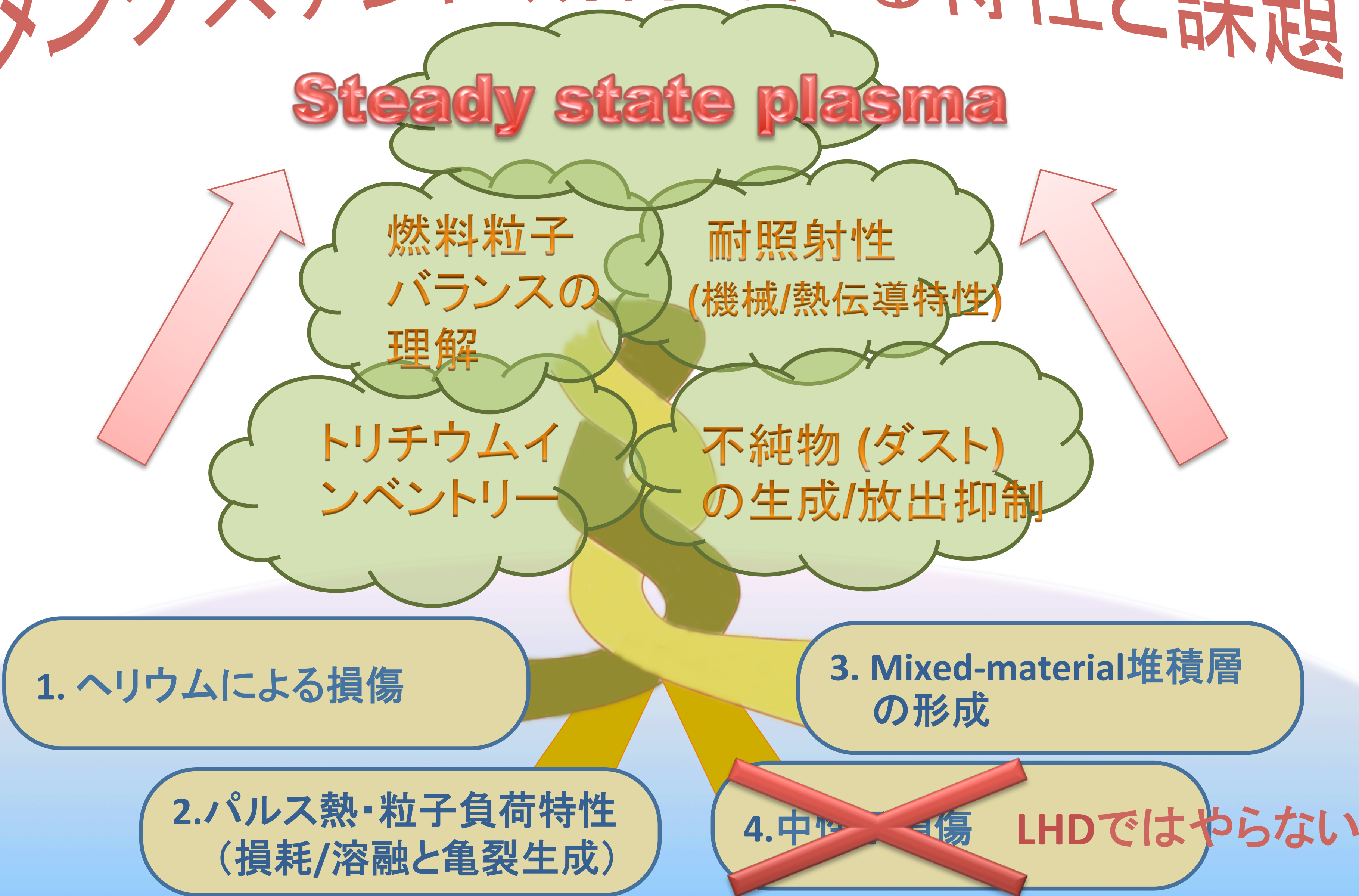
平成24 年度 ダイバータおよびPWI合同研究会 2012, 7/23-24 於:筑波大学

LHDにおけるタングステン研究の進展

核融合科学研究所

時 谷 政 行

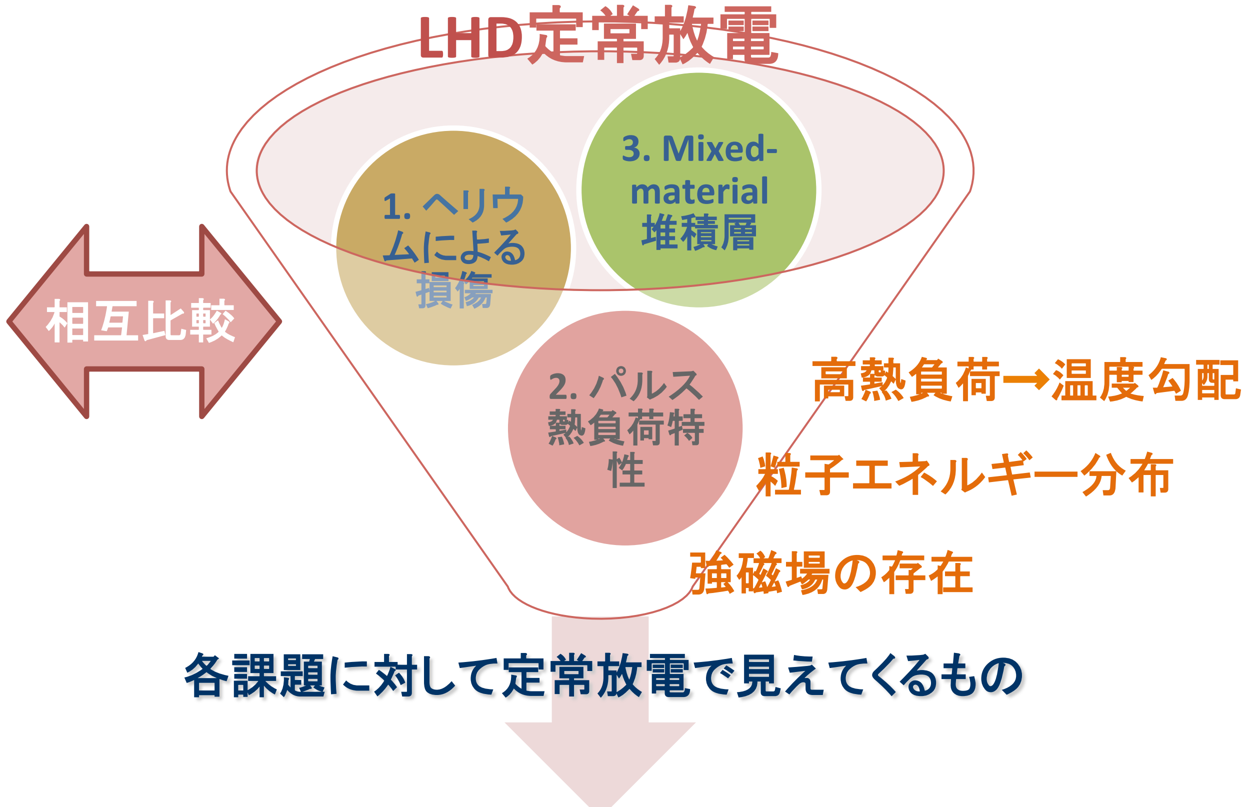
タングステンに期待される特性と課題



重要な4つの課題

LHDでは定常放電の特徴を生かした研究を

- GAMMA-10
- NAGDIS
- JUDITH
- MAGNUM-PSI
- Pilot-PSI
- QSPA
- JEBIS
- ACT



各課題に対して定常放電で見えてくるもの

1.ヘリウムによる損傷

- W-fuzzの生成
- アーキングの物理
- Heと水素同位体の定常的な同時照射効果の理解
- ダスト生成への影響

2.パルス熱・粒子負荷特性

- 溶融・損耗物体の移動機構
- 構造変化

3. Mixed-material堆積層

- 堆積層の成長と剥離/放出の時間スケール
- 堆積層による壁排気特性の時間スケール
- ダスト生成への影響

これまでのLHDのW研究で注力されてきたキーワード:

1. ヘリウムによる損傷

2.パルス熱･粒子負荷特性

3. Mixed-material堆積層

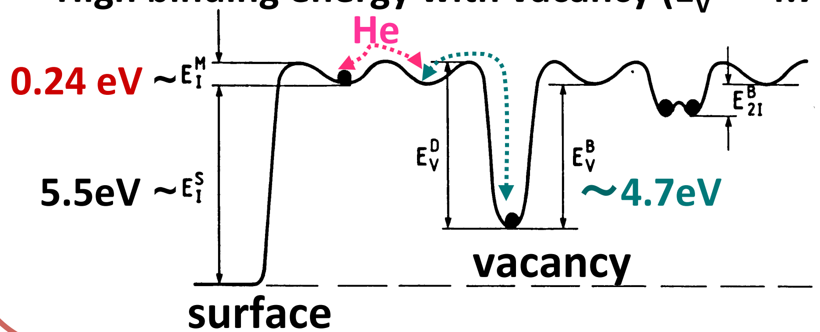
今日はこの部分を
短めに紹介

Fundamental studies of He irradiation

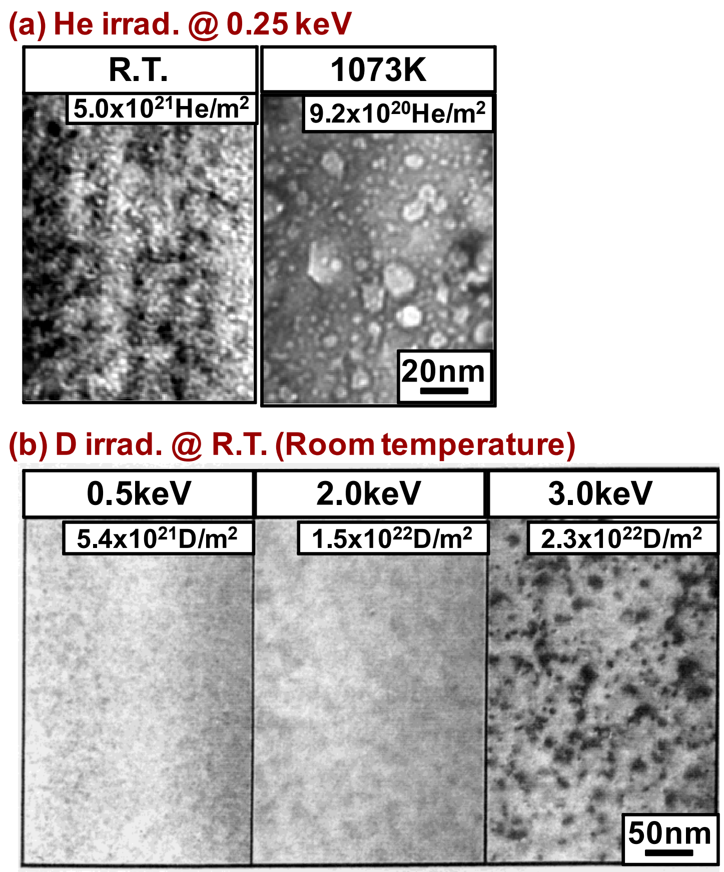
He has a strong interaction with lattice defects such as vacancies or dislocation loops.

Property of He in metal (e.g. W)

- High mobility (E_I^M~0.24eV)
- Low solubility (E_{2I}^B>0)
- High binding energy with vacancy (E_V^B~ 4.7eV)



So injected helium in metals are difficult to desorbs from the matrix.



TEM images of W irradiated with (a) 0.25keV-He^+ in different temperature and (b) various energy of D^+ in room temperature.

Special feature of the He irradiation effects to metals would be lead to change of the surface properties of the plasma facing component.

(e.g. H retention or Degradation of the material)

Damages by He on the LHD W divertor

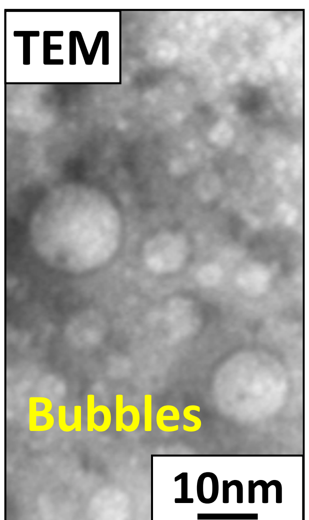
Some kinds of tungsten experiments exposed to the divertor plasma has been conducted.

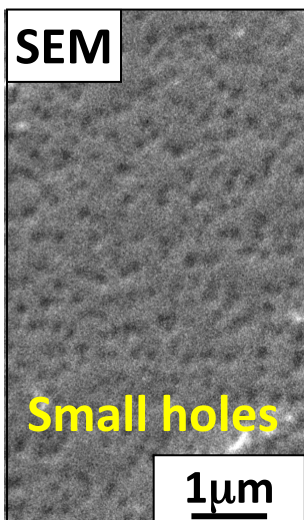
Typical parameters of divertor plasma

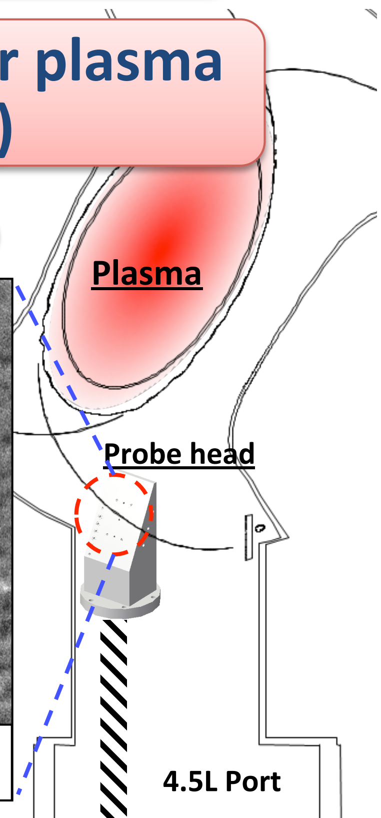
- $T_e \sim T_i \sim$ 10-20 eV
- $n_e \sim 10^{18}$ m^{-3}
- Particle flux ~ 10^{23} He/m^2s
- Av. heat flux ~ 3 MW/m^2

W exposed to divertor plasma (He discharge)

Exposed for only 1s (1 shot)

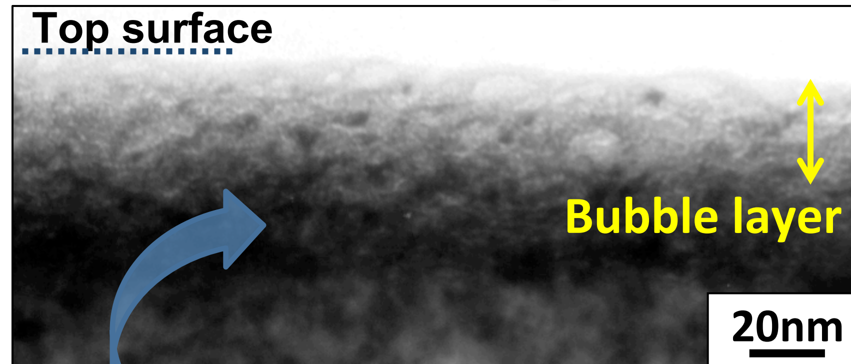


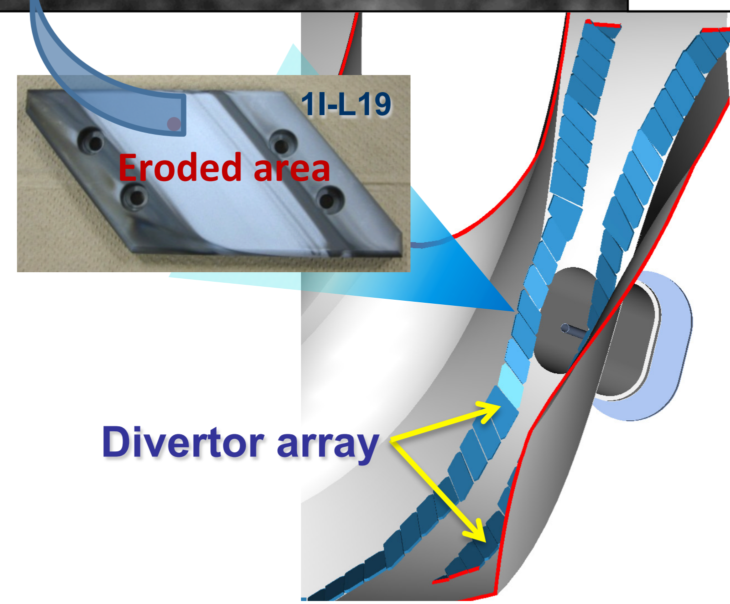




W-helical divertor

Cross-sectional TEM image





New PWI study "W-Fuzz expose to LHD divertor plasma"

Recently, the most intensive damages by helium on tungsten surface (W-fuzz) have been energetically reported in laboratory device.

Back ground

- W-fuzz structure is easily formed when the temperature is in the range 1000-2000 K, and the incident energy of helium is higher than 20 eV. Such a structure might be also formed in "<u>steady state plasma operations</u>" with helium gas puffing in LHD divertor armour in the near future.
- The results from the NAGDIS have revealed that such fine fiberform structure could make it significantly easier to trigger a unipolar arcing and exfoliate some areas of the W-fuzz structure itself. The initiation of arc could lead to a large scale exfoliation, which must be the serious impurity sources for fusion plasmas. However, irradiation tests of the W-fuzz in the large-sized, high-field plasma confinement devices have been scarcely conducted so far.

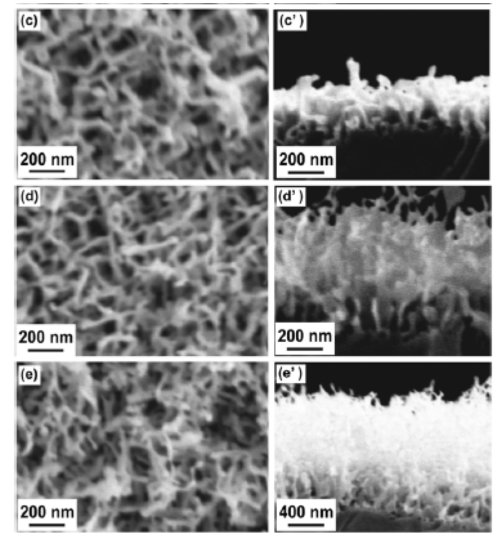


Fig. 1. Growth process of W-fuzz irradiated with low energy He ion (up to 5.5×10^{25}He/m^2) [1].

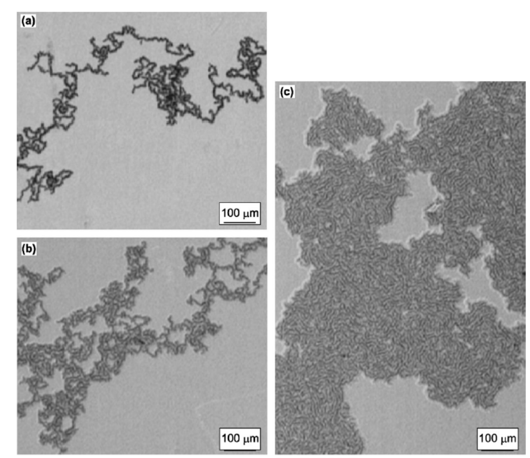


Fig. 2. Picture of the arc trails on W-fuzz specimen in NAGDIS [2].

[1] S. Kajita et al., Nucl. Fusion 49 (2009) 095005
[2] S. Kajita et al., J. Phys. Lett. A373 (2009) 4273-4277

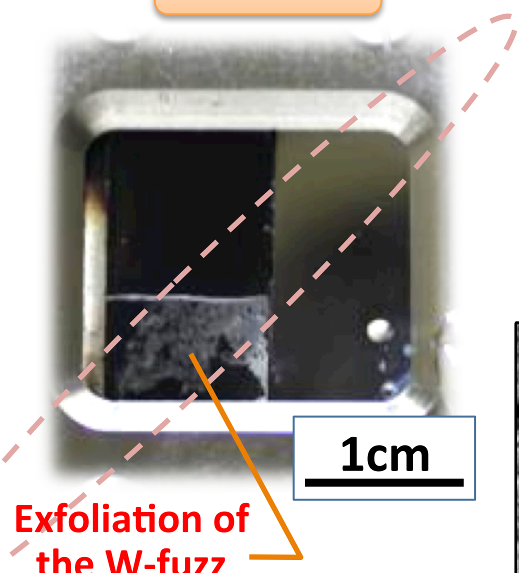
Exfoliation of the tungsten fiberform nano-structure by unipolar arcing in the LHD divertor plasma

M. Tokitani, Nucl. Fusion 51 (2011) 102001

Objective

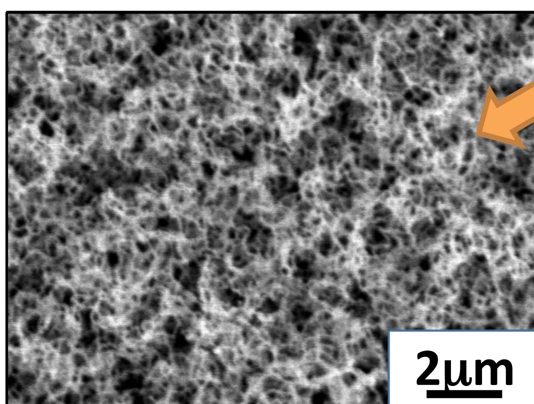
W-fuzz layer produced in the NAGDIS was exposed to the LHD hydrogen divertor plasma to investigate erosion of the W-fuzz due to the generation of the unipolar arcing on the fuzzy surface. After the exposure, surface analyses of the eroded/exfoliated area were performed by using SEM and laser microscope.

After

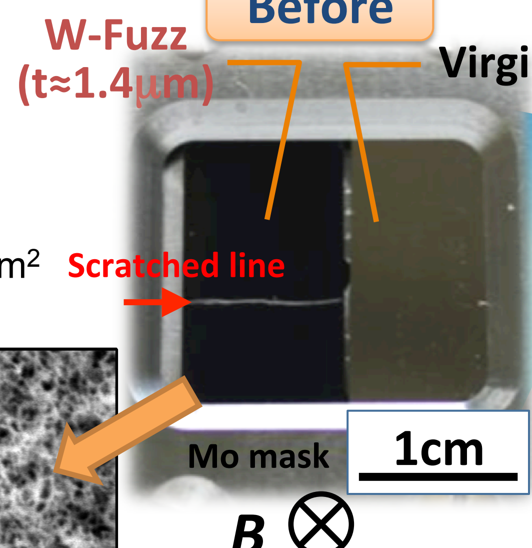


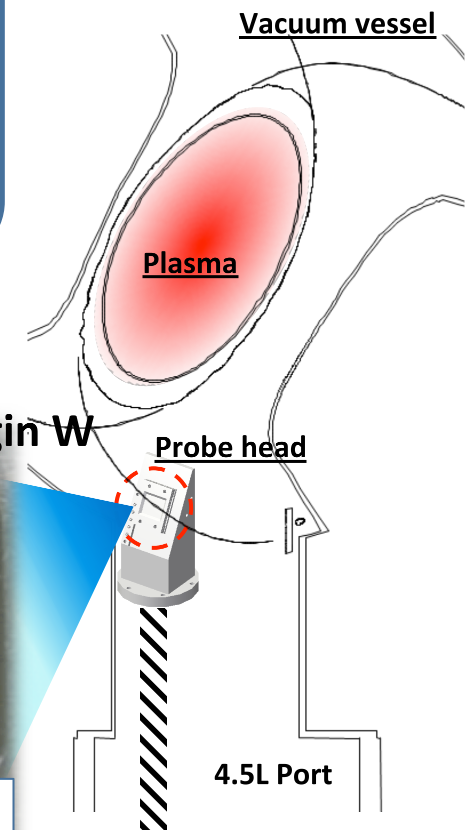
NAGDIS
He discharge

- Temp.: 1460K
- Flux: 1.21×10^{22} He/m^{2}s
- Fluence: 2.17×10^{25} He/m^2
- Irrad. energy: 57eV



Before

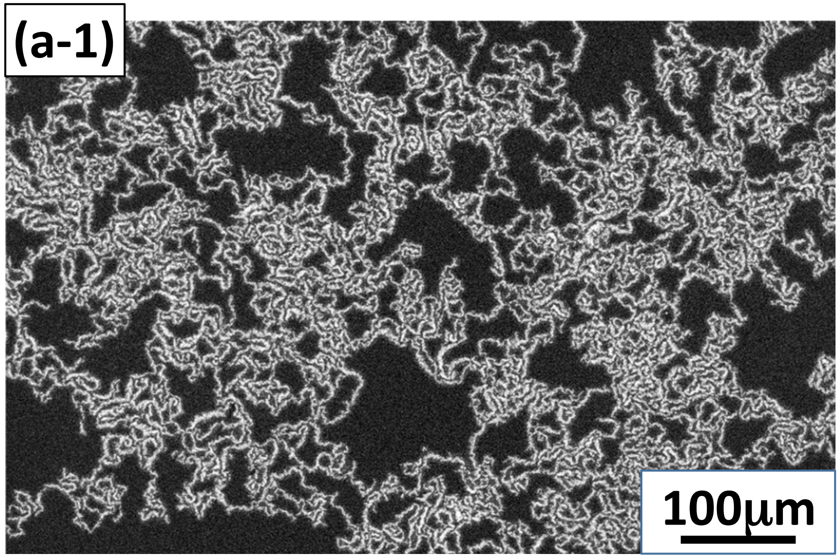


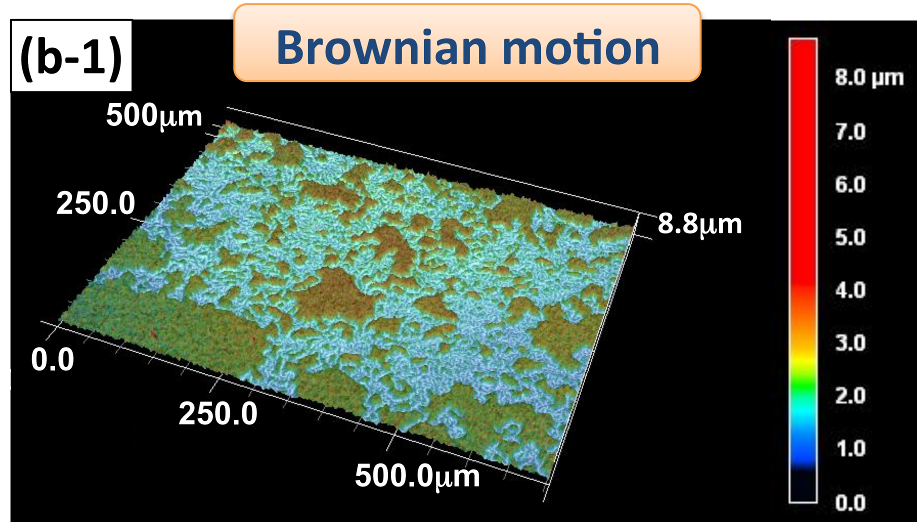


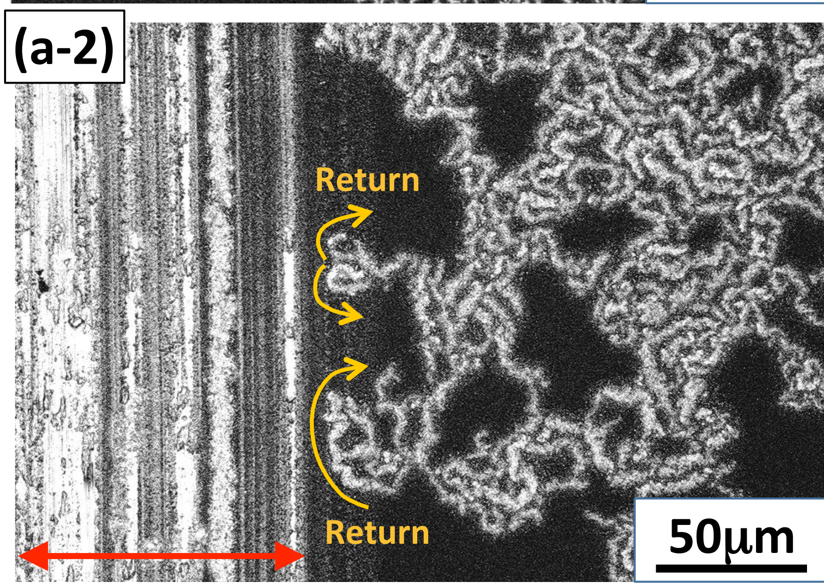
LHD
H discharge

- #103812 (1 shot only)
- NBI#1~5 (2s injection)

SEM and laser microscope image







Scratched line

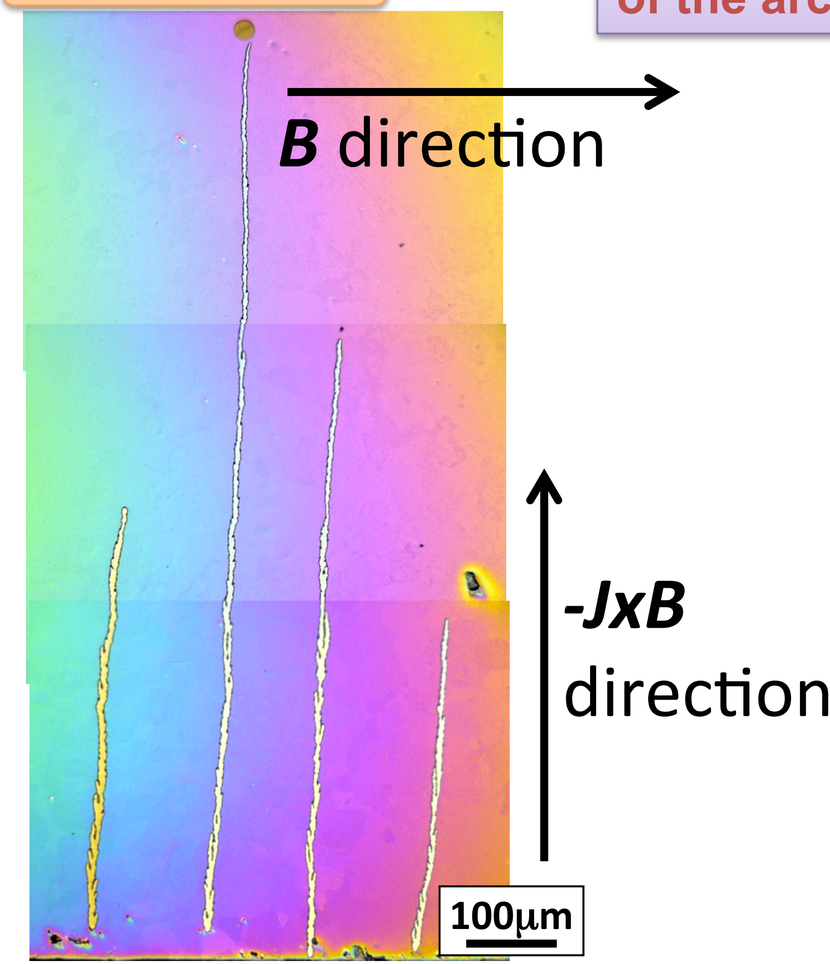
- Very fine and randomly moved unipolar arc trails were clearly observed. Spot size : 8μm (W) x 1μm (D).
- The fuzzy surface was exfoliated by continuously moving arc spots even for the very short exposure time (< 2s).
- This is the first observation of unipolar arcing and exfoliation of some areas of the W-fuzz structure itself in a large plasma confinement device with high-magnetic field.
- Their motion was almost equal to Brownian motion (Fractal analysis).
- Arc spots cannot go inside of the scratched line.

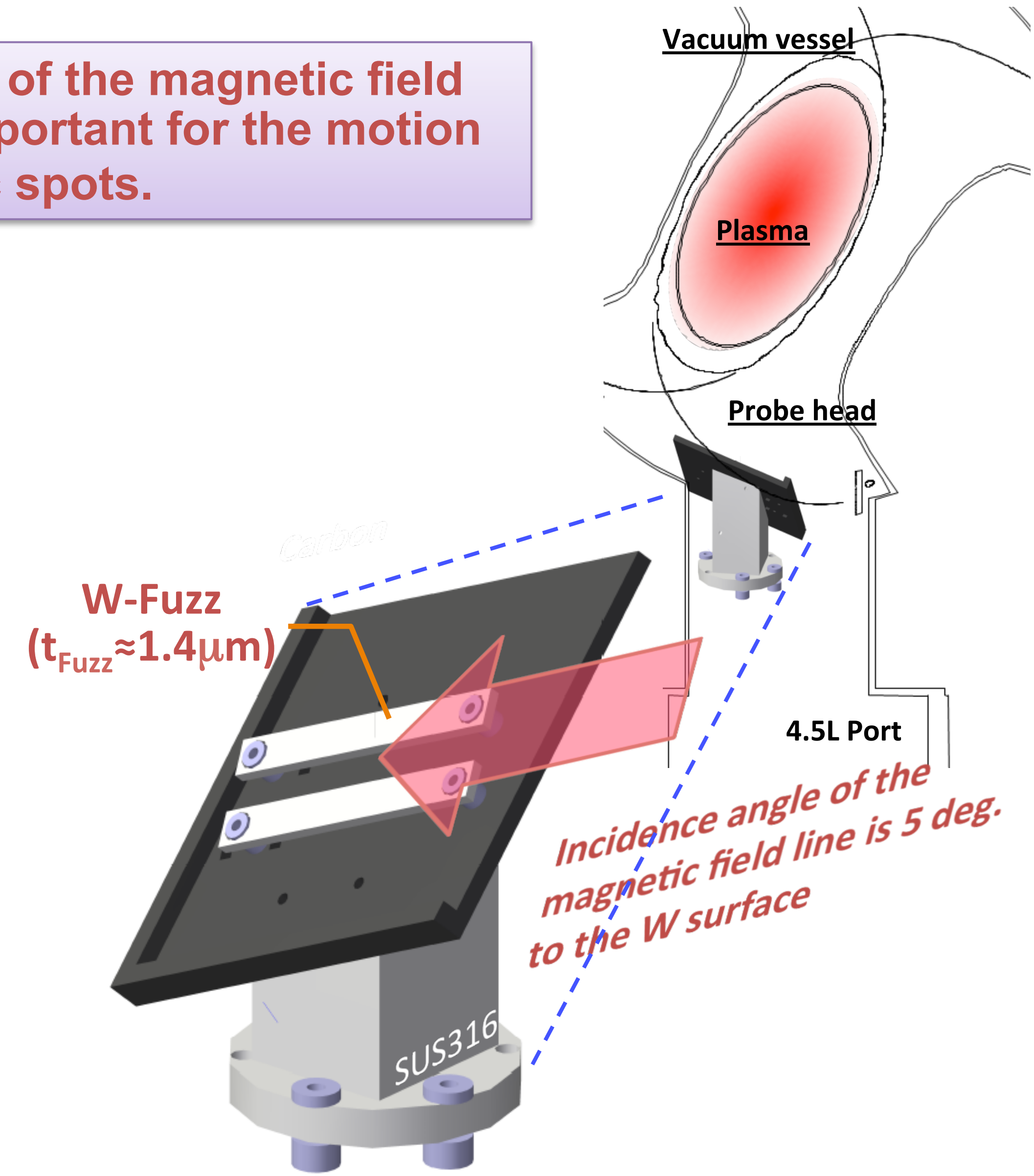
W-fuzz can lead to the initiation of arcing and subsequently cause a large scale exfoliation of W and provision of a significant W source.

Influence of the incidence angle for the moving direction

Direction of the magnetic field line is important for the motion of the arc spots.

Linear motion





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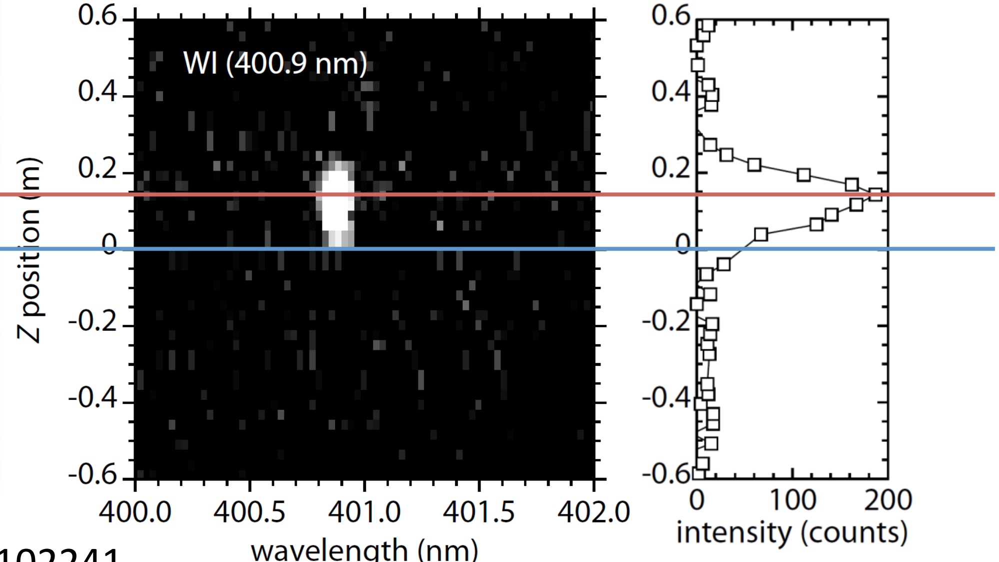
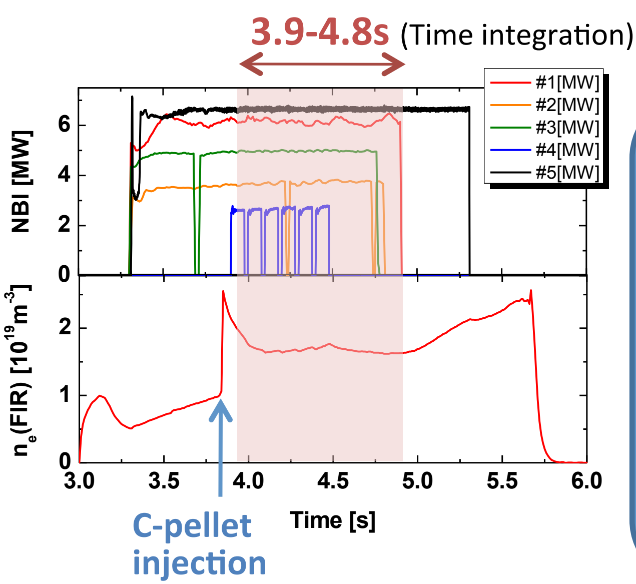
Identification of WI emission

~10^{17}W/m^{2}s以上で検出可能 by M. Goto

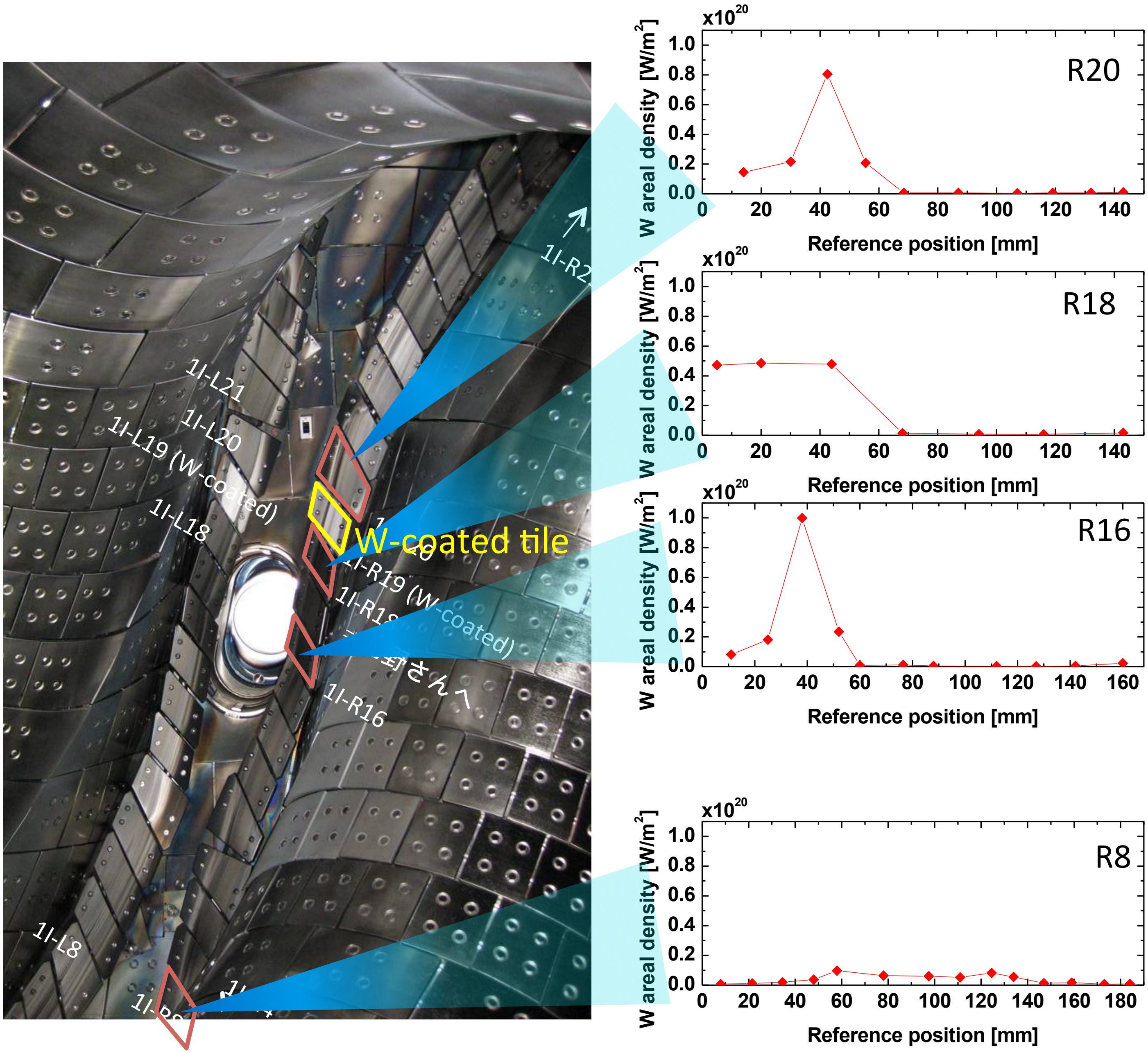


#102241
T_i~6keV
3.9s-4.8s (Time integration)



- Emission of WI line was clearly observed from the W-tile by visible spectrometer with the integration time between 3.9-4.8s.
- Spatial distribution of the WI emission does not show broader. This means that emitted W atoms not so deeply injected into the bulk plasma.
- Starting time of the WI emission seems to be close to the timing of the C pellet injection.
- Relationship of the C pellet injection and WI emission should be considered in subsequent analysis.

"Transport" and "Deposition" of W



- Distribution distance of the sputtered W atoms were not so far.
- Deposited W atoms were mostly located on the private region.

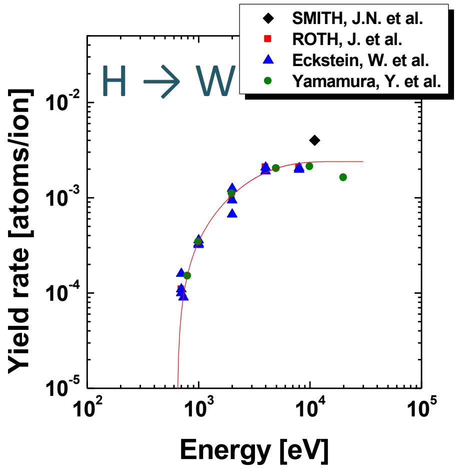
Wのスパッタリング損耗

1. 分光分析

- *Visible spectrometerは，W neutral fluxに対してどのくらいの感度があるのか？*
  - ~10^{17}W/m^{2}s以上で検出可能.
- *プラズマコア部でのWの蓄積の有無がどうにかして確認できないか？*
  - Visible, VUV 双方で明瞭なピークは確認できなかった.

2. 損耗に寄与する入射Hのエネルギー分布

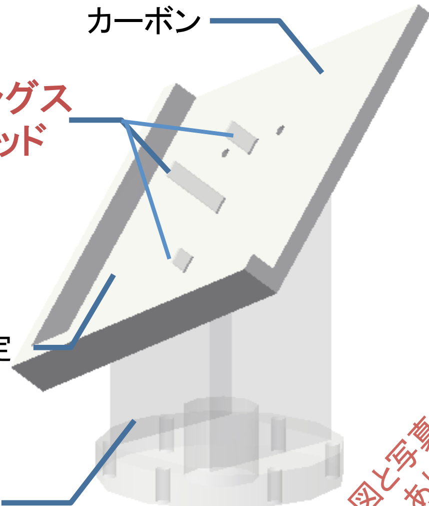
- *タイルへの入射粒子のエネルギー分布を評価してみては？*
  - 500eV以上：0.3%，400eV以上：3%，300eV以上：18%



パルス熱・粒子負荷による損耗/溶融

カーボン

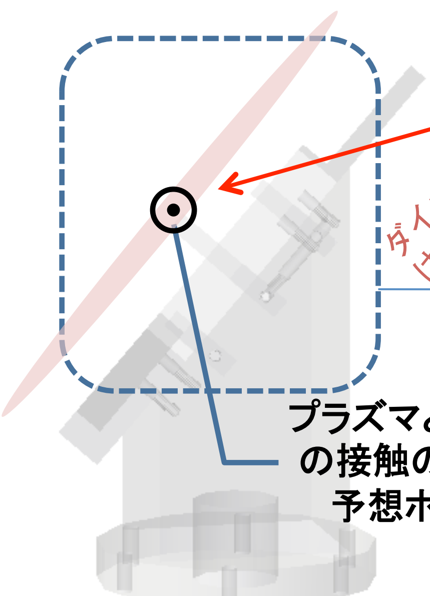
微小タングステンロッド



Mo bolt固定

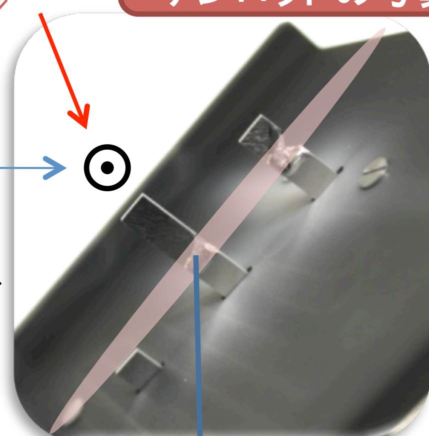
SUS316

ダイバータプラズマの磁力線は本図と写真では紙面にほぼ垂直方向(多少のずれあり)



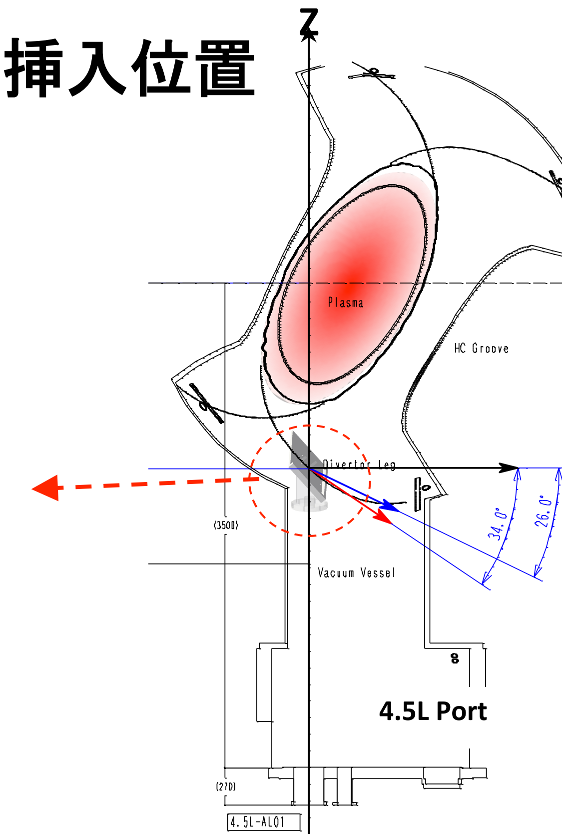
プラズマとWロッドの接触の当初の予想ポイント

照射後のタングステンロッドの写真



実際のストライクポイント

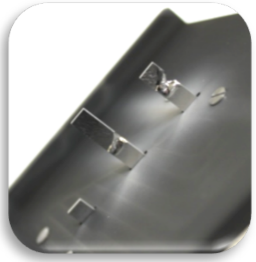
挿入位置

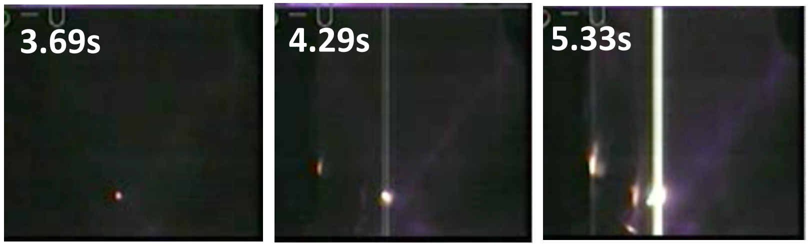


計測

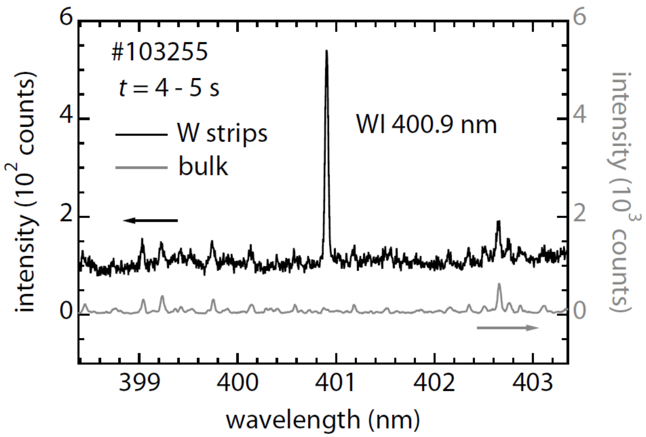
- 放電中のタングステン溶融物の飛散を可視カメラで観察.
- 放電中のタングステン元素の放出を分光器で観察.

実験結果と考察





タングステンロッド発光の時間変化 (#103256)

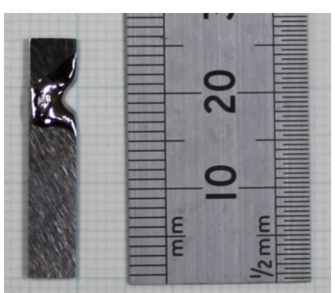


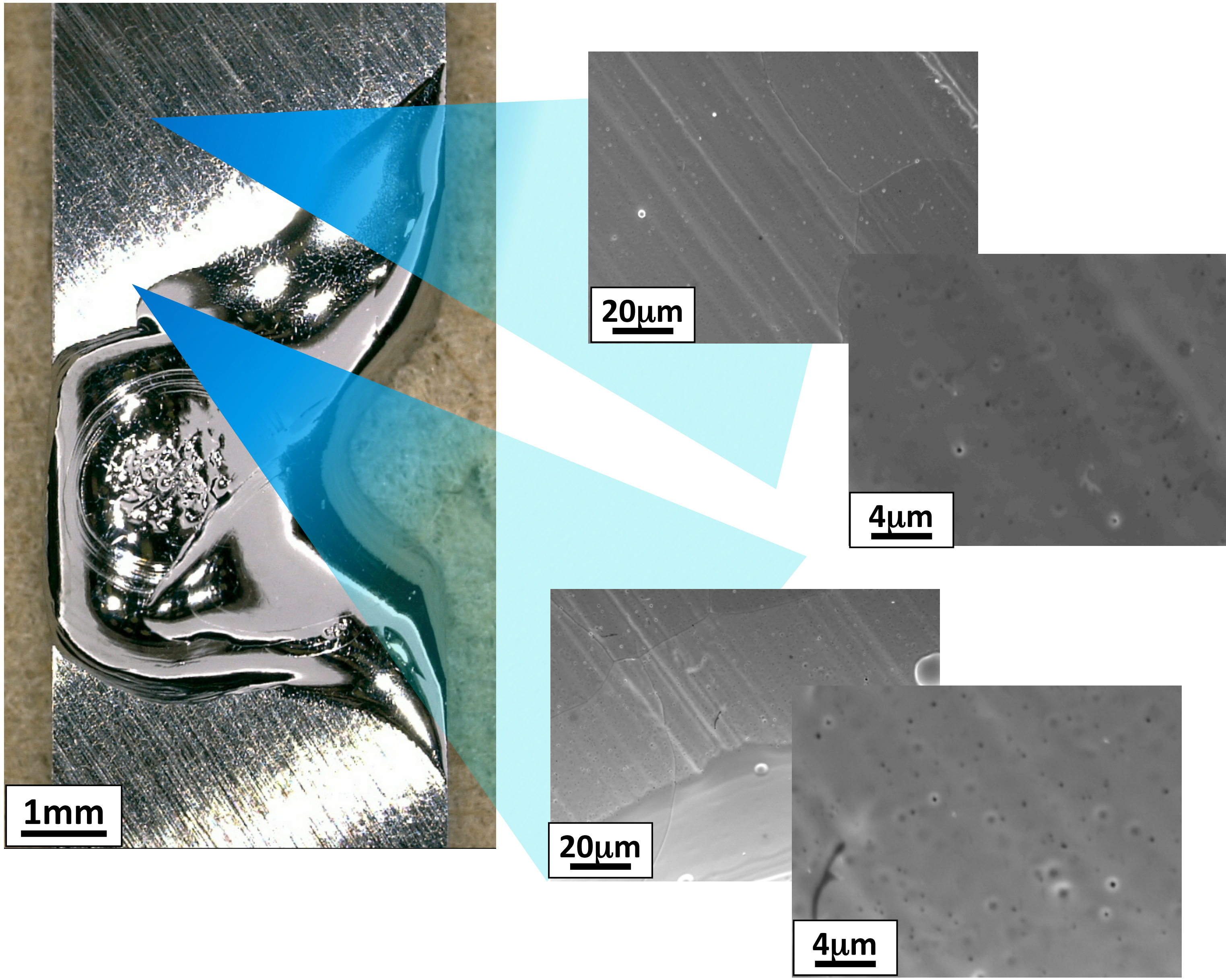
タングステンロッド周辺部とバルクプラズマ中でのWI(400.9nm)スペクトルの比較(#103255)

- 溶融が確認される2本のタングステンロッドにダイバータプラズマが照射された．黒鉛タイルとは接触していない．
- 放電時間の経過と共にロッド先端部の発光が顕著になる．
- W溶融物の飛散をカメラで確認することはできなかった．
- 可視分光計測より，タングステンロッド周辺部にWIラインスペクトルの強い発光が確認された．
- バルクプラズマ中での発光は見られなかったため，プラズマへの目立った影響はなかった．
- 黒鉛タイルに飛散したタングステンの付着が確認されたため，今後分析を行い，タングステン元素放出の方向依存性を導出する予定．

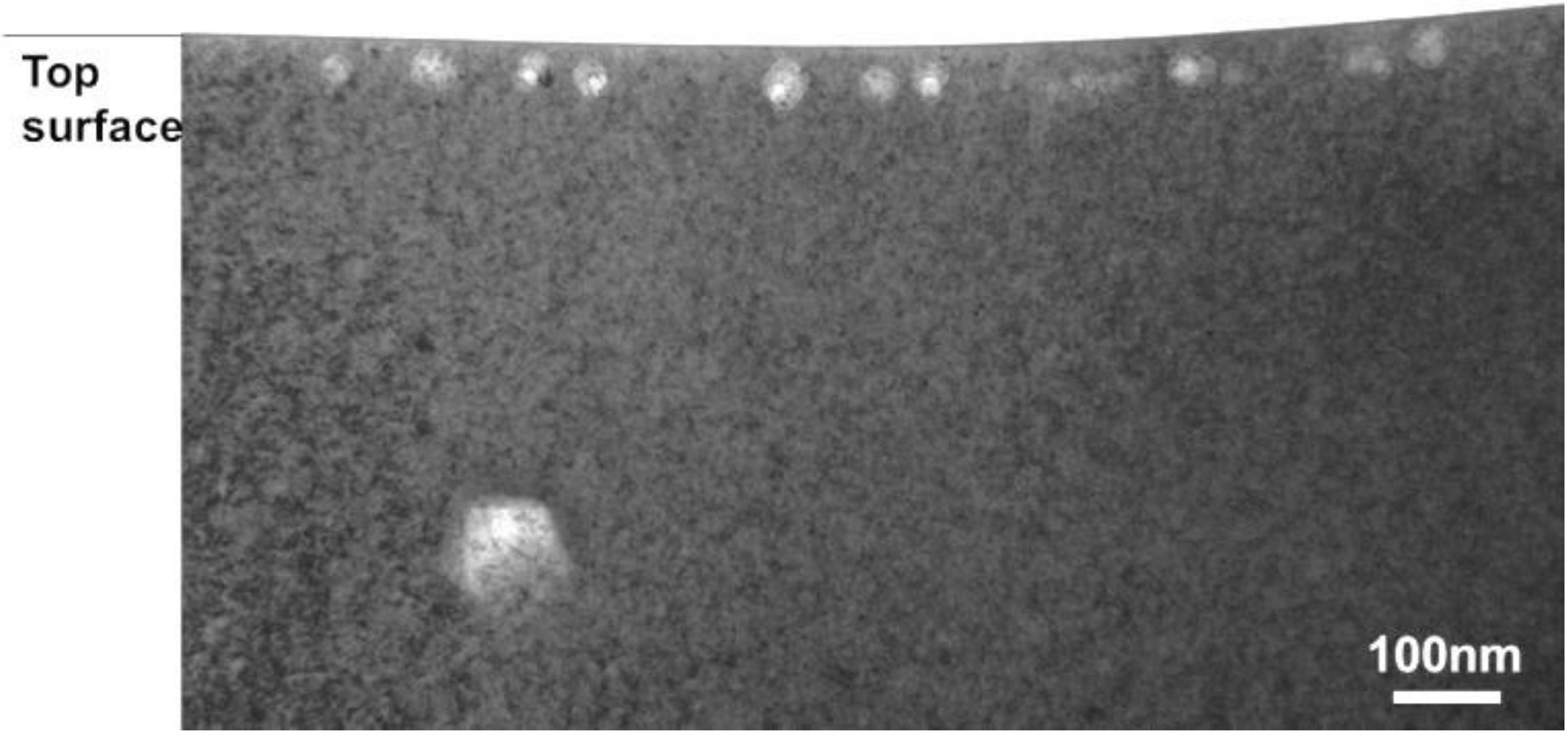
溶融層周辺の損傷特性

全体像





Hキャビティの形成



TEM images of W irradiated with LHD divertor plasmas.
100-200eV, ~10^{22-23} ions/m^{2}s.

H irradiation effects in LHD

Cavities are difficult to form in the H irradiation

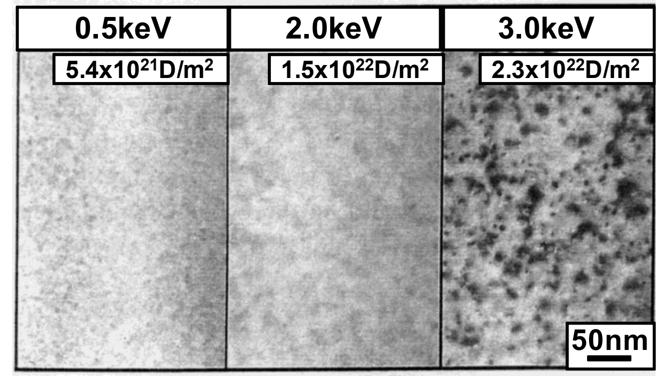
- In the hydrogen irradiation, no defects clusters are formed if the incident energy of the ion is less than 2keV [2]. In that case, injected hydrogen atoms can easily desorbed up to the 800 K [3,4]. Even if the energy is higher than the threshold value, formation of dense cavities has not been observed for the irradiations up to the order of ~10^{22} ions/m^2 [5].

Large size cavities were densely observed in LHD

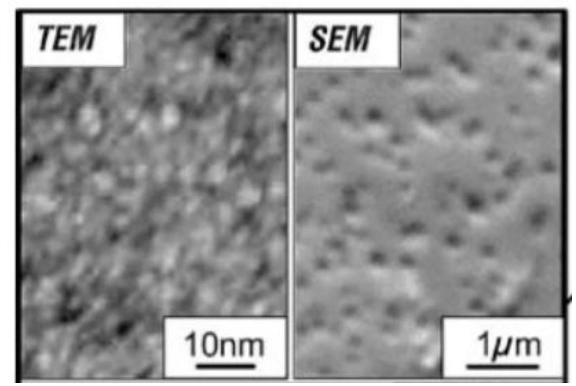
- Unexpected damages such as formation of fine cavities were observed in the LHD experiments
- Synergistic effects of high-heat and -particle loads would cause the undesired extensive damages for tungsten, and such damages change the retention property of the hydrogen isotope.

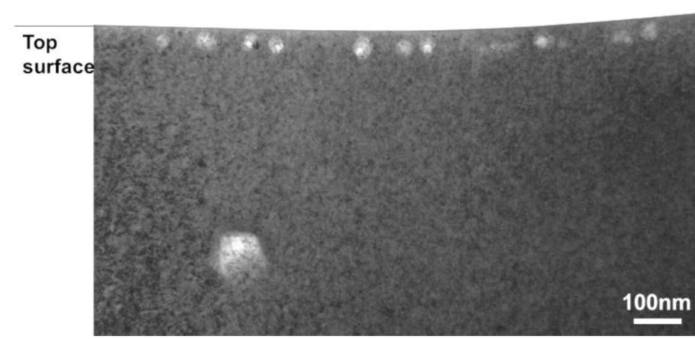
Systematic experiment of high-heat and -hydrogen loading in W

- Systematic experiment of high-heat and -hydrogen particle loading in tungsten focused on the microstructure change has never been done so far.
- **Change of the microstructure due to the high flux hydrogen plasma irradiation will be observed as a function of the flux, energy and temperature in MAGNUM-PSI. (TEXTOR collaboration)**



TEM images of W irradiated with various energy of D^+ in room temperature.





TEM images of W irradiated with LHD divertor plasmas. 100-200eV, ~$10^{22\text{-}23}$ ions/m^2s.

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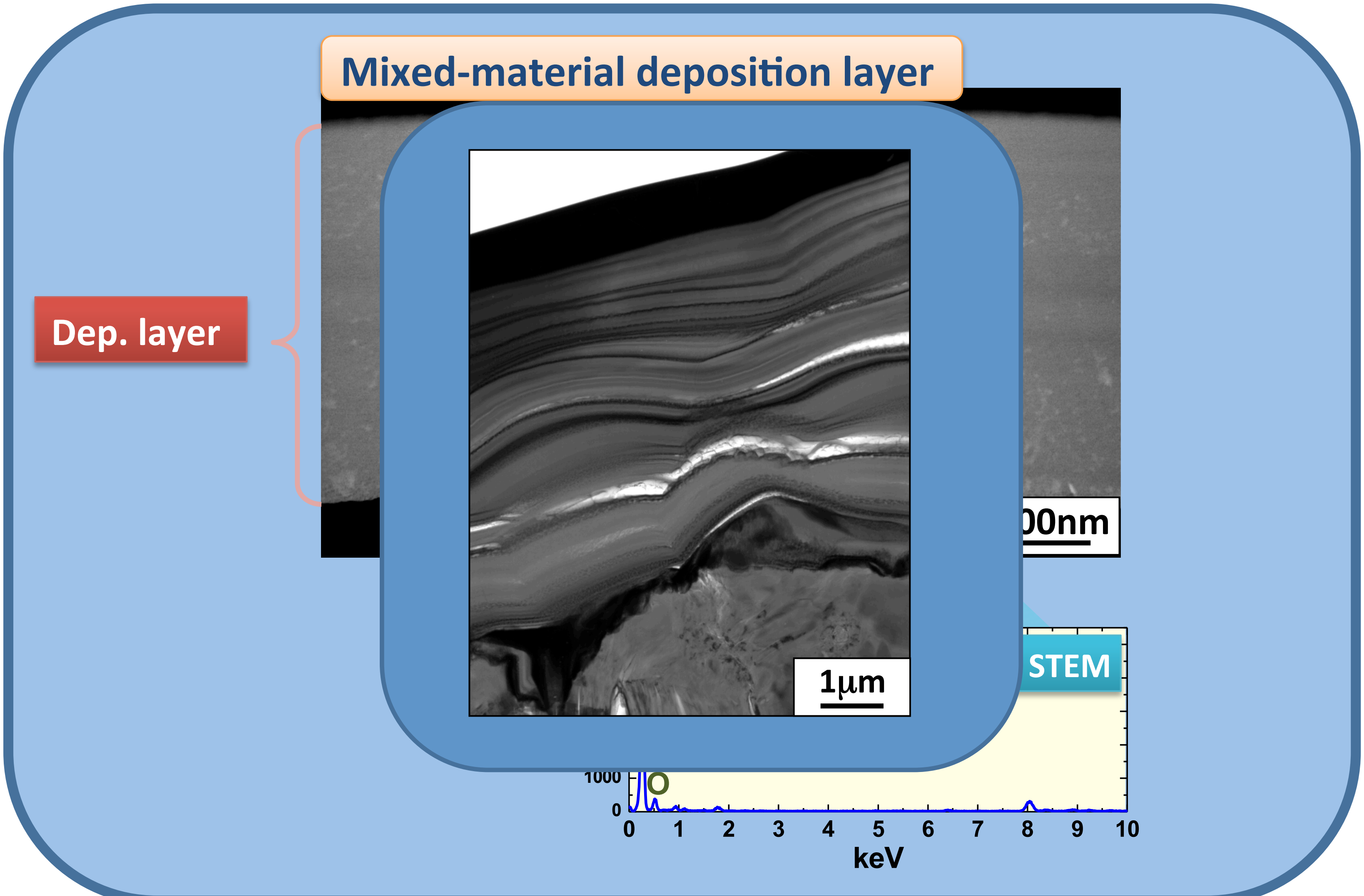
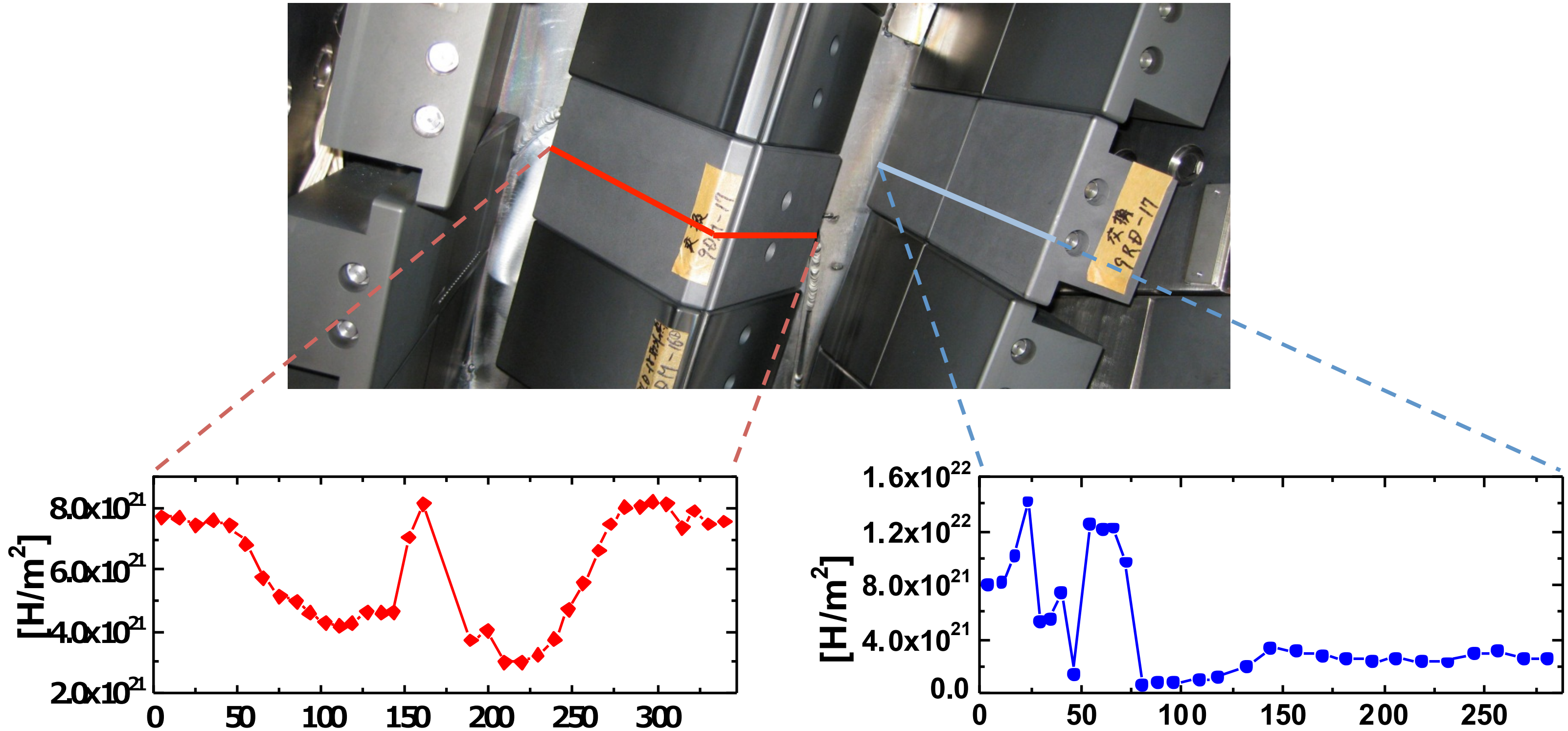
H areal profile



If we would want to suppress the hydrogen co-deposition with carbon, full W divertor system would be effective!!

H areal density Divertor and Dome



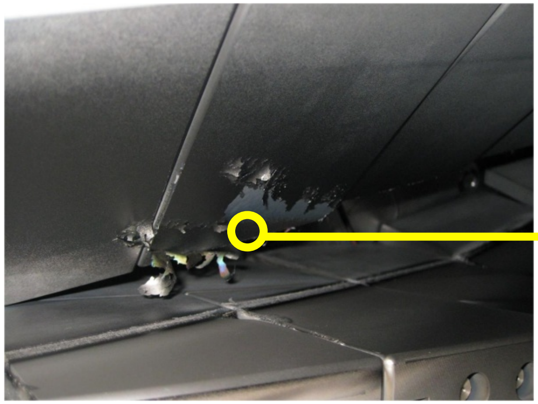
- **Dome**

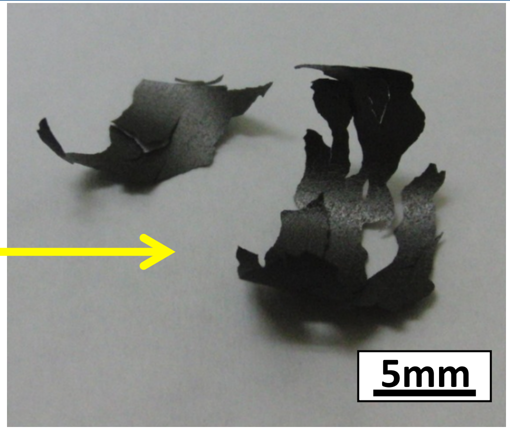
  High H retention was located at the Lower and top region.

- **Divertor**

  Highest H retention ~1.5×10^{22} H/m^2 was observed near the strike point. The value was about two times higher than that of the open divertor tile.

Mixed-material flakes on the "DOME"

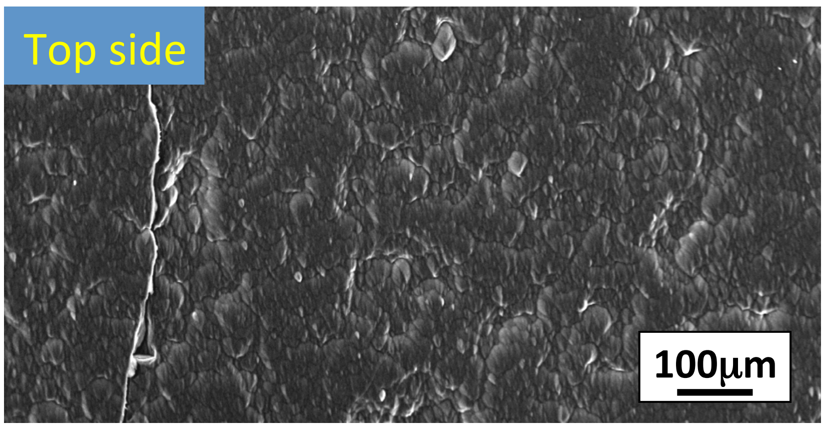


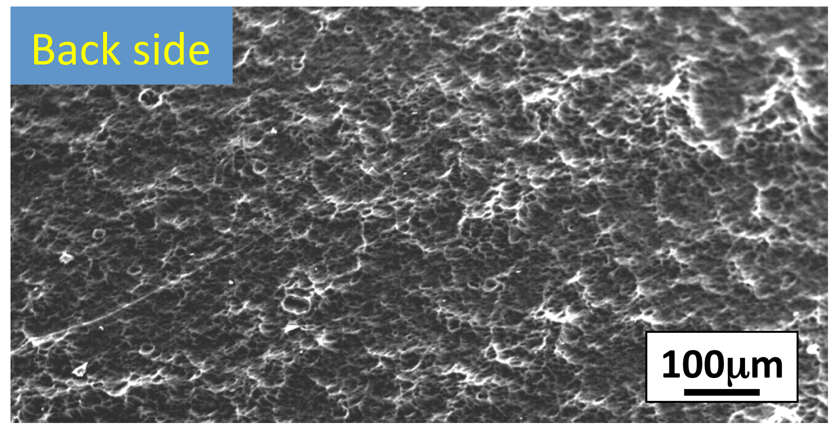


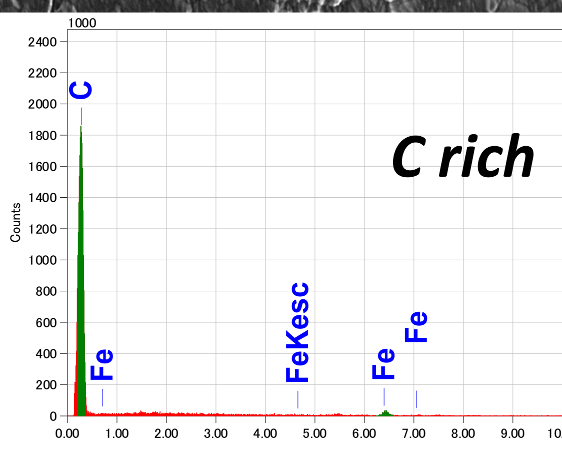
Firstly, Fe impurities were deposited, and then, majority of the impurity element was changed to the C at the late stage.

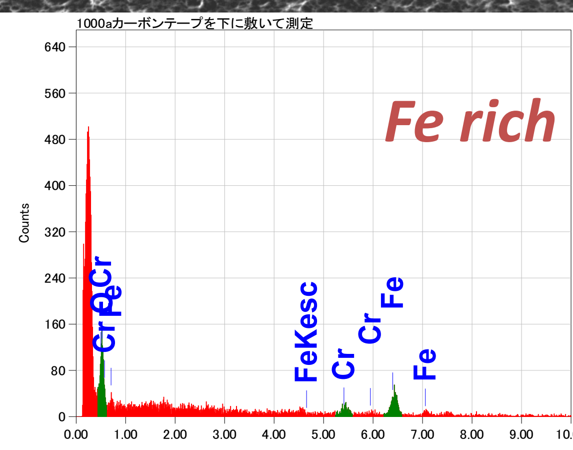
Fe deposition layer act as the initiator of the exfoliation of the whole deposition layer! [1]

[1] M. Tokitani et al., J. Nucl. Mater. 417 (2011) 668

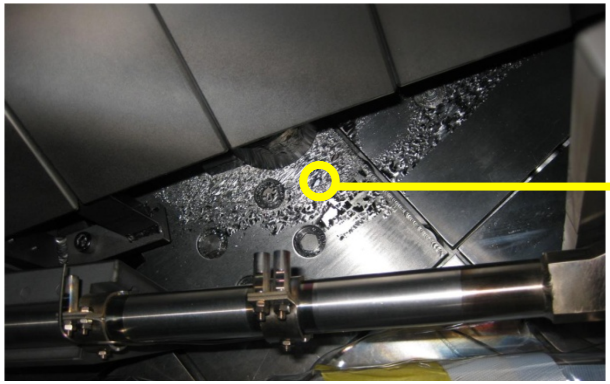


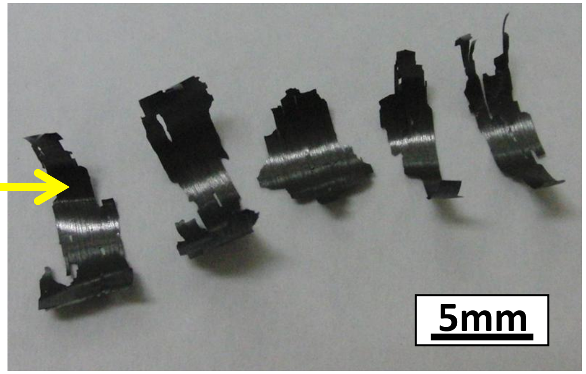


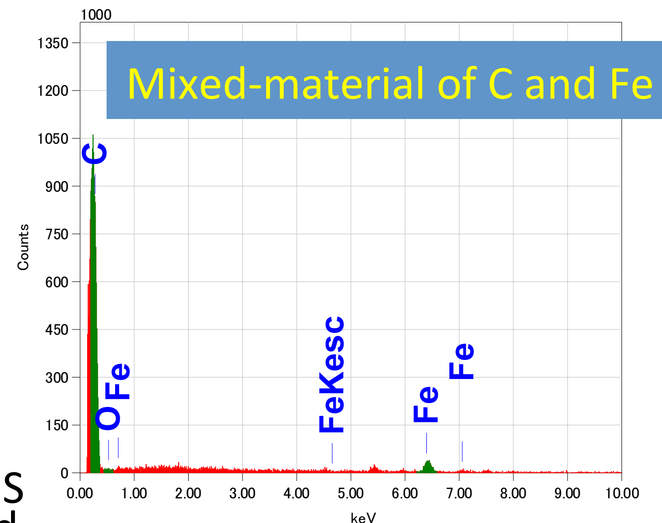




Mixed-material flakes on the "WALL"

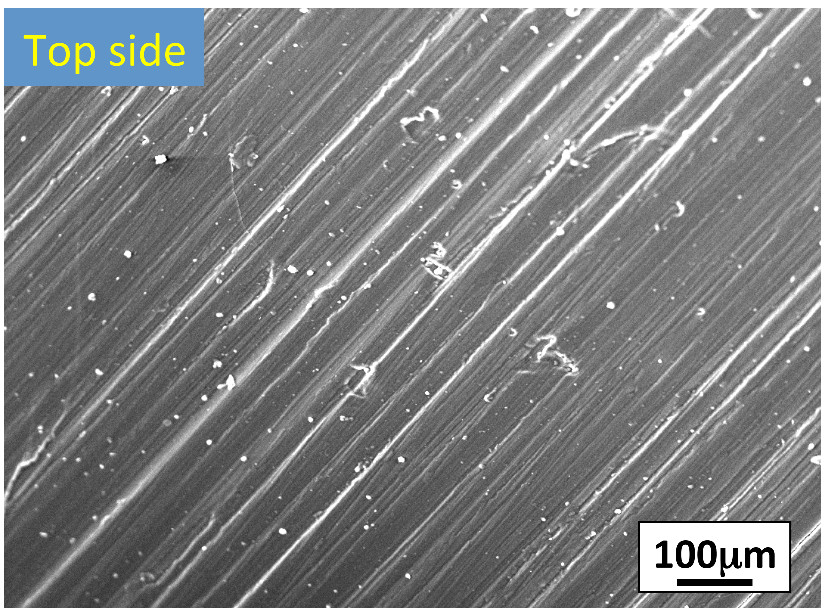


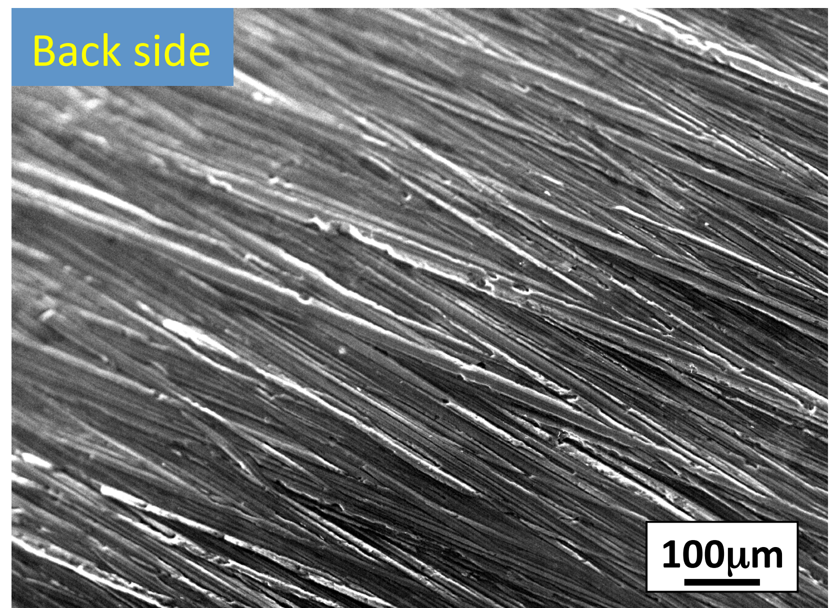




Fine scratched like structures were formed.

Such scratched structures would have been caused by the SUS substrate. Because SUS wall surface has the original scratched structures due to the mechanical polish.

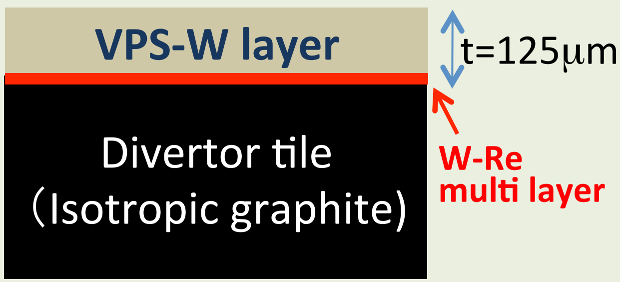


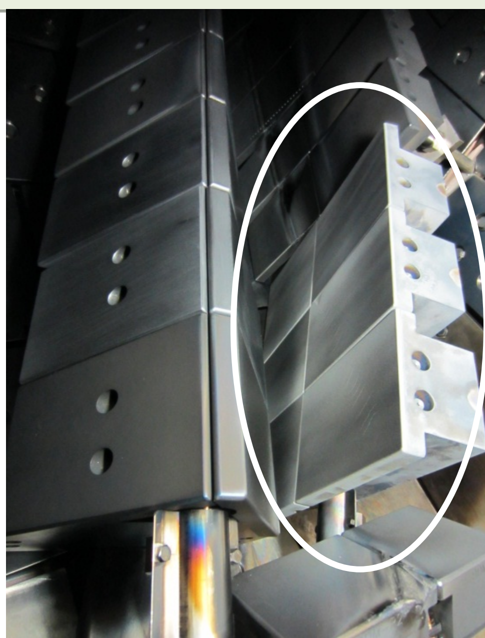


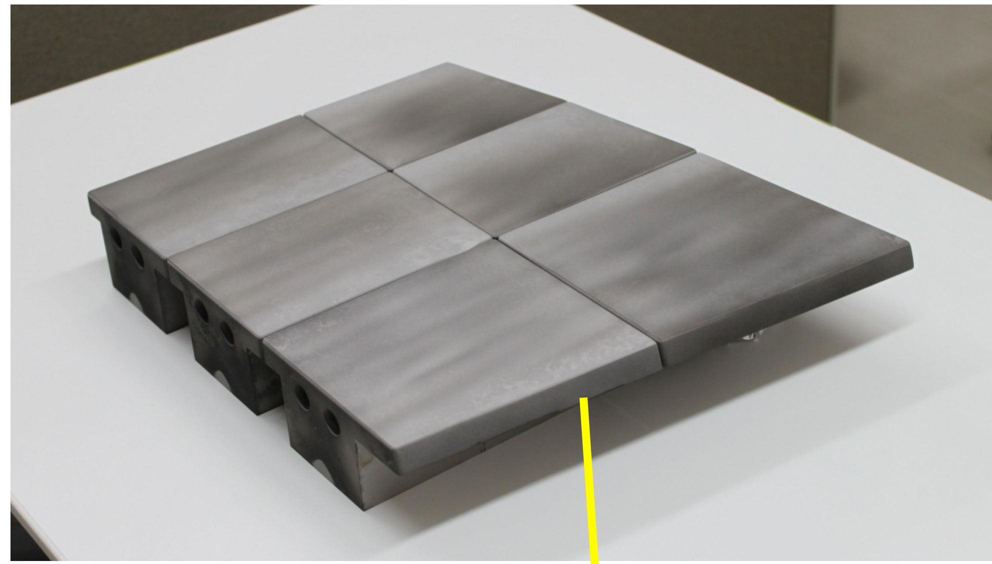
W-coated IG tiles were installed

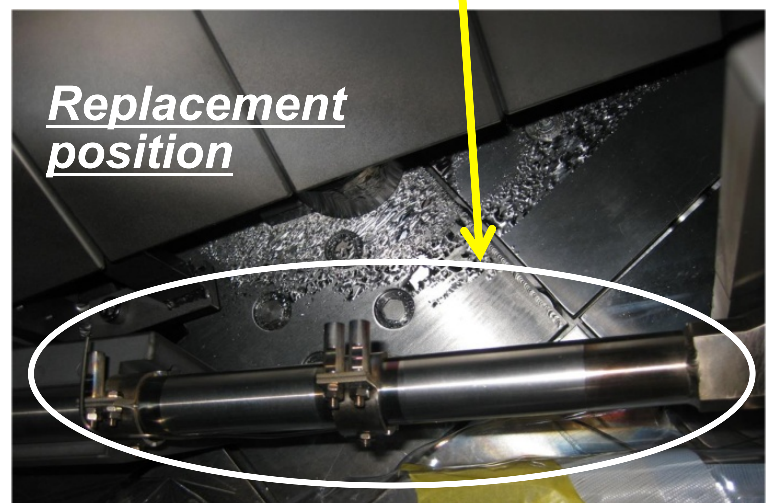
VPS-W coating will be prepared for 3 pairs of divertor tile.
H inventory and dust generation would be suppressed by VPS-W.

The VPS-W layer with 125μm thickness will be coated on the IG tile.









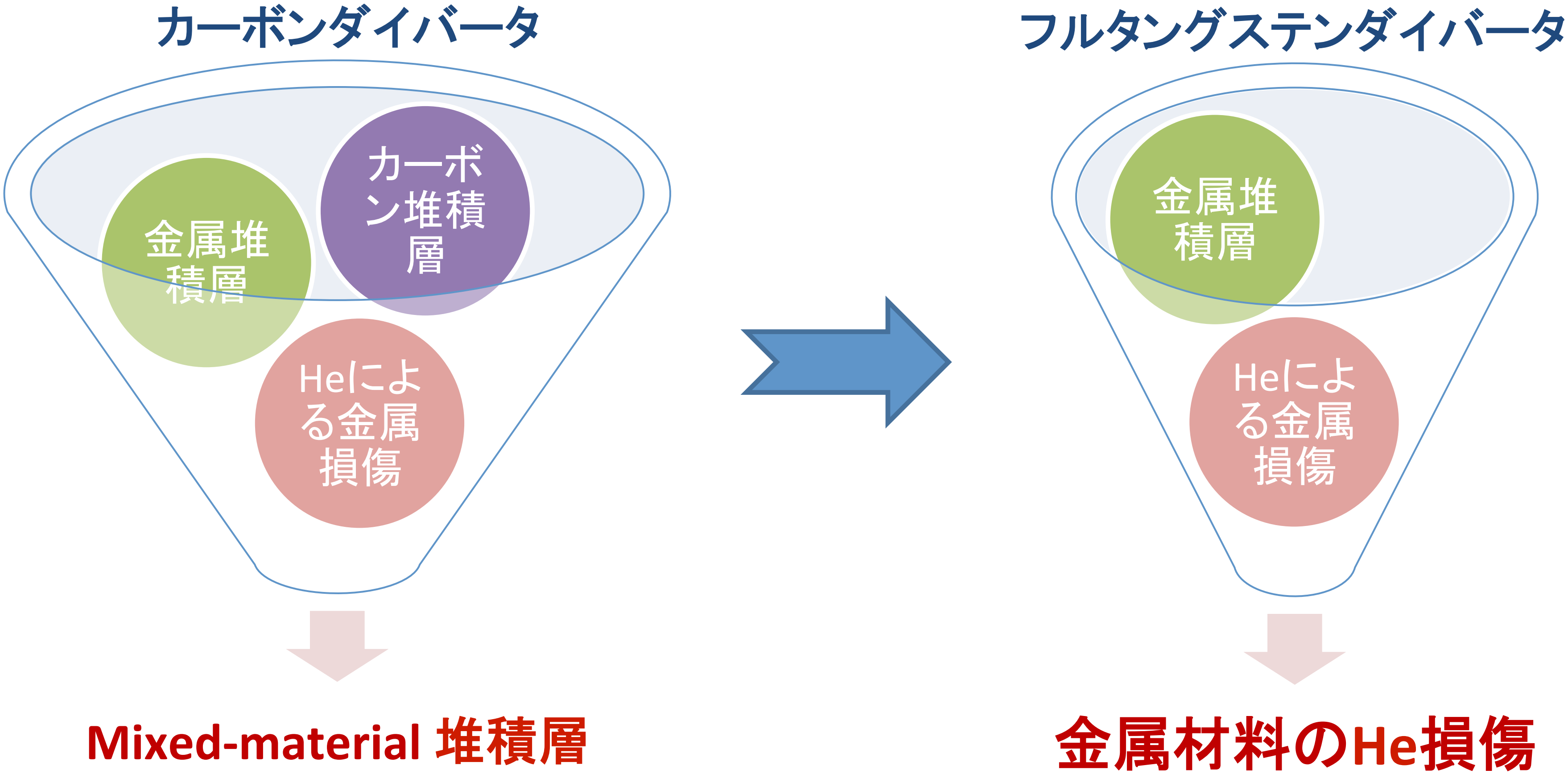
LHDフルタングステンダイバータのメリット

現象の単純化とそれによる物理的メリットは大きい.

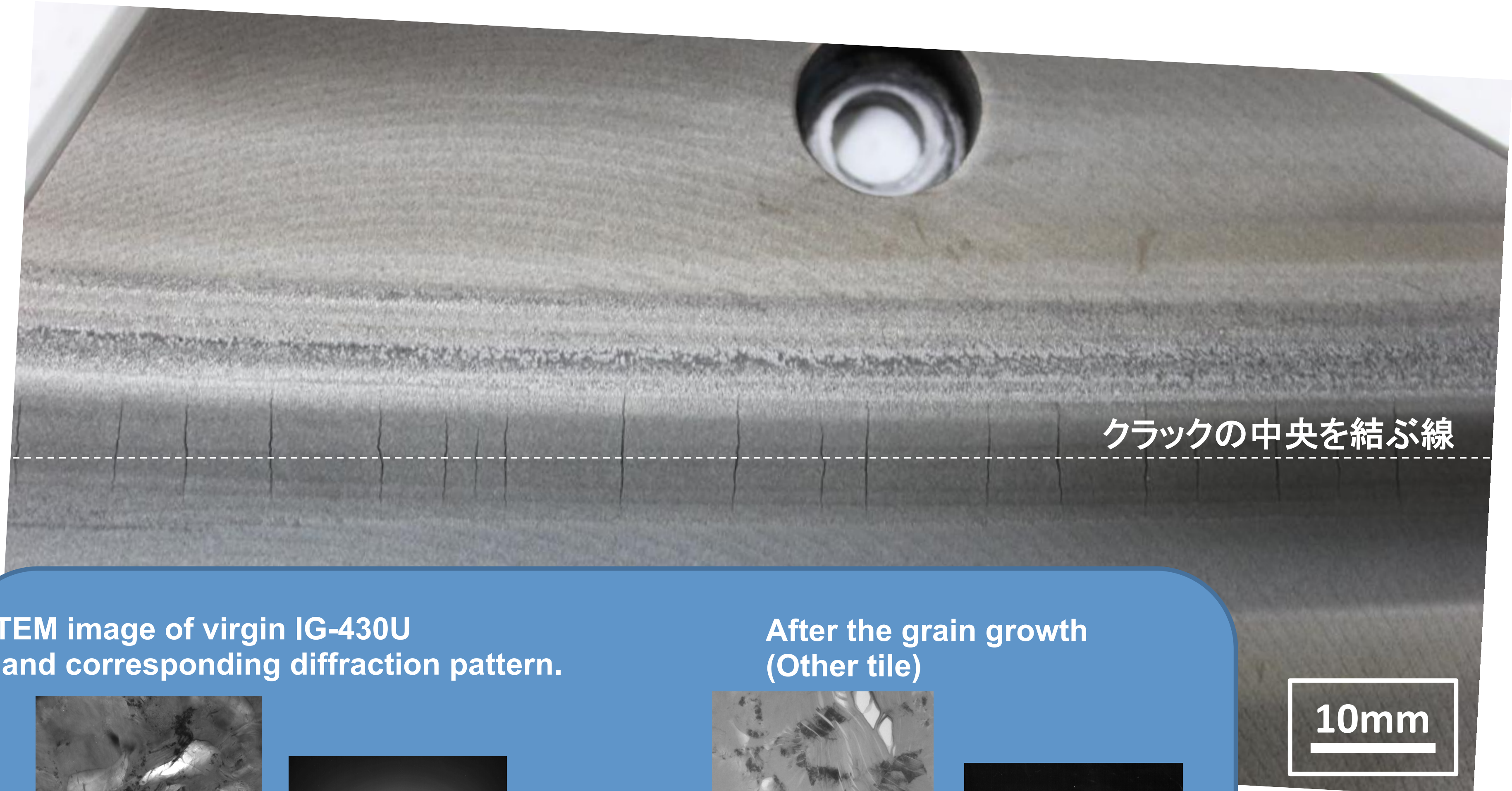
特に,

⇒・水素同位体リテンションの抑制

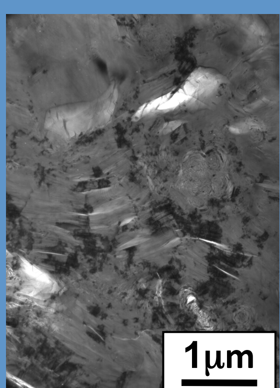
⇒・ダスト生成の抑制

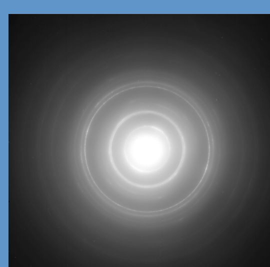


カーボンへの集中したクラックの発生



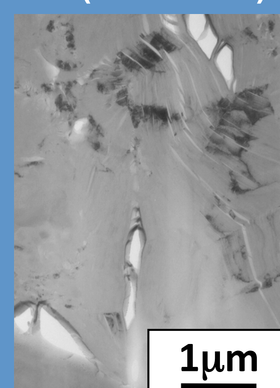
TEM image of virgin IG-430U
and corresponding diffraction pattern.

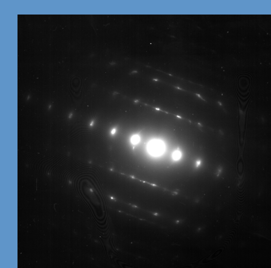




Different

After the grain growth
(Other tile)





定常放電とタングステンダイバータ

フルタングステンダイバータを導入した場合,

- 単位面積当たりの水素同位体の捕捉量が1桁以上減少.

  ⇒粒子コントロール(壁が突発的に燃料sourceとなることへの対策)が容易になる.

- 堆積層構造の単純化によりフレークの剥離によるダスト生成が減少する.

  ⇒不純物混入

(A)密度上昇によるプラズマ崩壊
(B)不純物(火花)による定常放電の停止

これらの課題の解決につながる可能性が大
導入したほうが良い.

導入時に問題となる課題

- バルク材料ではLHDの配管に設置するのは困難⇒W被覆黒鉛が現実的
- カーボン材料をプラズマに見込ませては意味がない. W被覆黒鉛では側面や裏面もWコーティングする必要がある. ⇒可能か?
- W被覆黒鉛は日本で製造できるメーカーがない. コストの面から国内メーカーで製造したほうが有利. メーカーと協力して製造法を開発する必要がある.

まとめ

1. ヘリウムによる損傷

- ダイバータプラズマへの照射実験により，Heによる損傷について知見が得られた.
- W-fuzz層の強磁場中での照射試験によりユニポーラアークによる剥離を確認した.

2.パルス熱・粒子負荷特性

- Wダイバータからの WIの放出を確認した．放出されたWは近傍のダイバータタイルに堆積・分布していることがわかった.
- 強い熱・粒子負荷によって水素放電においても巨大サイズのHキャビティが形成されることを確認した.

3. Mixed-material堆積層

- 閉ダイバータ部ではドーム，受熱板上に~10^{22}H/m^{2}の水素捕捉を確認．カーボンタイルが複雑に入り組んだ構造のため，Mixed-material堆積層の大規模な剥離が形成.
- 堆積層抑制のため今期より閉ダイバータの一部をW coated graphite tileに交換した.