Tritium issues in plasma wall interactions

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Tritium Science and Technology for Fusion
http://tritium.nifs.ac.jp/

Tritium
Radioactivity requires safety handling.
Limited resource requires effective breeding and recovery.

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Part I Tritium issues in a fusion reactor

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DT fusion reactor (Ignition and continuous burning)

\[ D + T = ^3\text{He} (3.7\text{MeV}) + n (14\text{MeV}) \]

To establish reliable and safe tritium fuel cycles and safe tritium confinement to build economic and safety fusion reactor

Encouraging young scientist and students

A brief introduction of our research project started at 2007 in Japan (Tritium is so important)
The main aim of this project is to establish tritium safety in a D–T fusion reactor. Since huge amount of radioactive tritium must be introduced into the reactor as a fuel, we are facing to lots of safety concerns newly appeared to be solved.

Main efforts will be to establish tritium safety in (1) a fueling system keeping continuous D–T burning, (2) tritium exhausting, recovering and refining processes, (3) a tritium breeding system with a breeding rate over 1.05, and (4) tritium monitoring and accounting systems.

In addition, easy isotopic exchange reactions of tritium with hydrogen in water and hydrocarbons result in the contamination of the systems, which require decontamination techniques. The project also aims to provide new insights into basic tritium science and technology.
Limited resource requires safety T breeding system compatible with power production and T breeding

- Recycling of fugue amount of T
- Safety confinement to avoid possible contamination
- Difficulty of extrapolation of limited experience of T handling to fusion system
- Poor understanding of isotope effect

Production of hazardous inorganic tritium

Contamination by permeation and leakage
  Multi step contamination

ITER at France and a Test reactor in Japan require large numbers of tritium experts.
2. Characteristics of a DT reactor as an energy source

Already 50 years has passed after finding nuclear reactions give energy.

Fission reactors are already established as energy sources.

Why much longer time has been required for fusion than fission?

Significant amount of energy is required to overcome Coulomb potential.

The first priority has been plasma confinement to establish DT burning, and we will soon attain $Q=10$ in ITER.

But this is not enough for a fusion reactor to be an energy source!!.

Lots of engineering issues are remained to be solved.

Tritium safety and economy are critical issues.
Hydrogen related 5 fusion reactions

\[
\begin{align*}
D + T & \rightarrow ^4\text{He} + n + 17.6\text{MeV} \quad (1) \\
D + D & \rightarrow ^3\text{He} + n + 3.25\text{MeV} \quad (3) \\
T + T & \rightarrow ^4\text{He} + 2n + 11.3\text{MeV} \quad (4) \\
D + ^3\text{He} & \rightarrow ^4\text{He} + H + 18.3\text{MeV} \quad (5)
\end{align*}
\]

Easiest reaction is DT reaction (1)
\[
D + T \rightarrow ^4\text{He} (3.5\text{MeV}) + n(14.1\text{MeV})
\]
plasma heating  Energy and T breeding

The D\(^3\)He reaction is very much attractive for no neutron production, though accompanying DD reactions do produce it.
### Comparison of fission and fusion as energy sources

<table>
<thead>
<tr>
<th></th>
<th>Fission reactor</th>
<th>Fusion reactor</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Energy Input</strong></td>
<td>Nearly zero</td>
<td>Huge energy required</td>
</tr>
<tr>
<td></td>
<td>Poor fueling efficiency</td>
<td>Poor fueling efficiency</td>
</tr>
<tr>
<td><strong>Energy conversion</strong></td>
<td>Energy carried by fission products (FP, heavy ions) (~170 MeV) is deposited in fuel pins.</td>
<td>Energy carried by neutron (14 MeV) must be converted in large volume of blanket system</td>
</tr>
<tr>
<td><strong>Fuel breeding and recovery</strong></td>
<td>One fission produces more than 2 neutrons, easy to keep chain reactions and to breed fuels.</td>
<td>To keep breeding ratio more than 1, we need neutron multipliers (Be, Pb).</td>
</tr>
<tr>
<td></td>
<td>Fuel pins retain both FP and new fissile and spent fuels are reprocessed to remove/recover them.</td>
<td>Tritium breeding and energy conversion must be done simultaneously.</td>
</tr>
<tr>
<td><strong>Nuclear Waste</strong></td>
<td>Long life radioactive FPs must be handled with special care and will be reposed deeply under ground.</td>
<td>Waste is limited to activated structure materials, could be recycled.</td>
</tr>
</tbody>
</table>
Coolant (water in LWR, Na in FBR)

Cross section of Fuel pin for FBR

\[ ^{235}\text{U} + n \rightarrow \sim^{120}\text{FP(1)} + \sim^{110}\text{FP(2)} + \sim2.4n + \sim180\text{MeV} \]

All energy is deposited in fuel pin

Little Tritium!!

For chain reaction

\[ n \rightarrow 2n \]
T resource is very limited → need T breeding

\[
\text{D} + \text{T} \rightarrow ^4\text{He} \ (3.5\text{MeV}) + n\ (14.1\text{MeV})
\]

plasma heating Energy and T breeding

– Deuterium can be extracted from natural water (SMOW (standard mean ocean water) contains 0.016% D)

– Tritium must be imported (limited) or bred internally from lithium
  • 56 kg tritium is required per GW year (thermal) of fusion power
  • About 100 g tritium is produced per year in a standard CANDU fission unit
  • 20 to 25 kg tritium (mainly in Canada) will be available for operation of ITER
  • Tritium must be bred by reactions in blanket systems
    \[
    ^6\text{Li} + n \rightarrow \text{T} + ^4\text{He} + 4.8\text{MeV}
    \]
    \[
    ^7\text{Li} + n \rightarrow \text{T} + ^4\text{He} + n \rightarrow 2.5 \text{ MeV}
    \]
    \[
    ^9\text{Be} + n \rightarrow 2n + 2^4\text{He} \rightarrow 2.5 \text{ MeV}
    \]
    \[
    ^{\text{A}}\text{Pb} + n \rightarrow 2n + ^{\text{A-1}}\text{Pb} \rightarrow 7 \text{ MeV}
    \]
  • Overall breeding ratio is expected to be above ~1.1 (must)

Very hard to attain
3. General safety issues in a fusion reactor

Fusion Safety Issues (General) are mostly owing to tritium and neutron activated materials because

- **The Fusion Process Is Inherently Safe**
  - No chain reaction
  - Reaction is thermally self-limiting
  - Limited to a few second burn without re-fueling
  - Power/energy densities in the reactor and plasma are low
  - Reaction products
    - Helium (totally inert)
    - Neutrons
      - Used to breed tritium
      - Absorbed in the surrounding material

- Most serious hazard involve the tritium fuel and activated dust from erosion of plasma facing components
Hazard and Containment

- Principle of defence-in-depth
  - Vacuum vessel
  - Cryostat
  - Building ventilation systems (sub-atmospheric condition)
- Passive safety features (natural physics) are used as extensively as possible
  - In case of active cooling system failure, decay heat from activated materials is low enough that all in-vessel components can be cooled by natural convection
  - Reactor “melt-down” is physically impossible

Environmental Impact

- Currently, materials are not optimized for low-activation under neutron irradiation
  - Can be recycled for re-use after 50-100 years
- In the future, material optimized for low-activation can be readily recycled for use in fusion power-plant reactors.
Public Safety

Emission of Tritium must be As Low As Reasonably Achievable (ALARA)

- **Under normal operation:**
  - Total releases will cause doses below 1% of that of natural background radiation: ~ 2 mSv/year, or 200 mrem/year.

- **Under the worst case:** the most severe hypothetical event, or the holy-Moses-oh-my-God-we-are-all-done-for scenario:
  - Fusion reactor site boundary dose will be less than 50 mSv (5000 mrem).

  - In comparison:
    - 50 mSv/year is the US NRC dose limit for adults working with radioactive material.
    - 100 mSv is considered “low-dose”; correlation with adverse biological effect (e.g. cancer) currently could not be established.
    - Plant workers and fire fighters battling the fire at Chernobyl received 700~13400 mSv of radiation; 20% of them died from radiation effects.

Concerns are coming from Tritium and nuclear activation
Tritium Abundance
(limited resources and regulation for safety)

Natural abundance
$3 \times 10^{18}$ Bq

Required in ITER
~ 5kg
$2 \times 10^{18}$ Bq

Production by cosmic ray
$1.5 \times 10^{17}$ Bq/year

Remaining in atmosphere
$2 \times 10^{19}$ Bq

Production by Atomic bombs
(1945~1973)
$3 \times 10^{20}$ Bq

Release from Nuclear Factory
$4 \sim 8 \times 10^{16}$ Bq/y

Regulation level

Quantitative analysis

Gravimetric
Radiation heat
Volumetric (PVT)
Disintegration (dps)

No single method can cover whole range.
Poor resolution inhibits cross-check
**4. Tritium as a fuel of a DT fusion reactor**

**Tritium** \( ^3\text{T} \rightarrow ^3\text{He} + \beta \text{ electron} \)

- **Half life** \( t_{1/2} = 12.323 \pm 0.004 \text{ years} \)
- **(about ~5.5% is disappearing in a year)**

**Maximum range of electron**
- **Air 6mm**
- **Metals \(<~1\text{mm} \)**

**Shielding of tritium radiation is not really a**
**issue (Except direct exposure of organs)**

- **Max energy** 18.6keV
- **Average energy** 5.7keV

- **Precise measurements of edge**
  **energy would give neutrino mass**

- **Fig. 16.2. A high-resolution spectrum of the tritium beta-particle kinetic energy (open**
  **circles). The closed circles are special low-energy points from another source. The spectrum**
  **was measured by J. J. Simpson, University of Guelph.**

**- Electrons emitted to neighboring molecules would enhance some chemical reactions.**
**- Effect of self irradiation would appear only at very high conc.**
**- Decay heat : 324 mW/1g could enhances \( \text{T} \) release from solid**
Detection of T is rather easy (ex. Imaging Plate Technique) to obtain the 2-D image or profiles of radioactivity image & profiles.

IP is a 2-D radiation detector with high sensitivity and resolution, utilizing a photostimulable phosphor (BaFBr:Eu^{2+}).

* Fig. Energy level diagram for the PSL mechanism in BaFBr:Eu^{2+}
Cross (Multi-step) contamination

Ex. Contamination by gloves in safety box

Metal plates exposed to D plasma in TPL and handled in a T handling glove box

Traces of glove fingers

Possible contamination by permeation
T precipitated on surface behaves differently from bulk T
(when bulk concentration is very low)

<table>
<thead>
<tr>
<th></th>
<th>Fe</th>
<th>Cu</th>
<th>Ni</th>
<th>Mo</th>
</tr>
</thead>
<tbody>
<tr>
<td>Photo surface</td>
<td><img src="image" alt="Fe" /></td>
<td><img src="image" alt="Cu" /></td>
<td><img src="image" alt="Ni" /></td>
<td><img src="image" alt="Mo" /></td>
</tr>
<tr>
<td>Tritium distribution</td>
<td><img src="image" alt="Fe" /></td>
<td><img src="image" alt="Cu" /></td>
<td><img src="image" alt="Ni" /></td>
<td><img src="image" alt="Mo" /></td>
</tr>
<tr>
<td>Tritium activity</td>
<td>704</td>
<td>2380</td>
<td>1070</td>
<td>811</td>
</tr>
<tr>
<td>H solubility at 873K (H/M)</td>
<td>$10^{-5}$</td>
<td>$10^{-5}$</td>
<td>$10^{-3}$</td>
<td>$10^{-7}$</td>
</tr>
</tbody>
</table>

At high concentration, both behave similarly.
But it is quite hard to detect bulk T
Why $T$ surface precipitation occurs?

Production of hazardous inorganic tritium

$T$ can easily radio chemically replace the ubiquitous lighter hydrogen isotopes, above all the protium (H) / deuterium in water and hydrocarbons in air

$$HT + H_2O = HTO + H_2 - \Delta G$$

$$HT + CH_4 = CH_3T + H_2 - \Delta G$$

Also, any solid surfaces absorb water molecules resulting in surface precipitation of $T$

$$M + HTO = MOT + \frac{1}{2}H_2 - \Delta G$$

Exposure of skin is not so important owing to thin penetration of $\beta$-electron, while tritium in organs are dangerous

In case, $T$ is going in your body, you should drink water to remove it. For that purpose, Beer is very good!
Tritium and ITER

• First fusion machine fully designed for equimolar DT operation
  – Tokamak vessel will be fuelled through gas puffing & Pellet Injection (PI)
  – Neutral Beam (NB) heating system will introduce deuterium

• Employing DT as fusion fuel has quite a number of consequences
  – It causes alpha heating of the plasma
  – The fusion reaction will eventually provide energy
  – Closed DT loop is required due to the small burn-up fraction
    • Primary tritium systems for processing of tritiated fluids
    • Auxiliary systems necessary for the safe handling of tritium
  – Multiple barriers vital for DT confinement
    • Atmosphere & Vent Detritiation are crucial elements in the concept

After all a rather complex chemical plant, i.e. the Tritium Plant of ITER is needed for deuterium-tritium fuel processing
• Dimensions
  – Length: 79 m
  – Width: 20 m
  – Height: 34 m

• Space occupation
  – HVAC: 18%
  – Detritiation systems 16%
  – Tritium processing systems 30%
  – Non Tritium Plant systems 21%
  – Non process areas 15%
Tritium issues relating fuel cycles and T breeding

- Tritium breeding with enough margin and compatible with energy conversion
- Limited resource of Tritium (CANDU reactors are the main source)
- Tritium recovery in fuel cycles and breeding systems and its refinement

Selection of breeding materials:
- Li$_2$TiO$_3$
- Li$_{0.17}$Pb$_{0.83}$
- Li$_2$BeF$_4$

Controlling composition and fine structure
Tritium relating issues in power generation and surroundings

- Physical confinement and Safety confinement
- Detritiation and/or decontamination
- Safety reposition

![Diagram of Tritium Cycle in Reactor Facility](image)
Difficulties related to tritium summary I

• Difficulty of detection and quantitative analysis measurement with high accuracy.

• No way to measure tritium in bulk except combustion detection and calorimetry.

• T behavior in a DT reactor might not be simulated by that in DD plasma machine

• Large mass difference among all hydrogen isotopes

• Tritium breeding must be compatible with energy conversion (or economic)

• Tritium is chemically very active and react with most of impurities, in particular water and hydrocarbon molecules, in air to make more hazardous.

• Permeation and leakage are unavoidable
Summary of Part I (Tritium in Fusion)

Amount to be handled $10^1 \sim 10^{17}$ Bq
monitoring 1kBq release

Temperature $10^1 \sim 10^9$ K
Pellet(20K), Gas at RT(300K), Plasma ($10^5 \sim 10^9$K)

Characteristics of Tritium
Chemistry of excited state and non-equilibrium thermodynamics
Effect of $\beta$ electron emission and/or radiation heat
Defect formation by electron excitation and He production
Adsorption, solution, diffusion and permeation in materials
Reacts with impurities to produce inorganic hazard

Difficulty in quantitative analysis (accountancy)
No way to measure tritium in bulk except combustion/calorimetry.
Counting of disintegration ($1 \sim 10^6$Bq limited to T near surface)
Mass and pressure measurements
Radiation heat measurement (accompanying large error)
Most of tritium problem is directly related to the safety of operators and/or professionals. But public safety does not seem to become significant problems.

Tritium handling system, which uses mostly established techniques, can be build for ITER or even reactor.

However, handling of huge amount of tritium in ITER gives somewhat different problems. (Mostly relating tritium behavior in tokamak)

→ Huge inventory in tokamak and its accountancy
→ Controlled fuelling of DT
→ Possible permeation and leakage leading to cross-contamination
→ Contamination of remote handling system

Most of tritium problem is directly related to the safety of operators and/or professionals. But public safety does not seem to become significant problems.

Tritium breeding must be compatible with energy conversion (or economic)

It should be mentioned that we are facing a world wide lack of experts in tritium science and technology.
Tritium in burning plasma

Inefficient fuelling, Inefficient fuel cycling system
- D and T are different in fuelling efficiency, escaping flux, pumping speed
- D and T must be separately fuelled

Difficulty in controlling DT ratio 1 in plasma to attain efficient burning
- D, T concentration
  - Quantitative evaluation of D and T in plasma center is not easy
  - Plasma opacity could disturb optical measurements like Thomson scattering

Feed back from neutron yield
- Possible but quite dependent on confinement time which could be significantly different for D and T,
- Influence of toroidal and poloidal inhomogeneity

Fugue in-vessel inventory
- Significant isotope effect among H, D and T due large mass differences
Molecular kinetics gives incident flux to wall surface under pressure $P$

Simple molecular kinetics tells that velocity for D and T at the same energy different. So as rotational and vibrational state are.

Maxwell-Boltzmann’s law gives

$$v = \sqrt{\frac{8RT}{\pi m}}, \text{ hence } \frac{v_H}{v_D} = \sqrt{2} \quad \frac{v_H}{v_T} = \sqrt{3}$$

Molecular kinetics gives incident flux to wall surface under pressure $P$

$$J = nv = \frac{P}{(2\pi mkT)^{\frac{3}{2}}} \quad \frac{J_H}{J_D} = \sqrt{2} \quad \frac{J_H}{J_T} = \sqrt{3}$$
Isotope effects

Mass ratio of H, D and T is 1:2:3

under the same pressure \( \nu_H / \nu_D = \sqrt{2} \) and \( \nu_D / \nu_T = \sqrt{3/2} \)

\( \phi_H / \phi_D = \sqrt{2} \) \( \phi_D / \phi_T = \sqrt{3/2} \)

to give the same flux \( p_H / p_D = 1 / \sqrt{2} \) and \( p_D / p_T = \sqrt{2/3} \)

Relating to

Different confinement Outgoing flux ratio would be SQR(2/3)

Impinging energy to wall surface ?

Reflection coefficient May be SQR(3/2) but no data for T

Recycling flux ratio Unknown retention time

Pumping speed ratio For mechanical pumping SQR(3/2)
Unknown for cryo-pump

Tritium retention (solubility, diffusivity and permeability, trapping effect)

Surface residence time
Part II  Tritium issues in plasma wall interactions

How to extrapolate results on hydrogen retention in present tokamaks to ITER and beyond

1. Tritium retention on plasma facing materials caused by DT experiments in TFTR and JET

2. Behavior of Tritium produced by DD reactions (could not be used to simulate behavior of T fuel)

3. Deuterium and hydrogen retention in JT-60 for understanding of DT fuel
Evaluation of hydrogen retention in present tokamaks is of high priority to establish a database and a reference for ITER (400 s...usually 10-20 s today).

- T retention constitutes an outstanding problem for ITER operation particularly for materials choice (low Z or high Z ?)

- A retention rate of 10% of the T injected in ITER would lead to the in-vessel T-limit (350/700g) in ~35/70 pulses. (every ~ 35/70 shots require removing of in vessel T)

- Retention rates of this order or higher (~20%) are regularly found using gas balance.

- Retention rate often lower (3-4%) are obtained using post mortem analysis

- T breeding can not compensate such high inventory
II-1. Tritium retention on plasma facing materials caused by DT experiments in TFTR and JET

TFTR: a limiter tokamak

JET: a divertor tokamak

DTE Campaign using MarkII-A divertor
Experience of DT discharges in TFTR

T retention is quite non-uniform in toroidal and poloidal directions as well as in material depth.

Different deposition profiles between upper and lower area.

Grey : CFC
White : Graphite
Experience of DT discharges in TFTR

- T retention is mostly in redeposited layers
- Inhomogeneous retention
- Machine dependent distribution

Concerns on T inventory in Carbon machines

TFTR
- TFTR bumper limiter
  - Tritium was mostly codeposited with carbon.
  - Heavier codeposition on the edge of the erosion dominated tiles [1].

Main source of the codeposition on the side was prompt deposition of carbon which was sputtered on the plasma-facing surface!

T retention profile on tile sides consists of two exponential decay components

\[ PSL(x) = C_1 \exp\left(-\frac{x}{\lambda_1}\right) + C_2 \exp\left(-\frac{x}{\lambda_2}\right) \]

- \( x \): Distance from front
- \( \lambda_1 \): Short-term decay length
- \( \lambda_2 \): Long-term decay length

Graphs showing PSL intensity vs. distance for different sides of the tile.
T retention in JET tiles measured by combustion method

- Tritium codeposits with carbon and other impurities at low temperature region
- No detailed profile
- Necessity to develop removal technique
Most of Tritium is in deposited carbon

Heavy deposition

Exfoliated region

Activity of metallic impurity

Exfoliated region

Line profil

IP image
Clear asymmetry owing to tile alignment

- **Local codeposition**
  - $\Rightarrow$ because of the codeposition of tritium and carbon sputtered at adjacent tile.
Tritium distribution on sides facing toroidal gap

- Narrow gap facing side
- Wide gap facing side

Details:
- Theta perp max 8°
- Chamfer angle α ~ 0.3°
- Tile 'A'
- Tile 'B'
- Tile length ~380mm
- Toroidal direction
- 12mm gap
- 3mm gap
Deposition on the toroidal gaps does not seem problem

- Little codeposition on the toroidal facing sides
- Clear In-Out asymmetry, but little difference between the both sides
CFC is a porous material allowing deep T penetration and its matrix and filler shows quite different tritium retention.

Front Surface

Rare Surface

Cross-sectional view:

PSL intensity [PSL/mm$^2$]
II-2. Behavior of Tritium produced by DD reactions

At the beginning we thought behavior of T produced by DD reactions should be similar to that of fueled T in tokamak.

We have found that was wrong.

Tritium produced by DD reactions in JT-60U, ASDEX-U and TEXTOR do not reflect behavior of fueled tritium.

Simultaneously we have found that

- Most of T produced by DD reactions (which initially have energy of 1MeV) do not fully loose their energy and are directly implanted into subsurface of the plasma facing materials (in present tokamaks).

i.e.

we can study behavior of high energy particles escaping from plasma like NBI particles and He ash.
Simulation of high energy Triton by OFMC code

Toroidally homogeneous

T distribution on W-shaped divertor tiles of JT-60U
Completely different T profiles at divertor area between JT-60U and JET
T profile is completely different from C deposition

Comparison between C deposition and tritium distribution profiles 2/27/2006
Tritium profile
Deposition profile
H and D profiles

Typical example for different behavior of T and H/D

Completely different

Deposited area

H and D profiles
Tritium profile
OFMC simulation
High energy Triton escaping from plasma

- 31% of produced tritons was lost and implanted to the wall.
- 12% of produced tritons impinged on the divertor region.
- Fluxes of the tritons were ~10^{14} s^{-1}.
  - In a case of neutron production of ~10^{11} m^{-2}s^{-1}: inner and outer midplane first wall.
  - ~10^{12} m^{-2}s^{-1}: inner divertor target and inner baffle plate.
  - ~10^{13} m^{-2}s^{-1}: dome top and the outer baffle plate.

Masaki et al. 15th PSI
Comparison of observed $T$ profiles and simulation

OFMC calculation

Cross section of the JT-60U vacuum vessel

Baffle plate

Divertor region

Tritium distribution by IP

Log scale
Tritium on dome top tile - full toroidal distribution -

# 240 pieces of tiles

# relation between tritium distribution and toroidal magnetic coils

Measured area

Relative activity [PSL/mm²]

Toroidal direction

Toroidal angle [degree]

JAERI
Tritium retention: comparison between observation and calculation

**Observation**

Long term tritium retention: Roughly 40% of produced tritium (18GBq)

<table>
<thead>
<tr>
<th>Location</th>
<th>Tritium concentration</th>
<th>Tritium retention</th>
</tr>
</thead>
<tbody>
<tr>
<td>Inner divertor</td>
<td>2 kBq/cm²</td>
<td>Divertor region</td>
</tr>
<tr>
<td>Dome top</td>
<td>60 kBq/cm²</td>
<td>10% of produced tritium</td>
</tr>
<tr>
<td>Outer divertor</td>
<td>250 Bq/cm²</td>
<td></td>
</tr>
</tbody>
</table>

**OFMC calculation**

31% of tritons produced by nuclear reaction are lost from plasma

<table>
<thead>
<tr>
<th>Location</th>
<th>Tritium concentration</th>
<th>Tritium retention</th>
</tr>
</thead>
<tbody>
<tr>
<td>Dome</td>
<td>6% of the produced tritons, ~0.7 MeV</td>
<td></td>
</tr>
<tr>
<td>First wall</td>
<td>1%, ~1 MeV</td>
<td>Divertor region</td>
</tr>
<tr>
<td>Divertor</td>
<td>3%, ~0.5 MeV</td>
<td>12% of produced triton</td>
</tr>
<tr>
<td>Inner baffle plate</td>
<td>1%, ~1 MeV</td>
<td></td>
</tr>
<tr>
<td>Outer baffle plate</td>
<td>20%, ~0.6 MeV</td>
<td></td>
</tr>
</tbody>
</table>
DT discharge experiments in JET and TFTR have shown that significant amount of tritium retained in redepoited carbon layers.

Behavior of T produced by DD reactions is completely different from that of fueled T.

For detailed understanding of T behavior and estimation of T inventory in ITER and a reactor, behaviors of D in various tokamaks have been extensively studied. And no we believe D behaves similar to T but is not so sure. (We don not how large isotopic effects are)

In anyway, large D retention in carbon redeposits make us to avoid carbon as PFM in a DT reactor.
Current estimation of T inventory in ITER is not saturated!

Tritium inventory in ITER plasma-facing materials and tritium removal procedures
J. Roth, E. Tsitrone, T. Loarer, V. Philipps, S. Brezinsek, A. Loarte, G. F Counsell, R. P. Doerner,
It is critically important whether hydrogen retention saturates or not.

Static retention: incorporated in redeposited carbon layers at plasma shadowed area
Dynamic retention: retained in plasma facing surface area both eroded and deposited

Which is large, Static or Dynamic?

- In Tore-Supra; Static >> Dynamic and $\frac{\partial S}{\partial t} > \frac{\partial D}{\partial t}$
- In JT-60U; Static > Dynamic but $\frac{\partial S}{\partial t} < \frac{\partial D}{\partial t}$
Concerns of large tritium retention in carbon materials minimize the utilization of carbon materials in ITER,

- Be ; First wall
- W ; Divertor dome and buffer plates
- C ; only for the divertor target.

However, utilization of tungsten blocks below their DBTT could result in the total failure of the machine through cracking of cooling pipes, we should keep carbon materials as an alternative for armor tiles even for a reactor.

(Matrical selections will be discussed in Wednesday evening)
Recrystallization to columnar grains results in cracking
Congruent melting with substrate metals leads cracking as well as melting
(Be and W would give same result)

Bulk W and Mo must not be used below DBTT

Break-up by thermal shock
However, utilization of tungsten blocks below their DBTT could result in the total failure of the machine through cracking of cooling pipes, we should keep carbon materials as an alternative for armor tiles even for a reactor. (Material selections will be discussed in Wednesday evening)

This motivate us to examine carbon erosion/deposition and H and D retention in plasma facing carbon materials of JT-60U in detail.
Remaining questions to be solved for application of carbon as PFM in DT machines are,

- Where and how much is carbon eroded and redeposited?
- Do erosion and redeposition saturate?

- Where the largest redeposition occurs, plasma facing surface, shadowed area or far remote area?

- Where is tritium (T) retained?
  How related to carbon deposition?
  How large is retention in eroded area and main chamber?
  Does T retention saturate?

- How to recover or remove the retained T?
II-3. Carbon erosion/deposition and D and H behavior in JT-60 for understanding of DT fuel

- CFC tiles
- Other tiles are isotropic graphite.

Inner Divertor Plates
Outer Divertor Plates
Dome Units
Inner Baffle plates
Outer Baffle plates
Inner pumping slot
Separatrix
Dome
Inner pumping slot
Dec. 1998 ~
Separatrix
Dome
Outer pumping slot
Normal plasma operation is done by D discharges with D NBI (DD discharges)

Usually each campaign terminated by HH discharges to remove T produced by DD reactions.

The temperature of plasma facing surfaces increased 50 -1000K owing to the plasma heating. Because of less heating power of H NBI, the temperature increments of plasma facing surface under HH discharges were significantly lower than that under the DD discharges.

### History of plasma operation and of plasma exposure of analyzed tiles

<table>
<thead>
<tr>
<th>Year</th>
<th>1997</th>
<th>1998</th>
<th>1999</th>
<th>2000</th>
<th>2001</th>
<th>2002</th>
<th>2003</th>
<th>2004</th>
<th>2005</th>
</tr>
</thead>
<tbody>
<tr>
<td>Operation</td>
<td>Inner side pumping</td>
<td>Both sides pumping</td>
<td></td>
<td></td>
<td></td>
<td>Both sides pumping</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Sample exposure period</td>
<td>Tiles for D,H retention study</td>
<td>Divertor</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>Collector probes for deposition, D,H retention studies in shadowed area</td>
<td>Main chamber</td>
<td></td>
</tr>
<tr>
<td>Temperature</td>
<td>~570 K baking (short pulse operation ~15s)</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>~420 K baking (long pulse operation ~60s)</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
Temperature rise owing to plasma heat load
Assuming toroidal symmetry,

Deposition : $0.55\ \text{kg (10.7 \times 10^{20} \text{C/s})}$

Erosion : $-0.34\ \text{kg (-5.7 \times 10^{20} \text{C/s})}$

Samples : 1997~2002

NB injection time :
$3 \times 10^4\ \text{s (outer dome wing:2 \times 10^4\text{s})}$

Mass balance of erosion and deposition

Materials density
Deposited layers : $0.91\ \text{g/cm}^3$,
Eroded region : $1.70\ \text{g/cm}^3$)

Assuming toroidal symmetry,
Deposition : $0.55\ \text{kg (10.7 \times 10^{20} \text{C/s})}$
Erosion : $-0.34\ \text{kg (-5.7 \times 10^{20} \text{C/s})}$
Missing : $0.21\ \text{kg (5 \times 10^{20} \text{C/s})}$

SEM observation and Micrometer measurements

Erosion/deposition profiles at divertor region

40% of the deposition on the divertor area must be originated from the main chamber wall.

**Growth of redeposited layers in JT-60**

Different growing mechanisms depending on discharge conditions, flux, temperature and so on.

<table>
<thead>
<tr>
<th>Shot No.</th>
<th>Divertor Discharges</th>
<th>Limiter Discharges</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>≥10MW</td>
<td>≤10MW</td>
</tr>
</tbody>
</table>

**Thickness**: Divertor discharges > limiter discharges

Lower power divertor discharges give columnar structure

Mechanism of Carbon Transport

Line of site deposition on outer dome wing

Tangential law \( \tan \alpha = 2 \tan \beta \)

\( \beta \) : angles of incident particles,

\( \alpha \) : preferential direction of columnar deposition

Deposition at inner target
Repetition of erosion and prompt redeposition

Deposition at outer dome
Direct transport of eroded carbon at outer divertor
Deposition thickness is proportional to a solid angle from outer divertor
\[ d \propto \frac{\sin \gamma}{L^2} \]

Erosion profiles of outer divertor tiles
Mechanism of Carbon Transport

\( ^{13}\text{C} \) deposition on surface first wall tiles was very small

\( ^{13}\text{C} \) deposition peak is slightly shifted toward the pumping slot than the peak position of C deposition

- Direction of the drift flux in the private region was toward the inner divertor (Reciprocating mach probe measurements)

\( ^{13}\text{CH}_4 \) puffing at outer divertor

Carbon transport through private region

Ishimoto et al. 12th ICFRM.
Deposition at remote area (Bottom of Divertor) caused by line of sight transport from eroded area

NB injection time: $8 \times 10^3$ s  
Average deposition thickness: $\sim 2 \mu m$  
Estimated density: $\sim 1.8$ g/cm$^3$  
Area: 3.8 m$^2$  
Total deposition: $\sim 0.013$ kg ($\sim 8 \times 10^{19}$ C/s)

Owing lower temperature (420K) operation (H+D)/C in redeposits is very high, 0.6 $\sim$ 0.8, which makes their structure amorphous like.
Carbon deposition pattern at remote area well corresponds Tritium profile

No Carbon deposition at far remote area evidenced by tritium retention

Little carbon exhaust owing to high temperature operation?

Reference:
Dome top tile ~ 60 kBq/cm²
(a) Outer divertor target

(b) Top  Outer dome wing  Bottom

5DV3cp

Photo

(c) 

(d)
Carbon deposition at tile gaps - Two different mechanisms -

Prompt redeposition of eroded carbon at front surface

e-folding length: 7 mm

Line of sight deposition from eroded area
Deposition at the deeper area and/or bottom of closed gap seems small.

\[ \text{H}^+ , \text{H}^* , \text{H}_2^+ , \text{H}_2^* \]

\[ \text{C}^+ , \text{CH}_n^+ \]

Wall

\[ \text{H}_2 \] Density

Gap

Deposition thickness

Deposition at gap between tile and base plate

Erosion
Summary on Carbon deposition observed in JT-60

1. Deposition at PFS
   - Large at inner divertor and outer dome wing
   - Different deposition mechanisms;
     Lamellar type, Columnar Type, Amorphous Type (High H/C case)

2. Deposition in the gap
   - The gaps at the first wall would be very small

3. Deposition at remote area
   - Can not be avoided.
     Mostly appeared at the line of site from the eroded area.
     Could be reduced by appropriate divertor geometry

4. Far remote area
   - Little deposition (Could be owing to high temperature operation)
Physical Erosion and Deposition

Net Erosion, physical and chemical

Repetition of erosion and prompt redeposition

Possible impurity transport through private flux region

Possible mechanisms of carbon erosion and transport (Summary)

Erosion?

tile analysis
QMB

Deposition

Line of sight deposition

Physical Erosion and Deposition

Remote area very small transport
Retention of D and H - Locations of analyzed tiles -

Poloidal and toroidal cross-sections of first walls

Having boronized layers

W shaped open divertor
Hydrogen retention determined by TDS

Peak temperatures well correlate with tile temperatures

Temperature/K

Redeposits on ID3
Redeposits on DM9
Eroded area OD1

Desorbed amount/10^{19}\,\text{molecules} \cdot \text{m}^{-2} \cdot \text{s}^{-1}

Temperature/K

1000K
800K
1400K
Comparison of H+D retention in near surface layers

Total retention $\sim 10^{23}/m^2$

- The differences of (H+D) retention among tiles are within a factor of 10.

<table>
<thead>
<tr>
<th>First wall tiles Above 600K</th>
<th>Divertor tiles Above 900K</th>
</tr>
</thead>
<tbody>
<tr>
<td>H retention</td>
<td>D retention</td>
</tr>
<tr>
<td>Sample name</td>
<td></td>
</tr>
<tr>
<td>BP1</td>
<td>BP1</td>
</tr>
<tr>
<td>ID2</td>
<td>ID3</td>
</tr>
<tr>
<td>DM6</td>
<td>DM7</td>
</tr>
<tr>
<td>OD1</td>
<td>OD3</td>
</tr>
<tr>
<td>Removal</td>
<td>OD1Dep</td>
</tr>
</tbody>
</table>

- H/D ratio is high for higher temperature tiles.
- Deuterium once retained in the wall during the DD shots was isotopically replaced by H under the HH discharges.
H+D retention in plasma facing surface layers is likely saturated
Redeposits on ID3
1000K
\[ H > D \]

Redeposits on DM9
800K
\[ H \sim D \]

Eroded area OD1
1400K
\[ H > D \]

Depth profile by SIMS indicates deep D implantation
Temperature dependence of H+D retention

Redeposited layers on DM (Dome)
- Lowest Temp.
- Inhomogeneous retention
- ID (Inner divertor)...
- Higher Temp (Thermally isolated from the substrate)
- Homogeneous distribution

Redeposited layers on ID

Eroded region
- OD (Outer divertor)...
- Highest Temp.
- Lowest Conc.

Benefit of high temperature operation

Surface hydrogen concentration is very likely saturated.
The saturated concentrations for the eroded area and redeposited layers were nearly the same and only depending on the surface temperature.
Gas
Graphite
Solid
Liquid

Determined by
○: IR
●: NMR
▲: EELS

A: C-H film ever produced

Hydrogen solubility (very small)

H/C=0.5
H/C=1.0

a: C-H (Solid phase)
Non-equilibrium phase

Equilibrium
Gausian
H_2 1 atm

C = S^1/2

Temperature

Graphite
(Solid)

Liquid

Gas

H

C

C_6H_6, C_2H_2

C_2H_4, C_2H_6

800K

-500K
H, D, T retention for eroded tiles

Retention during HH and replacement of D by H
Retention during DD discharges
Injection of high energy Triton

Lower temperature

Higher temperature

Concentration

Depth/ µm

Injection of high energy Triton

T x 10^8~13

H, D, T retention for eroded tiles

Retention during HH and replacement of D by H
Retention during DD discharges
Injection of high energy Triton

Lower temperature

Higher temperature

Concentration

Depth/ µm
H,D,T retention for redeposited tiles

Deposition under HH discharges

Lower temperature

Deposition under DD discharges

Injection of high energy Triton

$T \times 10^{8-13}$

Concentration

Depth /10 µm

H

D

Higher temperature

$T \times 10^{8-13}$
Summary of Part II (Erosion and H retention) Cont’d

1. Carbon erosion and deposition in JT-60U
   - Deposition occurs mostly at the inner divertor probably owing to repetitive process of erosion and prompt redeposition
   - Deposition at outer dome wing and divertor shadowed area is caused by line of sights from the eroded area
   - Deposition in tile gaps is not large, except open gaps connected to pumps

2. Retention and depth profiles of three hydrogen isotopes in JT-60U
   - Hydrogen retention at the plasma facing area is very likely saturated and would not linearly increase with time.
   - The isotopic ratios of retained hydrogen near surface layers are always equilibrated with incoming hydrogen fluxes (H/D/T).

3. Effects of high temperature deposition
   - Possible saturation of T retention at plasma facing surface and less T retention
   - Deposits at high temperature have less T and show strongly adhesion

4. Importance of Geometry
   - Tile alignment, Gap width, Divertor geometry could reduce erosion.
   - Plasma shaping could suppress erosion
- Saturation of H retention on the plasma facing surface would not allow linear increase of T retention with time.

- The isotopic ratios of retained hydrogen near surface layers are always equilibrated with incoming hydrogen fluxes (H/D/T).

- Depth attaining this equilibrium is quite thick owing to the porous nature of carbon materials and is increased by temperature rise.

- Hence tritium retention in plasma facing surfaces (both eroded and redeposited) would be significantly reduced by isotopic replacement by DD discharges subsequently made after DT discharges.

- All these results from JT-60 is promising to use carbon as PFM at high temperature (above 800K)
Since JT-60U had rather large magnetic ripple loss, the loss or injection of high energy triton to the deep in the first wall was appreciable. Different from the plasma particle injection which would cause the near surface saturation of hydrogen, the deep implantation with less flux could pile up for long time.

The deep implantation of energetic hydrogen could enhance hydrogen retention even for metallic first wall.
<table>
<thead>
<tr>
<th>Deposited area</th>
<th>Location</th>
<th>Deposition rate x 10^{20} atoms/s</th>
<th>H+D retention rate x 10^{19} atoms/m^2 s</th>
<th>(H+D)/C</th>
<th>D/H</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Inner divertor</td>
<td>~ 6</td>
<td>~ 1</td>
<td>~ 0.02</td>
<td>~ 0.4</td>
</tr>
<tr>
<td></td>
<td>Outer dome wing</td>
<td>~ 4.5</td>
<td>~ 6</td>
<td>~ 0.13</td>
<td>~ 1.2</td>
</tr>
<tr>
<td></td>
<td>Bottom of divertor (Base Temp. 420K)</td>
<td>~ 0.85</td>
<td>~ 6</td>
<td>~ 0.75</td>
<td>~ 3.6</td>
</tr>
<tr>
<td></td>
<td>First wall (low field side)</td>
<td>~ 0.0015</td>
<td>~ 0.0024</td>
<td>~ 0.16</td>
<td>~ 1</td>
</tr>
<tr>
<td>Eroded area</td>
<td>Inner dome wing</td>
<td>~ 1.5</td>
<td>~ 2</td>
<td>not evaluated</td>
<td>~ 0.07</td>
</tr>
<tr>
<td></td>
<td>Outer divertor</td>
<td>~ 4.2</td>
<td>~ 3</td>
<td>~ 0.07</td>
<td>~ 0.31</td>
</tr>
<tr>
<td></td>
<td>First wall (low field side)</td>
<td>not evaluated</td>
<td>~ 2-4</td>
<td>~ 0.0004</td>
<td>1 ~ 4</td>
</tr>
</tbody>
</table>

**Diagram:**

- **BP1, 2, 3:** Baffle Points
- **ID1, 2, 3:** Inner Divertor Plate
- **ID1, 3:** Inner Divertor Plate
- **Thermocouple DM3, 5, 6, 7, 9**
- **Inboard, Outboard**
- **Inner Baffle Plate, Inner Divertor Dome, Outer Divertor Plate**