding blanket programme for DEMO. Since blanket is not included in the integration on ITER, system design focused on blanket should be of importance to define feasible DEMO concepts. RAFM steels (e.g., EUROFER of EU, and F82H of Japan) are considered as the most promising structural materials of blanket. Establishment of the fabrication technology of RAFM in the DEMO relevant scale and the database for standardization for DEMO are critical issues. SiC_f/SiC ceramic composites are considered for advanced blanket concepts although the use of them in DEMO is likely to be restricted to functional material.

ii) Magnet

Progress in magnet technology was reviewed. It was pointed out that the maximum field of TF coils (B_{max}) tended to be reduced with the coil size, and that an advantage of high J_c conductor (such as Nb_3Al and Bi-HTS) in attaining high B_{max} would be lessened for large TF coils.

"System Issues" on DEMO design

i) Maintenance

Maintenance schemes are divided into two categories, 1) "in-vessel maintenance" in which most of replacement and testing of core components are carried out in the vacuum vessel, and 2) "hot cell maintenance" in which most of replacement and testing are done in the hot cell adjacent to the reactor hall. Hot cell maintenance with vertical or horizontal transport is expected to dramatically reduce the maintenance period using spare units because the most time-consuming processes such re-welding and its inspection can be done in the hot cell during the reactor operation. However, we are not confident with what scheme is most feasible and favorable to DEMO. Various conceptual studies on maintenance need to be carried out to assemble technical information for making a decision on the maintenance scheme.

ii) Safety

Based on previous studies, it was stressed that fusion's safety and environmental potentials were real. On the other hand, the assessment result would be dependent on material choice (structural, breeder, coolant, etc.). Therefore, each design option assessments have to demonstrate proper material choices and proper combination of materials toward minimization of radioactive inventories.

ii) Systemic issues

In relation with DEMO, several design tradeoffs were pointed out, including 1) reactor size vs. volt-sec supply, 2) complex tradeoffs regarding blanket, and 3) maintenance vs. TF coil support. Blanket design contains a lot of tradeoffs in various engineering aspects. Key factors for blanket design are 1) TBR, 2) structural strength, 3) β value (related with a conducting shell position) and 4) cooling. The first priority must be given to TBR among these factors. Accordingly, we may need to work out a possible compromise for the other factors to meet the required TBR.

“System Code”

Systems codes in EU and Japan were reviewed. The result of the codes can provide a rough guideline for the selection of favorable design parameters of a fusion reactor. Although the systems codes adopt different algorithms and models, benchmark may be required as a part of joint work.