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# Plasma Physics and Technology Aspects of the Deuterium–Tritium Fuel Cycle for Fusion Energy

*Summary of a Technical Meeting*

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# PLASMA PHYSICS AND TECHNOLOGY ASPECTS OF THE DEUTERIUM–TRITIUM FUEL CYCLE FOR FUSION ENERGY

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# PLASMA PHYSICS AND TECHNOLOGY ASPECTS OF THE DEUTERIUM–TRITIUM FUEL CYCLE FOR FUSION ENERGY

SUMMARY OF A TECHNICAL MEETING

INTERNATIONAL ATOMIC ENERGY AGENCY
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# FOREWORD

Fusion power plants are expected to use as fuel a mixture of two heavy isotopes of hydrogen — deuterium (D) and tritium (T) — (among other possible choices) which fuse to produce helium and energetic neutrons. Inside the fusion power plant, additional tritium will be 'bred', that is created from the reaction of the neutrons within a blanket containing some form of lithium covering the inner wall of the plasma chamber. However, several uncertainties remain regarding the physics and technology of the D–T fuel cycle in ITER and future fusion power plants.

In October 2022, the IAEA organized the Technical Meeting on Plasma Physics and Technology Aspects of the Tritium Fuel Cycle for Fusion Energy, attended by 39 participants from 9 Member States and by representatives of the ITER Organization. The topical area of focus was the complex interface of plasma physics and technology aspects of the D–T fuel cycle in magnetic fusion devices, from ITER to demonstration fusion power plants.

This publication provides an overview of the topic, a scope largely neglected but having a potentially large impact on the operations and feasibility of fusion as an energy source. It also presents contributed papers from experts representing a wide range of international magnetic confinement fusion R&D programmes working on the D–T fuel cycle.

In general, the important role of the fuel cycle for future fusion power plant operation has been underestimated in the past. This publication shows that additional R&D is needed in this complex area where the fuel cycle systems interact directly with the plasma. Overambitious plasma requirements can potentially result in the fuel cycle plant's becoming unattractively large. It is paramount that the developers of plasma scenarios and their operation strategies work closely with the developers of technology to ensure that their integration will meet the requirements of both sides. Ideally, this should be addressed in broad and international R&D programmes, that exploit the range of expertise in the fusion community.

The IAEA wishes to express its appreciation to all those who contributed to the drafting and review of this publication, in particular C. Day (Germany) and L.R. Baylor (United States of America). The IAEA officers responsible for this publication were M. Barbarino of the Division of Physical and Chemical Sciences and A. Khaperskaia of the Division of Nuclear Fuel Cycle and Waste Technology.

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# 1. INTRODUCTION

## 1.1. BACKGROUND

It is expected that first generation Fusion Power Plants (FPPs) will use a mixture of two heavy isotopes of hydrogen — deuterium (D) and tritium (T) — as fuel, which then fuse to produce helium and energetic neutrons. Inside the fusion power plant, additional tritium will be 'bred', i.e., created from the reaction of the neutrons within a blanket containing some form of lithium covering the inner wall of the plasma chamber. However, several uncertainties remain regarding the physics and technology of the D–T fuel cycle in ITER and future FPPs.

In October 2022, the Agency organized a Technical Meeting on Plasma Physics and Technology Aspects of the Tritium Fuel Cycle for Fusion Energy. The topical area of focus was the complex interface of plasma physics and technology aspects of the fusion fuel cycle, from ITER to DEMO plants.

## 1.2. OBJECTIVE

The objective of this publication is to provide an overview of plasma physics and technology aspects of the D–T fuel cycle in magnetic fusion devices, from ITER to demonstration FPPs. The TECDOC also provides contributed papers from experts representing a wide range of international magnetic confinement fusion research and development programmes working on the D–T fuel cycle.

## 1.3. SCOPE

The topical area of focus in this publication is the complex interface of plasma physics and technology aspects of the D–T fuel cycle in magnetic fusion devices, a scope largely neglected but having potentially large impacts on the operations and feasibility of fusion as an energy source. On the following page, Table 1 gives a representation of the topical scope of this publication.

## 1.4. STRUCTURE

Following the first section, Section 2 serves as an introduction to the fuel cycle, contextualizing its significance within the framework of a fusion power plant. Sections 3–5 provide summaries of the main topical areas of interface of plasma physics and technology aspects of the D–T fuel cycle in magnetic fusion devices, namely (i) plasma, burn control and fuel cycle technology; (ii) first wall and fuel cycle technology; and (iii) disruption mitigation and fuel cycle technology. Section 6 summarizes the main findings and conclusions developed in the discussion at the technical meeting. To conclude, subsequent to the primary text, a curated compilation of contributed summary papers showcased during the technical meeting is included, systematically arranged according to the corresponding technical session.

TABLE 1. TOPICAL AREA OF FOCUS: THE COMPLEX INTERFACE OF PLASMA PHYSICS AND TECHNOLOGY ASPECTS OF THE D–T FUEL CYCLE IN MAGNETIC FUSION DEVICES

| Tokamak Physics / Plasma Chamber Topic | Fuel Cycle Topic |
|---|---|
| **Plasma physics:** | **Fuel cycle technology:** |
| Burn fraction | Fuelling |
| Fuel delivery requirements | Isotope separation |
| Gas delivery in addition to DT (H, Ne, other) | Loop configuration (direct internal recycle) |
| Plasma resilience to impurity build-up | Vacuum pumping |
| Plasma particle transport | Fuel clean-up (purification) |
| **Plasma burn control:** | **Fuel/exhaust actuator:** |
| Plasma fuelling rate and isotopic mix control | Fuelling and impurities feed control |
| Power dissipation | Isotope separation and/or direct recycle |
| ELM control | Pellet and gas injection |
| **First wall:** | **Fuel cycle technology:** |
| Be, W, Mo, low activation steel | Fuel clean-up (impurity processing) |
| Low-Z first wall coatings | Be and activated product clean-up technology (off normal) |
| Liquid metals | In-vessel tritium recovery (conditioning and remote handling) |
| Dust | Tritium recovery from coolant |
| Redeposited layers | Inventory measurement |
| Tritium retention | |
| **Disruptions:** | **Disruption mitigation:** |
| Fuel and impurities released due to disruptions | Disruption mitigation technology |
| Optimization of disruption mitigation for fuel cycle compatibility | Fuel clean-up (purification and impurity processing) |
| | Isotope separation |
| | Vacuum pumping |

## 2. D–T FUEL CYCLE CONTEXT

As can be derived from the fusion reaction, an FPP would consume about 153 g of tritium per GW of fusion power in a full power day; this means that for each full power year, a 2 GW power plant would require about 112 kg of tritium, or 2.8 tonnes for a projected lifetime of 25 full power years. As it is not feasible to supply an FPP with such amounts from external sources, the fuel cycle of an FPP needs to enable tritium self-sufficiency. This involves: (i) generating in the breeding blanket all the tritium that it will be uses during operation; (ii) relying on a (to be minimized) initial supply of tritium (several kg) to start the plant and sustain plasma operation until the breeding blanket and its associated tritium systems are able to work solely with newly bred tritium; and (iii) to produce during operation a stock of tritium able to support the start-up of follow-on power plants. In general, an FPP will need to meet the requirements for all the important areas of safety in all stages of the lifetime to protect workers, the public and the environment from harmful effects of ionizing radiation and for the safety of facilities and activities that give rise to radiation risks. Application of the concept of defence in depth throughout design and operation and multi-layer confinement concepts will provide protection against transients, anticipated operational occurrences and accidents, including those resulting from equipment failure or human action within the facility, and events induced by external hazards.

Historically, the Tritium Systems Test Assembly (TSTA) at Los Alamos National Laboratory, United States of America was the first facility dedicated to the development and demonstration of technologies required for fusion D–T processing at a relevant scale. That facility design was launched in 1977, for a flowrate of 10 $Pa{\cdot}m^3{\cdot}s^{-1}$. It was commissioned in 1982, the first tritium was processed in 1984 and it was operated until 2001. It had the capability to recirculate tritium, although not at a steady state. The maximum tritium inventory was of 140 grams.

In 1993, the Tokamak Fusion Test Reactor (TFTR) at the Princeton Plasma Physics Laboratory, United States of America became the first world's magnetic fusion device to perform extensive experiments with plasmas composed of 50/50 D/T mixture, with an administrative tritium inventory limit of 5 g. It produced a world record of over 10 MW of fusion power output in 1994, exceeded by JET in 1997 with 16 MW of D–T fusion output power. TFTR stopped operation in 1997 while JET continues to operate with dedicated D–T operating campaigns to this day.

Since 1988, Japan has been operating the Tritium Process Laboratory (TPL) in Tokai, with an approved tritium inventory of 60 g, aiming to establish more complete tritium safety technology for a D–T fusion experimental device. TPL is used for exhaust detritiation R&D for ITER.

In Europe, the Tritium Laboratory Karlsruhe (TLK) at Karlsruhe Institute of Technology, Germany is celebrating 30 years of operation in 2023. It offers a closed loop architecture for recycling and purifying tritium and is licensed for 40 g. At TLK, the basic concepts of the ITER fuel cycle tritium processing technology were developed.

The JET experimental tokamak in Culham, United Kingdom is currently the only fusion device that can be operated with tritium. JET produced a record-breaking 59 MJ of sustained thermal fusion energy in 2021. Since the first demonstration of thermal fusion energy production in 1991, more campaigns have been conducted in 1997, 2004 and recently in 2022. The JET Active Gas Handling System (AGHS) is a tritium loop fuel cycle designed to service the experiments. The maximum inventory on site of tritium is 90 g.

The Hydrogen-3 Advanced Technology Centre (H3AT) is a future facility, currently under construction at Culham, United Kingdom. It is the most recent effort to demonstrate the operation of a closed loop fuel cycle at pilot plant scale in a continuous mode, as well as with that closest to the fuel cycle operation expected to be necessary for an FPP. It will also feature a down-scaled ITER relevant tritium loop, with a tritium inventory of 100 g.

ITER will drive the size of the fuel cycle on a new level, as it is being designed for a nominal throughput of 200 $Pa{\cdot}m^3{\cdot}s^{-1}$ DT and will operate with a total inventory of about 4 kg of tritium. The ITER research plan is currently under revision, but it is expected that the ITER fuel cycle will be in full operation more than 10 years from now. While the ITER fuel cycle architecture is based on a once-through closed loop, see Bonnett p. 37, it is still not self-sufficient, in the sense that it requires tritium being imported from external sources. One important driver of the ITER fuel cycle architecture is the requirement for experimental flexibility to enable a rich physics programme. The ITER fuel cycle architecture is also reflecting the need to process large throughputs of deuterium to feed the neutral beam systems, see Veltri p. 35, which may not be necessary for an FPP.

According to present knowledge, the fuel cycle of an FPP, with a complete blanket system that generates more tritium than being burnt, cannot be a simple scale-up from ITER. Instead, the FPP fuel cycle design needs to consider further tritium inventory reduction measures as a main design driver. This will lead to a multi-loop architecture, all operated in an integrated, fully continuous, and steady state manner, using a bypass to bridge the dwell time in between the plasma discharges if operated in a pulsed manner. This would add additional complexities, such as dynamic control (loops follow very different cycle times), and limited capability to provide D/T in ratios other than unity, see Day p. 29. Due to the dynamic character, tritium management and systems of accounting for and control of tritium within the containing systems will become a new challenge but be key towards achieving a licensable FPP. A generic sketch of an FPP relevant D–T fuel cycle is shown in Figure 1.

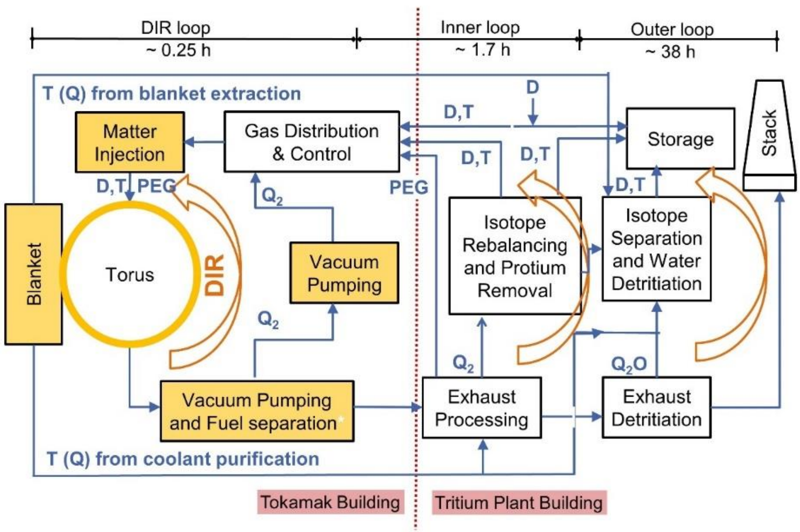


*FIG. 1. Sketch of the FPP-relevant D–T fuel cycle (Q is denoting any of the three hydrogen isotopes H/D/T).*

The FPP fuel cycle comprises four functional units. The breeding blankets produce tritium by absorbing the fusion neutrons and consuming lithium. The vacuum pumping systems remove the exhaust including the helium ash and the unburnt D/T fraction, keeping the burning plasma under vacuum, as well as generating the vacuum condition to start a new plasma discharge in the times between two plasma pulses. The matter injection systems provide the fuel to the plasma in sufficient amounts to keep the target core density constant at the level needed to sustain a burning plasma and deliver plasma enhancement gases to radiate power in order to control the heat flux to plasma facing components. While the latter three systems are in close interaction with the plasma and located in the tokamak building, the actual tritium processing systems are combined in a separate building – the tritium plant. These systems are doing the clean-up of the exhaust, recovering and extracting the unburnt tritium, removal of impurities including the helium, and providing the required 50/50 D–T mix to be delivered to the matter injection systems. Focus of the present document are the highlighted systems on the left side of Figure 1.

As indicated by the three arrows in Fig. 1, the fuel cycle comprises three different loops. The innermost or Direct Internal Recycling (DIR) loop provides a fast short-cut of unburnt fuel without separating to the level of pure isotopes, thus contributing to reduce the tritium inventory, while the second loop mainly provides protium removal. The third loop provides tritium recovery and removal from the remaining gas. Such a fuel cycle in its totality, under continuous operation conditions with D–T, and at the scale necessary for an FPP (with inventories in the multiple kg range) has not yet been demonstrated. The ITER fuel cycle will demonstrate a number of the outer loop technologies and include a complete isotopic separation of all hydrogenic species contained in the exhaust.

Just recently, the private company Kyoto Fusioneering together with Canadian Nuclear Laboratories have announced plans to build a D–T FPP fuel cycle technology demonstration platform by 2025 at the Chalk River site, Canada. It is planned to feature PbLi blanket relevant tritium extraction technology.

As shown in Table 1, this workshop is focussed on the interface area between the FPP fuel cycle and the plasma, with emphasis on the burn phase of plasma operation (ramp-up, flat-top, ramp-down). This challenging area defines the main engineering loads and size driving requirements on the fuel cycle.

Figure 2 summarizes schematically the different systems treated in this document. Even for technologies already demonstrated to work in a D plasma environment in large scale experimental fusion devices, it needs to be understood, how the demonstration extrapolates to an FPP scale DT plasma and, in particular, how the design can be made tritium-compatible and safe.

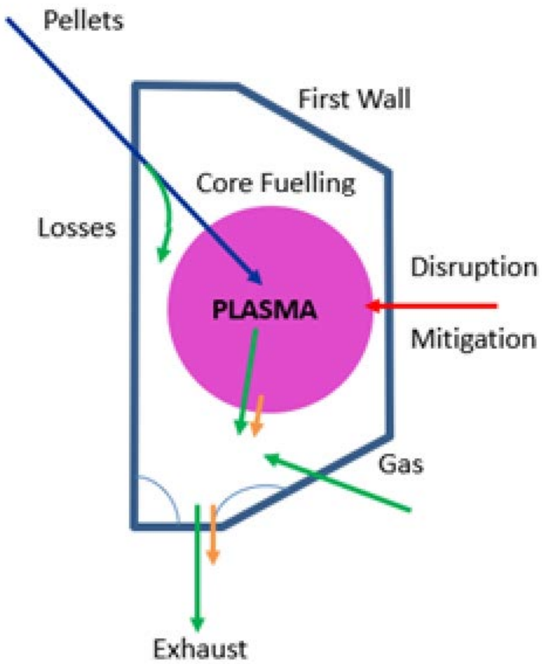


*FIG. 2. Interfaces between plasma and fuel cycle.*

## 3. INTERFACE BETWEEN BURNING PLASMA, ITS CONTROL AND THE FUEL CYCLE

The development of the control system for an FPP faces unprecedented challenges, see Schuster p. 15. The requirements for control reliability and accuracy are more stringent than on existing fusion experimental devices. This is because the large energy content that would result from any loss of plasma control in a tokamak based FPP may result in a disruption, which could damage the inner wall of the FPP. Due to the nonlinear dynamic behavior of the plasma, feedback control of the burn condition will be necessary to avoid undesirable transient performance and respond to changes in plasma confinement, impurity content or operational conditions, which could significantly alter the plasma burn. This would also assist in avoiding potentially disruptive plasma conditions due to thermal instabilities. At the same time, there exist more stringent lifetime issues for plasma facing components due to neutron and gamma radiation, erosion and re-deposition. Thus, the FPP control system needs to rely on a reduced set of very robust diagnostics. This manuscript, as illustrated in Fig. 2, focuses on the fueling and pumping components required as part of the fuel cycle and on their role as actuators in a burning plasma control environment.

Radiative power removal by seeded noble gases in the core and in the divertor will be a necessity for an FPP plasma, see Kallenbach p. 18. The fuel dilution caused by core and divertor radiators, together with other impurity sources such as He ash and eroded material from the plasma facing components, will need to be sufficiently low to maintain the required D–T burn rate. Ideally, this scenario would be integrated with no–ELM conditions, achieved either with the help of magnetic perturbations or pedestal tailoring via the combination of radiation cooling and intrinsic magnetohydrodynamic and turbulence mechanisms. However, it is understood that impurity transport is less problematic in a large burning plasma than in present day tokamaks due to the expected outward convection, see Dux et al. p. 22.

Core fuelling is essential to compensate for the particle losses from the plasma core due to collisional transport out of the confined plasma. This translates in the need for deposition of a given flow of fuel particles into the core, to keep the core density constant (to maintain a certain profile at high density with sufficient safety margin for any operational density limit). The fuel cycle technology providing this function is high velocity (of the order of 1 km/s), high frequency (in the order of 10 Hz) pellet injection of solid DT pellets into the plasma, see Lang et al. p. 31. As the particle source from the pellets is in discrete events, disturbances need to be controlled to levels that avoid plasma instabilities, which means the required repetition rates and pellet sizes will result from a trade-off. To employ the E×B drift effects that will act on the pellet deposited particles once ionized, the injection needs to happen from the high-field side of a tokamak based FPP to achieve efficient fuelling. This is difficult to access, and it results in additional losses from the pellets being transported through guiding tubes. The outlet of the injection tubes needs to be properly integrated in the blankets covering the first wall, with minimal curvature for efficient transport.

There exist two technology variants for pellet acceleration. One is based on the gas gun concept (as ITER will be), the other one uses a centrifuge. The former concept has been demonstrated with DT on TSTA, while the latter one has only been used in D–D experiments. The pellets will be formed by continuous cryogenic extrusion of frozen fuel ribbons.

The resulting plasma density achieved for a given pellet fuelling rate depends significantly on the fuelling efficiency, which is the fraction of pellet particles that are retained in the plasma during the injection process. This process is not properly understood for FPP relevant plasma conditions and, at present D–D experiments, is known to be a function of the pellet penetration. The actual pellet penetration in a burning plasma will be very shallow, given the anticipated high plasma edge pedestal electron temperature. The resulting mass deposition in the plasma from such shallow pellets has not been benchmarked by any codes in present machines, adding necessary additional margins towards higher flowrates.

A very important point, with a significant impact on the dimensioning of the isotope separation systems, is the tolerable amount of protium in the fuel. Current numbers are in the order of 1%. The higher this number, and the lower the additional protium amount that is added to the exhaust from the walls inside the plasma chamber, the lower the technology effort for protium removal becomes, see Day p. 29.

For an FPP relevant plasma, a radiating gas needs to be seeded in the core (at low concentrations) to help to reduce the power load to the divertor region, which puts an additional demanding requirement on pellet injection technology. It is currently under study how to admix such gas (for example xenon) to the fuel pellets; alternatively, a dedicated impurity injector may be necessary.

Gas injection into the main chamber volume is a technology that is being used for main core fuelling in all present-day machines but is not expected to be FPP relevant, due to the high density in the plasma scrape-off layer which results in ionization before entry into the core plasma, making fuelling efficiencies unacceptably low. For similar reasons, the wall recycle fluxes will also not significantly contribute to effective fuelling in an FPP environment. For an FPP relevant plasma, DT gas injection is foreseen to control the density of the divertor and of the scrape-off-layer, as well as sustain the separatrix density, independently of the core density. The role of the gas injection system will also be central in the plasma ramp-up phase, where the start of the discharge is achieved with gas injection for a limited period before pellet injection adds the necessary fuelling rate to achieve the burning plasma operating density.

The technology behind gas injection is simple. It entails piping with radiation hard modulating valves, sufficiently removed from the plasma to endure in such environment. This results in slow time response due to the conductance of the long gas tubes that makes divertor control more challenging.

The separation between gas injection for edge fuelling and pellet injection for core fuelling in ITER will allow the use of different D/T isotopic ratios for fuelling the edge and the core plasma, aiming to reduce the tritium throughput, see Loarte p. 13. But DIR and its consequential reduction of full isotopic separation results in a stiff isotopic composition of D/T, which limits the flexibility to exploit the relative mix of fuel as a virtual actuator. It is anticipated that for FPP relevant scenarios with significant power exhaust from the core, impurity concentrations at the plasma edge may be large enough to lead to an inwards flux of ions from the peripheral plasma into the confined plasma, thus coupling the edge/core fuelling and D/T ratios. JET DTE2 confirm that for very high density plasmas, 50/50 D/T fuelling was not necessary to obtain the highest D–T fusion rate. This suggests that DT mixing in the core to an optimal ratio is due to transport and not explicitly to the fuelling ratio, see Garcia p. 20.

Gas puffing into the divertor volume is expected to provide stable control of the plasma detachment, which is essential to ensure that the heat flux on the divertor target plates stays below the limits of the materials' resistance. For this purpose, the gas injection system will be embedded in an integrated control strategy to mitigate reattachment events. Feed forward control requires characteristic gas injection flow times in the order of the divertor neutral density response time. As the gas to be injected is mainly DT, and this limits the supply pressure of the injection pipes to be below atmosphere, the gas injection dynamics may be too slow. It is currently being actively researched how such an integrated controller could look like.

The gas puffing system will also be responsible for injecting radiative gases (such as argon) that work in the colder divertor area. The use of such plasma enhancement gases results in additional separation needs on the fuel cycle exhaust processing side. A general complexity of using noble gases in the plasma chamber comes from the activation caused by the D-T fusion-generated neutrons, which will require to add gamma radiation compatible system design features.

Vacuum pumping of the exhaust gas via slits and openings in the divertor cassettes acts only as a slow actuator on plasma operation. Different from current experimental fusion devices, the overall machine gas throughput necessary to maintain a given density and hence fusion power in an FPP is a fixed number. The pumping speed grows with increasing divertor pressures due to the associated higher neutral collisionality in the flow. The machine gas throughput unfolds as the product of effective pumping speed and divertor pressure. The divertor puffing rate influencing the divertor pressure is therefore closely related to the particle exhaust via vacuum pumping, see Varoutis et al. p. 33.

The vacuum pumping system needs to fulfil the different requirements during burn (high throughput, Pa range pressure) and dwell phase (from evacuation of the vacuum chamber with hot outgassing walls to plasma discharge re-start condition (mPa range) in minimum time). This combination of requirements can make operation very complex, see Schwenzer p. 44. The separation of these different requirements on separate systems is not considered to be an option, as the openings would take too much first wall's surface that would otherwise be used for tritium breeding.

## 4. INTERFACE BETWEEN FIRST WALL AND THE FUEL CYCLE

The first wall of an FPP will feature the tritium breeding blanket. From the fuel cycle point of view, its main function is to generate further tritium fuel via the reaction of the neutrons generated in the core of the DT plasma with lithium (enriched in the Li-6 isotope content). As the non-breeding structures of the plasma chamber will also absorb neutrons, there is a need to combine the lithium in the blanket with neutron multipliers for loss compensation. This will produce tritium breeding ratios (TBR) above unity (a TBR of unity is defined when every fusion neutron produces a triton from the lithium breeder).

In order to achieve the required TBR, big cut-outs in the first wall, such as for additional vacuum systems or openings for neutral beam/electron cyclotron heating systems, will need to be minimized. It will be necessary to minimize tritium losses from the plasma to its surrounding systems, such as the tritium breeding modules, plasma-facing components, cooling system, etc. Tritium will be retained in the traps (produced during manufacturing or induced by neutron irradiation) of the metal walls of these components and systems. Many of the related underlying critical processes, comprising tritium migration (erosion, transport, deposition and dust formation) and tritium retention (co-deposition, implantation, diffusion and permeation), see Brezinsek et al. p. 24, cannot be studied in present day devices. Linear plasma devices are being exploited to address tritium retention and permeation issues associated with high neutron fluence and high wall temperature, see Shimada p. 27. The design of the tritium breeding systems and plasma-facing components will therefore have a direct impact on the performance, operational safety and cost of any future FPP, as well as bear a strong impact on the fuel cycle design requirements. Isotope exchange reactions will lead to a significant build-up of tritium inventory in the wall. Neutron wall loads will lead to material damage (dpa, helium bubbles), which further increases the potential tritium uptake capability. A good understanding and the quantification of in-vessel tritium retention is necessary. Strategies of cleaning and in-vessel tritium recovery need to be demonstrated for use if one reaches the inventory limit, see Ashikawa p. 55.

Tritium trapped in the chamber materials can also act as a source term of tritium release during any unintended event. If coolant is lost under accidental conditions, the release of trapped tritium may occur due to temperature increase by decay heat. Hence, the trapping of tritium needs to be well understood and minimized, see Hatano p. 52. Tritium retention and removal was intensely studied as part of the DTE2 campaign in JET, see Douai et al. p. 56. In addition, outgassing from the first wall is the main load on the vacuum pumping systems during the dwell phase. Consequentially, if too strong, outgassing can become a design driving feature and potentially increase the dwell/burn time ratio.

In addition to the breeding functionality, the blanket will also be the main source of heat extraction. About 80% of the energy of the fusion reaction is carried by the neutrons. This, together with the energy produced by additional exothermic nuclear reactions, is transferred to the blanket coolant, which (in most concepts) is water or helium. The operation temperature of the blanket will be sufficiently high so that fractions of the produced tritium will permeate into the coolant. The fuel cycle will thus need to recover this permeated tritium from the coolant in addition to extracting the formed tritium from the blanket.

The FPP will be equipped with a primary and a secondary coolant loop for heat extraction and energy conversion. Such a configuration inevitably opens a path to tritium migration because heat removal and tritium production occur both in the blanket region where the presence of high temperatures, large metallic surface areas and high tritium concentrations facilitate the permeation from the blanket to the primary coolant, see Santucci et al. p. 49. Once in the primary coolant circuit, tritium will permeate either across the coolant tubes or into the secondary coolant loop, from which it can reach the external environment. The management of such tritium migration pathways is therefore a very important task, see Humrickhouse p. 47.

Finally, the blanket is also linked to the fuel cycle outer loop via the stream of extracted tritium. Different technologies are under discussion depending on the blanket concept and the choice of breeder material. For the fuel cycle, it is particularly relevant whether the extracted tritium comes in relatively clean form (which would be the case for some liquid breeder technologies like vacuum sieve tray or permeation against vacuum) or in small fraction in a large carrier gas stream (such as the helium purge in the case of a solid breeder).

## 5. INTERFACE BETWEEN DISRUPTION MITIGATION AND THE FUEL CYCLE

Disruptions are the result of rapid losses of the thermal and magnetic energy stored within the plasma. These phenomena, which can occur in tokamak based FPP, cause sudden curtailment of the plasma operation. Disruptions occur when: (i) a plasma stability limit is reached; (ii) when systems malfunction and plasma control is lost; or (iii) when unintended material enters the plasma. The mitigation of plasma disruptions in tokamaks is necessary due to the high magnetic field and high plasma current that can lead to deleterious effects on the internal components, due to the high heat flux from the fast dissipation of the plasma thermal energy and the magnetic stored energy, which lead to large forces. The disruption can also lead to the formation of several megaamperes of energetic runaway electrons during the current quench, which can damage the internal component surfaces.

The mitigation of disruptions is a critical area of current research. Several methods have been developed to potentially mitigate disruptions in ITER, see Baylor p. 41. These methods include injecting radiating material deep into the plasma to rapidly radiate the plasma thermal energy, as well as to densify the plasma to prevent the formation of runaway electrons during the current quench. The use of cryogenic pellets is attractive because these convert to gas, which can be pumped out. The quantity of material injected will be on the order of two orders of magnitude more than the content of the burning plasma and, therefore, pumping out this gas will take some time and impact the overall fuel cycle of a tokamak based FPP. Because disruptions are not present in stellarators, an FPP based on such type of fusion concept would not need to address this challenge in the fuel cycle.

The cryogenic pellet material injection in the form of shattered pellets is the most likely candidate for disruption mitigation (DM) and is being studied for ITER. Other solid materials such as Be, B, Li, and W in the form of dust or small pellets have been considered, but these will lead to dust or coatings that can trap tritium in the vessel and result in a large tritium inventory inside the machine. The dust can also lead to a possible toxic hazard or explosive mixture depending on the material used. Since the tokamak vessel gas pressure, following a disruption mitigation event using cryogenic pellets, will be in the 100 Pa range due to the amount of material injected to prevent runaway electrons, there will need to be a roughing pump system that can handle this gas load similarly to the initial pump down of the torus. This roughing pump system needs to be able to handle tritium at less than 1% levels and such technology is being developed for ITER. The time to pump out this gas will likely be more than an hour, which limits the recovery time of a tokamak FPP from a disruption, if the gas is the only deleterious result from the disruption.

The exhaust gas from the disruption event pump down will need to be processed to remove the tritium that was in the vessel at the time of the disruption. Achieving this with current methods can be an expensive and time-consuming process, that may need to be done as a batch process independent of normal fuel cycle operation, in order to get the FPP back in operation quickly. Following the disruption mitigation, in addition to the injected material from a cryogenic pellet, there will likely be additional gas that enters the vessel from the propellant gas that accelerates the pellet into the plasma. This adds to the amount of gas that needs to be pumped and processed. Deuterium, hydrogen (protium) and helium are the logical light gases to be used for this application, and each of them with advantages and disadvantages for their use and processing.

Finally, the DM system needs to be located relatively close to the plasma in order to have sufficient time response to mitigate a disruption. The ITER system will be located some 6 m away from the plasma, occupying valuable space that in an FPP may need to be utilized for blanket manifolds and tritium extraction processing equipment. The system will also need multiple penetrations through the vacuum vessel and breeding blankets to allow the material access to the plasma for suitable disruption mitigation. In an FPP, it will be necessary to keep the tritium breeding area high and reduce interface complications, thus requiring DM systems that minimize penetrations in the breeding blankets.

## 6. CONCLUSIONS

- In the currently assumed FPP relevant fuel cycle architecture with DIR, most of the exhausted unburnt fuel will not be isotopically separated. This will limit the flowrates of pure D/T, if plasma control requires it, while DIR will allow for larger DT throughputs without significantly increasing the integral tritium inventory of the plant.

- The fusion output from D–T during the DTE2 campaign in JET was found to tolerate larger deviations than expected from a 50/50 D/T composition of the fuel.
- The aspects of H management, i.e., the acceptable limits in the fuel for technology (pellet formation) or plasma physics (stable burn) reasons, can be design driving and potentially result in very large isotope separation systems.
- Processing large streams of deuterium with high purity represents a very large burden on the isotope separation systems and needs to be avoided.
- The amount of tritium that would be trapped on the first wall of an FPP and in the form of dust is not sufficiently quantified, and this is real challenge in terms of associated safety risks and tritium accountancy.
- The disruption mitigation systems have a significant impact on the fuel cycle pumping system, and therefore these need to be developed in joint integration.

To summarize, additional R&D is needed in this complex area of the fuel cycle systems directly interacting with the plasma. The important role of the fuel cycle for an FPP operation has been for long underestimated and the challenges outlined above need to be addressed for an FPP based on the D-T fuel cycle. Overambitious plasma requirements can potentially result in the fuel cycle plant becoming unattractively large. This is why it is paramount that the development of plasma scenarios and their operation strategies is conducted in close interaction with the technology developers, to ensure that the integration will work within acceptable requirements on both sides. Ideally, this needs to be addressed in broad and international R&D programmes, exploiting the different expertise in the fusion community.

## SESSION I: PLASMA CHAMBER AND TRITIUM BEHAVIOR

**PLASMA CHAMBER PARTICLE BALANCE AND PHYSICS OF FUEL BEHAVIOR**

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The presentation provided an overall description of the physics processes involved in plasma fuelling, impurity exhaust and in–vessel tritium retention and removal in fusion devices. Special emphasis was given to:

— The specificities of these processes related to the need for core–edge integration in FFPs, such as stationary power load control by radiative divertor operation and transient power load control (edge localized modes or ELMs);
— The differences between the present generation of fusion devices and the next generation of FPPs under design or construction, with specific examples from ITER integrated scenario simulations.

The main core–edge integration aspects for FPPs (stationary power control and helium exhaust) require operation in radiative divertor regimes, which are achieved by maintaining a high divertor neutral density and a high edge plasma density. Under such conditions, the total deuterium–tritium (DT) fuel and helium throughput can be controlled by varying the speed of the in–vessel pumps, since the neutral pressure is sufficiently high to provide efficient pumping. ELM control, on the other hand, is accompanied by an increased DT throughput whether because of the additional pellets required to trigger ELMs in a controlled way or to compensate for the deteriorated core particle transport when external 3–D fields are applied for ELM control.

Regarding plasma fuelling, the large spatial scale of FPP plasmas renders gas injection and recycling fluxes very ineffective to fuel the core plasma, for which pellet injection is needed. This poses specific integration issues with high radiative divertor operation since edge density transients may cause thermal instabilities in the plasma. Similarly, core DT density and D/T mix control issues arise since the DT fuel by pellets is peripherally deposited (typically in the last 20% of the plasma) and D and T need to transport into the core where fusion reactions take place. This process has typical timescales of 10 s in ITER and is determined by turbulent transport, which itself depends on the D/T ratio. Of particular importance, in this respect, is the control of plasma density and DT mix in the phases of the plasma scenario where access to and exit from burning plasma conditions takes place. In these phases, a too high/too low plasma density or a too high/too low D/T ratio can prevent access to burning plasma conditions altogether or to a fast exit from burn. The latter leads to unacceptable power fluxes to plasma facing components and the loss of radial position control.

On the other hand, the separation between gas injection for edge fuelling and pellet injection for core fuelling in ITER allows the use of different D/T isotopic ratios for fuelling the edge and the core plasma (pure D for edge fuelling and ~ 50–50 DT for core fuelling). This can significantly reduce the T throughput required to sustain the burning plasma since core fuelling is typically only 25–50% of the total fuelling required in high Q operation in ITER, where Q is the fusion power gain. In this respect, it should be noted that the separation between core and edge fuelling found in ITER may not apply to DEMOs for scenarios with significant power exhaust from the core, which are unlike ITER. In such cases, impurity concentrations at the plasma edge may be large enough to affect DT transport and lead to an inwards flux of ions from the peripheral plasma into the confined plasma thus coupling the edge and core fuelling and D/T ratios.

ITER operation and demonstration of burning plasma condition is expected to resolve most of the open issues in the area of plasma fuelling and impurity exhaust and thus provide a solid basis for the final design of the DEMO fuel cycle.

Another issue that considerably impacts the fuel cycle is that of in–vessel T retention. This is linked to plasma–wall interaction processes leading to formation of co–deposited layers, tritium implantation and permeation to cooling circuits, and dust production. In this respect, the dominant process in ITER is driven by the erosion of beryllium from the first wall and its co–deposition with T, rather than implantation in tungsten. Although T retention is expected to be very moderate in ITER, which is supported by the JET ITER–like–Wall experimental results, routine removal of the retained T by conditioning techniques and specific plasma operation will be required to remain under the T in–vessel inventory limit throughout ITER's lifetime. While the ITER experience in this area will be valuable for future DEMOs, it will leave a range of open issues. DEMOs are expected to operate with tungsten plasma facing components and high temperature cooling loops for which the dominant T retention processes are implantation in neutron irradiated tungsten and permeation into cooling loops; these processes only play a minor role in ITER.

## ISOTOPIC FUEL TAILORING AS ACTUATOR FOR BURN CONTROL IN TOKAMAK FUSION POWER PLANTS

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The regulation of the amount of fusion power produced by future FPPs will require precise control over the plasma density and temperature. Therefore, the control of the core–plasma kinetic state, usually referred to as burn control, arises as one of the most fundamental problems in nuclear fusion and will be critical to the success of burning–plasma devices like ITER. Due to the nonlinear coupled dynamics of the plasma, feedback control of the burn condition will be necessary to avoid undesirable transient performance, to respond to changes in plasma confinement, impurity content, or operation conditions, which could significantly alter the plasma burn, and to avoid potentially disruptive plasma conditions due to thermal instabilities. When the control objective is not only to regulate the plasma around a desired fusion power but also to have the capability of determining the operating point in terms of plasma energy, density and tritium fraction that provides the desired fusion power, a model–based control approach is needed to consider the high dimensionality and the coupled dynamics of the burning plasma. This control capability will be absolutely necessary in FPPs in order to keep the operating point far from stability and safety boundaries in order to avoid disruptions and protect the device while regulating the fusion power. Since the primary goal of a burn controller is usually to regulate volume–averaged properties of the plasma, zero–dimensional models are appropriate for control synthesis. However, if the goals were different for different regions of the plasma (e.g., core vs. edge), dynamic models capturing the spatial dependence will be needed for control synthesis.

A well–design burn controller should be able not only to achieve tight regulation around a desired operating point by rejecting perturbations in temperature and density but also to drive the plasma from one operating point to another during the burning plasma mode (e.g., different $Q$ or fusion power), to access to and exit from the burning plasma mode, and to handle the nonlinear coupling with other competing controllers using shared actuators. These control objectives demand not to neglect the nonlinear dynamics of the plasma during the model–based control synthesis process. In order to overcome the operability limits imposed by the linearization of the burn dynamics, nonlinear techniques for burn control have been proposed to account for the non–local character of the dynamics. Model–based control designs incorporating the nonlinear dynamics of the plasma have shown higher levels of performance, stability, and robustness against model uncertainties [1, 2]. These controllers utilize several actuators simultaneously, using auxiliary power modulation to prevent quenching, impurity injection to stop thermal excursions by increasing radiation losses, and fuelling modulation to regulate the density.

The relative mix of tritium fuel in the plasma was exploited more recently as a virtual actuator to cool the plasma during thermal excursions [3]. In this way, impurity injection, and the consequent reduction in fusion power because of enhanced radiation, needs to be used only in cases where the control of the relative content of tritium in the core is limited by severe particle recycling conditions at the wall, which is not what is expected for ITER. The isotopic fuel mix in the plasma is a critical FPP parameter as it has a major influence on the fusion power produced. Differences in deuterium (D) and tritium (T) transport and fuelling efficiency, as well as perturbation introduced by other sources of particles not under the burn controller such as neutral beams injections, may lead to a non–optimal fuel mix in the core even with an optimal

50:50 DT injection. Additionally, depending on the operating scenario, it may be desirable or even necessary to operate at a lower tritium fraction or vary the tritium fraction during operation. The regulation of the tritium ratio is possible thanks to a method of fuelling referred to as isotopic tailoring, in which the relative mix of deuterium and tritium injected by the fuelling system is varied in real time. The pellet injection system for ITER will include two separate injectors—one with pellets made of primarily deuterium and the other with pellets made primarily of tritium. A gas injection system will be used to supply deuterium at the edge of the plasma. Together, these systems will allow for isotopic fuel tailoring in real time. Recent experimental results in DIII–D have shown the in–vessel coil system as another effective actuator for burn control. The in–vessel coils can be used to generate non–axisymmetric magnetic fields that are able to reduce the energy confinement time, becoming in this way an alternative actuation mechanism for reduction of the plasma stored energy in case of thermal excursions, which can be combined with isotopic fuel tailoring [4]. The lack of an isotopic fuel tailoring capability in future FPPs will not prevent a well–designed burn controller from regulating the fusion power (note that control designs such as those proposed in [1, 2] do not exploit isotopic fuel tailoring) but it may limit the capability of the controller in terms of simultaneous regulation of other important plasma properties defining the operation point within the operation space due to the loss of actuation capability.

Unmeasurable variations over time (biases and drifts) in the tritium concentration of the fuelling lines could have an important impact on FPP operation. Limitations in the tritium plant subsystems could result in a lower tritium concentration. Even if the nominal concentration could be initially achieved, it might not be possible to sustain it for the total duration of long pulses. This not only imposes burn–control challenges but also raises concerns over having the fuelling capability to sustain long–pulse high–$Q$ operation in ITER. While methods have been successfully proposed to make the designed burn controllers more robust against these variations [5], the operational space can significantly shrink. The shrinkage of the operational space, illustrated in terms of plasma operation contour (POPCON) plots in [6], cannot be overcome by any control design and is fundamentally an actuation–related constraint. In other words, while robust controllers can still effectively regulate operation within the operational space, they cannot do anything against the shrinkage of the operation space itself. Limitations imposed by the tritium concentration in the fuelling lines together with actuator constraints and operational limits imposed by safe divertor operation [7] will at the end dictate the space where the FPP can operate.

An overall burn control solution with the capabilities described above will demand not only the design of a nonlinear controller but several other key components such as:

— A state observer to estimate in real time from limited–in–number and noisy diagnostics the plasma properties that are needed for feedback control [8];
— An online optimizer to determine in real time the references needed by the controller for all the controlled plasma states given the desired operating point by the operator [9];
— Adaptation mechanisms to estimate in real time those poorly known components of the model embedded in the controller in order to achieve a high level of performance robustness [9, 10];
— An actuator allocation algorithm to map in real time the request by the controller in terms of heating and fuelling to ITER's actuators while considering actuator dynamics and time–varying model uncertainties in the mapping [11, 12].

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## INTEGRATED POWER AND PARTICLE EXHAUST SCENARIOS

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Besides sufficiently high energy confinement, there are various requirements for an integrated DEMO plasma scenario: assuming 500 MW core loss power (400 MW from alphas + 100 MW for current drive and control), about 350 MW need to be radiated by a seeded noble gas inside the separatrix, depending on the actual H–L power threshold. About 100 MW divertor radiation are needed to achieve low enough temperatures for low tungsten sputtering rates and the achievement of divertor detachment. The fuel dilution caused by core and divertor radiators need to be sufficiently low to maintain the required DT burn rate. This scenario needs to be integrated with no–ELM (edge located mode) conditions, achieved either with the help of magnetic perturbations or pedestal tailoring by the combination of radiation cooling and intrinsic magnetohydrodynamic and turbulence mechanisms.

Inside the pedestal, the relation of radiated power and fuel dilution can be well predicted via atomic data and impurity transport calculations, with Xe, Kr and Ar being main candidates. Generally, a higher Z results in less dilution per radiated power but may cause a higher risk of central impurity peaking. A major fraction of the power crossing the separatrix needs to be radiated in the divertor to achieve the required low temperatures which facilitate momentum loss processes and result in very low tungsten sputtering rates. Candidate elements used in present day devices are here $N_2$, Ne and Ar. Nitrogen is probably not feasible in an FPP due to issues related to the formation of tritiated ammonia molecules. Neon exhibits a lowish radiative capability under divertor conditions of present devices in line with low $L_z$ values at temperatures below about 20 eV, in particular in comparison to nitrogen. In contrast, SOLPS–ITER modelling predicts a comparable radiative capability of neon and nitrogen for ITER divertor conditions. Argon is a good candidate as divertor and pedestal radiator, albeit a high divertor enrichment of argon is required to avoid excessive core fuel dilution. Here, further modelling is required for argon divertor radiation as well as its transport in the pedestal region.

An important interface parameter for the different requirements is the electron density at the separatrix. Quite generally, a higher separatrix density on the one side favours radiative cooling and divertor detachment, but on the other side may lead to degraded energy confinement and reduced current drive efficiency (which favours high $T_e$ and low density). Divertor power dissipation is closely linked to the divertor neutral pressure, and thus particle throughput for given pumping speed. Given the foreseen pumping speed for DEMO of ~ 100 $m^3/s$ and a divertor neutral pressure of ~ 10 Pa, a high enough particle throughput can be expected for the sustainment of the required helium removal even if some de–enrichment of helium in the divertor occurs. The separatrix density weakly rises with divertor neutral pressure, and even complete saturation is predicted for ITER by SOLPS modelling. Optimization of the separatrix density requires further work on divertor modelling, maybe including a more explicit treatment of turbulence, an improved pedestal model and core transport modelling including radiation and fast ion effects.

The largest uncertainty regarding an integrated scenario is related to the no–ELM requirement. Extrapolation of experiments in present day machine is hampered by the fact that a combination of high separatrix–and low pedestal collisionality is only possible in a large device of ITER size. This means that extrapolation needs to rely on first principle codes, whose validation is an ongoing task. One example is the empirical absolute pedestal density threshold for ELM suppression by resonant magnetic perturbations in present devices, which is well below the planned ITER and DEMO density values. Natural no–ELM scenarios have often an upper power threshold, beyond which ELMs reappear. In this case, combination with radiative power removal is mandatory.

**PLASMA CORE TRANSPORT OF D AND T AND IMPLICATIONS FOR THE FUEL CYCLE**

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During the first deuterium–tritium (D–T) campaigns performed in JET and TFTR in the 90's it became evident that DT plasmas behaved significantly different to D–D. In particular, it was obtained that the energy confinement time had a strong dependence on the isotope but as well on the plasma conditions. On the origin of such differences, it was found the heat transport driven by turbulence, which significantly changes from D–T to D–D. In a similar way to global confinement, particle confinement was also found to change from D–D to D–T.

The differences, in terms of particle transport, between D and T in DT plasmas were not studied in detail due to the inherent difficulties to measure core isotope content. After the 90's experimental DT campaigns, several theoretical studies analysed particle transport of T and D in DT plasmas. Gyrokinetic studies pointed out that D and T transport is asymmetric. When considering realistic plasmas conditions, T heat and particle transport is lower than D with an increased inward particle pinch. This might suggest an asymmetry in the D and T densities in DT plasmas, which could lead to difficulties for the alpha power control.

In the framework of the preparation of the JET D–T campaign, this topic has been deeply studied. Experiments were performed in H–D plasmas with different concentrations of hydrogen H and with D neutral beam injection (NBI). It was found that D and H concentration largely follow the concentration measured at the divertor, regardless the fact that there is a significant core D fuelling from the beams. Several theoretical and integrated modelling studies have unveiled the concept of fast ion mixing which explain the insensitivity of multi–ion plasmas to the exact location of the ion sources. The fast ion mixing is a characteristic of multi–ion plasmas which are dominated by the ion temperature gradient turbulence regime. In such conditions, the ion transport characteristics are not necessarily the same as the electron transport. In particular, the diffusion and pinch of ions become very high, allowing for fast ion transport. Therefore, ion densities are mostly determined by turbulent transport rather than by specific particle sources.

The D–T campaign, performed at JET in 2021, has provided an excellent testbed to study the D and T densities in DT plasmas. In a broad variety of plasma conditions, high and low current, high and low density, fuelled by neutral gas injection, the core D and T ratio seem to follow the edge ratio. This is determined by comparing the neutron rate calculated in TRANSP, assuming a D/T ratio equal to the measured one in the divertor and comparing it to the measured neutron rate. Good agreement is usually obtained, which would mean that the D/T ratio is quite similar and preserved from the edge to the core. These results are also obtained in conditions when only the T isotope is puffed and pure D NBI is used as heating, which means that core particle sources do not determine ion densities. As a consequence, core D/T ratio seems relatively easy to control with neutral gas injection, including in the case of very high density plasmas

The situation is less clear in DT plasmas at low input power (and hence low densities) where significant deviations between TRANSP and measured neutron rate are found. For the moment, no explanation for such a difference has been found but extensive sensitivity scans in TRANSP are being performed.

There was a set of DT experiments with D fuelling pellets into a plasma fuelled by T neutral gas injection aiming at 50/50 of D/T. By adjusting the pellets frequency, it was relatively simple to control the D/T ratio and obtain 50/50 D/T in the divertor. It is being analysed the consistency of the core D/T ratio with respect the edge D/T ratio.

**A SURVEY OF THE BEHAVIOUR OF IMPURITIES IN TOKAMAK PLASMAS**

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In an FPP plasma, impurities are present due to multiple sources. In the centre, He ash is produced by fusion reactions, at the plasma facing components impurity atoms are eroded and gaseous species are actively seeded to reach a tolerable heat exhaust.

The impurity density profiles will be the result of a combination of the strength of the sources and of the transport, determining the concentration levels and the profile shapes. For He and the eroded impurities, a low impurity confinement time needs to be achieved and the erosion yield needs to be low which requires a low plasma temperature at the plasma facing component. For all impurities, a hollow impurity profile in the confined plasma would be beneficial.

The radial impurity flux has a diffusive and a convective part and, in the absence of an impurity source inside the plasma, the shape of the profiles is given by the ratio of the two parts resulting in peaked profiles for inward convection and hollow profiles for outward convection. Accumulation of impurities, i.e., an extreme peaking being much larger than that of the main ions, needs to be avoided. For all impurities, turbulent transport is characterised by diffusive and convective parts of similar magnitude. Collisional (neoclassical) transport, however, has a convective part that rises linearly with the impurity charge and is strong for high–Z elements. The direction of the collisional convection depends on the relative strength of the main ion density gradient (inward drift) and the ion temperature gradient (outward drift) and the collisionality regime.

As a consequence of the strong increase with charge of collisional transport, the transport of highly charged impurities is generally determined by a combination of collisional and turbulent transport, where the neoclassical transport is rarely negligible. Plasma toroidal rotation leads to a poloidal asymmetry of the impurity density on the flux surface for high–Z elements at high Mach numbers, which strongly increases neoclassical transport and affects the direction of the convection (more inward for high collisionality and more outward for very low collisionality and at high rotation). Light impurities, like He, are usually more strongly governed by turbulent transport. The relative roles of collisional and turbulent transport depend on the plasma region. Collisional transport prevails close to the centre, where turbulent transport is weak, and in regions of reduced turbulence, like in the transport barrier of the H–mode pedestal. Collisional convection of highly charged impurities is usually important everywhere in the plasma in present experiments.

In present tokamaks, accumulation of high–Z ions at the plasma centre is regularly observed when the central heating power is too low in combination with a good confinement at the edge, like in the H–mode. In the H–mode pedestal, a strong inward convection of impurities appears which increases with the impurity charge and outward transport due to magnetohydrodynamic (MHD) phenomena like edge localized modes are required to control the impurity content. Turbulent transport leads to impurity profiles which are either mildly peaked or even hollow. The turbulent impurity diffusion is larger than the heat diffusion when the electron heat flux is in the range of the ion heat flux and is much smaller when electron and ion heat fluxes are very different. The increase of turbulent diffusion provided by central heating can therefore reduce the unfavourable impact of inward collisional convection and avoid, or at least limit, the accumulation.

The following arguments lead to a picture that impurity transport is less problematic in a large burning plasma than in present day tokamaks. Using the scaling of the energy confinement time and typical heating powers, it is found that impurity transport is more strongly driven by turbulence in a large burning plasma than in present tokamaks. Furthermore, it is expected that toroidal rotation will only lead to small Mach numbers and collisional transport of high–Z elements is not boosted by a poloidal impurity asymmetry. In the H–mode pedestal, a large temperature rise is needed to achieve a burning plasma while the density rise will need to be low since a high density at the separatrix is needed to obtain a cold detached divertor plasma. For these conditions, the neoclassical convection will be outwardly directed provoking hollow impurity profiles across the pedestal and edge MHD modes are then not needed to control the impurity peaking across the pedestal as is required in present tokamaks.

# PLASMA–MATERIAL INTERACTION IN THE MAIN CHAMBER OF FUSION POWER PLANTS: THE ROLE OF HIGH–Z AND LOW–Z WALL MATERIALS ON EROSION, DUST, FUEL RETENTION, AND FUEL RECOVERY METHODS

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Plasma-material interaction (PMI) imposes a number of challenges on the operation of a next step fusion device or FPP associated with the lifetime of components, the sustainability of the tritium cycle, and ultimately with safety aspects [1]. The underlying critical processes under steady-state plasma operation can be split into two categories:

— Erosion, transport, deposition, and dust formation described in general in the term material migration [2];
— Co-deposition, implantation, diffusion, and permeation labelled in general as fuel retention [3].

The role and strength of the individual process depend primarily on the choice and energy of plasma or projectile species (D, T and seeding species Ar, Ne, etc.) and plasma-facing or target materials (low-Z species like C, Be, Li etc.- and high-Z species like W. Mo, steel, etc.). DT fusion neutrons with 14 MeV energy can induce additional damage to the first wall materials in the main chamber as their energy lays above the damage threshold [4]. This results e.g., for W in an enhancement of the fuel retention above roughly half a dpa, which is in the order of the end of lifetime damage in the only full burning device under construction - ITER. In an FPP of DEMO size with much higher expected neutron dose, neutron-induced effects at 50 dpa will play the dominant role in the fuel retention process in metallic plasma-facing components (PFCs) like W [5]. Also, low-z or graphite-based PFCs neutron-damage would cause the second main contributor to in-vessel T retention in addition to the most critical co-deposition [6]. However, the different processes of PMI cannot independently be addressed, but they interact with each other, and define together potential limits in steady-state conditions driven either by tritium safety, tritium self-sustainability, first wall lifetime by erosion and ultimately dust formation.

In laboratory experiments e.g., in MAGNUM-PSI, PISCES-B, PSI-2 [7] the basic mechanisms and their dependences on main parameters like impact energies, fluxes and material temperatures can be studied and applied to benchmark codes like TMAP for fuel transport in materials or SD.TRIM for erosion under ion impact. However, the integral behaviour, thus how the processes interact with each other, and define potential limits including synergistic effects can only be addressed in toroidal devices. The experimental results in devices like ASDEX Upgrade or JET can benchmarked against full device PMI codes likes WallDYN [8] for fuel retention and ERO2.0 [9] for erosion and deposition and applied for future applications in ITER. DEMO or other future devices providing that predictive plasma boundary information are given from codes likes SOLP-ITER [10] or EDGE2D-EIRENE [11]. However, the experimental information and modelling predictions for the impact of neutron-damage effects are still rare and topic of present-day research [12] and plans for future studies in linear plasmas compatible with T or with neutron-damaged materials [7, 13].

Predominantly experimental results from toroidal devices are taken from the JET and ASDEX Upgrade tokamaks will allow describing the differences between plasma operation in low-Z and high-Z first wall in deuterium. Both facilities underwent a transition from low-Z (full-C) operation towards metallic operation with difference, that the ASDEX Upgrade [14] transfer was sequential over years into a full-W device, and JET from low-Z (full-C with Be) to the ITER-like configuration [15] with low-Z first wall made of Be and high-Z divertor made of tungsten in one large installation.

A massive reduction of the first wall erosion, material migration towards divertor, and dust formation with majority of first wall eroded material deposited on plasma-facing sides in the divertor and not in recessed areas like in graphite [16]. The main reason for the change with metallic devices is the absence of chemical erosion of graphite as primarily driver of erosion at low impact energies [3]. A reduction of fuel retention by a factor 10-20 determined by co-deposition and implantation have been identified in both device [16, 17, 18] when transferring from carbon-based materials towards the metallic device, but adaptation of the operational space was required with tungsten [14, 19]. These studies were performed in deuterium plasmas and without active cooling of PFCs. Transfer, to complete steady-state conditions and devices with actively cooled PFCs was done recently in the EAST and WEST tokamaks [20] confirming the essential observations in the short-pulse devices regarding the PMI processes in low and high-Z facilities. Comparison of the impact of different fuel isotopes has been experimentally addressed in particular with the aid of JET, which operated recently in H, D, T and permits to extrapolate previous non-T operation findings in PMI towards FPP-like conditions [21].

Experiments at those tokamaks are used as reference cases to benchmark the Monte-Carlo code ERO2.0, the workhorse for full 3D modelling of global plasma material interaction and impurity transport in the plasma boundary layer. Complementary WallDYN describes the surface-state conditions based on rate coefficients and is the unique tool applied to simulate and calculate on and in plasma-surface materials in steady-state conditions. Plasma boundary information up to the first wall is required to achieve highest reliability. Both codes were recently applied together to predict for ITER in full 3D the critical first wall Be erosion locations, the material migration paths, and the operational limiting tritium retention by co-deposition and implantation for discharges in the so-called pre-fusion plasma operation in H and fusion plasma operation in DT [22]. Safe operation without active T cleaning is possible until the third phase of full power operation according to these predictions, otherwise active cleaning with e.g., glow discharge cleaning or ECWC or ICWC will needed to recover T in the device. It should be stressed, that Be is the determining material for first wall erosion, material migration and dust formation as well as tritium retention in ITER. The situation will be different in an FPP-class device with much higher duty cycle and availability. Such a device will use high-Z walls, presumably W, for plasma operation to achieve the required lifetime of about five full plasma-burning years before critical components need to be exchanged [23].

The corresponding plasma edge (SOLPS-ITER) and plasma-material interaction modelling (ERO2.0) for the European DEMO is currently in progress in order to identify the boundaries in fuel retention, first wall erosion and dust formation, but without consideration of neutron impact. However, none of the present-day tokamak operation and associated simulations can directly contribute as benchmark to the assessment of the role of neutrons on the PMI processes. Here, accompanied research in dedicated laboratories, which can mimic the impact of neutrons by heavy ion, proton or fission neutron impact is mandatory. Currently only modelling is used to transfer the physics from fission neutrons in lack of a facility to study material damage by 14 MeV neutrons. Indeed, a new facility, IFMIF-DONES [24], is in plan in Europe to address the gap in PMI with neutron-damaged material at the high dpa range in DEMO. New facilities with the goal to address both critical open issues at once, the neutron damage as well as the tritium breeding capabilities in the first wall blankets are currently discussed in the world-wide fusion community.

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## PLASMA CHAMBER PMI – LINEAR PLASMA FACILITIES (TPE, IMPLANTATION AND IRRADIATED MATERIALS)

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This presentation addressed tritium (T) retention and permeation issues associated with high neutron fluence and high wall temperature, which are not fully addressed in ITER, with the use of linear plasma device and irradiated materials under US–Japan collaboration.

INL's initial studies had indicated the concern and possibility of deep (>>10 μm) tritium migration in bulk (0.1–0.3 dpa) neutron–irradiated tungsten several year ago. Recently, this deep tritium migration and trapping was experimentally measured up to 11 μm by nuclear reaction analysis with 0.016 dpa neutron–irradiated W. Now we can conclude that T can diffuse and be trapped in bulk neutron–irradiated tungsten at the elevated wall temperature above 773K. Also, the recent results from all tritium retention studies with neutron–irradiated tungsten shows that T retention can remain relatively high even at elevated temperature > 773K due to high binding energy (binding energy >1.4eV) associated with neutron–induced radiation damages (i.e., trap site) and deep T migration at elevated temp. Only limited data is currently available to understand the fluence dependence on T retention with neutron irradiated W for high ion fluence above $10^{26}$ m$^{-2}$, but the limited data suggested that T retention continued to increase at the square root of exposure time (~$t^{0.5}$).

Future work with neutron–irradiated plasma facing components will focus on:

- Effect of impurities (e.g., He, N, Ar, O etc.) on surface chemistry and tritium migration and trapping;
- Incident deuterium (D) ion fluence dependence above $10^{26}$ m$^{-2}$;
- Explore high radiation damage (>> 1 dpa). INL will restart D/He plasma experiment with RB* neutron–irradiated W and W alloys in January 2013.

INL received ~ 300 neutron–irradiated samples from the High Flux Isotope Reactor at the Oak Ridge National Laboratory and will receive additional >30 samples through the IAEA coordinated research project in the near future. This will help further minimize the uncertainty associated with tritium behaviour in neutron–irradiated tungsten. In future fusion demonstration and pilot plants, featuring (all tungsten) plasma chamber, plasma material interaction processes will give rise to potential T–retention concerns associated with:

- Higher radiation damage in blanket first wall than that in divertor due to closer to the plasma core;
- Synergetic effect with charge–exchange neutrals to first wall materials.

# SESSION II: TRITIUM FUEL CYCLE ENGINEERING SYSTEM DESIGN

**DEUTERIUM–TRITIUM FUEL CYCLE: OVERVIEW AND DEMO OBJECTIVES**

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A self–sufficient fuel cycle is indispensable to enable commercial fusion energy. This puts additional requirements on existing fuel cycle concepts. Driven by the need to reduce the tritium inventory in the systems to an absolute minimum, the work package Tritium–Matter Injection–Vacuum (TFV) of the European Fusion Programme has developed a novel three–loop fuel cycle architecture. 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 principal feasibility to be shown. This contribution is addressing the inner fuel cycle and how it is linked to the plasma.

The innermost loop implements the functionality of direct internal recycling (DIR), in which a large part of the hydrogenic gas fraction in the exhaust gas is separated close to the divertor in pure form, so that it can be immediately recycled to feed the matter injection systems without further gas processing. This shortcut is established inside the tokamak building. From the fuel cycle point of view, the DIR is able to compensate for a low burn–up fraction and a low fuelling efficiency of the plasma and results in reduced flowrates to be processed by the tritium plant. In the EU concept, to achieve such continuous separation, metal foil pumps are utilized that work by super-permeation of hydrogen through hot metal foils. A collisional plasma source is generating the suprathermal hydrogen atoms at the right energy to pass the barrier at the surface of the foil. This technology currently has still a low technical readiness, but experimental performance results in lab scale are very promising. As fallback solution, multi–stage cryogenic pumping can be used, however this does re–introduce the issue of inventory build–up and requires additional gas processing due to imperfections in the separation efficiency.

Vacuum pumping in the EU–DEMO fuel cycle architecture is established by continuous technologies based on mercury as operating fluid. For primary vacuum, linear diffusion pumps with ejector stages, as appropriate, are foreseen, while rough pumping is performed by liquid ring pumps that can compress to ambient or even higher pressures, if designed accordingly.

The second loop provides tritium recovery of the arriving gas (non–recycled hydrogens and impurities). The produced hydrogenic stream is undergoing protium removal and isotope rebalancing. This is achieved in a cyclic temperature swing absorption process which acts as a pre–isotope separation so that the classical isotope separation process by cryogenic distillation in the outer loop can be dimensioned smaller. The product stream of the temperature swing absorption–based isotope separation step can then be continuously returned to be cut with the DIR gas stream. The known implementation of the temperature swing absorption process, known as TCAP, suffers from large inventory build and poor scalability. The European programme is looking in process variants with new materials that do not need cryogenic temperatures.

It was found that with the three–loop architecture, the operational tritium inventory of the fuel cycle is less than 2 kg, with isotope separation and pellet injection systems being the largest contributors.

As regards interfaces with the plasma, the following points have been identified:

— The working point of the DEMO fuel cycle is given by the overall gas throughput of the machine – considering all technological losses through the scrape–off layer – and hence strongly determined by the product of burn–up fraction and fuelling efficiency. Both numbers are not expected to improve significantly. DIR is seen as very efficient tool to avoid excessive gas processing in the tritium plant;
— Due to the principle of direct internal recycling and saving the effort of a complete hydrogen isotope separation down to the level of the pure isotopes, the main gas supply to the plasma is mixed fuel (D/T around the 50/50 composition point). Addition of pure deuterium or tritium streams is possible within a few % fraction of the total throughput. Larger flowrates require a shift of the DIR operational point, and correspondingly reduce the overall efficiency of the fuel cycle;
— Any protium that enters the fuel cycle – mainly by outgassing from and isotope exchange with the first wall – needs to be removed by the fuel cycle at significant effort. The metric for this effort is the to be removed stream over the stream accepted for plasma fuelling. The current design works with an acceptable hydrogen content in the fuel of 1%, which is a limit given by plasma physics reasons;
— The matter injection requirements of the plasma need to be carefully identified, in trade–off studies given technology constraints of pellet or gas injection. Gas injection at the size of an EU–DEMO is associated with timescales (sec range) that are too long to be helpful for divertor control under transient conditions (msec range) (e.g., detachment control). It may therefore be necessary to develop a pellet–based actuator to control the sub–divertor density;
— The noble gases considered for radiative seeding (currently Xe for the core, and Ar for the divertor) undergo neutron activation. The Gamma dose rate and associated decay times can potentially result in significant shielding efforts inside the tritium plant, an aspect which should be considered in a trade–off study in the final choice of the seeding gas;
— The EU DEMO team has developed tools based on the Boltzmann equation to describe the sub–divertor gas dynamics in a self–consistent way with the plasma boundary. This allows for a neat definition of the vacuum pumping needs on the basis of the to be extracted particle fluxes rather than of prescribed pumping speeds or divertor pressures.

## PLASMA FUELLING ON ITER AND NEW REQUIREMENTS FOR DEMO

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Matter injection systems in ITER and DEMO are expected to cover, apart from the tasks to enable for disruption mitigated plasma shut discussed elsewhere, requirements with respect to:

- Core particle fuelling;
- Edge localized mode (ELM) control and mitigation (as a necessity in ITER, as a potential option in EU–DEMO);
- Provision of "plasma enhancement gases" (PEGs);
- Fuelling during plasma ramp–up and ramp–down.

These tasks are to be shared between the pellet injection system and the gas injection system. Core particle fuelling is attributed solely to pellets in order to access and maintain a core density level adequate for a high fusion gain. The injection of core fuelling pellets will be performed in both devices from the torus inboard side via a guiding tube system. For efficiency reasons, it is envisaged to make use of pellet core particle fuelling tool already during the final part of the current ramp–up and initial part of ramp–down phase in DEMO.

ELM control and mitigation by pellet pacing is foreseen in ITER. For this task a dedicated injection path from the torus outboard will be provided; yet inboard injection will be an option for this task, too. For EU–DEMO, since currently an alternative plasma scenario without strong ELMs is envisaged, there is no dedicated request for ELM pacing yet. However, the option to provide pacing pellets is tried to be kept on board with low added effort to the fuelling system. Thus, no dedicated pacing path from the outboard side is foreseen in EU–DEMO.

Currently, the pellet launching system under development for ITER, expected to serve simultaneously for core fuelling and ELM pacing, is the main driver of the FPP relevant pellet technology. Aiming on a particle throughput in the order of $10^{23}$/s delivered for up to one hour, comprising both deuterium and tritium (T), it matches also already expected EU–DEMO needs. Dedicated experimental and modelling efforts are under way at ORNL to develop high throughput steady–state single– and twin–screw extruders. The ITER pellet launching system (PLS) will be laid out to supply pellets of different sizes and composition via different injection trajectories. For this, it will comprise a set of repetitive gas gun accelerator units located in three casks placed at three different toroidal locations at building level B1. Most precious knowledge and results have been obtained at ORNL with respect to the T handling technology. This ranges from the operation of T compatible pumps for up to one year and the experience–based improvements of the layout, the development of T compatible relief valves to the production and acceleration of T pellet. The latter proofing adequately sized pellets can indeed be formed and accelerated up to the relevant speed range.

For EU–DEMO, for ongoing design work the ITER solution is adopted as an interim assumption; yet here the PLS housed in an adequate moveable cask is placed above the torus vessel finally yielding a better performance than ITER. Other main decisions are also still pending. As pellet source, either a screw extruder or a gas dynamical extruder is considered. To accelerate the pellets, usage of a stop cylinder centrifuge or a high–speed double stage gas gun is suggested. For the reference case, employing a centrifuge in a cask delivering pellets via a guiding tube system, proper space reservations in the computer–aided design model are already in place. Both for the centrifuge and high–speed gas gun solution possible engineering setups covering different inboard injection configurations have been worked out. Scenario modelling showed both variants are expected to achieve adequate core fuelling with a modest particle flux in the order of $10^{23}$/s – provided the pellet size is suitable chosen and missed–out pellets can be replaced in due time. In case the centrifuge concept, successfully used for 30 years at AUG, is selected, optionally pacing pellets and even pellets with admixed PEGs can be delivered by the same PLS with minor extra effort. The according design of a multi–tasking PLS will be already applied at JT–60SA requesting simultaneous core density and ELM rate control. Hence, the JT–60SA system currently under design and manufacturing could act as potential EU–DEMO prototype. In order to close the still existing technology gap for EU–DEMO, a Direct Internal Recycling Development Platform Karlsruhe–Pellet Injection Engineering Test Bed (DIPAK–PET) mimicking the DEMO fuel cycle is currently set up. Its main task is the dedicated survey of PLS actuator development issues following the recommendations of the TFV (Tritium–Matter Injection–Vacuum) Design Review Panel related to technology. This is to address the following gaps:

— Quantification of transfer losses in guiding system;
— Enduring pellet production with repeatable reliable high performance;
— Tritium handling capability in the relevant operational regime.

In summary, the pellet systems share on the matter injection task covers mainly core particle fuelling. ELM pacing is a firm request yet only for ITER but considered as a low effort option still at EU–DEMO. The EU–DEMO consideration relying on the capability of a multi–actuator approach would then even allow the inclusion of admixed pellets. A dedicated European test bed will be set up in order to close the identified still existing technology gaps.

**PLASMA EXHAUST AND VACUUM PUMPING ON ITER AND OTHER DEVICES**

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The divertor system of a fusion device is always a compromise which needs to meet power exhaust, particle exhaust and neutron shielding requirements at the same time. The design space of the tokamak particle exhaust function results from a number of requirements, such as geometrical parameters (for instance the divertor cassette configuration, and the position of the pumping port relative to the divertor), the effective pumping speed that can be provided, the intra– and inter–cassette gaps which define the recycle flow pattern of the divertor cassettes as well as from the presence of the dome which is mainly defined by neutron shielding. Hence, a feasible divertor design needs to properly consider particle transport physics. Furthermore, since the neutral density in the private flux region (PFR) is expected to be high enough to justify viscous flow conditions, the corresponding gas collisionality increases and therefore a nonlinear (i.e., collisional) neutral particle transport treatment is required.

In that context, a numerical tool called DIVGAS (divertor gas simulator) has been developed at Karlsruhe Institute of Technology. The DIVGAS code is based on deterministic and stochastic numerical solvers of the Boltzmann equation. The aim of this code is to investigate and reliably describe the flow conditions in the particle exhaust of a fusion device. That said, DIVGAS considers all the physics and engineering aspects of plasma-wall interactions in the divertor, which influence the generation of neutral particles at the targets and consequently the overall flow behaviour of the particle exhaust, including the attached vacuum system. For validation purposes, the DIVGAS code has been implemented to model the neutral gas flow in the JET sub–divertor. Moreover, DIVGAS has been applied for simulating the particle exhaust of ITER, JT60SA, AUG, EU–DEMO, DTT and recently of the stellarator W7–X.

This contribution exemplifies the main workflow, which uses DIVGAS process to provide a self–consistent coupling between the sub–divertor volume and the vacuum systems. DIVGAS requires the total neutral flux (i.e., fuel gas and impurities) in the PFR as input boundary condition. This information is usually provided by an edge plasma code. Additionally, the actual 2D/3D divertor geometry is introduced. The outcome of the simulation is given by the total pumped throughput, the recycle flow from the divertor to the core plasma, and the distribution of the neutral pressures in the whole sub–divertor area, which directly points to the required total pumping speed, distributed among a certain number of pumps located in the divertor pumping ports.

Based on the aforementioned workflow, this contribution aims to highlight the impact of the main design drivers, illustrated by corresponding results, for various machines and divertor configurations (i.e., Single–Null, X, Super–X and liquid metal divertors) on the particle exhaust. In all cases, important design directions for achieving high pumping efficiency are presented. More specific the main conclusions related to divertor design recommendations, are the following:

— By using the divertor particle exhaust neutral code DIVGAS, the throughput number can be translated in a (cross–section multiplied by the capture coefficient) number for pumping;
— The neutral–neutral collisions in the sub–divertor area significantly influence the particle exhaust behaviour and by no means could be neglected;

— In the case of a divertor without a dome, a strong outflow of the molecules towards the core is observed – Due to the viscous effects, complex field lines with vortices are observed;
— It is preferable to have a dome/liner with maximum height and with a large pumping port size – In terms of cost a larger pumping port is more preferable than a very efficient pump;
— The position of the pumping port is preferable to be in the middle of the cassette for reduced outflux. The inter–cassette gaps have a moderate influence on the pumping efficiency;
— A “long–leg” divertor (i.e., X divertor or super X divertor) has a higher pumped flux compared to the SN divertor, due to higher neutral compression.

## NEUTRAL BEAMS AND THE REQUIREMENTS THEY PLACE ON THE FUEL CYCLE

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The use of negative ions is mandatory to produce high–energy beams of neutral atoms to be used in large fusion machines, like ITER and beyond. Negative ions are generated in a plasma ion source (typical plasma density and temperature are in the order of 1e18 $m^{-3}$ and few eV respectively) by a complex interplay of physicochemical processes requiring a careful control of the main parameters, including the source pressure and gas purity. In particular, the main production channel relies on the conversion of positive ions or neutrals emitted from the plasma on the surfaces of the ion source chamber. The negative ion yield has an exponential dependence on the difference between the work function of the material and the electron affinity of the negative ion [1]. The latter is about 0.75 eV for H– and D– ions, while the work function depend on the surface material, temperature. The surfaces of the ions source are typically coated with refractory materials to reduce the plasma sputtering. In the case of ITER ion source molybdenum coating was proposed, whose work function is about 4.3 eV Experimentally, it was observed that the in–situ evaporation of alkali metals like caesium (Cs) on the surface tend to lower the work function of the material down to about 2.2 eV, so that the negative ion production is dramatically increased [2]. This enhancement of negative ion yield is a key factor to achieve the required current densities to achieve the ITER neutral beam injection (NBI) requirements. Unfortunately, the work function achieved by this means is quite unstable during the plasma discharge, especially due to the sputtering of the thin Cs coating from the surfaces by the plasma ions. In particular, it was observed experimentally that the performances of the machine are strongly lowered when deuterium plasma is used, owing to the larger mass of the plasma particles and more effective sputtering of Cs. In particular, the change in work function is accompanied by large fluxes of co–extracted, causing to intolerable power fluxes to the grid of the accelerator [3].

The concerns on this detrimental effect on the negative ion production has driven the requirement to have an extremely pure gas feed into the ion source, down to 200 ppm (0.02%). The presence of certain amount of tritium in the gas, in fact, would cause a similar amount of tritium ions and atoms present in the discharges that could enhance the sputtering of caesium.

More recently, some experiment carried out in IPP, Garching, have shown that the presence of small fractions of $D_2$ gas in the H2 discharge has only a moderate effect on the negative ions and co–extracted electron current. The experimental conditions used (Filling pressure 0.43 Pa of $H_2$ with 0.03 Pa $D_2$) cannot be fully indicative of the situation of ITER ion source ($D_2$ filling pressure <0.3 Pa, with <0.02% $T_2$), but it is enough to show that the present requirements on gas purity are too restrictive. Probably tritium traces <0.5% are tolerable without any impact on the source performances, representing a >20 fold relaxation with respect to present requirements.

Another possible issue that was highlighted, is the acceleration of $T^-$ ion beams in the source, and their dumping onto uncooled parts of the beamline (that was optimized to deal with re–ionized protium and deuterium beams, falling at slightly different positions in the beamline, owing to their different gyro radius). Preliminary calculation shows that with the gas purity level suggested above, the associated power fluxes will be rather small (about 20 kW of power dumped on the front–end components and neutral beam, NB, duct). This number appear to be well manageable, but a careful check of the $T^+$ trajectories against the front–end components is advisable, to highlight possible weak point in the present cooling systems of such components.

Finally, we discuss the topic of the tritium retention into the surfaces bombarded by this stray T+, T– or T0 beams. Due to the high energy of such particles (0.8–1 MeV) they are implanted in the beam facing components (typically made in copper) at a depth of few microns. The particle flux on some of these components (namely, the panels of the residual ion dump, RID) can be rather high (in the order of 2.5e17 1/s), corresponding to an accumulated amount of tritium of more than 40 grams over the ITER lifetime. In reality, a saturation of the material is expected, so that the after a certain time, an equilibrium is reached, and a tritium atom will be released for each incoming ion. Assuming that such saturation happens when 25% of copper surface is filled with tritium (as reported for the surfaces bombarded with Tritium beams during experiments at JET [4]) that would correspond to a much smaller amount of retained tritium (about 0.5 grams). Moreover, a "clean–up" effect will be present, due to the presence of much larger fluxes of D–/D+ impinging in the same areas and replacing the tritium atoms in the material. This clean–up cannot be perfect, as implantation depth of deuterium and tritium, ions are different, but a large fraction of the 0.5 grams mentioned above are probably replaced by deuterium atoms.

In summary, the present requirement for $D_2$ gas purity in the ions source (<200 PPM of $T_2$) was set on the basis of the concerns about the degradation of source performances and the location of power loading of re–ionized T+ atoms. Experiment at IPP show that purity levels in the order of 0.2–0.5% are probably acceptable. Tritium ion implantation would happen in the in–vessel component of the NBIs (ion source, RID, neutralizer), but qualitative consideration suggest that this would not be a serious issue.

We conclude that a strong relaxation of gas purity requirements is possible, from 200ppm to 2000 or even 5000 ppm that remains anyway quite challenging for the Isotope Separation System. The views and opinions expressed herein do not necessarily reflect those of the ITER Organization.

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## 2. ACKNOWLEDGMENTS

Author wants to acknowledge U. Fantz, B. Heinemann (IPP), D. Boilson, S. Willms, (IO), E. Sartori (RFX) and R. Hemsworth, for fruitful discussions and A. Litvinov, K. Roux (IO) for computer–aided design support in the molecular gas flow simulations.

## DEUTERIUM–TRITIUM FUEL CYCLE CONSIDERATIONS FOR PLASMA PHYSICS

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To provide some context of the relationship between the design and operation of an FPP and its fuel cycle, a brief overview of the ITER fuel cycle is provided first. As a reminder, the aspects driving the design and technology of the ITER fuel cycle are:

— Magnetic plasma resists injection of replacement fuel in the core – this dictates the need for core fuelling and the use of high–speed pellet injection;
— He “ash” and impurity build–up quench the fusion reaction which requires the need to extract and process the plasma exhaust;
— Tritium is radioactive, expensive, and only a fraction (~1%) of it burns per pass – this means that a once through process is not possible and it needs to be recycled;
— Vacuum needs to be maintained around the plasma to provide the conditions necessary to sustain a plasma and remove impurities.

The ITER fuel cycle has been designed to address each of these aspects together with the additional needs of the heating systems, the ITER Research programme and overriding safety objectives. The complete ITER fuel cycle with all its various subsystems including the hot cell detritiation systems and the interface to the tritium breeding test modules is shown in Figure 3.

The ITER fuel cycle has demanding requirements arising from its multiple, simultaneous, and varying missions. It needs to simultaneously provide FPP fuelling products (low purity $D_2$ and $T_2$ separately as pellets and gas), neutral beam neutralizer gas (medium purity $D_2$), neutral beam ion source gas (high purity $D_2$ and $H_2$) and discharge protium (with very low $T_2$ concentrations). These are to be provided in batches while the feed flow from the plasma exhaust varies cyclically.

Tritium as a material is hazardous, expensive, rare and has proliferation concerns. These lead to an overall objective to reduce inventory as far as reasonably practicable. Normally for complicated process plants, the use of feedstock and product storage provides buffering that helps to simplify control and smooth operating conditions. Though in the case of ITER, this is not possible for the fuel cycle given that inventory needs to be minimised. Without buffering capacity, faster processing times are required, with increased linking of control loops resulting in compounded reliability issues and the risk of increasingly unstable control as the damping effect of “capacitance” is reduced.

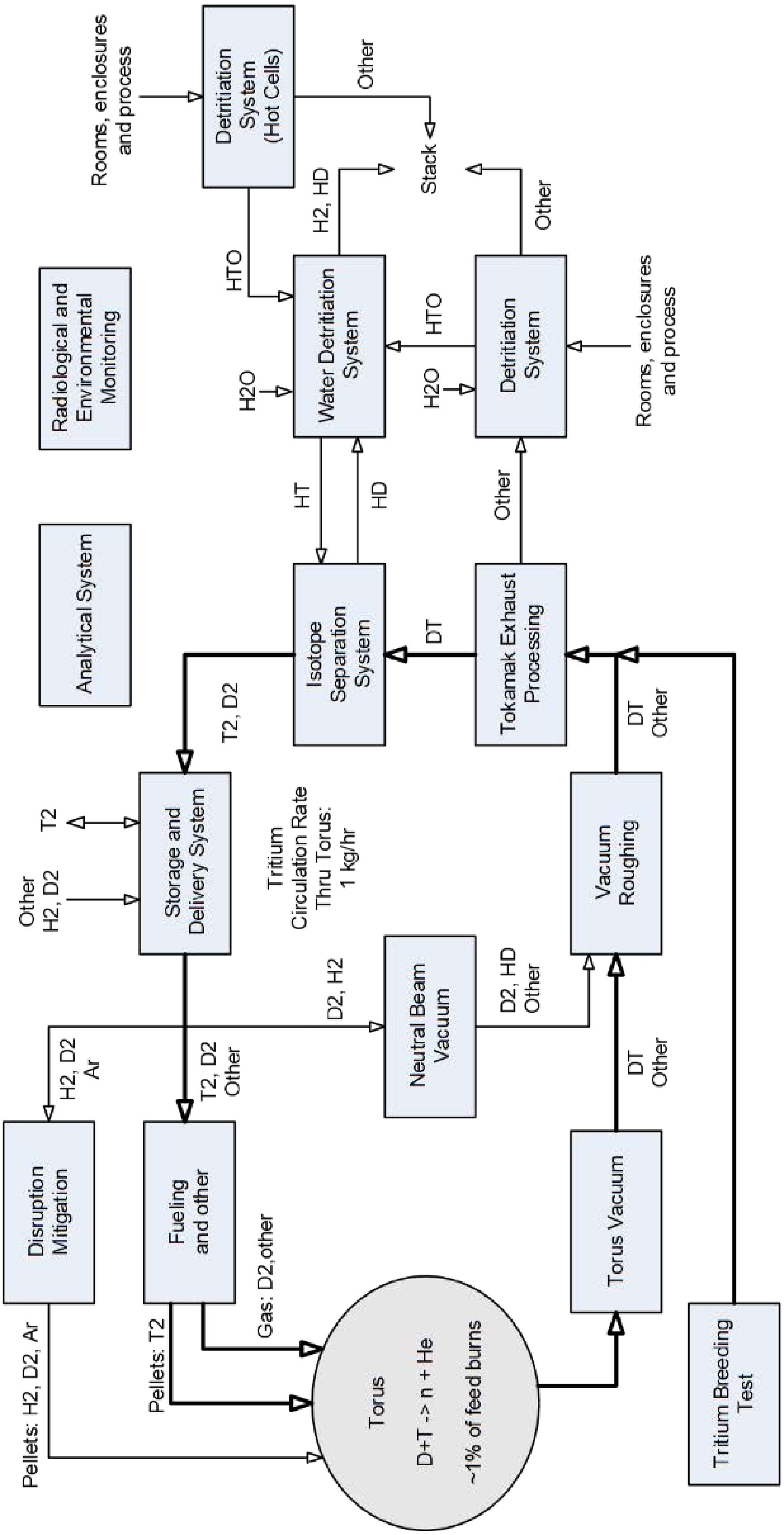


*FIG. 1. Detailed view of the ITER Fuel Cycle.*

The following section explains key relationships between physics decisions and the fuel cycle. The burn fraction (the fraction of tritium burnt divided by the tritium sent to the FPP for fuelling) is a major driver for the design of the fuel cycle. The tritium throughput of a fuel cycle is inversely proportional to the burn fraction. If it could be increased the throughput of tritium for fuelling purposes could be reduced, leading to smaller equipment and less inventory in the fuel cycle.

How the plasma core is fuelled and controlled has significant impact on the configuration of the fuel cycle. In ITER, there is a requirement to fuel separate T and D pellets to the FPP, this leads to the need for the $H_2$, $D_2$, or $T_2$ exhaust to be processed through an online isotope separation system (ISS). If there is no need to fuel separately and a fixed ratio of T to D is acceptable for the operation of the FPP, then direct recycling (i.e., bypassing of the ISS) is possible to some extent.

The purity of the product streams from the fuel cycle drives the complexity, size, and inventory of the systems. Lowering purity requirements means that fewer separation stages are required in the Isotope Separation System, less reflux is needed and subsequently less inventory is held up in the system. Lowering purity requirements will also help to relax processing times.

The isotope separation system is a cascaded cryogenic reactive distillation process. Concentration profiles are established in each of the four columns to produce the various product streams with different concentrations of the hydrogen isotopologues. Significant perturbation to these profiles can arise when the feedstock changes, e.g., switching from plasma exhaust (i.e., DT) to neutral beam regeneration gas (e.g., $D_2$). These perturbations can lead to unsteady state conditions in the columns resulting in difficulties to maintain product purities.

The neutral beams at ITER require three product streams from the fuel cycle, medium purity $D_2$ for the neutralizer, and high purity $D_2$ and $H_2$ for the ion sources ($H_2$ as an option for the diagnostic neutral beam). Due to inventory limits set for safety reasons on the neutral beams, the neutral beam cryopanels require frequent regeneration. This introduces a significant flow of $D_2$ to be introduced to the Fuel Cycle which will challenge the ability of the ISS to maintain gas purities. As a result, the ITER fuel cycle has the capacity to store the gas for processing during the silent hours overnight. Though this is a challenging mission as the time available is limited for warm up, processing and cool down operations with limited cryogenic cooling capacity available.

The selection of first wall material can influence the minor impurities found in the plasma exhaust. For example, naturally occurring uranium in beryllium can produce volatile fission products like radioiodine, whereas activation of calcium can create $^{37}Ar$. If these radioisotopes are released from the first wall, they will need to be processed in the fuel cycle.

The choice of seeding gas has multiple considerations for the fuel cycle. Selection of Argon or Nitrogen will result in more frequent 300K regeneration of the torus vacuum pumps leading to daily overnight processing of the air–like materials in the fuel cycle. The increased flow of this material leads to larger equipment to process impurities like tritiated methane and reduces residence time for decay storage (for Argon after it has become activated as $^{41}Ar$). Nitrogen also creates tritiated ammonia that will tend to stick to cool surfaces and will accumulate tritium contamination. The use of Neon avoids the frequent 300K regeneration and the subsequent challenges for the fuel cycle, though it is rare and expensive.

Finally, erosion products of the Pellets in the flight tubes needs to be managed. The quantity of erosion increases with pellet speed and so this needs to be considered in the design of the fuel cycle. The views and opinions expressed herein do not necessarily reflect those of the ITER Organisation.

**PELLET ELM PACING AND DISRUPTION MITIGATION IMPACTS ON THE FUSION FUEL CYCLE**

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The mitigation of edge localized modes (ELMs) using rapid frequent pellets and disruptions using material injection both have impact on the deuterium–tritium (DT) fusion fuel cycle because of the additional exhaust gas that needs to be pumped from the machine and processed to prevent becoming part of the normal operating fuelling material.

ELMs are edge instabilities in toroidal confinement plasmas (primarily tokamaks) that are driven by peeling ballooning modes driven by pressure and current gradients in the edge plasma of high confinement mode (H–mode) edge pedestal configuration. These instabilities result in filaments of plasma being ejected from the edge into the surrounding magnetic field layer outside that surrounds the plasma. The result of ELMs is an intense high heat flux filament of plasma that reaches the divertor in sub millisecond time scales that can lead to localized divertor material ablation and melting. Thus, to prevent damage shortening the lifetime of a tokamak–based FPPs these ELMs need to be eliminated or reduced greatly in intensity. One method that has been demonstrated in today's moderate size tokamak experiments is the use of small shallow injected pellets to trigger ELMs on demand at a much higher frequency than they otherwise normally occur. This can result in less intense heat flux to the divertor to prevent material damage. This technique is known as pellet ELM pacing and is an active research topic to understand how it may extrapolate to ITER burning plasmas and beyond. A side benefit from the pellet ELM pacing is that pellet material that is ablated in the edge plasma flows in the scrape of flayer when the triggered ELM occurs. This have been shown in multiple machines to reduce the impurity content in the plasma in both carbon-walled and ITER like metal-walled machines. In today's experiments the material is generally $D_2$, the same as the plasma fuel and thus the flux of fuel into the plasma chamber is increased from non–ELM pacing operation. Non–fuel pellets of room temperature solid material such as Li, B, and C have also been used to pace ELMs. These generally lead to a higher Z effective in the core plasma that is undesirable and result in dust and coating in the plasma chamber that could be a tritium retention by–product in a DT burning plasma. Thus, we are not considering this impact here on the fuel cycle.

In order to trigger an ELM, the pellet needs to be of sufficient size and velocity to penetrate into the pedestal region of the plasma and produce a localized pressure excursion that results in the ELM being triggered. Pellets used for this process generally do not result in net fuel being deposited directly into the plasma. The neutral particles that end up in the divertor can result in slow time scale fuelling from neutral penetration and ionization in the edge plasma. In a large burning plasma this fuelling effect is expected to be minimal due to the opacity to neutrals penetrating the plasma. Large fuelling pellets also penetrate beyond the pedestal region and generally also result in ELMs being triggered.

Pellet ELM pacing resulting flow rate can be larger than the fuelling flow rate and therefore can have a strong impact on the content of material in the fuel cycle. In ITER the fuelling flow rate from pellets is projected to be 30–70 $Pam^3/s$ is $5–9x10^{21}$ atoms/s. The ELM pacing needs for ITER ate projected to be a 40 Hz repetition rate to keep the heat flux below 0.5 $MJ/m^2$ and keep tungsten (W) impurity content in the plasma tolerable.

This additional input does not fuel the plasma but will end up as neutrals in the divertor and needs to be pumped away, so pumping/reprocessing requirements significantly increased. If $D_2$

only pellets are use, the direct internal recycling (DIR) recirculation loop in an FPP needs significant T supplement – thus it will be a big challenge to maintain DT mix while minimizing the overall T inventory.

Divertor detachment control is also complicated by this additional periodic non–axisymmetric particle source in the divertor. The question to be answered is will pellets impact a detached divertor with radiation instability events? AUG experiments have started to examine this in semi–detached plasma and DIII–D has plans for dedicated experiments in the next year.

Disruptions are a sudden loss of plasma confinement in tokamaks triggered by magnetohydrodynamic instabilities or from unintended material injection. They can cause material damage in the chamber from the thermal energy radiation and conduction and from impacts of runaway electrons that can form during the current quench that occurs after the thermal quench. The way to mitigate disruptions is to inject large amounts of radiating material to dissipate the thermal energy and to strongly densify the plasma to prevent runaway electrons from forming. In general, the material injected turns to gas so that it can be pumped out. The preferred method selected for ITER is shattered pellet injection using cryogenic pellets of hydrogenic and noble species.

The injection of this material when a disruption is detected to be coming will cause a rapid increase in vessel pressure as much as 1 mbar if all the material available to ITER is used. This will impact cryopumps if they are the primary pumps since the valve in front of the pumps are much too slow to close on the milliseconds time scales of a disruption mitigation event. Other pump types such as super permeable membranes would just quit pumping and a roughing pump system would need to be switched in to pump out the resulting gas. The normal path of exhaust gas to recirculate and feed the fuelling system would need to be switched off so that the mitigation material would not get entrained into the fuel cycle. Alternative non cryogenic pellet mitigation material injection that has been proposed for faster time response but leads to dust formation, which needs to be handled somehow ($T_2$ retention and dust flow).

The pump out by the roughing system would need to cope with 0.1–1% T contained in the gas from the DT in the plasma and any cryopumps in operation that immediately regenerate. Technology for roughing the vessel has been somewhat developed for ITER (Roots pumps backed by screw pumps). This technology would need further improvement to handle the breakdown of oil and keep oil from entering the process gas. Alternatively, a liquid metal ring pump developed by the Karlsruhe Institute of Technology looks feasible if Hg is allowed to be used in the fuel cycle. The time to pump out the gas to the level needed for restarting the plasma ($< 10^{-5}$ mbar) would likely take multiple hours since up to 1000 bar–L of gas will need to be removed and processed to remove the T. This reprocessing could be done over a longer period of time provided sufficient ballast volume to contain the gas is available.

The mitigation material injected for ELMs and disruptions has significant impact on the exhaust pumping systems for an FPP vessel and can also impact divertor detachment control and the time to restore operating conditions and reprocessing load. If $D_2$ is used for ELM pacing the this will lead to $D_2$ rich fuel in DIR loop that will need to be blended with $T_2$. This is unsustainable for long durations and thus DT would need to be used for this application the same as for fuelling pellets. The space needed for disruption mitigation also affects the area for breeding blanket modules and thus needs to be minimized. Thus, there are significant design and technical challenges to the fuel cycle for these transients and an ELM and disruption free FPP is really needed.

# SESSION III: TRITIUM BEHAVIORS AND DEMO FUEL CYCLE

## EU–DEMO FUEL CYCLE OPERATION MODELLING AND DESIGN

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### 1. INTEGRAL ASPECTS

Design and optimization of the fuel cycle in early design phases is driven by the objective of minimizing tritium inventories held up in the processing systems and piping. The EU–DEMO Fuel Cycle can be assessed by a functional view in terms of the requirements for tritium processing in its loops and systems. There are four primary tritium processing loads:

- Circulation of the machine gas throughput;
- Isotopic rebalancing of the machine gas throughput;
- Processing of tritium bred in breeding blankets;
- Recovery and processing of tritium from non–primary pathways.

The EU–DEMO fuel cycle accommodates these processing streams in its loop–based architecture. The tritium processing load can be expressed by a fuel cycle performance metric for each processing load that reflect how the underlying function determines the required tritium processing load in the affected systems.

#### 1.1. Machine Gas Throughput

The machine gas throughput is provided to the torus by systems for pellet injection and gas injection and extracted from the torus by its vacuum systems. Hydrogen isotopes are separated from non–hydrogens by the fuel separation system and exhaust processing system. The total amount of fuel that needs to be provided to and extracted from the machine is derived from the fuel demand of the plasma core while also considering the gas contributions stemming from non–ideal fuelling efficiencies and fuelling contributions for maintaining divertor detachment or edge located mode pacing. The amount of deuterium–tritium (DT) burned in the plasma core is approximately less than 1% of the total machine gas throughput. This is then also reflective of the helium ash content in the exhaust gas. By improving the fuelling efficiency, the tritium throughput and correspondingly inventories in affected systems can be reduced.

#### 1.2. Fuel Rebalancing

One of the key aspects of a DEMO will be the operation at a fixed operation point and plasma scenario which will be fuelled with an optimal D/T ratio as required by the plasma (while correcting for any isotope effects in the transport from the fuelling systems to the core). In order to maintain this isotope ratio any disproportionate sinks or sources of the fuels isotopic constituents need to be compensated. Deuterium and tritium are burned at equimolar rate (neglecting D–D reactions), allowing for adjustments proportional to the burn–up without necessitating isotopic separation. Next to deuterium and tritium, the protium content in the fuel needs to be managed as well. In order to achieve a steady state protium content in the fuel that is below the acceptable limit the same amount of protium that enters into the fuel needs to be removed continuously. Protium removal is achieved by processing a bleed stream of the circulated machine gas throughput in the Isotope Rebalancing and Protium Removal system backed up by the Isotope Separation System. The amount of protium that can be removed is dependent on the total amount present in the bleed stream (and consequently also the circulated fuel). Thus, by allowing for higher protium levels or minimizing protium ingress, the bleed stream and tritium inventories in the associated systems can be reduced.

### 1.3. Tritium Extraction

The tritium processing load that arises from tritium breeding is fairly constant for a power plant of given fusion power and only varies with the tritium breeding ratio. Most current technology selections for tritium breeding and tritium extraction require the use of purge gas and hydrogenic doping agents. The latter necessitate the isotopic separation of the extracted tritium, incurring significant tritium inventories in the Isotope Separation System. Reductions of the fuel cycle tritium inventory can be achieved if tritium can be extracted in a composition with greater or equal tritium fraction as required for fuelling and a protium fraction below the allowable fuel limit.

### 1.4. Tritium Recovery

Tritium may enter secondary enclosure or fluids by permeation or leakage. In order to combat the build–up of tritium these areas are served by detritiation systems, such as exhaust detritiation, glove box detritiation, and coolant purification. The overall rates at which tritium is lost from the primary system is subject to stringent minimization by the system design and consequently the lowest of all tritium processing loads. For the optimization of the fuel cycle these contributions are nevertheless of great interest, as the associated process paths are energy intensive. Employed technologies rely on the conversion of airborne tritium to water, requiring additional water detritiation to recover tritium from its oxide form and subsequent isotope separation.

## 2. FUEL CYCLE DYNAMICS

The dynamics of the inner loops of the EU–DEMO fuel cycle are governed by the fuelling pattern of its fuelling systems (pellets, or pellets and gas injection/puffing), which in turn act on the requested fuel demand of the plasma. Gas injection can feature significant delay times depending on the location of valves and employed line size. For the design and optimization of these systems an understanding of the required fuelling patterns, especially during plasma ramp–up and ramp–down is required.

A second driver of the fuel cycle dynamics in its inner loops is the dwell phase and associated dwell pumping. The achievable dwell pressure is limited by the outgassing rate in the plasma chamber. During plasma operation species in the plasma may be implemented into the first wall or adhere to it. Next to fuel species and impurities generated therefrom, dissolved protium is always present in metal systems and will readily outgas from hot surfaces at low pressures.

During the dwell phase, alterations in the composition of the residual gas species in the plasma chamber will occur. Outgassing will predominantly increase the share of hydrogenic species while the fraction of plasma enhancement gases and helium will drop significantly due to the lack of a source term.

If very low concentrations of inert species and impurities are desired for the start of the next discharge, purging the plasma chamber with DT gas is an available option. This will result in lower partial pressures of all species not actively injected at the cost of an overall higher pressure at the end of the dwell time.

The inner loops of the EU–DEMO fuel cycle operate fully continuous not relying on the use of metallic getter beds or other discontinuous storage methods. This means that all pressure oscillations occurring in the plasma chamber and its port areas during burn/dwell cycling need to be buffered in the gas collection and buffering system to guarantee stable conditions for the composition control systems and fuelling systems. In order to minimize these oscillations, dwell bypass streams are employed keeping all areas at constant pressures that are not required to be evacuated for dwell pumping. The size of the employed buffer vessel then needs to be large enough to host the evacuated amount of gas while staying within the designated pressure ranges.

## TRITIUM MIGRATION PREDICTIONS AND PATHWAYS IN THE FUSION CORE

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Fuelling, exhaust, breeding, and processing of large amounts of tritium is one of the significant technical challenges facing future deuterium–tritium (DT) FPPs. Such FPPs will burn tritium at a rate of ~152 g/GW–d, and therefore need to produce it in the breeding blanket at the same or higher rate in order to close the fuel cycle. While the tritium throughput in the "inner" (plasma fuelling/exhaust) fuel cycle needs to be much (~20–200x) higher due to the low burn fraction in the plasma, tritium produced in the blanket is nevertheless a particular safety concern due to the desirability of high coolant temperatures in electricity–generating plants, and the propensity of tritium (as other hydrogen isotopes) to permeate through metals and other structural materials at high temperatures.

While successful closure of the DT fusion fuel cycle requires that tritium losses via this permeation mechanism remain small, safety and environmental considerations are typically a more stringent limitation. In the United States, limits on tritium releases are determined by the resultant radiation dose to a maximally exposed individual offsite, not to exceed 1 mSv/yr from all sources during normal operations (0.1 mSv/yr in the Department of Energy Fusion Safety Standard). It therefore depends on characteristics of an FPP site; a generic ITER site considered in the US Fusion Nuclear Science Facility (FNSF) study would reach the Department of Energy (DOE) limit at a release rate of 0.29 g T/yr.

Predicting such permeation loss rates is an important part of FPP design and safety analysis, and codes such as the Tritium Migration Analysis Program (TMAP) have been developed for this purpose. Such codes can be used to build a coarsely–modalized model of an FPP and its ancillary systems that tracks tritium migration through each of them and computes their tritium permeation rates. Key permeation paths include:

— Through structures separating breeder and coolant zones within the blanket itself, if these are separate (i.e., in a dual–cooled design);
— Through ex–vessel primary piping in both the breeder/purge gas and coolant loops;
— Through primary heat exchanger surfaces to secondary coolant loops;
— Through secondary system piping to subsequent heat transfer systems or the environment.

Key transport phenomena include diffusion, where the tritium flux $J$ is driven by a concentration gradient $\partial C/\partial x$ with diffusivity $D$ as the proportionality constant $\left(J = -D(\partial C/\partial x)\right)$, and solution, where concentration at a solid surface is proportional (via Sieverts' constant $K_S$) to the square root of the tritium partial pressure in the contacting fluid $\left(C = K_S\sqrt{P}\right)$. Combining these (in a 1D cartesian geometry) gives the classical permeation flux, $J = DK_S\left(\sqrt{P_1} - \sqrt{P_2}\right)/x$, where the product $DK_S$ is known as the permeability. A more general boundary condition treats competing dissociation (rate constant $k_d$) and recombination (rate constant $k_r$) processes at the interface, $J = k_d P - k_r C^2$. This reduces to Sieverts' law when diffusion is rate limiting, but results in a permeation flux linearly dependent on pressure $(J = k_d P/2)$ when surface processes themselves are rate–limiting. In flowing liquids, turbulent or conventional diffusion may be rate–limiting relative either of these phenomena, in which case a concentration gradient exists between the centreline or bulk flow concentration $(C_0)$ and the near wall concentration in the liquid ($C_1$); the permeation flux ($J = k_T(C_0 - C_1)$) is determined by a mass transport coefficient ($k_T$) which can be determined from analogous heat transfer correlations.

Permeation loss calculations for the FNSF design predicted a baseline permeation loss rate of 6.18 g/y, primarily through primary system PbLi piping. This exceeds the desired 0.29 g/y generic site DOE limit, which occurred despite a number of attractive design features that were shown to have reduced this significantly. Foremost among these were a concentric hot/cold leg piping configuration, in which permeation out of the hot leg (FPP exit) simply entered the surrounding annular cold leg; most of this tritium is then swept directly back into the blanket. High efficiency tritium extraction systems were also shown to be important, and a hypothetical vacuum permeator design achieving 95% extraction efficiency was outlined.

Predicted permeation rates in excess of desired limits in the FNSF design emphasizes the need for robust barriers to tritium permeation. Ceramic oxide coatings have long been investigated for this purpose; these show impressive performance in the laboratory (~1000x or more permeation reduction) but significantly degraded performance when tested in FPPs. While this remains an important R&D need, reliability of safety systems will in the meantime require more robust solutions, e.g., locating heat transfer piping and equipment in sealed and actively detritiated enclosures. A similar but more compact solution would be achieved by enclosures that conform tightly to piping, e.g., "guard pipes" as illustrated in Fig. 1. In the FNSF, a slow helium sweep gas (e.g., 0.1 m/s at low pressure) in this annular region surrounding the concentric hot and cold legs was able to reduce permeation losses by over two orders of magnitude without detrimental effects on heat transfer.

| He purge velocity (m/s) | Outer pipe temp (C) | Inner pipe temp (C) | Heat loss (W) | Tritium loss (g/y) |
|---|---|---|---|---|
| N/A | - | 417 | - | 4.21 |
| 0.1 | 94 | 416 | 65 | 0.014 |
| 1.0 | 353 | 414 | 2,083 | 1.19 |
| 10.0 | 350 | 414 | 21,350 | 0.58 |

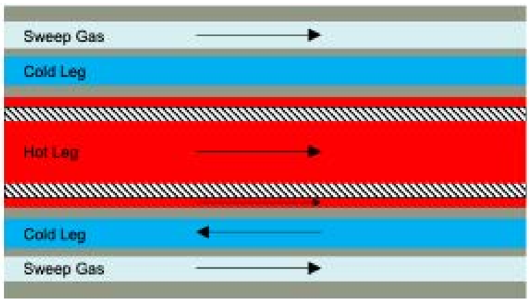


*FIG. 1. FNSF guard pipe design (right) and impacts on total tritium permeation for a range of flow rates (left).*

Finally, some comments are in order on the phenomenon of trapping; other contributions to this meeting highlighted the apparent lack of any observable saturation of trapped tritium in plasma–exposed surfaces, which has implications for total tritium inventory and potentially impacts the ability to close the fuel cycle. In the FNSF study, minimal plasma–driven permeation to the first wall coolant was predicted relative to that permeating from the breeding blanket. A wide variety of similar such analytical studies were surveyed, each looking at the impact of various factors such as power, temperature, parameter uncertainties, surface condition, etc. Predictions of this permeation rate consequently varied widely, from insignificant (μg/d) to significant (grams/d) levels. This remains a topic in need of some further experimental and analytical scrutiny.

## TRITIUM MIGRATION PATHWAYS AND ROLE OF THE COOLANT PURIFICATION SYSTEM

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Central requirements for DEMO are the production of net electricity and the operation with a closed fuel cycle. Thus, the machine will be equipped with a primary and a secondary coolant loop for heat extraction and energy conversion, and with a breeding blanket and a fuel cycle respectively for tritium production and processing. Such configuration inevitably opens a path to tritium migration because heat removal and tritium production occur both in the blanket region where the presence of high temperatures, large metallic surface areas and high tritium concentrations facilitate the permeation from the blanket to the primary coolant. Once in the primary coolant circuit, tritium permeates either across the coolant tubes and into the secondary coolant loop from which it easily reaches the external environment. A schematic view of this tritium migration issues is illustrated in Fig. 1. Obviously, the entire phenomenon is of safety relevance for DEMO and for the future fusion power plants, therefore a series of activities have been performed and many are still ongoing with the intent to identify and implement effective strategies to mitigate the tritium permeation.

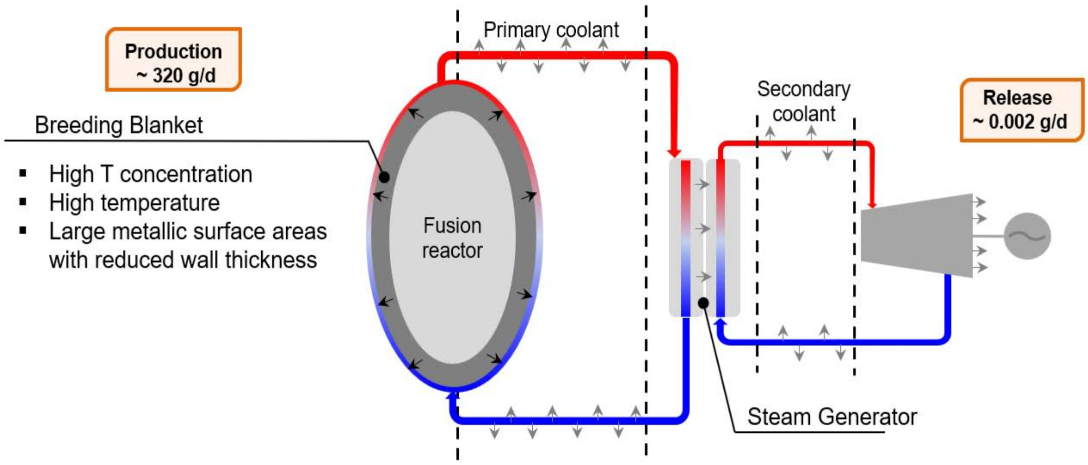


*FIG. 1. Schematic view of the tritium migration issues (rearranged from Ref. [1] with permission).*

This contribution has illustrated the development of tritium migration models at system level in DEMO, the status of the activities regarding the mitigation strategy, such as the development of anti–permeation barriers and coolant purification system, and the analysis performed in the frame of the so–called key design integration issue (KDII) #2 in which the tritium permeation has been evaluated with respect to several operating limits.

One of the first model treating the issue of tritium migration at system level in DEMO was the fusion–devoted tritium permeation code, FUS–TPC, code developed in MATLAB and inspired to the Sodium–Cooled Fast Reactors Tritium Permeation Code [2,3]. FUS–TPC represented the first example of a unified code for three different blanket concepts: helium cooled lead lithium, helium cooled pebble bed and water–cooled lithium–lead. Therefore, this provided the possibility to perform sensitivity analyses in the three blanket concepts. Main outcomes of such analyses can be summarized in three different groups. First the activity allowed to homogenize the inputs, the assumptions and the parameters used in the design of the three blankets. Second, the work identified some relevant aspects related to materials, such as:

- The impact of the large uncertainty values of the LiPb solubility constant on the evaluation of tritium permeation and inventory;
- The need to develop effective anti–permeation barriers.

Finally, about the tritium processing, the results of the analyses clearly showed that:

- Tritium extraction from blanket needs to be performed with an efficiency higher than 80%, this is a very demanding requirement especially for the water–cooled lithium–lead concept;
- The primary coolant requires a dedicated system, named coolant purification system (CPS), for tritium recovery by processing a certain coolant fraction.

The technologies to be adopted in the helium CPS can rely on the ones developed for fission, in the high temperature gas cooled reactor, and in ITER, while for the water CPS it was clear that the amount of coolant routed in the CPS needs to be reduced to avoid large water detritiation systems which are very energy demanding.

In the frame of the EUROfusion programme (FP8), the new reference tool for the tritium migration analysis at system level in DEMO became EcosimPro, developed by the Centre for Research on Energy, Environment and Technology (CIEMAT), that is an object–oriented modelling software. With this code several improvements are made compared with the FUS–TPC. For instance, the model now considers:

- The integration of the secondary coolant included the intermediate heat exchanger and the steam generator;
- The tritium permeation from pipes into rooms useful to assess the expected tritium concentration in different area;
- The presence of H2, HT and T2 and their co–permeation even if such phenomenon requires experimental validation;
- The effect of doping agents like H2 and/or H2O addition inside the coolant;
- An updated design of the entire primary heat transfer system.

Regarding the mitigation strategy, activities related to the development of anti–permeation barriers and pre–conceptual design of the coolant purification system have been presented. Concerning the anti–permeation barriers, the research is mainly dedicated to the deposition of alumina coatings via different deposition techniques. For instance, by using the plasma laser deposition, dedicated permeation experiments have measured a permeation reduction factor up to 1000. About this topic important steps to be performed are related to:

- The development of deposition techniques able to cover complex shapes;
- The validation of the effectiveness of the coatings under neutron irradiation.

Referring to the CPS the pre–conceptual design has identified suitable solutions and processes either for helium or water coolant. The reference case for He–CPS is based on the scale up of technologies used in ITER, while the alternative solution relies on the use of novel non–evaporable getter material which are characterized by relatively high hydrogen embrittlement limit. For water CPS an initial activity was necessary to identify a suitable bypass value. In fact, tritium removal processes from water are very energy demanding therefore water amount to be routed inside the CPS needs to be minimized. For such reason, anti–permeation barriers able to provide a permeation reduction factor higher than 100 are necessary. The processed identified for water CPS is water distillation, the preliminary design of such column has been presented in terms of dimensions, energy consumption and tritium inventory.

Last part of this contribution was dedicated to show the activity performed in the frame of the KDII #2 which was carried out to addresses some specific issues of the design of breeding blanket and ancillary systems related to the use of helium or water as a coolant for the blanket and impact on the overall plant design. In this frame the methodology used for the tritium migration issue was the follow:

- Collect input from different plant designers (breeding blanket, tritium extraction and recovery, coolant purification system, anti–permeation barriers, primary heat transfer system, power conversion system, etc.);
- Identify a reference scenario and suitable ranges for the sensitivity analysis;
- Perform tritium simulation analysis at system level with the most updated plant architecture;
- Analyse the simulation results with respect to some operating limit such as:
  a. Release in tokamak building;
  b. Threshold to activate the detritiation system;
  c. Threshold for nuclear classification of the secondary/intermediate systems according to the Equipment Sous Pression Nucléaire (ESPN) Directive [3].

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## TRITIUM IN CHAMBER MATERIALS, TRAPPING AND RELEASE

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Tritium trapping in the camber materials reduces efficiency of fuel recycle of an FPP using deuterium (D) and tritium (T) as fuels. In addition, tritium trapped in the chamber materials can be a source term of tritium release during potential accident of a fusion power plant. DT fusion reactions generates high energy neutrons and helium ions, as depicted in Eq. (1) below:

$$D + T \rightarrow He\ (3.5\ MeV) + neutron\ (14\ MeV) \tag{1}$$

The chamber material becomes radioactive after the operation of an FPP due to neutron–induced nuclear transmutation. Therefore, decay heat is generated in the chamber materials. If coolant is lost under accidental conditions, the release of trapped tritium may occur due to temperature increase by decay heat. Hence, the trapping of tritium needs to be well understood and minimized.

ITER will use beryllium (Be) for first walls in the main chamber and tungsten (W) in the divertor region. The discharge duration in ITER will be ~1000 s. Under such conditions, co–deposition with sputtered material (mainly low–Z Be) can be the main mechanism of tritium retention. The fuel retention in Be and W walls has been examined in ITER–like wall experiments in JET (JET–ILW) [1]. On the other hand, the current JA–DEMO design uses W for both first wall armour layer and divertor to generate steady state plasma [2]. In this case, deep penetration of tritium into W by diffusion is possible. In addition, displacement damages are accumulated in the material. Hence, trapping at the bulk of W can make significant contributions to tritium retention in addition to co–deposition. In the presentation, tritium trapping in ITER–type device and DEMO was discussed by reviewing the publications from JET–ILW experiments and those on D retention in neutron–irradiated W.

JET has performed ITER–like wall experiments with Be limiter tiles in the main chamber and bulk W and W–coated carbon–fibre–composite (CFC) tiles in the divertor region since 2011 [1]. Three experimental campaigns with DD discharges were performed in 2011–2012 (ILW1), 2013–2014 (ILW2) and 2015–2016 (ILW3) [1]. Be limiters have a castellated structure for better durability against heat load [3]. The bulk W tiles have lamellae structure. D retention on the plasma–facing surfaces and in the gaps were examined using various experimental techniques including ion beam analysis, secondary ion mass spectrometry and thermal desorption spectrometry (TDS) [1]. A small amount of tritium was formed by DD fusion reactions, and distribution of tritium was examined using imaging plate technique, β–ray induced X–ray spectrometry, TDS and full combustion technique [4]. Co–deposition with Be, C, O and other metallic impurities was the main mechanism of fuel retention in JET–ILW [4,5]. The concentration of fuels in deposition layers increased with increasing C content [4]. The co–deposition occurred also in the grooves of castellated structure of Be tiles [3] and gaps in the W lamella structure [4]. Thermal desorption from W–coated CFC tiles and Be tiles peaked at 400–600 °C and continued up to >800 °C [6,7].

To understand fuel penetration into bulk W under DEMO–like conditions, W samples were irradiated at High Flux Isotope Reactor (HFIR), Oak Ridge National Laboratory, US [8] and Belgium Reactor 2 (BR2), SCK·CEN, Belgium [9] and exposed to D plasma using linear plasma machine called Tritium Plasma Experiment (TPE), Idaho National Laboratory, US [10] and Compact Divertor Plasma Simulator (CDPS), Tohoku University, Japan [11]. Deep penetration of D into neutron–irradiated W (~100 μm) was observed after plasma exposure at 473–773 K for 60–400 minutes [12]. D retention was proportional to the square root of exposure time [12], as expected from the model proposed by Wampler and Doerner [13]. The concentration of D was reached [D]/[W] ~ 0.01 after neutron irradiation to 0.3 dpa and plasma exposure at 473 K [14]. Steep temperature gradient is expected to be developed under plasma exposure of W monoblocs in DEMO; the plasma–facing surfaces of W monoblocs reach very high temperature due to heat flux from the plasma, while the materials around cooling pipes are kept at around ~573 K depending on coolant. The calculation using diffusion analysis code TMAP [15] showed that fuel particles diffuse to the relatively cold region around cooling pipes during exposure to steady state plasma and trapped there. In other words, the cold regions around the cooling pipes can be the largest reservoir of tritium in the vacuum chamber. Neutron irradiation under hydrogen gas atmosphere is in progress [16]. Helium seeding in the D plasma resulted in formation of dense, nano–sized bubbles beneath the surface and significant reduction in D retention [17], as observed for non–irradiated W [18]. Addition of gases like N, Ne, Kr, Xe etc. to the plasma has been proposed for divertor detachment and edge located mode/disruption mitigation. These elements may remove helium implanted layers by sputtering of W. The stability of the thin He implanted layer during operation of DEMO should be carefully evaluated. FPP mitigation of irradiation effects has been observed by addition of Re and Cr in W.

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## TECHNIQUES OF TRITIUM DECONTAMINATION ON PLASMA–FACING WALLS IN DEMO

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In fusion DEMOs, tritium (T) decontamination scenario before maintenance begins is a key issue. Hence, it is important that T decontamination under vacuum conditions before opening the plasma vacuum vessels. Currently, JA–DEMO team has not yet determined the allowable value of residual T in the vacuum vessel, but it is necessary to indicate a candidate T decontamination technique. Furthermore, the construction of a short–term maintenance scenario that includes the T decontamination process after plasma operation is stopped is also important for fusion DEMOs.

Three kinds of candidate techniques of T decontamination are considered in vacuum conditions:

— Temperature control by decay heat and baking/cooling;
— Active wall conditionings, such as glow discharge, ion cyclotron wall conditioning, and electron cyclotron wall conditioning;
— A selection of working gas and vacuum pressure.

An isothermal desorption method is required for the tritium decontamination on the in–vessel wall with the limited temperature range in DEMO. In JA–DEMO, the upper limitation of blanket and structural materials is 300–350 degrees due to the coolant by water. In the second topic, results of surface analysis the specimen exposed to the ion cyclotron wall conditioning (ICWC) are shown. The top surfaces less than 10 nm have damage as an indication of particle interactions by ICWC. And no bubbles due to lower temperature and fluences to produce bubbles. Such thin layer damages are an advantage of ICWC. For the third topic, tritium removal by isotope exchanges between $H_2O$ and HTO are shown. When specimens with tritium were exposed to air, quickly surface tritium was removed by the isotope exchanges. Under Ar gas condition, the tritium desorption without effect by $H_2O$ was observed. For tritium recycling and reusable, pumped tritium in the vacuum vessel is better before the vacuum vent of the vessel. However, the number of these data is limited due to experimental conditions, and a combination of a)–c) is important for effective tritium decontamination. In addition, monitoring methods of retained tritium on specimens are shown. Monitoring requires processing by a remotely operated system under vacuum conditions in the vessel.

Tritium decontamination and monitoring methods are related mainly to the hardware designs of DEMO. Hence it is required to make a list of requirements in the early phase.

**EXPERIENCE WITH TRITIUM RETENTION AND REMOVAL IN JET–DTE2**

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JET is the largest tokamak in use and currently the only one capable of handling radioactive tritium (T). It operates since 2011 with the ITER–like wall (ILW), which consists of a tungsten (W) divertor and a beryllium (Be) main chamber. After a decade of enhancements, tritium–related upgrades, and training [1], as well as preparatory experimental campaigns in deuterium (D), hydrogen (H) then T, JET has operated the second deuterium tritium experimental campaign (DTE2, after DTE1 in 1997 with carbon–based plasma facing components, PFCs), aiming at demonstrating high DT fusion power for 5 sec. with the ILW and improving predictions for ITER DT [2].

About 1 kg tritium has been supplied to five tritium injection modules and two neutral beam injections (NBIs) boxes by the JET Active Gas Handling System (AGHS) during the T and DT campaigns. The latter were conducted on cycles of three–four weeks of operation followed by one week of tritium reprocessing and accounting. A global gas balance was performed to assess the in–vessel T retention after one day of operation [3] by subtracting the amount of actively pumped neutrals, as determined after operation by pressure–volume–temperature (PVT) and residual gas analysis (RGA) of the gas released from the cryopump regeneration to 80K, from the amount of injected T through the gas injection systems. Long–term outgassing experiments completed the study, evidencing a faster decay of the T partial pressure compared to D. No major change on the so–determined long–term T retention compared to previous data in D could be evidenced. The global T accountancy by AGHS is however still in progress.

After DTE2, a sequence of complementary fuel recovery methods was successfully operated to remove T from the PFCs. It consisted of baking the main chamber under vacuum at 320°C, followed by isotopic exchange with ion cyclotron wall and glow discharges in $D_2$ at this temperature, preferentially accessing T retained in the main chamber. This ensured efficient T removal while preventing production of 14 MeV neutrons. 20 seconds long diverted plasmas with up to 16 MW NBI and ion cyclotron resonance heating power were then operated in different magnetic configurations with the main chamber at 200°C. A particular configuration with a Raised Inner Strike Point on the inner divertor tiles had been developed and applied in order to target in particular the inner divertor baffle region, where the majority of the retained fuel is known to reside in thick Be deposited layers [4].

The isotopic ratio T/[H+D+T] through the clean–up the sequence was measured in the exhaust using mass spectrometry and with the JET sub–divertor optical Penning gauge, as well as with high resolution Optical Emission Spectroscopy from the H, D, T Balmer alpha line intensities in ICWC or diverted plasmas. However, the sensitivity of these diagnostics doesn't allow measurements below 1% T in D. Moreover, deconvolution of mass spectra in presence of three hydrogen isotopes is cumbersome. Therefore, as soon as diverted plasmas could be operated in the clean–up sequence, the isotopic ratio T/[H+D+T] was inferred from neutron spectroscopy, using beam–plasma reactions either during short blips or long phases of D–NBI. Hence, T/[H+D+T] was finally found to be ~$10^{-4}$ in H–mode D plasmas, well below the 1% target set by the allocated $5 \cdot 10^{19}$ 14 MeV fusion neutrons budget for the following D campaigns. Monitoring of the isotopic ratio was pursued in the D campaign following the clean–up sequence and evidenced further decrease of the plasma T content.

In total, about $4 \cdot 10^{23}$ T atoms were removed from JET PFCs, among which ~45% by baking and 50% by ICWC and GDC, the remaining 5% being removed afterwards from PFCs already depleted from T by limiter and diverted plasmas. Similar results had been obtained in a qualification experiment prior to the T campaign, where D was removed from PFCs using baking, as well as ICWC, GDC and plasma operated in H [5]. Though removal by baking seems to be less effective for T than for D, ICWC and GDC promote extra removal in both cases.

Access to T buried in co–deposited layers at the upper part of the inner divertor was clearly evidenced from the increased neutron rate and elevated surface temperatures above 1200°C in plasmas that had the inner strike point raised onto this area.

A similar Raised strike point scenario has also been investigated in ITER (10 MA deuterium L–mode with 20 MW Electron Cyclotron Resonance Heating). The temperature of ITER monoblocs at the strike point was estimated using heat loads calculated by SOLPS–ITER and the measured thermal resistance of JET deposits as assessed from infra–red measurements in plasmas with raised inner strike point. Although the ITER divertor will be actively cooled, it is estimated that Be co–deposits on the inner divertor may be heated up to 800°C using raised strike points, efficiently allowing for fuel outgassing.

Still, to fully the likely re–deposition of released material needs to be assessed in JET, especially in the new raised inner strike point configuration, which has never been simulated by edge plasma and impurity transport codes. For this purpose, high fidelity numerical simulations are on–going to assess Be and T migration using the Monte–Carlo erosion/migration code ERO2.0 [6].

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## LIST OF ABBREVIATIONS

| | |
|---|---|
| **AGHs** | active gas handling system |
| **CFC** | carbon fibre composite |
| **CPS** | coolant purification system |
| **DIR** | direct internal recycling |
| **DIVGAS** | divertor gas simulator |
| **DM** | disruption mitigation |
| **DTE** | JET deuterium tritium experimental campaign |
| **ELM** | edge localized mode |
| **FNSF** | Fusion Nuclear Science Facility |
| **FPP** | Fusion Power Plant |
| **FUS–TPC** | fusion–devoted tritium permeation code |
| **GDC** | glow discharge cleaning |
| **ICWC** | ion cyclotron wall conditioning |
| **ILW** | ITER–like wall |
| **ISS** | isotope separation system |
| **KDII** | key design integration issue |
| **MHD** | magnetohydrodynamic |
| **NBI** | neutral beam injection |
| **PEG** | plasma enhancement gas |
| **PFC** | plasma facing component |
| **PFR** | private flux region |
| **PLS** | pellet launching system |
| **PMI** | plasma material interaction |
| **Q** | any hydrogen isotope |
| **RID** | residual ion dump |
| **TBR** | tritium breeding ratio |
| **TDS** | thermal desorption spectrometry |
| **TFV** | tritium–matter injection–vacuum |
| **Z** | atomic number |

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