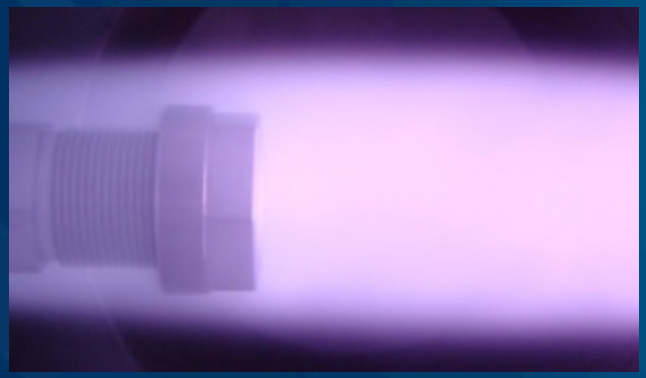
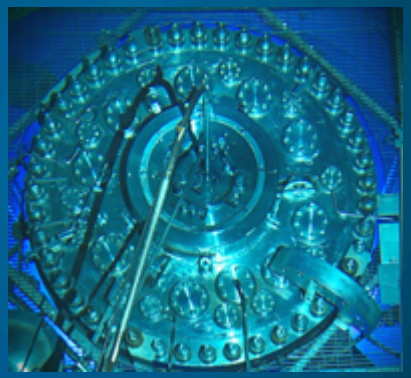
2nd Research Coordination Meeting (RCM) of the Coordinated Research Project (CRP) on
"Plasma-Wall Interaction with Irradiated Tungsten and Tungsten Alloys in Fusion Devices"

Status update on MERCOR-TMAP integration, Tritium Plasma Experiment Modification, and PHENIX program for PMI in irradiated tungsten





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STI #: INL/MIS-15-XXXXXX | Seoul, Korea | 9/9/2015

U.S. DEPARTMENT OF ENERGY | Office of Science

Outline

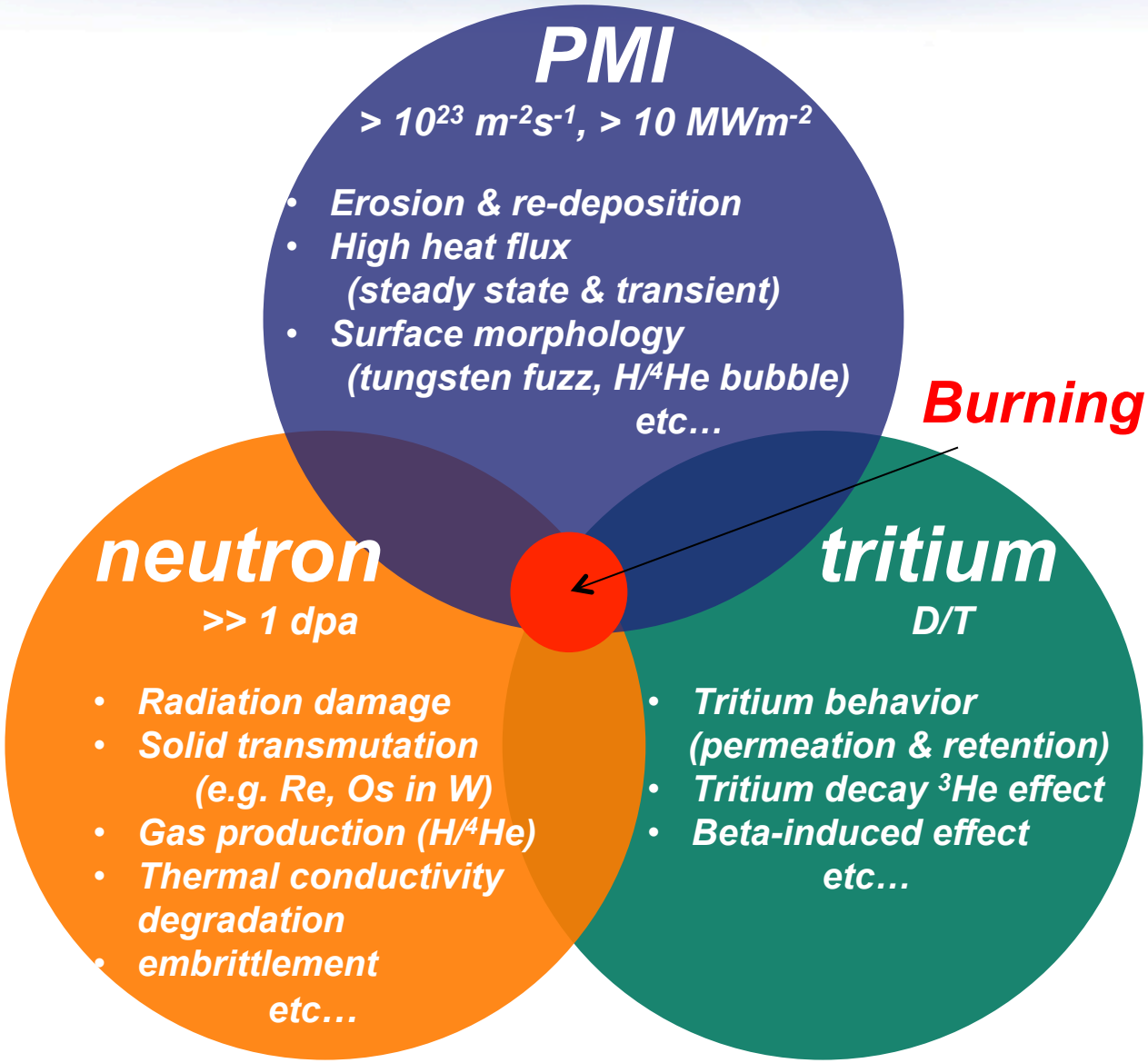
1. Motivation for this CRP

2. Progress since 1st RCM
   - *Defect annealing of neutron-induced damage in tungsten*
   - *High energy (2 keV) D ion effects on D retention*

3. Status update on:
   - *HFIR neutron-irradiated tungsten under PHENIX program*
   - *TMAP-MERCOR integration*
   - *Tritium Plasma Experiment (TPE) modification*
   - *Other supporting experiment for this CRP*

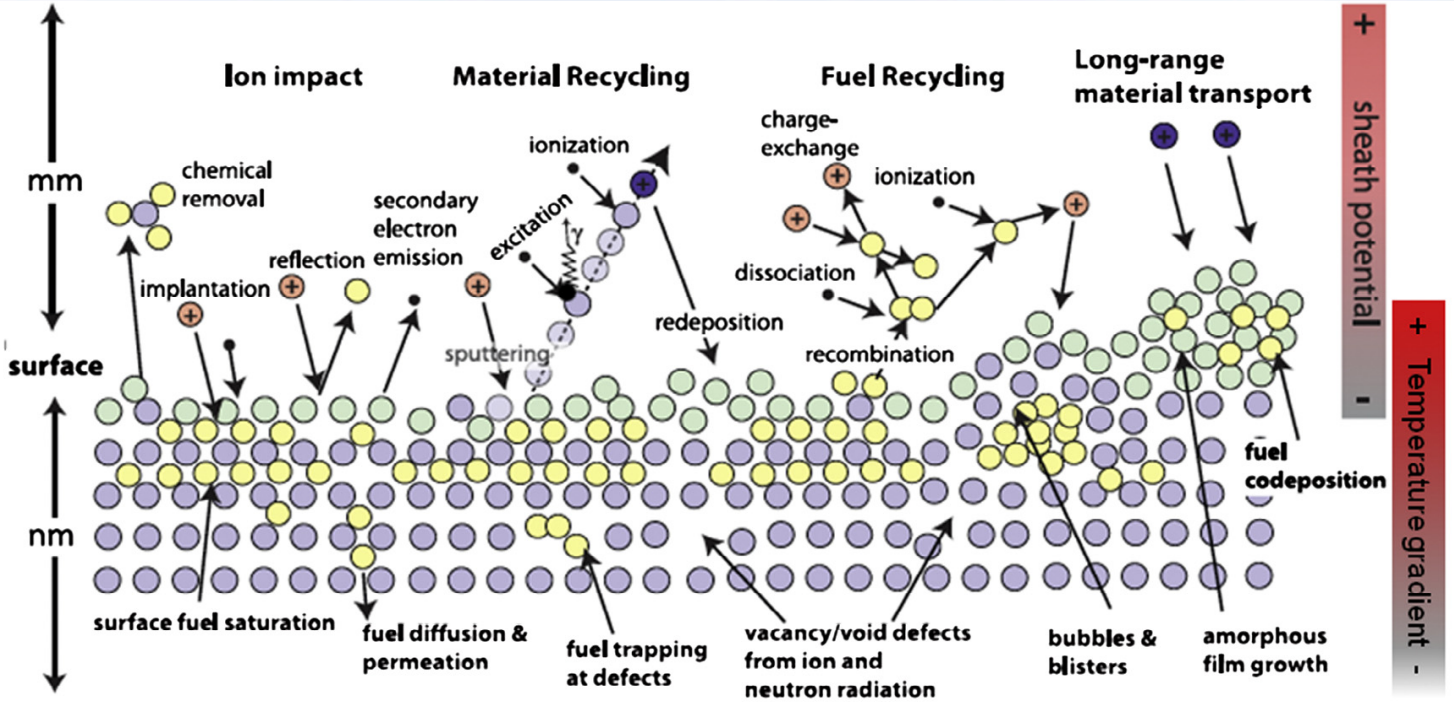
Motivation:

Combined/Synergistic effects in Burning Plasma Long Pulse PMI



Motivation:

Tritium retention in ITER D/T, DEMO, future fusion reactor



PMI

T retention

>10μm ~ mm

Concern in bulk (>10μm) W:

- *Neutron-induced damage in bulk W*
- *Tritium diffusion/permeation to bulk W at high temp.*
- *Less ^{4}He effect in bulk W*
- *Tritium decay ^{3}He might play addition role in bulk W*

➔ Bulk effects can play a major role in “T retention in neutron-irradiated W”.

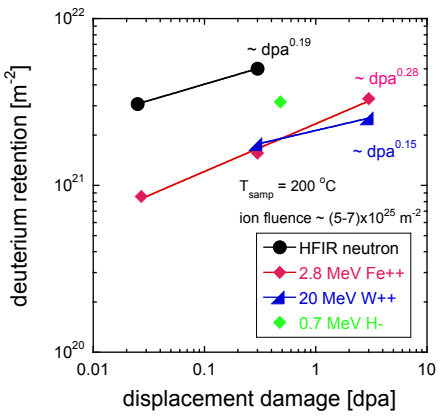
INL's planed research for this CRP (F43021)

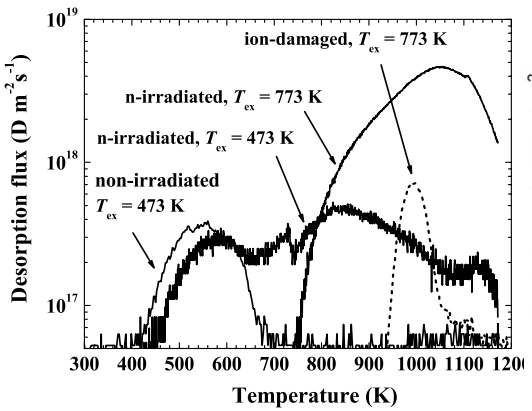
- Title: "**Tritium behavior in neutron-irradiated tungsten using TPE divertor plasma simulator and TMAP mass transport code**"
- Objectives: to investigate tritium behavior (retention as well as gas-driven and plasma-driven permeation) in neutron-irradiated tungsten utilizing unique capabilities of the INL:
  - ***Tritium Plasma Experiment (TPE) divertor plasma simulator***
  - ***Tritium Migration Analysis Program (TMAP) mass transport code***
  - ***HFIR neutron irradiated tungsten specimens***
    - Under US-Japan PHENIX collaboration, large number (>100) of tungsten specimens (e.g. single crystal W, ITER grade W, W/Re alloy) will be irradiated with neutrons at high temperature (500, 800, 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.

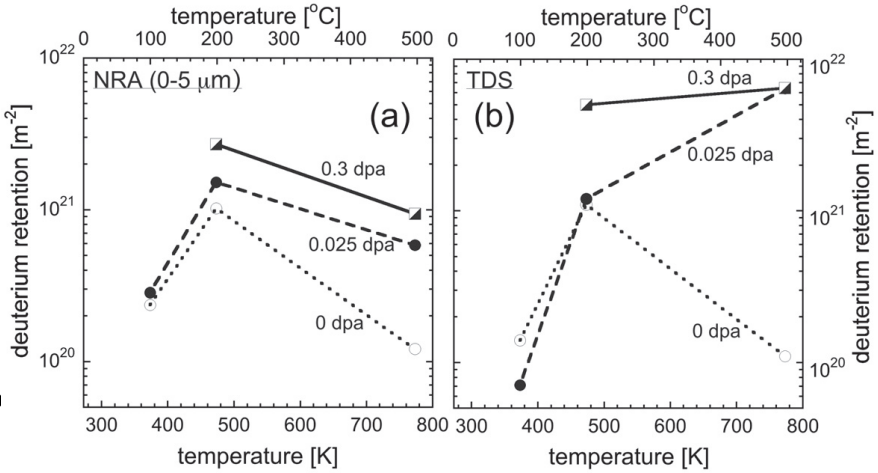
History of HFIR irradiation in tungsten:

Tritium retention in HFIR neutron-irradiated tungsten

- ✓ **US-Japan TITAN program (2007-2013):**
  - Low-temperature (<100 °C) low-dose (0.025 & 0.3 dpa) HFIR neutron-irradiated tungsten
  - Tritium was trapped in bulk (>10 μm), and retention increases at high temp. (500 °C)
  - Measurement of microstructural evolution and characterization of radiation defects and defect annealing before/after plasma exposure is required to reveal trapping mechanism.
- ✓ **US-Japan PHENIX program (2013-2018):**
  - High-temperature (500, 800, & 1200°C) medium-dose (0.3 &1.5 dpa) HFIR neutron-irradiated tungsten RB* irradiation with Gd thermal neutron shield
  - Positron annihilation spectroscopy (PAS) at ORNL and INL will characterize radiation defects and before/after plasma exposure to reveal defect annealing mechanism.
  - Nuclear reaction analysis, thermal desorption spectroscopy, and PAS is used to determine tritium migration depth and trap density for tritium retention assessment.







M. Shimada et.al. *Fus. Eng. Des.* **87**(2012) Y. Hatano et.al. *Nucl. Fusion* **53**(2013) M. Shimada et.al., *Nucl. Fusion* **55**(2015)

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Defect annealing of neutron-induced damage in tungsten

- Question: **Ref: M. Shimada et.al., Journal of Nuclear Materials 463 (2015) 1005–1008**
  - *Can we anneal the neutron-induced damage in tungsten at 900 °C ?*

- Methodology in this research/paper:
  - *HFIR neutron-irradiated tungsten (0.025 dpa @ 70-80 °C irradiation)*
  - *Repeated D plasma exposure and TDS (up to 900 °C) for 4 times*
  - *Measure changes in TDS spectra*
  - *Model with TMAP for changes in trap concentration*

- Plasma exposure conditions:

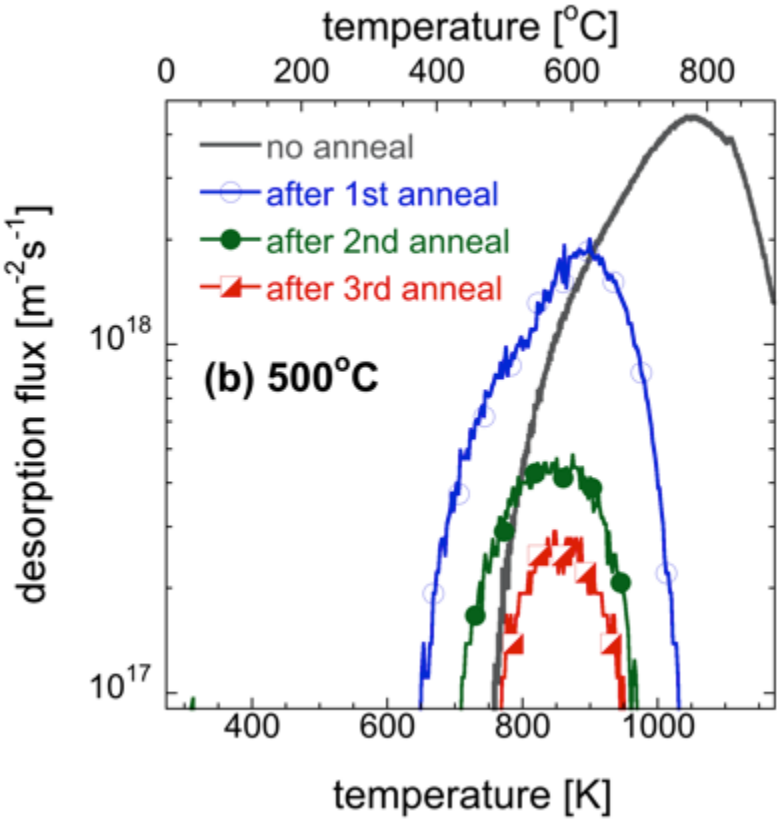
Sample ID	Weight [gram]	Sample size[c] (diameter / thickness) [mm]	HFIR irradiation dose [dpa]	TPE exposure temperature [°C]	Ion fluence[d] [m^{-2}]			
					1st TDS	2nd TDS	3rd TDS	4th TDS
Y102	0.80	6.0 / 0.15	0.025	100	1.0E+26	1.0E+26	6.3E+25[f]	4.2E+25[f]
Y103	0.89	6.0 / 0.16	0.025	200	9.7E+25	1.0E+26	1.0E+26	9.8E+25
Y105	0.83	6.0 / 0.15	0.025	500	1.2E+26	1.0E+26	6.1E+25[f]	9.8E+25

- Thermal desorption spectroscopy condition: 10 °C/min up to 900 °C

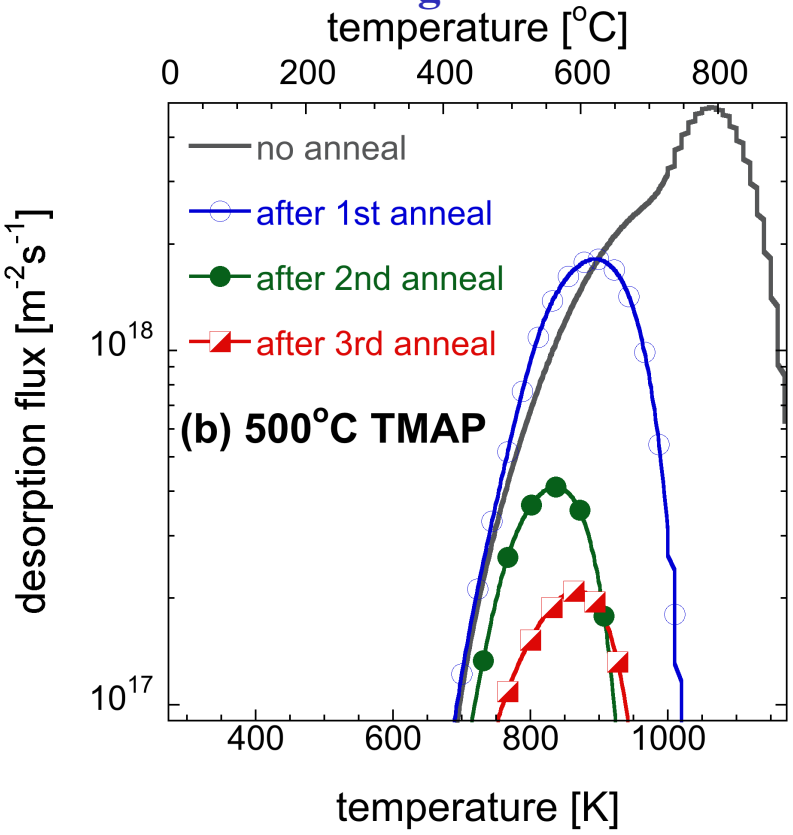
Defect annealing of neutron-induced damage in tungsten

- Thermal desorption spectra of 500 °C exposure sample
  - *Reduction in TDS peak as each annealing at 900 °C for 0.5 h*
  - *Lowered TDS peak from 800 °C to 600 °C in after 1st anneal*

Experimental result of TDS



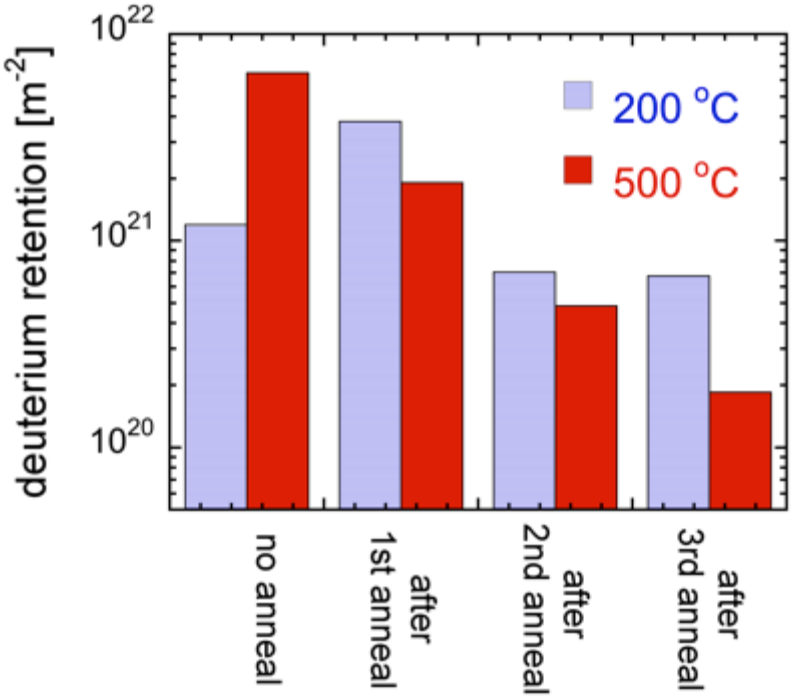
Modeling result of TDS



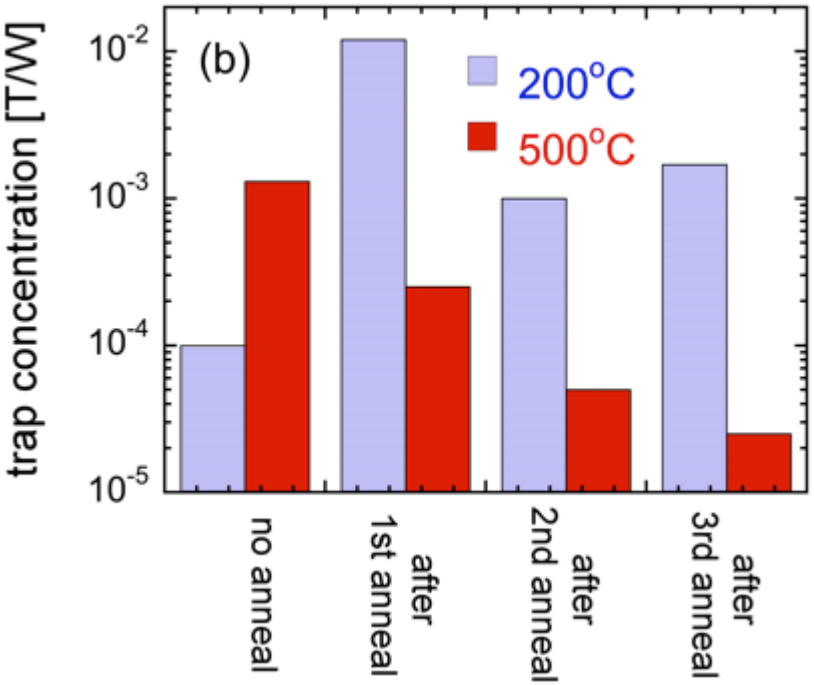
Defect annealing of neutron-induced damage in tungsten

- Deuterium retention of 500 °C exposure sample (red line)
  - *Reduction in total deuterium retention with annealing at 900 °C for 0.5 h*
  - *TMAP modeling also support reduction in trap concentration with annealing at 900 °C for 0.5 h*

Experimental result of TDS



Modeling result of TDS



Defect annealing of neutron-induced damage in tungsten

- Conclusion from this research:
  - *Deuterium retention decreases approximately 70% for at 500 °C after each annealing at 900 °C for 0.5 h*
  - *TMAP modeling revealed the trap concentration decreases approximately 80% after each annealing at 900 °C for 0.5 h.*
  - ***The neutron-induced radiation damage were NOT annealed out completely even after the 3rd annealing.***

- Concern for this CRP
  - *This is results from D exposure results (not He/D/T results)*
  - *Existing of ^{4}He in near-surface radiation-damage site probably make it more difficult to anneal radiation damages*
  - *Existing of tritium decay ^{3}He in bulk radiation-damage site also make it more difficult to anneal radiation damages*
- Concern for high wall temperature operation (>500 °C) of DEMO
  - *Diffusion of tritium increases whereas annealing effect may not take place until very high temperature (1000 °C)*
  - *Tritium retention might be high in high wall temperature operation*

High-energy (2.0 keV) D ion effects on D retention

Presented by C.N. Taylor at PFMC-15

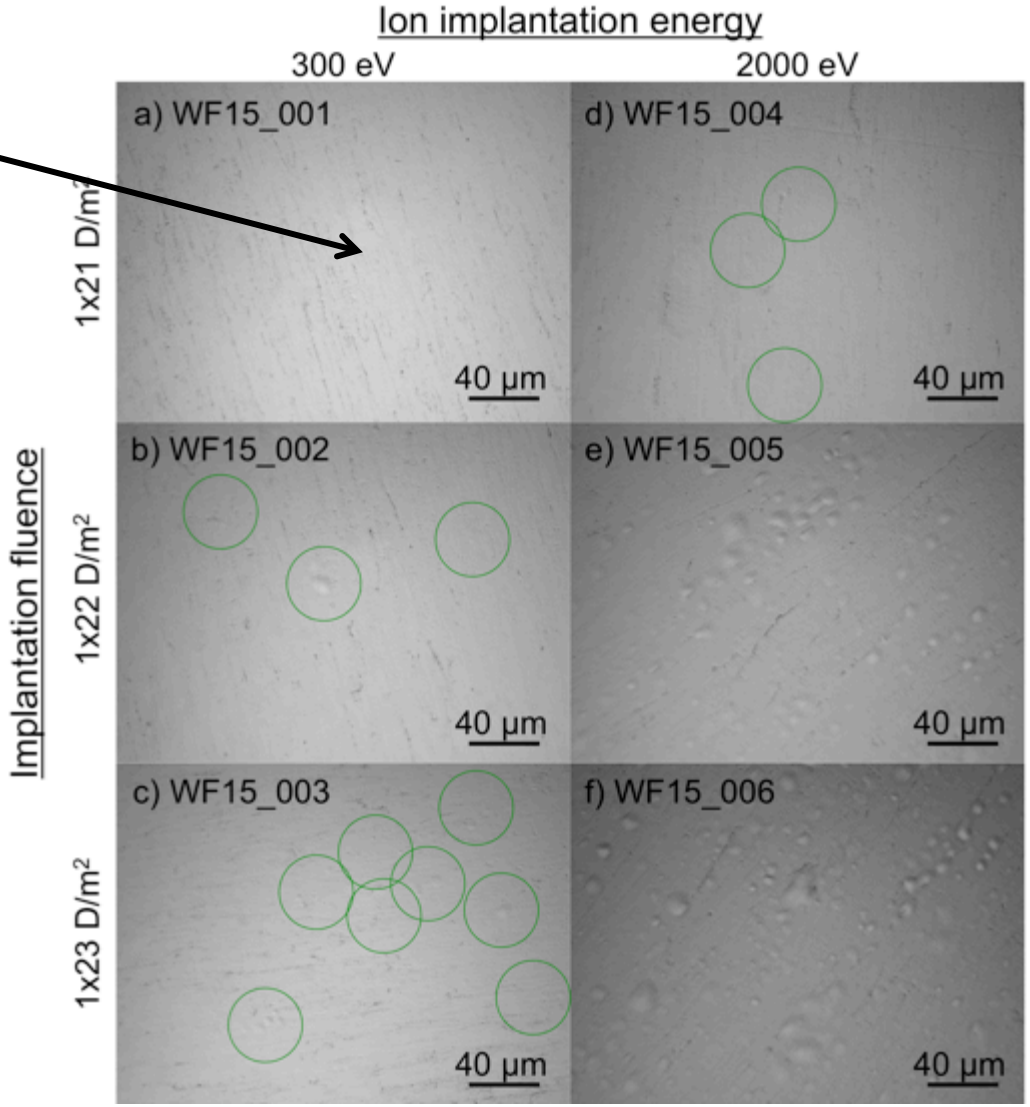
- Question:
  - *Can high-energy (2.0 keV) D ion create radiation-damage in tungsten ?*
  - *To what energy range (0.5-10.0 keV) can we use our ion beam for D retention study?*
- Methodology in this research/paper:
  - *Annealed at 900ºC for 0.5 h prior to ion beam exposure*
  - *Expose to two different energy (0.3 keV vs. 2.0 keV) below or above damage production threshold energy of 933 eV/D at RT (20 ºC)*
  - *Measure changes in TDS spectra*
- Ion beam exposure conditions:

Sample	Beam	Ion energy (eV/D)	Temp. (°C)	Flux (D/m^2s)	Fluence (D/m^2)
WF15_001	D_3^+: 900 eV	300	RT	2×10^{18}	1×10^{21}
WF15_002	D_3^+: 900 eV	300	RT	2×10^{18}	1×10^{22}
WF15_003	D_3^+: 900 eV	300	RT	2×10^{18}	1×10^{23}
WF15_004	D_3^+: 6000 eV	2000	RT	2×10^{19}	1×10^{21}
WF15_005	D_3^+: 6000 eV	2000	RT	2×10^{19}	1×10^{22}
WF15_006	D_3^+: 6000 eV	2000	RT	2×10^{19}	1×10^{23}

- Thermal desorption spectroscopy condition: 30 °C/min up to 900 °C

Optical microscopy observation

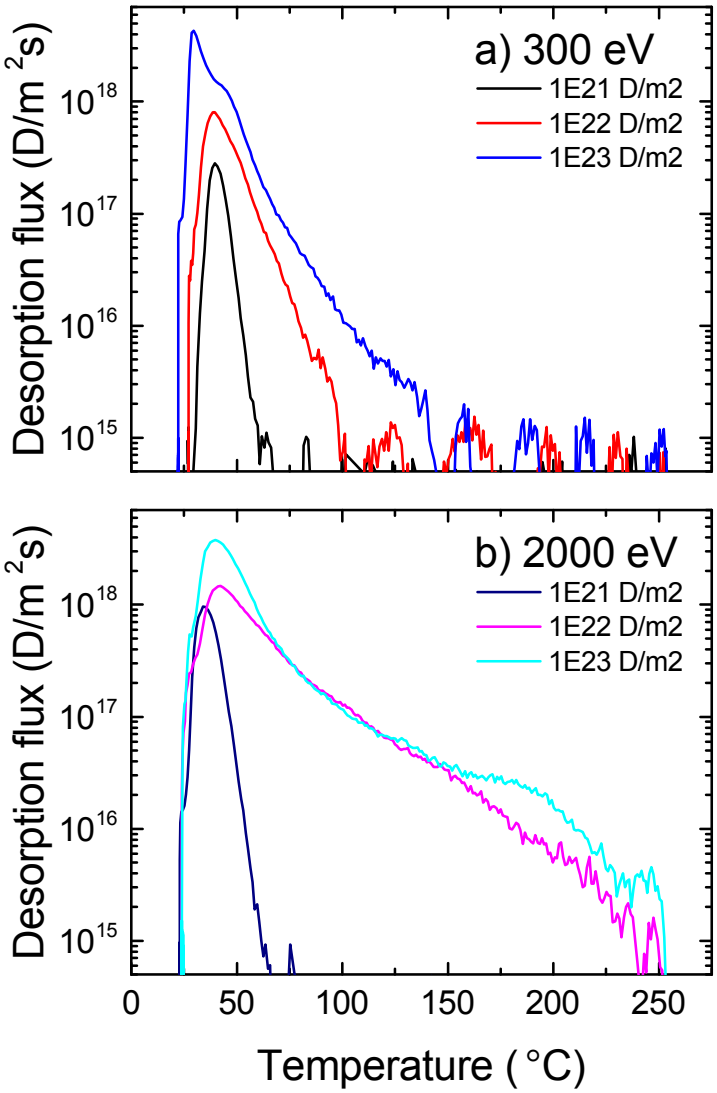
- Observations
  - *Large blisters (> 5μm) were observed in all samples except for low fluence & low energy*
  - *Larger blisters (up to 20 μm) at 2000 eV.*
  - *More blisters at higher fluence.*
  - *Blister of 300 eV are elastic deformation*
- Conclusions
  - *Blister formation mechanism is not due to displacements.*
  - *Blister formation is due to pressure induced stress from gas precipitation.*
  - *But displacements can help create larger blister formation.*
  - *Small (< 5 μm and not visible with optical microscope)blister exists in <111> grain [1]*
- Future work
  - *SEM, TEM measurements*



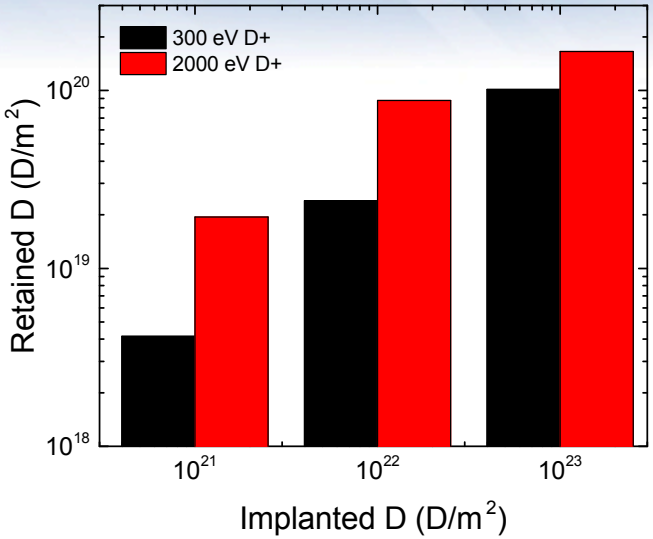
Ref: R.D. Kolasinski, M.Shimada et.al. JAP 2015

Deuterium desorption from 0.3 and 2.0 keV D ion exposure

- Desorption conditions
  - *Ramp rate of 0.5ºC/sec to 900ºC*
- Observations
  - *The primary desorption peak (< 100ºC) is very high and is deuterium trapped in dislocation type trapping site due to low detrapping energy.*
  - *Similar to the peak observed in unannealed W by R.A. Anderl (Anderl et.al. JNM2001)*
  - *Desorption from blisters may be playing small role at low energy exposure (300 eV).*
  - *As the fluence increases, the primary desorption peak becomes wider, creating more dislocation type trapping site.*
  - *Displacement with 2000 eV beam creates additional trapping site around 200º C*
  - *Desorption from blister (peak around 200 ºC) still play minor role in high fluence, high energy case.*



Deuterium desorption from 0.3 and 2.0 keV D ion exposure



- Observations
  - *Similar retained D fluence (fractional retention up to 0.4 %) with Haasz (Haasz et.al. JNM 97), but different TDS spectrum.*
    - The primary desorption peak (< 100ºC) is attributed to deuterium trapped in dislocation type trapping site in our results due to low detrapping energy.
  - *Displacement with 2000 eV beam increase the D retention by a factor of 4 at low fluence (fractional retention up to 2.0 %)*
  - *D retention in the low temperature peak starts to saturate at high fluence (10^{23} m^{-2})*
    - Now fractional retention decreased to 0.1 %.
- Conclusions/future work:
  - *High energy (2000 eV/D) beam does enhances D retention for low fluence, so low energy beam (< 933 eV/D) should be used for low fluence study.*
  - *Change annealing temperature or exposure temperature to eliminate/minimize the low desorption peak (< 100ºC), and with SEM, TEM, PAS measurement*

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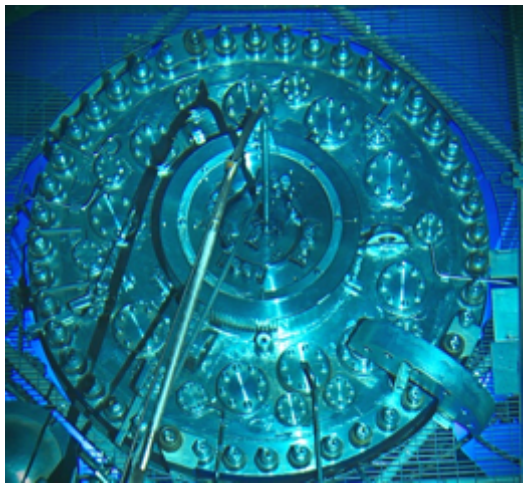
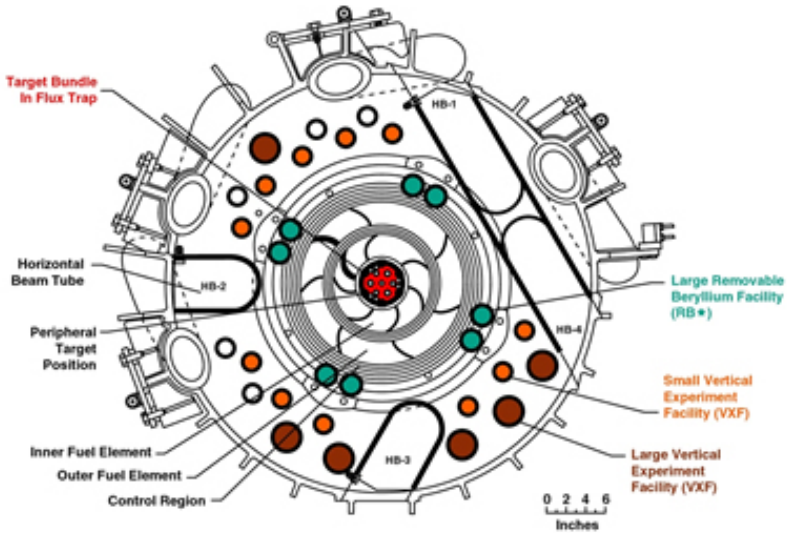
Status update on: HFIR neutron-irradiation under PHENIX (April 2013 – March 2019)

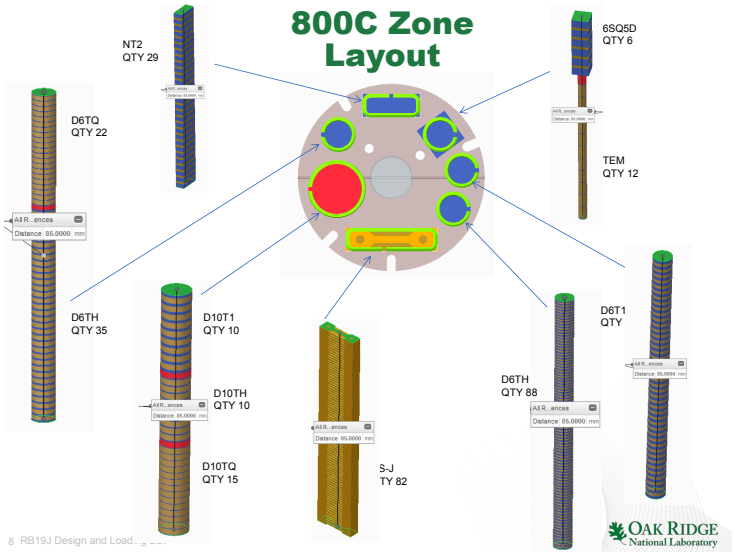
- Capsule design completed in June 2015
- Capsule irradiation planed from February 2016 to April 2017 (6 cycle irradiation)
  - *Position:* *Removable Beryllium* (RB*) irradiation port at HFIR*
  - *Dose:* *1.0 ~ 1.5 dpa*
  - *Temperature:* *500, 800, and 1200 °C*
  - *Uniqueness:* *with Gd thermal neutron shielding*
- Removal from HFIR and disassembly at hot cell around July 2017
- PIE from August 2017 – March 2019

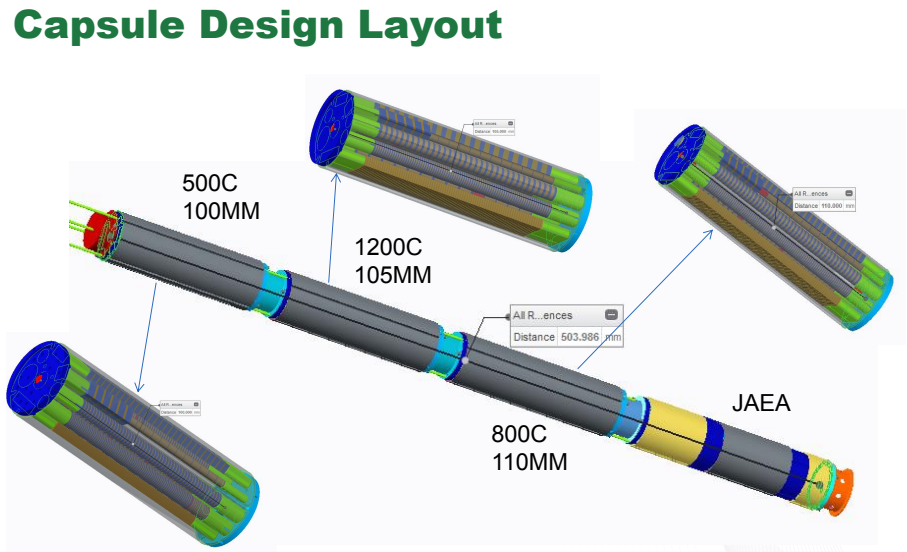




Capsule design and specimen matrix in PHENIX

- RB19J capsule design
  - *Cost sharing with JAEA-ORNL F82H irradiation program (300 °C zone)*
  - *500, 800, 1200 °C temperature zone for PHENIX tungsten irradiation*
- Specimen matrix for PHENIX Task 3: “Tritium behavior in irradiated tungsten”
  - *D6TQ : 6 mm OD, 0.25mm thick discs : ~ x10 each temp. zone*
  - *D6TH : 6 mm OD, 0.5 mm thick discs (standard) : ~ x80 each temp. zone*
  - *D6T1 : 6 mm OD, 1.0 mm thick discs : ~ x10 each temp. zone*
  - *D10TQ : 10 mm OD, 0.25 mm thick discs : ~ x15 each temp. zone*
  - *D10TH : 10 mm OD, 0.5 mm thick discs : ~ x10 each temp. zone*





PIE plans in PHENIX Task 3: "Tritium behavior in irradiated tungsten"

- **(x10)D6TQ, (x10)D6TH, (x10)D6T1**
  - *Thickness dependence on retention (i.e. long time TPE exposure)*
  - *Thickness dependence on gas-driven permeation*
- **(x15)D10TQ, (x10)D10TH**
  - *Thickness dependence in plasma-driven permeation*
- **(x70)D6TH**
  - *Temperature (500-1200 °C), ion flux (1e22-1e23 $m^{-2}s^{-1}$), ion fluence (1e26-1e28 m^{-2}) dependence on retention*
  - *He effect (He, D, T) on retention*
  - *Deuterium gas exposure (500-950 °C, < 1e6 Pa) for trap mechanism*

NOTE: Actual test matrix will be determined based on the number of intact HFIR irradiated samples after its shipment to INL.

PIE plans in PHENIX Task 3: *"Tritium behavior in irradiated tungsten"*

1) PIE at ORNL
   - *Positron Annihilation Spectroscopy (PAS) coincidence lifetime measurement for defect characterization (sharing data from Task 2)*
   - *Transmission Electron Spectroscopy (TEM) for defect characterization (sharing data from Task 2)*
2) PIE at INL
   - *PAS Doppler broadening measurement for defect characterization prior to and post plasma exposure and TDS*
   - *Plasma (and/or ion) exposure with TPE (and/or) NIMIIX*
   - *Gas-driven tritium permeation with TGAP*
   - *Thermal desorption spectroscopy (TDS)*
   - *Deuterium gas exposure with SGAS*
3) Nuclear reaction analysis (NRA) at U. Wisconsin-Madison or SNL-MN
4) Supporting PIE at JA Oarai Center

Status update on: Recent development in TMAP from TMAP4

- Multiple trap model (TMAP4 modified version)
  - Ref: Brad J. Merrill, M. Shimada, P.W. Humrickhouse, *J. Plasma Fusion Res Series* **10** (2013) 71
  - Fit with:
    - 1 trap density of 0.35 at.%
    - 6 detrapping energy (0.9, 1.0, 1.1, 1.2, 1.4, and 1.6 eV)

J. Plasma Fusion Res. SERIES, Vol. 10 (2013)

Simulating Tritium Retention in Tungsten with a Multiple Trap Model in the TMAP Code

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(Received: 10 May 2012 / Accepted: 1 September 2012)

Accurately predicting the quantity of tritium retained in plasma facing components is a key safety issue for licensing future fusion power reactors. Retention of tritium in the lattice damage caused when high energy neutrons collide with atoms in the structural material of the reactor's plasma facing components (PFCs) is an area of ongoing experimental research at the Idaho National Laboratory (INL) under the US/Japan TITAN collaboration. Recent experiments with the Tritium Plasma Experiment (TPE), located in the INL's Safety and Tritium Applied Research (STAR) facility, demonstrate that this damage can only be simulated by computer codes like the Tritium Migration Analysis Program (TMAP) if one assumes that the lattice damage produced by these neutrons results in multiple types of hydrogen traps (energy wells) within the material, each possessing a different trap energy and density. Previous attempts to simulate the quantity of deuterium released from neutron irradiated TPE tungsten targets indicated that at least six different traps are required by TMAP to model this release. In this paper we describe a recent extension of the TMAP trap site model to include as many traps as required by the user to simulate retention of tritium in neutron damaged tungsten. This model has been applied to data obtained for tungsten irradiated to a damage level of 0.025 dpa in the High Flux Isotope Reactor (HFIR) at the Oak Ridge National Laboratory (ORNL) after exposure to a plasma in TPE.

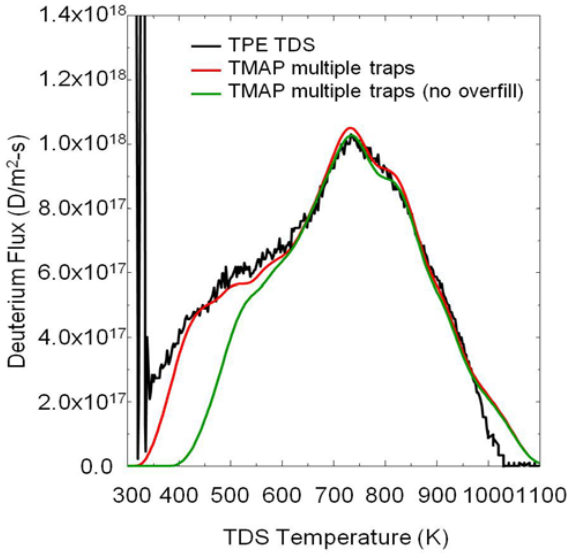


Fig. 4. Comparison of TMAP simulation with multiple traps and the TDS spectrum for neutron irradiated tungsten exposed to a TPE plasma at 200ºC.

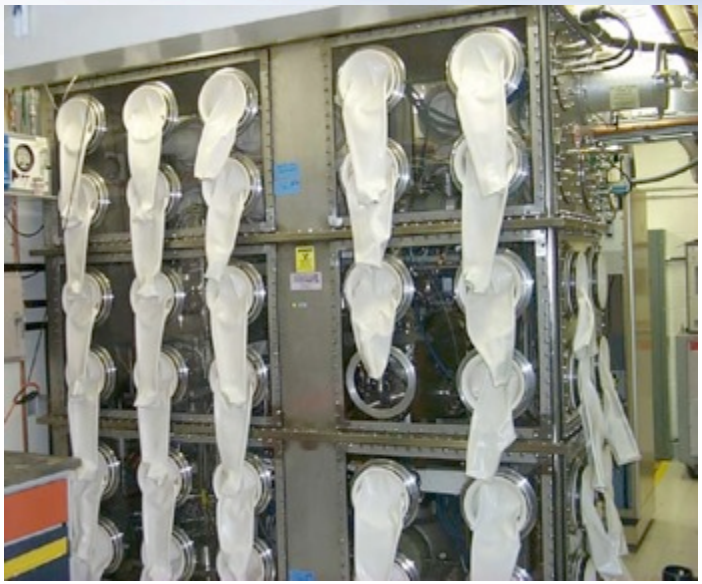
Status update on: TMAP integration into MERCOR safety code

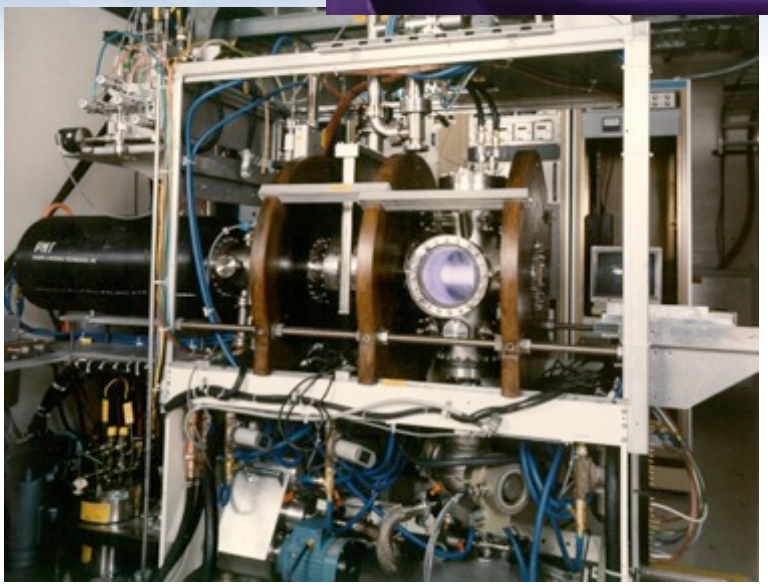
- MELCOR
  - *A fully integrated, engineering level thermal-hydraulics computer code that is being developed at Sandia National Laboratories for the U.S. Nuclear Regulatory Commission, and models the progression of accidents in fission reactor (e.g. BWR, PWR, and advanced reactor concept)*
- MELCOR for Fusion
  - *Modified by INL FSP to model the progression of accidents in fusion power plants, including a spectrum of accident phenomena such as reactor cooling system and containment fluid flow, heat transfer, and aerosol/dust transport*
  - *Used in ITER's NSSR 1&2, GSSR, RPrS (after being pedigreed and placed in INL Software QA system as a QL1 safety code), US DCLL TBM, reactor design studies APEX (Li and FLiBe), ARIES AT/SC/ACT, LIFE, and the JA and EU DEMOs.*
- Tritium Migration Analysis Program - TMAP
  - *A tritium migration code that treats multi-specie surface absorption and diffusion in composite materials with dislocation traps, plus the movement of these species from room to room by air flow within a given facility*
  - *Used in safety analyses for ITER, US DCLL TBM, reactor design studies APEX (Li and FLiBe), ARIES AT/SC*

Status update on: TMAP integration into MERCOR safety code

- TMAP integration into MERCOR 1.8.6
  - The natural development path for both of these codes (MERCOR and TMAP) is to merge their capabilities into one computer code to provide a more comprehensive safety tool for analyzing accidents in fusion reactors.
  - A new version of MELCOR for fusion is under development based on the last F77 MELCOR 1.8.6 (released in 2009) that combines the capabilities of MELCOR for ITER (1.8.2) and MELCOR Multiple Fluids (1.8.5).
  - This version was chosen because of its double precision accuracy and ease of source modification compared to MELCOR 2.x (F95 is under SNL-NM configuration control).
  - TMAP is being imported into this MELCOR code version in order to produce a more self-consistent safety accident analysis capability (also an update for the TMAP code, making it more advanced than user requested TMAP7), and will be presented in on-coming ISFNT-12
  - In the near term, the development and validation of this version of MELCOR for fusion will continue at INL. When released, the user of this code will have a comprehensive tool for analyzing accidents in fusion reactors.

Tritium Plasma Experiment - TPE





Unique capabilities

- TPE is at Safety and Tritium Applied Research (STAR) facility, LTHC 3 nuclear facility.
- 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: 4.0×10^{22} $m^{-2}s^{-1}$)
  - (c) the ability to handle radioactive materials (STAR limit: < 100 mR/hr = 10 μSv/hr)
  - (d) the ability to handle beryllium

Comparison of plasma parameters among Existing and proposed US Linear Plasma Devices

	PISCES-B (UCSD)	TPE (INL)	MPEX (ORNL)
Deuterium ion flux: Γ_i ($m^{-2}s^{-1}$)	10^{21}–10^{23}	10^{20} – 3.7x10^{22}	>10^{23}
Incident ion energy: E_i (eV)	20–300 (bias)	50–200 (bias)	??
Electron temperature: T_e (eV)	4 – 40	5 – 20	3-50
Ion temperature: T_i (eV)	2 – 5	2 - 5	1 - 200
Electron density: n_e (m^{-3})	10^{18}–10^{19}	10^{16} – 3.5x10^{18}	10^{18}-3x10^{19}
Max. heat flux: P_{max} (MW/m^2)	5	~1.2	20
Plasma diameter (mm)	75	50	120
Max. specimen size	ϕ ~ 25.4 mm disc	ϕ ~ 50.8 mm disc	100 x 100 mm plate
Pulse length (s)	Steady state	Steady state	Pulse and Steady state
Activated targets	No	Yes	Yes
Tritium	No	Yes	No
Beryllium	Yes	Yes/No*	Yes
Permeation capability	No	Yes**	No
Ion incident angle	Normal	Normal***	Inclined and Normal
Plasma source (cathode)	Reflex arc (LaB_6)	Reflex arc (LaB_6)	Helicon (no cathode)
Year of operation	Since 1988	Since 1989	Proposed phase
Unique capabilities	In-situ surface analysis, transient surface heating, beryllium testing	Tritium use and diagnostics, neutron irradiated materials	Electrodeless plasma (Helicon + ECH + ICH) minimizes plasma contamination by impurity

NOTES:
*: Beryllium has been extensively tested in TPE during it tenure at TSTA, LANL in 90's, 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.

Status update on: TPE modification (electrical system upgrade)

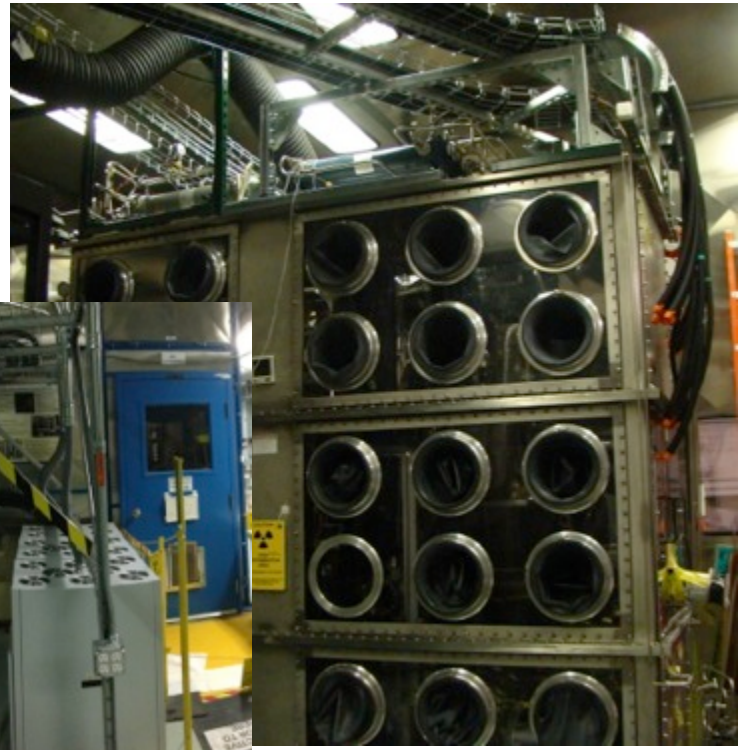
- Background and *Safety issue in operation with previous TPE configuration*
  - *Heat issue (up to 95-100 F for tritium operation) in contamination area (CA)*
  - *No space to put chair and desk in the current control room (inside Permacon)*
  - *Exposure to tritium and beryllium*
  - *Existing/old power supply unable to remote control and setup safety features*
- *Decision was made in 2013 to setup new power supplies and control room outside of Permacon in order to eliminate the above four safety issues.*

Current power supply location and control room (inside Permacon)

New power supply location and control room (outside Permacon)

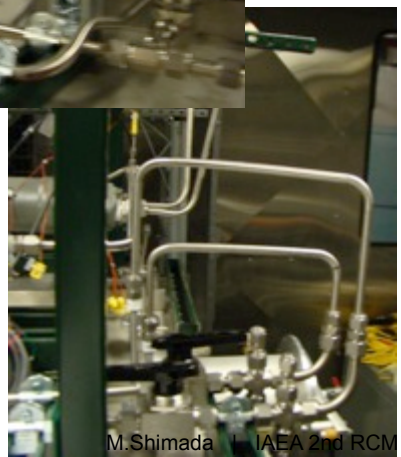
Status update on:
TPE modification (electrical system upgrade)

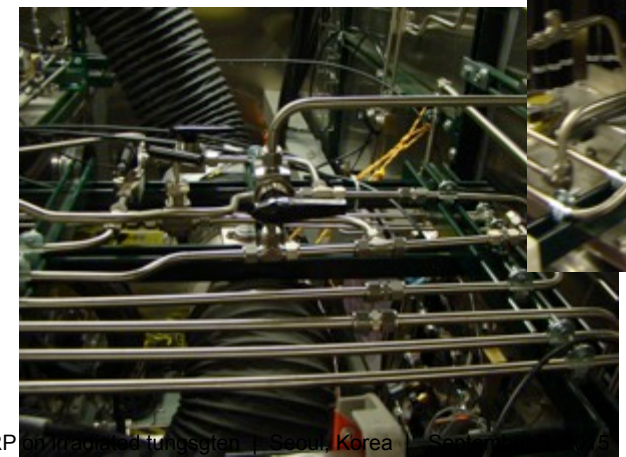
- **Completed tasks:**
  - *Installation of electrical breakers, disconnects, AC supply lines completed in July 2014*
  - *Installation of DC cables (total length: 1000 ft ~ 300m) in Permacon (contamination area) completed in September 2014*
  - *Installation of new cooling manifold inside TPE's Ventilated Enclosure (high contamination area) completed in December 2014*
  - *Revision of the Work Control Document has been approved in February 2015*
  - *Establishment of new radiological work permit (RWP) for whole body entry into Ventilated Enclosure (high contamination area) in February 2015*



Status update on: TPE modification (electrical system upgrade)

- **Completed tasks:**
  - *Removal of old cooling lines, and installation of new cooling lines/flow meters and pressure gauges inside Permacon (outside Ventilated Enclosure)*
- **On-going activities:**
  - *Installation of data acquisition system and safety interlock is underway*
  - *Installation of new DC cables, cooling lines, pressure gauges, and thermocouples inside Ventilated Enclosure (High Contamination Area) is underway*
  - *Installation of uninterrupted power supplies (UPS) as standby power for loss of electricity during full power plasma operation*
  - *First plasma after TPE modification is expected in September/October 2015.*



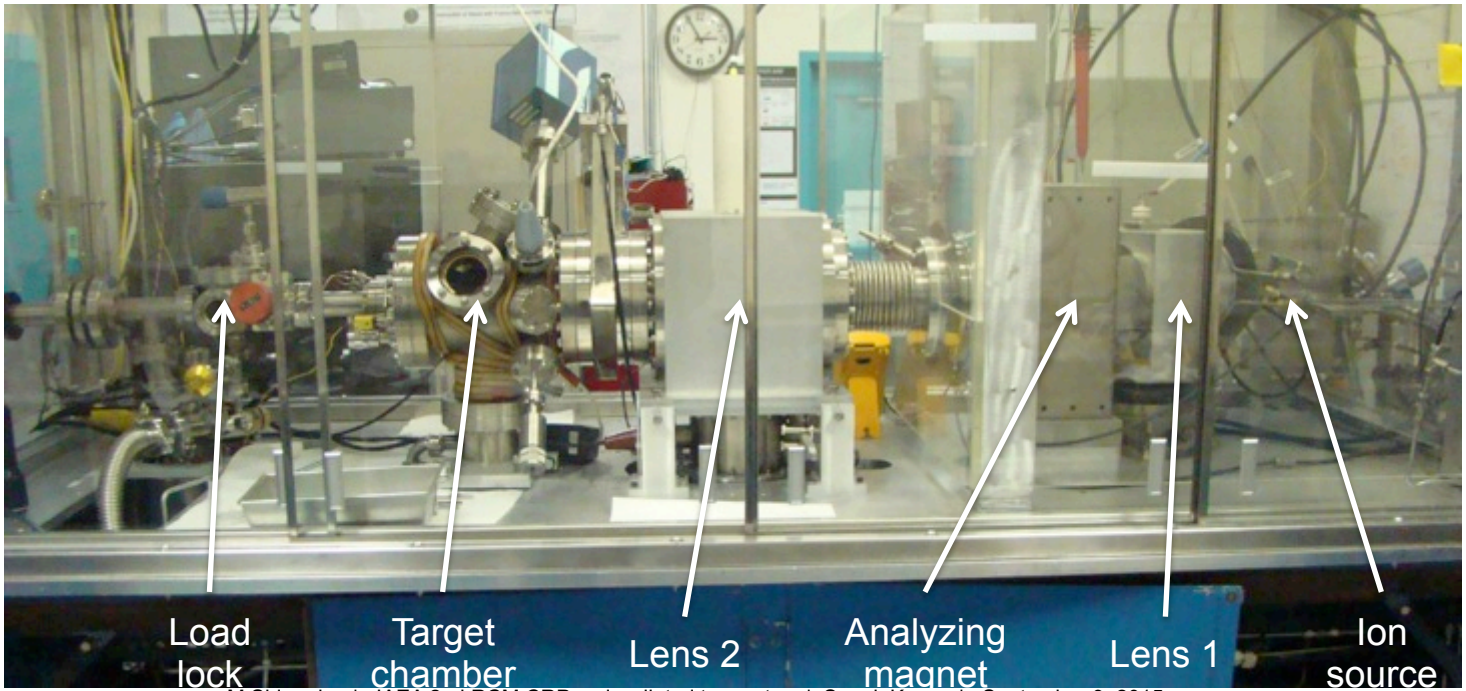


Status of supporting experiment at STAR for this CRP

- Neutron Irradiated Materials Ion Implantation eXperiment – NIMIIX
  - *To investigate PMI on low flux/fluence condition*
- Tritium Gas Absorption Permeation – TGAP
  - *To investigate gas-driven tritium permeation on irradiated tungsten*
- Static Gas Absorption System – SGAS
  - *To investigate deuterium trapping in irradiated tungsten*
- Thermal Desorption Spectroscopy – TDS
  - *To measure total D and He retention in irradiated tungsten up to 1100 °C*
- Doppler Broadening Positron Annihilation Spectroscopy – DB-PAS
  - *To qualitatively characterize radiation defect in irradiated tungsten*

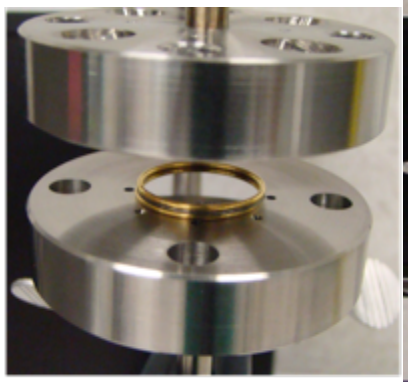
Neutron Irradiated Material Ion Implantation eXperiment

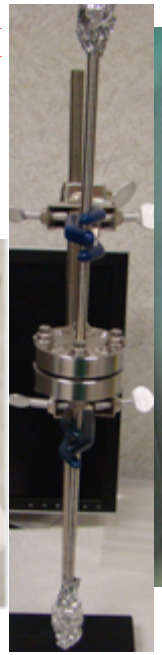
- NIMIIX is located in non-tritium side at the STAR
  - *Base pressure: low 10^{-8} torr*
  - *Capable of handling neutron irradiated materials*
  - *Ion is created in duoplasmatron ion source*
  - *Mass analyzed beam*
  - *Flux: $1x10^{19}$ $m^{-2}s^{-1}$*
  - *Species: D, He*

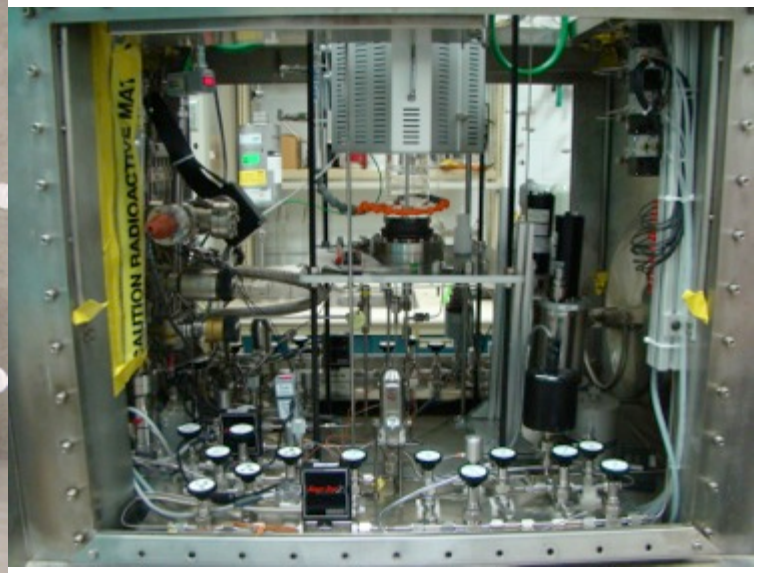


TGAP:
Tritium permeation in irradiated tungsten

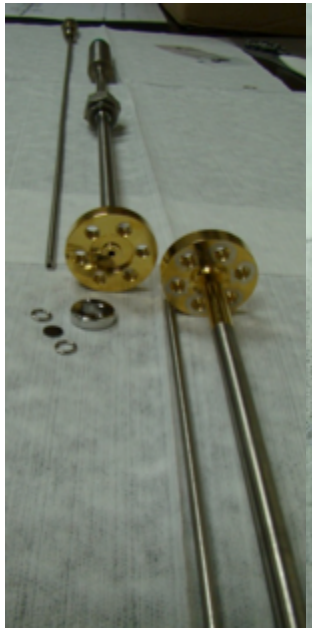
Test section for NFRI ARAA







Test section for PHENIX W

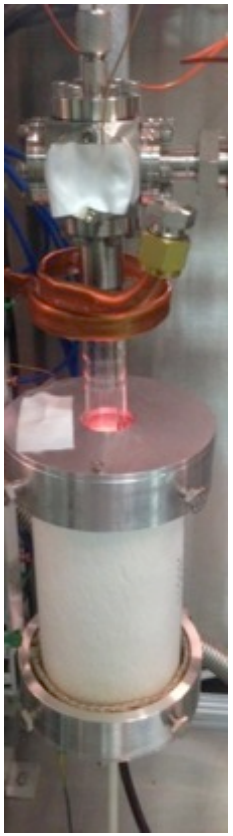


Unique capabilities

- Designed to measure tritium transport properties (e.g. diffusivity, solubility, and permeability) in activated materials at realistic fusion sweep gas conditions (e.g. low tritium <10 Pa & hydrogen partial pressures < 1000 Pa, moderate < 700 C)
- Capable of testing liquid or ceramic breeder materials (e.g. PbLi, Li_2TiO_3, etc.) and disc shaped metal specimens (W, RAFM steels, etc.)

SGAS:
Deuterium trapping/absorption in irradiated tungsten

- This experimental apparatus studies deuterium gas absorption in materials. It is located inside Laboratory fume hood (STAR RM 103)
- **Unique capabilities**
  - Sub atmospheric absorption (< 0.1 MPa) up to 950 C
  - Utilize three calibrated capacitance manometers (0.002, 1.3, 133 kPa)
  - Capable of testing liquid or ceramic breeder materials (e.g. PbLi, Li_2ZrO_3, KO functional materials etc.) and PFC/structural material (W, RAFM steels, etc.)
  - Capable of testing neutron-irradiated materials

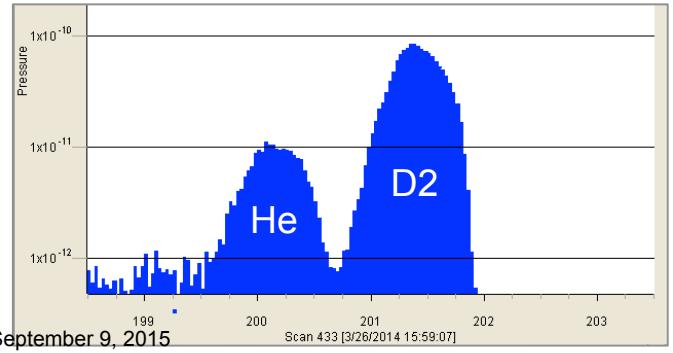


Thermal desorption spectroscopy (TDS)

- Both TPE and NIMIIX study retention.
- To investigate retention,

$$R = \frac{\Phi_{out}}{\Phi_{in}}$$

- Heat samples up to 1100ºC
  - *Vacuum infrared tube furnace*
  - *Linear ramp rate*
  - *6 calibrated leaks (x3 He and x3 D2)*
- Analyze residual gases
  - *Two quadrupole mass spectrometers*
  - *High resolution QMS can distinguish:*
    - D2: 4.0282 amu
    - He: 4.0026 amu



Doppler Broadening Positron Annihilation Spectroscopy

- Positron annihilation spectroscopy
  - *Diagnostic to assess relative defect concentration.*
  - *Correlate with TDS data.*
- First PAS measurements at STAR: April 2015
  - *Commissioned LN2 auto fill system.*
  - *Zero dose measured on dosimetry rings.*
  - *Problems with detector. Returned to IRC. Found incorrect vent plug. Testing Will return to STAR in July.*
- Future collaboration with Brian Wirth (UTK) and ORNL.

Summary of INL activities for this CRP

- TPE will be restarted with electrical system next month
  - *This enable safer plasma operation and achieve much higher ion flux density (> 10^{23} $m^{-2}s^{-1}$), will be ready for new (> 100) neutron-irradiated tungsten specimens*
- PHENIX will start HFIR neutron-irradiation to W from Feb.2016 to April 2017
  - *This irradiation will provide large number (> 100) of one-of-a-kind neutron-irradiated tungsten specimen*
  - *Plasma-driven D retention in neutron-irradiated tungsten will be investigated with TPE, NIMIIX*
  - *Gas-driven D retention in neutron-irradiated tungsten will be investigated with SGAS*
  - *Plasma-driven T permeation in neutron-irradiated tungsten will be investigated with TPE*
  - *Gas-driven T permeation in neutron-irradiated tungsten will be investigated with TGAP*
- By the 3rd RCM (spring 2017), we should have some preliminary results from these new PHENIX specmens

Supporting slides

Critical remaining issues:

1. Radiation damage recovery temperature
   - Recovery temperature
   - Activation of defect migration
   - Defect characteristic etc.
2. Trapping mechanism(s) of tritium in radiation damage
   - Maximum trap concentration,
   - Trap concentration profile throughout material (on the order of mm),
   - De-trapping energy of tritium from trap site
   - Need to distinguish radiation damage from transmutation effects etc.
3. Diffusion mechanism(s) of tritium to bulk (>10 μm) tungsten under ITER/ DEMO divertor relevant condition
   - High ion flux (>10^{22} $m^{-2}s^{-1}$)
   - Mixed (He, D, and T) plasma
   - High temperature (>500 C)
4. Mitigation of deep diffusion of tritium to bulk (>10 μm) tungsten
5. Removal of tritium from bulk (>10 μm) tungsten

NOTE:

Neutron-irradiation is the best (and only) option to create high dose (> 1 dpa) radiation damage in bulk (>10 μm) W

Enhanced diffusion to simulate plasma implantation

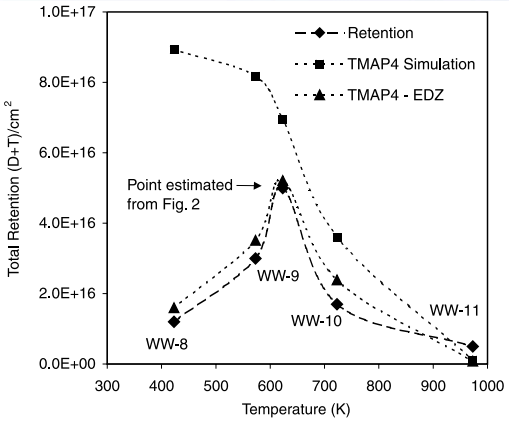
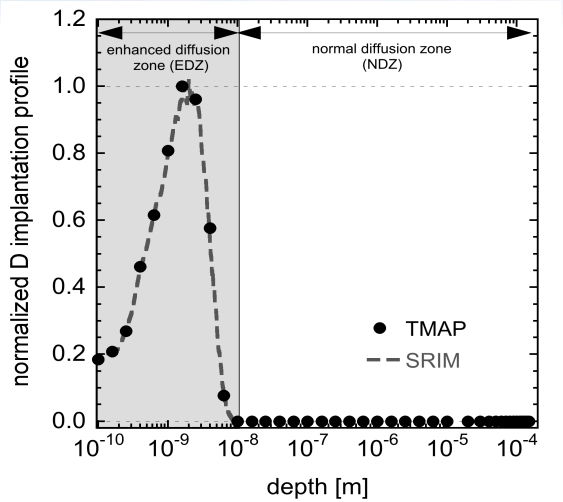


Fig. 1. Experimental retention results (◆), TMAP4 simulation using best-fit parameters (■), and TMAP4 retention results using $C_0 = 0$ and enhanced diffusion in the implant zone (▲).



- TMAP does not treat plasma surface physics (e.g. erosion, sputtering and deposition).
- TMAP can simulate the ion implantation with ion implantation profile
- Venhaus and Causey[1,2] artificially increased hydrogen diffusivity in the Enhanced Diffusion Zone (EDZ) (<10 nm) during plasma implantation in order to reduce the mobile concentration and retention at low temperature in tungsten.
  - *During plasma implantation:*
    - Enhanced diffusivity (D_{diff}^{EDZ}) of (0.5-2.0)x10^{-9} m^2/s[1] matches the experiment well.
  - *During thermal desorption:*
    - Diffusivity (D_{diff}) by Fraundelder[4]

References:
[1] R.A. Causey, et. al. J. Nucl. Mater. 266 (1999) 467.
[2] T. Venhaus and R.A. Causey, J. Nucl. Mater. 290 (2001) 505.
[3] T. Venhaus and R.A. Causey, Fus. Technol. 39 (2001) 868.
[4] R. Frauenfelder, Vac. Sci. Tech.,6 (1969) 338.