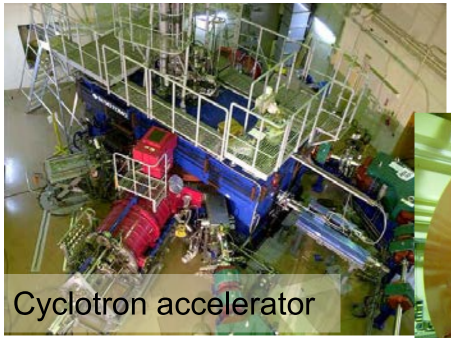
Tungsten IEA Meeting, ICFRM-17 Aachen
Monday, October 12, 2015 Konf 4

Activities of W-material Development, Evaluation and Irradiation in Japan

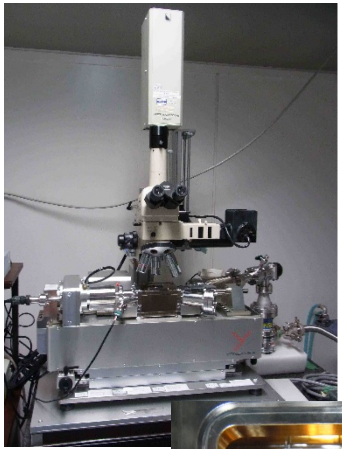
A. Hasegawa Tohoku Univ.

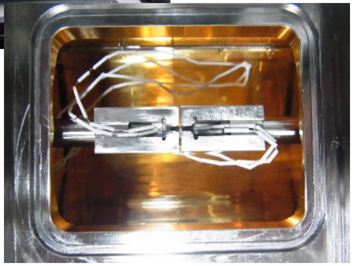
Activity for Fusion Reactor Materials in Tohoku University

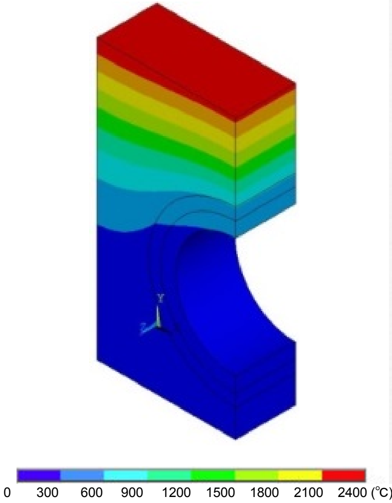
- Tungsten material development for plasma-facing components
- Irradiation effect on fusion reactor materials (i.e. RAFM and W)
- Analysis of the structural response of fusion reactor components by computer simulation
- Development of the residual life prediction and mechanical property investigation technology for fusion and fission reactor component materials

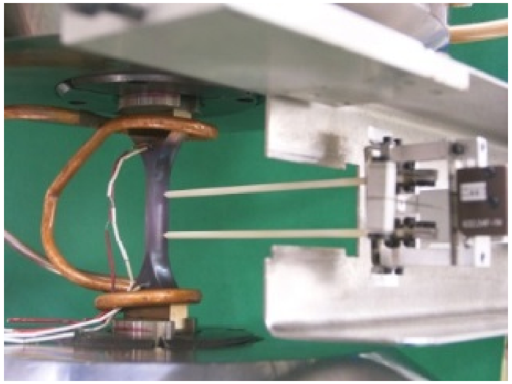

Cyclotron accelerator


Dynamitron accelerator



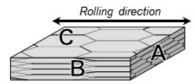




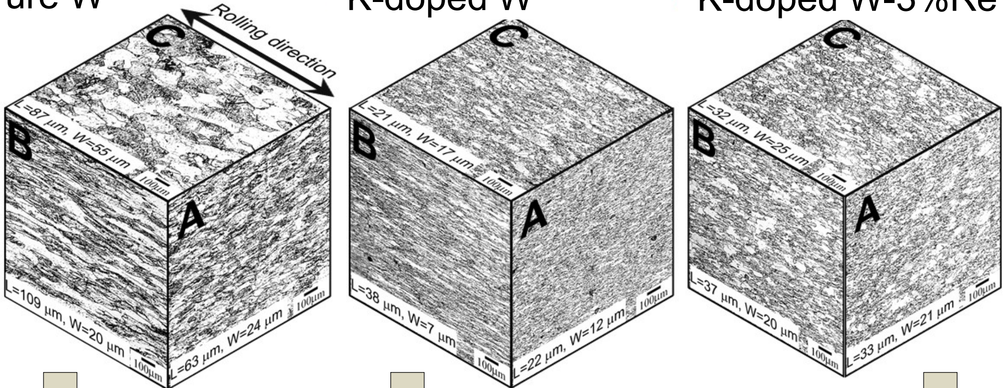


Tungsten Material development

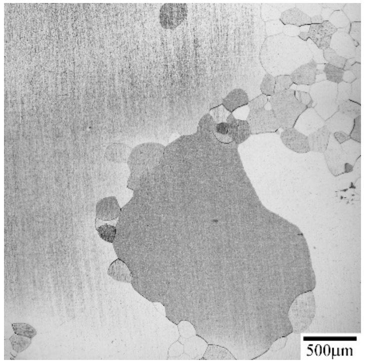
The W alloys which shows higher recrystallization temperature and tensile strength than pure W had been developed. The W-alloy development is supported by NIFS.

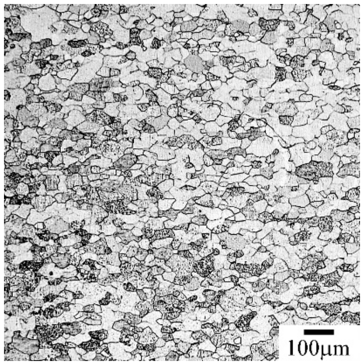


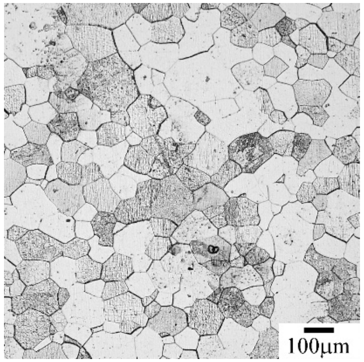
Pure W　　K-doped W　　K-doped W-3%Re



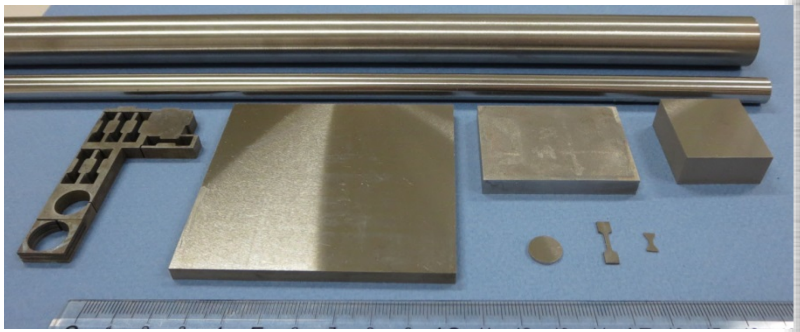
After 2300℃　1h



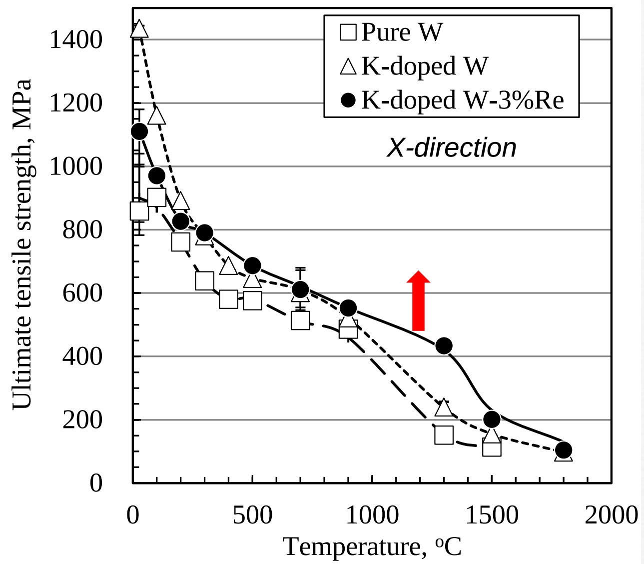




M. Fukuda, to be published



Powder metallurgical processed materials



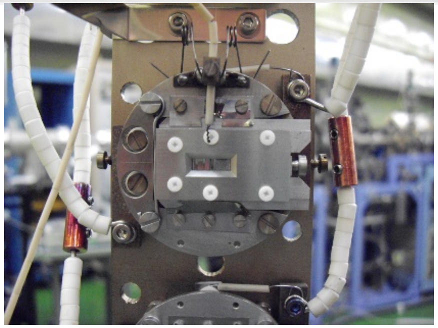
M. Fukuda et al, Fus. Sci, Technol., 68 (2015)

Irradiation Effect

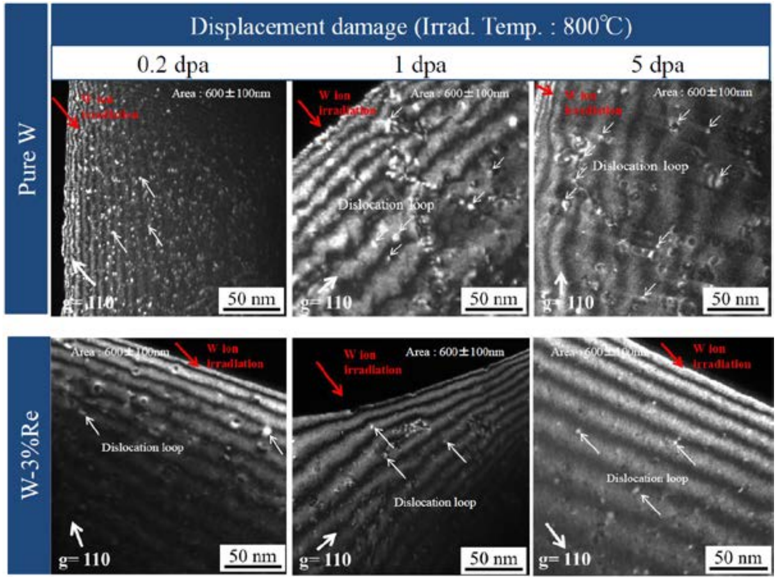
The effects of neutron and ion irradiation on the microstructure and mechanical property in W materials

Examined materials

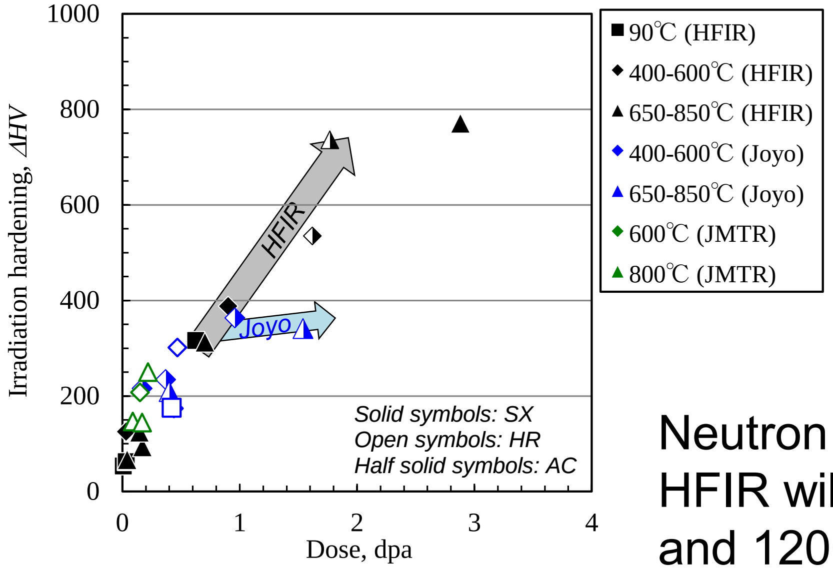
- Pure W
- K-doped W
- W-3%Re
- K-doped W-3%Re
- La-doped W-3%Re



W-ion irradiation at 500 ℃and 800℃.
Dpa: 0.2,1 and 5dpa



T. Hwang, Po 3-31



M. Fukuda, Po 1-65

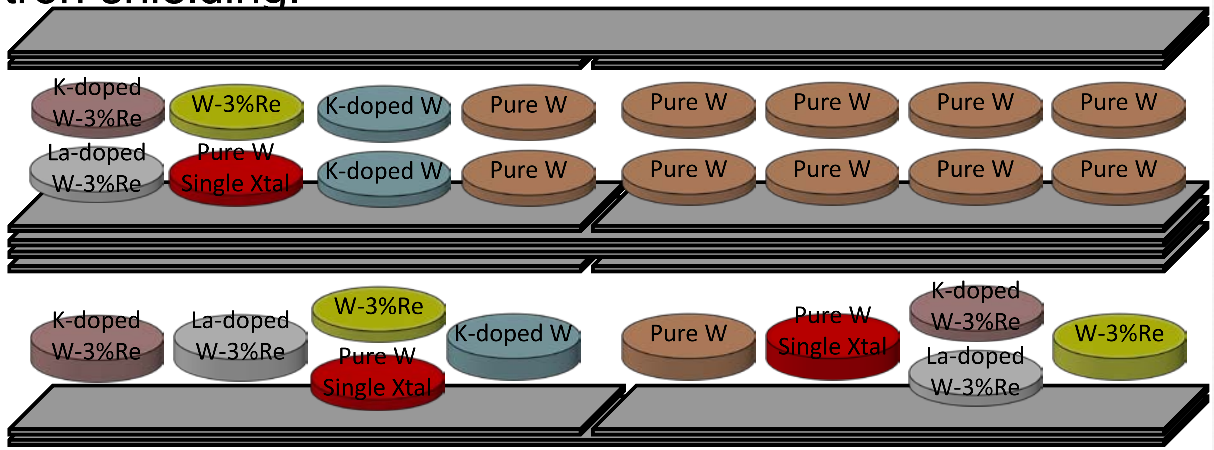
Neutron irradiation experiment in HFIR will be performed at 500, 800, and 1200 ˚C to ~1.5 dpa

PHENIX Project

2013-2018 PHENIX JAPAN-US JOINT PROJECT

TOHOKU UNIVERSITY

Neutron irradiation to various tungsten materials was performed in HFIR rabbit capsules without thermal neutron shielding.



Examined materials

- Pure W
- K-doped W
- W-3%Re
- K-doped W-3%Re
- La-doped W-3%Re

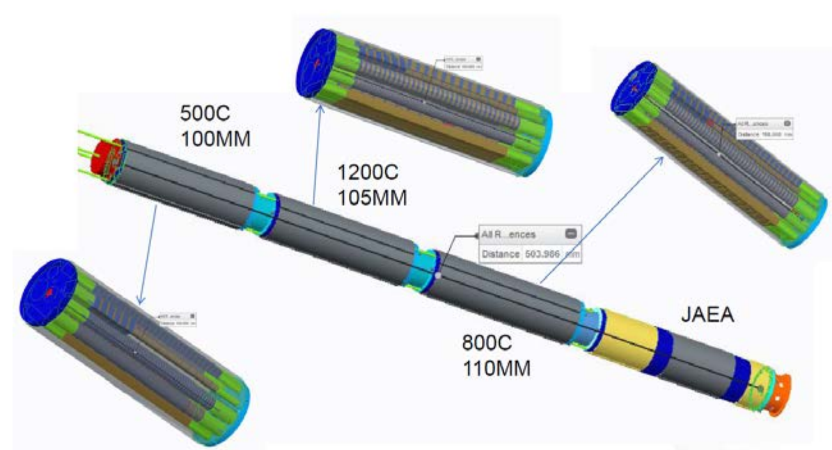
The neutron irradiation in HFIR with thermal neutron shield to simulate the fusion reactor irradiation condition will be started in this FY.

Specimen types:

Small tensile specimen (SS-J)

3mmΦ disk, 10mmΦ disk

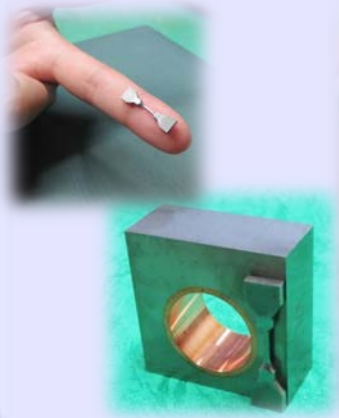
Capsule Design Layout

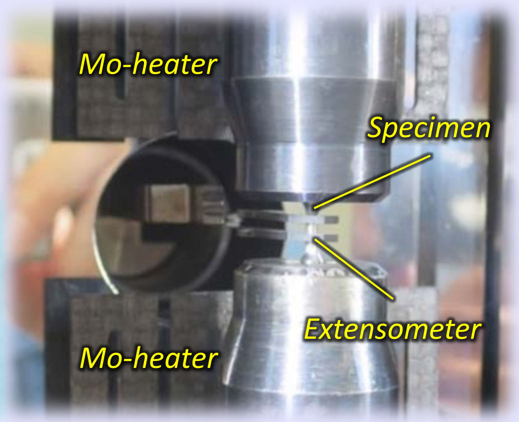


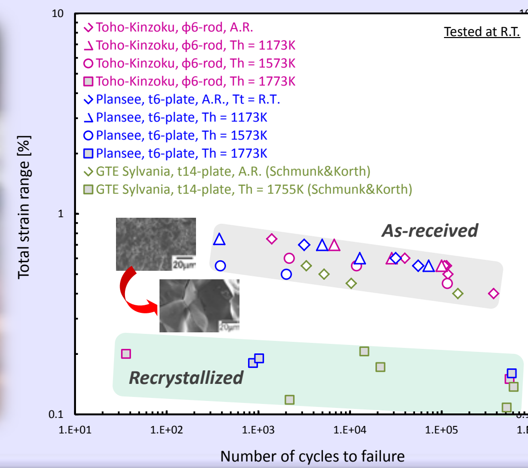
Development of Fatigue Test Technology

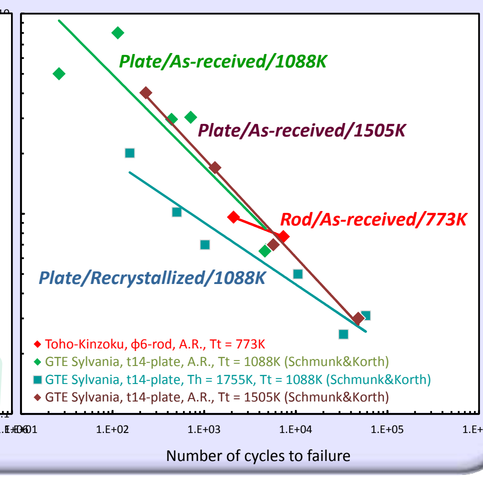
S. Nogami, Parallel session 6B, Thursday, 8:30~

High Temp. LCF Test Technology using Small Specimen for Reliable Fatigue Life Database of W and CMC

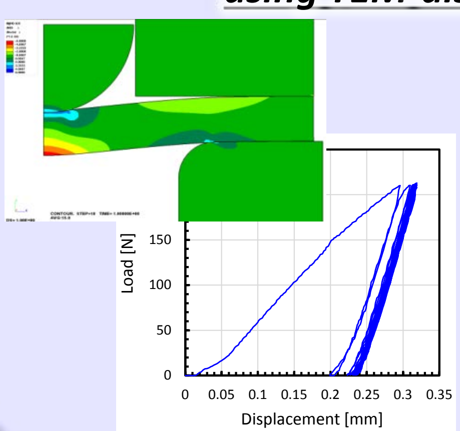


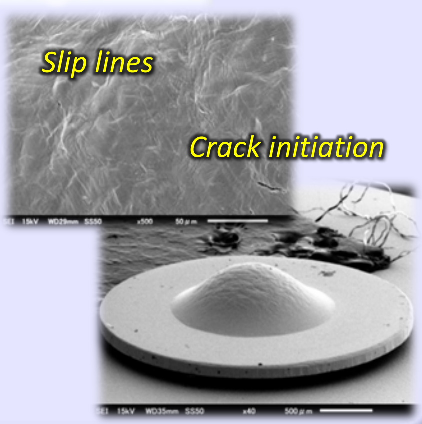




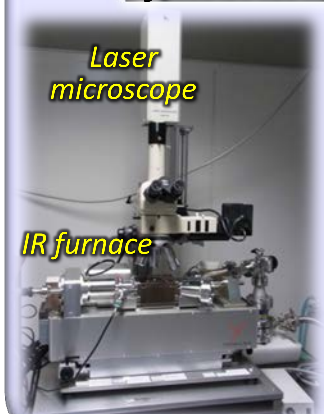


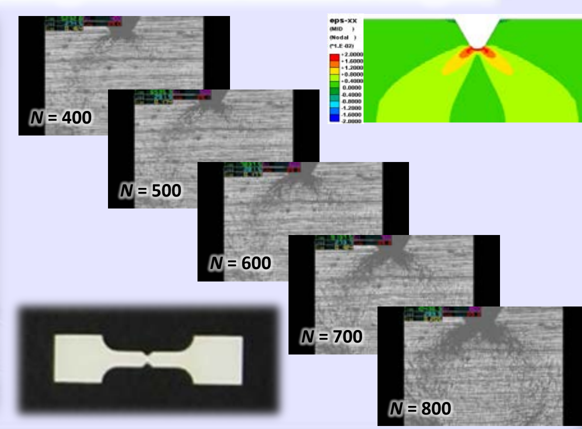
Small Punch Fatigue (SPF) Test Technology using TEM-disc Specimen





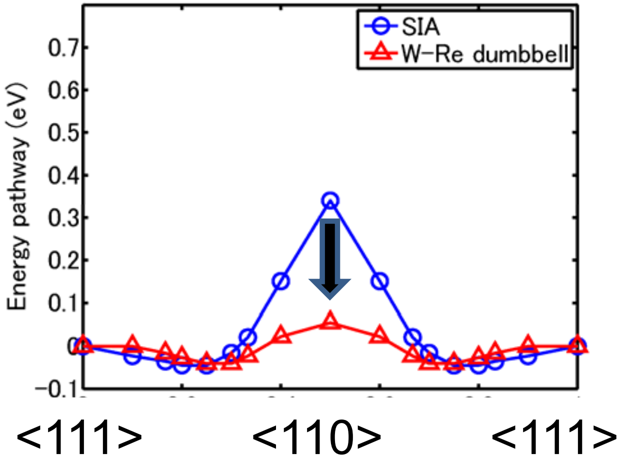
In-situ Observation Technology of Material Degradation under Fatigue



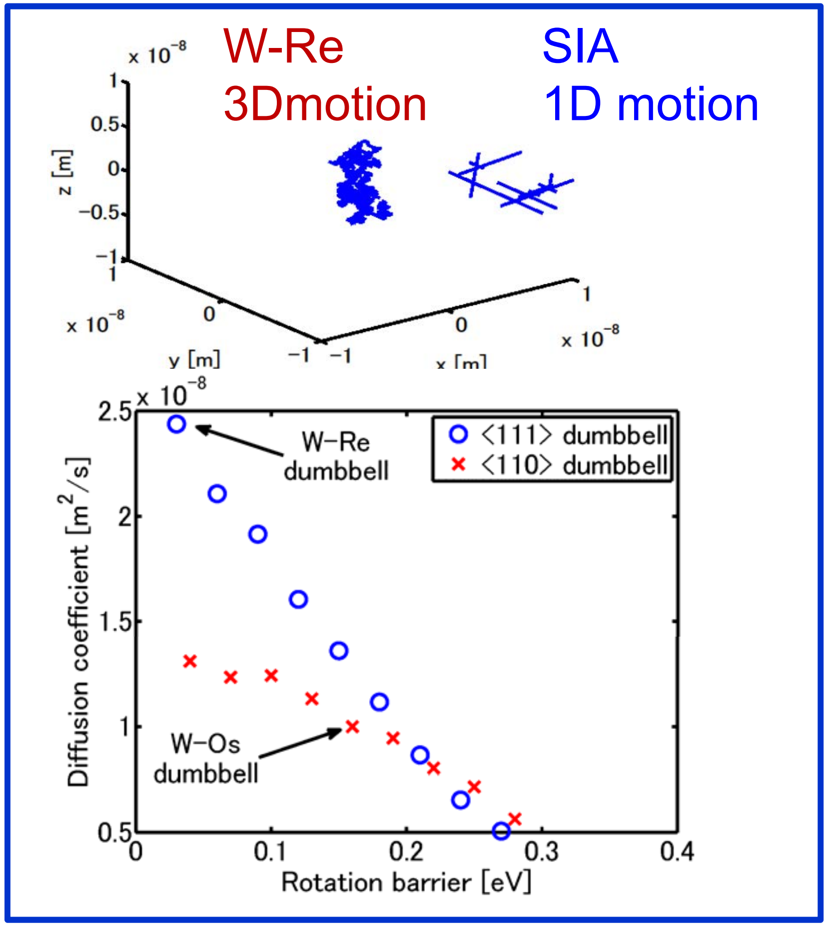


KMC simulation of Re and Os interstitials in W (T. Suzudo, Poster 4-64 on Thursday)

Ab initio results



KMC results



Highlights

Low rotation barriers of W-Re/Os mixed dumbbells cause 3D motion and large diffusivity, which probably promoting precipitation.

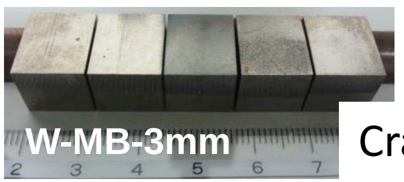
T. Suzudo, M. Yamaguchi, A. Hasegawa, Modeling Simulation Mater. Sci. Eng. 22 (2014) 075006; J. Nucl. Mater. (2015) accepted.

W R&D activities in IAE, Kyoto University

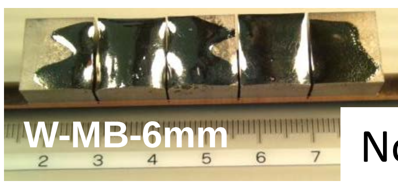
1. Damage analysis of high-heat loaded W-MB (Okunishi)
2. Effects of recrystallization on the DBTT of pure W (Kobayashi)
3. Ion-irradiation effects on pure W (Zhang)

1. W-MB (from JAEA)

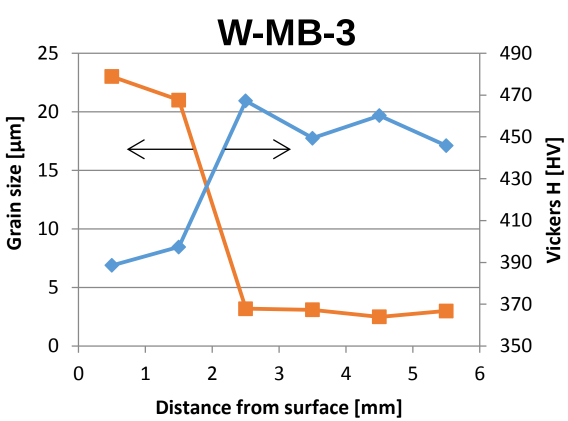
～30MW/m², 1000 cycles



Cracks

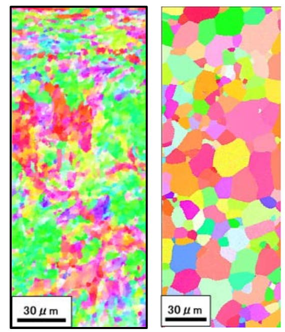


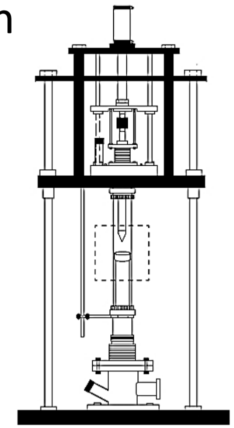
No crack

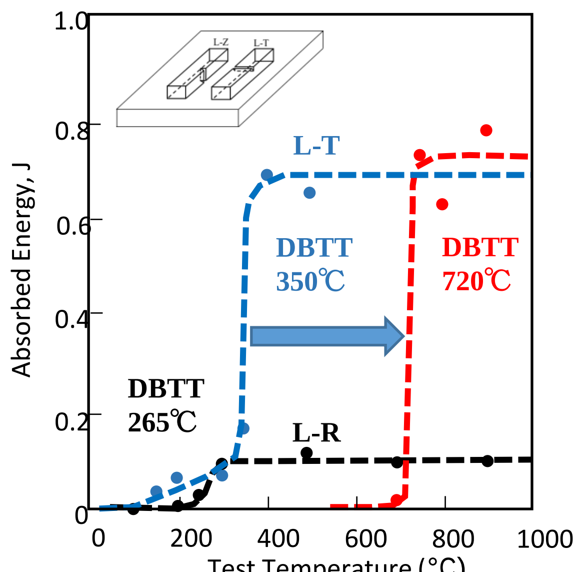


2. Recrystallization

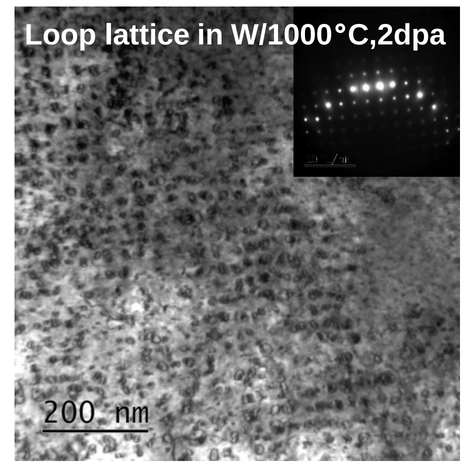
(1400°C, 1 hr)

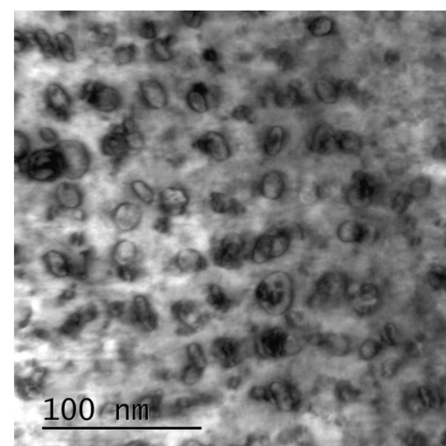






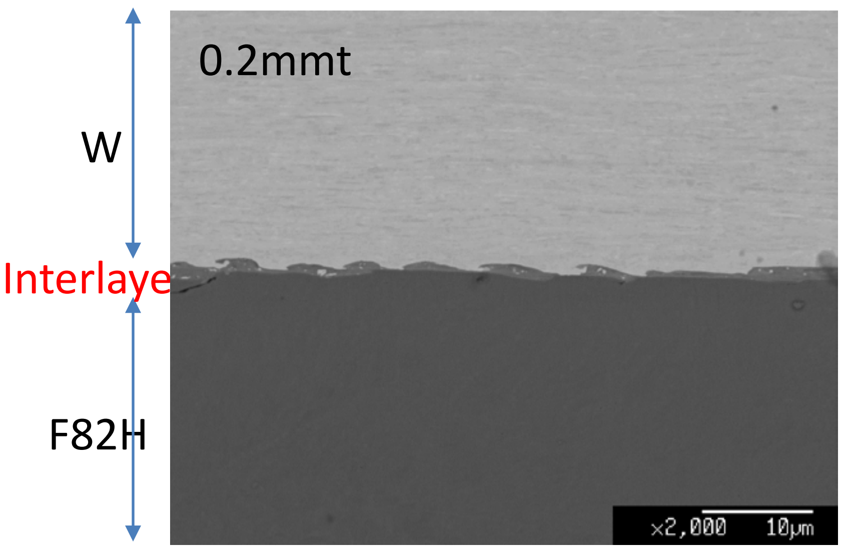
3. Ion-irradiation effect



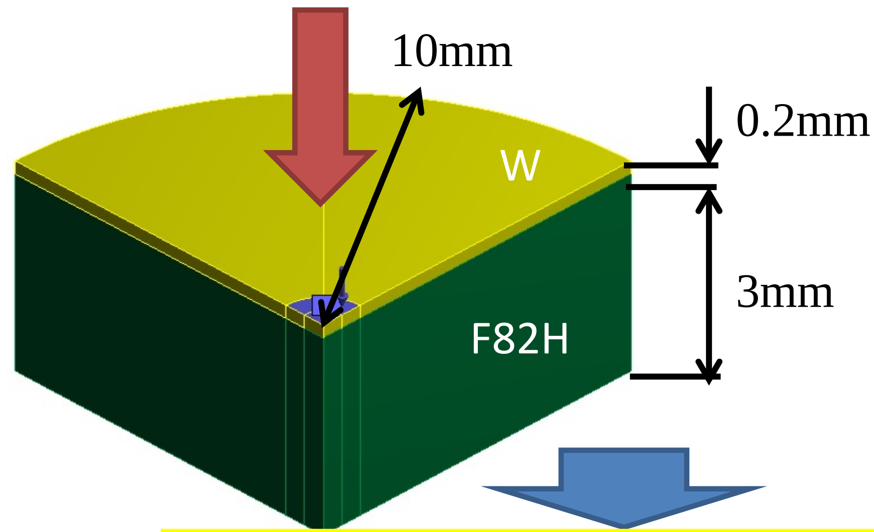


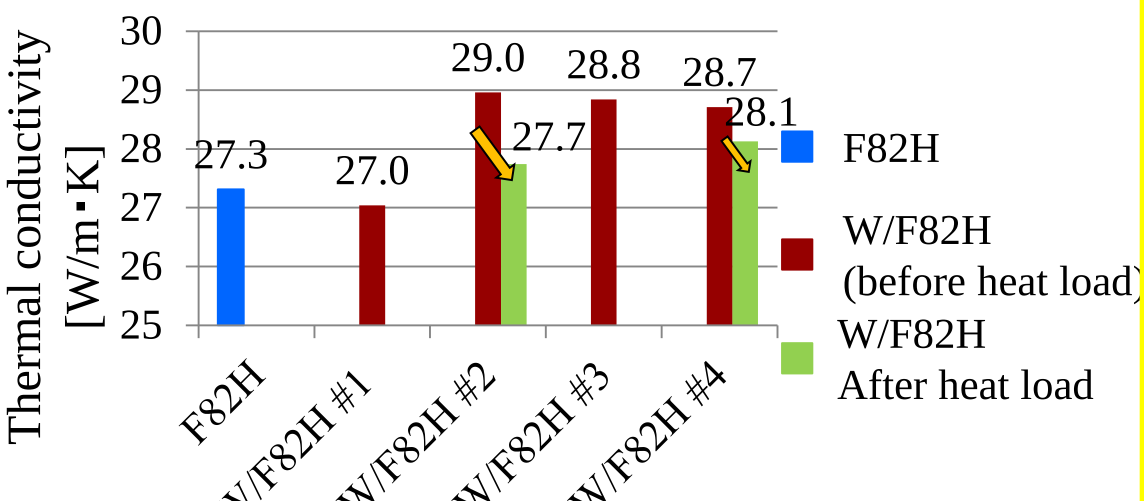
Laser heat load tests on underwater explosive welding of W coated on F82H (Kyoto U. & Kumamoto U.)

W/F82H



YAG Laser (1.2mmϕ) with 29.6MW/m²





Laser irradiation (4cycle up to 550°C at F82H) reduced the thermal conductivity slightly but crack was NOT observed.

Microstructure and D retention of W exposed to Mixture Plasma

M. Miyamoto et al., Nucl. Fusion 49(2009)065035, JNM 415(2011)S657, JNM 438(2013)S216

- He seeding to D plasma causes suppression of blister formation and significant reduction in D retention.
- The co-deposits with high ratio of D/Be (~0.03) is obtained at the Be deposition condition.
- D retention is suppressed again at Be erosion condition, regardless of He seeding.