THIRD EDITION

TRITIUM SCHOOL

BOOK OF ABSTRACTS

18 - 22 March 2024

Palais du Pharo
58 Bd Charles Livon, 13007 Marseille

Online and Onsite Event

www.titans-project.eu

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Day 01

Monday 18th March

8:00-9:30	Registration	
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Day 02

Tuesday 19th March

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12:00-12:50 (50 min)	L10 - State-of-the-art review of detritiation treatments applicable to metallic tritiated waste	Karine Liger (CEA, France)
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Day 05

Friday 22nd March

ITER visit

Departure of the bus to Cadarache at 8:30, from Marseille.

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TRITIUM PROCESS TECHNOLOGY FOR FUSION FUEL CYCLES

Fusion energy holds the promise of harnessing energy from fusing atoms together to generate net electrical power. The most commonly studied fusion reaction is deuterium-tritium (DT) fusion which creates neutrons and helium in a magnetically confined and heated plasma. With low amounts of DT in the plasma actually fused and the need to remove the helium “ash”, tritium and deuterium must be continuously withdrawn from the reactor and recycled. Since tritium is consumed in the fusion reaction it must also be produced in sufficient quantities to sustain the fusion process.

This presentation will give a high-level overview of the DT fusion fuel cycle, waste gas processing, and tritium stripping and tritium recovery technologies. The importance of tritium recovery technologies within the context of environmental accumulation of tritium from fusion energy will also be discussed. The implications of different fusion approaches on the fuel cycle will also be discussed.

Lecture 02

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ASSESSMENT OF TRITIUM AND HYDROGEN TRANSFERS IN NEW GENERATION FISSION REACTORS

Due to its high diffusivity in metals and especially for high temperature conditions, the tritium produced in the core of fission reactors is likely to migrate easily through the different circuits. Potential release of tritium in gaseous form, by permeation through pipings and components walls, is also an important phenomena to be evaluated for the control of safe operating conditions and of environmental impact. In the case of new generation Sodium Fast Reactors (SFR), the control of tritium inventories and releases is an important issue due to its higher effective transfer into sodium through stainless steel claddings, and due to its atomic form once dissolved in sodium.

A global review of the different transfer contributions and physico-chemical phenomena to be considered for tritium (and more generally hydrogen isotopes) in SFR is presented as well as the equivalent behavior of stable hydrogen produced in tertiary circuit in the case of classical water/steam cycle (Rankine thermodynamical cycle). Physical models used for the evaluation of the different transfers and phenomena are presented as well as the qualitative influence of major parameters.

Calculation tools developed for the evaluation of Tritium (and hydrogen) transfers and inventories are described. Illustration results are presented in terms of tritium activities and hydrogen concentrations in each circuit of the reactor, but also in terms of transfer fluxes and releases. In addition, the distribution of the various transfer contributions (between primary, secondary and tertiary circuits) is analyzed as well as the typical influence of key parameters.

An application is also presented for the assessment of the potential effect of controlled hydrogen dissolution in secondary sodium in order to propose efficient solution for enhanced tritium trapping in sodium purification systems (cold traps).

Lecture 03

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TRITIUM IN CIRCUITS

Tritium, being a scarce and radioactive element, is a valuable fuel for nuclear fusion reactors, requiring maximum traceability and containment. One of the common issues encountered in tritium transport circuits is its permeation through materials. Tritium is a lightweight element with extraordinary diffusion properties, making it highly capable of permeating structural materials. Consequently, the main objective of these circuits is to control the inventory and prevent permeation into the environment.

Several control strategies have been studied to manage tritium leaks in circuits, such as anti-permeation barriers, the impact of isotopic co-permeation, and chemical equilibrium, in both fusion and fission programs. At the same time, techniques like permeators, getters, and cold traps have been employed to control inventory and recover tritium for reuse.

This is where the ability to predict tritium behaviour and the development of models to study transfer phenomena become particularly crucial. In this lecture, scenarios such as breeding blankets of nuclear fusion reactors, the lithium loop of the early neutron source IFMIF-DONES, and the balances and inventories of a fast sodium-cooled reactor will be analysed. Different control techniques applied to their respective tritium transport models will be explained.

KEYWORDS

- Tritium
- Hydrogen
- Transfers
- Inventory
- Releases
- Modelling

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THE FUNDAMENTALS OF TRITIUM HANDLING

Storage, process monitoring, and scavenging form the basis of all tritium handling systems. This presentation will discuss uranium as a storage medium, ionization chambers as the backbone of monitoring tritium movement in process loops, and recovery of trace levels of tritium from inert and air-bearing streams for emission reduction.

Several factors influence the loading rate of getter beds. These include the host medium, the thermal capacity of the containment, and the gas transmission rate from a vessel to the storage device. The interrelationship of these factors will be discussed with the objective to design responsive storage beds that can reproducibly absorb tritium up to their design capacity. Uranium is the medium of choice against which the advantages and disadvantages of using other storage media need to be assessed. Pyrophoricity is often raised as an issue of concern. Responses to air ingress accident scenarios will be shown to be benign events in correctly designed storage beds. [1]

Ionization chambers afford a simple and robust technique for monitoring the movement of tritium in process loops. These devices can be integrated directly into process loops without compromising the leak-tightness of the process systems. To first order, ion pair production in these chambers relies uniquely on the concentration of tritons in the carrier stream. However, detailed interpretation of the measurements needs to consider several mitigating factors: operating pressure, type of gas, impurity content, and surface effects. Additionally, under higher carrier gas flow conditions there may be a need to understand the collection and the precipitator efficiencies of the design. [2]

Emission reduction from facilities relies on the extraction of low concentrations of tritium from effluent streams before they are discharged to the environment. This scavenging depends on the composition of the effluent streams. Non-reactive streams (those that contain negligible quantities of oxygen) are typically passed through getter beds that have ultra-low hydrogen vapor pressures. Hydrogen containing tritium can be released from the getters and collected for tritium recovery if necessary.

[1] G.R. Longhurst, et al., Fusion Technol. 21, 1017 (1992)
[2] N. P. Kherani and W. T. Shmayda, Fusion Technol. 21, 340 (1992)

Air bearing streams rely on catalytic oxidation traditionally known as ‘burn and dry’ schemes. In these cases, tritium containing species are converted to tritiated water which is collected on driers. Once regenerated the condensate can be processed for tritium recovery or can be land disposed. This presentation will discuss both of these approaches.

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ITER TRITIUM PLANT

ITER that is presently in the construction phase is a significant scale-up of previous deuterium-tritium burning plasma experiments. The ITER Tritium Plant needs to deliver large quantities of tritium and deuterium to the fuelling systems, and process continuously the vacuum vessel exhaust gas with high efficiency to ensure emissions are minimized.

Since the first conceptual design of the ITER Tritium Plant more than 20 years ago, the tritium processing systems designs have largely matured and evolved. This results from the deeper and better i) analysis of the processing requirements of each system, ii) understanding on how all the systems working together, iii) consideration of all the technical and safety constrains.

This presentation will summarize current progresses and plans for the tritium processing system.

Lecture 06

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THE TRITIUM FUEL CYCLE IN A FUSION POWER PLANT

The fusion community is currently working on the design of a demonstration fusion power plant. A key system is the fuel cycle which for the first time has to integrate a full scale breeding blanket system so that tritium self-sufficiency can be demonstrated. It implies that, after start-up, the tritium required by the plant in any operational phase is provided internally by the tritium bred in the blankets, and no tritium being supplied from external anymore.

Consequentially, the main driver to design the architecture of the tritium fuel cycle of a fusion power plant is the systematic reduction of tritium inventory in all parts of the system. This requires the continual recirculation of gases in loops without storage, avoiding hold-ups of tritium in each process stage by giving preference to continuous over batch technologies, and immediate use of tritium released from tritium breeding blankets. In order to achieve this goal, a number of novel concepts and technologies had to be found and their feasibility to be shown.

The lecture uses the European DEMO as example for a fusion power plant [1], although the main conclusions do qualitatively also hold for other magnetic confinement configurations. It starts from a functional analysis of the fuel cycle and introduces the results of a technology survey and ranking exercise which provided the prime technology candidates for all system blocks. The main boundary conditions for the systems are described based on which a novel three-loop fuel cycle architecture was developed and the required operational windows of all subsystems were defined. Dimensionless figures of merit are introduced with which the operational point of the fuel cycle can be described.

To achieve goals, a number of low-readiness technologies have to be utilized which require substantial maturation efforts. Metal foil pumps are introduced in the divertor pumping to separate a pure DT stream which is then immediately recycled to feed the pellet injection systems. Continuous re-injection of the exhaust gas can artificially increase the wall recycling coefficient and hence allows an increase in the burn-up fraction, which results in a reduction of gas throughputs needed to maintain stable plasma operation at acceptable fuel dilution.

[1] Chr. Day et al., The pre-concept design of the DEMO tritium, matter injection and vacuum systems, Fusion Engineering and Design 179 (2022) 113139.

To increase the core fuelling efficiency, optimization potentials in matter injection technology are being exploited. Dynamic tritium confinement approaches will be used wherever possible to limit discharges. The tritium accountancy system will have to rely on modern, real-time and online tritium instrumentation, not currently available. Consequentially, to validate the derived architecture, various R&D lines were established, selected results of which are reported, together with the key technology developments.

The lecture informs on the current status in the field. It is shown how the DEMO requirements are satisfied and affect system level performance. Examples are given for integration issues and how they were solved.

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TRITIUM PROCESSING IN BREEDING BLANKETS AND TEST BLANKET SYSTEMS IN DEMO

Light nuclei fuse to create heavier nuclei during fusion. This process releases energy that not only powers the stars but can also give humanity baseload electricity that is safe, sustainable, and devoid of greenhouse gas emissions.

Enabling technology for magnetic fusion must be at the cutting edge of an impressive range of fields, from remote handling to cryogenics, balance of plant, and high-power microwave sources, to name a few. The Fuel Cycle and the Breeding Blanket (BB) are the nuclear core of any fusion reactor relying on the DT reaction. Achieving high confidence in obtaining tritium self-sufficiency in DEMO and qualifying all the necessary auxiliary technologies, including the complete fuel cycle, is a key part of the Roadmap to Fusion Electricity in Europe [1]. Currently, for DEMO, two BBs have been selected as potential candidates for the integration in the reactor. They are the Water-Cooled Lithium Lead (WCLL) and the Helium Cooled Pebble Bed (HCPB) BB concepts. The two BB variants together with the associated ancillary systems drive the design of the overall plant. The Tritium Extraction and Recovery (TER) system, one of the BB auxiliary systems, is crucial in enabling the fulfilment of the tritium-self-sufficiency requirement [2].

The WCLL BB adopts a liquid metal, i.e., the eutectic alloy PbLi (15.7 at. % Li) as tritium breeder which serves also as tritium carrier up to the tritium extraction unit. Two technologies are currently considered as the most promising for tritium extraction from PbLi, the gas-liquid contactor and permeator against vacuum (PAV) technologies. In the GLC, a flow of helium (or helium plus a small percentage of hydrogen) is put into direct contact with PbLi in a counter-current and the tritium is removed by the stripping gas. Instead, in the PAV system a membrane separates the PbLi from the vacuum because of the concentration gradient [3].

[1] A. J. H. Donné et al., European Research Roadmap to the Realisation of Fusion Energy, EUROfusion, 2018
[2] G. A. Spagnuolo et al., FED 173, 112933, 2021, DOI: 10.1016/j.fusengdes.2021.112933
[3] M. Utili et al., Energies 16 (13), 5231, 2023, DOI: 10.3390/en16135231
[4] M. Draghia et al., FED 193, 113784, 2023, DOI: 10.1016/j.fusengdes.2023.113784

As the HCPB concerns, the TER system consists of a helium loop with the main purpose to purge the HCPB BBs pebble bed (Li4SiO4 + 35 mol% Li2TiO3) removing the tritium from where it originates allowing then its recovery. The tritium recovery from Q2 and Q2O forms from the purge gas is realized by adsorption of the Q2O form in the Reactive Molecular Sieve Bed (RMSB) followed by Q2 adsorption in the Cryogenic Molecular Sieve Bed (CMSB)/getter beds (Q stands for any H/D/T). During the regeneration of the RMSB and CMSB/getter beds the tritium will be recovered in the Q2 form and finally is processed in the Tritium Plant [4].

In this work, a comprehensive overview of the BB systems and the relevant tritium processing requirements as well as possible technological solution is given.

Lecture 08

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TRITIUM MANAGEMENT IN WATER-COOLED LITHIUM-LEAD BREEDING BLANKET FOR THE EUROPEAN DEMO REACTOR

One of the most ambitious goals of fusion energy is to ensure fuel self-sufficiency of future D-T fusion power plants. Tritium consumption for a 2000 MWth fusion power reactor is 112 kg per full power year [1]. It is clear that efficient characterization of the processes and engineering solutions to manage and control tritium transfer and release is a critical factor in the success of fusion electricity deployment. As far as the supply of tritium to D-T fusion power plants is concerned, one solution is to produce tritium in the blanket surrounding the area of the D-T fusion reactions.

The eutectic alloy PbLi is the breeder of one of the two Breeder Blankets (BB) concepts candidates for the EUROPEAN DEMO, the WCLL (Water Cooled Lithium Lead) [2]. The analysis of tritium generated into BB, the definition of strategy to reduce tritium permeation into PHTS (Primary Heat Transfer System) of the reactor and the design and characterization of the system devoted to extract the tritium generated into the breeder, the Tritium Extraction and Removal (TER) System [3]. Three technologies are analysed for the use in DEMO as TER: Gas Liquid Contactor (GLC), Permeation Against Vacuum (PAV) and Liquid Vacuum Contactor (LVC) [4].

The management of tritium into WCLL BB of DEMO reactor and the design of TER system and integration in tritium building are reported, together with their secondary systems.

[1]. NI, M., et. al., Tritium supply assessment for ITER and demonstration power plant, Fusion Engineering and Design 88 (2013) 2422-2426.
[2] F. Cismondi, G.A. Spagnuolo, L.V. Boccaccini, et. al., Progress of the conceptual design of the European DEMO breeding blanket, tritium extraction and coolant purification systems, Fusion Engineering and Design (157), 2020, 111640.
[3] M. Utili, S. Bassini, L. Boccaccini, et. al, Status of Pb-16Li technologies for European DEMO fusion reactor, Fusion Engineering and Design (146) Part B, 2019, 2676-2681.
[4] D. Demange, R.Antunes, O.Borisevich, L.Frances, D.Rapisarda, A.Santucci, M.Utili, Tritium extraction technologies and DEMO requirements, Fusion Engineering and Design (109–111) Part A, 2016, Pages 912-916.

Lecture 09

Author: F. Di Fonzo
X-nano, Italy

MULTIFUNCTIONAL PERMEATION BARRIER LAYERS AS KEY ENABLING TECHNOLOGY FOR GENERATION IV AND FUSION NUCLEAR POWER PLANTS

DEMO will be fueled by a tritium-deuterium mix, thus the in-situ production of tritium to sustain the fusion reactions assumes a critical importance. To have a proper tritium balance and guarantee radiological safety, it will be mandatory to reduce to negligible the quantity of tritium leaving the breeding zone by permeation through the steel structures. We report on the latest results about amorphous aluminum oxide (Al_2O_3) barriers deposited by Pulsed Laser Deposition (PLD), which is currently considered as one of the most promising solutions to achieve this target. Optimized PLD coatings underwent several fusion relevant experiments. First, deuterium permeation tests showed an unprecedented Permeation Reduction Factor (PRF) > 10^5 at 450 °C, at the limit of the instrumental sensitivity.

In addition, the PRF values are only slightly affected by both high energy electron irradiation (1,8 MeV) and thermal cycling (for several days). Moreover, coated samples underwent corrosion tests involving both Pb 16Li (up to 7000 h, 550 °C, static and flowing conditions) and lithium based ceramic pebbles (550 °C and 800 °C, 730 h), which are the two breeding materials considered for DEMO. Despite the excellent adhesion and the lack of corrosion signs, post-test analyses accomplished by X-ray Diffraction (XRD), Atom Probe Tomography (APT), X-ray Photoelectron Spectroscopy (XPS), and Focused-Ion-Beam Time-Of-Flight Secondary Ion Mass Spectrometry (FIB-TOF-SIMS), highlighted the interaction between the ceramic coating and lithium.

Thermodynamic considerations and basic characterization of the coatings after exposure, allowed to have new insights about this process. In addition, coated samples were tested in DEMO relevant conditions, i.e. tritium permeation tests in a nuclear fission reactor, exposing also the samples to flowing Pb 16Li. Prospects of PLD will be critically discussed and its impact on DEMO BB analyzed.

Lecture 10

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STATE-OF-THE-ART REVIEW OF DETRITIATION TREATMENTS APPLICABLE TO METALLIC TRITIATED WASTE

During its operating and dismantling phases, a fusion reactor using deuterium and tritium as fuel will generate different types of tritiated metallic waste for which suitable and appropriate outlets must be provided. Depending on the relevant national waste management regulation and waste acceptance criteria of final repositories, the management of this tritiated waste combines a cooling phase, waste cutting, sorting, treatments, conditioning steps and storage (with potential interim storage for tritium decay before transfer to a final repository) or possible reuse (circular economy) for waste with an activity below a clearance level.

The present lecture gives an overview of existing detritiation processes that could possibly be used on solid metallic tritiated waste, and investigates their technical relevance and readiness as a preliminary treatment prior to the final storage of the waste.

Their potential benefits lie in a reduction of waste management constraints (e.g. radioprotection in the facilities managing the tritiated waste, interim storage duration and surface area, transportation constraints...) together with a lower environmental impact of tritium release, an opportunity of material recycling (circular economy), and a possible reuse of extracted tritium. Nevertheless, though solutions exist for all types of metallic waste, the implementation and operation of these processes can be costly, especially for irradiating waste.

Furthermore, even though there exists a number of efficient detritiation processes, most of them were only developed at laboratory scale and still require an increase of Technology Readiness Level (TRL) to be confident for industrial-scale use. The correct balance between the benefits and drawbacks of using a detritiation process is thus to be assessed on a case-by-case basis.

Lecture 11

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HARD WASTE DETRITIATION SYSTEMS FOR 3RD TRITIUM SCHOOL

Tritium presents specific challenges and opportunities when present in solid radioactive waste due to it's mobility and the complexity of characterisation. This impacts the techniques applied for quantification, storage, treatment, packaging and disposal.

Within the United Kingdom, a disposal solution for hard Intermediate Level Waste (ILW) in the Geological Disposal Facility is decades away. Regulation of site operators, by the Environment Agency, requires the application of 'best available techniques' to minimize the activity of radioactive waste produced on the premises and the duration of waste accumulation.

A process of removing tritium from waste (detritiation) has been developed and exploited for reducing the radiological inventory of hard operational fusion waste. The process has been successful in down categorisation of ILW and enabled UKAEA to dispose of fusion waste as Low level Waste (LLW) in existing disposal sites instead. However, the existing detritiation system has limitations and is not suitable for all types of hard waste expected in future fusion machines, such as dusts and swarf, hard plastic, and other materials and forms. Additionally, the environment, operating conditions and amount of fuel in future fusion experiments will be different from current machines. Therefore, the requirements for future detritiation systems need consideration.

Lecture 12

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TRITIATED RADWASTE MANAGEMENT IN FRANCE: STATUS AND PERSPECTIVES

Andra is the French National Agency for Radioactive Waste Management. A regulatory framework defines the status, missions and activities of the Agency. After a short reminder of Andra' scope of activities, the lecture will focus on the radioactive waste management solutions already set up and in operation in France and on the on-going project development in Andra.

Tritium issues and limitations related to tritiated waste conditioning and disposal in CIRES and CSA facilities will be described. The case of possible tritium limitations for CIGEO tritiated waste acceptance will also be presented.

The current French strategy for solid tritiated waste management will be reminded as planned in France through the application of the French National Radioactive Materials and Waste Management Plan (the so called PNGMDR). Established by the government, this plan describes the management strategy applicable to all radioactive materials and waste over a period of 5 years by identifying needs and setting objectives to improve this management.

In the 09/12/2022 prescriptive order officializing the latest version of the plan (PNGMDR N°5 for the 2022-2026 period), specific demands of analysis for waste difficult to manage or particular waste as tritiated waste, were required: first to identify management options for all tritiated waste in France, and secondly to construct and compare pertinent management scenarios by performing a multi-criteria analysis (MCA). At the light of the MCA results, the objective will be to confirm, update and/or consolidate the current strategy for solid tritiated waste management.

From these strategic tasks, the need to further study and develop the proposed optimizations (to reduce interim storage duration, to minimize out-gassing rates, to facilitate tritiated waste acceptance to repositories, ...) will be highlighted and then put in perspective for (possible) future fusion reactors development.

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LEARNINGS FROM DISMANTLING AN OBSOLETE TRITIUM INSTALLATION AND DECOMMISSIONING TRITIUM LABORATORIES

In recent decades, SCK·CEN embarked on two significant tritium dismantling and decommissioning initiatives. Between 2003 and 2009, the decommissioning of two rooms originally designated as tritium laboratories at SCK·CEN took place[1]. These laboratories were originally established in 1975 to handle a tritium inventory of 37 TBq, with a primary emphasis on managing tritium as HTO. For the decommissioning of the first laboratory room, the strategy was geared towards maximizing material free release. Conversely, a more pragmatic approach was adopted for decommissioning the second laboratory room, prioritizing speed despite generating additional waste. Notably, more than a decade after these decommissioning efforts, the tritium release limits underwent alterations.

In 2017, a dismantling study commenced for the Variable Neutron Shield (VNS), an installation situated within the BR2 research reactor's reactor building. It's important to note that the VNS installation was unrelated to the tritium laboratory. At the outset of this dismantling study, the VNS had been non-operational for over two decades. This installation had been responsible for substantial tritium production through the neutron irradiation of pure He-3 gas. Tritium generated in this process was subsequently captured using titanium retention traps (Ti-getters). Each Ti-getter had a maximum retention capacity of 370 TBq, but the actual tritium content in the five existing Ti-getters remained unknown. Additionally, there was a tritium-containing experimental NaK getter stored separately. NaK is a liquid metal known for its reactivity with oxygen, water, or NaK oxides, and the tritium content of this NaK getter remained undisclosed. To proceed with the dismantling exercise and find appropriate disposal or recycling methods for these getters, it was imperative to characterize them properly. However, this characterization necessitated the dismantling of the getters from the VNS installation.

[1] Dylst, Kris ; Slachmuylders, Frederik ; Gilissen, Bart et al. / Comparison of different strategies for decommissioning a tritium laboratory. In: fusion engineering and design. 2013, Volume 88, Issues 9–10, 2013, Pages 2655-2658
[2] Vankrunkelsven, Arno ; Dylst, Kris ; D'Joos, Yves. / TUG tritium decommisioning experience. 2023 - 21st meeting of the UK Tritium Users Group, Bristol, United Kingdom.22 p.

In a subsequent phase, aside from the getters, several components of the VNS installation underwent disassembly. Firstly, a robust stainless steel bellow was disassembled into manageable sections employing diverse cutting techniques. Simultaneously, highly contaminated items were extracted from the VNS installation and characterized to prepare them for detritiation in an external facility.

Furthermore, an important task involved initiating the detritiation process for the NaK getter, ensuring the safe removal of tritium from this particular component in order to prepare it for chemical neutralization.

Sharing the insights gained from these projects can offer valuable perspectives on tritium installation decommissioning, encompassing aspects such as waste management, cost considerations, labor implications, the impact of chosen strategies, applicable release limits, and the distinctions between decommissioning installations with or without prior operational knowledge.

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TRITIUM RETENTION IN FUSION DEVICES: FROM JET TO ITER AND DEMO

Tritium management in a fusion reactor is one of the key ingredients in a successful, sustainable and save operation. There is an interplay between tritium breeding, tritium fuelling, tritium burn-up and tritium retention and a surplus of tritium is required to ensure operation of a reactor. However, most of the experience gained is based on deuterium plasmas in present-day devices and only a small amount of DT plasmas with full cycle analysis exists. Nevertheless, it is possible to derive from deuterium experiments e.g. in JET and laboratory experiments mimic e.g. neutron damage, information about what can be expected in reactor conditions.

Here, we focus on the assessment of fuel retention in the reactor vessel, thus predominantly related to plasma-facing components. Results from JET will be presented and analysed with respect to the main mechanisms: implantation and co-deposition. Extrapolations to the foreseen ITER operation shown and the changes between short-term and long-term retention discussed. Finally, the changes in mechanisms responsible for fuel, or better tritium, retention in a fusion reactor introduced.

This contribution is providing furthermore an insight how the fuel retention is quantified in present day devices and how the quantification can be done in future reactors at high neutron dose rate. The challenges and potential solutions will be introduced, and the required research path indicated.

Lecture 15

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TRITIUM IN PLASMA FACING COMPONENTS IN FUSION DEVICES: RETENTION AND TRANSPORT

The uptake of T into the first wall of fusion devices has important consequences for their operation: The so-called recycling of hydrogen isotopes (HIs) at the surface, determined by the balance of impinging vs re-emitted HIs, strongly affects the particle, momentum and energy balance in the scrape off layer region of the plasma which is in contact with the wall[1]. The retention of HIs that diffuse deep into the bulk of the first wall material where they are immobilized by trapping at lattice imperfections results in further challenges: While this can lead to a large inventory of T which poses a safety concern in machines like ITER, for future fusion reactors, this loss of T into the wall by trapping is a concern w.r.t T-self-sufficiency which is a key requirement for the operation of fusion reactor[2]. Future fusion reactors must breed enough T to sustain their fuel requirements and also produce excess T for starting up other reactors.

The trap sites that the HIs bind to are both intrinsic in nature and are also generated by the nuclear environment. Intense n-irradiation results in displacement damage and the transmutation products (e.g. He) also form trap sites for HIs. To describe the transport of HIs through the first wall not only the trapping processes but also the transport across multi-material interfaces between different wall materials must be included.

Since the conditions at the first wall of a fusion reactor cannot be replicated in laboratory experiments, numerical models are required that describe the processes involved in the transport of T from the plasma into bulk and finally out into the coolant. Currently the codes that can span the necessary time scales are so called diffusion trapping codes[3]. They require a large number of input parameters which have to be validated against experiments.

[1] R. Schneider, et al., Contrib. Plasma Phys. 40 (3-4), 2000 p. 328
[2] G.R.Tynan et al, Nucl. Mat. and Energy Vol. 12, 2017 p. 164
[3] A. H. M. Krom , A. Bakker, Metall. Mater. Trans. B 31B, 2000 p. 1475

The lecture will review experimental results on the formation of traps in current candidate materials for the first wall of fusion devices: Tungsten & EUROFER and how these experimental results are used to validate the simulations codes. Then, as a numerical exercise, a simple estimate on the T-self-sufficiency requirements for a DEMO like machine will be presented. This will show how the available experimental data and current state of the art diffusion trapping codes are used to make predictions for future fusion devices.

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TRITIATED DUST: THEIR IMPACT ON TOKAMAK OPERATION

During a fusion reactor operation and due to plasma wall interactions, dust will be created. Different types of dust are produced from almost spherical micrometer particles induced by high heat flux interaction with metal to fractal nano dust. The dust properties especially their ability to be covered by an oxide insulating layer and their surface topology deeply affect their tritium inventory. Due to tritium beta decay, dust are rapidly positively charged. Dust physico-chemical properties and radioactive electrical self-charging process have numerous consequences in term of operation and safety that will be highlight.

In this contribution, we will first list how dust are created in a fusion reactor. We will show that inevitably fractal nano-dust will be produced. The results presented in this first part, obtained mainly in dusty plasma laboratory, will be compared with the current observations obtained in fusion reactors.

Then the dust properties (composition, size and morphology) will be presented. The dust tritium inventory will be then addressed. The electrostatic self-charging state of the dust due to tritium decay will be then calculated taking into account the geometrical characteristics of the different particles.
In the following of this presentation, we will finally address more specifically different open issues related to these tritiated dusts as: 1- The impact of the electrostatic self-charging on the adhesion and on the mobilization of tritiated particles during normal operation or accident as a lost of vacuum accident. 2- The metrology to put in place for the monitoring of occupational exposure of workers 3- The waste management of metallic tritiated dust as well as some proposed technical solutions.

Lecture 17

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CORROSION OF MATERIALS IN FUSION AND FISSION REACTORS

Corrosion of materials within fusion and fission reactors represents a critical challenge in the pursuit of clean and sustainable energy sources. This contribution provides an overview of the multifaceted issue of corrosion in nuclear reactors, highlighting the distinct challenges faced by both fusion and fission technologies. In fission reactors, the corrosion of structural materials and fuel cladding can lead to safety risks and reduce the efficiency and lifespan of the reactor. Research efforts have focused on developing corrosion-resistant materials and innovative cooling systems to mitigate these issues.

In contrast, fusion reactors face unique challenges related to the extreme conditions within the breeding blanket zone. The high-energy neutrons produced in fusion reactions can induce radiation damage and transmutation of materials, leading to embrittlement and corrosion. The breeding zone, where tritium is generated from lithium, plays a pivotal role in ensuring the continuous fuel supply for the fusion reaction. However, it is subjected to a harsh environment characterized by high temperatures, intense neutron flux, and energetic radiation. These conditions can lead to several corrosion-related problems, including the erosion of structural materials, degradation of coolant systems, and the potential release of tritium, a radioactive isotope.

Efforts to address corrosion in the breeding zone have given rise to a range of innovative solutions. These include the development of specialized materials such as advanced ceramics, tritium-compatible coatings, and novel coolants capable of withstanding extreme conditions. Additionally, careful reactor design and maintenance strategies are essential to mitigate corrosion and ensure the sustained performance of fusion reactors.
This abstract emphasizes the importance of ongoing research to understand and mitigate corrosion in both fusion and fission reactors, as the success of these technologies is critical for meeting future energy needs while minimizing environmental impact and ensuring safety.

[1] R. Schneider, et al., Contrib. Plasma Phys. 40 (3-4), 2000 p. 328
[2] G.R.Tynan et al, Nucl. Mat. and Energy Vol. 12, 2017 p. 164
[3] A. H. M. Krom , A. Bakker, Metall. Mater. Trans. B 31B, 2000 p. 1475

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ACTIVE GAS HANDLING SYSTEM SUPPORTING JET FOR THE DTE2 AND DTE3 CAMPAIGNS

The Active Gas Handling System (AGHS) was originally constructed and commissioned in the 1990's to facilitate the tritium fuel cycle of the Joint European Torus (JET). During the Deuterium-Tritium Experiment (DTE) campaign, a stock of 20g of tritium was repeatedly recycled to feed a total of ~100g to JET[1].

Following the successful delivery of DTE in 1997 and the Trace Tritium Experiment (TTE) in 2003, JET switched to H & D plasmas. AGHS continued limited operations to provide exhaust detritation to JET, but the cryogenic and reprocessing subsystems were mothballed.

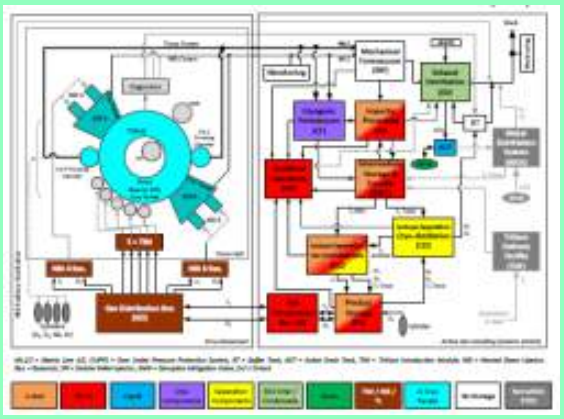
Upon the conception of the DTE2 campaign, AGHS underwent over 5 years of substantial recommissioning activities to bring the plant back to a fully operational state. A series of upgrades were also undertaken to allow AGHS to accommodate the far more ambitious experimental program and to reprocess water in-house. The tritium stock was increased to ~70g.

DTE2 (2020-2021) and the subsequent clean-up campaign saw AGHS operate 24/7 for over 14 months, supplying a total of ~1 kg from the recycled stock. AGHS then provided fueling to JET for the much shorter 8-week DTE3 campaign (autumn 2023), which is the last tritium operation of JET before the torus begins repurposing and decommissioning activities in 2024.

AGHS will continue to provide exhaust detritiation for the foreseeable future while JET is decommissioned. Some AGHS subsystems will be permanently decommissioned, with new glovebox laboratory systems integrated into the facility to enable future R&D activities.

[1] R. Lässer et al., Fusion Eng. Des 47 (1999) 173

This tutorial lecture will give an overview of the AGHS-JET fuel cycle and the plant sub-systems. This will be followed by some of our experience gained from DTE2/3 upgrades and operations, and conclude with some operational accountancy data illustrating a batch-mode tritium fuel-cycle in practice.



Schematic Flow Diagram of Torus Systems and Active Gas Handling Systems (DTE2/3).

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TRITIUM ANALYSIS OF JET PLASMA FACING MATERIALS

Much has been learnt from operating fusion devices with deuterium plasmas, however fusion energy will require operation of devices and power plants in deuterium-tritium fuel mixture. Therefore, the transition from analysis of deuterium and low levels of tritium in plasma facing materials retrieved from fusion machines for study to higher level tritium analysis is necessary in terms of developing both the analysis techniques and laboratory capability for handling radioactive samples containing tritium in combination with activated materials.

Within the fusion community tritium analysis of plasma facing materials (PFCs) removed from tokamak devices has developed through tritium arising from deuterium-deuterium rection and in the case of JET, tritium from the first deuterium-tritium experiments (DTE1) in 1997. The possibility for radiochemical analysis allows for different techniques to be used for tritium detection and quantification alongside the many techniques already deployed for analysis of fuel in plasma facing components. For example, in the case of quantifying tritium inventory desorbed from a heated sample “thermal desorption”, the choice of detection method may be by mass spectrometry often deployed from deuterium detection, or alternatively by a more sensitive radiochemical analysis such as liquid scintillation counting method.

Having demonstrated these ex-situ tritium analysis techniques on JET samples over several decades, the challenge for the coming years will be to realise these measurements for higher tritiated samples coming from the JET DTE2 and DTE3 experiments held in 2020 and 2023 respectively, as was the case in DTE1. Planning for sample removal in 2024 and future PFC analysis relies on estimations for tritium retention coming from knowledge of deuterium retention from earlier ex-situ measurements following deuterium plasma operations. Based on this analysis which assumes 0.19% of injected fuel is retained in PFCs, the tritium retention on individual tile components from JET will be 0.5 – 500 GBq (1.4-1400 µg T or 3x1017 – 3x1020 T atoms) depending on location within the vessel. These PFCs will need to be downsized to smaller samples of the order 1cm3 for analysis to bring T inventory into the range ~1 – 0.1 GBq/sample.

[1] See the author list of “Overview of T and D-T results in JET with ITER-like wall” by CF Maggi et al. to be published in Nuclear Fusion Special Issue: Overview and Summary Papers from the 29th Fusion Energy Conference (London, UK, 16-21 October 2023).
[2] See the list of contributors to JET tile and dust analysis at the International Fusion Energy Research Centre website at https://www.iferc.org/index.php/achievements-rd/.

In addition to ex-situ analysis the first in-situ measurements of tritium in a fusion device have been carried out during the JET DTE3 operations and tritium clean-up in 2023 making use of the final diagnostic installation on JET using laser induced desorption and detection by quadrupole mass spectrometers (LID-QMS).

The tritium techniques used in the analysis of JET plasma facing materials will be presented. The application of knowledge from fuel retention measurements of plasma facing components to provide tritium retention and inventory assessments following the JET DT operations and tritium clean-up will be discussed along with the future requirements for analysis techniques and laboratory capabilities needed to enable exploitation of scientific data and preparation for decommissioning.

Lecture 20

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NON-DESTRUCTIVE TRITIUM MEASUREMENT IN SOLIDS

Tritium emits low energy β-rays (≤18.6 keV), and the escape depths of those β-rays are a few micrometers in low atomic number materials and a few hundred nanometers in high atomic number materials. Hence, a combination of different techniques is necessary for the evaluation of tritium content in a solid.

An imaging plate (IP) sensitive to the low energy β-rays has been extensively used for quantitative imaging of tritium distributions on the plasma-facing tiles used in fusion devices [1] [2]. This type of IP has a bare phosphor layer and its detection limit is as low as several hundred Bq/cm2 (~1011 atoms/cm2) without shielding for background radiation. Because of the short escape depths of β-rays, this type of IP is suitable for detection of tritium in thin layers on surfaces such as deposition layers, oxide files, etc.

Tritium retention in deeper regions can be evaluated by detecting x-rays generated by material-β-rays interactions because of larger escape depths of x-rays. An IP with a thin coating layer of polyethylene terephthalate selectively detects x-rays and provides an image of tritium distribution in a bulk without interference of tritium in an oxide layer [3]. The measurement of energy spectrum of x-rays using a semiconductor detector and analysis by considering generation and attenuation of x-rays in a specimen allow non-destructive depth profiling of tritium [4]. This technique has been called β-ray induced x-ray spectrometry (BIXS) and applied not only for measurements of tritium retention but also for evaluation of diffusion coefficients of tritium in different materials at relatively low temperatures [4].

A total tritium content throughout the bulk of specimen can be evaluated with calorimetry. In this technique, an increase in specimen temperature due to decay heat of tritium is measured. Because the amount of heat generated by tritium decay is very small (0.9 μW at 1 GBq), a high sensitivity heat sensor is required [5] [6].

[1] K. Miyasaka et al., J. Nucl. Mater. 290–293 (2001) 448–453.
[2] S. E. Lee et al., Nucl. Mater. Energy 26 (2021) 100930.
[3] Y. Hatano et al., Fusion Sci. Technol. 60 (2011) 361–364.
[4] M. Matsuyama et al., J. Nucl. Mater. 307–311 (2002) 729–734.
[5] M. Matsuyama and M. Hara, Fusion Sci. Technol. 54 (2008) 182–185.
[6] M. Matsuyama et al., Fusion Eng. Des. 85 (2010) 2045–2048.

Lecture 21

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LIBS AS AN *IN-SITU* DIAGNOSTIC IN FUSION DEVICES

Laser-Induced Breakdown Spectroscopy (LIBS) is a pulsed laser technique dedicated to the determination of the composition of a material. This technique is based on the solid to thermal plasma conversion of the sample. Illuminated with a sufficiently high irradiance, the material heats, liquefies, then emits vapors which become plasma by inverse Bremsstrahlung during the pulse. The temperature levels reached are of the order of 10,000 to 15,000 K. The conversion being very rapid, the plasma produces a significant overpressure leading to the production of a shock wave propagating in the ambient gas. During this propagation, the plasma cools while emitting radiation first made up of a continuum, then of ionic and atomic spectral lines. The spectroscopic study of this radiation makes it possible to deduce the multi-elemental composition of the plasma, and therefore of the material. This can result from comparisons with experimental databases that have been previously identified and implemented when the user has already indications on the composition. In the case of completely unknown compositions, the experimental spectrum must be recalculated under conditions of local thermodynamic equilibrium. As a result, the plasma experimentally obtained must be as close as possible to the local thermodynamic equilibrium and uniform. Its electron density has to be sufficiently high and a noble background gas should be preferred.

To be correctly implemented, this method requires to be well understood, particularly from the point of view of the underlying physics. This physics is very rich because it combines phase change, laser-matter interaction, unsteady diffusion of thermal energy in multiphase media, plasma physics, thermodynamics of critical states and out of thermodynamic equilibrium. We will illustrate what the numerical simulation can predict. We will be particularly interested in the question of the modification of the concentration profiles of tritium induced by the implementation of LIBS in the case where this atom is present in various materials (silicon wafer then steel) using the code ATLAS (Analysis of Tritium diffusion under LASer pulse).

[1] A. Favre et al, "First LIBS measurement of tritium in a solid sample", submitted to J. Nucl. Mater.

These descriptions will illustrate the use of this technique in laboratory in the case of silicon. This was in fact recently implemented at CEA Saclay with very interesting results[1]. These results will be illustrated and discussed in the light of the results of the ATLAS numerical simulation code.

The technique was also implemented at CEA Cadarache on the WEST reactor in a first version. The objective is to determine the composition of the metal deposits produced by the machine. The results obtained are currently being analyzed. They will be presented. They will lead to the optimization of the implementation through a second version of the device where different limitations put forward by the first version are solved.

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EPIDEMIOLOGICAL STUDIES OF TRITIUM EXPOSURE

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 very 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. 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, presently available epidemiological findings do not provide reliable evidence that the risk to health of tritium exposure has been seriously underestimated[2]. Recently, however, a small cohort study of tritium workers in France has been conducted using tritium-specific doses, which found some indication of a possible effect of tritium exposure for a limited number of cancers, but these results were based on very small numbers of deaths[3]. This French study[3] illustrates what could be done with large numbers of tritium workers in an international collaboration.

[1] M.P. Little and R. Wakeford, J. Radiol. Prot. 6-32 (2008) 28
[2] United Nations Scientific Committee on the Effects of Atomic Radiation, UNSCEAR 2016 Report, Annex C (2017)
[3] S. Martin and C. Ségala, Radiat. Res. 284-292 (2021) 195

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HUMAN EXPOSURE TO TRITIUM: BIOKINETIC MODELS AND BIOLOGICAL EFFECTS

Tritium is of natural or anthropogenic origin. It is produced as tritiated hydrogen, tritiated water or tritiated organic molecules. As a low-energy beta emitter, tritium is generally considered to be of low radiotoxicity. Tritiated water makes up the majority of tritium releases from nuclear facilities. Its biokinetics in humans essentially involve a retention period of 10 days, which decreases when people drink a lot of water. A small fraction is metabolised and integrated, along with hydrogen, into all biomolecules (proteins, lipids, carbohydrates, nucleic acids). This fraction has a longer half-life of several dozen days. Current uncertainties are limited and marginal in terms of implications for protection standards. The International Commission on Commission on Radiological Protection (ICRP) does the updating of the retention and excretion functions of radionuclides periodically. Tritium is no exception to these revisions, which are taken up by the international radiation protection authorities (IAEA, WHO and EURATOM in particular).

In terms of health, recent syntheses (UNSCEAR report) have highlighted several difficulties and/or uncertainties concerning the assessment of the effects of tritium exposure. The need to reassess the current biokinetic model for intake of tritiated biochemical substrates, 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.
The experimental data come mainly from cellular and animal studies after exposure to tritiated water (HTO). In laboratory mice and rats, exposure to tritium can induce both deterministic and stochastic effects (cancer or heritable effects). However, to date, there is no epidemiological evidence of stochastic health effects being induced by tritium exposure in humans.

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DOSIMETRY OF TRITIUM IN HUMANS AND NON-HUMAN BIOTA

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 wR. For tritium and all beta radiation, wR 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 wT. 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 2023.

[1]. ICRP 2016, Occupational Intakes of Radionuclides, Part 2. Ann ICRP 45 (3/4)
[2]. ICRP 2003. Relative Biological Effectiveness, Quality Factor and radiation weighting factor. Ann ICRP 33(4).

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 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 wB 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 wB, might be warranted.

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BIOKINETIC MODELS FOR TRITIUM

Biokinetics is the time-dependent behavior of incorporated substances in living organs: uptake, distribution, retention, and clearance. It partly determines the radiation dose received from intakes of radionuclides. Biokinetic models are therefore required to interpret bioassay measurements for radiation protection. The principles of biokinetic modelling will be introduced with examples from nuclear medicine. The main models developed by the International Commission on Radiological Protection [1] will be presented: human respiratory tract, alimentary tract, examples of element-specific systemic models. Their use in radiation protection will be explained. A focus will be made on models applicable to tritium [2], addressing the issues of different chemical forms [3], binding to organic material and cellular distribution.

[1] ICRP. Occupational Intakes of Radionuclides: Part 1. ICRP Publication 130. (2015) Ann. ICRP 44(2)
[2] ICRP. Occupational Intakes of Radionuclides: Part 2. ICRP Publication 134. (2016) Ann. ICRP 45(3/4), 1–352
[3] Smith R., Ellender M., Guo C., Hammond D., Laycock A., Leonard M.O., Wright M., Davidson M., Malard V., Payet M., Grisolia C., Blanchardon E. Biokinetics and Internal Dosimetry of Tritiated Steel Particles (2022) Toxics, 10 (10), 602. DOI: 10.3390/toxics10100602

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ASSESSING IMPACT OF TRITIUM ON AQUATIC ENVIRONMENT

Compared to other radionuclides, Tritium (^3H) is abundantly released in the environment from the nuclear facilities. This rather paradoxical radionuclide is deemed to be extremely mobile in the environment as well as in the biological systems. Considering the amount of ^3H released, its physical and chemical properties, there is growing scientific and regulatory concerns for its potential impact on humans and non-human biota (NHB). Despite the concern, very limited knowledge is available pertaining to its potential biological effects on NHB as well on humans. The limited available information pertaining to ^3H radiobiological studies exhibit inequality for different taxonomic groups and species. Among the NHB, the studies are heavily inclined towards marine bivalves, fish and rodents. Further limitations relate to the scarcity of field relative to the laboratory studies, and lack of reports using different forms of tritium (e.g., HTO and organically bound tritium: OBT).

In addition, the inherent limitations to determine the absorbed radiation doses in varieties of NHB with different shapes and sizes hinders our understanding of 'dose-response' relationships, in common with all the radionuclides. Within these constraints, tissue-specific bioaccumulation and responses at different levels of biological organisations (viz., molecular to individual or population levels) following exposures to ^3H, have been reported for early and adult life stages. In contrast to human or mammalian studies, the potential transgenerational, bystander effects, germ-line mutations, epigenetic effects, and the application of rapidly developing "omics" (i.e., transcriptomics, proteomics and metabolomics) approaches have been very limited. Appropriate adoption of these approaches could help to fill these knowledge gaps and further elucidate the relationships between molecular level events with higher level effects through the development of radiation specific adverse outcome pathways.

[1]. Smith, J. et al., (2023) The Risks of Radioactive Wastewater Release. Science (1979), 382, 31–33. https://doi.org/10.1126/science.adi5446.
[2]. Jha, A.N. (2021) Nuclear power: how might radioactive waste water affect the environment? The Conversation. https://theconversation.com/nuclear-power-how-might-radioactive-waste-water-affect-the-environment-159483.
[3]. Ferreira, M. F. et al., (2023) Tritium: Its Relevance, Sources and Impacts on Non-Human Biota. Sci Total Environ 2023, 162816. https://doi.org/10.1016/j.scitotenv.2023.162816.

The environmental behaviour of different forms of ^{3}H, including mechanisms and rates of OBT formation also need further elucidation to better define and understand its potential long-term impacts. Exposures to multiple physical (e.g., temperature, hypoxia) and chemical stressors (e.g., nanoplastics, organics) and particles associated with dismantling of nuclear facilities (e.g., steel and cement), to improve environmental risk assessments for the radionuclides should be given important considerations for future studies. These could be combined with emerging modelling approaches including artificial intelligence (AI).

Finally, given the quantities of ^{3}H discharged globally, an important technical goal should be to minimise its production and discharge.

[4]. Dallas, L. J. et al., (2016) Exposure to Tritiated Water at an Elevated Temperature: Genotoxic and Transcriptomic Effects in Marine Mussels (M. Galloprovincialis). J Environ Radioact., 164, 325–336. https://doi.org/10.1016/j.jenvrad.2016.07.034.
[5]. El Zrelli, R. et al., (2023) PET Plastics as a Trojan Horse for Radionuclides. J Hazard Mater, 441. https://doi.org/10.1016/j.jhazmat.2022.129886

Lecture 27

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TRITIUM DOSIMETRY: MODELING APPROACHES AND BIOLOGICAL DATA ANALYSIS AT THE SUBCELLULAR/CELL POPULATION SCALES

The radiotoxicological and radiobiological consequences of tritium contamination strongly depend on the speciation of tritiated products. Tritium commonly occurs as tritiated water (HTO) or organically-bound tritium (OBT), but can also exist in many other forms, depending on nuclear-related construction materials, including tritiated particles of different nature and sizes. Furthermore, different exposure pathways are possible under accidental exposure scenarios (i.e., inhalation, skin absorption and/or ingestion). Overall, the chemical speciation determines, for a given exposure pathway, both the kinetics associated with the distribution of tritiated products at the organ/tissue level (including clearance) and the associated distribution at the cellular and subcellular levels, which can be highly inhomogeneous.

Considering the average range of beta (β) electrons emitted by tritium decays in biological tissue (0.5 µm, corresponding to an average decay energy of 5.7 keV), it is clear that both the distribution of energy deposition events at the subcellular level and the cumulative energy ("dose") deposited in target cells will vary substantially when different tritiated products with different distributions are considered. Both these two pieces of information are necessary to correctly assess tritium dosimetry and correlate dose to tritium biological effects.

In this lecture, we will review the main concepts at the basis of sub-cellular dosimetry for tritiated products, and more generally, for short-range emitters and their related biological effects. We will focus on the possible modeling approaches and simulation tools that are available to achieve a thorough assessment of energy deposition following internal contamination with radioactive particles, and to obtain indications on the associated biological damage.

[1] M. Siragusa et al., Rad. Res., 188(2), 204–220 (2017)
[2] D. Alloni et al., Rad. Res., 182, 322–330 (2014)
[3] A. Mentana et al., Rad. Res 199(1), 25–38 (2023)
[4] J. Chen, Radiat. Prot. Dos. 156(3), 372–5 (2013)
[5] G. Baiocco et al., Rad. Res. 195(3), 265–274 (2021)
[6] A. Mentana et al., Toxicology in Vitro 92, 105656 (2023)

In particular, we will review a.o. track structure calculations [1] [2], radiation transport and micro/nanodosimetry [3] [4], and analytical approaches [1] [3] [5]. We will also discuss how to choose which model is more suited to provide information to be integrated with experimental results, depending on the characteristics of the setup in use (e.g. *in vitro* standard "2D" vs. "3D" cell cultures). Finally, we will introduce an example of a possible approach to analyze biological data from *in vitro* tritium exposure that specifically tackle the inhomogeneity of tritium source distributions, as expected in case of insoluble micrometric particles of tritiated metal.

Lecture 28

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SKIN ABSORPTION OF TRITIUM AFTER CONTAMINATION FROM TRITIATED WATER AND TRITIATED POWDERS

Skin absorption of tritiated water (HTO) was studied on volunteers and in ex-vivo experiments with Franz cells since '50 with the aim to define the potential systemic exposure after skin contamination. Moreover, HTO has been used in many experimental setting to evaluate the integrity of the skin before skin absorption studies on penetration and permeation of chemicals from the skin. For that reason, many data are available on this topic that could help to predict the risk of exposure through the skin after contamination with tritiated powders derived for dismantling atomic power plants.

Previous studies on volunteers demonstrated that HTO diffused into the skin reaching the whole body and contamination was studied analyzing tritium in urine after hours and days from exposure. The maximum urine concentration was measured between 3 and 6 hours after the exposure [1]. The body water was estimated through literature that gave a rapport with body area of W/A=23.6 L/m^2. HTO was absorbed through the skin from the liquid or the vapor phase [1]. ICRP [2] in 1995 calculated that 1% of HTO activity per m^3 in air is absorbed through the skin per minute considering data from Osborne (1966) and Hill and Johnson (1993). In this condition skin contributed for one third of the total HTO intake when a subject is active (higher breathing intake) while at rest the amount of HTO absorbed through the skin is the same than that absorbed through inhalation.

Moreover, from Osborne data, the protective effect of personal protective equipment was analyzed confirming the reduce of HTO penetration using skin protective devices.

Eakins et al in 1975 demonstrated on volunteers that exposure of skin to tritium-gas-contaminated surfaces caused absorption of tritium, as HTO and OBT (organic bound tritium). The effective dose was estimated to range from 8.7 × 10^{-12} to 9.7 × 10^{-12} Sv/Bq absorbed(Johnson and Dunford, 1985).

[1] R.V Osborne, Health Phys. 1527-37 (1966) 12
[2] International Commission on Radiological Protection (ICRP), Part 4 (1995a) 71, Oxford, Pergamon Press.
[3] AAVV Health Effects, Dosimetry and Radiological Protection of Tritium, Minister of Public Works and Government Services Canada 2010
[4] R.L. Hill RL and J.R. Johnson, Health Physics 628-647 (1993) 65
[5] J.D. Eakins, W.P. Hutchinson, and A.E. Lally, Health Physics 213-214 (1975)

In vitro data, using human skin and Franz cells, demonstrated a flux of HTO through the skin of about 0.15 MBq/cm²/h (±0.008 SE) applying 2 mL of a solution of 0.1MBq/mL (Davies et al. 2004) and of about 0.0076 MBq/ cm²/h applying 0.0017 MBq per cells. Lag time (the time needed to have a constant flux through the skin) was between 1 and 2 hours of contact. Considering available data on HTO skin absorption, the exposure of hands (40 cm²) for 8 hours/day for 200 days per year, a concentration of tritiated powder of 10 mg/m³ in air (maximum allowed concentration for powders exposure) with activity of 1MBq/mg we can calculated a possible exposure for workers in which skin exposure was much lower than inhalation route. Different exposure scenarios will be used to estimate received dose through the skin route.

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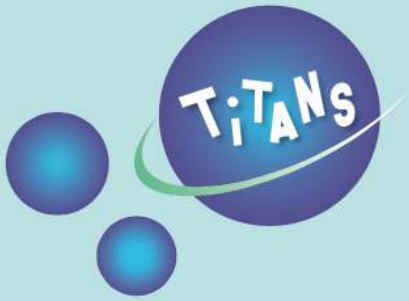

TRITIUM IMPACT AND TRANSFER IN ADVANCED NUCLEAR REACTORS

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