Japan-US Joint Project

Technological Assessment of Plasma Facing Components for DEMO Reactors

PHENIX

PFC evaluation by tritium Plasma, HEat and Neutron Irradiation eXperiments

2013~2018

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Background (1)

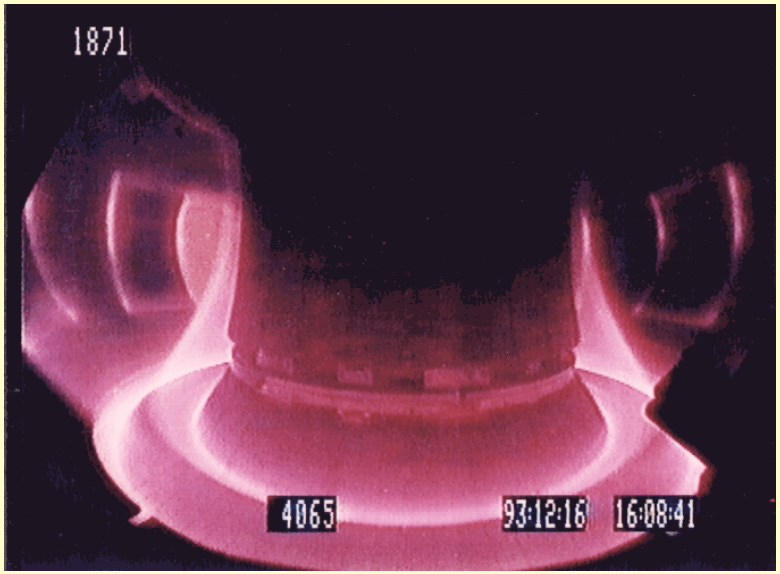
Plasma-facing materials (PFMs) of fusion reactors will be exposed to extremely high heat flux, high energy neutrons and ions of deuterium, tritium and helium.

$$D + T \rightarrow {}^{3}He\ (3.5\ MeV) + n\ (14\ MeV).$$

Because of high melting point, high heat conductivity and low erosion rate by hydrogen isotopes, W-based materials are recognized as primary candidates of PFMs.

Carbon has been used extensively as PFMs in short-pulse DD devices. However, carbon cannot be used in DT fusion rectors because of insufficient tolerance to 14 MeV neutron irradiation and high T retention.

Fusion community has selected W as a new PFM candidate and been studying properties of W.



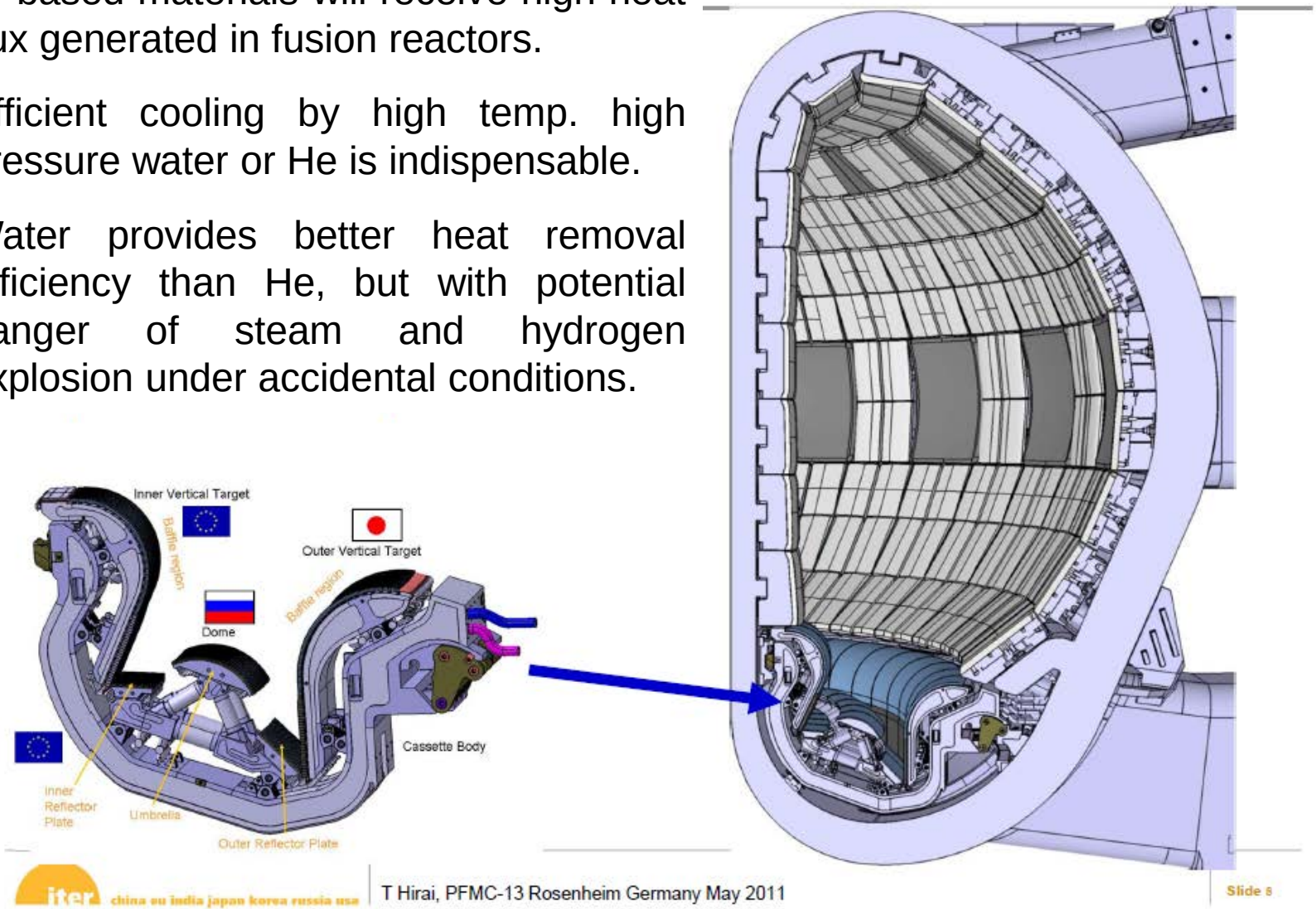
ASDEX-U Tokamak under discharge
(IPP Garching, Germany)

Background (2)

W-based materials will receive high heat flux generated in fusion reactors.

Efficient cooling by high temp. high pressure water or He is indispensable.

Water provides better heat removal efficiency than He, but with potential danger of steam and hydrogen explosion under accidental conditions.



Cross-section of ITER vacuum vessel

Summary of the project

The goal of this project is to evaluate the feasibility of He gas-cooled divertor with tungsten material armor for DEMO reactors. Main research subjects are listed below;

1. Heat transfer mechanism and modeling in He-cooled systems, improvement of cooling efficiency and system design.
2. Response of tungsten layered materials and advanced tungsten materials to steady state and pulsed heat loads.
3. Thermo-mechanical properties measurement of tungsten basic materials, tungsten layered materials and advanced tungsten materials after neutron irradiation at elevated temperatures relevant to divertor conditions (500-1200 ^{o}C).
4. Effects of high flux plasma exposure on tritium behavior in neutron-irradiated tungsten layered materials and advanced tungsten materials.
5. Evaluation of feasibility (under ~10 MW/m^2 heat load with irradiation of plasma and neutrons) and safety (tritium retention and permeation) of He-cooled PFCs and clarification of critical issues for DEMO divertor design.

Structure of this Project

Task 1
（IR facility, ORNL）
Heat Load Tests
Heat Transfer
System Evaluation

Material Properties, Neutron-irr. Samples

Tritium Behavior

Task 2
(HFIR, Oak Ridge NL)
Neutron-irrad. Effects
Microstructure
Physical Properties

Neutron-irradiated samples

Task 3
(TPE, Idaho NL)
Plasma-Surf. Interac.
Tritium Behavior

Alternative facilities with similar capabilities are also acceptable.

Research Subjects for Task1, Task2, Task3

Task1 **Task2**

Property Improvement of W-alloy
Assessment of Complex
Heat Load Effect

Task2

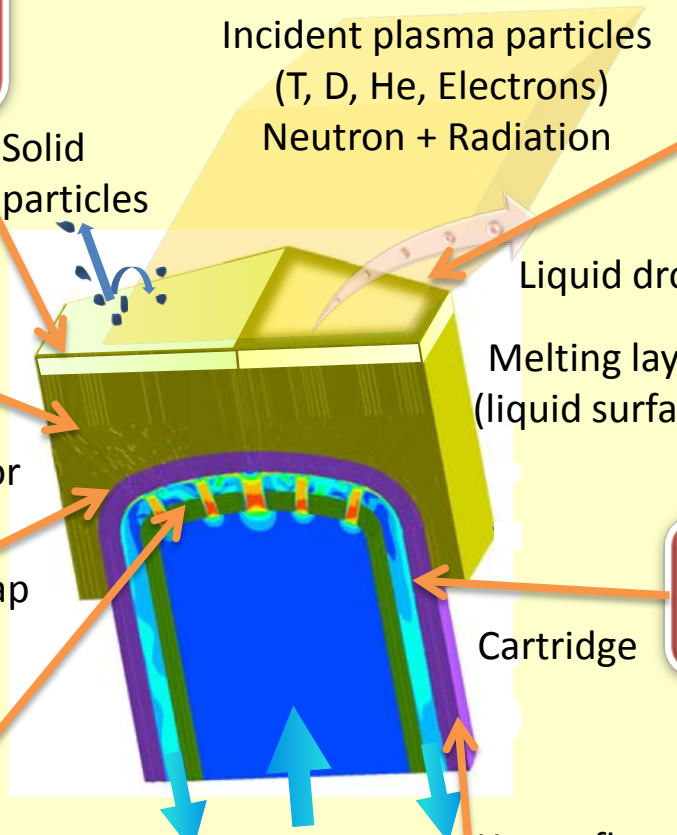
Neutron Irradiation
Effect on
Thermomechanical
Property

Task2

Development of Bonding
between Armor and
Structural Materials

Task1

Understanding of Heat
Transfer Mechanism
Optimization of Cooling
Configuration



Task3

Assessment of
Tritium Retention in
Armor Material

Task3

Assessment of Tritium
Permeation to Coolant

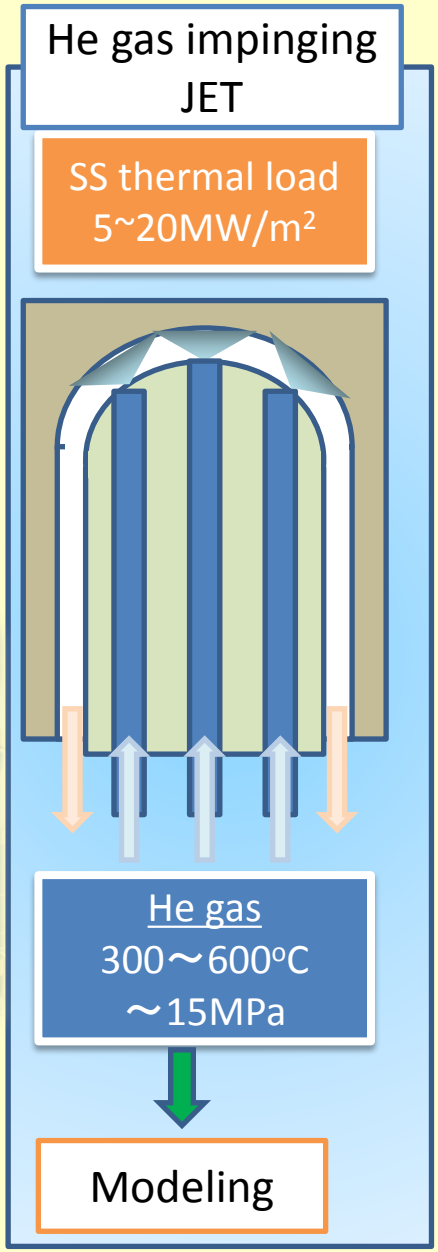
Task1

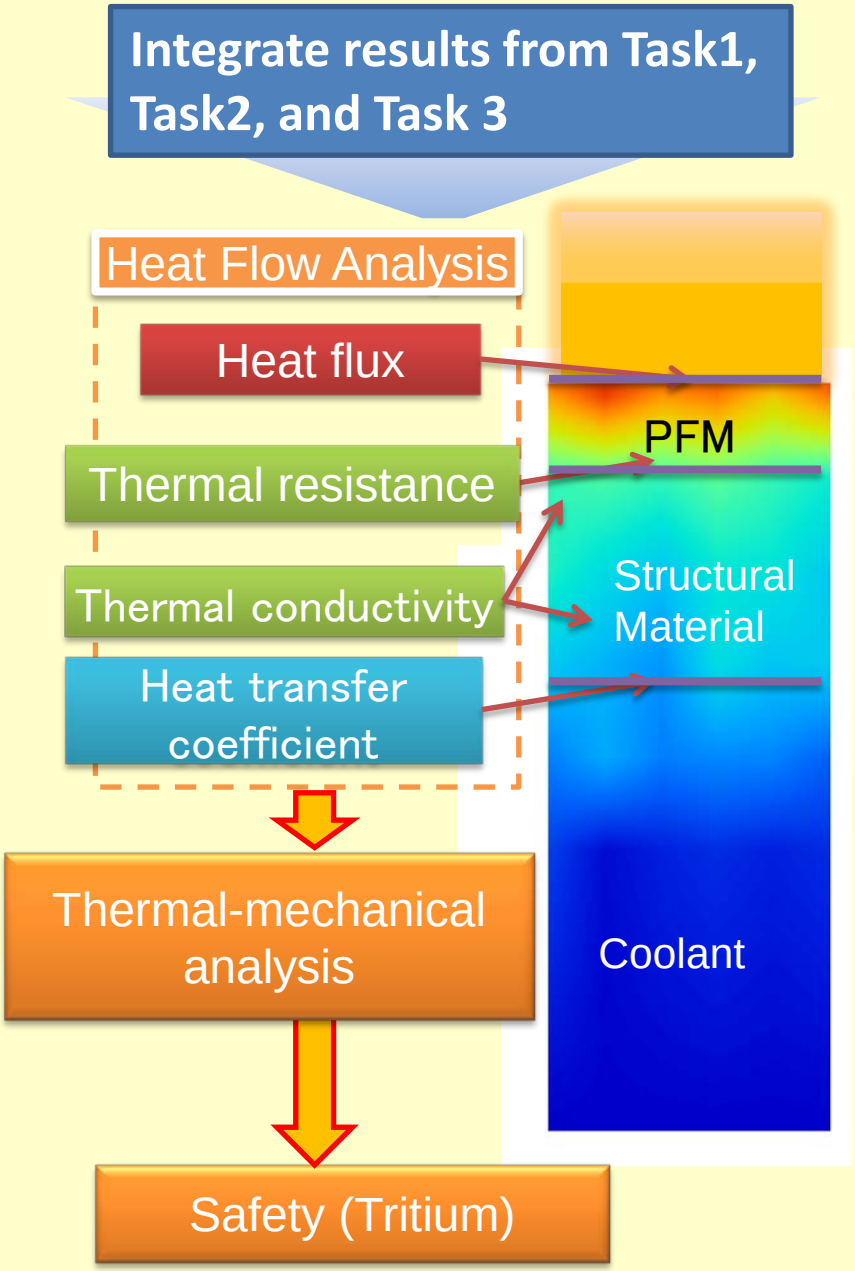
Assessment of Soundness of Sealed
Structure with High Pressure/High
Temperature Helium

Task 1 details

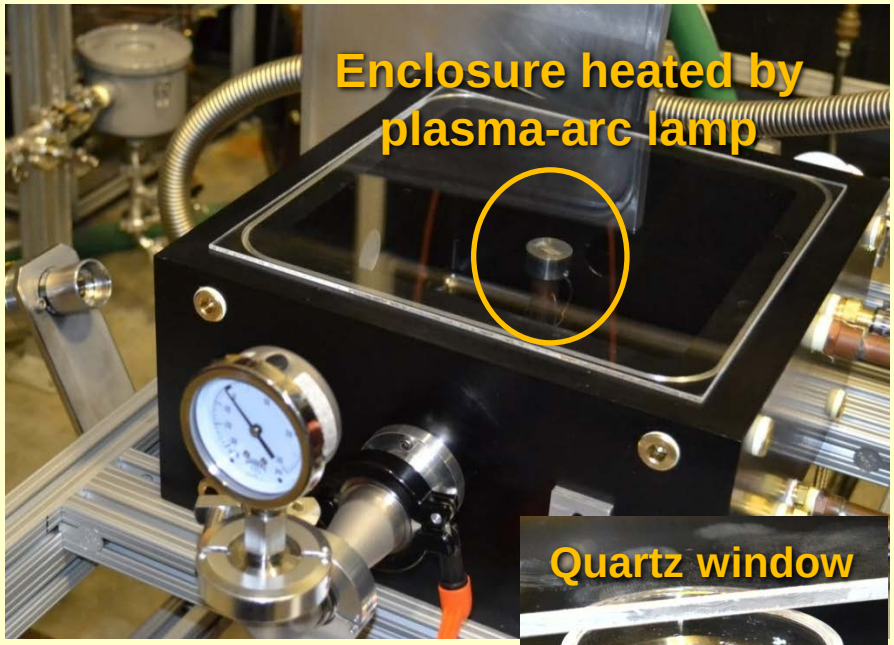
Plasma Arc Lamp in ORNL for heat load tests

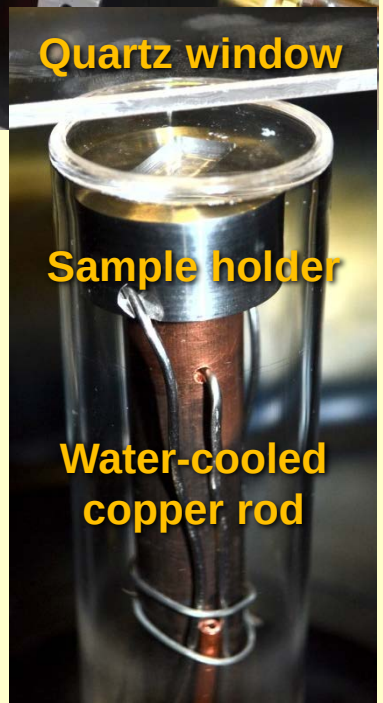
He loop in GIT for heat transfer tests





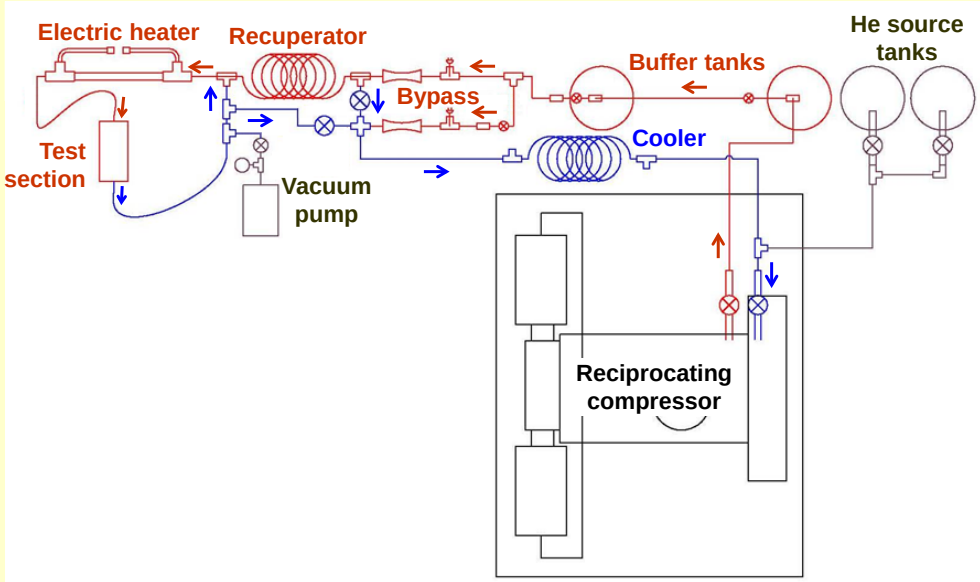
Task 1 2013-2014 Highlights





Design and fabrication of test section for heat load tests of neutron-irradiated specimens.

Heat load tests of non-irr. Specimens.



Upgrading of GIT He loop test sections to increase available temperature and heat flux.

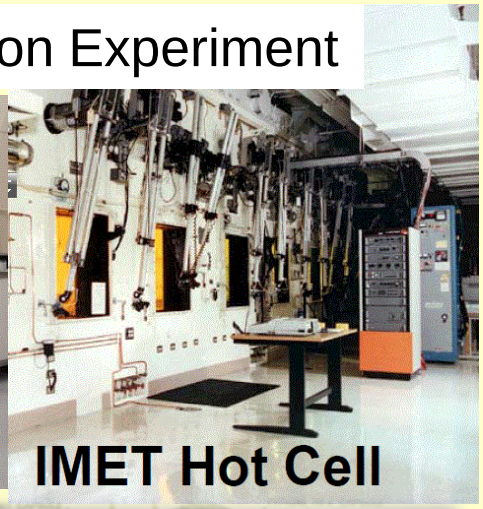
Direct numerical simulation of single impinging jet heat transfer was performed.

Task2 details

Material selection and modification

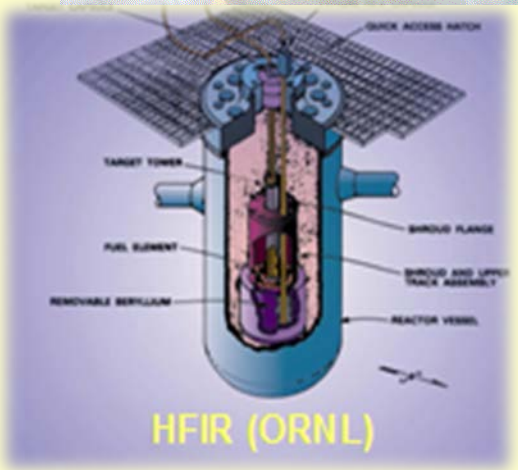
- W based material, advanced W material including alloy, surface treatment and composites: developed by US and JP
- Layered material: led by JP
- Information exchange and feedback for development

Post Irradiation Experiment



Neutron Irradiation

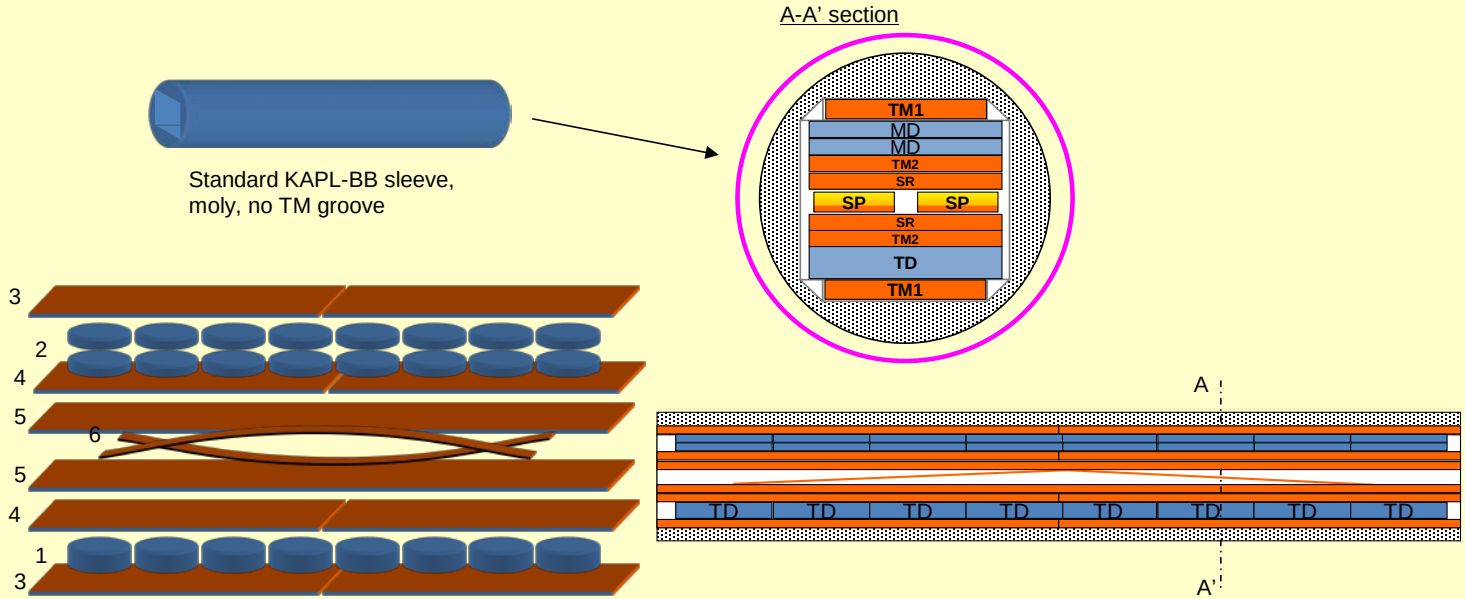
- Temperature: 500℃～1200℃, Fluence: ～1.5 dpa
- ~0.3dpa irradiation w/o thermal neutron shield
- ~1.5dpa irradiation with thermal neutron shield
- Feedback to material development
- High fluence（～15 dpa）: Utilizing ion irradiation



Post irradiation experiment

- LAMDA, IMET: thermal conductivity, toughness, strength, microstructure
- IR Facility: hear load
- Oarai, Japan: supporting

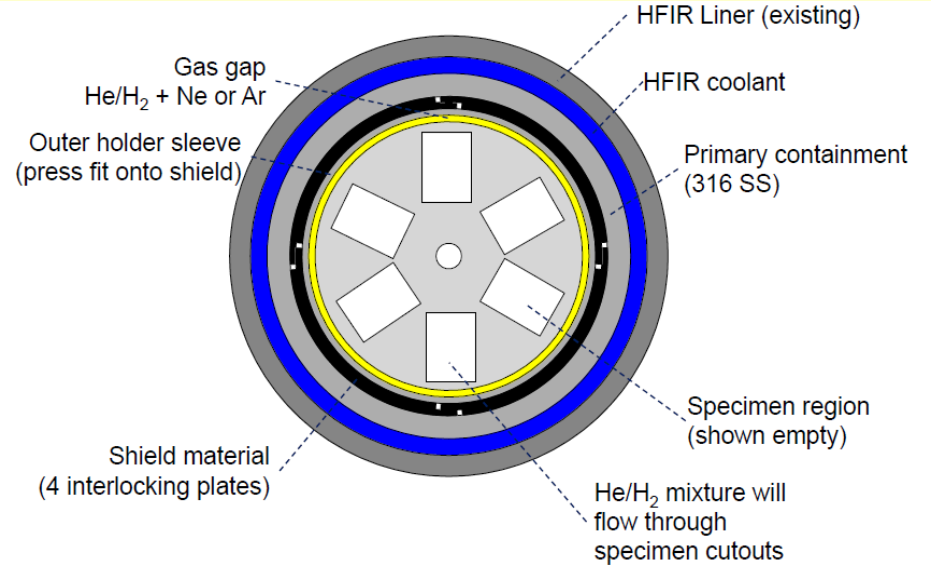
Task 2 2013-2014 Highlights



Design of low-dose rabbit capsule has been completed and irradiation will start Oct., 2014.

Design of high-dose irradiation capsule with thermal neutron shield is in progress.

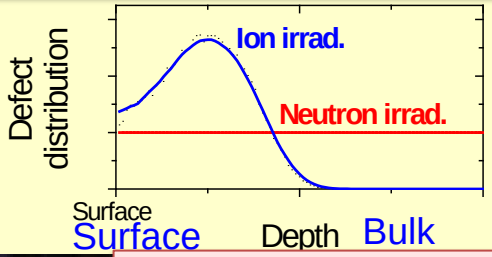
Development of PIE techniques is also in progress with non-irr. specimens.

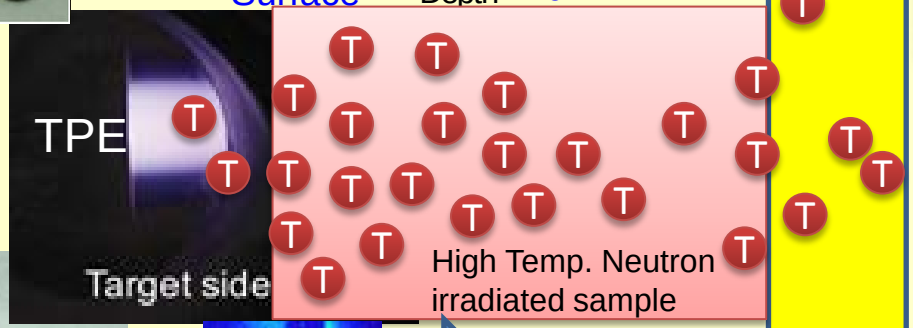


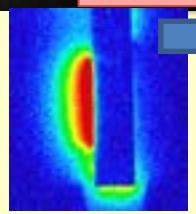
Task3 details

Neutron irradiation at higher temperature (500 – 1200 °C) is performed to simulate divertor conditions and the retention of hydrogen including tritium is also studied in high temperature region after neutron irradiation.

Neutron irradiation at HFIR and sample transport to INL

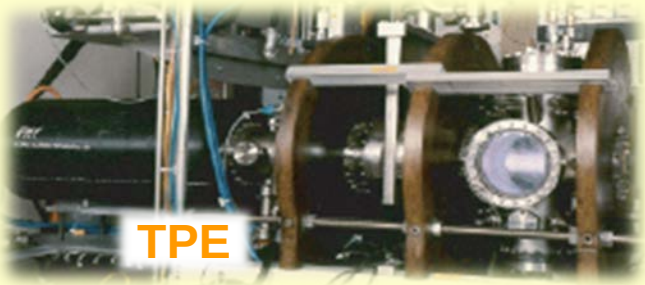




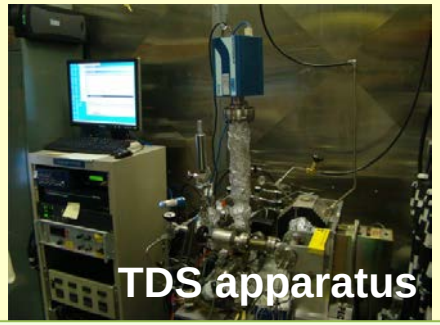


Diffusion behavior

Retention behavior (TDS)



Exposure to T plasma to measure T retention in neutron irradiated W.



- ✓ **Tritium diffusion coefficient**
- ✓ **Trap density (Tritium retention)**
- ✓ **Activation energy for desorption**
- ✓ **Recombination coefficient**

Tritium retention and permeation rate evaluation by modeling and simulation (TMAP etc)

Safety analysis

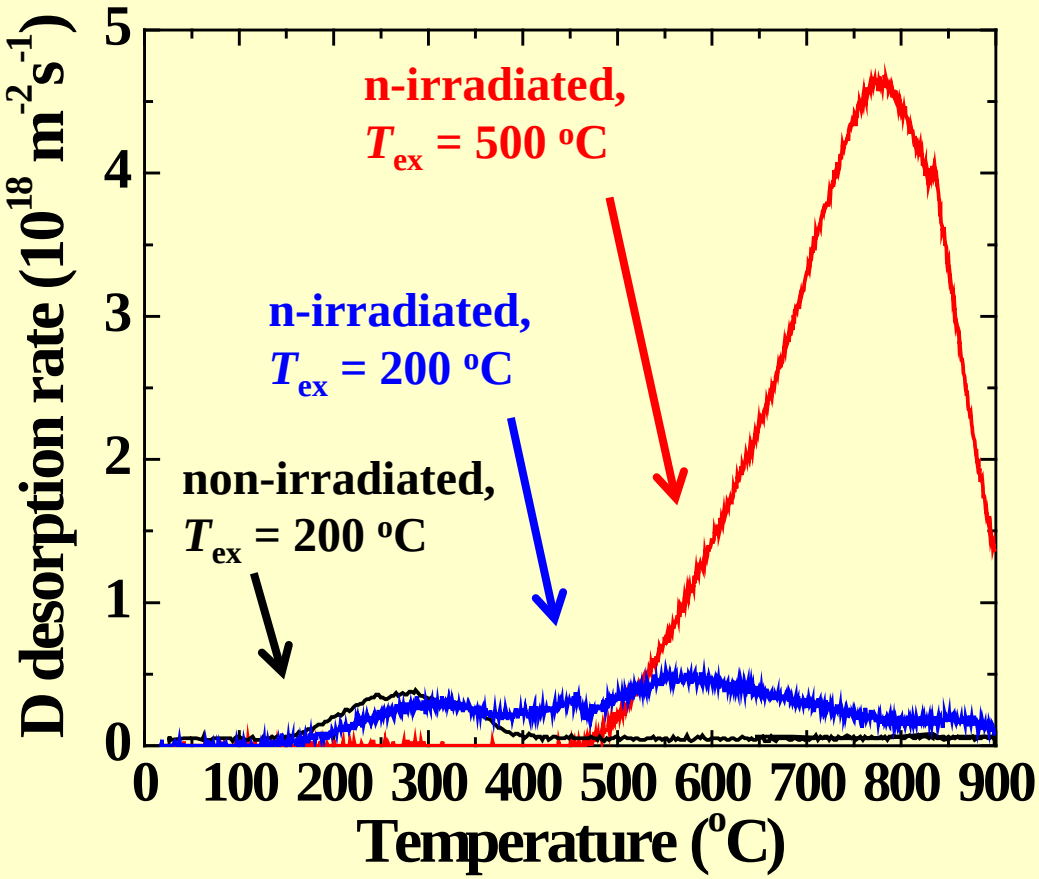
D retention in neutron-irradiated W

Result of previous project TITAN

In the previous project TITAN, W specimens were irradiated with neutrons in HFIR at around 50 °C without thermal neutron shielding and then exposed to plasma at 200 and 500 °C.

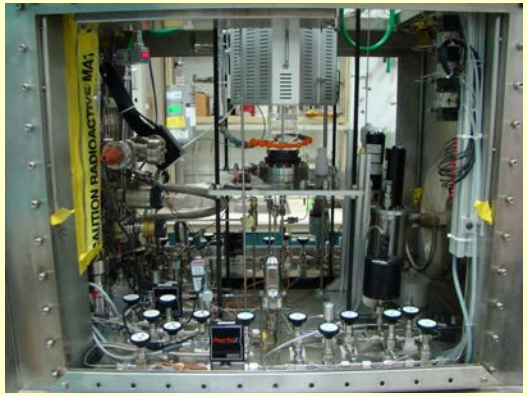
Significant increase in D retention was observed after neutron irradiations due to trapping effects by radiation-induced defects.

In the current project PHENIX, effects of high-dose irradiation at high temperature under thermal neutron shielding are examined.



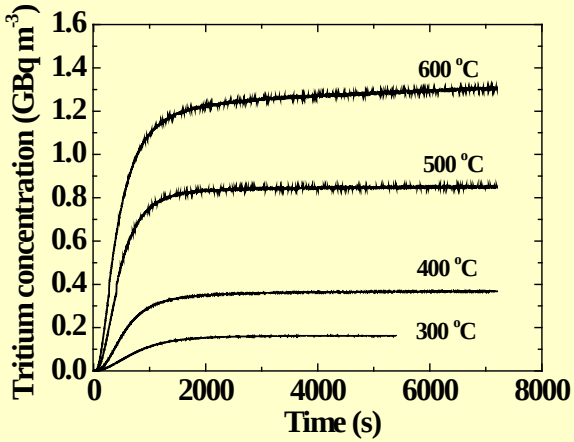
TDS spectra of n-irr. and non-irr. W at T_{ex} = 200 and 500 °C.

Task 3 2013-2014 Highlights



New tritium gas-driven permeation set up (TGAP)

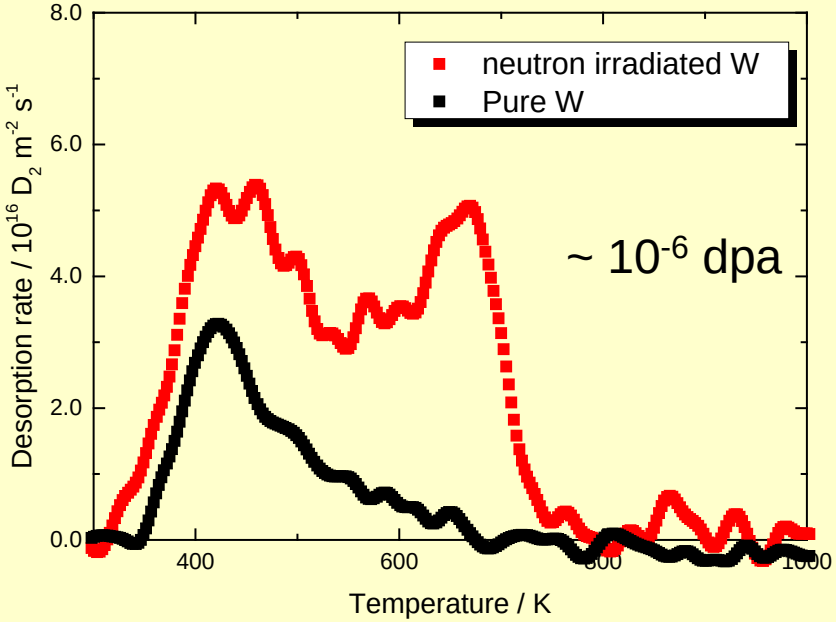
TGAP benchmark test was successful. (INL)



T_2 permeation through Fe

Deuterium retention in W irradiated with low-flux 14 MeV neutrons and high-energy (MeV) heavy ions was examined in Japan (Shiuzoka U.) and INL.

Clear effects of irradiation of 14 MeV neutron was observed at the dose as low as 10^{-6} dpa.



D_2 TDS spectra for 14-MeV neutron irradiated W

Summary

- ✓ In PHENIX project, we investigate
  1. Heat transfer mechanism and modeling in He-cooled systems.
  2. Thermo-mechanical properties of tungsten basic materials after neutron irradiation at elevated temperatures relevant to divertor conditions (500-1200 ^{o}C).
  3. Effects of high flux plasma exposure on tritium behavior in neutron-irradiated tungsten.
  4. Evaluation of feasibility (under ~10 MW/m^2 heat load with irradiation of plasma and neutrons) and safety (tritium retention and permeation) of He-cooled PFCs.
- ✓ Numerical simulation codes and experimental techniques have been developed to study neutron irradiation effects on thermo-mechanical properties of and tritium behavior in W.
- ✓ Neutron irradiation starts soon, and PIE starts in 2015.