Tritium Focus Group meeting
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Tritium Plasma Experiment and its role in PHENIX program

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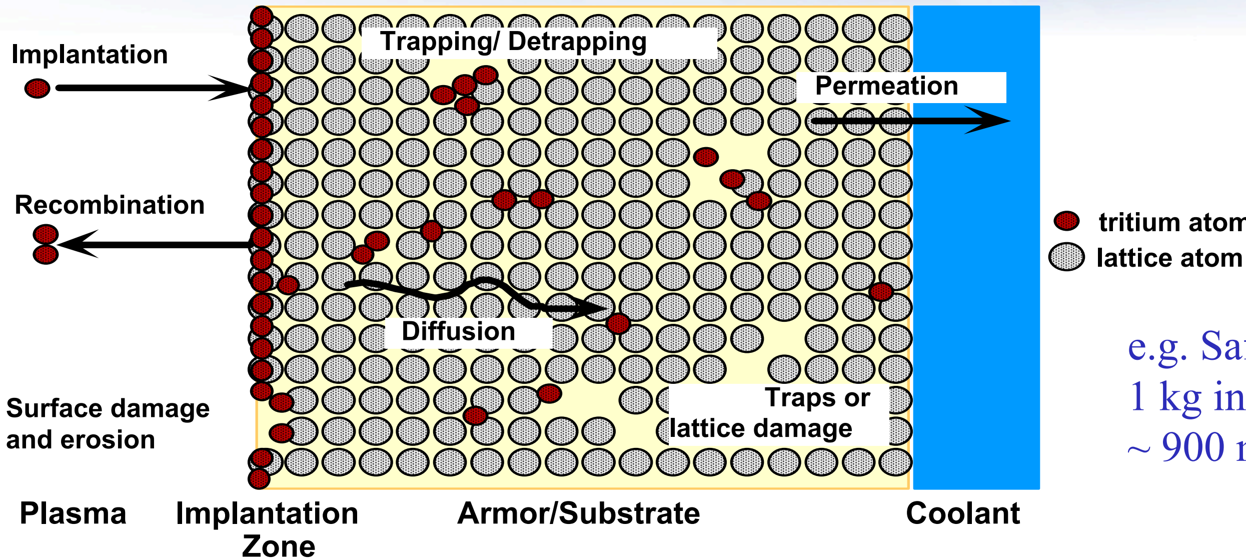
Outline:

1. Motivation
2. Tritium Plasma Experiment
3. INL/STAR's role on US-Japan collaboration
4. Role of TPE in PHENIX project
5. TPE modification and development of plasma-driven permeation

Safety concern: "in-vessel inventory source term"



Reference: ITER GSSR 2004

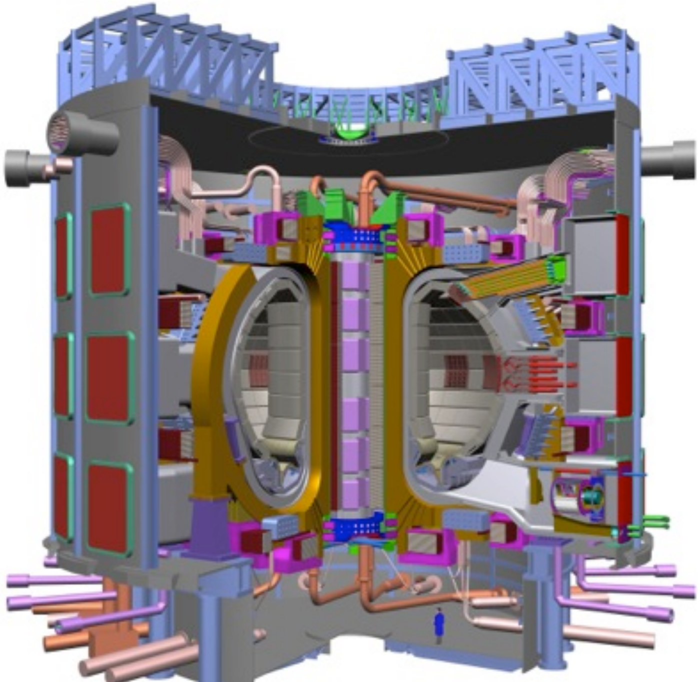
e.g. Safety Limit in ITER:
1 kg in-vessel tritium inventory in
~ 900 m^2 PFC surface area

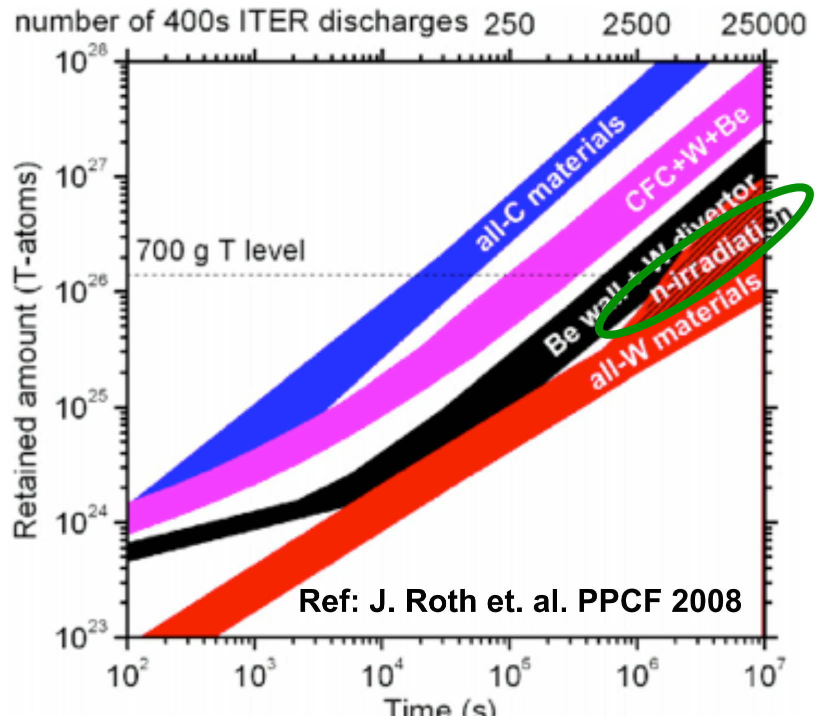
- Challenges in neutron-irradiated plasma facing components (PFCs)
  - Radiation damages (vacancy, vacancy-cluster, void etc.) will be created by 14 MeV throughout PFCs thickness, becoming trapping site for tritium
  - Large amount of tritium can be trapped in vacancy-cluster as gas form, leading to bubble formation, and blister formation in metal
  - Tritium behavior in the fusion nuclear environment is not fully understood

➔ There exists large uncertainty in tritium retention assessment in neutron-irradiated PFCs

Safety in material selection:

- *Safety plays a major role in material selection:*
  - *e.g.* Carbon was excluded to use in the tritium phase of operations due to unacceptable levels of tritium retention in co-deposited carbon layers.
- *Question: How about tritium in tungsten?*
  - *Tungsten, a candidate PFC for the divertor in ITER, is expected to receive a neutron dose of 0.7 dpa by the end of operation in ITER, and >10 dpa in FNSF and DEMO.*





Ref: J. Roth et. al. PPCF 2008

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Brief history of TPE and host tritium facilities (1/2)

I. 1983-early 1990's: Tritium Research Laboratory (TRL), SNL-Livermore

- Established as the ***Tritium Plasma eXperiment (TPX)*** and operated for 10 years
- RF driven plasma (390 MHz) up to 200 W; axially magnetized to ~ 150 G, plasma density ~$3x10^{11}$ ions/cm^3, Te ~ 10 eV, on-sample ion flux 10 mA/cm^2
- Performance: T throughput ~ 0.1g/day; experiment placed in a high-velocity ventilation hood for T contamination control; pumping system exhausted to TRL vacuum effluent recovery system, diagnostics included Langmuir probes, QMS (plasma species and permeation species), in-situ AES
- Decision was made to upgrade the TPX, and then close the TRL in 1992

II. early 1990's-2002: Tritium Systems Test Assembly (TSTA) at LANL

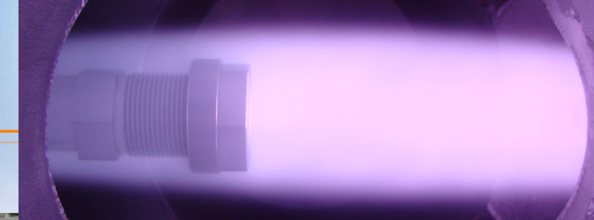
- Rename as the ***Tritium Plasma Experiment (TPE)***, and upgraded to hot cathode reflex arc w/ LaB_6 source, returned to tritium operation in 1995, and operated for 7 yrs
- Performance: Increased maximum T throughput to ~ 0.5 g/hr; direct-feed of T from TSTA facility, or local T source from a U-Bed; T effluent captured on U-Beds, ion fluxes up to 1 A/cm^2 and 100 - 200 eV energy, increased pumping speed to 2200 l/s, diagnostics included Langmuir probes, QMS.
- System placed in a glovebox with atmosphere T monitoring and purge gas control
- Decision was made to close the TSTA and relocate the TPE.

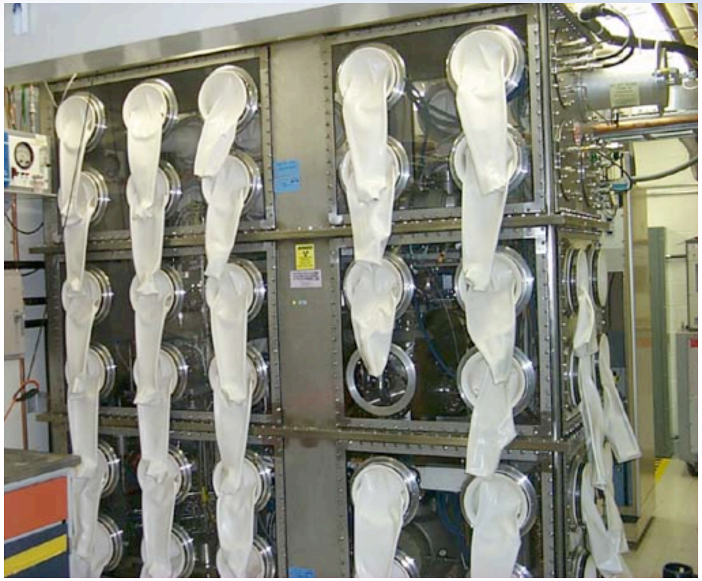
Brief history of TPE and host tritium facilities (2/2)

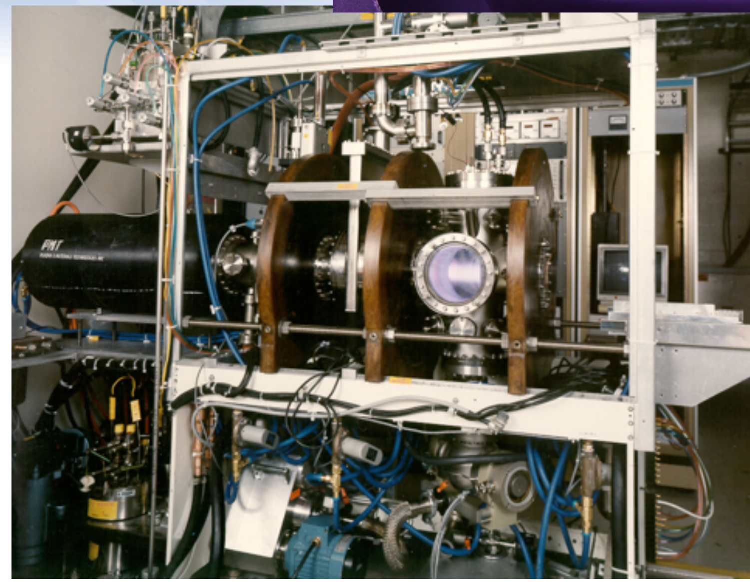
III. 2002-present: Safety and Tritium Applied Research (STAR) facility, INL

- Tritium contamination level as high as 300,000 dpm / 100 cm^2 located within instrument racks and power supply chassis (CA limit is 10,000 dpm / 100 cm^2).
- Decontamination efforts unsuccessful at reducing levels below CA limit.
- Substantial facility modifications were made to build a PermaCon enclosure (CA boundary), re-route and expand electrical service, modify facility ventilation, extend the fire suppression system into the PermaCon.
- Returned to deuterium operation in 2005, and returned to tritium operation in 2009.
- Performance: maximum T throughput ~ 0.05g/day; experiment placed in a ventilated enclosure (HCA boundary) and Permacon enclosure (CA boundary); local T source from a 300 cc cylinder; T effluent captured on U-Beds, ion fluxes up to 1 A/cm^2 and 100 - 200 eV energy, decreased pumping speed to 900 l/s, diagnostics included Langmuir probes, QMS, and optical spectrometers.
- *New capabilities at STAR*:
  - Handling of "neutron-irradiated materials"
  - Cutting tritiated material in ventilated enclosure
  - Plasma-driven tritium permeation capability

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: 4.0×10^{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
- Plasma-driven tritium permeation capability (under development)

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.

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Past and present US-Japan collaborations:

I. JFY2001-2006 (Apr.2001-Mar.2007):

– The second Japan/US Program on Irradiation Test for Fusion Research (JUPITER-II)

- Corrosion and purification of molten salt (FLiBe)
- Mass transport of tritium in FLiBe

II. JFY2007-2012 (Apr.2007-Mar.2013):

– Tritium, irradiation, and thermofluid for America and Nippon (TITAN)

- Mass transport of tritium in lead lithium eutectic and development of tritium permeation barrier materials
- Tritium retention in HFIR neutron-irradiated tungsten

III. JFY2013-2018 (Apr.2013-Mar.2019):

– PFC evaluation by tritium plasma, heat, and neutron irradiation experiment (PHENIX)

- Tritium behavior (retention, diffusion, and permeation) in HFIR neutron-irradiated tungsten

PHENIX 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.

PHENIX project

Idaho National Laboratory

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 INL)
Plasma-Surf. Interac.
Tritium Behavior

Alternative facilities with similar capabilities are also acceptable.

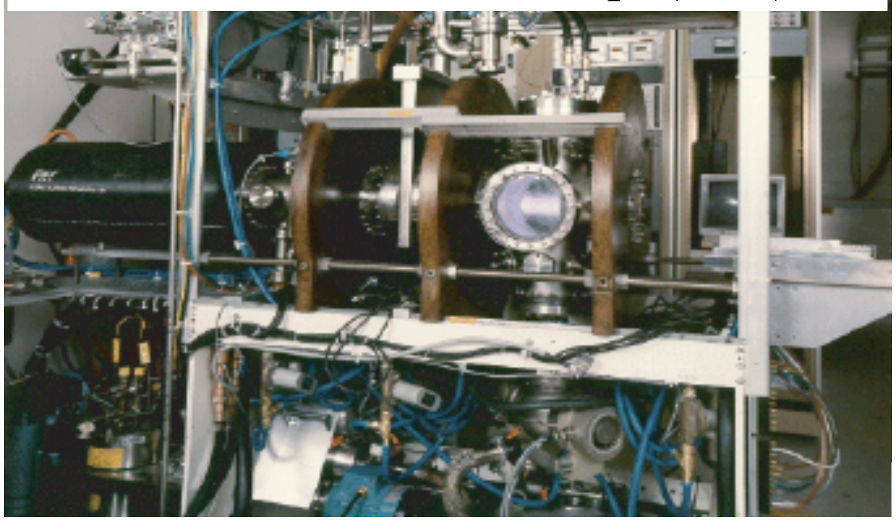
Primary facilities (US)

Task 1 : IR facility

Task 2 : HFIR

He loop in GIT is under consideration

Task 3 : Tritium Plasma Exp. (TPE)

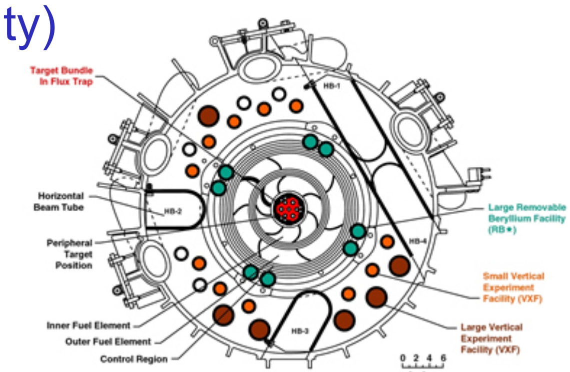


Research at ORNL and INL

New challenges in PHENIX project (2013-2019):

Tritium behavior in realistic fusion nuclear environment

- Tritium behavior (retention and permeation):
  - Deuterium/tritium retention by high-flux TPE plasma
  - Gas-driven permeation in TLLE
  - Plasma-driven permeation in TPE
- Realistic plasma conditions in divertor:
  - High plasma exposure temperature (500-1000C)
  - Mixed (D/T/He) plasma
  - High flux (10^{22} – 10^{23} $m^{-2}s^{-1}$), high-fluence (10^{26} – 10^{27} m^{-2})
- Realistic fusion neutron irradiation
  - Larger irradiation port (removable beryllium facility)
  - HFIR irradiation with thermal neutron shielding
  - High irradiation temperature (500-1300C)
  - Deuterium gas environment



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TPE modification

- *Background and Safety issue in TPE operation with previous setup*
  - *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 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 of TPE modification (outside CA):

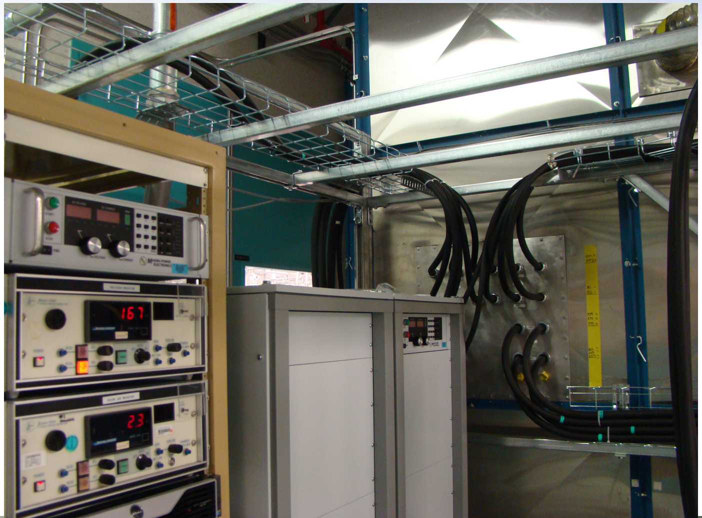


Photo of Highbay area on September 18 after DC cables installation

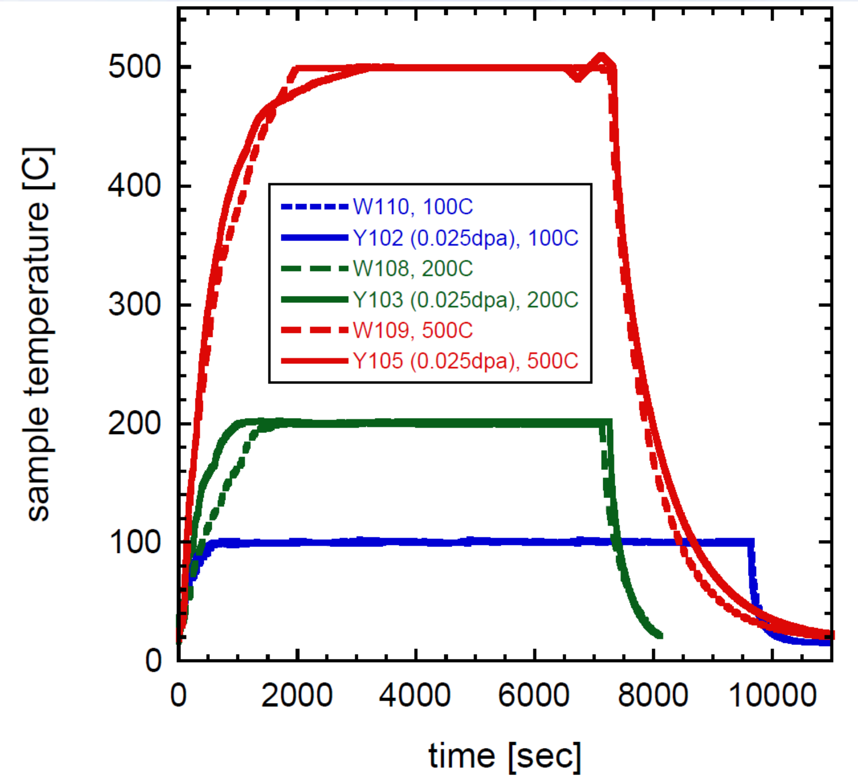
Status of TPE modification (inside CA):

Photo of inside CA in September 17 after DC cables installation

Status of TPE modification:

- Installation of electrical breakers, disconnects, AC supply lines was completed in July 2014.
- Installation of DC cables (total length: 1000 ft ~ 300m) in Permacon (Contamination Area) was completed in September 2014.
- Installation of data acquisition system and safety interlock is underway.
- Installation of new cooling lines and manifold inside Ventilated Enclosure (High Contamination Area) is underway.
- Installation of new pressure gauges and thermocouples inside Ventilated Enclosure (High Contamination Area) is underway.
- Work Control Document will be revised in October – November 2014.
- First plasma after TPE modification is expected in December 2014.

Temperature profile with old target holder

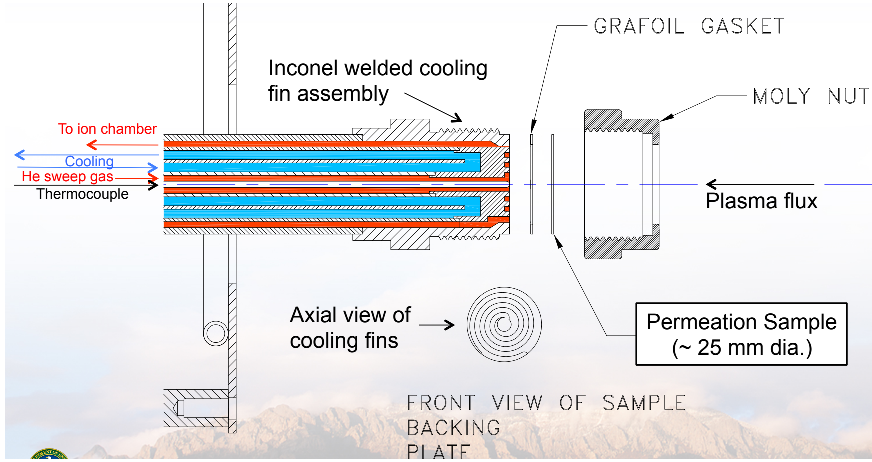


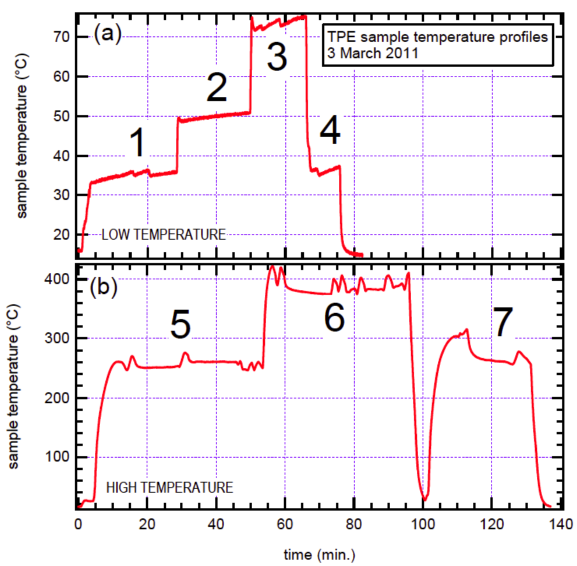
- Sample temperature is determined by plasma heating and heat conduction to the cooling plate
- At higher temperature (> 500 C), it takes 20-30 min to reach the desired temperature and 20-30 minutes to cool down to room temperature
- Maximum sample temperature obtained with old target holder was 700 C

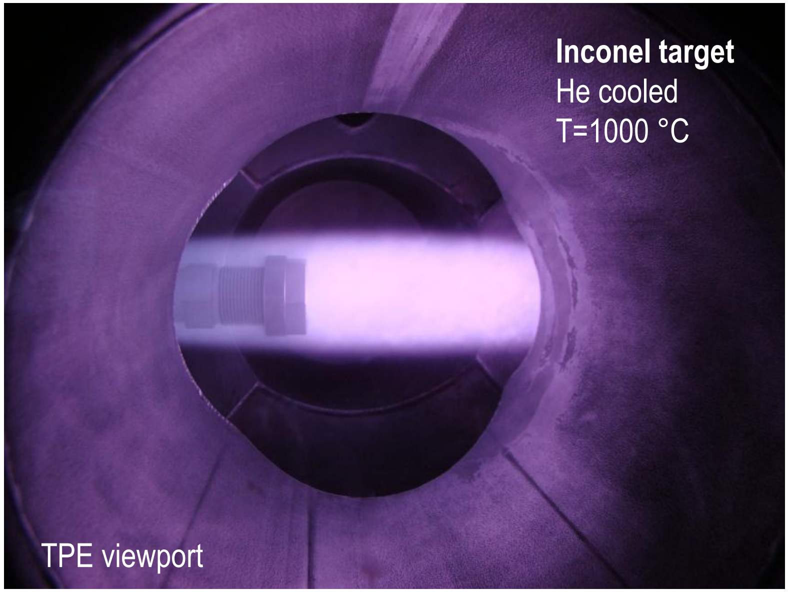
Development of plasma-driven tritium permeation in TPE

Motivation: There is no plasma (ion)-driven permeation data from high flux device

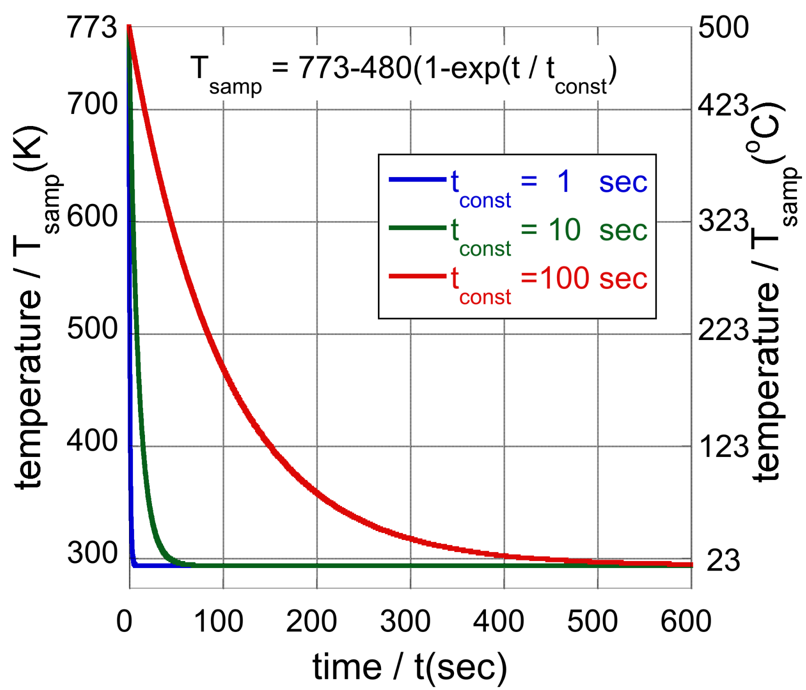
- Key challenge for plasma-driven permeation: stable operation at high temperature.
- Developed two retention stages (Cu and Inconel) to test new design concept.
- Leveraged concentric cooling channel design from PISCES.
- Successful testing Inconel target to T=1000 °C using He cooling.







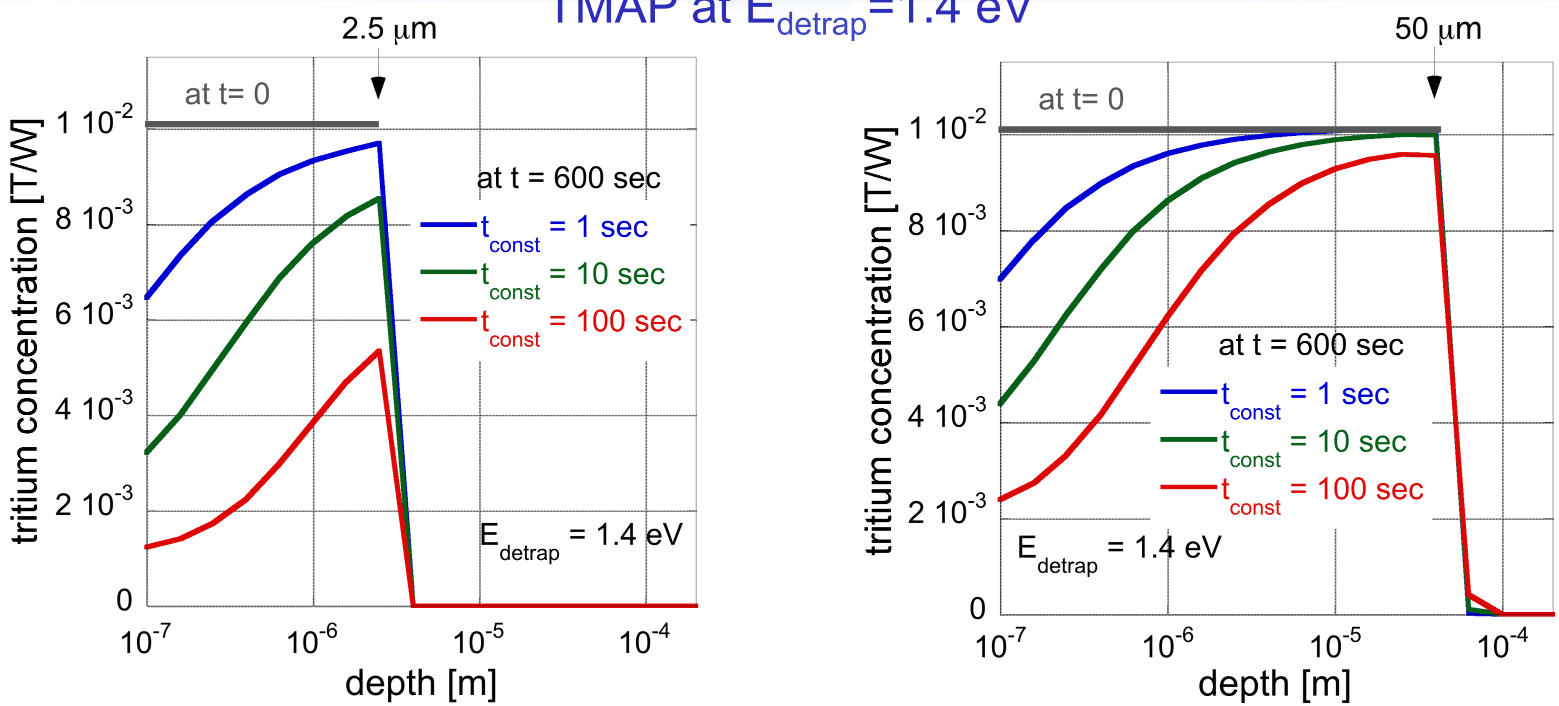
Effect of temperature profile during cooling down



- We need to know temperature profile not only during plasma/ion implantation but also during cooling down especially for high temperature case (> 500°C)
- There exists the machine dependence when comparing the D/T retention data
  - Low flux machine (< 10^{21} $m^{-2}s^{-1}$), t_{const} > 100 sec due to active heating
  - High flux machine (> 10^{22} $m^{-2}s^{-1}$), 10 < t_{const} < 100 sec due to active cooling
  - t_{const} depends on the location (divertor plate, strike point, dome) in ITER

Ion-damage W vs. neutron-irradiated W

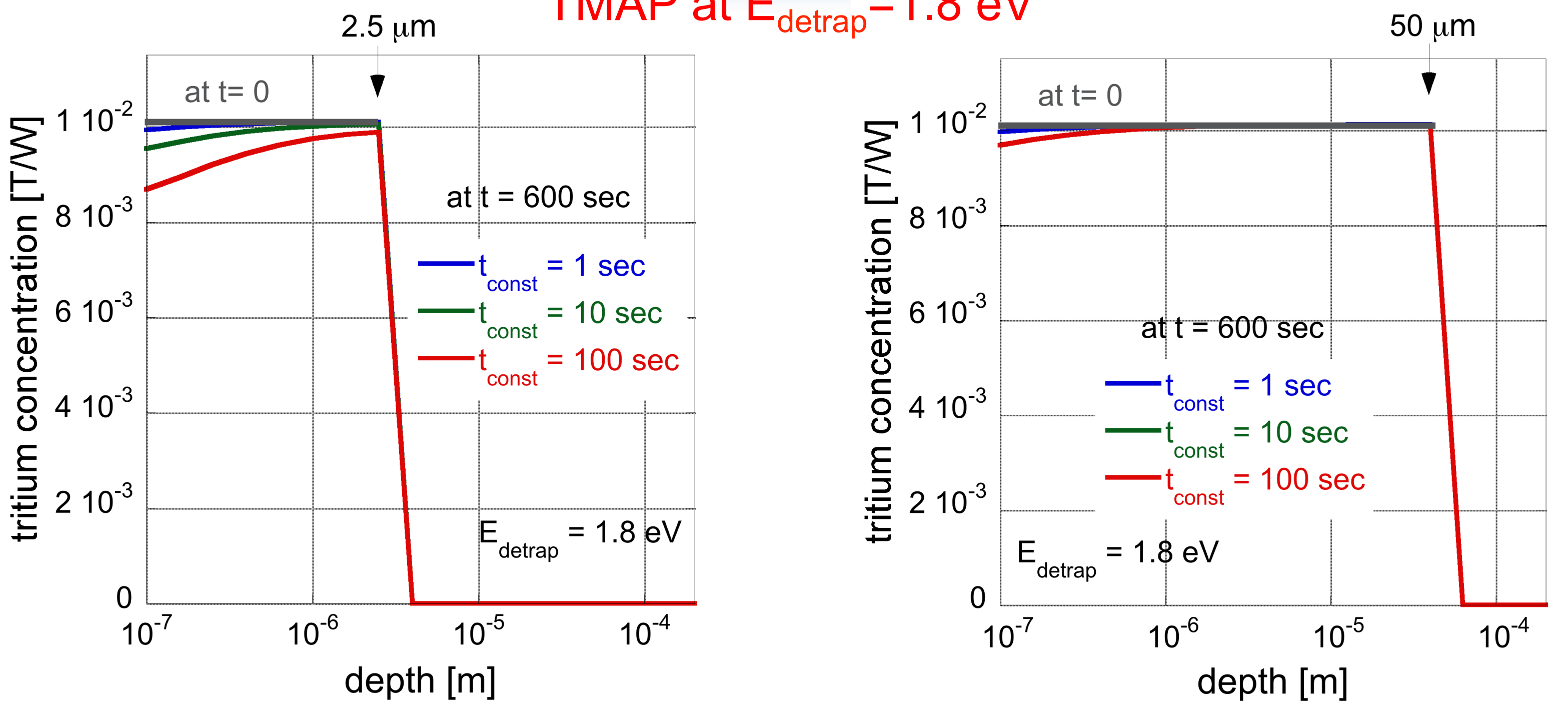
TMAP at E_{detrap}=1.4 eV



1. The trap concentration will most likely saturate at 1 at.% at > 1 dpa
2. T is trapped in ion-damaged range (2.5 μm) vs. neutron-irradiated (>50 μm)
3. T depth profile at high (> 500C) temperature:
   - For E_{detrap}=1.4 eV, Most of T can be desorbed during ramp-down if t_{const} >100 sec

Ion-damage W vs. neutron-irradiated W

TMAP at E_{detrap}=1.8 eV



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3. T depth profile at high (> 500C) temperature:
   - For E_{detrap}=1.4 eV, Most of T can be desorbed during ramp-down if t_{const} >100 sec
   - For E_{detrap}=1.8 eV, T is still trapped during ramp-down if t_{const} <100 sec

Summary:

- Tritium Plasma Experiment (TPE) is the unique linear plasma device to study tritium behavior in plasma-facing components (PFC) materials.
- With US-Japan collaboration PHENIX project, TPE will investigate tritium retention in neutron-irradiated tungsten.
- Electrical systems and power supplies are replaced to enhance operator safety in TPE
- TPE plan to restart by the end of December 2014.
- Plasma-driven permeation holder will be tested in 2015.