Status of plasma facing material studies and issues toward DEMO

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Development of Physics and Technology of Stellarators/Heliotrons en route to DEMO

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Acknowledgements : S. Masuzaki, T. Morisaki, M. Yoshimura (NIFS)

Outline

- Comparison of issues for tungsten and graphite
- Helium effects on tungsten and possible protection method
- Neutron effects on graphite and tungsten
- New tungsten development (UFG-W)
- Issues related to PWI in helical system for DEMO
- Summary

Graphite (CFC) as plasma facing materials

- Advantages of graphite as PFM's
  - High heat shock resistance
  - Non-melting feature
  - High thermal conductivity
  - Low impact on core plasma performance
- Critical Issues
  - Reduction of erosion (related to dust formation)
    - Chemical erosion decreases above 700 °C
    - Redeposition control (reduction of net erosion)
    - Material mixing could work (Be layer greatly reduces erosion)
  - Avoidance of Material degradation by neutron irradiation
    - Dimensional change, reduction of thermal conductivity
  - Reduction in T retention in redeposition layers
    - In DEMO, this would not be a big problem.

Tungsten as plasma facing materials

- Advantages of tungsten as PFM’s
  - High melting point (3693 K)
  - Low sputtering yield (high threshold energy for light ions)
  - High thermal conductivity
  - Low tritium retention
- Critical Issues
  - Avoidance of material degradation under fusion environments
    - Plasma Irradiation (D/T,He ions)
    - Pulsed heat load (ELM’s, disruptions, etc.) [Tokamak]
      - These should be suppressed in DEMO.
    - Neutron irradiation
      - Embrittlement, swelling, transmutation
  - Avoidance of core accumulation

T behavior in DEMO

- Problems of T retention would not be serious….
  - Wall temperature will exceeds 600 °C.
  - But, if coolant temperature is low (ex. 300 °C for water), hydrogen isotope trapping near coolant tubes may not be negligible.
- T permeation to coolant and dynamic retention effect should be considered.
  - T recovery system from the coolant will have heavy load, if significant permeation flux exists.
  - Diffusion barrier of T on inner surfaces of coolant tubes will confine T in wall materials, which could increase T retention (dynamic retention, only existed during plasma exposure). The effect of this dynamic retention on the degradation of materials should be investigated.
  - Design of high heat flux components and first wall of blankets need to take this mobile T effect into consideration.

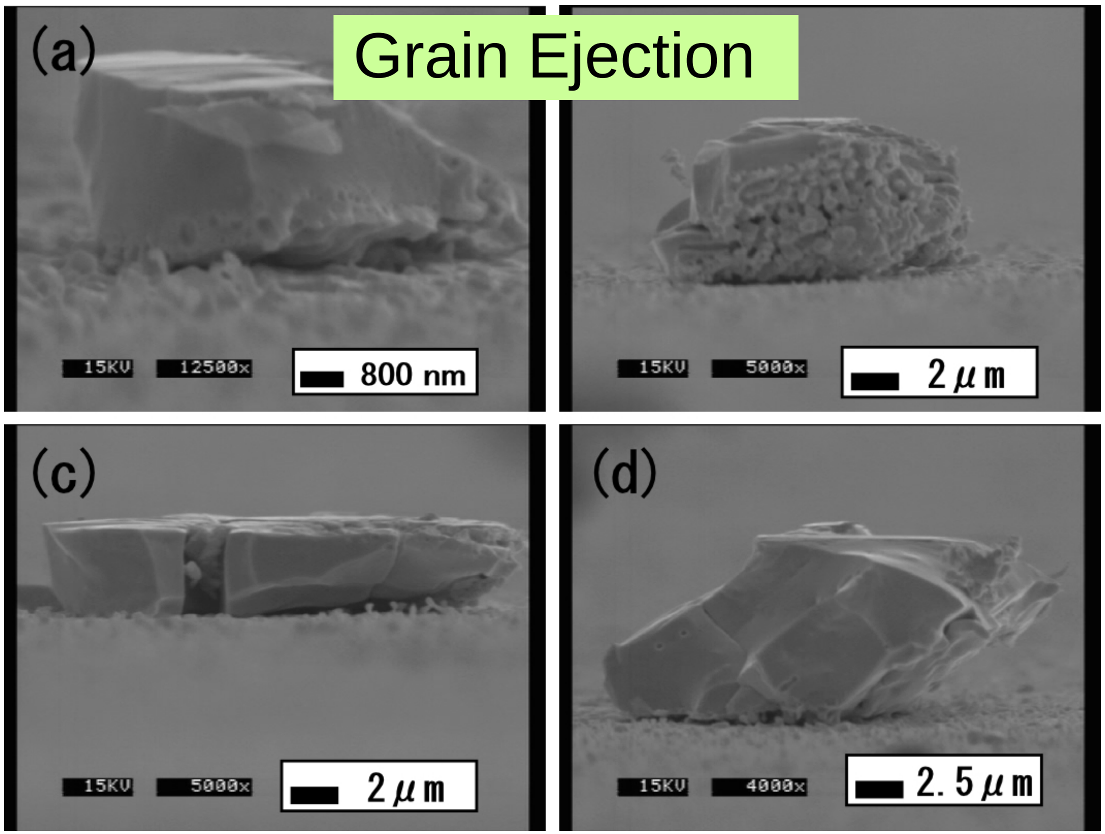
Helium effect on W

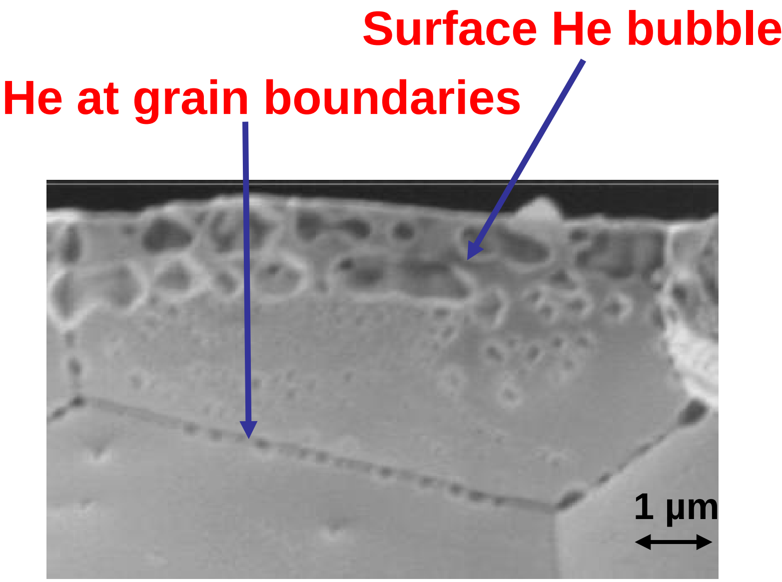
- Helium effect on surface roughening of tungsten becomes significant over 800 °C.
  - Nano-fiber (cotton-like) morphology appears at relatively low temperature.
  - At higher temperature, bubble structure grows together with recrystallization.
  - These surface modification will probably lead to enhanced erosion and dust generation, which would not be acceptable.
  - This effect also appears under mixed plasma (D, T & He (5-10%), actual burning plasma) conditions.

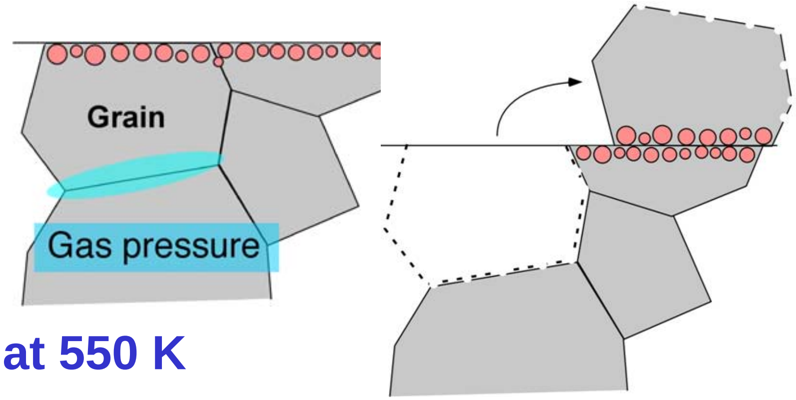
He ion irradiation effects (~1600 K)

NAGDIS-II, Nagoya Univ.

Surface He bubble formation and recrystallization with He bubbles at grain boundaries could cause enhanced erosion and dust formation







He exposure at <u>1600 K</u>, then D plasma exposure at 550 K

D. Nishijima et al., J. Plasma Fusion Res. 81 (2005) 703.

Effect of He plasma on various grades of W

M. Baldwin (UCSD), TITAN Workshop 2008

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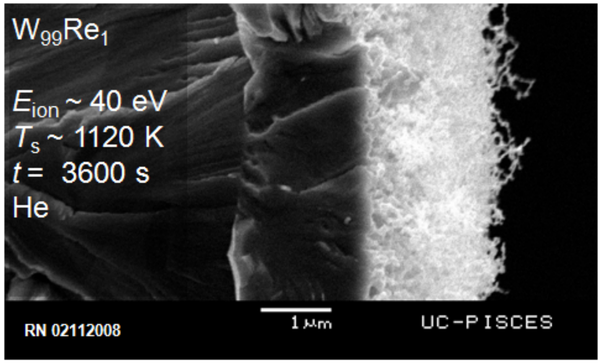

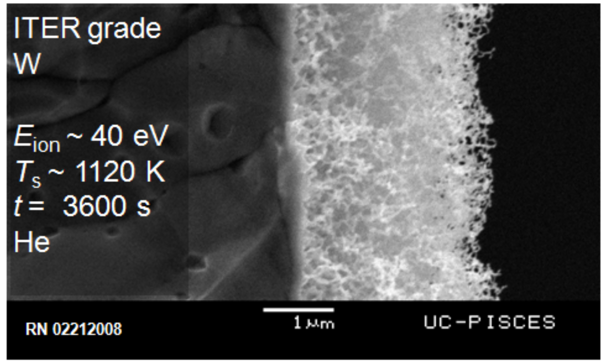

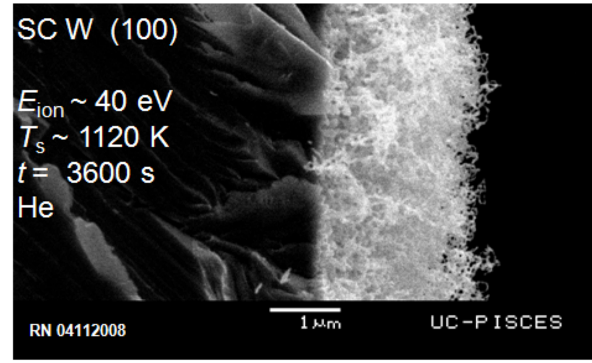

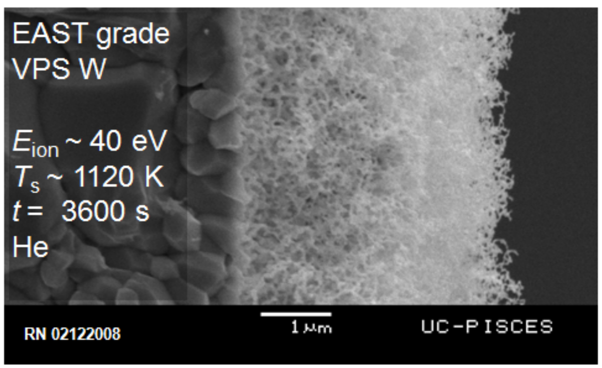

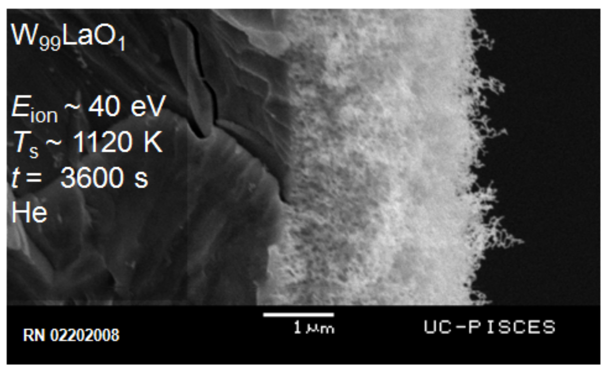

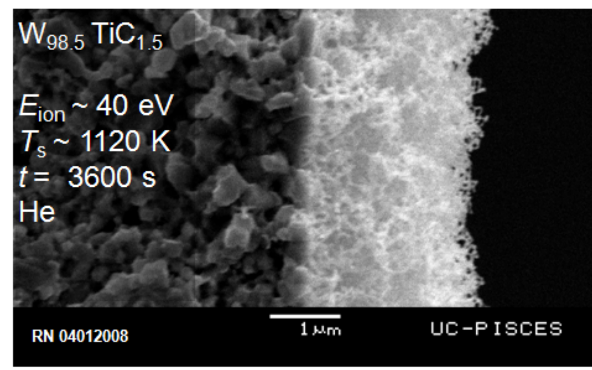

He plasma effects take place for any tungsten material

He ion effects at elevated temperatures would not be inevitable.

Plasma deposited Be and C layers completely inhibit nano-morphology at ~1150 K.

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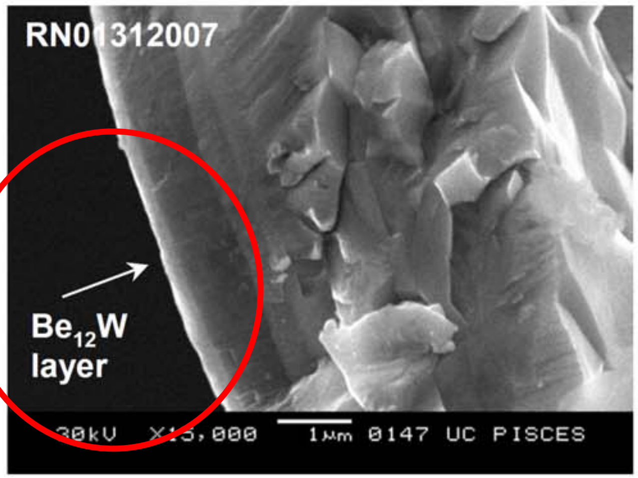
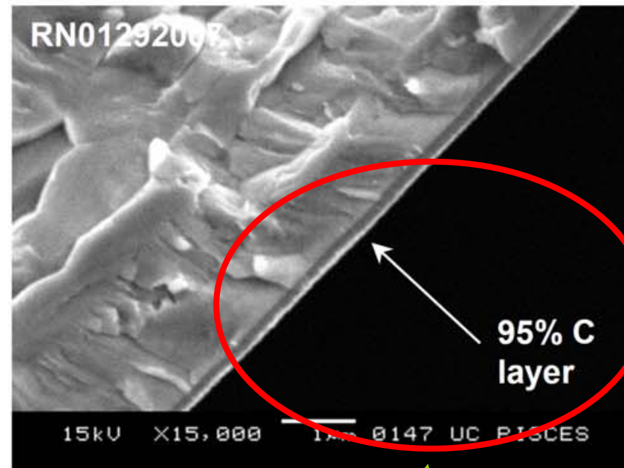
49th Annual meeting of DPP (2007)
M. Baldwin et al.

E_i = 15 eV, T_s = 1150 K, Fluence = 10^{25} He^+/m^2

D$_2$-He plasma with Be	D$_2$-He plasma with C
n_{He+}/n_e ~ 10 %, n_{Be+}/n_e ~ 0.5 %, Δt = 5000 s	n_{He+}/n_e ~ 10 %, n_{C+}/n_e < 0.1 %, Δt = 3600 s





- Surface layer composition determined by x-ray microanalysis (WDS).
- At E_i = 15 eV, Be and C deposited on W are not sputtered away.

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UCSD Jacobs | Mechanical and Aerospace Engineering

•Be-W alloy and W-C layers (< μm) inhibit He induced morphology.

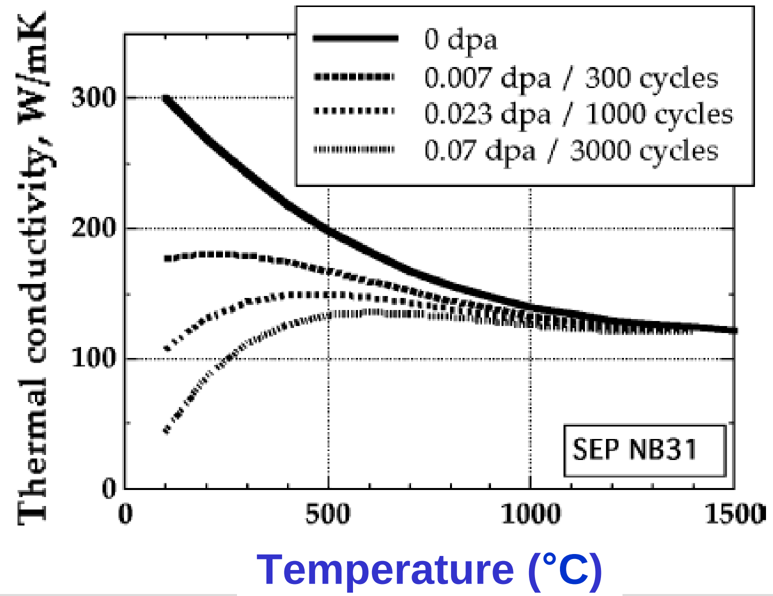
This could be the key technology to use solid wall materials (tungsten) in DEMO.

Protection of wall surface

- Surface low Z layer is effective.
- Choice of low Z material
  - Carbon: High erosion and dust formation.
  - Beryllium : Relatively low melting point (1278 °C). Mixed layer formation with W, leading to enhanced erosion of tungsten.
  - Boron : not easy to form thick coating due to brittleness
  - Boron would be the candidate, but needs more investigation.
    - This idea was originally proposed by N. Noda, then C. Wong.
- Deposition area control and dust collection
  - Deposition control and regular dust collection (if any) would be needed.

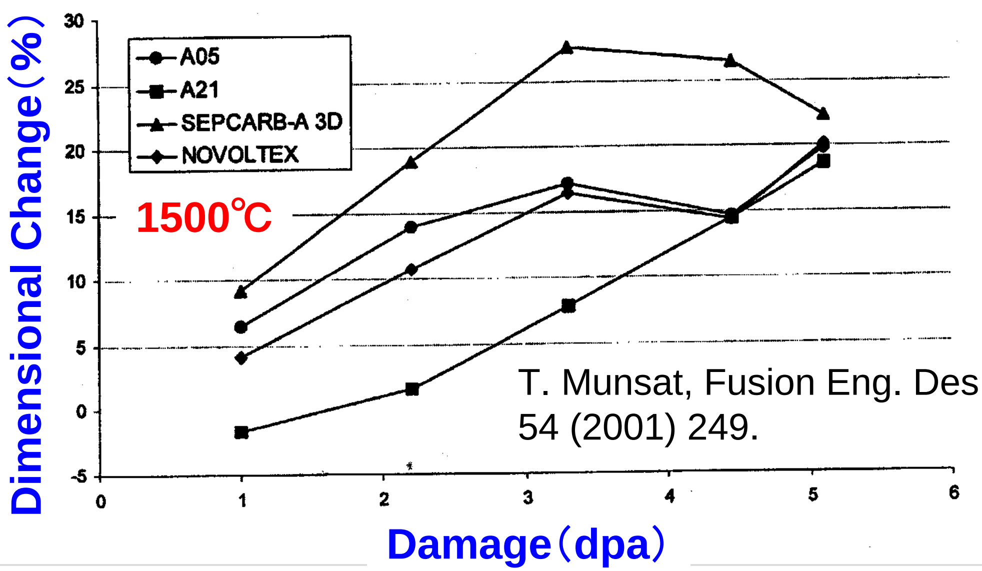
Neutron effects on graphite

V. Barabash et al.,
J. Nucl. Mater. 313-316 (2003) 42.



- Reduction of Thermal conductivity
  - Degradation is remarkable at low temperatures.
  - No significant changes above 1000 °C.

- Dimensional change
  - Recoiled atoms make additional graphite planes.
  - Carbon fibers expand radially and shrink axially.
  - This effect is increased with temperature.



T. Munsat, Fusion Eng. Des. 54 (2001) 249.

Neutron Effects of tungsten

- Neutron irradiation effects
  - Increase in DBTT (Ductile Brittle Transition Temperature)
  - Reduction in thermal conductivity due to lattice damage
  - Void swelling
  - Increase in T trapping
    - Not significant at elevated temperatures (>600 °C)
- 14 MeV neutron effects should be studied.
  - Almost no data for tungsten irradiated by 14 MeV neutron.
  - Helium production becomes significant at this energy.
- High dose effect (> 10 dpa) should be studied.
  - Transmutation (W → Re → Os) is not negligible for DEMO reactors.
    - Increase in Re concentration reduces thermal conductivity.

Endurance test of PFCs for years of operation

- What kind of test conditions are needed?
  - Complicated conditions for divertor
    - Heavy irradiation by 14 MeV fusion neutron
      - radiation damage, transmutation, He production
    - High heat flux →Thermal stress (irradiation creep?)
    - High fluence He (&D,T) ion irradiation from edge plasmas
  - ITER engineering phase can provide opportunity for studying high ion fluence effects. But neutron fluence is not enough.
- What is the most realistic method for the test?
  - CTF-like device, IFMIF with plasma sources, using first phase of DEMO, or something else?

New tungsten development

- Preferable property for tungsten
  - High toughness and high yield strength at elevated temperature
  - High recrystallization temperature
  - Negligible increase in DBTT by neutron irradiation
- UFG-W (Ultra Fine Grained W) Dr. Kurishita (Tohoku Univ.)
  - High recrystallization temperature
  - Highly resistant for neutron irradiation
  - Preferable properties under high flux plasma exposure
    - almost no D blistering, observed for PM (powder-metallurgy) tungsten
    - no enhancement of D retention
      - Latest results from the high density plasma device (UCSD)
- Development of fabrication technique of mono-block size UFG-W is planned.

UFG-W : high resistance to neutron irradiation

H. Kurishita, et al., J. Nucl. Mater. 377 (2008) 34.

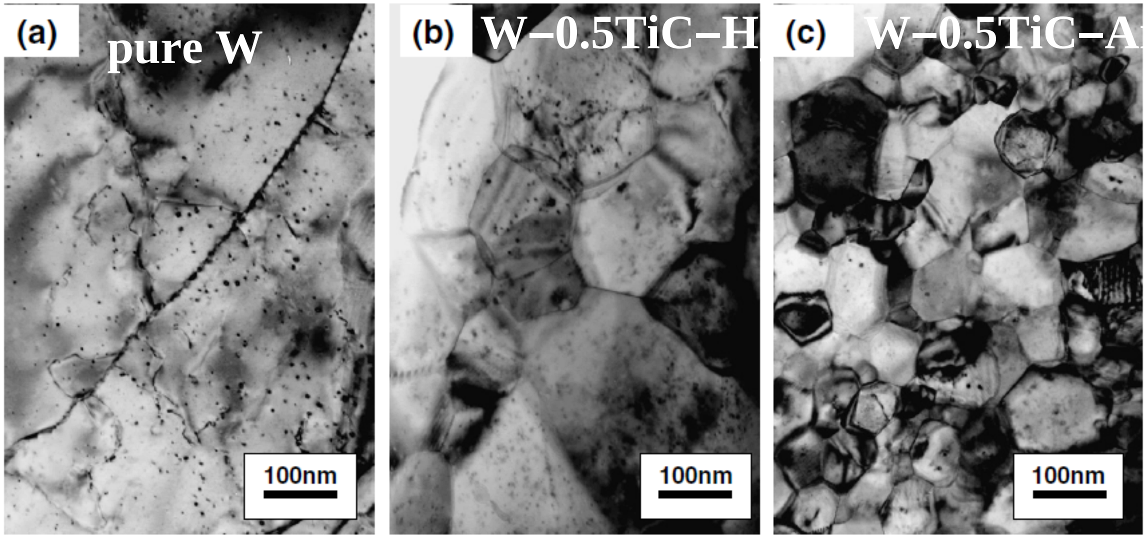


Fig. 7. TEM bright-field images of microstructures in (a) pure W, (b) W–0.5TiC–H_2 and (c) W–0.5TiC–Ar after neutron irradiation at 873K to 2 × 10^{24} n/m^2 in JMTR.

It also showed less neutron induced damage (black dots in photo).

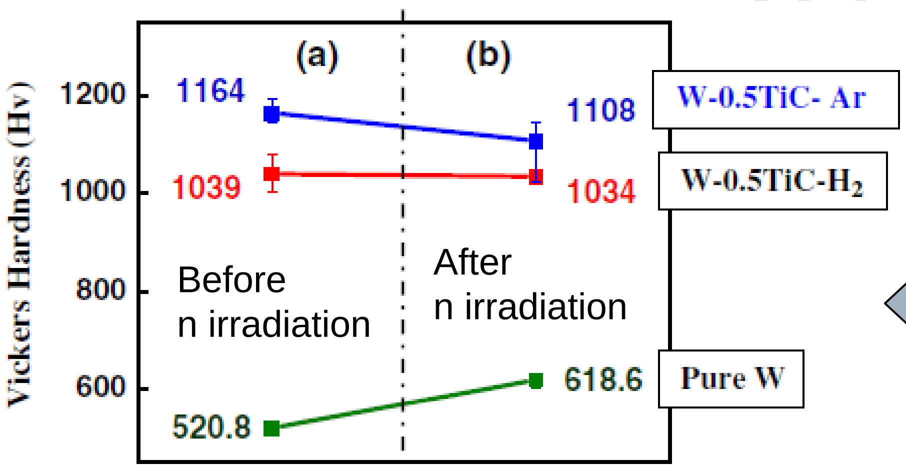
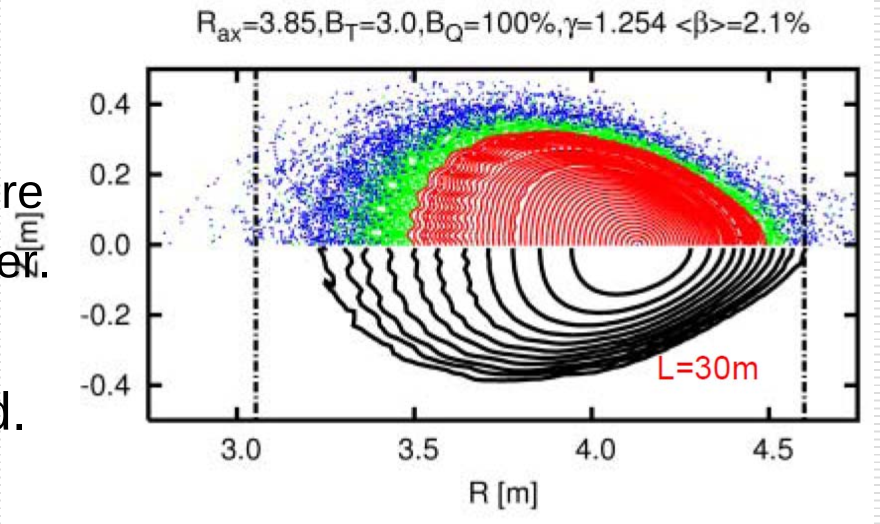


Fig. 9. Vickers microhardness number before and after neutron irradiation for pure W, W–0.5TiC–H_2 and W–0.5TiC–Ar.

UFG-W showed less hardening than pure W by neutron irradiation.

PWI related issues for DEMO in Helical system

- Avoidance of impurity accumulation in core plasmas
  - Erogodic layer will play a major role for impurity screening.
    - Impurity outward flux from the core should not be blocked by this layer.
  - High confinement mode without impurity accumulation is required.



- Power and Particle control
  - Reduction of heat load to less than ~10 MW/m^2 and effective He ash removal should be realized simultaneously.
  - Can cooling gas (Ar, Ne) puffing be avoided for reduction of erosion?
- Erosion and redeposition control
  - Reduction of net erosion
  - Low-Z protective layer formation by deposition control

Impurity Hole in LHD

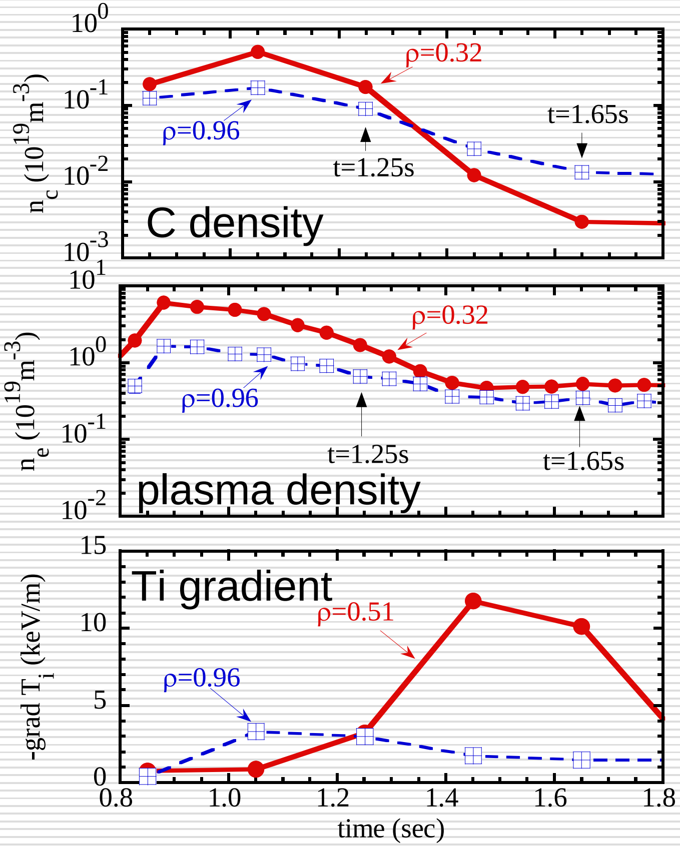
M. Yoshimura, S. Masuzaki (NIFS)

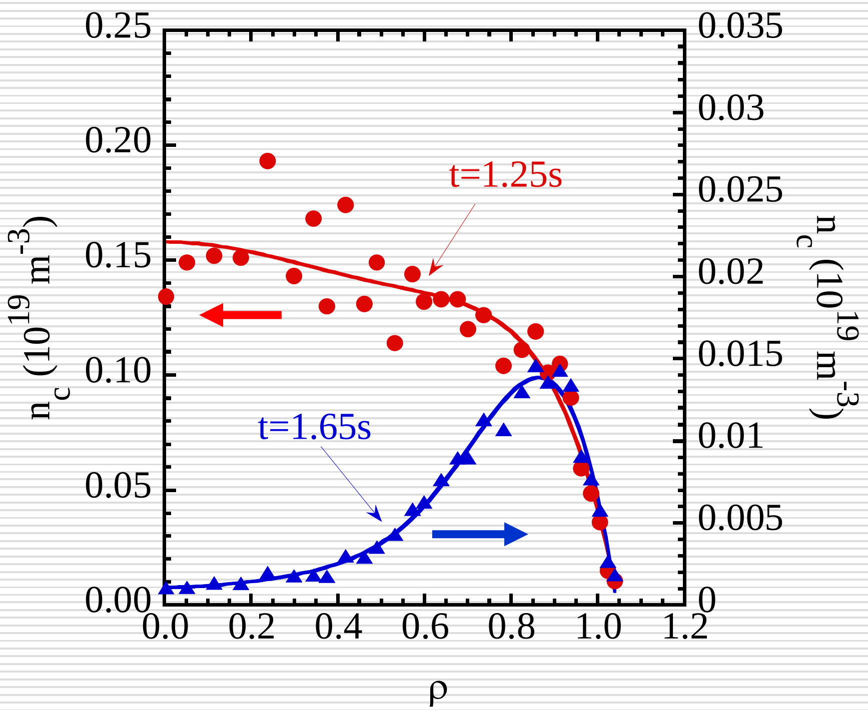
A carbon pellet (a size of f1.4x1.4mm) is injected at t=0.8sec into NBI sustained plasma.

The gradient of Ti increased during the decay phase of the density.

The carbon density near the plasma core decreases faster than that at the edge.

Extremely hollowed profile of carbon impurity is produced





Carbon density profile of LHD

Summary

- Material issues for DEMO
  - Graphite (CFC)
    - Erosion and dust formation, material degradation by neutron, T retention
  - Tungsten
    - Material degradation by He and neutron, avoidance of core accumulation
    - Surface protection by low Z coating could be effective.
- 14 MeV neutron effects must be studied.
  - Testing of PFC’s must be made under year-long complicated conditions.
    - High heat flux, heavy neutron irradiation, and high plasma ion flux.
- In helical system, PWI issues are important for DEMO
  - Avoidance of impurity accumulation in high confinement plasmas
  - Power and particle control
  - Erosion and redeposition control