Plasma Surface Interaction Issues in Magnetic Fusion Chamber

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Acknowledgement: K. Tobita (JAERI), Y. Katoh (Kyoto Univ.)
A. Hasegawa (Tohoku Univ.)

2003 US-J Joint Workshop on
High Heat Flux Components and Plasma Surface for Next Fusion Devices
& High Power Density Devices and Designs

Manresa Hotel, Port Townsent, Washington, USA
July 28 – July 31

Workshop Agenda

Outline

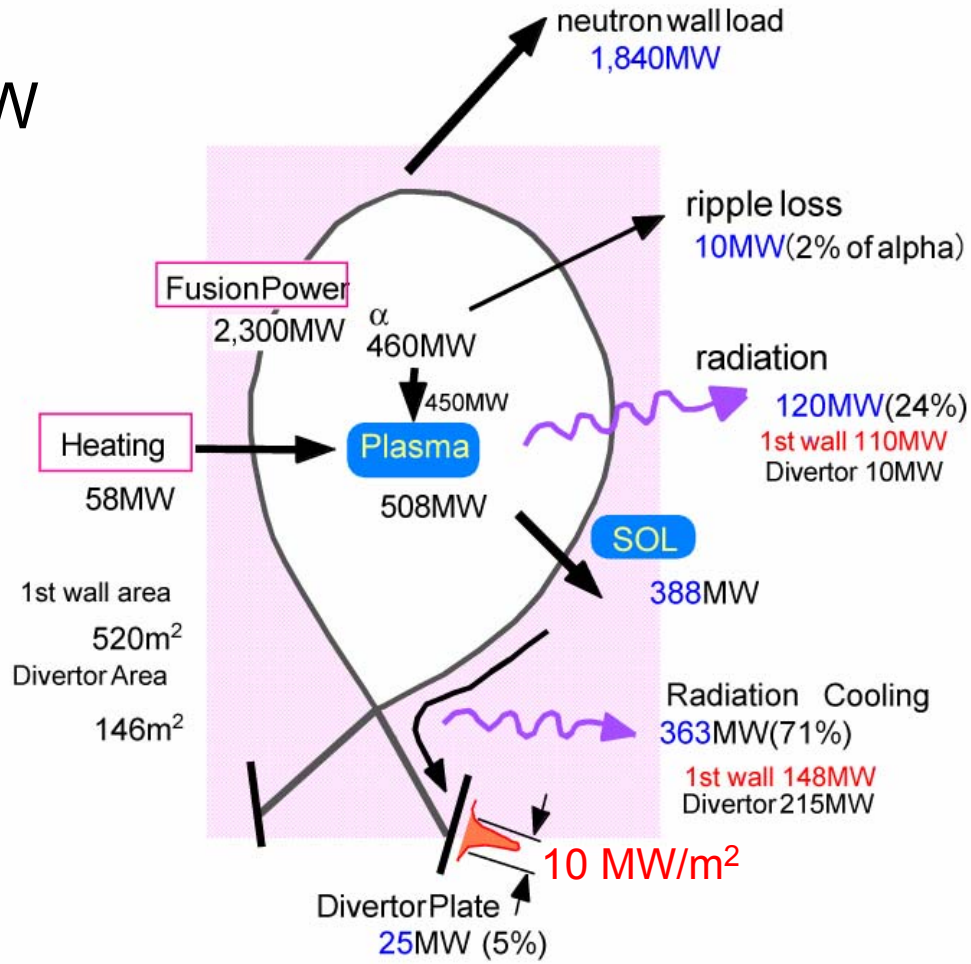
- Reactor Parameters
- Particle Load to FW
  - Feature of CX neutrals to FW (ITER and DEMO)
- Alpha Particle Ripple Loss
  - Reduction of ripple loss by Ferromagnetic Inserts
  - Simulation Exp. for Ripple Loss Alpha Effects
- Blanket FW design
- Sputtering Erosion of FW
  - Sputtering of Low Activation Materials
- Plasma Stabilization by Blankets
  - Effect of Ferromagnetic Material on Plasma Stabilization (RWM)
- Reflection of Synchrotron Radiation on FW
- Summary of FW material characteristics

Power Flow in DEMO

Most of the power flow to FW could be done by radiation.

Particle fluxes may not contribute heat flux, but they play important roles in erosion and surface modification.

Strong radiation cooling (about 95% of heating power (α + ext. heating) is needed to reduce divertor heat flux.



JAERI DEMO Design

Konishi et al. Fusion Eng. Des. 63-64 (2002) 11-17.

Reactor Design Parameters

	ITER-FEAT Exp. reactor	SSTR Demo	DEMO (JAERI)	SEAFP Comm	A-SSTR2 Comm	CREST Comm	ARIES-RS Comm
R (m)	6.2	7	5.8	9.5	6.2	5.4	5.52
a (m)	2	1.75	1.45	2.09	1.5	1.59	1.38
I_p (MA)	15	12	12	10.4	12	12	11.3
B_t (T) on-axis	5.3	9	9.5	7.8	11	5.6	8.0
B_t (T) max	12	16.5	20	12.8	23	12.5	15.8
T_e (keV)	9.8	17	18	10	19	15.4	18.7
$\langle n_e \rangle$ (10^{20} m^{-3})	1.04	1.4	1.94	1.62	2.7	2.1	2.11
β_N (%)	1.9	3.5	4	3.5	4.2	5.5	4.8
P_{fus} (GW)	0.5	3	2.3	3.0	4.0	2.97	2.17
N_{load} (MW/m^2)	0.57 (ave) 0.78(max)	3 (max)	3 (ave) 5 (max)	2.1 (ave)	6 (ave) 8 (max)	4.5 (ave) 6.5 (max)	4.0 (ave) 5.7 (max)
Heat Load [FW] (MW/m^2)	0.25 (ave) 0.5 (max)	1 (max)	0.5 (ave) 1.0 (max)	0.4 (ave)	1.0 (ave)	1.2 (ave)	0.4 (max)
Heat Load [DV] (MW/m^2)	10	7	10		4	10	6

ITER-FEAT -> DEMO -> Commercial

fusion power and neutron load : increase by a factor of 4 to 8.

machine size : similar or even smaller (for economical competitiveness)

→high β_N (4.2 - 5.5) compared with ITER-FEAT (1.9)

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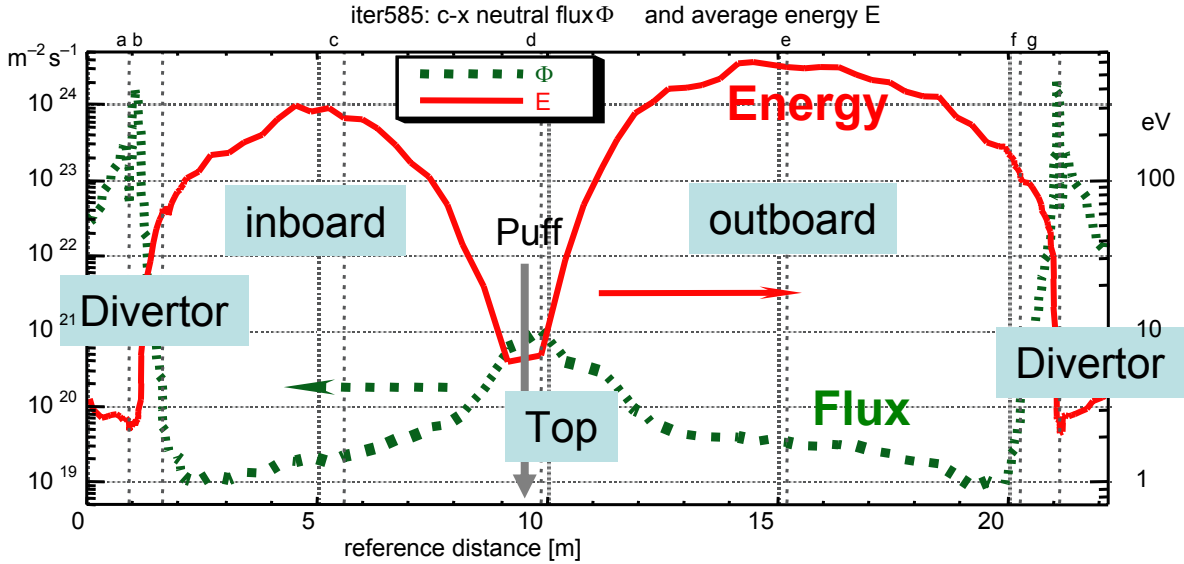
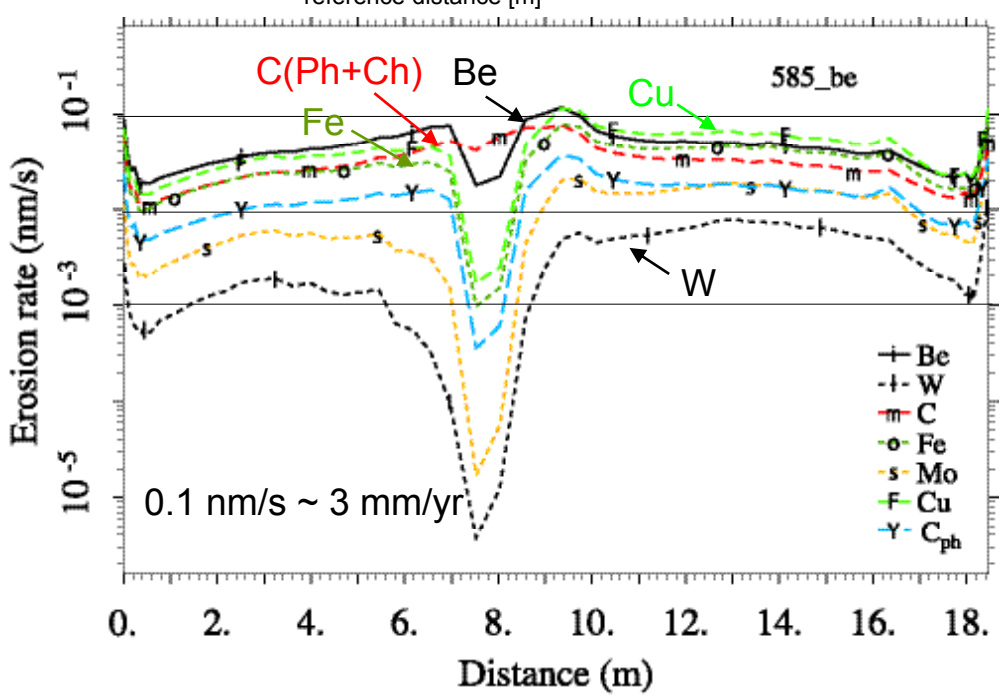
CX neutral and Erosion Rate (ITER)





Calculated by B2-EIRENE code.

Maximum erosion takes place near gas puff position.

Max. erosion rate of Be is 3.5 mm/burn-yr.

Erosion rate of W is about one order of magnitude lower.

R. Behrisch et al., J. Nucl. Mater. 313-316 (2003) 388.

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Extrapolation from ITER to DEMO

Separatrix density n_s scaling proposed by A. Kukushkin.

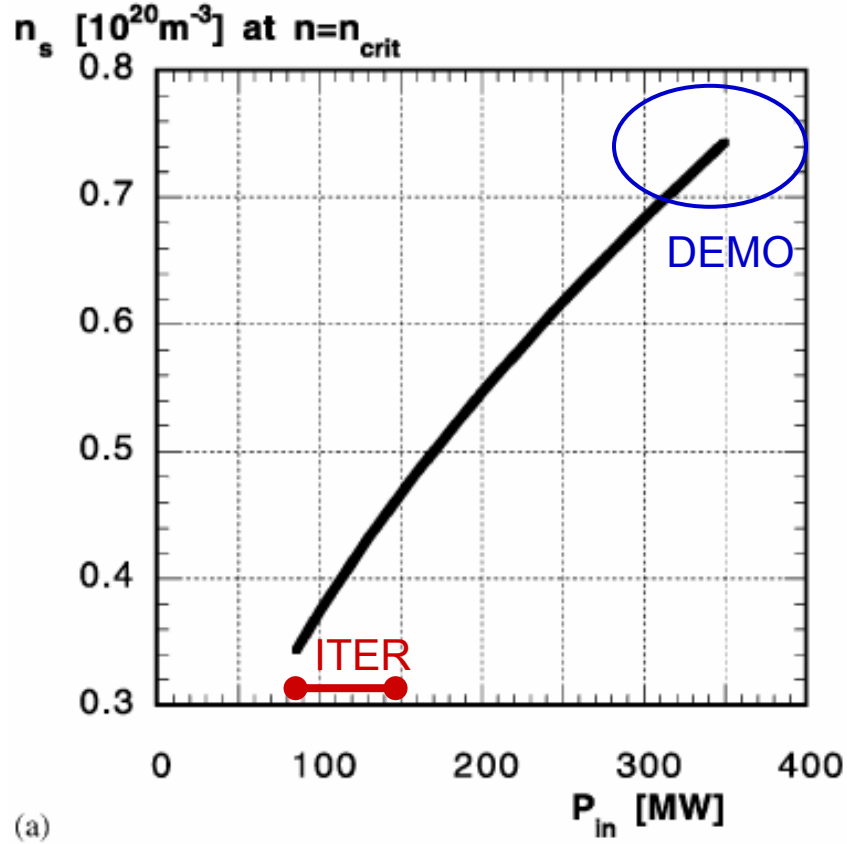
$$n_s \propto f_{fuel} P_{in}^{0.55}$$

$$f_{fuel} = \frac{(\Gamma_{puff} + 1.18(\Gamma_{DT} - \Gamma_{puff}))}{\Gamma_{DT}}$$

f_{fuel} : fuelling eff. of SOL

P_{in} : input power to SOL
$(P_a + P_{aux} - P_{rad})$

For ITER size DEMO reactor, separatrix density would increase to (0.7-0.8) x 10^{20} m^{-3}.

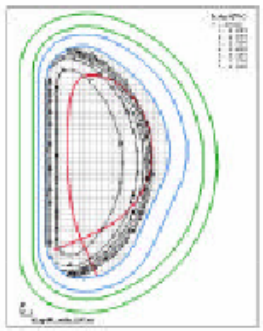
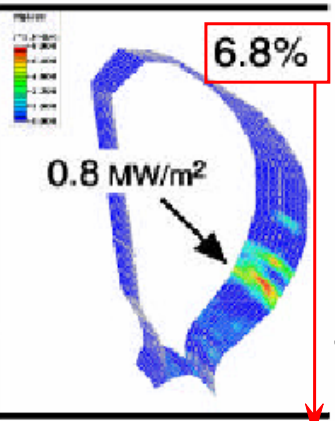
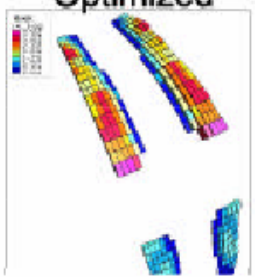
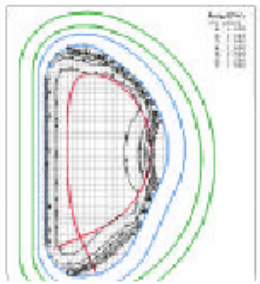
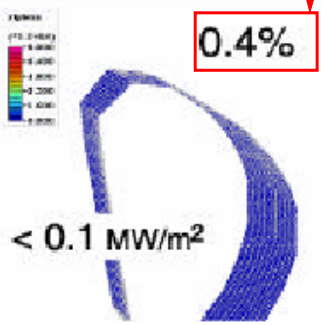


Separatrix density vs. input power to SOL

A. Kukushkin et al., Fusion Eng. Des. 65 (2003) 355

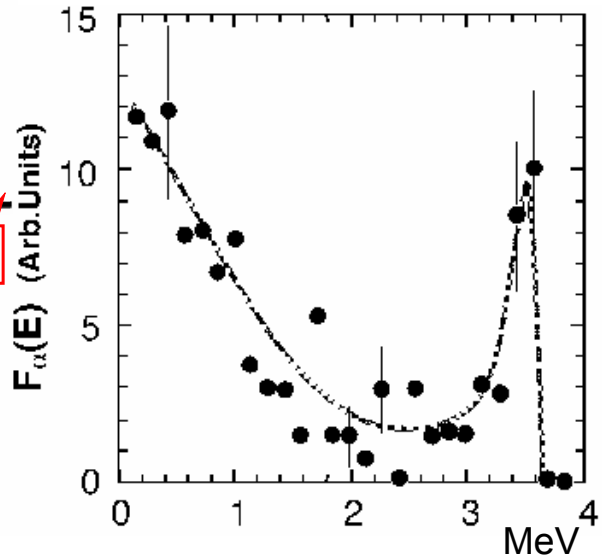
Alpha Particle Ripple Loss Reduction

ITER- FEAT ripple loss reduction
by Ferromagnetic Inserts
By Tobita (JAERI)

Ferromag. Inserts	TF Ripple	Ripple Loss
Large Toroidal Field Ripple None		
Optimized 		

Alpha particle ripple loss decreases drastically by ferromagnetic inserts.

For DEMO reactors, ripple loss can be less than 0.4 MW/m^2, which corresponds to about 10^{18}/m^{2}s.



Energy Distribution of Loss α

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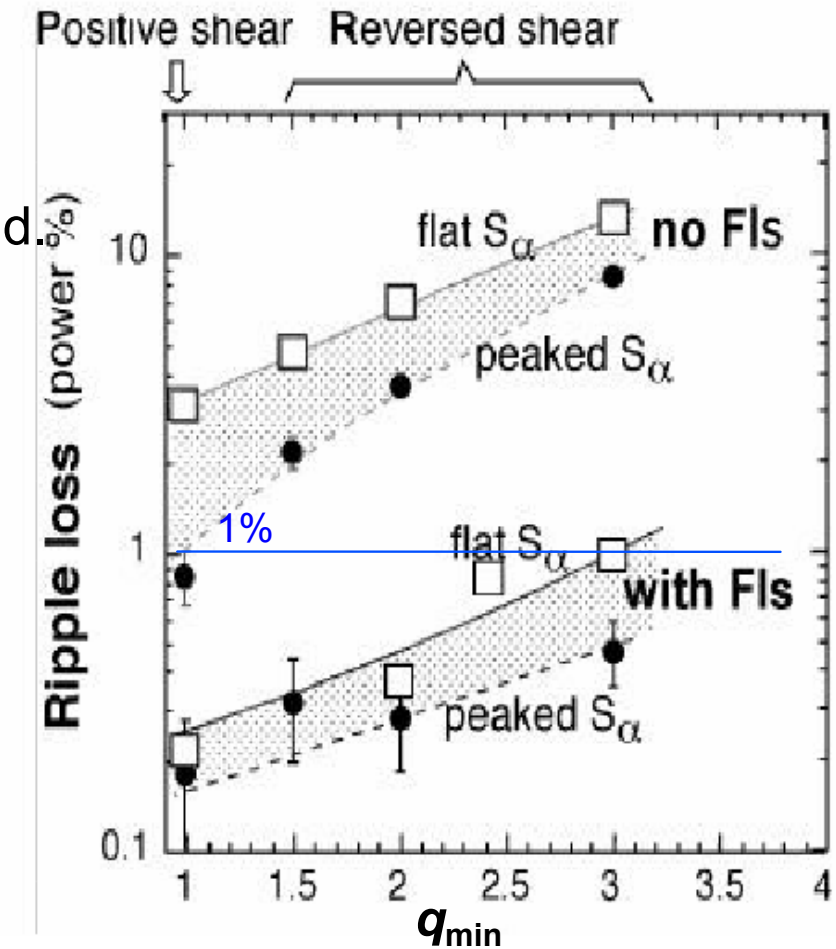
Ripple Loss for high confinement mode

In the case of reversed shear mode and current hole mode, ripple loss becomes large due to low poloidal field.

When FI are installed, ripple loss can be reduced less than 1% even with Flat S_α and high q_{min} RS case.

Ripple loss α-particles do not cause serious sputtering erosion, but can cause significant effects on material degradation (blistering, embrittlement, etc.).
For DEMO, total fluence during blanket life would be ~10^{26} m^{-2}.

Critical fluence of blister appearance by mono-energetic ion beam →~10^{22} m^{-2}



Ripple loss power vs. minimum q for ITER-FEAT configuration. S_α : alpha particle production profile

K. Tobita et al., Plasma Phys. Control. Fusion 45 (2003) 133.

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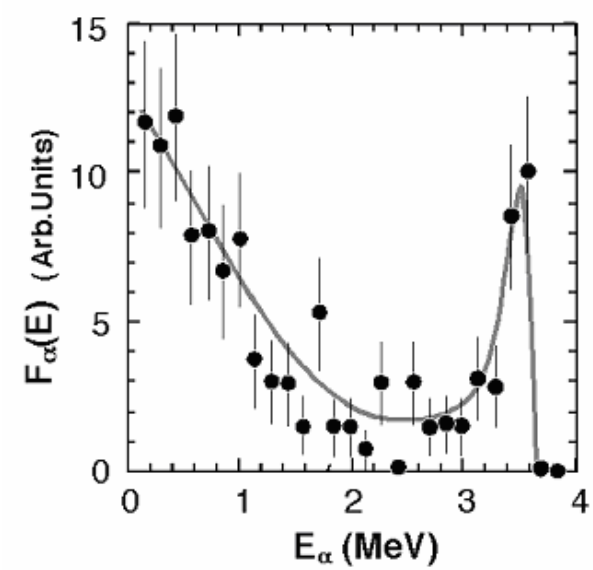
Blistering Behavior of F82H and W for Plasma Facing Components for DEMO Reactor Application

A.Hasegawa, K.Tanaka, K.Abe

Dept. of Quantum Science and Energy Engineering, Tohoku University, JAPAN

Objective

Study on blistering of first wall material caused by ripple loss particles in DEMO reactor.



Predicted energy distribution of ripple loss alpha particles. K.Tobita et al.

Irradiation Conditions

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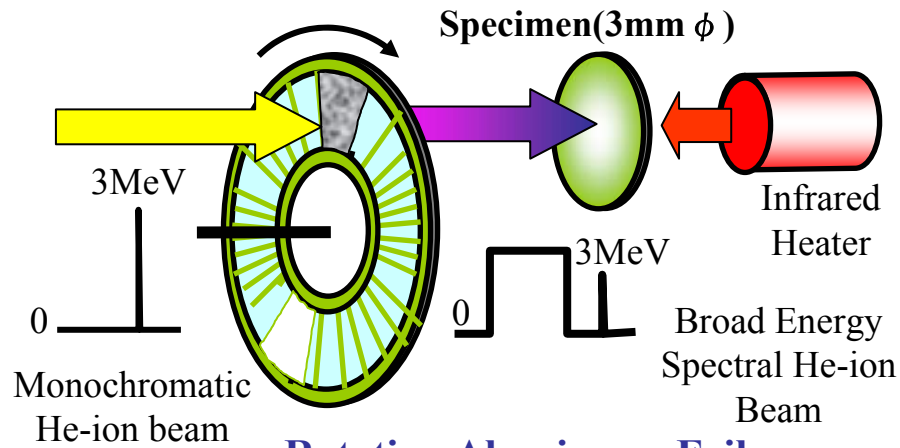
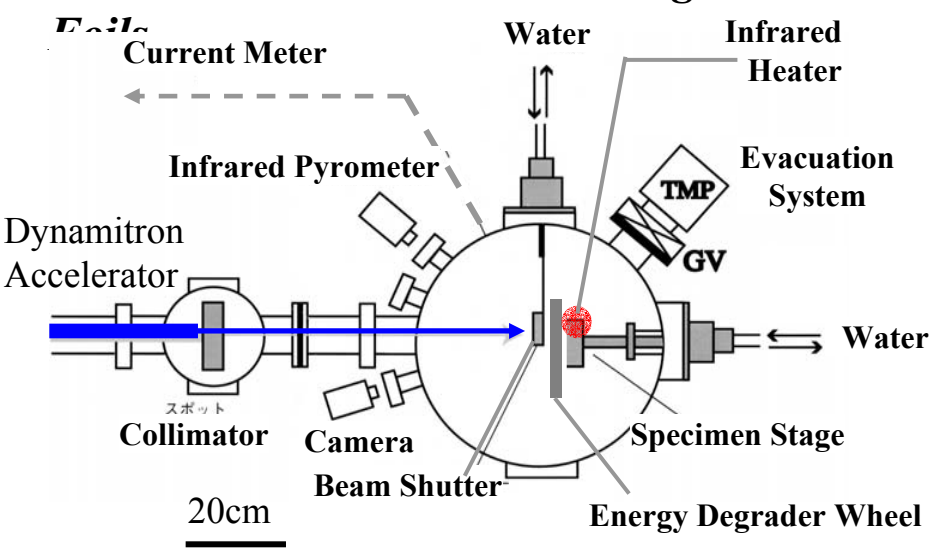
- *Particle : $^{4}He^{+}$*
- *Accelerated Energy : 3.0MeV*
- *Implanted Temperature : RT & 550℃*
- *Fluence : 10^{22}～10^{23} He/m^2*
- *Energy Degrader :*

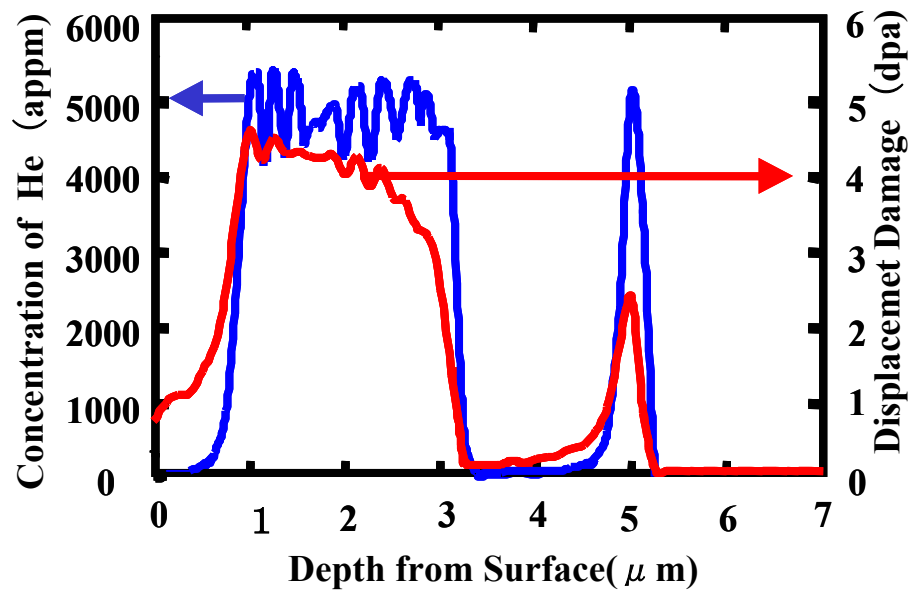


Rotating Aluminum- Foils Degrader

Rotating Aluminum-



Schematic View of Irradiation System



Calculated Depth Distribution in Fe by TRIM code

Experimental Conditions

●**Examined Material :**

（1） Reduced Activation Ferritic Steel : F82H IEA Heat

（2） Powder Metallurgical Tungsten

Recrystallization at 1300℃ for 1hr

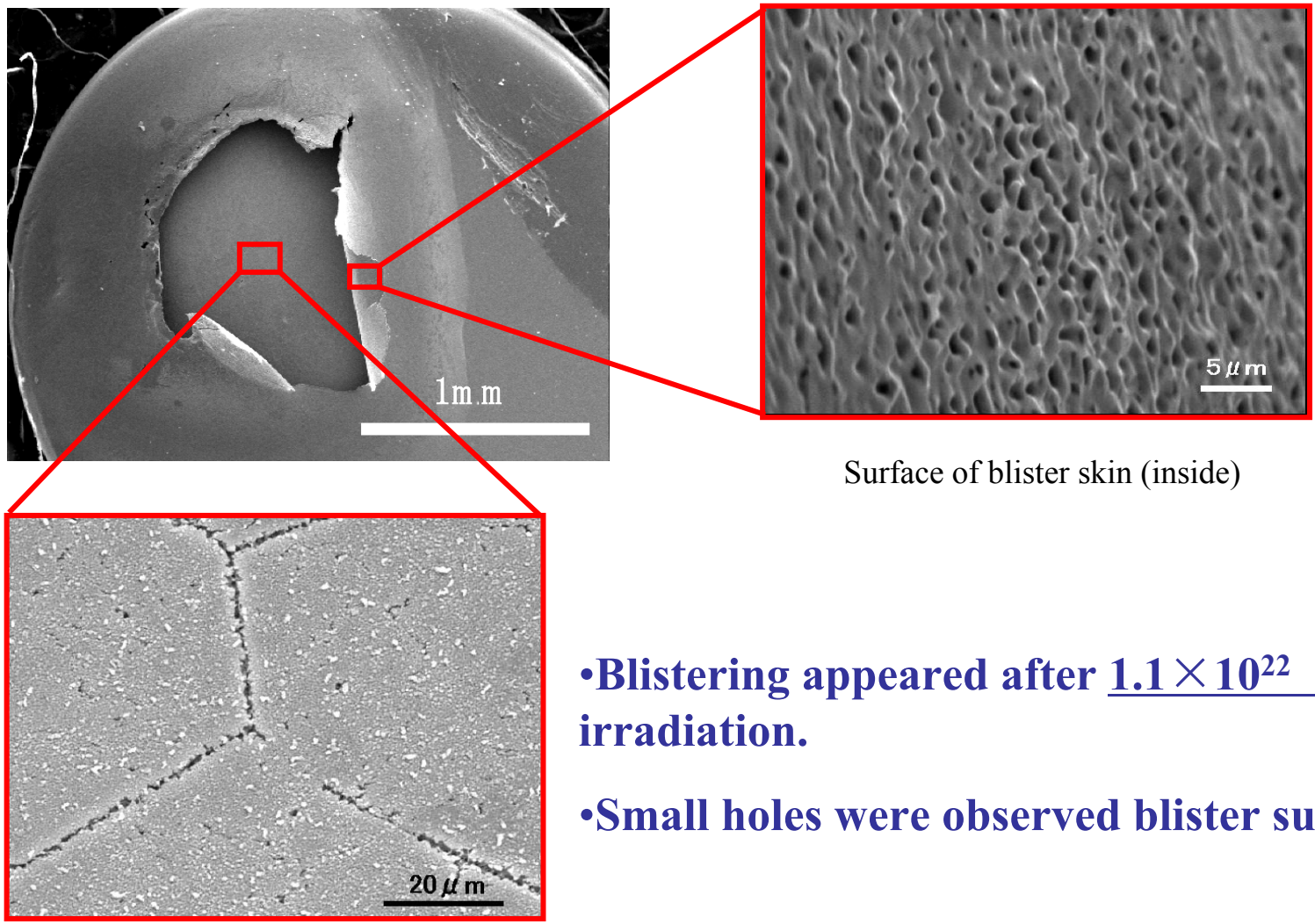
Shape of Specimen : 3mm ϕ x 0.2mmt Disk

Before irradiation, specimen surface was polished mechanically and electro-polished to remove surface worked layer.

●**Surface observation : Optical microscope and SEM**

Results : F82H

- **Monochromatic 3MeV He-ion beam irradiation at 550℃**



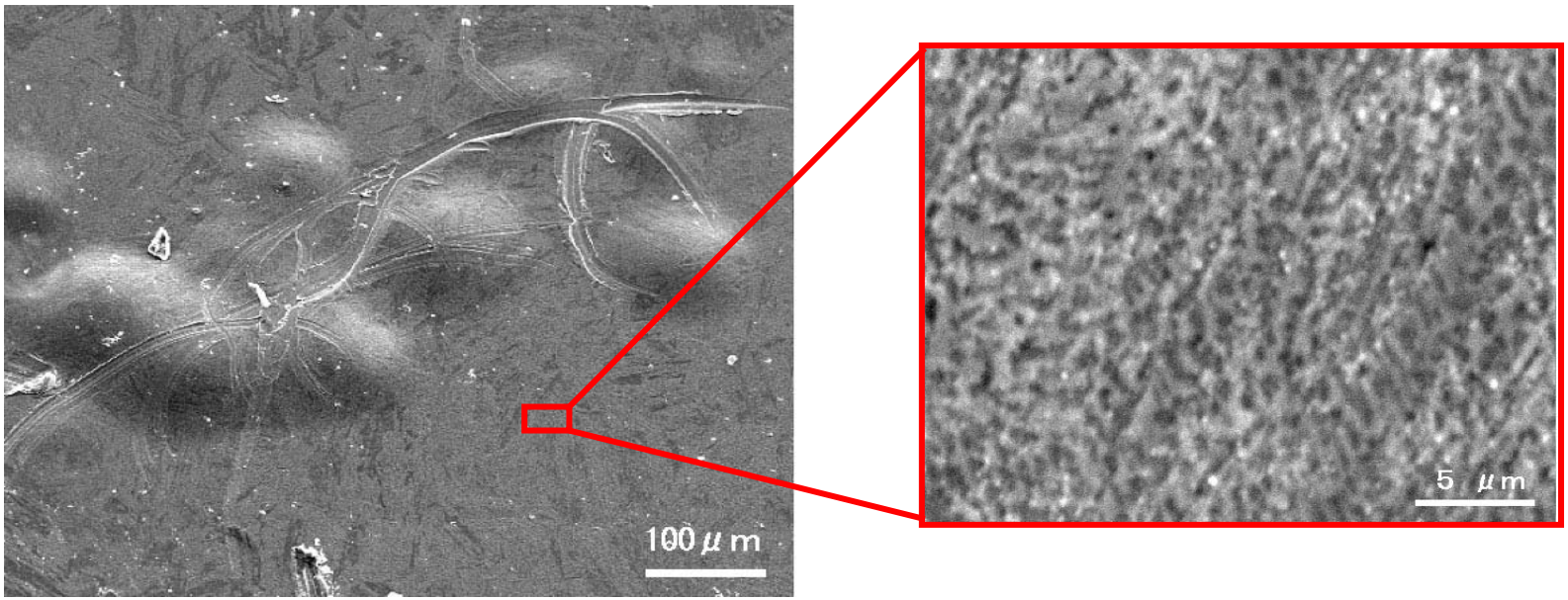
Surface of blister skin (inside)

Bottom of blister

- **Blistering appeared after 1.1×10^{22} He/m^2 irradiation.**
- **Small holes were observed blister surface.**

Results : F82H

- **Energy degraded beam irradiation at 550℃**



- **Small blister appeared after 4.5x10^{22} He/m^2.**
- **Exfoliation was not observed in this specimen.**

Results : W

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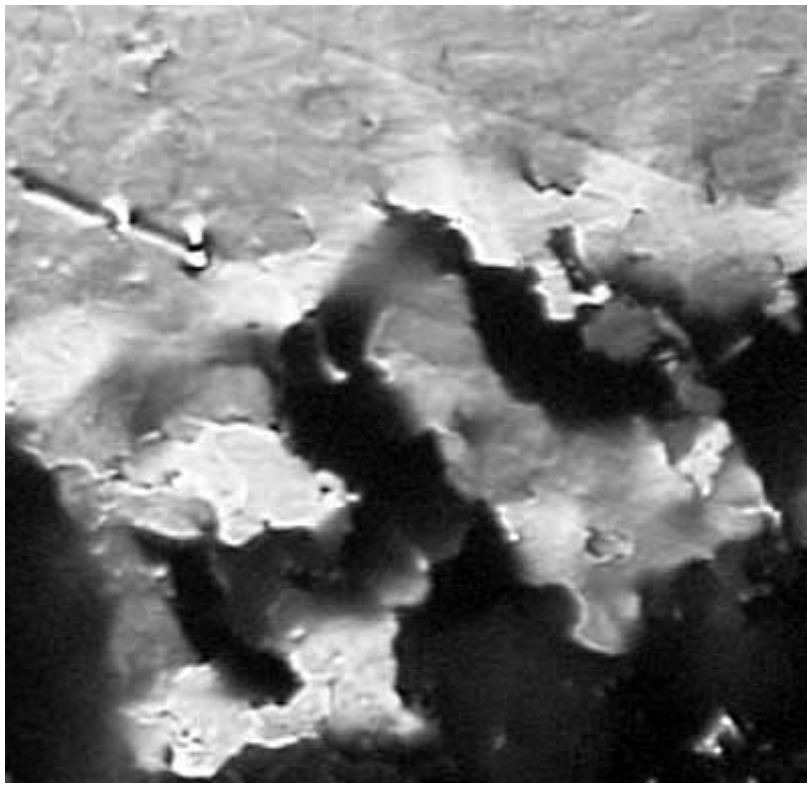
- Monochromatic 3MeV He-ion beam irradiation at 550℃

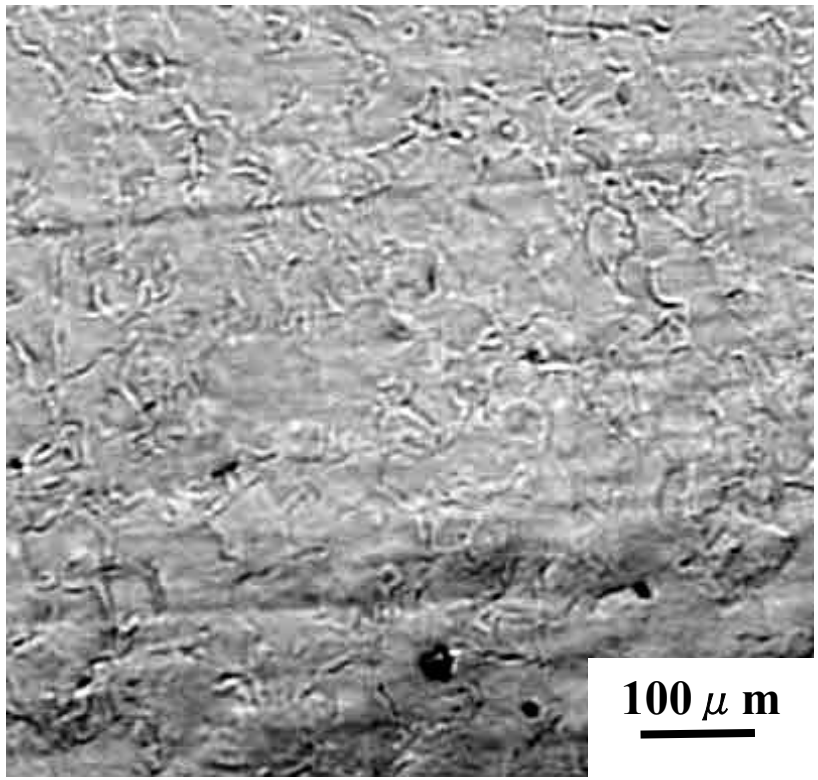
Fluence : 2.6×10^{22} He/m^2



- Exfoliation was observed.

- Energy degraded beam irradiation at 550℃

Fluence : 1.6×10^{23} He/m^2



- Blistering was not observed up to 1.6×10^{23} He/m^2 irradiation.

Blanket first wall design

	He Cooled Pebble Bed	Water Cooled Pebble Bed	Water Cooled Pb-17Li	Self-Cooled Flibe	Self-Cooled Li	He Cooled Li	Self-Cooled Pb17Li (TAURO)	He Cooled Pebble Bed (advanced)
Device	Tokamak DEMO	Tokamak DEMO	Tokamak DEMO	Helical FFHR-2	Tokamak	Tokamak LAR design	Tokamak SEAFP	Tokamak
Structural Material	ODS steel RAF	F82H,ODS RAF	EUROFER RAF	V-Alloy	V-Alloy	V-Alloy (W coating)	SiC$_f$/SiC	SiC$_f$/SiC
Fusion Power (GW)	3.6	2.3	3.6	1.0		5.3	3.0	4.5
Neutron Load (MW/m^2)	4.4(max)	5.0(max)	6.6(max)	1.7 (ave)	10 (max)	11 (max)	2	3.5 (max)
Surface Heat Load (MW/m^2)	0.8 (max)	1.0(max)	1.2(max)	0.1 (ave)	2(max)	2.73(max)	0.5	0.6(max)
FW thickness (mm)	5	3	4	5	4	1.5+1(W), tubing	3	3
FW temperature (¡C)	630	~ 600	590	750	754	697	<1300	913
Coolant Pressure	He 8 MPa	H_2O 25MPa	H_2O 15.5MPa	Flibe 0.6 MPa	Liq. Li 0.5 MPa	He 15 MPa	Pb-17Li 1.5 MPa	He 8 MPa
Tritium Breeder	Li Ceramics	Li_2TiO_3	Pb-17Li	Flibe	Li	Li	Pb-17Li	Li_4SiO_4
Neutron Multiplier	Be	Be ($Be_{12}Ti$)	none	Be	Be	none	none	Be
Reference	L. Giancarli et al. (2000)	T. Konishi et al. (2002)	L. Giancarli et al. (2000)	A. Sagara et al. (2001)	Y. Gohar et al. (2001)	C. Wong et al. (2000)	Y. Golfier et al. (2000)	L. Bocaccini et al. (2000)

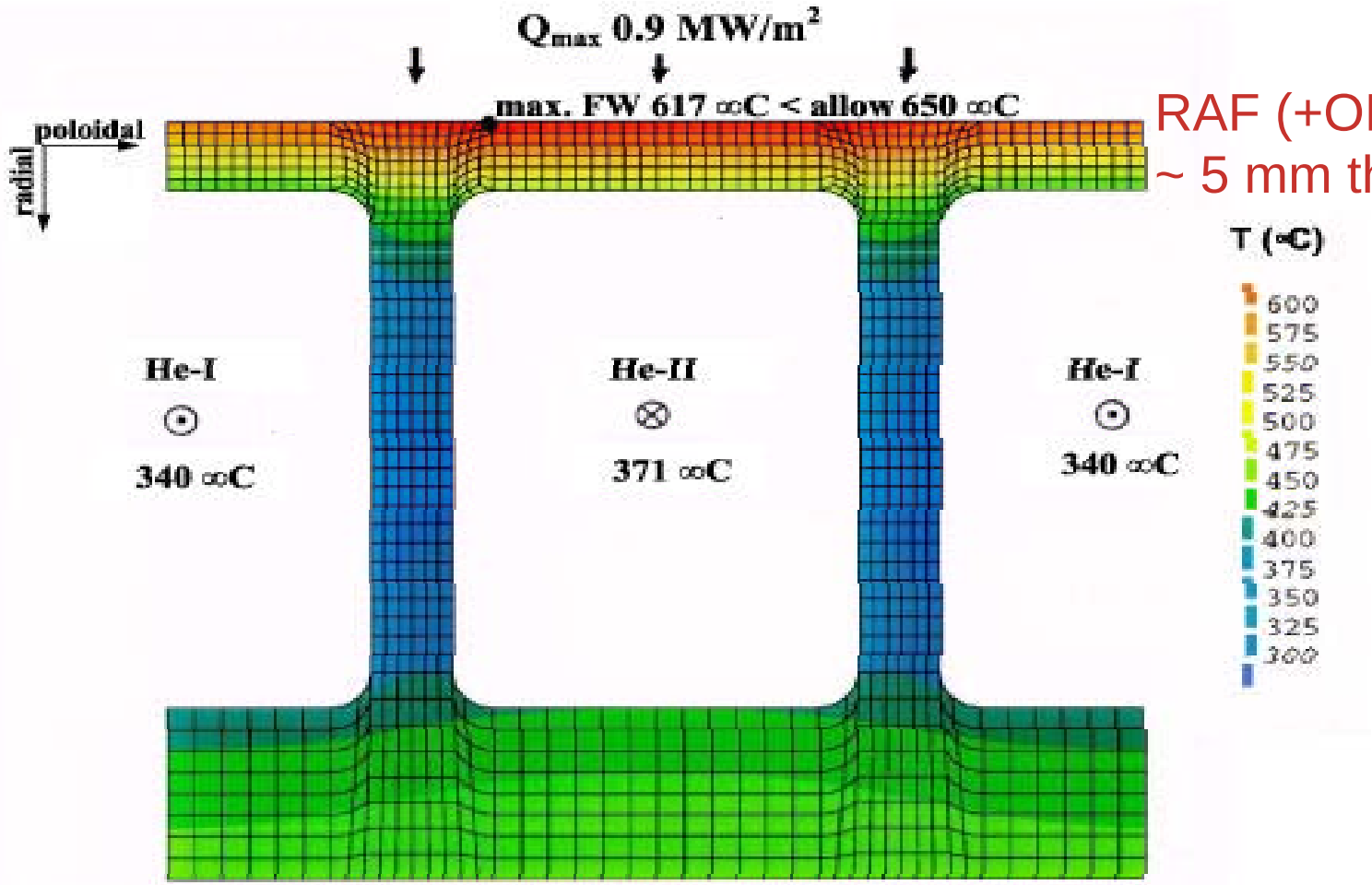
For RAF blanket system, maximum surface heat load is ~1 MW/m^2.

For V-alloy blanket system, maximum heat load is ~ 2MW/m^2 due to its good surface heat capability.

For SiC/SiC blanket system, operating temperature is ~1,000°C, but its acceptable heat flux is ~0.5 MW/m^2 (conservative estimation).

FW thickness is only 3-5 mm, very vulnerable to erosion.

Example of Temp. Distribution of FW



RAF (+ODS)
~ 5 mm thickness

P. Norajitra et al., Fusion Engineering and Design 58-59 (2001) 629-634

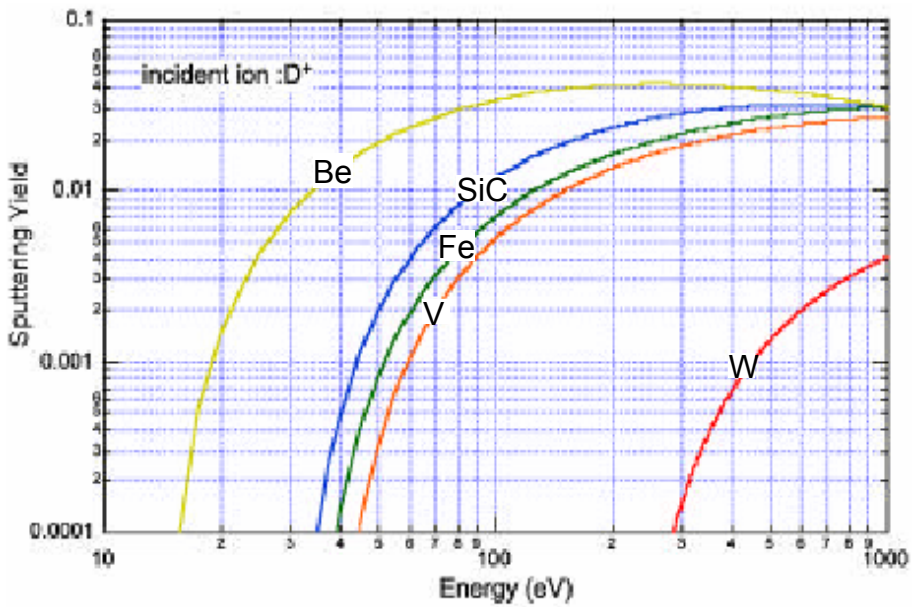
(Fig. 2: Temperature distribution in the first wall at the torus centre for the reference case)

Sputtering Erosion (Low Activation Mater.)

Erosion yield of LAM are similar (RAF, V-Alloy, SiC/SiC). Erosion rates are ~2 mm/burn-yr (ITER).

Considering FW thickness (3-5 mm), this erosion rate is unacceptable.

Low Z impurities and inert gas for divertor cooling could enhance sputtering, especially for W.



Physical sputtering yield of Be, W, SiC, Fe (RAF), and V (V-alloy)

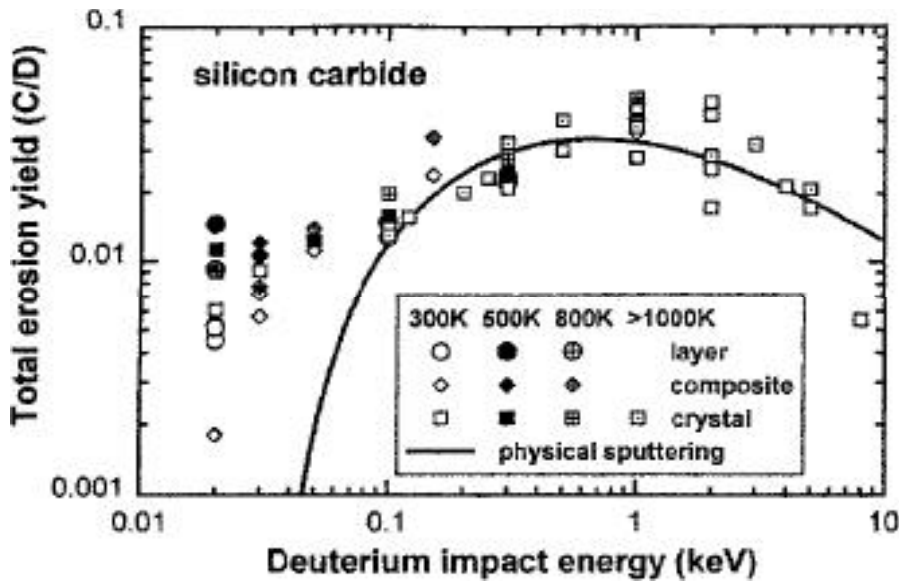
Non-volatile impurities such as C tend to complicate erosion process due to its accumulation and diffusion.

Runaway self-sputtering for W must be avoided. It could occur in the case of only 35 eV T_e in the edge plasma.

Chemical Erosion of SiC

Chemical sputtering occurs for SiC in the energy of less than 100 eV.

Due to this chemical erosion, the erosion of SiC/SiC composite would not decrease significantly with the decrease of impact energy, less than 100 eV.



M. Balden et al., J. Nucl. Mater. 290-293 (2001) 47.

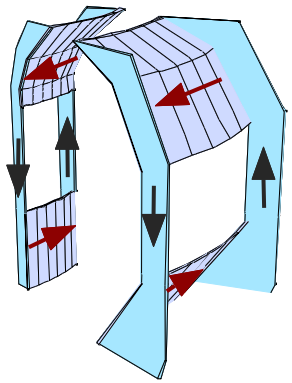
20 eV D: yield is ~ 0.5 % at 300 K
~ 1.5 % at 500 K
~ 1.0 % at 800 K
(for β-SiC)

Higher temp. data (>1000 K) is needed for correct evaluation.

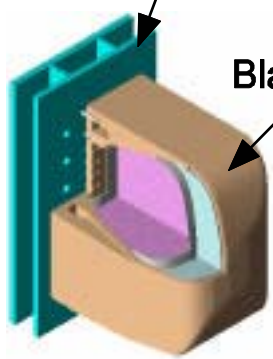
Plasma Stabilization

Conducting materials for plasma stabilization are needed in blanket system. One of the reason is that β_N for fusion reactors will exceeds free boundary limit (3.5). Suppression of positional instability, ideal kink mode, resistive wall mode (RWM) is needed.

Resistive wall mode (RWM) could be a serious problem for steady-state plasma. RWM becomes unstable with FW close to the plasma with growth time comparable to field penetration time.



Saddle Shape Coil



For stabilization of RWM, plasma rotation and/or stabilization coils with feedback control system are needed.

Resistive Wall Mode (RWM)

Resistive wall mode (RWM) could be a serious problem for steady-state plasma. RWM becomes unstable with FW close to the plasma with growth time comparable to field penetration time.

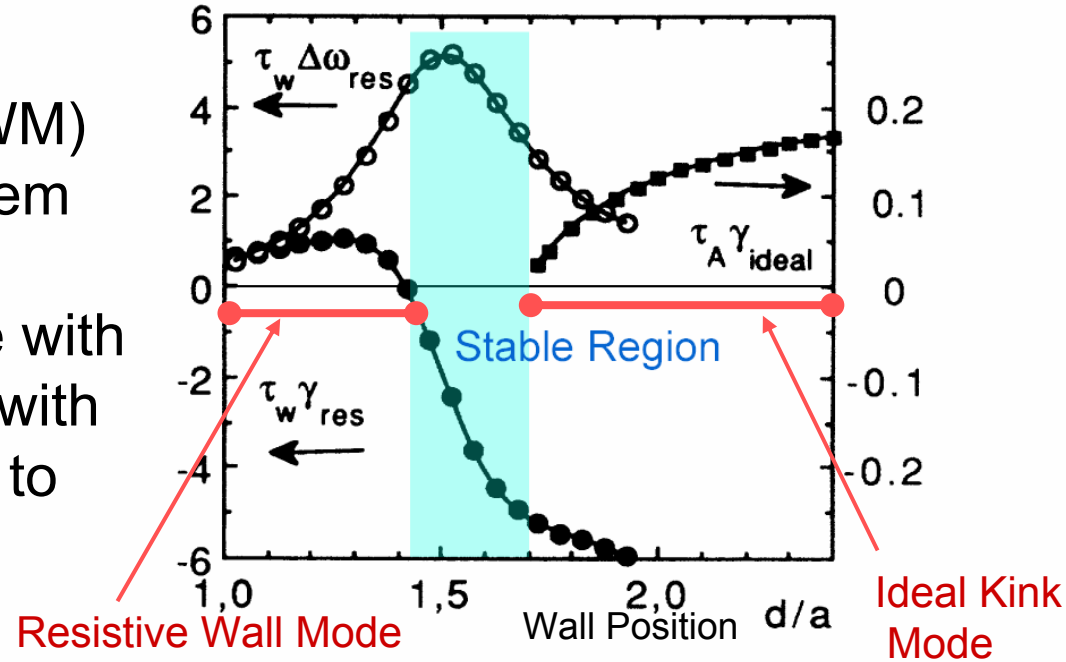
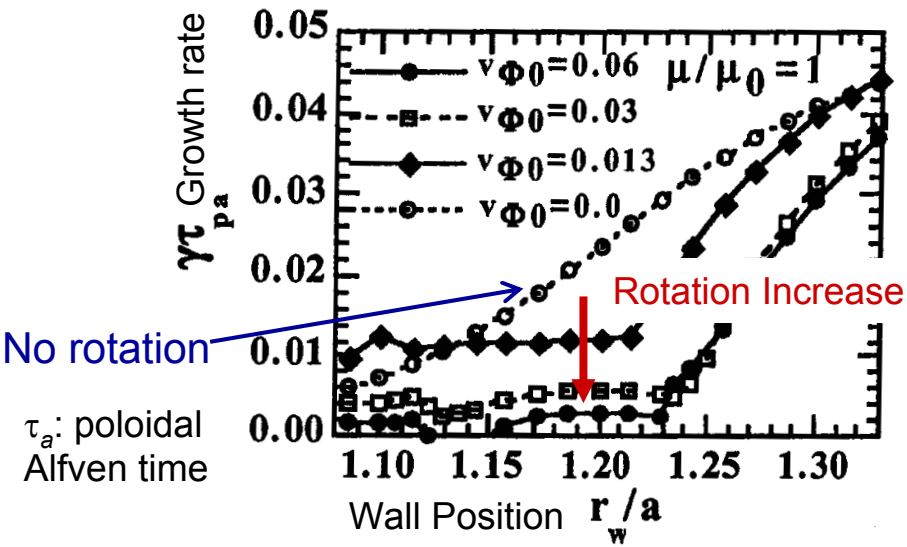


FIG. 1. Growth rate γ_{res} and slip frequency $\Delta\omega_{res} = \omega_{rot} - \omega_{res}$ of resistive wall mode and growth rate of plasma mode γ_{ideal} versus wall radius for the $n=1$ mode with pressure about 30% above the free-boundary limit and $\omega_{rot}/\omega_A = 0.06$.

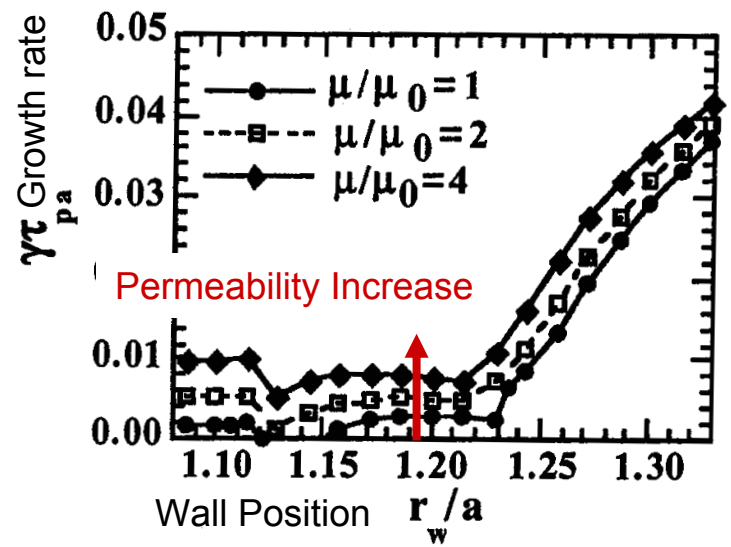
A.Bondeson et al., Phys. Rev. Lett. 72 (1994) 2709.

For stabilization, plasma rotation and/or stabilization coils with feedback control are needed.

Effects of Ferromagnetism of Wall on RWM



Growth Rate vs. Wall Position
(Relative permeability is 1)



Growth Rate vs. Wall Position
(Relative rotation speed is 0.06 V_A)

G. Kurita et al., in proceedings of 19th IAEA Fusion Energy Conf (2002) IAEA-CN-94/FT/P1-06

As relative permeability (μ) of the wall material increases, growth rate of RWM increases.

More effective stabilization could be needed for the FW with higher permeability such as RAF.

Reflection of Synchrotron Radiation (SR)

Synchrotron radiation loss plays an important role in the power balance.

$$P_{syn} \sim n_e^{1/2} T_e^{5/2} B_T^{5/2} (1-R_{ref})^{1/2}$$

-> Reflectivity of the first wall R_{ref} affects a necessary condition for energy confinement time.

For the DEMO plant by JAERI,

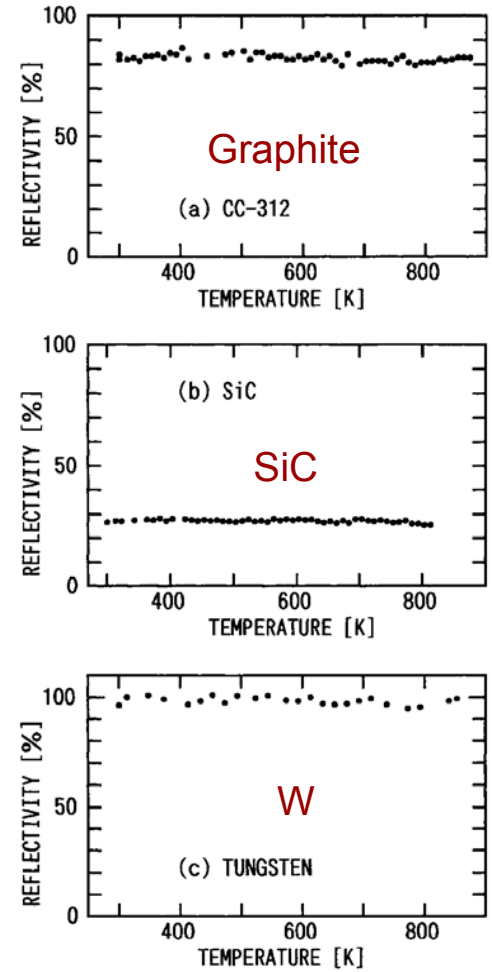
H_H = 1.8 needed for R_{ref} = 0 (No reflection)

H_H = 1.5 needed for R_{ref} = 1 (Complete reflection)

Metallic FW would be preferable for effective use of SR.

However, ion irradiation tends to roughen the surface, which reduces reflectivity of SR.

This reflectivity degradation could be recovered by metal coating.



Reflectivity (443μm)

Takeda et al., JJAP 34 (1995) L849.

Low Activation Mater. as FW

- RAF (Reduced Activation Ferritic Steel)
  - Sputtering Erosion → not low (energy reduction <100eV could reduce erosion)
  - Surface heat capability →good (~1MW/m^2)
  - Plasma stability →Possibility to lower achievable beta due to RWM
- Vanadium Alloy (ex. V-4Cr4Ti)
  - Sputtering Erosion → not low (energy reduction <100eV could reduce erosion)
  - Surface heat capability →very good (~2MW/m^2)
  - Chemical reactivity →Oxygen deteriorate V-alloy even at ~300ppm.
    →Surface coating could be needed.
- SiC_f/SiC composite
  - Sputtering Erosion → not low (energy reduction <100eV may not reduce erosion due to chemical erosion)
  - Surface heat capability→not good (~0.5MW/m^2) (CVI method)
    Newly-developed materials could improve this.
  - Plasma Stability →Necessity for stabilization conductor (back plate)

High Z material (W) as FW

- Sputtering Erosion → Acceptable
  - Impurity effects need to be care.
  - Runaway sputtering must be avoided.
- Surface Heat Capability → very Good (2-4 MW/m^2)
- Material Degradation→Serious Concern
  - Blistering by D/T and He (thermal ion and α-particle ripple loss)
    - Enhancement of effective erosion yield by flaking and vaporization
  - Embrittlement
    - Neutron irradiation embrittlement
    - H/He embrittlement

Comments on Liquid Metal FW

- Stabilization of RWM
  - Flowing liquid metal FW could be effective for RWM suppression, because it acts as semi-superconducting wall.

    τ_{wall} ~ 130 ms (0.1 m thick Liq. Li at 300ºC)

- Reflection of Synchrotron Radiation
  - Liquid metal FW could be advantageous for lowering requirement for confinement time if the surface is smooth enough to reflect SR.

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