Tritium behavior in neutron-irradiated tungsten using TPE divertor plasma simulator and TMAP mass transport code

Masashi Shimada
Brad J. Merrill
Chase N. Taylor
Fusion Safety Program, Idaho National Laboratory, U.S.A.

Outlines

1. Motivation for this IAEA CRP (F43021)
   - Motivation
   - INL’s proposed research
2. Unique capabilities of Fusion Safety Program, INL
   - Tritium Plasma Experiment (TPE) divertor plasma simulator
   - Tritium Migration Analysis Program (TMAP) mass transport code
   - HFIR neutron-irradiation under US-Japan PHENIX program
   - Other capabilities (NIMIX and TGAP)
3. Research highlights in “PSI with irradiated tungsten”
   - HFIR irradiation tungsten under US-Japan TITAN program
4. Summary

Assessment of T retention in neutron-irradiated W

- High energy ion beams have been used to simulate displacement damages by 14 MeV fusion neutron, and recent studies provided us three trends/assumptions:
  1. The trap concentration will most likely saturate at > 1 dpa
  2. Tritium will most likely stay with in a few micro meters from the surface
  3. Small tritium retention from damaged W at high (> 500°C) temperature
- If the above three trends/assumptions are true (?), tritium retention in W will be small.

INL’s proposed research for this CRP (F43021)

- Title: “Tritium behavior in neutron-irradiated tungsten using TPE divertor plasma simulator and TMAP mass transport code”
- Objectives: to investigate tritium behavior (retention as well as gas-driven and plasma-driven permeation) in neutron-irradiated tungsten utilizing unique capabilities of the INL:
  - Tritium Plasma Experiment (TPE) divertor plasma simulator
  - Tritium Migration Analysis Program (TMAP) mass transport code.
- Under US-Japan PHENIX collaboration, ITER grade tungsten will be irradiated with neutrons at high temperature (300, 600, 1000, and 1200 °C) up to 1.5 dpa at High Flux Isotope Reactor (HFIR) with thermal neutron shielding to better represent the fusion neutron energy spectrum

Tritium Plasma Experiment - TPE

Unique capabilities

- TPE is contained within double enclosure (PermaCon Box and Glovebox)
- TPE is unique in that it combines four specialized elements:
  - (a) the ability to handle tritium (max. T inventory: < 1.5g in STAR)
  - (b) a divertor-relevant high-flux plasma (max. ion flux: 4.0x10^17 m^-2 s^-1)
  - (c) the ability to handle radioactive materials (STAR limit: < 100 mR/hr = 10 μSv/hr)
  - (d) the ability to handle beryllium
- Plasma-driven tritium permeation capability (under development)
### Comparison of plasma parameters among Existing and proposed US Linear Plasma Devices

<table>
<thead>
<tr>
<th>Device Type</th>
<th>Plasma Source</th>
<th>Ion Incident Angle</th>
<th>Beryllium</th>
<th>Tritium</th>
<th>Electron Temperature: $T_e$ (eV)</th>
<th>Deuterium Ion Flux: $j_D$ (m$^{-2}$s$^{-1}$)</th>
<th>Electron Density: $n_e$ (m$^{-3}$)</th>
<th>Ion Temperature: $T_i$ (eV)</th>
<th>Max. specimen size</th>
</tr>
</thead>
<tbody>
<tr>
<td>TMAP</td>
<td>Reflex arc (LaB$_6$)</td>
<td>Normal***</td>
<td>No/Yes</td>
<td>Yes</td>
<td>2–5</td>
<td>10$^{18}$–10$^{20}$</td>
<td>10$^{18}$–10$^{19}$</td>
<td>2–5</td>
<td>120</td>
</tr>
<tr>
<td>INL-TPE</td>
<td>Helicon</td>
<td>Inclined and Normal</td>
<td>No</td>
<td>Yes</td>
<td>3–5</td>
<td>10$^{20}$–10$^{22}$</td>
<td>10$^{19}$–10$^{20}$</td>
<td>1–200</td>
<td>120</td>
</tr>
</tbody>
</table>

**NOTES:**
- *: Beryllium has been extensively tested in TPE during its tenure at TSTA, LANL in the 90s, but it has not been actively tested in INL.
- **: Tritium plasma-driven permeation capability is under development with the SNL/CA collaboration.
- ***: Incident angle can be varied upon target holder design, and the current target holder is designed for normal incidence only.

### Trinitium Migration Analysis Program - TMAP

- **TMAP** does not treat plasma surface physics, such as sputter or sputtered material re-deposition. TMAP’s basic equations are:

#### Surface Movement

**Non-equilibrium**

$$ J_m = -K_J p_m - K_J C_m + \alpha J_p $$

**Equilibrium**

$$ J_m = K_J p_m - K_J C_m $$

#### Bulk Diffusion

$$ \frac{\partial C_m}{\partial t} = D \frac{\partial^2 C_m}{\partial x^2} + S_m - \sum_{i} \frac{C_i}{\Delta t} $$

#### Bulk Trapping

$$ \frac{\partial C_m}{\partial t} = \alpha_i f_i C_m = \alpha C $$

**Unique capabilities**

- Designed to measure transport properties (e.g., diffusivity, solubility, and permeability) of tritium at realistic blanket conditions (e.g., low tritium partial pressure $< 1000$ Pa).
- Capable of testing liquid breeder material (e.g., PbLi and FLiBe) and disc shaped metal.
- Uniform temperature ($+/-15$ C) within the test section utilizing 12" tube furnace.

### Neutron Irradiated Material Ion Implantation Experiment (NIMIX)

**Unique capabilities**

- One of very few high-energy ion implantation devices that can handle neutron-irradiated materials.
- Moderate ($10^{18} – 10^{20}$ ions/cm$^2$) ion fluence to study plasma surface interaction in the first wall.
- Specializes in neutron-irradiated materials.

### Tritium GAs Permeation - T-GAP

**Unique capabilities**

- Designed to measure transport properties (e.g., diffusivity, solubility, and permeability) of tritium at realistic blanket conditions (e.g., low tritium partial pressure $< 1000$ Pa).
- Capable of testing liquid breeder material (e.g., PbLi and FLiBe) and disc shaped metal.
- Uniform temperature ($+/-15$ C) within the test section utilizing 12" tube furnace.

### Experimental procedures

1. W rod fabrication (A.L.M.T. Co., Japan) and sample preparation (slicing to 0.2 mm, polishing, and annealing at 900°C for 0.5 hour) at Univ. Toyama, Japan.
2. Neutron irradiation at 50-70°C to 0.025 Tm and 0.3 dpa at the HFIR.
3. 1st D plasma exposure at 100, 200, and 500°C to 5x10$^{20}$ ions cm$^{-2}$ at TPE.
4. 1st D depth profile measurement via NRA at U. Wisconsin-Madison.
5. 2nd D plasma exposure at 100, 200, and 500°C to 5x10$^{20}$ ions cm$^{-2}$ at TPE.
6. 2nd D depth profile measurement via NRA at U. Wisconsin-Madison.
7. D retention measurement via TD$\delta$ 10 Cm to 900°C.

### Deuterium depth profiles in HFIR neutron-irradiated W

<table>
<thead>
<tr>
<th>Depth (m)</th>
<th>0.1 Tm</th>
<th>0.3 Tm</th>
</tr>
</thead>
<tbody>
<tr>
<td>100</td>
<td>0.005</td>
<td>0.01</td>
</tr>
<tr>
<td>200</td>
<td>0.01</td>
<td>0.02</td>
</tr>
<tr>
<td>500</td>
<td>0.02</td>
<td>0.04</td>
</tr>
</tbody>
</table>

Ref: H. Schultz, Mater. Sci. Eng. '91

---

**Outlines**

1. INL’s proposed research for this IAEA CRP (F43021)
   - Motivation
   - INL’s proposed research
   - Expected outcomes
2. Unique capabilities of Fusion Safety Program, INL
   - Tritium Plasma Experiment (TPE) divertor plasma simulator
   - Tritium Migration Analysis Program (TMAP) mass transport code
   - HFIR neutron-irradiation under US-Japan PHENIX program
   - Other capabilities (NIMIX and TGP)
3. Research highlights in “PSI with irradiated tungsten”
   - HFIR irradiation tungsten under US-Japan TITAN program
4. Summary
Experimental procedure:

1. **1st TPE (@INL)**
   - Flux: 5x10^{21} m^{-2}s^{-1}, Fluence: (4-5) x10^{25} m^{-2} each TPE exposure, (8-10) x10^{25} m^{-2} in total
   - Ion energy: 100 eV
   - Plasma exposure temperature: 100, 200, and 500 C
   - Dose: 0, 0.025, and 0.3 dpa

2. **1st NRA (@U of Wisc.)**
   - NRA provides (near surface) trap concentration
   - An implantation depth of 3 nm estimated by Wampler and TPE implantation flux
   - Frauenfelder measured diffusivity and solubility and Anderl's determined surface jump diffusion coefficient

3. **2nd TPE**
   - Enhanced Diffusion Zone (EDZ) (<10 nm) during plasma implantation in order to reduce the mobile concentration
   - Often, the mobile concentration is measured in the ion fluence of 10^{26} m^{-2} range

4. **2nd NRA**
   - NRA provides (near surface) trap concentration
   - Different calculation methods and assumptions are used for various cases

5. **3rd NRA**
   - NRA measurements and trap concentration calculations are made

6. **Final NRA**
   - NRA measurements and trap concentration calculations are made

To apply TMAP to TPE results, proper data for tungsten diffusivity, solubility, and surface recombination are required.

An implantation depth of 3 nm estimated by Wampler and TPE implantation flux (5x10^{21} D+/m^{2}-s), reflection coefficient of 0.5, and flux and temperature histories were adopted for this study.

An implantation depth of 3 nm estimated by Wampler and TPE implantation flux (5x10^{21} D+/m^{2}-s), reflection coefficient of 0.5, and flux and temperature histories were adopted for this study.

During plasma implantation:

- **Enhanced diffusion (D_{diff})** of (0-2.0) x10^{-5} m^{2}/s matches the experiment well.
- **Dissociation coefficient (D_{diss})** by Anderl

During thermal desorption:

- **Dissociation coefficient (D_{diss})** by Anderl
- **Dissociation coefficient (D_{diss})** by Anderl
- **Dissociation coefficient (D_{diss})** by Anderl
- **Dissociation coefficient (D_{diss})** by Anderl

References:

1. Deuterium/tritium retention by high-flux TPE plasma
   - Deuterium retention at high plasma exposure temperature (500 °C)
   - Mixed (D/T/He) plasma
   - High flux (10^{22} – 10^{23} m^{-2}s^{-1}), high-fluence (10^{26} – 10^{27} m^{-2})

2. Realistic fusion neutron irradiation
   - Larger irradiation port (removable beryllium facility)
   - HFIR irradiation with thermal neutron shielding
   - High irradiation temperature (500-1200C)
   - Deuterium gas environment

Challenges in DEMO environments:
- W and W alloy, one of the promising candidate materials for first wall and divertor PFCs, are expected to receive dose up to >10 dpa at high temperature (>500 °C)
- T will expected to migrate and find the trapping site in the bulk, increasing T retention
- We need to understand the physics of tritium behavior in irradiated W and find the mitigation/removal technique

Summary:
- The FSP/INL possesses the unique capabilities (TPE, TMAP, NIMIX, and TGAP) to study the PSI in irradiated tungsten, and will utilize them for this CRP (F43021)
- FSP/STAR continues to study the PSI in irradiated tungsten under the US-Japan PHENIX collaboration (April 2013-March 2019)
- Through this CRP, we hope to achieve the successful development of scientific understanding of tritium behavior in neutron-irradiated tungsten, which can be a major breakthrough in fusion technology and safety.
- This knowledge would minimize the uncertainty in tritium in-vessel inventory levels, and better understanding of tritium in-vessel inventories and ex-vessel releases will support licensing assessments.