Book of Abstracts

25th-28th March 2019

Jožef Stefan Institute

Ljubljana, Slovenia

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TRANSAT
TRANSversal Actions for Tritium

First Tritium School

Book of Abstracts

http://transat-h2020.eu/transat-tritium-school/

25th-28th March 2019

Main Lecture Hall

Jožef Stefan Institute

Ljubljana, Slovenia

TRANSAT received funding from the Euratom Research and Training Programme 2014-2018 under grant agreement n° 754586.

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Acknowledgement:

The TRANSAT project received funding from the Euratom Research and Training Programme 2014-2018 under grant agreement n° 754586.

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		17:50h	Discussion	17:35h	Discussion	L - 40 + 10 min O - 30 + 5 min C - 20 + 5 min	
				20:00h	Tritium School Dinner		

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Stefan's Days Lectures

The Jozef Stefan Institute is celebrating its 70th anniversary this year and in honour of this occasion special invited lectures will be held by accomplished speakers from around the world. You are warmly invited to attend these lectures as they are open to the general public.

Monday, 25th of March:

INVITED LECTURE: **COMRADESHIP OR HOW TO FEEL LIKE A FISH IN THE WATER**
Mrs. Zdenka Badovinac, Director of Modern Gallery, Ljubljana — **Great Hall, at 13.00**

EXHIBITION: **IRWIN: NSK GUARDS AND PROCESSIONS**
Introductory address: Prof. Jadran Lenarčič, Director J. Stefan Institute
Honorary address: President of the Republic of Slovenia Mr. Borut Pahor
Gallery of the Institute, at 14.00

Tuesday, 26th of March:

INVITED LECTURE: **WILL LIFE GO LIFE ONE DAY?**
Prof. Bart De Moor, KU Leuven, Belgium — **Great Hall, at 12.00**

INVITED LECTURE: **MITOCHONDRIA CALCIUM SIGNALLING IN CELL LIFE AND DEATH**
Prof. Rosario Rizzuto, Rector University of Padua, Department of Biomedical Sciences, Italy — **Great Hall, at 13.30**

Wednesday, 27th of March – 70th Anniversary Celebration of J. Stefan Institute

WELCOME SPEECH: **Prof. Jadran Lenarčič, Director J. Stefan Institute**

HONORARY SPEECH: **Prime Minister of the Republic of Slovenia, Mr. Marjan Šarec**

INVITED LECTURE ON FLYING ROBOTS: **Prof. Vijay Kumar, Dean of Eng., Penn University, USA**

GOLDEN AWARDS of J. STEFAN
Linhart Hall, Cankarjev dom, Ljubljana, at 19.00 - Entry with a ticket

Thursday, 28th of March:

INVITED LECTURE: **4D PRINTING AND BIO-PRINTING: THE "MASS" IS NOT YET OVER!**
Prof. Jean-Claude André, LRGP-UMR 7274 CNRS-UL and INSIS-CNRS — **Great Hall, at 12.00**

Friday, 29th of March:

INVITED LECTURE: **THE TECH GIANTS ARE HARVESTING YOUR DATA. SHOULD YOU CARE?**
Prof. Geoff Webb, Monash University, Australia — **Great Hall, at 13.30**

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Scientific programme, Monday 25th of March

8:30-9:50	Registration		
9:50-10:00	Welcome – prof. dr. Jadran Lenarčič - Director of JSI, dr. Christian Grisolia (CEA)- TRANSAT coordinator,		
Topic : Tritium management and detection		Chair : Christian Grisolia CEA	
10:00-10:50	Different detection techniques tor tritium inventory	Pascal Fichet (CEA, France)	50 min (L1)
10:50-11:40	Reducing releases from tritium facilities	Walter Shmayda (Univ. Rochester, US)	50 min (L2)
11:40-12:30	Tritium management in DEMO breeding blanket	Ion Cristescu (KIT, Germany)	50 min (L3)
12:30-14:45	Lunch – Opening of Institute open days at 13:00		135 min
Topic : Tritium migration, management		Chair : Ion Cristescu KIT	
14:45-15:35	Tritium migration in breeding blankets in fusion technology	Carlos Moreno (CIEMAT, Spain)	50 min (L4)
15:35-16:10	Lithium enrichment needs for fusion and fission applications	Katharina Battes (KIT, Germany)	35 min (O1)
16:10-16:30	Coffee break		20 min
16:30-17:05	Feedback and perspectives issued from the OBT WG inter-laboratory exercises	Nicolas Baglan (CEA, France)	35 min (O2)
17:05-17:40	Numerical and experimental investigation of yttrium metal as a getter material for the hydrogen hot trap of IFMIF/DONES	Sebastian Hendricks (CIEMAT, Spain)	35 min (O2)
17:40-18:10	Discussion on tritium detection migration, management		30min

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DIFFERENT DETECTION TECHNIQUES FOR TRITIUM INVENTORY

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Tritium is a radionuclide very difficult to analyse mainly because of its energy, type of emission (β^-) and its chemical and physical behaviour. In all kinds of developments concerning the tritium (dismantling processes, fusion development, environmental control…), it is essential to characterize the radionuclide with high accuracy and a minimum of uncertainty. Elevated concentrations of tritium in the environment associated with producing, handling and managing the radioactive form of hydrogen at nuclear facilities are of great public concern. At present time in France, wastes containing tritium come mainly from defence, research activities and nuclear power plants. Concerning NPP internationally, PWR produces less tritium than other reactors such as CANDU. This production of tritium in the nuclear industry is currently an important subject and tritium inventory must be deeply investigated. In the very near future with the development of fusion machines such as ITER, that will use Deuterium and Tritium as fuel, production of tritiated wastes and needs to assess the exact inventory of tritium will be crucial. The presentation will focus on the different chemical forms of tritium, which induce very different ways for the development of analytical techniques to provide the tritium inventory. Different analytical techniques already exist commercially but mainly consider some applications to control tritiated wastes in particular chemical form and for particular activities. Some international regulations already exist and will be described to assess the different ways of tritium investigations for public acceptance.

However, with new fusion technologies such as ITER, tritium inventory inside the machine still remain a high challenge, and the investigation of tritiated wastes will require of course standard techniques but also new technological breakthrough.

A nearly complete panel of standard techniques and of new technologies will be described during the talk. The autoradiography technique, a technique commercially developed for researches in biology is currently developed in CEA France because of its high potentialities to investigate tritium in situ. New developments of this technology will be presented and particularly the technique using phosphor screen and another one providing results in real time with new SiPM detectors.

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REDUCING RELEASES FROM TRITIUM FACILITIES

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All facilities that handle tritium gas produce tritiated water. Sources range from outgassing, to decontamination, to deliberate conversion of elemental tritium to tritiated water. Decontamination efforts strive to reduce cross-contamination and personnel doses. Gas to water conversion processes strive to reduce emission to the environment. The water activities range from a few µCi/L (tens of kBq/L) to a few Ci/L (tens of GBq/L) depending on the operation. Facilities that handle larger quantities of tritium can easily generate tritiated water streams with activities approaching kCi/L (TBq/L).

The common practice presently is to intercept, collect, and segregate these streams by activity. Low activity streams are assayed to ensure compliance with discharge limits and released. Intermediate streams are immobilized, packaged, and buried. High activity streams are reduced over metal getters to prevent beta-induced hydrolysis of water and stored as hydrides. While these practices meet current regulatory requirements, they leave the facilities open to expensive disposal costs, increasing on-site inventories and future liabilities. It is unlikely that larger tritium facilities could remain compliant with discharge regulations without active tritium extraction from the effluent streams.

Typically, the volume of water collected is inversely proportional to its activity ranging from thousands of litres of low activity water to a few litres of high activity water. At one end of the concentration spectrum, high throughputs are required to process large-volume low-activity water. At the opposite end, radiation tolerant materials are needed to handle high-activity low-volume water. Several processes designed to extract tritium from water processes have been evaluated for technical and economic feasibility. Combined Electrolysis Catalytic Exchange (CECE) shows promise by addressing many of the requirements described above: throughput, radiation resistance, economic viability, technical simplicity and facile coupling to isotopic separation systems.

This lecture will discuss:

- tritium oxide release from surfaces,
- options to convert tritium gas effluents to water as a strategy to mitigate emission.
- enrichment and recovery of tritium gas based on two CECE systems: a 7 m^3/hr alkaline electrolysis cell and a 20 m^3/h proton exchange membrane (PEM) electrolysis cell. Both systems are coupled to an isotope separator.

This material is based upon work supported by the Department of Energy National Nuclear Security Administration under Award Number DE-NA0001944, the University of Rochester, and the New York State Energy Research and Development Authority. The support of DOE does not constitute an endorsement by DOE of the views expressed in this article.

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TRITIUM MANAGEMENT IN DEMO BREEDING BLANKET

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The development of commercial fusion power production using deuterium and tritium is ongoing worldwide since decades and the European version of DEMO will undergo conceptual design between 2021 and 2027.

Among the different ways to provide electrical power, the nuclear fusion will only be publically accepted if the environmental impact is at tolerable levels. Auxiliary power requirements of fusion power reactors will need to be optimized, and heat will need to be efficiently converted to electrical power through usage of high temperature steam. On the other hand, heat might need to be intermittently "stored" to account for pulsed plasma operation, on the expense of the temperature level available for steam generation. Tritium is highly mobile, and its management as far as containment and confinement are concerned becomes more difficult with increasing temperatures of structural materials; any effluents and releases shall be kept at an absolute minimum. Therefore, tritium containment and confinement equipment and procedures need to be well integrated into the design and into operation of fusion power reactors.

The inventory of tritium, the processing throughput required, and the tritium consumption in fusion power reactors is unprecedented. Almost 56 kg of tritium is consumed per GW- year (thermal) of fusion power and 1 GWel reactor will therefore burn about 170 kg per year. Correspondingly, tritium breeding from lithium at a rate of almost 0.5 kg per day is required. As a result, the breeding and tritium processing systems will contain tritium at inventory levels of more than 10 kg, and will need to handle tritium throughputs of more than 1 kg per hour. Such amounts and throughputs may raise questions on tritium management and control. The tritium quantities in effluents and releases shall in any case be very low and intrinsically well below the accuracy limits of tritium tracking and accountancy, which can be anticipated for fusion power reactors. This of course does not mean that effluents and releases do not need to be measured in real time and quite accurately.

The paper will be focused on the topics of tritium breeding and in the identification of the tritium sources as far as permeation and escape into the environment are concerned. In addition, the main barriers for mitigation of tritium release into the environment will be as well introduced.

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TRITIUM MIGRATION IN BREEDING BLANKETS IN FUSION TECHNOLOGY

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Tritium as fuel in nuclear fusion reactors and radioactive element requires maximum traceability and containment. Being a scarce element in nature, it needs to be produced in the reactor itself.

The breeding blankets are the devices responsible for regenerating tritium and facilitate recovery towards the reinjection in the reactor by its ancillary systems. In ITER, the TBMs (Test Blanket Modules) will test the viability of these systems. Moreover, the breeding blankets will have to provide the full fuel for the demonstration power plant DEMO.

Tritium as a light element is able to permeate through structural materials. The main problem due to this characteristic is the permeation towards the coolant and the subsequent emission to the environment. This is what has motivated the creation of models of these phenomena using simulation tools.

Since the 80s, tools like TMAP, DIFFUSE and other 1D codes laid the foundations for more complex system codes, such as EcosimPro.

The system level programs offer considerable advantages because its object-oriented nature. Codes like EcosimPro facilitates implementation of these processes, offering the synergy of various disciplines (e.g., transport, control, chemistry, hydraulics, etc.) and the robustness of its equation-solving algorithms. EcosimPro and its TRITIUM_LIBS libraries enable the user to build simulation models from a simple permeation membrane to complex systems like a breeding blanket device and its auxiliary systems both ITER and DEMO conditions.

Likewise, the computing power of current workstations allow a comprehensive 3D analysis at the component level in tools such as ANSYS or COMSOL. A modelling routine can be established with this set of tools where the 3D code gives an optimization of the migration processes that will be implemented in the simulation at system level.

The state of the art of the tools, the different problems to be addressed, the results and the analysis will be discussed in this lecture.

[1] F.R. Urgorri, C. Moreno, E. Carella, D. Rapisarda, I. Fernández-Berceruelo, I. Palermo and A. Ibarra, Tritium transport modelling at system level for the EUROfusion dual coolant lithium-lead breeding blanket, Nucl. Fusion., Volume 57, Number 11

[2] F. R. Urgorri, C. Moreno, E. Carella, J. Castellanos, A. Del Nevo, and A. Ibarra, Preliminary system modeling for the EUROfusion water cooled lithium lead blanket., Fusion Sci. Technol., 71, (444-449), 2017.

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[4] E. Carella and C. Moreno. Ecosimpro detailed documentation package. Technical Report EFDA-D-2L3JCZ v2.0, EUROfusion, 2015.

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LITHIUM ENRICHMENT NEEDS FOR FUSION AND FISSION APPLICATIONS

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In future fusion reactors the huge amount of energy that is released by merging the hydrogen nuclei deuterium and tritium will be used to generate electricity. Deuterium is widely available in seawater, whereas tritium, which is a β-radiator, only exists in nature at negligible amounts, so that it will be generated inside the reactors by a nuclear reaction in so-called breeding blankets.

The most suitable material for that purpose is the lithium isotope ^{6}Li. Unfortunately, natural lithium consists to 92.5 % of ^{7}Li and only to 7.5 % of ^{6}Li. This means that the required ^{6}Li has to be enriched starting with natural lithium. For the use in future fusion power plants, an amount of approx. 60 t of 90 % enriched lithium per GWel. will be needed. At the moment, it is unclear where this material will come from: It seems that the knowledge of how lithium isotopes can be separated in technical scale does either not exist anymore (^{6}Li enrichment was done mainly for military purposes in the 1950s and 60s) or it is not available to the fusion community (process details are restricted and not published).

On the other hand, in nuclear fission, ^{7}Li is required in pressurized water reactors. Here, lithium hydroxide is added to prevent the cooling water of the reactor core from becoming acidic and thus obviate corrosion and possible failures of pipes and other infrastructure. In addition, ^{7}Li is added to demineralizers, special water purifiers, for filtering out radioactive contaminants from the cooling water. For these purposes ^{7}Li needs to be enriched to very high purity as ^{6}Li would react with nuclear material under release of tritium. For the same reason, the molten salt reactors currently planned to be built in China require ^{7}Li at a purity above 99.995%.

The objective of this talk is to discuss the requirements on lithium isotope separation for both, fusion and fission applications, and propose suitable enrichment processes that could fulfil the requirements. For this purpose, a review of existing lithium isotope separation processes has been conducted and the results been discussed using a systems engineering approach.

FEEDBACK AND PERSPECTIVES ISSUED FROM THE OBT WG INTER-LABORATORY EXERCISES

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Organically bound tritium (OBT) has become of increasing interest within the last decade, with a focus on its behaviour and also its analysis, which are both important to assess tritium distribution in the environment. After the first OBT International Workshop which was held in France in May 2012, an international working group was created. As a result, four OBT exercises were organised; the 1st one on potatoes was conducted in 2013 by the Canadian National Laboratory (former AECL) with about 20 participating labs from around the world, the 2nd one on sediment was organised in 2014 by Southampton University (GAU) on a sediment with again about 20 participating labs the third one on wheat was organised in 2015 by the CEA with about 25 participating labs and the fourth one on grass was conducted by Cernavoda NPP in 2017 with about 25 participants.

These exercises allow withdrawing very positive conclusions as the results are in progress demonstrating that; (i) analytical skills are increasing in all participating labs, (ii) matrix change doesn't seem to affect the quality of the results and (iii) there is more and more interest to participate in these exercises. In addition, the fifth exercise on fish is ongoing with results anticipated early 2019 and the sixth one in preparation to be launched in 2020.

In addition to these analytical progresses, aiming to further improve our tritium community's knowledge upon topics such as tritium analysis, tritium migration and transfer, several points of importance are discussed within the OBT WG:

- to develop Certified Reference Material's (CRM's) providing analytical support for optimising and validating analytical procedure
- to associate a model inter-comparison with an analytical one and to gather attendees capabilities to organise field experiments?

The results and conclusions from these works started in 2012 will be presented and discussed here.

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NUMERICAL AND EXPERIMENTAL INVESTIGATION OF YTTRIUM METAL AS A GETTER MATERIAL FOR THE HYDROGEN HOT TRAP OF IFMIF/DONES

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Ensuring a secure and reliable execution of the future neutron irradiation facility IFMIF/DONES requires its liquid lithium loop to be purified from hydrogen isotopes which are generated during operation. In order to fulfil the demanded limits of hydrogen isotope concentration in the loop an yttrium pebble bed is thought to serve as a hydrogen hot trap. In the past several experiments have been done with the purpose to measure those system specific parameters which determine the final efficiency and removal rate of the hydrogen trap like the solubility and diffusivity of hydrogen in lithium and yttrium [1,2,3,4] as well as the mass transfer coefficient between a fluid and a solid pebble bed [5]. In addition, numerical studies have been executed estimating the final absorption flux into the trap by assuming either a constant or concentration dependent trap efficiency [6]. However, this rather weak assumption does not allow to reproduce or explain experimental data in a quality that would be sufficient for a reliable design of a tritium trap for IFMIF/DONES. For this purpose, a detailed numerical simulation model of tritium transport into the yttrium pebble bed is required and has been developed from scratch within the framework of this work using the software EcosimPro.

The tritium absorption flux into the pebble bed is simulated by solving Fick's second law for a number of spherical bodies. As a boundary condition for the concentration at the pebble surfaces the equilibrium of partial pressures of tritium in liquid lithium and yttrium is considered. A second boundary condition is given by the equilibrium between the pebble surface diffusion flux given by Fick's first law and the mass transfer flux of tritium in liquid lithium. The tritium mass transfer coefficient for flowing liquid lithium through an yttrium pebble bed could be numerically estimated within the scope of this work and is found to mainly determine the lithium flow rate and the trap efficiency.

Using the developed numerical model of the tritium trap, it was possible to qualitatively simulate the tritium inventory build-up of a simplified model of the purification loop of IFMIF/DONES in which tritium is generated with a constant rate. In contrast to previous calculations which assumed a constant trap efficiency it is found that an increase of yttrium mass in the trap can only slow down the build-up of tritium inventory in the loop but will never lead to stationary tritium content. This implies that the trap efficiency decreases during operation by several orders of magnitude depending on the applied mass of the yttrium bed. This result does not conform with former estimations used for previous designs of the tritium trap.

In order to test the numerical model developed in this work it has been attempted to reproduce the experimental results of tritium removal rate measurements published in the articles [6,7]. The experimental and numerically predicted values match within a small margin, which even decreases when varying the chosen pebble diameter. Minor discrepancies are most likely to origin from unknown geometric dimensions of the disc-like shaped pebbles used for the experiments. The present work also shows a conceptual design of an experiment for the purpose of measuring hydrogen removal rates of a miniature yttrium pebble bed in contact with flowing liquid lithium and thus of optimizing the accuracy of the developed numerical simulation model.

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[5] E.J. Wilson and C.J. Geankopolis , Ind. Eng. Chem. Fundamen. 9–14 (1966) 5
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Scientific programme, Tuesday 26th of March

Topic : Tritium inventory and control		Chair : Sabina Markelj JSI	
8:30-9:20	Hydrogen isotope retention and transport in neutron-irradiated tungsten	Yuji Hatano (Toyama University, Japan)	50 min (L5)
9:20-10:10	Modelling of tritium inventory in plasma facing component in fusion devices	Klaus Schmid (IPP, Germany)	50 min (L6)
10:10-10:25	Coffee break		15 min
10:25-11:15	Tritiated dust in tokamak	Christian Grisolia (CEA, France)	50 min (L7)
11:15-11:50	Analysis of fuel retention in plasma-facing components. Experience from D-T operation of JET with carbon walls, current approach and preparation for next campaigns	Marek Rubel (KTH, Sweden)	35 min (O3)
11:50-14:45	Lunch – Lecture – Institute open days at 13:30		175 min
Topic : Tritium inventory and control		Chair : Pascal Fichet CEA	
14:45-15:20	Measurement of tritium on W divertor tiles used in JET-ITER like wall campaigns using imaging plate and β-ray induced x-ray spectrometry	Sun Eui Lee (Toyama University, Japan)	35 min (O4)
15:20-15:45	Activities at the university of Latvia for tritium measurements from functional and plasma facing materials	Gunta Kizane (University Latvia, Latvia)	25 min (C1)
15:45-16:05	Coffee break		20 min
16:05-16:40	Permeation measurements using a getter layer and ion-beam based detection	Thomas Schwarz-Selinger (IPP, Germany)	35 min (O5)
16:40-17:15	JSI Fusion for Energy (F4E) activities involving tritium measurements	Ivan Kodeli (JSI, Slovenia)	35 min (O6)
17:15-17:50	Learnings from dismantling an obsolete tritium installation and decommissioning tritium laboratories	Kris Dylst (SCK CEN, Belgium)	35 min (O7)
17:50-18:20	Discussion on Tritium inventory and control		30 min

HYDROGEN ISOTOPE RETENTION AND TRANSPORT IN NEUTRON-IRRADIATED TUNGSTEN

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Fuel retention in tungsten (W) is one of dominant factors determining tritium (T) inventory in vacuum vessels of future fusion reactors. Permeation of T to coolant may result in uncontrolled T release to the environment. Hence, fuel retention and transport in W are important for assessment of safety and T economy of fusion reactors.

It is known that defects induced in W by neutron irradiation act as strong traps against hydrogen isotopes and lead to significant increase in hydrogen isotope retention [1,2]. The objective of this presentation is to share fundamental knowledges and highlights of latest researches on hydrogen isotope behaviour in neutron-irradiated W. The contents of the talk are:

1. Fundamentals of hydrogen trapping in defects in metals and peculiarity of W;
2. Deuterium retention and transport in W irradiated with neutrons at relatively low temperatures;
3. Influences of irradiation temperatures;
4. Influences of transmutation and alloying; and
5. Remaining issues.

Very strong influence of traps on hydrogen isotope retention and diffusion in neutron-irradiated W will be demonstrated. Mitigation effects by alloying [3] and He seeding in plasma [4] will be also presented. Experimental results on neutron-irradiated W given in the talk were mainly acquired in Japan-US Collaboration Program TITAN Project (2007–2012) [1,2], PHENIX Project (2013-2018) [5] and Collaboration Program of Institute for Materials Research, Tohoku University [6].

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[2] Y. Hatano et al., J. Nucl. Mater. 438(2013)S114.
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MODELLING OF TRITIUM INVENTORY IN PLASMA FACING COMPONENTS IN FUSION DEVICES

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The uptake and transport of hydrogen isotopes HI (Hydrogen, Deuterium and Tritium) in metals used as first wall materials in fusion devices has key implications on their operation: The so called recycling dynamics of implantation & reflection of energetic HI's and effusion of molecular species at thermal energies from the wall back to the plasma, defines momentum and energy balance in the scrape of layer region of the plasma which is in contact with the wall [1]. The retention of HI's that diffuse deep into the material and are immobilized there via trapping at defects poses two further challenges: Firstly a large T inventory is a safety concern in current fusion experiments like ITER. Secondly in a future fusion power plant the retained T is no longer available for burn the plasma core and this loss has to be compensated by increased breeding of T in the blanket. Thus putting even more strain on the already tight T budget defined by the tritium breeding ratio in a fusion reactor. [2]

Therefore, understanding and being able to make quantitative predictions about the transport and retention of HI's for future fusion devices is of key importance. This requires models that can quantitatively describe the tritium inventory in plasma facing components of fusion devices. To make calculations on the required length and time scales of mm to cm and hours respectively, so called diffusion trapping codes are used today. They treat the transport and retention of HI's in metals by dividing the HI into two populations: The solute atoms which are located at interstitial lattice sites and move quickly through the material via diffusion and the trapped atoms which are immobilized by being trapped at a lattice imperfection. In the simplest form the traps have single occupancy and HI's have to de-trap by a thermally activated Arrhenius process to join the solute population, to further migrate through the material. In these models the inventory of hydrogen is then determined by diffusion limited filling of the traps. [3]

The lecture will first explain the main concepts behind these models and show typical experiments that are used to generate the required input data. In particular caveats and ambiguities arising in the interpretation of experiments by these models will be discussed. Then the concept of isotope exchange will be introduced, showing that the classical picture of single trap occupancy: only one HI per trap, is not sufficient to explain the isotope exchange at low temperatures found in experiments. Finally, current areas of research will be presented: In particular the evolution of trap densities and the influence of HI's on the formation of traps will be discussed.

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TRITIATED DUST IN TOKAMAK

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During ITER operation and due to plasma/wall interaction, dust (e.g. small particles are created. They have variable sizes ranging from nanometres to tens of microns. The properties of these particles, such as their ability to be covered by an insulating oxide layer or their surface topology that affects their tritium inventory, are essential to describe their behaviour in the tokamak. After having recalled the various processes of creation of dust occurring during tokamak operation, we will endeavour to describe the physicochemical properties specific to tritium tungsten particles. We will recall their tritium inventory which varies according to their specific surface area. We will then specify how these tritiated particles acquire, over time, a positive electric charge, a load that we will determine. These electrostatic properties modify the adhesion of dust on the surfaces on which they are deposited. We will argue that in the case of a single particle, adhesion is enhanced. However, if the tritiated particle is part of an aggregate (a pile of particles), the adhesion remains unknown but should be very low. Due to the limited free path of the β emission in the material, the tritium inventory transported by the aerosol created during their suspension in the air during a "Loss of Vacuum Accident" for example cannot be measured in real time by conventional methods. A new strategy is necessary for measurements at the workplace, for example, or during release into the environment. We will also present the results of toxicity studies observed during in vitro exposure of lung cells to untritiated/tritiated tungsten particles of 100 nm. Finally, after the collection of these powders by vacuuming, it is necessary to avoid their spreading into the environment. We will specify the different technical solutions envisaged to immobilized these particles.

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ANALYSIS OF FUEL RETENTION IN PLASMA-FACING COMPONENTS. EXPERIENCE FROM D-T OPERATION OF JET WITH CARBON WALLS, CURRENT APPROACH AND PREPARATION FOR NEXT CAMPAIGNS

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There is a vast experience after the operation of the Joint European Torus (JET) with the equimolar deuterium-tritium mixture in years 1997-1998 (DT Experiment-1) [1,2]. Following that campaign, a significant amount of hydrogen as isotopes, both deuterium and radioactive tritium, remained in the vessel wall because of co-deposition with carbon thus forming fuel-rich layers particularly in remote (and cold) areas of the divertor, in regions shadowed from the direct plasma impact. This has had crucial consequences for fusion science and technology: the major change of the wall composition in JET [3,4] and then in ITER [5], i.e. the replacement of carbon plasma-facing components (PFC) by a metal wall composed of beryllium in the main chamber and tungsten or tungsten-coated carbon in the divertor. It has also led to the development of: (i) fuel removal methods [2,6]; (ii) in-situ monitoring of fuel retention by means of laser-based methods [7]; (iii) erosion-deposition studies using tracers and various wall probes [8,9].
This contribution deals with three topics: (a) a reminder of strategy in material handling and fuel measurements in PFC after the DTE1 operation from JET with carbon wall (JET-C); (b) techniques used in studies of tritium from JET-ILW and (c) preparation and installation of erosion-deposition diagnostics prior to the D-T operation in years 2019-2020.

- Ad (a) A large number of limiter and divertor tiles were retrieved from JET-C after DTE-1, when the total tritium inventory exceeded 1 TBq. The development and application of methods to section (cut) big tiles was a crucial step to prepare samples of reduced radioactivity level. This eventually enabled T and D analyses by various methods, such as full combustion followed by scintillography of tritiated water, and nuclear reaction analyses to determine deuterium contents in carbon- and beryllium-rich co-deposits.
- Ad (b) Deuterium is the main fuel species in JET-ILW operated with D2 fuelling, but tritium is a serious contaminant originating both from residue after the DTE-1 and that produced in the D-D fusion. Methods used in the study of D and T will be briefly reviewed.
- Ad (c) Next campaigns with tritium (TT and DT) are planned at JET-ILW in 2019-2020. Retrieval, sectioning and analyses of tungsten and beryllium tiles contaminated with significant T amounts will pose unprecedented difficulties. The level of difficulties may be alleviated when small-size items would be taken for ex-situ studies to determine fuel retention [10]. Therefore, a number erosion-deposition monitors (wall probes) were installed. This also includes a new category: mirrors pre-damaged by ion bombardment (2 and 20 dpa) to simulate neutron-induced surface damage. Such probes will be used to determine retention in damaged materials.

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MEASUREMENT OF TRITIUM ON W DIVERTOR TILES USED IN JET-ITER LIKE WALL CAMPAIGNS USING IMAGING PLATE AND B-RAY INDUCED X-RAY SPECTROMETRY

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The Joint European Torus (JET) performed ITER-like wall campaigns with beryllium (Be) plasma-facing wall and divertor of bulk tungsten (W) and W coated carbon-fiber composite (CFC) tiles. Until now, three experimental campaigns were performed in 2011-2012 (ILW-1), 2013-2014 (ILW-2) and 2015-2016 (ILW-3) [1]. After each campaign, post-mortem analysis was carried out for investigation of erosion and deposition of wall materials, dust generation, deuterium and tritium retention, etc.

In this study, by using imaging plate (IP) and β-ray induced x-ray spectrometry (BIXS), tritium retention in W-covered CFC divertor tiles used in JET ILW-1 and ILW-3 was examined within the frame work of the ITER Broader Approach Activity in the International Fusion Energy Research Center (IFERC), National Institutes for Quantum and Radiological Science and Technology (QST), Rokkasho Japan. The specimens examined were disks (17 mm diameter) cut from the tiles. Also, tritium depth profiles in tiles were calculated by applying Monte Carlo simulation tool kit Geant4.

The major results obtained were;

- In the case of ILW-1 tiles, strong enrichment of tritium in Be deposition layers was observed. The tritium retention of the inner divertor region was higher than the outer region due to heavier deposition.
- Contrary, tritium concentration in Be deposition layers formed on ILW-3 tiles was significantly low compared with ILW-1 tiles. As a result, tritium distribution on ILW-3 tiles was more uniform than that on ILW-1 tiles.
- It is plausible that higher input power and higher energy deposition on the ILW-3 tiles enhanced tritium desorption from Be deposition layers.

This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014-2018 under grant agreement No 633053 and the ITER Broader Approach Activities. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

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ACTIVITIES AT THE UNIVERSITY OF LATVIA FOR TRITIUM MEASUREMENTS FROM FUNCTIONAL AND PLASMA FACING MATERIALS

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In the frame of EUROfusion consortium programme, Institute of Chemical Physics and Faculty of Chemistry of the University of Latvia are performing investigations of tritium behavior and release from neutron irradiated functional breeder blanket zone materials and plasma facing components of the fusion reactors, in order to extend the knowledge about materials for the use in International Thermonuclear Experimental Reactor and DEMOnstration power plant.

Tritium determination has been performed in beryllium used as plasma facing protection material of the Joint European Torus vacuum vessel in form of tiles and pebbles irradiated in nuclear reactor (HFR, Petten, the Netherlands) fission spectra as neutron multiplier.

Tritium release characteristics from neutron irradiated beryllium pebbles shows importance of material microstructure, fabrication method, irradiation temperature on the accumulated tritium amount and release characteristics. Analysis of tritium in a plasma facing tiles indicates to importance of positioning in the vacuum vessel as well as position of analysed sample in a separate tile on tritium accumulation and distribution on a surface and in bulk of a sample [1].

Tritium accumulation, temperature programmed thermodesorption analysis of neutron irradiated tritium breeding material, lithium containing ceramic pebbles, allow to reveal different forms of tritium: HTO, HT.

Behavior of tritium in carbon fiber composite materials have been used as divertor and plasma facing materials in fusion reactors and up to now were analysed by wide spectra of physical methods.

During active operation of fusion reactor, plasma-wall interactions (erosion, formation of dust, fullerenes and long-chain hydrocarbons, tritium retention) occur. Tritium distribution in plasma facing tiles and dusts was determined with full combustion and liquid scintillation method, desorption process is analysed by TDS to investigate tritium retention mechanisms. Complex system of TG/DTA in couple with FTIR analysing the combustion products allow to determine the combustion pathways and characterize the materials behavior under action of elevated temperatures as well as atmosphere (air, argon).

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PERMEATION MEASUREMENTS USING A GETTER LAYER AND ION-BEAM BASED DETECTION

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A novel method to measure permeation of hydrogen isotopes through tungsten is presented. Unlike common methods where permeating hydrogen isotopes need to be measured in a dedicated setup in situ, this method collects the permeated hydrogen in a getter layer deposited on the back side of the tungsten sample. The amount of hydrogen retained in the getter and hence the permeated hydrogen is then measured ex situ by ion beam analysis methods [1]. The method is applicable for gas driven as well as ion or plasma driven permeation experiments. As it needs no special installation, it is ideally suited to measure permeation for different conditions or on different setups such as those typically used for deuterium-retention measurements. It is also not limited to laboratory studies but can be applied in fusion devices, too. As IBA methods do not only allow to derive total amounts retained in the getter but also the depth distribution of trapped deuterium in tungsten the method also provides information about the defect distribution, which is necessary to interpret the permeation results.

Results will be presented for plasma driven permeation of deuterium through 24.5 µm thick tungsten foils at 300 K and 450 K [2]. The D(3H,p)4He nuclear reaction was used to quantify the permeated deuterium. Getter layers of zirconium, titanium or erbium were tested for this case on the unexposed side of the foil. A cover layer system on the getter prevents direct loading of the getter with deuterium from the gas phase during plasma loading. In addition, it enables the distinction of deuterium in the getter and at the cover surface.

Special emphasis was put on the influence of sub-surface damage evolution on the permeation. Microstructural analysis by scanning electron microscopy, assisted by focused ion beam, revealed sub-surface damage evolution at 300 K exposure temperature. This damage evolution was correlated with a significant evolution of the deuterium amount retained below the plasma-exposed surface. Although both of these phenomena were observed for 300 K exposure temperature only, the deuterium permeation flux at both exposure temperatures was indistinguishable within the experimental uncertainty. The permeation flux was used to estimate the maximum ratio of solute-deuterium to tungsten atoms during deuterium-plasma exposure at both temperatures and thus in the presence and absence of damage evolution. Diffusion-trapping simulations revealed the proximity of damage evolution to the implantation surface as the reason for an only insignificant decrease of the permeation flux.

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JSI FUSION FOR ENERGY (F4E) ACTIVITIES INVOLVING TRITIUM MEASUREMENTS

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In the future fusion reactors, such as ITER or DEMO, tritium will be produced by bombardment of lithium atoms with neutrons. Several types of special Tritium Breeder Modules (TBM) will be installed in the ITER reactor to demonstrate the self-sufficiency of tritium production. Due to its high interest for future fusion devices several activities within the European fusion programme (Fusion for Energy – F4E) studied tritium production issues in the past. The following activities involving JSI participation in the last 15 years will be presented:
1) Helium-cooled Pebble Bed (HCPB) Tritium Breeder Module (TBM) Mock-up benchmark experiment [1] was performed in 2005 at the 14-MeV d-T Frascati Neutron Generator (FNG) at ENEA Frascati to experimentally verify the accuracy of Tritium Production Rate (TPR) calculated using the modern nuclear codes and data for the HCPB, one of the two blanket designs developed within the European fusion programme. JSI participated with the pre- and post-analyses using the sensitivity and uncertainty codes.
2) Helium-Cooled Lithium-Lead (HCLL) TBM Mock-up experiment [2-4] was performed in 2008, likewise at ENEA FNG facility to test the tritium self-sufficiency and the accuracy of the computational tools for the analysis of the 2nd European TRP concept.
3) Experimental investigation of the potential use of the 55Mn(n,γ)56Mn reaction as a tritium production monitor [5]: foils of certified reference materials Al-1%Mn and Al-0.1%Au, TLD(LiF) and Li2O were irradiated in different irradiation channels in the JSI TRIGA research reactor, both bare, and under Cd and boron-nitride to experimentally investigate the relationship between the 55Mn(n,γ)56Mn reaction and the tritium production rate in 6Li(n,t).

The FNG benchmarks were performed in cooperation between the ENEA Frascati, KIT, TUD and JSI, and the Mn measurement involved the cooperation between JSI and AGH, Krakow, Poland.

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LEARNINGS FROM DISMANTLING AN OBSOLETE TRITIUM INSTALLATION AND DECOMMISSIONING TRITIUM LABORATORIES

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In the last decades two major tritium dismantling / decommissioning projects were started at SCK•CEN.

Between 2003 and 2009 two rooms that served as tritium laboratory at SCK•CEN were decommissioned [1]. The tritium laboratories were initially commissioned in 1975 for a tritium inventory of 37 TBq, with a strong focus on handling tritium as HTO. For the first laboratory room, the decommissioning strategy was to free release as much materials as possible. For the decommissioning the second laboratory room a more pragmatic approach was used. At the expense of extra waste generation, the decommissioning could be done faster.

In 2017 a dismantling study started for the VNS (Variable Neutron Shield). The VNS installation is located in the reactor building of the BR2 research reactor and is not related to the tritium laboratory. At the start of this dismantling study the VNS installation wasn't been operational for more than 20 years. In this installation substantial amounts of tritium were generated by neutron irradiation of pure He-3 gas. The formed tritium was removed from the He-3 gas flow by sending it through titanium retention traps (or Ti-getters). The maximum retention capacity of each Ti getter was 370 TBq, but the actual tritium content of each of the 5 present Ti-getters was unknown. Three of these Ti-getters were still connected to the VNS installation whilst two others were stored separately in stainless steel containers. Also a tritium containing experimental NaK getter was stored separately. NaK is a liquid metal that can react heavily with oxygen, water or NaK oxides. Also the tritium content of the NaK getter is unknown. A proper characterization together with finding disposal routes for these getters was indicated as an important first step in this dismantling exercise. But this required dismantling getters from the VNS installation.

Sharing the learnings from these projects can provide useful perspectives for decommissioning of tritium installations: from the waste, cost and labour impact of the used strategy to the difference between decommissioning an installation from which operational knowledge is present or not.

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Scientific programme, Wednesday 27th of March

Topic : Radiotoxicity/ecotoxicity		Chair : Veronique Malard CEA	
8:30-9:20	Overview of tritium effect	Laurence Lebaron-Jacobs (CEA, France)	50 min (L8)
9:20-10:10	Assessing ecotoxicological impact of tritium	Awadhesh Jha (Plymouth, UK)	50 min (L9)
10:10-10:30	Coffee break		20 min
10:30-11:20	Epidemiological studies of tritium exposure	Richard Wakeford (University Manchester, UK)	50 min (L10)
11:20-11:55	Tritiated and non tritiated ITER-like tungsten particles in lung-derived cells: An epigenotoxic study	Chiara Uboldi (CEA, France)	35 min (O8)
11:55-12:20	Determination of radiological parameters (H-3) of drinking water in Poland in 2010-2015	Agnieszka Fulara (CLOR, Poland)	25 min (C2)
12:20-14:20	Lunch Break		120 min
Topic : Radiotoxicity/ecotoxicity		Chair : Awadhesh Jha UK	
14:20-15:10	Biokinetics of low levels of tritium as HTO or OBT and its genotoxicity relative to gamma-radiation in a laboratory mouse model	Dmitry Klokov (CNL, Canada)	50 min (L11)
Topic: Tritium dosimetry		Chair : Awadhesh Jha UK	
15:10-16:00	Dosimetry of tritium in humans and non-human biota	Francois Paquet (IRSN, France)	50 min (L12)
16:00-16:20	Coffee break		20 min
16:20-17:10	Tritium dosimetry: Modelling approaches at the sub-cellular scale	Giorgio Baiocco (UNIPV, Italy)	50 min (L13)
17:10-17:35	Uncertainty of internal dose estimation from tritium exposure	Anna Pantya (Energia, Hungary)	25 min (C3)
17:35-18:05	Discussion on Radiotoxicity/ecotoxicity/dosimetry		30 min
20:00 -	Tritium School Dinner – Ljubljana city center – Restaurant Vodnikov hram https://goo.gl/maps/pLLC91MsFP62		

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OVERVIEW OF TRITIUM EFFECTS

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Whatever its origin, tritium is extremely mobile in all biological systems and in the environment, and exchanges with hydrogen atoms within biological molecules (DNA, proteins...). Tritium, a low energy beta emitter, is considered as a low radiotoxicity element based on experimental studies. Its average path in water is low. Nevertheless, questions remain as to the level of risk to be attributed because of its high density of heterogeneous distribution ionisation. Drinking water and food are sources of tritiated water (HTO) and organically bound tritium (OBT). According to the publication n°56 of the International Commission on Protection against Ionizing Radiation (ICRP), approximately 90% are in the form of HTO and approximately 10% in the form of OBT after incorporation and transformation of the tritiated molecules [1]. However, extensive human and animal data underline the value of identifying a compartment for slowly eliminated tritiated molecules, corresponding to tritium incorporated into biological structures (DNA) of slowly renewing tissues (non-exchangeable OBT) (Harrison et al., 2002). In addition, studies on tritium biokinetics in rodents show that the behaviour of OBT differs from that of HTO.

The experimental data come mainly from cellular and animal studies after exposure to tritiated water (HTO): tritium can cause early cellular lesions and an excess of cancers even more marked when the cumulative dose and dose rate are high [3]. However, few studies analyse the biological consequences of OBT exposure. Moreover, the results differ greatly depending on the experimental protocol.

The need to reassess the current biokinetic model, the high density of heterogeneous distribution ionisation of tritium, the relevance of the dose, and the lack of data at environmental concentrations do not call into question the low radiotoxicity of tritium, but may eventually lead to review estimates.

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ASSESSING ECOTOXICOLOGICAL IMPACT OF TRITIUM

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Tritium, a radioisotope of hydrogen (half-life ~ 12 years) is ubiquitously distributed in the environment, both as a result of natural phenomenon (e.g. interaction of cosmic rays with nitrogen in the atmosphere) and anthropogenic activities (e.g. discharges from nuclear reprocessing and power plants). Compared to other radionuclides, tritium is discharged in huge quantity by nuclear establishments (routinely or accidentally), mainly as tritiated water (HTO) ultimately in the hydrosphere to which organisms are constantly exposed. This raises the environmental health concerns. Environmental monitoring and assessment of potential impact of HTO on human and natural biota is therefore important from both scientific and regulatory perspectives. Despite the concern, not enough information is available in the literature assessing the potential bioaccumulation and detrimental impact of HTO on the natural biota.

Adopting an integrated approach, we have evaluated the potential biological impact of HTO on different life stages of aquatic invertebrates. Our initial studies had suggested that HTO is capable of inducing genetic damage in adult and early life stages of marine mussels at much lower internationally recommended dose rate. We also demonstrated that organic (i.e. tritiated glycine) and inorganic tritium (i.e. HTO) differentially accumulate in the tissues and induce different levels of genetic damage in the haemocytes of mussels. In order to estimate the radiation dose following exposures to HTO, we adopted 4 different methods of dose estimation in adult mussels. Our study suggested that dose estimation software, ERICA tool is useful for estimating radiation dose. To maximise the accuracy for dose estimation, it was realised that it is essential to quantify the activity within the organism as this would remove any assumptions about concentration ratio (CR). The use of the ERICA tool with water activity concentrations only, tends to over-estimate dose (due to conservative assumptions for CR). This has some implications as overestimation of dose could lead to artificially inflated parameters (e.g. ED_{50}) which in turn could lead to underestimation of radiotoxicity.

Potential impact of radionuclides should be assessed along with other abiotic and chemical factors. Temperature is an abiotic factor of particular concern for assessing the potential impacts of radionuclides on marine species. We assessed the tissue-specific accumulation, transcriptional expression of key genes and genotoxicity of HTO to marine mussels at either 15 or 20°C over a 7-day time course with varying sampling time. Our study suggested a significant induction of DNA strand breaks showing temperature- dependent time shift. At 15°C, DNA damage only significantly elevated after 7d in contrast to 25°C where a similar response was observed after only 3d. Transcription profiles of *hsp* 70, *hsp* 90, *mt* 20, *p*53 and *rad* 51 genes indicated potential mechanisms behind this temperature-induced acceleration of genotoxicity. We further studied the interaction of HTO with two different concentrations of zinc (Zn), as environmentally relevant metal, in the presence of dissolved organic ligands (humic acid as dissolved organic carbon-DOC) and elevated temperature. Mussels were exposed for 14 d to these mixtures to investigate (a) 3H partitioning in the soft tissues and (b) DNA damage in haemocytes. Overall, results suggested a clear antagonistic effects of Zn on HTO-induced DNA damage at all Zn concentrations used. The interaction of DOC with 3H was variable, with strong 3H-DOC associations observed in the first 3 d of the experiment. Overall, the study highlights the importance of potential mixture effects in the environmental risk assessments of radionuclides.

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EPIDEMIOLOGICAL STUDIES OF TRITIUM EXPOSURE

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The health effects of tritium are of interest because it is a radioisotope of hydrogen, a pure beta-particle emitter, and the electron ejected from the triton is of low energy so has a short range and is densely ionizing relative to most other beta-particles. Consequently, a number of epidemiological studies have been conducted of those exposed to tritium occupationally and in the environment [1]. Although workers exposed to tritium offer an opportunity to examine potential risks to health, quantification of tritium-specific doses has been carried out for only a few studies, so that the conclusions that may be drawn from epidemiological studies in terms of tritium exposure risks are limited. Studies of environmental exposures are even more difficult to interpret reliably because tritium-specific doses are hardly ever available. However, an international collaborative effort to study workers exposed to tritium, which uses tritium-specific dose estimates, may be capable of meaningfully assessing the risk to health from exposure to tritium [2]. Nonetheless, the presently available epidemiological evidence does not indicate that the risk to health of tritium exposure has been seriously underestimated [2].

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TRITIATED AND NON TRITATED ITER-LIKE TUNGSTEN PARTICLES IN LUNGDERIVED CELLS: AN EPIGENOTOXIC STUDY

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The high density and the elevated melting point were some of the reasons that made tungsten the material chosen to interact with the plasma of the ITER thermonuclear fusion reactor (www.iter.org). Although its robustness and low plasma sputtering yield might guarantee a limited erosion of the tokamak inner wall during plasma operation, tungsten particles (W-Ps) will nevertheless be formed. Because, following a loss-of-vacuum-accident (LOVA), W-Ps might be potentially released into the environment and induce accidental or occupational exposure, the study of their cytotoxic and epigenotoxic potential results of great importance for the safety and well-being of those working in or living around the fusion facility.

For this reason, plasma sputtering and laser ablation ITER-like W-Ps were synthesized and labelled with tritium, and their potential toxicity was assessed on human-derived immortalized bronchial cells (BEAS-2B cells). The physicochemical characterization showed that plasma and laser derived ITERlike W-Ps particles had similar mean size diameter (100-200 nm), but they had a different specific surface area.

Plasma and laser produced W-Ps exerted different degrees of cytotoxicity depending on the presence/absence of hydrogen on their surface. Differences were observed also in their genotoxic potential, which was investigated by performing the centromeric cytokinesis-block micronucleus cytome test (CBMN-cyt) and the alkaline comet assay. While both types of ITER-like W-Ps induced micronuclei formation and primary DNA damage, the laser ablation-derived ones seemed to have a slightly stronger genotoxic potential on BEAS-2B cells. These results might be correlated to the enhanced oxidative stress induced by laser W-Ps compared to the plasma counterparts. Furthermore, DNA methylation analysis revealed that both plasma and laser W-Ps had some epigenetic effects on BEAS-2B cells.

This work is supported by the A*MIDEX project (ANR-11-IDEX-0001-02) funded by the "Investissements d'Avenir" French Government program, managed by the French Research Agency (ANR).

DETERMINATION OF RADIOLOGIGAL PARAMETRS (H-3) OF DRINKING WATER IN POLAND in 2010-2015

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Systematic radiochemical measurements of the radioactive isotopes were performed in drinking waters in Poland. The investigations of tap water radioactivity in 2010-2015 were performed in largest cities of Poland. We studied water originating from 33 cities.

Collected and analyzes were performed in the waters coming from 82 water treatment plants.

The origin of most water samples was from surface water from rivers and lakes and deep water. In each sampling point a quantity of 20 liters of water was taken. ^{137}Cs and ^{90}Sr were determined in the same 15 l sample. In the rest 5-liters sample, the tritium activity and total alfa and beta radioactivity were determined.

The method used to determine tritium in water was based on the method of electrolytic tritium enrichment. The enrichment consists in electrolysis of water sample with addition of 20% NaOH solution. Water undergoes decomposition into oxygen and hydrogen, whereas the quantity of NaOH remains the same.

Due to different ionic mobility of tritium and hydrogen, decomposition of water particles not containing tritium takes place faster. Therefore, in the process of electrolysis, the percentage of HTO in the residue is becoming higher.

Application of this method allows to reduce tritium detection limit in water from about 10 Bql^{-1} in direct measurements to 0.5 Bql^{-1}.

The tritium concentration in drinking water ranged from values below detection limit (0.5 Bql^{-1}) to 5.8±0.9 Bql^{-1}. The average concentration of tritium calculated for all water samples tested in 2010-2015 was 1.2±0.2 Bql^{-1}.

According to the Ministry of Health regulations issued on 29 of march 2007, concerning the quality of drinking water designed for public consumption, tritium concentration in drinking water must not exceed 100 Bql^{-1}.

The analyzes show that the tap water tested meet the requirements specified in the regulations Ministry of Health of 29 March 2007.

The work has been performed for National Atomic Energy Agency.

BIOKINETICS OF LOW LEVELS OF TRITIUM AS HTO OR OBT AND ITS GENOTOXICITY RELATIVE TO GAMMA-RADIATION IN A LABORATORY MOUSE MODEL

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It has increasingly been recognized that both nuclear fission and, in the future, fusion technologies will play a substantial role in the environmentally sustainable low-carbon energy production. However, nuclear energy technologies involve the production and use of significant quantities of tritium, a highly volatile radioisotope of hydrogen (3H). Public concerns regarding 3H releases into the environment and potential health impact are very high. They are poorly addressed by current knowledge on biological effects and health risks associated with the exposure to low levels of 3H. Canadian Nuclear Laboratories, in collaboration with the French Institut de Radioprotection et de Sûreté Nucléaire (IRSN, France), embarked on a large scale mouse in vivo study to explore biokinetics and relative toxicity of low levels of 3H present in drinking water for 1 or 8 months in the form of HTO or organically bound 3H (OBT) [1]. The concentrations of 3H were 10 kBq/L (the WHO action level limit of 3H in water is 7 kBq/L), 1 or 20 MBq/L. The biokinetics experiments revealed that, in contrast to the current ICRP model, HTO and OBT did not differ significantly in their biokinetics characteristics and this was not dependent on 3H concentration. Using the biokinetics data, exposure dose rates were determined to carry out external gamma-irradiation study at matching dose rates and cumulative doses in order to evaluate toxicity of 3H relative to gamma-radiation. Cumulative exposure doses were calculated to be 0.01, 1 and 20 mGy at 1 month of exposure, and 0.08, 8 and 160 mGy at 8 months of exposure. Biological effects were estimated by gross pathological examination of 9 different tissues and organs, as well as by histological, cellular, cytogenetic and molecular measurements of features and markers of toxicity, inflammation and DNA damage. Summarized results show that: 1) neither HTO nor OBT exposures resulted in detectable biological alterations at the lowest dose of 10 kBq/L; 2) 1 (for OBT) and 20 (for HTO and OBT) MBq/L concentrations triggered biological responses, whereas gamma-radiation at matching dose rates did not cause a response; 3) biological responses to 3H were tissue specific, with some being deleterious and some ostensibly beneficial [1]. When estimated using chromosomal aberrations in peripheral blood lymphocytes, relative biological effectiveness (RBE) of OBT was higher than that of HTO and increased as the dose decreased [2]. It should be noted, however, that the high RBE at low doses is not supported by the observed lack of any effect at the lowest 3H concentration of 10 kBq/L and may be explained by the assumptions and approximations of the mathematical and experimental models. Overall, our results provide new information that informs tritium radioprotection standards and help design future studies to obtain mechanistic insights into the biological effects of low levels of 3H and associated health risks.

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DOSIMETRY OF TRITIUM IN HUMANS AND NON-HUMAN BIOTA

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During this last decade, there has been much debate about the dosimetry of tritium. Most of the debate centered on the radiation weighting factor to be applied, due to the very short-range of the beta particles emitted after disintegration of tritium, and also because of its heterogeneous distribution in tissues when incorporated as chemical forms that have a high affinity for DNA. The origins of the debate were partly a misunderstanding of the tools developed to calculate doses, and of their ranges of applicability.

The International Commission on Radiological Protection (ICRP) has proposed a methodology to calculate doses resulting from incorporated radionuclides based on the type (alpha, beta, gamma, neutrons) and energy of radiation, but not on specific elements. As a consequence, the doses resulting from the incorporation of tritium are calculated as for every other element of the periodic table. In the range of low doses, which may induce stochastic (cancer/heritable) effects in humans, the quantity effective dose E(50) serves for optimization procedures and for the demonstration of compliance with doses limits. Effective dose is calculated through a series of steps: the absorbed dose is defined as the mean energy imparted to matter of mass dm divided by the mass dm; then, the equivalent doses to individual target organs or tissues are calculated as a sum of absorbed doses, weighted by the radiation weighting factor w_R. For tritium and all beta radiation, w_R is set equal to 1; and finally, the effective dose is defined by a sum of tissues equivalent doses, weighted by their respective tissue weighting factor w_T. It is important to note that tissue and radiation weighting factors used for the calculation of the effective dose are defined for stochastic effects only. A revision of the effective dose coefficients for tritium for workers has been published in 2016 [1]; those for the members of the public are to be published in 2019.

In the range of doses that may induce deterministic effects (tissues reactions), the quantity to be used is the mean absorbed dose to the organ or tissue, weighted by an appropriate value of the Relative Biological Effectiveness (RBE) for the radiation and for the biological endpoints of concern. A large variation of RBE is observed according to the endpoint considered and is described in relevant ICRP Publications [2].

For the dosimetry of tritium in non-human biota, work has just been completed and will be published soon by ICRP. Whereas protection of humans has focused on avoiding deterministic and stochastic effects, protection of biota has largely focused on tissue reaction endpoints relevant to population viability. A review of RBE data relevant to biota for tritium has reported values centered around 1.5–2 compared with X-rays, and 2–2.5 compared with gamma rays. Lower values are observed for deterministic effects compared to stochastic effects. It is therefore proposed that for protection purposes, radiation weighting factors w_B for biota regarding tritium and all low LET radiations should be set to 1, and used to modify the absorbed dose rates to relevant Reference Animals and Plants (RAPs). Use of a single value of 1 for all low LET radiations is consistent with the approach taken to protection of humans. A caveat is made that if exposures to tritium beta particles, or to other low energy, low LET radiations, are within or close to the derived consideration reference level (DCRL) band, additional review, and possible modification of w_B, might be warranted.

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[2] ICRP 2003. Relative Biological Effectiveness, Quality Factor and radiation weighting factor. Ann ICRP 33(4)

TRITIUM DOSIMETRY: MODELING APPROACHES AT THE SUB-CELLULAR SCALE

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When we talk about radionuclide intake, dosimetry is a very peculiar concept: biokinetic models for individual elements and their radioisotopes are used to calculate the total number of radioactive decays occurring within specific tissues, organs or body regions (source regions) during a given period of time. Dosimetric models are then used to calculate the deposition of energy in all important organs/tissues (targets) from each source region, taking account of the energies and yields of all emissions. Dose coefficients can be given, as values of committed equivalent dose and committed effective dose per unit intake of specified radionuclides by ingestion or inhalation (units: Sv Bq^{-1}).

This whole scheme of calculation relies on simplifying assumption, with severe limitations in case of a non-homogeneous distribution in tissue/cells of radionuclides emitting short-range decay products. Tritium might be regarded as an extreme case of a radionuclide for which:

- decay products are β electrons with an unusual short range (average energy of 5.7 keV, corresponding to 0.5 μm in water/tissue, $E_{\beta\text{-max}}$ = 18.6 keV), much shorter than cell nucleus dimensions; this means that the sub-cellular location of tritiated products (and their congruence with more radiosensitive targets) is of utmost importance in determining the biological effects;
- the chemical speciation is also crucial in determining the effects of its radioactive decay, because of the very wide range of compounds in which the tritium atom may be firmly bound, that again results in different sub-cellular distributions and distribution kinetics.

In this lecture we will review the main concepts at the basis of sub-cellular dosimetry for short-range emitters and related biological effects, as well as which are the calculation approaches and simulation tools at our disposal (a.o. track structure calculations [1, 2], micro/nanodosimetry [3], analytical approaches [1]) to achieve a thorough assessment of energy deposition following contamination with tritiated particles.

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UNCERTAINTY OF INTERNAL DOSE ESTIMATION FROM TRITIUM EXPOSURE

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Tritium may exist in several chemical and physical forms in workplaces, common occurrences are in vapour or liquid form (as tritiated water) and in organic form (e.g. thymidine) which can get into the body by inhalation or by ingestion. For internal dose assessment it is usually assumed that urine samples for tritium analysis are obtained after the tritium concentration has reached equilibrium inside the body fluid following an intake.

The measurement of tritium (tritiated water) can be performed with Liquid Scintillation Counting method. Uncertainties of urine measurement (e.g. colour quench effect on the actual efficiency) are well known and these cause only a few percent deviation in the measured activity. However, during the estimation of the internal radiation dose due to tritium several other factors have to be considered, e.g. the time of intake, the time span between sampling and last intake, route and pattern of intake and the chemical form of tritium. These parameters have a significant influence on the final result of the dose estimation. Generally, the uncertainty of the measurement itself is usually significantly lower than the uncertainty of the model used for the dose estimation in internal dosimetry. Sources of uncertainty will be presented in this report.

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Scientific programme, Thursday 28th of March

Topic : Tritium Waste		Chair: Karine Liger CEA	
8:30-9:20	The effect of soft housekeeping wast composition on the accuracy of common tritiated waste measurement techniques	Robert Vale (UKAEA, UK)	50 min (L14)
9:20-10:10	French strategy for solid tritiated waste management	Karine Liger (CEA, France)	50 min (L15)
10:10-10:25	Coffee break		15 min
10:25-11:15	UK approach to tritiated waste processing and disposal	Dave Coombs (UKAEA, UK)	50 min (L16)
11:15-11:50	H3AT: Tritium advanced technology	Damian Brennan (UKAEA, UK)	35 min (L17)
11:50-13:00	Lunch - Lecture – Institute open days at 12:00		70 min
13:00-15:00	Adjourn - lab tour Accelerator - optional		

THE EFFECT OF SOFT HOUSEKEEPING WAST COMPOSITION ON THE ACCURACY OF COMMON TRITIATED WASTE MEASUREMENT TECHNIQUES

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Fusion and fission industries generate soft operational wastes which are contaminated with tritium from different processes. Regulators require appropriate characterisation of these wastes to identify and quantify the levels of tritium (and other radionuclides), which may be present, to reduce the opportunity for release into the environment. Currently characterisation of lower activity soft waste occurs through

1. Taking of small mass solid samples for tritium analysis
2. Monitoring of off gas within a package
3. Taking of larger mass solid samples for soaking and analysis of the soak liquid

Research being carried out as part of H2020 TRANSAT will investigate characteristics of different polymers (PU, PP, PVC, PE) together with cellulose through an experimental programme of work. The nature of tritium (OBT,HTO, HT) found within the samples will be used to assess the appropriateness of the above characterisation techniques for specific material types.

TRANSversal Actions for Tritium

FRENCH STRATEGY FOR SOLID TRITIATED WASTE MANAGEMENT

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The lecture will focus on tritiated waste management strategy as it is planned in France, in particular for future ITER tritiated waste.

Currently, most tritiated waste inventory is due to military applications, civil research activity and small producers like pharmaceutical and medical research laboratories. Due to the research on fusion reactor using tritium as fuel and in particular due to the development of ITER (International Thermonuclear Experimental Reactor), the volume of solid tritiated waste is expected to increase more than 10 times in the next 40 years. Hence, a first step of the lecture will be dedicated to the description of the inventory and characteristics of current and foreseen tritiated waste.

Then, the French general approach for nuclear waste will be described and the specific challenges posed by this radioactive waste containing tritium will be presented [1]. The feedback of nuclear waste final repository operation will give examples of issues raised by tritiated waste storage. A detailed description of the solutions planned for the various waste categories is given as well as the implementation expected for the ITER tritiated waste, including the features of a future interim storage facility [2].

Several options to reduce temporary storage duration [3-5] and to minimise out-gassing rates and tritium discharges into the environment are under study [6] and will be presented. Finally, the first lessons learned for fusion development and their extrapolation to future reactors are outlined.

[1] J. Pamela, et al., Fus. Eng. and Des. 89 (2014) 2001
[2] D. Canas, et al., Fus. Sc. and Tech. 67:2 (2015) 290
[3] M. Kresina, et al., Proceedings of 29th Symposium on Fusion Technology (2016)
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UK APPROACH TO TRITIATED WASTE PROCESSING AND DISPOSAL

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Fusion research is exciting and so too are the challenges of managing the liabilities generated from this important work. The impact to the environment from tritiated waste, is considerably less than that from conventional nuclear power and will not place a burden on future generations.

In the UK, staff working on the Joint European Torus (JET) optimise waste management within the UK's regulatory framework to drive new standards in 'Best Available Techniques' (BAT) and application of the waste hierarchy; delivering environmentally responsible outcomes. Provenance, waste characterisation and robust practices are key, but so too is innovation to maximise recycling and minimise disposal. For the first time on an industrial scale, UKAEA has created processes to remove tritium from 'real' fusion waste and recover it for reuse in fusion research; thereby closing the fusion fuel cycle.

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H3AT: TRITIUM ADVANCED TECHNOLOGY

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A new tritium facility is being built at the UKAEA's Culham Science Centre near Oxford, UK. The new facility is named H3AT standing for Tritium Advanced Technology. This £40m facility is part of an £86m investment in Culham by the UK Government under the banner of a National Fusion Technology Platform (NFTP). The other half of the NFTP are the Fusion Technology facilities comprising a Joining and Additive Manufacturing Laboratory, a Materials Technology Laboratory to develop and qualify new materials and a Module Test Facility for thermomechanical and electromagnetic testing of components under fusion relevant conditions.

At the heart of the H3AT facility will be a fusion relevant 100g tritium processing loop comprising of a depleted uranium tritium storage bed and distribution system, a palladium membrane reactor based impurity processing system, a cryogenic distillation isotope separation system, a CECE water detritiation system and an air stripping column based atmosphere detritiation system. The loop will be able to operate in closed cycle to simulate a complete fusion fuel cycle. Provision will be made to allow alternative technology subsystems to be added in the future. The loop will also be able to feed a flexible suite of enclosures and glove boxes to allow a broad range of tests and experiments for fusion tritium research in support of ITER, the EUROfusion DEMO programme and for non fusion tritium R&D.

The facility will also contain dedicated laboratories to develop materials detritiation processes, a tritium wet chemistry laboratory and a facility to handle other beta emitters notably Carbon 14.

This presentation will describe the new H3AT facility in terms of its technical capabilities and the progress to its realization.

Editors: Sabina Markelj, Mathilde Bazin-Retours, Matic Pečovnik

Conference logo: Mathilde Bazin-Retours, LGI Consulting, France

Website: Mathilde Bazin-Retours, LGI Consulting, France

Printed in March 2019

Number of printed copies: 80

Publisher: "Jožef Stefan"Institute, Jamova cesta 39, Ljubljana, Slovenia

Ljubljana 2019

Price: 0,00 €

The workshop was supported by the TRANSAT project, which received funding from the Euratom Research and Training Programme 2014-2018 under grant agreement n° 754586.

CIP - Kataložni zapis o publikaciji
Narodna in univerzitetna knjižnica, Ljubljana

546.11.027*3(082)

TRITIUM School (1 ; 2019 ; Ljubljana)
Book of abstracts / First Tritium School, 25th - 28th March 2019 ; [editors Sabina Markelj, Mathilde Bazin-Retours and Matic Pečovnik]. - Ljubljana : Jožef Stefan Institute, 2019

ISBN 978-961-264-147-4
1. Markelj, Sabina
299224832