Fusion Engineering and Design 179 (2022) 113139

Contents lists available at ScienceDirect

Fusion Engineering and Design

journal homepage: www.elsevier.com/locate/fusengdes

The pre-concept design of the DEMO tritium, matter injection and vacuum systems

Chr. Day [a,*], K. Battes [a], B. Butler [b], S. Davies [b], L. Farina [e], A. Frattolillo [e], R. George [b], T. Giegerich [a], S. Hanke [a], T. Härtl [c], Y. Igitkhanov [a], T. Jackson [b], N. Jayasekera [b], Y. Kathage [a], P.T. Lang [d], R. Lawless [b], X. Luo [a], C. Neugebauer [a], B. Ploeckl [d], A. Santucci [e], J. Schwenzer [a], T. Teichmann [a], T. Tijssen [b], S. Tosti [e], S. Varoutis [a], A. Vazquez Cortes [a]

[a] Karlsruhe Institute of Technology, Karlsruhe, Germany
[b] CCFE Culham Science Centre, Abingdon, UK
[c] EUROfusion Consortium, Project Management Unit, Garching, Germany
[d] Max-Planck-Institut für Plasmaphysik, Garching, Germany
[e] ENEA, FSN Dept, Frascati (RM), Italy

ARTICLE INFO

Keywords:
Tritium
Fuel Cycle
Vacuum
Pellet injection

ABSTRACT

In the Pre-Concept Design Phase of EU-DEMO, the work package TFV (Tritium – Matter Injection – Vacuum) has developed a tritium self-sufficient three-loop fuel cycle architecture. Driven by the need to reduce the tritium inventory in the systems to an absolute minimum, 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 extracted 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 paper starts from a functional analysis of the fuel cycle and introduces the results of a technology survey and ranking exercise which provided the prime technology candidates for all system blocks. The main boundary conditions for the TFV systems are described based on which the fuel cycle architecture was developed and the required operational windows of all subsystems were defined. To validate this, various R&D lines were established, selected results of which are reported, together with the key technology developments. Finally, an outlook towards the Concept Design Phase is given.

1. High-level challenges in fuel cycle development for DEMO

1.1. Overall plant requirements and their effect on the fuel cycle

The work package Tritium – Matter Injection – Vacuum (TFV) is within the EUROfusion Consortium responsible for the EU-DEMO (that will be simply called DEMO in this paper) fuel cycle development in line with the European Research Roadmap to the Realisation of Fusion Energy [1] and the DEMO mission requirements. The following subset of guiding requirements with relevance for the fuel cycle have been used:

- The DEMO fusion core shall adopt a tokamak architecture.
- The DEMO plant architecture as well as critical technologies and materials shall be extrapolable to a fusion power plant (FPP).
- The duration of flat top shall be at least 2 hours.
- In Plasma Operation Phase, DEMO shall breed sufficient tritium to fuel its plasma without requiring supply from external sources throughout its planned operational schedule (including all planned maintenance, stand-by and dwell intervals).
- The DEMO plant shall aim to provide a considerable fraction of the tritium required for the start-up of another fusion plant.
- The average amount of electricity during flat top delivered into the grid subtracting the power consumed by all plant systems shall be at least 300 MW.
- The DEMO capital cost and the DEMO total annual operating cost shall be minimised.
- The DEMO plant shall not require the evacuation of any members of the public under any failure or accident scenario.

* Corresponding author.
E-mail address: christian.day@kit.edu (Chr. Day).

https://doi.org/10.1016/j.fusengdes.2022.113139
Received 24 October 2021; Received in revised form 6 March 2022; Accepted 8 April 2022
Available online 23 April 2022

- The DEMO plant shall minimise dose rates in all manned areas in accordance with the As Low As Reasonably Achievable (ALARA) principle.
- The DEMO plant shall be designed for an acceptable decommissioning cost and duration.
- The DEMO plant shall minimize the amount of radioactive waste produced.
- The DEMO plant shall be able to detritiate efficiently structures, fluids and RH equipment.

These requirements were directly translated into the organisation of the work in the Pre-Concept Design (PCD) Phase of the fuel cycle. The above listed set of requirements in this combination is first-of-its-kind. The aspects of tritium self-sufficiency, plant availability and safety for the expected size of DEMO are posing an unprecedented challenge. Due to this novelty, it was decided to build the DEMO fuel cycle architecture not on existing fuel cycles but to develop it from first principles. As will be outlined in the following, all technologies applied have to be feasible, allow start-up and licensing of the fuel cycle and must be economically attractive.

1.2. Ensure technology feasibility

It is trivial to say technologies for fuel processing have to fulfil the foreseen task. Nevertheless, this has to be explicitly mentioned here, as very often suggested technologies reach physics limitations if utilized in other applications. A technology might also become unfeasible if the system size (in many cases coupled with the tritium inventory) exceeds a value where the replenishment of the losses (decay, permeation etc.) would ask for a tritium breeding rate that cannot be provided due to physics and geometrical (available surface for tritium breeding inside the reactor) limitations of the machine. We performed a quantitative technology rating to check and ensure feasibility. To our knowledge, this is the very first time such a structured approach to determining the starting points for architectural development of a fusion fuel cycle, as detailed in this paper, has been undertaken in such a systematic manner. The fuel cycle as described in detail here is the main outcome of the PCD phase and becomes the starting point for the Concept Design (CD) Phase.

1.3. Allow start-up and licensing

The tritium inventory for the fuel cycle start-up is a very important parameter. If this value is very large, the availability of tritium from other global sources might not be sufficient [2, 3]. The effect of tritium availability for fuel cycle start-up is very difficult to extrapolate over several decades, but the main conclusion for our work presented here is that the tritium inventory in the fuel cycle must be minimized.

A large tritium inventory would also mean a potentially large source term. As DEMO is likely to be a nuclear licensed facility, stringent safety requirements have to be fulfilled. This requires an assessment of accident scenarios with a given radioactive inventory; only once this is undertaken to the satisfaction of the regulator, can a license be obtained. In this field, Return of eXperience (RoX) can be expected to be gained from ITER. There is a very strong driver to push the tritium inventory in the DEMO fusion fuel cycle towards the absolute minimum, ideally of the same order as in ITER or less. A discussion of the start-up inventory and the tritium lifetime is presented at the end of section 4.

1.4. Guarantee economic attractiveness

Technologies for the processing of tritium in the fusion fuel cycle are neither new nor unknown. From the past decades, there is definitely experience available but mainly limited to small flows where system size, tritium inventories, scalability towards industrial scale plants and energy consumption played no or only a minor role. Between DEMO and ITER, there is a gap between the amount of tritium that has to be processed in the fuel cycle. The main reason for this gap is (i) the relatively small pulse time and availability of ITER, in particular in the first decade of operation, and (ii) the lack of representative tritium breeding systems at ITER (there will be a factor 10^4 of the tritium needed to obtain self-sufficiency related to the tritium bred by the test blanket modules in ITER [4]). It is theoretically possible to scale the known systems towards DEMO scale, but the price to pay would be a very large footprint potentially leading to very large capital expenditure (CapEx) and excessive energy consumption linked with high operational expenditure (OpEx). This contradicts the stakeholders' requirement of minimized costs and does not allow technology extrapolation towards an FPP. However, even if not all technologies developed in the past can be applied in DEMO, all TFV work is based on the maximum return of experience from ITER, as will be clearly indicated for all possible cases.

This paper is organized as follows. Following this introduction, the main boundary conditions and interfaces are summarized in Section 2. Section 3 will then outline the current system block architecture of the DEMO fuel cycle on loop level, whilst Section 4 will treat in detail each of the systems for which the technology choice will be discussed, selected R&D achievements will be presented and an outline of future work to be performed in the CD Phase will be given. Section 5 will then look into integration aspects and the new facilities planned for technology maturation and demonstration, followed by a short summary in Section 6.

2. Identification of fuel cycle main boundary conditions

2.1. Fuel cycle interfaces

The fusion fuel cycle is normally subdivided into the inner fuel cycle and the outer fuel cycle. The latter comprises the breeding blankets including their Tritium Extraction and Removal Systems (TERS), while all remaining systems belong to the inner fuel cycle [5]. The fusion plasma is the central system that provides the neutron flux to the blankets and defines the exhaust gas throughput and composition to the inner fuel cycle; it also interacts with the first wall and produces a certain tritium inventory there, which has to be considered as additional to that in the fuel cycle. The coolant of the blankets needs detritiation that will be provided by the coolant purification system (CPS); the CPS may also service the coolants of vessel and divertor in which case it is a system that connects the inner and the outer fuel cycle. Moroever, there is a direct interlink between the coolant purification system and the balance of plant.

This holistic conception of the fuel cycle is reflected by the organisation of the work in EUROfusion. Here, several work packages are strongly linked to TFV, namely Breeding Blanket (BB) [6], Safety (SAE) [7], and Balance of Plant (BOP) [8]. In order to allow an efficient working manner, external interfaces have to be defined. In this context, 'external' refers to the work package (i.e. TFV), thereby not violating the holistic view that is taken by TFV in general. Fig. 1 gives a systematic

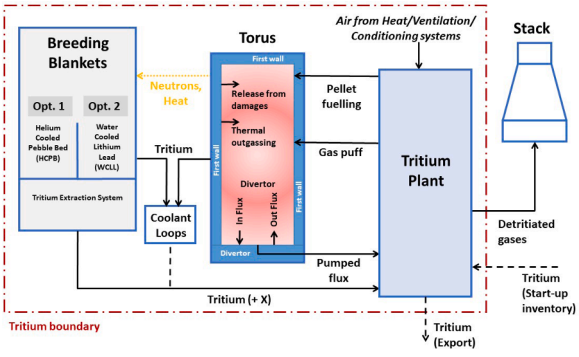

Fig. 1. Interfaces to be considered in the fuel cycle.

overview of the most important interfaces.

The central interface is to the plasma: The charged particles from the plasma core are neutralized on the divertor plates (divertor influx) and some of them are backscattered towards the plasma again (divertor outflux). The difference has to be pumped out by the torus pumping system (pumped flux). The definition of this interface is of prime importance as it defines the fuelling requirement and, hence, also the load on the vacuum pumps and the fuel cycle. This interface is described in more detail in Section 2.2.

Another interface with strong impact on the fuel cycle concerns the influx from tritium breeding. Here, tritium is being bred and extracted from the blankets resulting in a very large mixed gas stream that includes impurities and carrier gas, potentially with doping agents, depending on the chosen blanket concept (Section 2.3). In the EU programme, two most promising blanket concepts were identified, namely the Water–Cooled Lithium Lead concept (WCLL) and the Helium-Cooled Pebble Bed (HCPB) concept where the requirements on tritium extraction and clean-up are very different. In TFV, both concepts are considered from the perspective of the interfaces to the fuel cycle.

In view of interfaces, BB is responsible for breeding and tritium extraction from the blankets. The interface on TFV side is the CPS for the blanket coolant media (water or helium), and the so-called Tritium Conditioning System (TCS), where the gases from TERS are received for further processing. This means that two layouts of TCS and CPS must be developed translating in different architecture variants. Moreover, as DEMO is foreseen to play the role of a component test facility for the breeding blanket, it must incorporate the ability and the flexibility to accommodate for testing at least one type of advanced tritium breeding blanket concept. This means that the fuel cycle may well be required to be compatible with two completely different blanket variants, the down-selection of which is foreseen for the Gate G2, in the second half of the CD Phase [9].

The tokamak and the tritium plant are huge multi-level buildings with a footprint of several 1000 m^2 and house tritium in the kg range and large quantities of hydrogen. It is obvious that nuclear safety plays a major role in the design of this plant and the included subsystems. Hence, there is another important interface with safety (SAE), where accidental scenarios are defined and simulated to demonstrate that tritium releases across the tritium boundary stay within acceptable limits, where 'acceptable' means ALARA below a target value.

2.2. Governing streams in the inner fuel cycle during burn

The governing input stream to the fuel cycle is the machine gas throughput across the divertor, which results from the particle transport via the plasma edge and the gas outside the plasma core flowing through the scrape-off layer.

To enable stable plasma pulses over long periods of time the tokamak has to operate in steady state conditions with regards to its gas throughput, essentially requiring the pumping of all matter injected into the torus vessel (considering any sinks and sources in the torus itself). During flat-top operation, this gas throughput is primarily given by the requirement of keeping the plasma core density at the specified level as well as limiting the helium ash concentration in the plasma core below a certain limit (of the order of 7% [10]), together with the losses associated with pellet production and transport into the core, and resulting from the density gradient of the plasma edge at given diffusivity. Moreover, there are additional throughput contributions that arise from plasma control purposes (Gas puffing & ELM pacing pellets if necessary). As no finalized plasma scenario exists at the moment, the fuel cycle was developed assuming ranges for the individual contributions, as discussed in [11, 12]. In sum, these contributions yield a bounding envelope of 265 – 430 $Pa{\cdot}m^3/s$ of gas that has to be exhausted from the torus. This explains why this paper is not organised around a single design point but describes more principal considerations that hold for a broader design space.

Although unburnt fuel with a molar D:T ratio of approximately 1 to maximize the fusion power gain is clearly the dominating fraction in the exhaust gas, it has to be well understood that the tokamak exhaust is a multi-species mixture of complex nature. These will all appear in some quantity in the exhaust:

- All six hydrogen isotopologues (H_2, HD, HT, D_2, DT, T_2) (in this paper we use the collective term Q2 if we want to speak about such hydrogenic species in general, different from the three pure isotopes protium (H_2), deuterium (D_2) and tritium (T_2)).
- Plasma Enhancement Gases (PEG) comprising inert noble gases necessary to enhance radiation from the core plasma and divertor. The current DEMO reference scenario [10] assumes Xe as core radiator and Ar for divertor seeding. Nitrogen is not included to avoid associated corrosion problems and substantial additional complexity that would arise from the formation of tritiated ammonia and the corresponding need to add a chemical removal stage to deal with this issue. Additionally, PEGs may be necessary to allow coupling of ion cyclotron resonance heating (ICRH) if this will be exploited at DEMO. Finally, they may be also utilized for disruption mitigation or runaway electron suppression, and for fast plasma shut down.
- ^{4}He ash from the fusion reaction.
- ^{3}He from the decay of tritium. Due to its scarcity and growing importance, for example potential upcoming needs in quantum computing, it is under consideration to export this as a commercial product.
- Q2O, CQ4, CO/CO_2 and the like from interaction of tritium with walls.
- Activated isotopes and decay products, mainly from plasma enhancement gases.
- Air and water which leak into the fuel cycle loops.
- Protium and other matter leached out of the tokamak walls, blanket first wall and limiters.

While the hydrogenic and PEG flowrates depend much on the plasma scenario, the helium flowrate is given by the defined fusion power of 2 GW, which, via the number of fusion reactions needed to provide this, directly translates in $7.2{\bullet}10^{20}$/s which is equivalent to a flowrate of 2.6 $Pa{\cdot}m^3/s$.

In order to design the divertor pumping systems, the pressure at the location where the pump ducts connect to the divertor ring has to be known. The pressure distribution in this subdivertor region again depends on the chosen divertor geometry (design of targets, existence and shape of the dome), magnetic configuration and detachment state. To tackle this strongly coupled problem, a workflow has been established that extracts the subdivertor information directly from the flux information at the separatrix as found from a plasma code (such as SOLPS) [12, 13]. By this innovative particle-based approach (based on the Direct Simulation Monte Carlo method such as implemented in DIVGAS [14]), the fuel cycle is self-consistently coupled to the chosen plasma scenario, which has not been the case in the past. This approach turned out to be very helpful in finding solutions to the key design integration issue 7 Design and feasibility of a pumping concept based on tritium direct recycling´ [15]. Note that the requirement on to be pumped fluxes in a FPP is significantly more constrained than in a physics fusion device. This is due to the fact that the given fusion power translates directly in a given throughput (=injected particle rate), and hence divertor pressure and pumping speed cannot be treated independently, while in today's devices the injected particle rate is considered to be an actuator which can be changed if the divertor plasma requires this.

2.3. Governing streams in the outer fuel cycle during burn

The outer fuel cycle serves to efficiently extract the bred tritium from the blankets and to transfer this to TCS. The required amount of tritium

to be produced, irrespective of the chosen blanket concept, is directly related to the fusion reaction that consumes the tritium. The 2 GW fusion power can thus be converted into a mass flowrate of 0.32 kg/d of tritium (full power day). It is to be noted that this flowrate is by more than one order of magnitude lower than the tritium contained in the torus exhaust gas input to the inner fuel cycle.

Even under steady state conditions, tritium is being trapped via interaction with walls, so that the blankets are designed considering a certain Tritium Breeding Ratio (TBR) above unity, defined as number of tritons produced per neutron received from the fusion reaction. The working number was 1.05 [16]. As shown in Section 4.15, some models predict a lower acceptable limit of the order of 1.025. At lower TBRs, the decay reduces the tritium more quickly than that being produced by the blankets.

For the design of the fuel cycle, it has to be considered that the tritium will be purged out of the breeder material using a helium stream in different flowrates for HCPB and WCLL configurations (if gas-liquid contactor technology is applied, the current lead technology candidate for WCLL [6]). This purge gas stream will then be further processed in a dedicated tritium extraction system for which there exist several technology variants [17, 18] that end up in significantly different flowrates and compositions. This is why a TCS is foreseen which treats the incoming streams to form a defined input to the inner fuel cycle.

2.4. Dwell time requirements

Besides the requirements on the fuel cycle arising during plasma pulse, additional requirements originate during dwell times. Dwell phases are needed in order to load the central solenoid and to provide the necessary start vacuum for the plasma discharge. Obviously, in order to allow an efficient and economically attractive operation scenario of a pulsed fusion power plant, the length of the plasma pulses shall be maximized and the dwell time in between the pulses must be reduced to an absolute minimum. Early considerations [19] indicated that to reach the standard 0.5 mPa pre-fill pressure level required to start a new discharge, took too long a time. This was the reason to launch an R&D programme to characterise a breakdown procedure assisted by Electron Cyclotron Resonance Heating (ECRH) [20] which has now become a reference. Based on this, the target time of 600 s has been defined as a requirement for the dwell pump-down time to be achieved by the divertor pumping system.

While for the early phase of operation the thermal protium outgassing of the first wall and vessel materials dominates, in later phases influences coming from neutron damage and plasma driven implantation increase the outgassing source term in magnitude and composition [21]. Regarding the particularly unwanted protium term, experimental outgassing measurements of untreated tungsten samples at 300°C (the expected first wall temperature for both WCLL and HCPB during the dwell phase) show an outgassing flux of $5.54{\cdot}10^{-5}$ $(Pa{\bullet}m^3)/(s{\bullet}m^2)$ after 10 h [19]. With a plasma surface area of 1419 m^2 as well as a geometric factor of about 1.85 to account for the actual 3D surface of the first wall (as used in ITER) an integral protium outgassing rate of approximately 0.15 $Pa{\cdot}m^3/s$ during the dwell phase is obtained.

It has to be noted that the fuel cycle contains a bypass around the tokamak so that the cyclic nature of the pulses will not directly translate in cyclic loads to the tritium plant.

3. Fuel cycle architecture development

3.1. A looped fuel cycle including the direct internal recycling concept

The main driver for the development of a novel fuel cycle architecture is the minimization of the tritium inventory. This was found to be necessary as the scaling of the ITER systems would have resulted in an inventory of the order of well above 10 kg [22] that is believed to be unacceptably high to meet the requirements discussed in Section 1.

The novel architecture is based on the Direct Internal Recycling (DIR) concept, developed by KIT [22, 23], where the tritium plant systems are bypassed for a large fraction of the unburnt fuel in the machine exhaust gases. The DIR concept reflects on the typical need of a fusion power plant, namely, to always repeat a given plasma, fuelled by a DT mixture at given composition, so that there is no need to separate the exhausted hydrogen isotopologues down to the level of the pure isotopes. However, the expected change of DT composition between fuel injection and exhaust gas must still be able to be compensated within the fuel cycle. DIR keeps the gas load to the tritium systems low and, hence, reduces their size drastically. The most important figure of merit to characterize DIR is the DIR ratio (R_{DIR}), which is defined as the recycled hydrogenic gas (Q2) stream bypassing the tritium plant divided by the overall hydrogen exhausted from the torus. It was shown that a DIR ratio of 0.8 is an optimum reflecting the fact that further load reduction on the inner fuel cycle side of the tritium plant would be overcompensated by loads coming from the outer fuel cycle side (blanket) [24].

As a result of the PCD Phase, a three-loop fuel cycle architecture has been developed [11, 12]. The main feature of these loops are significant differences in the residence times (or process speed), as will be quantified in Section 4.15. It was sought during the development of the fuel cycle architecture to maximize the gas load on loops with short residence times. Fig. 2 shows the three loops that are currently foreseen: the Direct Internal Recycling Loop (DIRL), the Inner Tritium Plant Loop (INTL) and the Outer Tritium Plant Loop (OUTL).

A full functional overview of the systems included in the current fuel cycle architecture is given in Fig. 3. This arrangement of the system blocks is the basis for the detailed description in Section 4 of this paper.

All three loops are part of the inner fuel cycle, while OUTL, however, connects directly to the outer fuel cycle. It must be noted that the concentration of tritium in the flows through the different loops decreases with the residence time. However, due to the size of the systems and the corresponding hold up, the tritium inventory still increases with increasing residence time. The system blocks shown in Fig. 2 are not exhaustive and shown only for illustration.

The current fuel cycle architecture considers a steady state operation of all systems wherever possible and features a dwell time bypass to decouple plasma operation from the tritium processing systems as far as possible. Steady state operation of the systems is advantageous for operation and control reasons: it must be kept in mind that the fuel cycle systems form a very large process plant with huge subsystems that need hours or even days until constant values for temperature or concentration profiles are reached. It is also important in the operation of the fuel cycle to decouple the three loops as much as possible.

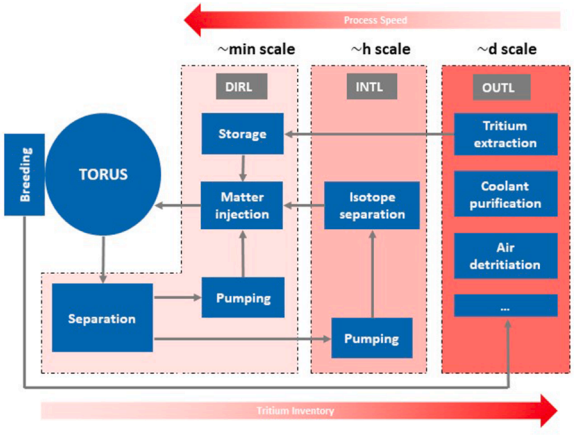

Fig. 2. Systematic illustration of the interlink between residence time and tritium inventory in the three-loop fuel cycle.

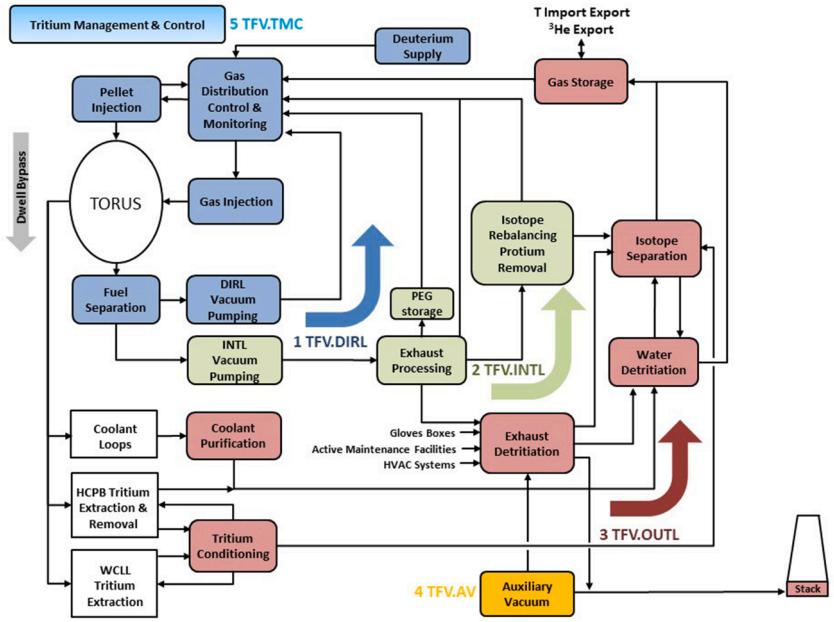

Fig. 3. Functional block diagram of the fuel cycle.

3.2. The direct internal recycling loop (DIRL)

3.2.1. Requirement on the DIRL

Following the DIR concept, the DIRL takes the unburnt fuel and recycles it directly - via the matter injection systems - back into the machine. It is the most important loop for the minimization of the tritium inventory in the fuel cycle. Due to the high gas throughput in the DIRL at R_{DIR}=0.8, special focus on low processing times (continuous processes) and optimized inventories has been given in the PCD Phase.

3.2.2. System block description

In the DIRL, six functional system blocks are included (blue in Fig. 3). These are:

- Fuel separation
- DIRL vacuum pumping
- Gas distribution, control & monitoring
- Deuterium supply
- Gas injection
- Pellet injection

Fuel separation is the most important technical system to provide a sharp gas separation of hydrogenic species from the exhaust gas. Without this system the DIR concept cannot be realized. There are performance aspects that have to be met: one is the separation efficiency, which denotes what fraction of the incoming gas can be separated out, and described by R_{DIR}. The other is the separation sharpness (the selectivity of hydrogenic separation), which denotes how pure the separated stream is with regard to hydrogens. To fully exploit the potential of the DIR concept, the separation sharpness should ideally be unity.

Vacuum pumping is a system that is contained in any fuel cycle and has the potential to accumulate large hydrogenic inventories, depending on the selected technical solution. In principle, the vacuum pumping system must fulfil two completely different tasks, namely, to keep the required pressure level in the subdivertor at the throughput to be pumped during plasma burn, and in the dwell period to pump down the torus to a vacuum level required for the start of the next pulse in a given time. These two functions do not necessarily have to be combined, but it would be advantageous to do so, as long as the performance in any of the two phases is not significantly compromised. For proper dimensioning and integration of the vacuum pumping systems, the exact geometry of the pump duct including the divertor has to be known (to consider the effect of conductance losses) and the expected gas flows and pressures must be known, as the pumps are very sensitive to variations. Vacuum pumping is a large system with several pumping stages. They occupy most of the lower ports, and the number of ports necessary is key information for the other systems potentially located there (such as for remote handling). While the flux to be injected by fuelling can theoretically be increased as needed by adding more pellet injectors, this does not work for vacuum pumps. Due to the rarefaction of the gas under the typical divertor pressures, there is a clear physics limit for how much can be pumped through a given opening cross-section, namely the (finite) black hole pumping speed.

In the DIRL, the pumped gases are recycled directly back to the machine without further processing, if separation sharpness allows to do so. However, before re-injection into the torus, the gas has to pass a gas collection & buffering system, which is basically a short-term storage system and a gas distribution, control & monitoring system (a complex valve and instrumentation system), where the gas is mixed with fresh fuel and where the desired gas composition for fuelling is adjusted, see Fig. 4. The target composition here will be somewhat different to the 50/50 target composition in the core, to reflect isotope dependent phenomena in pellet formation, transport, deposition and ablation as well as in vacuum pumping. Injection of PEGs can come, mixed with fresh fuel, in gaseous form via the gas injection system (as expected for radiative seeding of the divertor) or admixed to the pellets if the PEG shall be deposited in the core.

3.3. The inner tritium plant loop (INTL)

3.3.1. Requirement on the INTL

The INTL takes all gases that are not being recycled in the DIRL. The technologies foreseen in this loop are working continuously and quasi-continuously, which means that the residence times are an order of magnitude higher than in the DIRL.

3.3.2. System block description

In the INTL, four functional system blocks are included (green in Fig. 3). These are:

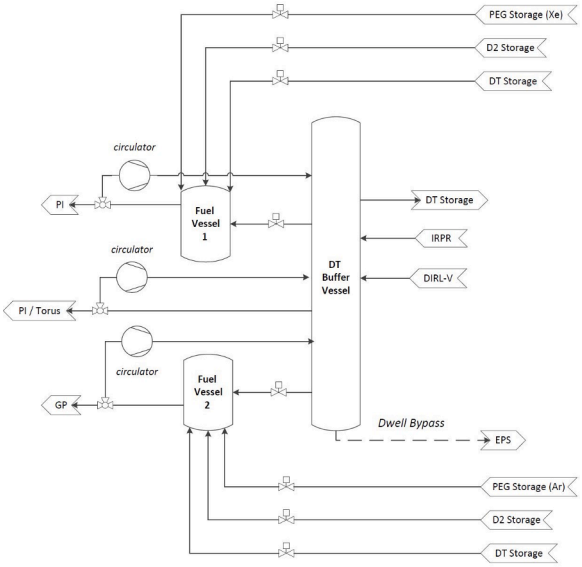

Fig. 4. Process scheme for gas distribution, control and monitoring.

- INTL vacuum pumping
- Exhaust processing
- PEG storage
- Isotope rebalancing & protium removal

Vacuum pumping is comparable with the DIRL vacuum pumping system, with one major difference: The throughput that has to be processed is approx. 4 times smaller.

The Exhaust Processing System (EPS) cleans up the exhaust gas and removes plasma enhancement gases and other impurities from the hydrogens. As the PEGs get activated from the fusion neutrons, dedicated measures (e.g. delay tanks and the like) have to be considered in the tritium plant design [25].

The pure hydrogen leaving the system is (partially or completely) transported towards the Isotope Rebalancing and Protium Removal system (IRPR) that adjusts the isotopic composition in the DT fuel mixture to the desired value, while also removing excess protium. The protium content to be removed by IRPR is another aspect that puts an upper boundary to the feasible R_{DIR} range. The system has to handle gases with a very high tritium content (~50%) where only moderate shifts in the isotopic composition are required. It has to route largely detritiated protium to the Isotope Separation System (ISS) in the OUTL.

3.4. The outer tritium plant loop (OUTL)

3.4.1. Requirement on the OUTL

The OUTL is responsible for processing all tritiated streams that do not constitute the circulated fuel. These arise from multiple sources in different tritium concentrations and chemical forms. Most notable is the requirement for steady state processing of bred tritium extracted from the breeding blankets to ensure its availability in fuel quality. As tritium may also permeate into the coolant loops, a fraction thereof has to be continuously detritiated to minimise build-up. Other requirements include the detritiation of air from buildings (from the Heating, Venting and Air Conditioning systems (HVAC)) and the Active Maintenance Facilities (AMF), dry inert gas (plus impurities) from glove boxes, and other gas streams that can carry trace amounts of tritium in order to recover the contained tritium as well as to allow the safe discharge of these streams with ALARA tritium levels. As all of these steps produce tritiated water, water detritiation is also required. Finally, safe storage of hydrogen isotopes is required, especially in a form that allows the accountable import and export of tritium.

3.4.2. System block description

In the OUTL, six system blocks are included (red in Fig. 3). These are:

- Exhaust detritiation
- Coolant purification
- Water detritiation
- Tritium conditioning
- Isotope separation
- Hydrogen storage

The Exhaust Detritiation Systems (EDS) are the only systems that discharge to stack. They receive gaseous streams that may contain tritium in trace amounts. These arise from ventilation of building areas that house tritium processing systems or the tokamak, the active maintenance facilities and glove-boxes, as well as the waste streams of the INTL. Water in the form of humidity in these streams is sent to the Water Detritation System (WDS), which receives all liquid aqueous streams that arise in all areas of the fuel cycle, with the main contributors being the coolant purification system and exhaust detritiation system. WDS produces a hydrogen stream of very high purity. The second product of the system is then a stream with elevated tritium concentrations that is sent to the ISS for tritium recovery.

The isotope separation system receives hydrogen streams from the IRPR, WDS and blankets (via TCS) that contain tritium not yet in fuel quality. The main task of this system is the production of a tritium free protium stream (discharge via WDS) and a protium free tritium stream (that may still contain deuterium) for fuelling. The system is linked directly to hydrogen storage.

The TCS receives the outlet streams from the tritium extraction of the breeding blankets and performs the necessary processing steps to make the streams acceptable for hydrogen isotope separation while discarding tritiated water to WDS and returning carrier gas (such as helium purge gas) to their source.

The hydrogen storage system receives all pure hydrogen streams that are not required for immediate reuse. It also marks the entry and exit point of tritium into or out of the fuel cycle. The hydrogen storage includes also a pure deuterium storage and a helium-3 removal system.

4. Detailed system block descriptions incl. selected technologies

4.1. Introduction to system block description

The development status of system blocks that are part of the fuel cycle and under the responsibility of TFV is described in this Section 4, which is organised as follows. For each system block, the technology selection will be presented in the first subsection to make clear why a certain technology has finally been chosen. In some cases, a fallback technology for risk minimization reasons has been kept. For the chosen technologies, an R&D line was developed for the PCD Phase, thereby emphasising those with a low maturity level, and where one could not build on RoX from ITER. Selected R&D results are presented in the second subsection. The issues and questions that still have to be addressed on the way to a Pre-Concept DEMO design are summarized in the final subsection.

4.2. Methodology of technology selection

The technology selection starts with a pairwise comparison to weight different requirements and continues with the calculation of a quality rating to rank the technology candidates. This approach is an acknowledged and presumed unbiased method to obtain an independent ranking in an arbitrary list of different requirements [26]. For the development of the fuel cycle, two different requirement sets have been assumed, depending on whether the systems are in the tokamak building or in the tritium plant building, see Fig. 4.

The actual quality rating is conducted in such a way that it is checked

Table 1
Technology quality rating for system blocks located in the tokamak building.

Category	Weighting p	Fuel Separation				Primary Pumping						Rough Pumping						Core Fuelling			
		Metal Foil Pump		Multi-stage Cryopump		Vapour Diffusion		Cryo-adsorption		Getter		Rotating Positive Displacement		Reciprocating Positive Displacement		Cryo-condensation		Centrifuge Pellet Injection		Gas Gun Pellet Injection	
		q	q x p	q	q x p	q	q x p	q	q x p	q	q x p	q	q x p	q	q x p	q	q x p	q	q x p	q	q x p
Required tritium compatibity level achievable	**23**	3	69	3	69	4	92	4	92	3	69	3	69	3	69	4	92	2	46	2	46
High system reliability by design (no complex solution)	**18**	2	36	3	54	4	72	3	54	3	54	3	54	2	36	3	54	2	36	3	54
High safety level and failure tolerance achievable	**25**	3	75	3	75	4	100	4	100	4	100	3	75	3	75	3	75	3	75	3	75
No or low regular maintenance required	**11**	2	22	3	33	4	44	2	22	3	33	3	33	2	22	3	33	3	33	3	33
Good remote maintainability	**20**	3	60	2	40	3	60	1	20	3	60	4	80	4	80	4	80	4	80	4	80
Conventional technologies widely used	**9**	2	18	3	27	3	27	3	27	3	27	3	27	3	27	2	18	4	36	4	36
Maturity Level as of now	**4**	1	4	4	16	4	16	4	16	3	12	4	16	4	16	4	16	4	16	4	16
Scaling to desired system size straight forward	**13**	4	52	4	52	4	52	4	52	3	39	4	52	3	39	3	39	3	39	3	39
High performance at low size	**8**	4	32	3	24	4	32	0	0	1	8	4	32	2	16	3	24	3	24	3	24
High magnetic field resistance achievable	**25**	2	50	3	75	4	100	4	100	3	75	4	100	4	100	4	100	4	100	4	100
High radiation resistance achievable	**25**	3	75	4	100	4	100	4	100	1	25	4	100	4	100	4	100	4	100	4	100
Low investment costs (low CapEx)	**6**	2	12	2	12	3	18	1	6	2	12	3	18	2	12	1	6	3	18	3	18
Low operation costs (low OpEx)	**10**	3	30	0	0	2	20	0	0	4	40	3	30	4	40	1	10	3	30	3	30
Hardware components independant of manufacturer	**9**	3	27	3	27	4	36	3	27	0	0	3	27	2	18	4	36	3	27	3	27
Low tritium inventory	**24**	4	96	0	0	4	96	0	0	0	0	4	96	4	96	0	0	3	72	2	48
Low demand of complex infrastructure	**10**	2	20	0	0	2	20	0	0	3	30	3	30	4	40	0	0	3	30	2	20
Sum	**240**	**678**		**604**		**885**		**616**		**584**		**839**		**786**		**683**		**762**		**746**	
QUALITY RATING W (%)		**70.6**		**62.9**		**92.2**		**64.2**		**60.8**		**87.4**		**81.9**		**71.1**		**79.4**		**77.7**	

to what extent a technology candidate meets the categories that had been ranked and weighted previously by the pairwise comparison. For this purpose, the score value obtained for each category is taken and multiplied with a rating number q. The value of the rating number has to be chosen from q = 0 (dissatisfying), 1 (inadequate), 2 (sufficient), 3 (good) or 4 (very good, ideal). Finally, the quality rating W is the sum of all products given in % related to the maximum number of points that would result if all categories would had achieved a rating of q=4. Hence, the quality rating does not only allow to compare different technologies but also indicates how much the found quality is away from an ideal (100%) solution. It is therefore also an important input to prioritise R&D given a limited budget.

Table 1 summarises the technology quality rating results in the form of a Pugh matrix for all 4 tokamak system blocks. Table 2 is showing the corresponding results for the tritium plant systems. For reasons of clarity, the tables only show the leading technologies; other investigated technologies are mentioned in the text. Earlier versions of such quality rating for selected technologies have been published previously [11, 27]; the findings in terms of the ranking order are very much in line with what is presented here. Compared to the earlier assessments, increased quality values resulted in the areas where additional R&D was performed over the last years, such as vapour diffusion pumping (increase from 70% to 90%) or temperature swing absorption (increase from 70% to 80%).

4.3. Fuel separation

The sharp separation of hydrogenic fuels from the tokamak exhaust close to the divertor is the key to the DIR concept. However, this puts some additional requirements on the technology selected to perform the separation task. Thus, the fuel separation technology has to be able to handle large processing streams, work in vacuum, which excludes the classical permeation membrane technology, fit inside the pump duct (2 × 2 × 3 m) and has to be (ideally perfectly) hydrogen-selective. Furthermore, the technology has to be tritium-compatible, radiation and neutron hard, and has to work reliably in strong magnetic fields. Safety and maintenance pose further constraints, which have to be carefully considered when deciding the best candidate fuel separation technology.

4.3.1. Technology selection

Taking the above-named criteria into account, two candidates have been selected, which can fulfil the task of hydrogen separation in the harsh environment present in the tokamak vicinity. The first candidate is a metal foil pump (MFP), which works on superpermeation. The second is a multi-stage cryopump working with stage-wise regeneration and with the capability for gas separation, different to that in ITER. Clearly, both technologies are novel and only existed as embryonic ideas when the European DEMO programme started, as the function of fuel separation at this position was not required in previous fuel cycle architectures. It, therefore, became a focal activity in the R&D programme to increase technology maturity in this field.

A definition of superpermeation is given in [28], relating the permeability of a membrane during superpermeation to that of an opening of the same size. Several metals are fit for superpermeation, especially those of group 5 (Nb, V, Ta) [29]. With this technology, the barrier to the absorption of hydrogen in a surface-limited permeation process is overcome by delivering the absorption energy to the hydrogen in advance, i.e. by dissociating it before its impingement on the surface. Historically, incandescent filaments have proven to be excellent sources for atomic hydrogen and contributors to high permeation fluxes. However, they only work properly in very low pressure regimes (< 0.1 Pa) and are prone to lifetime issues. Subsequently, we sought alternative methods for energization in the gas phase and identified a suitable candidate from cold plasma technologies. In literature, there are reports of plasma-driven permeation experiments working at lower pressures [30–32] but not in the range beyond 10 Pa. Hence, another challenge that comes with the development of an MFP is finding an industrially available, scalable plasma source that can be used in our application in combination with large metal foil surfaces.

The multi-stage cryopump technology is based on condensation of the heavier and cryosorption of the lighter gases at different cryogenic temperatures, but spatially distributed in different chambers of the pump. This pre-separation allows for separate stage-wise regeneration of the chambers that can be isolated from each other via gate valves. This concept has existed for some time [33].

Table 2
Technology quality rating for system blocks located in the tritium plant building.

		Helium CPS						Water Detritiation						HVAC / AMF Exhaust Detritiation				EPS Impurity Removal & Processing						EPS PEG separation			
Category	Weighting p	Getter Bed		Molecular Sieve Bed		Cryo Molecular Sieve Bed		Water Distillation		Water Distillation + CECE		Combined Electrolyses and Catalytic Exchange		Catalytic Conversion + Molecular Sieve Dryer		Catalytic Conversion + Wet Scrubber		Catalytic Converter + Cryogenic Distillation		Catalytic Converter + Pd Alloy Diffuser		CatalyticConverter + Getter Bed		Cryogenic Distillation		Zeolite Membranes	
		q	p x q	q	p x q	q	p x q	q	p x q	q	p x q	q	p x q	q	p x q	q	p x q	q	p x q	q	p x q	q	p x q	q	p x q	q	p x q
Required **tritium compatibity** level achievable	**20**	3	60	3	60	3	60	3	60	3	60	2	40	2	40	3	60	3	60	3	60	3	60	4	80	3	60
High system **reliability by design** (no complex solution)	**10**	3	30	3	30	2	20	4	40	2	20	3	30	4	40	4	40	2	20	3	30	3	30	4	40	3	30
High safety level and failure tolerance achievable	**19**	3	57	3	57	1	19	3	57	3	57	3	57	4	76	4	76	2	38	2	38	3	57	3	57	3	57
No or low regular maintenance required	**13**	3	39	3	39	2	26	3	39	2	26	2	26	2	26	4	52	3	39	3	39	3	39	3	39	2	26
Conventional technologies widely used	**6**	3	18	3	18	3	18	4	24	3	18	3	18	4	24	4	24	3	18	4	24	2	12	4	24	3	18
Maturity level as of now	**6**	3	18	3	18	3	18	3	18	3	18	3	18	4	24	4	24	3	18	3	18	2	12	3	18	2	12
Scaling to desired system size straight forward	**14**	4	56	4	56	2	28	2	28	3	42	3	42	4	56	4	56	3	42	3	42	3	42	3	42	3	42
High performance at low size	**11**	3	33	3	33	4	44	2	22	4	44	3	33	2	22	2	22	2	22	3	33	2	22	3	33	2	22
Low investment costs (low CapEx)	**8**	2	16	2	16	2	16	3	24	2	16	3	24	2	16	3	24	2	16	3	24	3	24	3	24	3	24
Low operation costs (low OpEx)	**13**	3	39	3	39	1	13	2	26	4	52	2	26	3	39	2	26	1	13	3	39	3	39	3	39	3	39
Hardware components independant of manufacturer	**5**	2	10	3	15	3	15	4	20	3	15	3	15	3	15	3	15	3	15	3	15	2	10	3	15	2	10
Low tritium inventory	**22**	2	44	3	66	2	44	1	22	3	66	3	66	3	66	3	66	1	22	4	88	1	22	2	44	3	66
Low demand of complex infrastructure	**9**	3	27	2	18	2	18	3	27	2	18	3	27	3	27	3	27	2	18	3	27	3	27	3	27	2	18
Sum	**156**	**447**		**465**		**339**		**407**		**452**		**422**		**471**		**512**		**341**		**477**		**396**		**482**		**424**	
QUALITY RATING W (%)		**71.6**		**74.5**		**54.3**		**65.2**		**72.4**		**67.6**		**75.5**		**82.1**		**54.6**		**76.4**		**63.5**		**77.2**		**67.9**	

		Isotope Rebalancing & Protium Removal						Isotope Separation					
Category	Weighting p	Membrane-Coupled Temp. Swing Absorption		Cryogenic Distillation		Pressure Swing Adsorption		Pressure Swing Adsorption		Thermal Diffusion		Cryogenic Distillation	
		q	p x q	q	p x q	q	p x q	q	p x q	q	p x q	q	p x q
Required **tritium compatibity** level achievable	**20**	4	80	4	80	4	80	3	60	3	60	4	80
High system **reliability by design** (no complex solution)	**10**	3	30	3	30	3	30	2	20	2	20	3	30
High safety level and failure tolerance achievable	**19**	3	57	2	38	2	38	1	19	3	57	2	38
No or low regular maintenance required	**13**	3	39	3	39	2	26	3	39	3	39	4	52
Conventional technologies widely used	**6**	2	12	4	24	3	18	2	12	2	12	4	24
Maturity level as of now	**6**	2	12	4	24	2	12	2	12	2	12	4	24
Scaling to desired system size straight forward	**14**	4	56	4	56	4	56	4	56	2	28	4	56
High performance at low size	**11**	4	44	2	22	3	33	2	22	2	22	2	22
Low investment costs (low CapEx)	**8**	3	24	3	24	2	16	3	24	3	24	2	16
Low operation costs (low OpEx)	**13**	3	39	1	13	2	26	3	39	3	39	2	26
Hardware components independant of manufacturer	**5**	2	10	4	20	3	15	3	15	3	15	4	20
Low tritium inventory	**22**	3	66	2	44	2	44	2	44	3	66	2	44
Low demand of complex infrastructure	**9**	4	36	2	18	2	18	3	27	3	27	3	27
Sum	**156**	**505**		**432**		**412**		**389**		**421**		**459**	
QUALITY RATING W (%)		**80.9**		**69.2**		**66.0**		**62.3**		**67.5**		**73.6**	

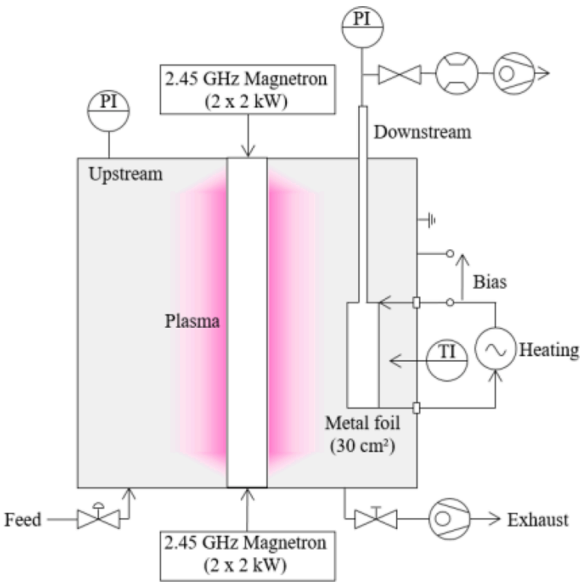

Fig. 5. Sketch of experimental facility HERMESplus.

Table 1 shows the result of the quality evaluation. Due to the imperfect separation sharpness for the multi-stage cryopump [34], circulating some residual noble gases in the DIR loop is unavoidable with this technology. This makes it difficult to generate well-defined pellets at the requested cryogenic mechanical strength, and, if they are generated, would add to the impurities in the plasma. Eventually, it might be necessary to combine the cryopump with an additional purification technology to treat the gas stream during the regeneration of the pumps. Based on the high permeation fluxes that can be achieved in principle with superpermeation [29], the scaled-up MFP would occupy less space and complexity than the cryopumping system. Overall, the MFP reaches a higher quality rating of 71% vs 63% of the multi-stage cryopump, prompting us to focus the research on the more promising MFP technology. In the PCD Phase activities for DEMO the design and feasibility of a pumping concept based on tritium direct recycling has been identified as one key design integration issue; in this context the comparison of the new technologies is repeated in the light of the experimental work done in recent years [15].

4.3.2. *Selected R&D achievements*

In contrast to cryopumping, metal foil pumping cannot look back on years of technological application and clearly started from a very low technology readiness level. Breaking down the MFP's pumping process to its core, we find a 5-step sequence comprising (i) energization of hydrogen, (ii) transport to the foil in vacuum, (iii) upstream surface interaction and absorption, (iv) bulk diffusion, and eventually (v) thermal re-emission on the downstream surface [35]. The two main components in this interplay are the hydrogen energizer and the metal foil.

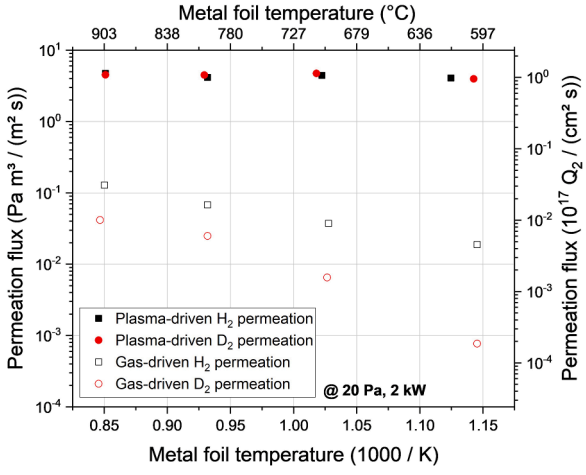

Fig. 6. Permeation fluxes with H_2 and D_2 plasma vs the Nb metal foil temperature.

Provision of a versatile experimental facility. An experimental setup has been developed that can accommodate a new, scalable, cylindrical microwave plasma source, which can also work at pressures > 10 Pa (*Duo-Plasmaline* from the company Muegge, Germany) [36]. In this experimental setup (HERMES*plus*), resistive heating of the metal foil in a wide range is possible. An exchangeable, cylindrical metal foil module with an area of approximately 30 cm^2 is completely immersed in the plasma. The modules with niobium and vanadium, respectively were manufactured by the group of Alexander Lifshits from Bonch-Bruevich Saint Petersburg University of Telecommunication. A sketch of the facility is shown in Fig. 5.

Demonstration of superpermeation fluxes with a microwave plasma source. In this experimental configuration, we were able to demonstrate high permeation fluxes and superpermeation with its characteristic temperature independence, always using the same niobium foil module. Fig. 6 shows permeation fluxes obtained with H_2 and D_2 plasma individually over a temperature range from 900 to 600°C, at constant microwave power and pressure. The observed plasma driven permeation (PDP) fluxes of several $(Pa{\cdot}m^3)/(m^2{\cdot}s)$ are comparable with what is usually found in the filament configuration [29], but is achieved at the higher pressure range, as needed for the metal foil pump. Also shown are Gas Driven Permeation (GDP) values which are orders of magnitude lower.

Isotope dependence. Ideally for the direct internal recycling, the fuel separation technology should not result in major changes of the isotope ratio D/T. As this has only been tested using filaments [37] and not plasma sources, we dedicated an experimental campaign to the comparison of H_2 and D_2 PDP, the results of which are plotted in Fig. 6. In the shown temperature range, PDP fluxes of H_2 reached 4.3 $(Pa{\cdot}m^3)/(m^2{\cdot}s)$ ±5.3 % and 4.4 $(Pa{\cdot}m^3)/(m^2{\cdot}s)$ ± 5.1 % for D_2. The measured difference between the two gases agrees well; hence, we observe no isotopic effect for PDP of the two isotopes. Note that for GDP, there is a rather strong isotopic effect. The process responsible for the generation of the GDP flux is limited by the sticking and dissociation of molecules on the surface, hinting at a significantly smaller molecular sticking coefficient of D_2 as compared to H_2.

Compression. The prime functionality of the MFP is to separate, but due to the superpermeation principle, which works even against the neutral pressure difference, it also has the capability to compress. In the experiments, compression ratios up to 100 have been measured [35].

Dependence on microwave power. The plasma input power causes a linear change in permeation fluxes up to a level at which power saturation is achieved. In HERMES*plus*, which features a 300 mm plasma column, this is reached at about 2750 W. Increasing the power beyond this value only increases the measured microwave reflection as more of it is reflected at the skin depth in the plasma. The highest achieved permeation flux so far was 6.6 $(Pa{\cdot}m^3)/(m^2{\cdot}s)$ for H_2, which is the range to be expected for superpermeation.

Influence of external magnetic field. Being operated in the lower pump duct of DEMO, the MFP is subjected to strong magnetic fields from the tokamak coils. Maximum flux densities there reach 1.2 T [38]. The tokamak stray magnetic field has been found to have strong gradients inside of the pump duct, where the MFP will be located. Due to this high flux density, it is necessary to critically assess the impact on the individual components of the MFP and its working principle. The metal foil material itself (our own investigations have been using V and Nb) is fairly thin (0.1 mm thickness) and not supposed to take mechanical loads. While the material is paramagnetic and does not react

significantly to the magnetic field itself, it has to withstand forces when acting as a conductor during resistive heating. The resulting shear stress on a foil will require a support structure to avoid denting and deformation.

The main species responsible for superpermeation is atomic hydrogen. Its movement is not affected by magnetic fields nor by ion-drag due to the low degree of ionization in the MFP cold plasma. Hence, hydrogen atoms find their way to the metal foil unaffected by magnetic fields under the assumption of proper plasma ignition. This assumption is backed by experiments with the candidate plasma source testing the ignitibility of plasma in strong magnetic fields of up to 250 mT. In the experimental setup, it has been possible to test field gradients and fields that are either purely perpendicular or parallel to the plasma source. Generally, we found that the plasma can always be ignited at low pressures (10 Pa) if sufficient power is used and that increasing power input always helps ignition. Furthermore, we observed the plasma source's dielectric outer surface temperature does not increase significantly when a magnetic field is switched on. A discussion of experimental results can be found elsewhere [38].

Modelling and simulation. We developed a 2-dimensional, axisymmetric hydrogen plasma simulation and modelled the experimental set-up in HERMES*plus* to investigate plasma characteristics like composition, electron density, temperature and behaviour in magnetic fields [38]. The obtained performance results have been integrated into a Monte Carlo-based vacuum simulation via a change in the permeation probability through a modelled foil boundary on the outside and a plasma boundary on the inside. The most important variables in this simulation are the permeation probability through the foil and the excitation probability on the plasma boundary that can be extracted from our experiments. With this model, it is possible to predict DIR ratios and pumping speed. And with this information, we elaborated a workflow that consistently tailors the complete vacuum pump train considering the conductances of the port and the pumping speeds of the roughing pumps.

4.3.3. Open issues and questions to be addressed in CD phase

Continuation of magnetic field testing. The tests performed under an external magnetic field have shown that the influence on the superpermeation process as such will be manageable. However, the plasma shape (volume, elongation) and ignition showed a strong dependency, which must be investigated under more representative external field conditions (non-uniform, gradients, drift effects). To address this point, the magnetic field test campaign will continue.

Steady state superpermeation. The MFP has to operate over large time spans without maintenance (years) and with only occasional in-situ surface treatment without severe degradation of performance. Here, limitations in the experiments have been seen which, however, we relate with the experimental set-up and the unfavourable ratio of vessel surface to metal foil rather than with a physics problem. To address this point, a re-design of the plasma source will be pursued and HERMES*plus* will be upgraded. Sensitivities against impurities on back and rear side of the foil will have to be investigated.

Modelling and design tool. The third activity will focus on an extensive campaign to validate the plasma model by optical emission spectroscopy, and with this to support the vacuum simulation programme by deriving an excitation probability function with respect to the axial position in the MFP as the pressure drops along it. The resulting transient code will become a major tool for the evaluation of the detailed design of an MFP in the future.

Engineering design of a MFP. Complementary to performance and physics investigations, a technical mock-up will be built to study mechanical integration issues such as foil connections or cooling. The final product of this R&D line will be a prototype pump for testing.

4.4. Vacuum pumping

The main function of the vacuum pumping systems is during burn to transfer the exhaust gas and to compress to close to ambient conditions, and during dwell to pump down the plasma chamber.

4.4.1. Technology selection

The vacuum pumping system is subdivided into the primary and rough pumping systems, as no pumping technology is capable of pumping over the whole operational pressure range. Functional analysis for burn and for dwell have shown that dedicated pumping systems are not required for the technologies eventually selected, so the decision was taken to fulfil the pumping duties during burn (pumping high throughputs in the Pa range) and dwell (pumping lower throughputs dominated by outgassing down to the mPa range) by one and the same pumping system.

The pumping system has to be localized close to the torus downstream of the fuel separation system in order to keep conductance losses due to the piping to a minimum. The primary pumping takes the gases as coming from fuel separation, and operates in the pressure range up to several 100 Pa. The rough pumping system compresses the gas mixture to around atmospheric pressure. A technology screening was performed for primary as well as rough pumping to find the most appropriate pumping technologies. The leading candidates of the quality rating are listed in Table 1, details also for other checked technologies have been published in [39].

4.4.1.1. Primary pumping. Five technologies were considered for primary pumping: Getter pumping, cryo-adsorption pumping, continuous cryopumping, turbomolecular pumping and vapour diffusion pumping. Both second place technologies have achieved a quality rating of ~ 60%. Cryo-adsorption (the ITER technology choice) and getter pumping are gas binding pumps and thus have to be operated in a discontinuous operating cycle, which makes them less attractive. An essential difference between cryopumping (condensation and adsorption) and getter pumping is that the latter involves absorption and/or dissolution, even solid phase changes, which adds the risk of potentially non-recoverable tritium inventory. However, developments of new getter materials with a significantly increased hydrogen pumping capability [40] may make this an attractive option in the future, if a good solution to integrate this with an additional pumping system for non-getterables can be found. For accumulation pumps, continuous pumping can only be achieved by operating several pumps in parallel. Cryo-adsorption pumps have been identified as the fallback solution for DEMO but a simple upscaling of the ITER pumping system is not possible due to a prohibitively high tritium inventory.

Vapour diffusion pumping which achieved the highest quality rating mainly because of its reliability, low maintenance requirements, high performance and low radioactive inventory, has been chosen as most suitable for primary pumping. The working principle of vapour diffusion pumps is based on the momentum transfer from a high-speed operating fluid vapour to the pumped gas. Mercury is used as the operating fluid because it is perfectly tritium compatible and has been successfully used in diffusion pumps in the past. It is currently planned to implement two types of diffusion pumps in DEMO: Linear Diffusion Pumps (LDPs) and cylindrical Booster Pumps (BPs). The BP is used to close the pressure gap between the LDP outlet and the rough pumping system. It is expected that the use of BPs will be sufficient in the fuel separation permeate stream due to the compression achieved by the fuel separation process itself.

4.4.1.2. Rough pumping. Four technologies were considered for rough pumping.

Cryo-condensation pumping (the ITER technology choice) works by condensing the gas to be pumped on cooled surfaces. In order to pump

hydrogen a complex, costly cooling infrastructure is required. Additionally, these pumps typically accumulate the condensed gas internally and would therefore feature a high tritium inventory in DEMO. Thus, they are deemed not attractive.

Rotating positive displacement pumps are continuous pumps that operate by compressing the gas via a rotating movement. Due to the lower complexity, this is the rough pumping technology that is found to be most appropriate for DEMO. Different pump types using this principle exist. An in-depth comparison revealed that liquid ring pumps (LRPs) are the best choice for DEMO [39]. These work by an eccentrically mounted paddle wheel that is driven by a motor in order to create a liquid ring around the rotor. Due to the eccentric mounting, the gas compartment's volume between the liquid ring and the paddle wheel changes during a turn of the wheel which is used to compress the gas. For tritium compatibility, mercury can be used as the operating fluid. This has the positive side-effect that the integration with the proposed mercury vapour diffusion pumps becomes easier.

4.4.2. *Selected R&D achievements*

As vapour diffusion pumps are a technology that has not yet been employed in fusion applications, it is essential to identify and address potential show-stoppers for the application and consider these in detail. Here we have to face the situation of fading and lost expertise, as mercury has been gradually substituted by specifically designed silicone/mineral oils since the 1930s, and oil diffusion pumps have in the meantime been largely replaced by turbomolecular pumps in many industrial and laboratory setups. Oil based pumps are not tritium compatible.

Similarly, a mercury liquid ring pump has never been built before. As the vacuum performance of such a roughing pump does not depend strongly on the operating fluid, the performance risk associated with changing towards mercury is considered to be low. However, the increased density of mercury (13.6 kg/l) compared to the usually used water (1.0 kg/l) has significant consequences on the mechanical design of the rotary equipment.

Provision of a versatile experimental environment. For developments of the pumping systems the facility THESEUS was designed, built and commissioned at KIT [39]. The pumps investigated in this facility were commercial pumps, but modified to make them mercury and tritium compatible. Complementary to THESEUS, a dedicated mercury lab was constructed [41]. This was utilized to develop handling procedures for mercury, validate processes for cleaning of mercury-contaminated components and finally demonstrate safe operation in a mercury environment.

High throughput pumping with mercury diffusion pumps. The fusion application is characterized by throughputs / inlet pressures which are untypically high for ultra-high vacuum pump technologies necessary to provide primary vacuum. It has therefore been tested which end pressures can be achieved at such conditions [42]. It was found that vapour diffusion pumps can be operated at high pressures and throughputs when an ejector stage is included, which is however untypical for this pump type.

Performance demonstration of mercury liquid ring pumps. Proof-of-principle tests have been performed with a modified pump to derive an understanding of the design criticalities under mercury operation conditions. The experiments showed the allowable rotor frequency range so that liquid ring pumps can operate smoothly and without suffering strong vibrations when using mercury as the working fluid.

Operation of mercury diffusion pumps in high magnetic fields. This was addressed in a set of analytical estimations of magneto hydrodynamic (MHD) effects on mercury. It was found that the electric conductivity of the vapour is so low that the influence of MHD effects is negligible. For liquid mercury, weak MHD effects are expected but based on the estimations no significant influence on the operating principle is expected.

Mercury contamination of the system upstream of the mercury diffusion pump. This issue is not specific to the application in fusion and generally solved by installing a cooled baffle above the inlet of the diffusion pump. As the purity requirements of DEMO are very high it was concluded that a multi-stage baffle is necessary [23, 43]. If found necessary it is possible to install a third, gold- coated stage to adsorb residual vapour. The design of such a baffle has to compromise high conductance and low contamination rate.

Tritium compatible designs. It is known that the rotary part of rough pumps has complex issues to be made tritium compatible in particular concerning reliable sealing and lubrication [44, 45]. Building on the experience gained in the proof-of-principle tests, a fully mechanical tritium compatible pumping train (MTPS) consisting of a cylindrical mercury booster pump and two mercury LRPs in series was designed, constructed and operated at KIT [46, 47]. The LRPs were designed with an industry partner (Hermetic Pumpen, Germany) for operation with mercury and tritium and proved that this technology is available for the application in the fusion environment. The BP design is based on a modified commercial oil diffusion pump. It was originally planned to test MTPS in the JET DT campaign 2021 but this did not finally happen due to a change of the R&D focus in the JET programme. However, MTPS has been operated in THESEUS for approx. 100 h up to June 2021 without any issues. Fig. 7 illustrates the measured hydrogen throughput of the MTPS at varied boiler power of the mercury booster pump. It is revealed that one and the same pump allows for different gas throughputs if the boiler power is adjusted accordingly, and the inlet pressure scales less than linearly with the throughput. Such experiments will be used to estimate energy consumptions of the pumping systems in the future.

Modelling and simulation. Diffusion pump designs have been developed by experience in the first half of the last century so that there is no tool available that would allow a systematic approach to design them for a new operating fluid. This is why we established a modern simulation methodology based on the Direct Simulation Monte Carlo method. As a first step a parametric study of single-stage small LDPs was

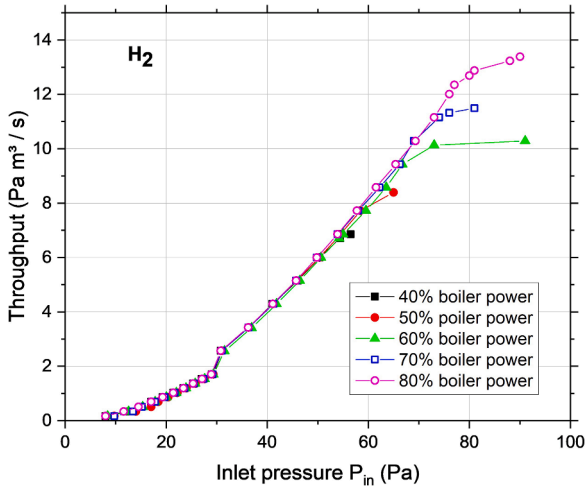

Fig. 7. MTPS pump characteristics.

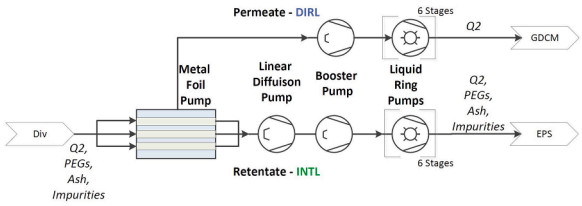

Fig. 8. Pre-Concept configuration of the vacuum pumping systems.

performed to verify the applicability of this methodology [48]. It was found that all essential properties of diffusion pumps could be reproduced by the simulations. Additionally, the simulations showed that the gas species dependency of the pumping speed is less than according to Graham's law. Moreover, it could be demonstrated that the simulation model can be utilized to optimize geometric and operational features of the LDP.

Similarly, the open literature of liquid ring pumps is very much limited and provides empirical correlations for water, at best. Hence also here, a model is under development, which properly describes the main characteristics of the two-phase flow in the rotor volume.

Integration of all pump types. The integration of all three vacuum pump types (MFP, LDP/BP and LRP) in a DEMO-wide pumping system has to consider the different compression ratios at the different pressure levels and throughputs. Fig. 8 illustrates the pre-concept configuration.

4.4.3. Open issues and questions to be addressed in CD phase

After successful proof-of-principle testing it is now necessary to verify that these pumps can be manufactured and will perform according to prediction.

Characterization of nozzle flows. As the pumping effect in vapour diffusion pumps is primarily based on the momentum transfer from the mercury vapour jet to the gas, a more detailed investigation of nozzle(s) producing the jet is necessary and is under preparation in order to understand better their influence.

Advancement of modelling and simulation capability and its validation by experiment. It has to be demonstrated that the simulation methodology is capable of describing larger, multi-stage LDPs and BPs. As there is practically no literature on such pumps available, a test pump has to be built that allows a rich set of diagnostics to provide experimental data for code validation. If necessary, improvements to the methodology have to be incorporated. Optimized designs for the LDP and BP then have to be derived from the collective results and experience of the above activities.

Demonstration of back-streaming management. A particularly important aspect is the back-streaming. A suitable baffle has to be developed for the LDP inlet and its performance validated. Special focus has to be placed on maximizing the conductance whilst keeping the mercury back streaming rate below a tolerable limit.

In this context, one should also note the different implications of mercury use. With regard to the handling of mercury, there is substantial experience available in mercury industry (mercury waste treatment, reprocessing of batteries) to which TFV created a link. A biomonitoring procedure was developed and, using the experimental facilities at KIT, it was possible to demonstrate safe procedures that clearly limit the workers exposure below the allowed concentrations. Regarding the waste aspects, a programme has recently been accomplished that provides a decommissioning routine for activated mercury [49]. Finally, there are strict legal limitations on the use of mercury in all countries

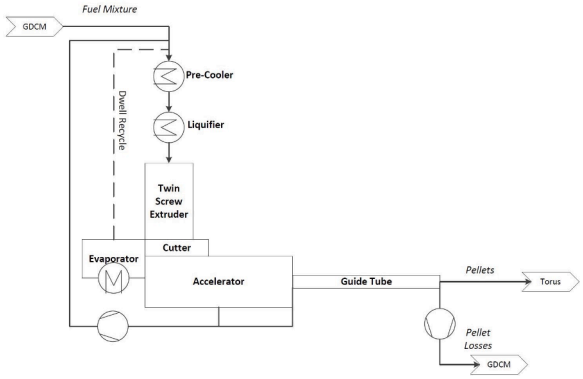

Fig. 9. Pre-Concept configuration of the pellet injection system.

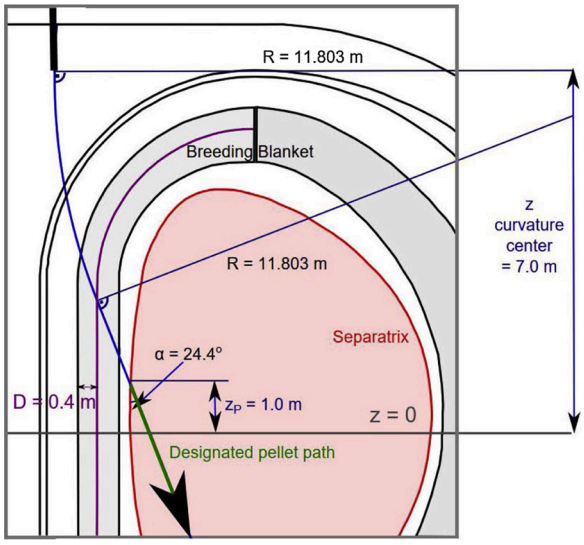

Fig. 10. Poloidal cross section of the HFS injection geometry [52].

that signed to the Minamata convention (EU 2018/852 in Europe). In this aspect, the KALPUREX process [23] is currently under study for inclusion in the annex that lists the allowed mercury applications.

4.5. Matter injection

Matter injection comprises the input of material to the torus vessel for several purposes: fuelling the burning plasma, potential ELM control, support the ramp-up and ramp-down of plasma as well to provide PEG for e.g. radiative power dissipation and/or divertor buffering. Depending on the purpose, matter is to be injected in a gaseous or solid (cryogenic) state. Hence, the employed technology is different.

4.5.1. Technology selection

Gas injection for radiative seeding, divertor buffering and for the ramp-up/ramp-down phase will be realised in a very similar way to ITER, relying on radiation hard valves, and gas distribution systems. But it is widely accepted that simple gas injection is not suitable for plasma core fuelling in DEMO, contrary to today's fusion devices. The injected neutral gas is not able to penetrate the separatrix, it would be reflected and pumped away without contributing to the fusion process.

A dedicated study was carried out in order to assess the further proposed matter injection technologies [50]. The basic requirements for fuelling purposes are the ability to deliver the appropriate fuel mix to the plasma, to provide sufficient particle flux and ensure safe operation. For

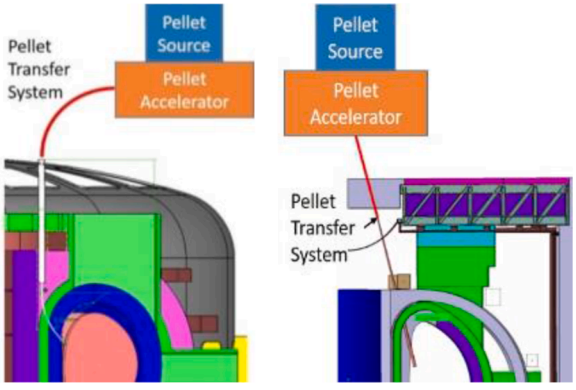

Fig. 11. Generic view of sub system core fuelling composed of pellet source, pellet accelerator and pellet transfer system for the two variants that are kept for the CDP [51].

the quality rating a set of technologies was investigated based on the same criteria as for all tokamak systems. As Table 1 shows, the prime candidate chosen for core fuelling is the conventional pellet injection, for which two variants exist that gained a very similar rating, so that both were further developed within the TFV R&D programme. The alternative techniques, such as compact tori injection, railgun pellet injection or unmagnetized plasma jet injection, showed clearly lower quality ratings. They all have in common the fact they do not meet the minimum requirements and hence they are not prime candidates to be considered for the CD Phase.

For pellet injection the main rating categories are tritium inventory, which will always be significant as the fuel comes as a cryogenic solid, and tritium compatibility, where some technology elements within pellet injection have already been studied for ITER. Additional important aspects are operational reliability, compatibility with injection geometries, and high fuelling efficiency, which is one of the key performance indicators of the fuel cycle, see Section 4.14.1.

4.5.2. Selected R&D achievements

A summary of the results achieved in PCD Phase is published elsewhere [51].

Injecting pellets to the magnetic High Field Side (HFS) is the only way to set up a feasible technical system. The HFS injection requires the use of guiding tubes. The injection location is of great importance for the fuelling efficiency. Key geometric parameters are the penetration point of the separatrix and the direction of movement. With the additional information on the pellet speed, the particle deposition process can be modelled. All these parameters are connected: the target penetration point (represented by the vertical distance "z" from the equatorial midplane, see Fig. 10) and the aimed injection direction require a certain bending radius of the guiding tube. This is due to geometrical constraints by the magnets. The maximum pellet speed is then limited by the bending radius of the guiding tube, beyond this limit, the mass loss becomes too large. On the other side, the pellet deposition in the plasma is more favourable for large impact angles, which, however, are associated with a small bending radius and hence limited pellet speed. This is why thorough optimizing activities were performed in order to assess the parameter range [52].

Pellet injection scheme. For the HFS injection a pellet speed up to 1200 m/s is envisaged, within the typical speed range of centrifuge or single-stage gas gun, the two successful technology candidates for pellet acceleration according to Table 1. An alternative injection scheme proposes a Direct Line of Sight (DLS) guiding tube [53]. Due to geometrical constraints, the penetration angles are smaller and the fuelling efficiency is less. This can be compensated at least partially using a double stage gas gun with a speed of up to 3000 m/s. Both schemes, see Fig. 11, will be further developed in CDP.

Pellet jitter. Inevitably, there is a distance between the pellet source and the plasma boundary. The travel time to overcome this distance creates a delay and the speed scatters the pellet arrival time causing a jitter in fuel supply [54]. This aspect has been investigated for the case of the gas gun in a collaboration between ENEA and ORNL [55].

Closed loop modelling for burn control. Since the plasma core fuelling actuator plays a key role in burn control needs, recent investigations have been focused on the development of adequate actuation and control tools [56, 57]. For this purpose, the pellet system accuracy in terms of timing must be good enough to qualify it as an actuator in the plasma control loop. It has turned out that great care is required to select an appropriate acceleration principle (centrifuge vs gas gun).

Initially, $6{\cdot}10^{21}$ atoms, the size of a fuelling pellet in ITER was considered as a reasonable choice. Modelling activities taking into account the feedback from the plasma (closed loop) indicated this is unfavourably strong for DEMO due to control issues [58]. Hence, the design size was adapted accordingly to $2{\cdot}10^{21}$ atoms. This size still has a considerable impact on plasma stability, challenging the diagnostics and the plasma control system. After all, an approach has been developed to calculate the fuelling impact of a pellet entering the plasma depending on the point of penetration as well as its direction and speed. This model includes the feedback of the plasma on the injected pellet. This tool can be applied to any future DEMO plasma scenario.

Gas injection architecture. An assessment of the design of the gas manifold designated for ITER has indicated that this solution seems appropriate for serving DEMO [59]. With regard to flux control, a more suitable solution was worked out based on a separation of the on/off function (valve) from flow adjustment. The latter is controlled by an adjustable orifice and a regulation valve operated in choked flow mode.

Ramp-up and ramp-down. Core fuelling during ramp-up and ramp-down requires a hand-over from gas injection to pellet injection. In the early phase of a discharge, the plasma is not hot and dense enough to absorb an injected pellet. First studies have been performed to understand the cross-over from gas to pellet injection in the ramp-up phase [51].

4.5.3. Open issues and questions to be addressed in CD phase

The Pre-Concept Design identified some important issues, currently not addressed on today's fusion devices. In order to close this knowledge gap, a European pellet test bed will be constructed in the CD Phase (DIPAK-PET), as described in Section 5.3.

Quantification of mass losses. Most importantly, the knowledge of mass loss issues in curved guiding tubes is incomplete with respect to required speed ranges and possible bending geometry. The results from the new test bed are important input parameters for the internal fuel cycle efficiency. The main aim is to provide methods of how to optimize the guiding tube design and benchmark it to the experiments in DIPAK-PET.

Pellet fabrication. There is a lack of knowledge in the development of systems for pellet fabrication and accelerating technology on a scale larger than that available at present fusion devices. The main deficits are regarding long pulse capability, performance repeatability and uniformity, precise acceleration to high speed and tritium handling capability. Early in the CD Phase a decision for a suitable technology must be taken. This will also consider an innovative new approach by thermal pellet formation [60]. The DIPAK-PET will then confirm this technology choice.

Gas injection dynamics. In the CD Phase, studies regarding the dynamic performance of the gas injection are foreseen, taking representative flow parameters and gradients as a basis to show feasibility in terms of the required response times. Furthermore, the sputtering effects of charge exchange neutrals under DEMO conditions have to be better understood.

4.6. Exhaust processing

The main function of the Exhaust Processing System (EPS) is to separate the exhaust gas into three streams: a deuterium and tritium stream, for re-use as fuel; PEG stream(s) to be re-used; and the remaining de-tritiated waste stream – unwanted by-products resulting from the fusion process.

4.6.1. Technology selection

EPS will be the first part of the tritium plant in line to receive gas from the tokamak downstream of the pumping systems. It can be split into two stages as shown in Fig. 12:

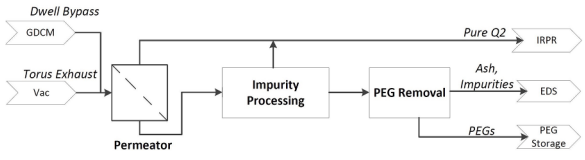

Fig. 12. Pre-Concept configuration of the exhaust processing system.

- Tritiated impurity removal and processing, to chemically separate and recover Q2 from Q-bearing impurities arising in the exhaust such as water, or hydrocarbons (particular in the early phases of operation with tritium).
- PEG removal and separation, to recover PEGs for re-use from the remaining gas flow. Detritiated waste products (such as helium, CO, CO_2) are then routed to exhaust detritiation for final clean-up and discharge.

Before these stages pure Q2 gases (bulk T_2 and D_2, and any H_2) are recovered and fed into the DIRL using permeator technology for which ITER operating experience and improvements to the technology are expected over the coming years.

4.6.1.3. Tritiated impurity processing. The analysis of technologies versus key design criteria identified the catalytic converter coupled with Pd alloy diffuser clearly as the best-suited technology for impurity processing. Such technology requires two different converter stages running at different temperatures and including different materials. Here, the first stage converts (cracks) the Q2O into Q2 and water and the second stage converts the CQx into carbon and Q2. It scores very well against all criteria. However, a regeneration of the beds is needed from time to time adding oxygen to convert the carbon produced into CO_2.

The other technologies investigated also included converter stages but different methods of separating the produced hydrogen. They all scored less. The use of cryogenic distillation suffers from large energy consumption and increases the tritium inventory, the same as the use of getter beds that would also have the problem with the oxygen present in the gas flow.

4.6.1.4. PEG separation. For PEG separation, continuous technologies like cryogenic distillation and zeolite membrane cascades have been considered to be viable. The best score with Xe and Ar as PEGs has been obtained by cryogenic distillation, as this is a simple and well-known separation method that is much more robust and technically less challenging that membrane cascades. Furthermore, this method can be adapted more easily to the flow rates that must be separated. It must be noted that in the stage of PEG separation, the tritium content is very low, so that the inventory in cryogenic distillation does not play a major role.

4.6.2. Selected R&D achievements

EPS Pre-Concept Design. The direct internal recycling loop DIRL reduces the load on EPS as a whole, and the quantity of Q2 to be processed and recovered in the fuel recovery subsection.

Before the first stage, fuel recovery must take place using permeator technology since it reduces both the tritium inventory and the sizing for the other subsystems of exhaust processing. A Pre-Concept Design of this system has been made based on palladium silver membranes in circular arrangements of 500 mm length and 10 mm diameter at a lumen pressure of 200 kPa and a shell pressure of 1 kPa [61]. It was found that a relatively small system of a two-stage arrangement where the retentate of the first stage is feeding the second stage was sufficient to provide an overall efficiency of well above 99% for the expected range of DEMO throughput. Additionally, multiple permeators are operated in parallel to increase the membrane area and thus the total achievable throughput. For the current DEMO baseline scenario without using nitrogen as a PEG, 31 permeators are needed for the first stage of hydrogen extraction, and 4 permeators for the second set. The tritiated impurity processing stage via converters is integrated between the two permeator stages. PEG separation will be done as the final step of EPS. To place tritiated impurity processing upstream requires that it has to be sized for handling PEGs as the physical presence of PEG molecules does reduce the efficiency. On the other side, this configuration supports the performance of the downstream PEG separation technologies considering that there are only negligible amounts of tritium, hydrocarbons or water present. For the expected amounts of PEG, we consider the positioning of PEG separation as last step to be the best compromise.

Experimental characterisation of membranes for PEG separation. Tests have been undertaken using a purpose-built experiment to investigate the diffusion and selectivity of gases through porous membranes [62,63]. The obtained selectivity values were still low, so that membrane cascading would be necessary, if this technology were to be chosen.

4.6.3. Open issues and questions to be addressed in CD phase

Currently, the development status of EPS suffers from missing information and decisions on the feed streams, which is outside the responsibility of the fuel cycle. Once this information is made available, concept design work on known technologies can readily start to fix the lay-out of the whole EPS architecture.

Exhaust monitoring requirements technology. As a lesson learnt from JET operation, it is important to be able to monitor exhaust gas exiting the tokamak to detect any hazardous species that may form unnoticed in the tokamak (e.g. halogens). The requirements for such a subsystem, technology options and a proposed concept design are all required.

4.7. Isotope rebalancing and protium removal

A fuel cycle normally contains cryogenic distillation to separate the incoming hydrogenic gas stream into the different pure hydrogen isotopes. However cryogenic distillation suffers from the inventory issue associated with the liquid hold-ups, which can negatively affect safety and operating costs and is responsible for unwanted long processing times. At ITER, a total of four distillation columns and several equilibrators are envisaged to provide the necessary flexibility. A large number of stages (of the order of 100) is needed for the columns, which makes the operation complex and costly.

The required functionality in DEMO is different, there will be a repetition of the same plasma discharge, and this is sufficient to mainly provide the fuel mixture, rather than separating and then combining the isotopes again. This situation brings back a number of technologies that have been discarded at ITER, because they do not achieve the purity requirements that cryogenic distillation can achieve, but have the advantage of shorter processing times. This approach has resulted in having an additional dedicated system block for isotope rebalancing rather than separation and leaving the classical isotope separation function for detritiating streams with only trace tritium.

4.7.1. Technology selection

The isotope rebalancing and protium removal (IRPR) system in DEMO is a system block upstream of the classical isotope separation block; it has to provide the capability to induce a shift in the D/T isotope ratio of the feed stream taken from the fuel recovery subsystem of EPS, as well as remove protium. The produced fuel stream with adjusted mixture composition directly feeds the DIRL gas collection and buffering system, as does the permeate gas stream from fuel separation, so that the required composition for fuel injection is reached. It is a requirement to reduce the fuel protium content to below 1%. This requirement is design driving and a certain fraction of the flow from fuel recovery may not be treated in IRPR but bypassed and recirculated. Due to this flexibility, IRPR is the main system that allows the fuel cycle to slowly adjust the nominal mixture composition, if requested. In the DEMO fuel cycle

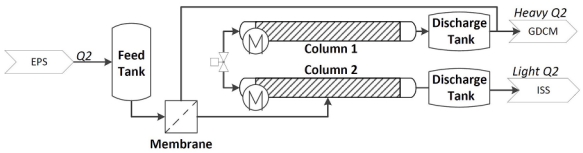

Fig. 13. Pre-Concept configuration of the IRPR system.

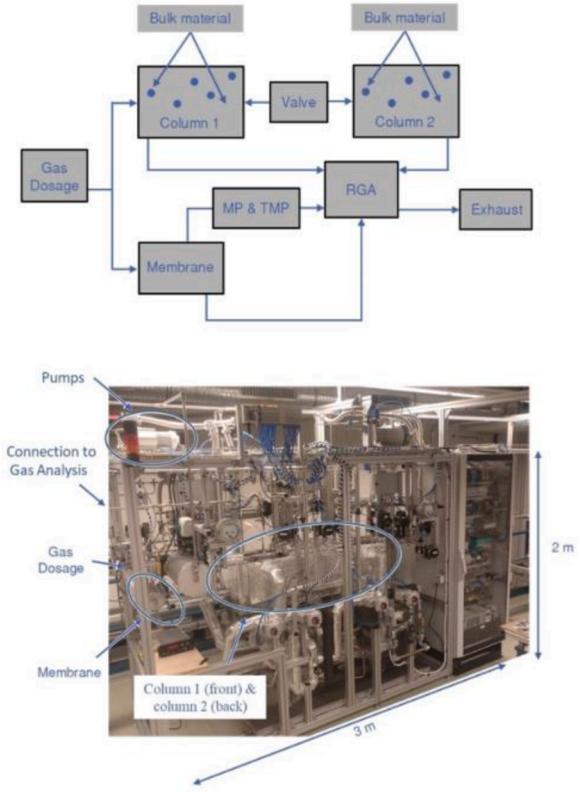

Fig. 14. MC-TSA demonstration facility HESTIA.

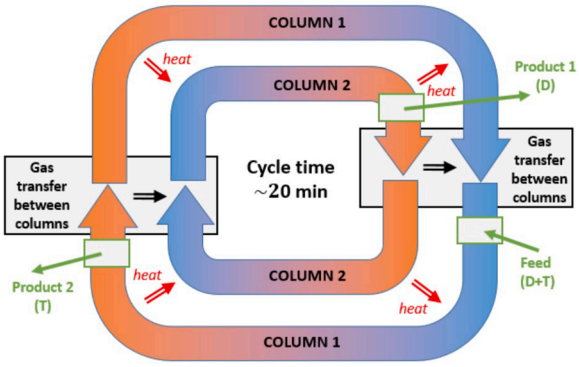

Fig. 15. MC-TSA process scheme.

development programme, the two IRPR functionalities were initially considered independently. It was later concluded, that both tasks are best handled by the same technology [64].

Considered candidate technologies included plasma separation processes, quantum sieving, laser isotope separation, cryogenic distillation or temperature swing absorption. The latter technologies ranked high, whilst the others lacked either a suitable demonstration of past applications in relevant conditions or were found to have an overall low technology readiness level (such as pressure swing adsorption, see also similar ranking results described in [27]). This is often coupled with other drawbacks such as high energy consumption, limited tritium compatibility, or very high complexity.

Out of all options, a newly tailored process, the Membrane-Coupled Temperature Swing Absorption (MC-TSA), as shown in Fig. 13, scored highest due to its very high separation performance, the possibility to operate at moderate temperatures and pressures and good tritium compatibility.

MC-TSA can be considered to be an improvement of the thermal cycling absorption process initially developed at Savannah River [65]. The process results in two product streams, one enriched in the heavier isotopes, the other enriched in the lighter isotopes, and employs two columns filled with metallic sorbent material with opposite isotope effect that establish a concentration profile across the length of the column if the gas is thermally cycled between the two columns. Gas is fed via a zeolite membrane which further enhances the isotope separation efficiency based on optimized usage of isotope effects. In the novel configuration, further R&D is required in the selection and optimization of active materials and optimal operating conditions giving it a comparably low TRL. As a fall-back solution we also developed a concept for an IRPR system block based on cryogenic distillation [64].

4.7.2. Selected R&D achievements

The performance and sizing of the MC-TSA system depends strongly on the absorption materials used. R&D efforts are under way to identify and characterize suitable materials for TSA [66].

Demonstration of the MC-TSA process. The TSA process coupled with a membrane stage operating at the desired temperature and appropriate materials is a new development. Hence, a test facility was constructed which allows the study of detailed operational process parameters and corresponding sensitivities. The facility HESTIA (Hydrogen Experiments with Temperature Initiated Absorption) is shown in Fig. 14.

The TSA process was demonstrated to work well along the scheme drafted in Fig. 15. The separation principle of the TSA is quasi-continuous. This means it works batch-wise with closed and open cycles but results in steady state in a continuous output of the processed gas portions. A detailed description can be found elsewhere [66].

The correlation between the separation efficiency and the associated effort, such as the number of cycles, and indirectly the tritium inventory, is the main figure of merit. The efficiency of the separation increases strongly with the increasing number of cycles and then evolves asymptotically, whilst the inventory is growing linearly so that an optimum must be identified.

Material characterization. The process combines two materials with opposite isotope effect, one that absorbs preferentially the lighter isotopologue, and the other which has a higher affinity to the heavier isotopologue. To reduce cycle times, they should both have corresponding properties in a temperature range between ambient and 200°C and at pressures between rough vacuum and max several bar. In order to find a good pair of materials, the thermodynamic equilibrium data of absorbed hydrogen (pressure-concentration diagrams at varying temperatures for the resulting alloy phases depending on the concentration range) must be known as well as additional kinetic effects. For this purpose a custom Sievert apparatus (Material Investigations for Absorption, MAIA) was developed and constructed. For the material which preferably absorbs lighter isotopes, Pd coated on Al_2O_3 spheres has been shown to work. For the complementary material with opposite isotope effect, a $Ti_{40}Cr_{60}$ alloy was successfully characterised.

4.7.3. Open issues and questions to be addressed in CD phase

The two facilities only became available in the end of the PCD Phase. There will be a need to further exploit them for process optimization.

Continued material development and characterization. Although the principal feasibility of the process could be successfully demonstrated, both materials show improvement potential. As regards Pd, alternative materials with similar isotope effect at lower cost would be highly desirable. Similarly, the investigated TiCr alloy did show issues regarding stability. A future material is expected to have fast ab/desorption and hydride formation kinetics at high embrittlement limits that would provide a large operational window and long lifetime of the material. A detailed materials development programme has been launched with support of industry.

Optimization of process parameters. A parameter with significant improvement potential compared to the current situation is the heat

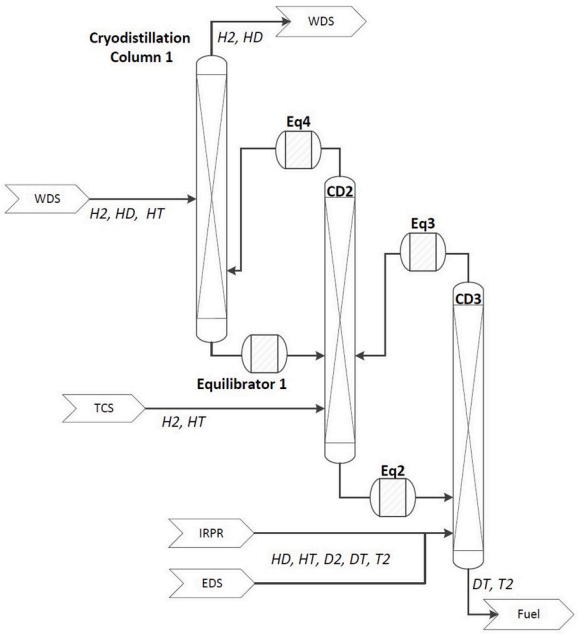

Fig. 16. Pre-Concept Design proposal for the isotope separation system.

Table 3
Column design parameters of existing cryogenic distillation systems used for hydrogen isotope separation as well as values adopted for the PCD Phase in DEMO.

	ITER [67]	JET [68]	TSTA [69]	Used Value
HETP (cm)	5.5	6	5	**5**
v_{gas} (m/s)	0.125	0.05-0.1	0.089	**0.1**
h_l (vol%)	10	14.5	10-15	**15**

transfer, which finally limits the possible column size and packing. High heat transfer coefficients would pave the way to reduced cycle time, number of cycles and tritium inventory. Although the currently existing model developed on the Aspen® Modeling Platform was able to predict the basic dependencies, it will have to be expanded to include kinetic effects of isotopologue adsorption and transients for the heating and cooling cycle.

4.8. Isotope separation

For a fuel cycle architecture with a separate IRPR block, the isotope separation system (ISS) receives a pure hydrogen stream containing trace amounts of tritium from WDS, a higher tritiated hydrogenic stream from TCS, as well as the protium-rich waste stream from IRPR and G-DS. The Q2 therein is purified to fuel quality by removing excess protium and deuterium which also requires the treatment of mixed isotopologues (primarily HT). This product stream is then fed to a short-term storage vessel as part of fuel storage from where it can be used to compensate for burn-up or sent to storage beds. Streams stripped of tritium are sent to WDS for final detritiation before discharge.

4.8.1. Technology selection

The following technology candidates for fulfilling the requirements of ISS were assessed through literature review studies: Ad/absorption-based separation, membrane-based separation including quantum sieving effects of metal-organic frameworks, liquid/vapour contacting with isotope exchange reaction, ionisation and ion separation, thermal diffusion techniques, and cryogenic distillation.

The leading results of the technology selection for ISS are summarised in Table 2. Cryodistillation remained as preferred and established technology with a quality rating of 74%. It can process continuous high input flows but cannot achieve very high purities without having a large inventory. However, this is not a problem as the production of very pure gases is not a requirement: In the current configuration, ISS does not produce pure D_2, and tritium is extracted as DT. Protium is routed via the WDS for final detritiation which implies that it must not be ultra-pure. Similarly to the ITER ISS design, isotopic equilibrators are used between columns to promote the formation of pure isotopologues.

4.8.2. Selected R&D achievements

ISS Pre-Concept Design. A pre-conceptual system layout that meets the requirements outlined above has been developed. Fig. 16 shows a potential design for an ISS system, consisting of a set of cryogenic distillation columns (CD1-3) and isotopic equilibrators (Eq1-4).

The first column serves as a bulk protium removal column optimized for high throughputs instead of purity, with the primary goal of reducing the flowrate on subsequent columns. The top product of H_2 and HD is then sent to WDS for final detritiation. The second column receives the bottom product of the first together with the input stream from TCS, removing H_2 and HD as top product which is refluxed into the first column, and concentrating HT, D_2, DT, T_2 at the bottom. This product stream is sent to the third column, where also the IRPR waste stream is introduced and T_2 and DT are enriched as final product. The top product of HT and D_2 is refluxed into the second column. Isotopic equilibrators are used between all columns to break up HT and DT mixed isotopologues to improve the achievable separation. Design data for the columns have been found to have a considerable impact on the achievable performance and required sizing of the columns, directly affecting their tritium inventory. Most important here is the Height Equivalent of a Theoretical Plate (HETP) affecting the required column height, the required vapour velocity v_{gas} in the packing to avoid drying or flooding dictating the column diameter as well as the specific liquid hold-up h_l of the packing directly affecting the tritium inventory. Table 3 lists these design parameters for isotope separation of existing hydrogen isotope separation systems based on cryogenic distillation. Based on these design values the system design will be progressed into the CD Phase.

4.8.3. Open issues and questions to be addressed in CD phase

The following work is envisaged in the CD Phase of DEMO. These activities are required to address the major technical, integration and manufacturing risks associated with the proposed ISS design; significant additional activities may be needed to adequately demonstrate concept maturity even after 2027.

Modelling. Construct a complete steady state model of cryodistillation, which will require literature data from experiments, but provides the first proof that the proposed system can fully meet the system requirements. Subsequently construct a full dynamic model of complete system implementation. Some dynamic instabilities in the feed flows to this system are envisaged, particularly in the event of a Loss of Vacuum Accident (LOVA). Therefore, dynamic modelling will be needed

Table 4
EDS and DS systems considered in the fuel cycle.

System	Served client(s)	Main gas species
HVAC-EDS	Heating, Venting and Air Conditioning systems	Wet air
G-DS	Residual gases of EPS PEG removal system	He, PEGs
O-DS	Oxygen from WDS	Oxygen
AMF-EDS	AMF Heating, Venting and Air Conditioning systems	Wet air
GB-DS	Glove boxes in tritium plant	Argon
SVS-DS	Service Vacuum System	All gases
HVAC-SEDS	Safety system of HVAC-EDS	Wet air
GB-SDS	Safety system of GB-EDS	Argon

Table 5
EDS technologies and system loads.

System	Technology	Approx. Flowrate	Duration
HVAC-EDS	Wet scrubber	~ 10 000 Nm^3/h	Always active but load may vary during burn/dwell.
G-EDS	Getter beds	< 1 Nm^3/h	
O-EDS	Molsieve dryer	~200 Nm^3/h	
AMF-EDS	Wet scrubber	6000 Nm^3/h (+ Outflow from SVS-EDS)	
GB-EDS	Molsieve dryer	some 100 Nm^3/h	
SVS-EDS	Molsieve dryer	< 100 Nm^3/h	
HVAC-SEDS	Wet scrubber	To be determined	Active in case of emergency, otherwise stand-by.
GB-SEDS	Wet scrubber	To be determined	

to demonstrate that safety and integration criteria are met.

Experimentation. Construct scaled-down implementation of complete system and run bench scale and pilot plant studies, both inactive (protium and deuterium) and active (tritium based) to allow extraction of model parameters (e.g. updated values of Table 3). The H3AT facility (see Section 5.3.2.) is promising to provide the opportunity to perform such testing in the second half of this decade.

4.9. Exhaust detritiation

Here, systems that interface directly to the stack (Exhaust Detritiation System, EDS) are distinguished from systems that only indirectly interface the stack (Detritiation System, DS) via an EDS. EDS systems require the use of wet-scrubber columns as final barrier, whereas DS may utilize different technologies to avoid the generation of tritiatied water. The purpose of the EDS is to capture any tritium, from waste process gas or ventilation (HVAC) systems and convert into tritiated water. This tritium is prevented from being released or discharged from the facility into the environment and will instead be made available for re-use. The sources of tritium carrying gas are various and, hence, a number of EDS and DS systems are needed to serve all clients. The EDS also contributes to maintaining dynamic confinement for relevant parts of the DEMO plant, providing a depression ensuring any leaks are inwards to areas of higher contamination.

Table 4 lists all systems considered for the DEMO fuel cycle. The by far largest systems are HVAC-EDS and AMF-EDS. Note that each system will comprise a number of (parallel and identical) units, whose number will be defined during the CDA.

The discharge of tritium (and indeed all radioactive and otherwise toxic / harmful products) must be minimised in accordance with the ALARA principle and must as a minimum requirement be below the legal discharge limits. The EDS system is a key part of ensuring this requirement is met.

4.9.1. Technology selection

In the PCD phase, work has focused mainly on the largest systems, namely HVAC-EDS and AMF-EDS. For these two systems, a ranking has been elaborated and is given in Table 2. As data for DEMO are not yet available, the values were estimated using a linear scale-up of corresponding ITER data, primarily from HVAC and Hot Cells, yielding an overall processing flowrate during normal operation of approximately 12000 Nm^3/h. When also considering the differences between DEMO and ITER not captured by linear scale-up (such as much larger heat transfer systems and extensive tritium breeding systems), the following species and flowrates (given in norm volumetric units related to 1 atm and room temperature) are anticipated, see Table 5.

Table 6
DEMO discharge limits.

Release source term	Demo release design objective(g/year)	Demo release design objective(TBq/year)
Tritium as HT	9	3333
Tritium as HTO	0.9	333
Total	9.9	3666

At JET and at ITER, EDS ensures detritiation of air and waste process gases by catalytic oxidation to produce tritiated water vapour, and it has been decided to rely on the same approach at DEMO. At JET, free hydrogen species are converted to water vapour in a low-temperature recombiner (150°C) vessel packed with a platinum catalyst. Compounds such as hydrocarbons are broken down and converted to water vapour in the high-temperature recombiner (500°C). All water vapour must be removed from the gas stream prior to discharge. For this process step, a ranking analysis was performed, as shown in Table 2 above.

Water Capture with molsieve dryers packed with 5Å molecular sieve is a technology used at JET. However, experience at JET points to issues [70]. For ITER molsieve technology was disqualified because of severe safety and reliability issues. Also the amount of molsieve and energy required for cycling the dryers becomes very large. The alternative are wet scrubbers [71] which have advantages over adsorption dryers in terms of detritiation, operating costs and energy consumption. In this technology, isotopic exchange occurs between the tritiated water vapour and liquid water fed into the column containing structural packing. Wet scrubbing columns have the advantage that they can be operated continuously and may produce less tritiated water for processing than molsieve dryers. As general rule, we have considered that all gases that are being discharged through the stack have to pass a wet scrubber column, fed with purified and non-tritiated water.

For some applications where no discharge to stack is required and no oxygen will be present, a solution based on getter materials is also an option and has been considered in the technology selection process.

4.9.2. Selected R&D achievements

Requirements for DEMO EDS. Two important parameters related to EDS are: (i) the allowable discharge limits and (ii) the achievable decontamination factor (DF). Knowing the incoming source term to EDS and its DF, it is possible to confirm whether the discharge limits will be respected. The following release design objectives, detailed in the DEMO Plant Safety Requirements Document, are proposed during normal operations for DEMO [72], see Table 6. A detritiation efficiency of greater than 99% is set as a design constraint for DEMO detritiation systems. It is expected that different DF numbers will be set for different subsystems and plant states; for example ITER has set DF > 1000 for vent detritiation, whilst secondary containment and room detritiation DF is set at >100. Further to these figures, all detritiation systems should reduce

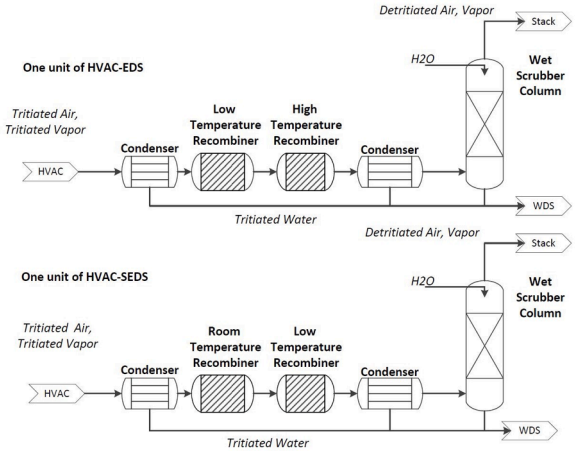

Fig. 17. Example of unit architecture of HVAC-EDS/SEDS.

tritium in accordance with the ALARA principle.

EDS outputs are essentially two streams: (i) detritiated gas, mostly air but also argon, oxygen, helium from the fusion reaction, and possibly unwanted PEGs and other impurity gases. Effectively, the flowrate of gas out of EDS is the same as that entering; (ii) tritiated water, to be sent to a water detritiation system. The flow rate dictates the sizing of the WDS. To estimate it, it was assumed that the gas relative humidities and temperatures are as per ITER max values for each EDS subsystem. This is assumed to dominate the water term, with the converted Q2 making up only a small fraction.

The quantity of tritium passing into EDS and therefore WDS is not a well-defined quantity and will certainly vary with time. Out of all current contributions, only the EPS effluent is well known for the current design point, accounting for approx. 0.53 g/h of tritium. Therefore a working design target of 1 g/h for all inputs streams is specified, leaving 0.47 g/h for the remaining contributions. This then would require a DF of at least 1000 to stay within release limits. However this should mark a stringent upper limit and steps should be taken to minimize tritium losses into secondary containment and building areas.

DEMO HVAC/AMF EDS Pre-Concept Design. Each system is comprising a number of identical trains and with a nominal throughput of 3000 Nm^3/h, redundancy aspects have to be considered. As shown in Fig. 17, the safety system configuration would be very similar, but uses only room and low temperature recombiners and no high-temperature recombiners.

The incoming stream first sees a pre-condenser, to remove water vapour at inlet (drain & route to WDS) dropping the dew point of incoming gas / air stream. Then follow two catalytic recombiners: the first one for conversion of diatomic tritium species to HTO vapour, and the second one for conversion of tritiated compounds such as hydrocarbons to HTO vapour. A following condenser (cooled by chilled water) is needed to capture a portion of tritiated water routed to active drains tanks / WDS. Wet Scrubber Columns utilising isotopic exchange will produce further HTO and detritiated gas / air with high DF. Additional to this, each train would contain items not shown such as isolation valves, a HEPA filter at the inlet, process heaters, economizers to improve energy efficiency, instrumentation, and possibly a chemical or acidity control system for the condensate.

Depending on the scalability of technology more trains could be added, but there should always be one spare to account for planned and unplanned maintenance. Wet scrubber columns may be very tall and thus each train may have more than one column fed in series, to ensure more practical building layout dimensions.

Because the S-EDS is a key safety related plant, designed to mitigate the result of fault and accident scenarios on the torus and active / tritium facilities, it must be designed in accordance with the requirements to come in the future from the DEMO safety team to ensure all safety criteria have been considered and met.

4.9.3. Open issues and questions to be addressed in CD phase

The following are topics that should be explored in order to better define and size the DEMO EDS concept design.

Accident analysis for Tokamak Vacuum Vessel and active / tritium facilities, with outputs including source terms, release fractions, etc. This information will be required for sizing the S-EDS system. An impact study of the found accident gas loads on recombiners in S-EDS has to identify optimal recombiner temperatures, balancing high DF versus safe operation.

DEMO building sizing and leak tightness requirements will be needed, in order to properly size the Normal plus Safety Detritiation Systems.

Confirmation of the AMF concept at DEMO, its integration and its EDS requirements. At ITER this system doubles the size of EDS plant (the plants are completely separate).

4.10. Water detritiation

The WDS is a system for the treatment of tritiated water. The WDS will treat water arising from EDS and arising from the tritium extraction recovery and coolant purification systems. A WDS system will typically feed into an isotope separation system (ISS) for final recovery of tritium from hydrogen and deuterium.

4.10.1. Technology selection

The results of the technology ranking are also included in Table 2. There was not a clear winning technology which is due to the fact that the requirements on WDS are complex and very difficult to capture quantitatively in the PCD phase. This situation was taken as justification to further look at both the leading technologies. It is likely that a future concept design of WDS will have to combine different technologies, as will be discussed below.

4.10.1.5. Combined electrolysis and catalytic exchange (CECE). In the CECE process tritiated water is filtered and then fed into an electrolyser to separate into oxygen and hydrogen gas streams. The hydrogenic gas is then transferred into a liquid phase catalytic exchange (LPCE) column, which is operated with clean water in a counter-current direction; isotopic exchange occurs with the outlet hydrogen stream being detritiated and the water enriched with tritium. One fraction of the hydrogen generated in the electrolyser passes through a palladium permeator to ensure purity and feeds an ISS cryodistillation column for isotopic separation. CECE technology is commercially available. CANDU reactors operated in Korea use CECE for water detritiation, as does ITER.

4.10.1.6. Water distillation. Water distillation columns can be used as the primary means for detritiation [73] which is well-established for heavy water upgraders used in CANDU reactors and also proposed for Fukushima clean-up. The general drawback, however, is that water distillation columns tend to become very large if high purities are required. This can be overcome if water distillation technology is applied for pre-concentration purposes only.

The principle is based on the distillation of water to produce a concentrated stream of HTO, the detritiated water may then be discharged whilst the concentrated HTO would require storage or further processing to produce tritium fuel, possibly using electrolysis to separate out the HT and oxygen. In this case, some of the arguments presented in favour of water distillation over CECE, such as inherent safety (no flammable gas production), lower tritium inventory and better overall energy consumption then may not hold. In order to produce gaseous HT a small LPCE column could alternatively be used. This scheme feeds clean gas to run an LPCE column, producing a tritium enriched hydrogen gas stream to feed to isotopic separation. The hydrogen from the isotopic separation then feeds back to the LPCE column to reduce the clean feed gas demand.

4.10.2. Selected R&D achievements

Identification of WDS processing requirements. It is expected that the DEMO EDS will provide the main feed of tritiated water into the WDS. Predicting how much tritiated water will require processing under normal operations led to a range of values up to 186 kg water per hour,

Table 7
DEMO WDS inputs.

Water source	Water (kg/h)	Tritium (g/h)
EDS condensate (including hot cells EDS)	150	1.0
CPS	34.4	0.4
TERS	0.2	0.01
Materials Detritiation	1.5	0.1
Totals	186	1.51

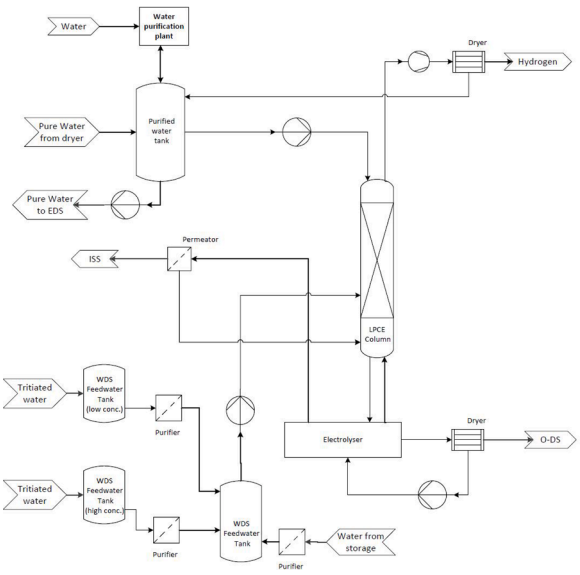

Fig. 18. DEMO WDS CECE process.

including 1.5 g/h of tritium. For reference, ITER WDS is sized to process up to 60 kg/h [74].

There are a number of EDS systems for use during plant incidents, which would further produce a large quantity of tritiated water (with much higher than usual tritium concentrations). It would not be practical to size WDS to handle all possible sources of water simultaneously, therefore it must be assumed there will be a large buffer tank (or tanks) available for temporary storage of tritiated water prior to feeding into WDS. Indeed, the ITER plant has two 90 m^3 HTO storage tanks available for this purpose.

Permeation of tritium into cooling water loops will necessitate cleaning up of cooling water, as described in the following Section 4.11 for the breeder blanket coolant in the case of WCLL. According to preliminary estimations described in [74], the tritium entering the DEMO coolant system will be 0.4 g per day (assuming a permeation reduction factor PRF =100); to remove and recover this amount of tritium, CPS will produce a significant quantity of tritiated water, a figure is given in [75] as 34.4 kg/h water, to which the load from the wet scrubber columns still has to be added, which is possible only once the design has progressed.

Table 7 summarises the inputs to the WDS.

WDS Pre-Concept Design. The proposed process is illustrated in Fig. 18. Tritiated water is fed at a point along with the height of the LPCE column, the exact height being determined by more detailed studies to optimise the isotopic exchange. Demineralised water is fed into the top of the column and the water mixes and travels downwards. The gases and vapours travel upwards through the column in counter current fashion, and the isotopic exchange occurs, with tritium (plus any deuterium) moving into the water and hydrogen moving into the gas

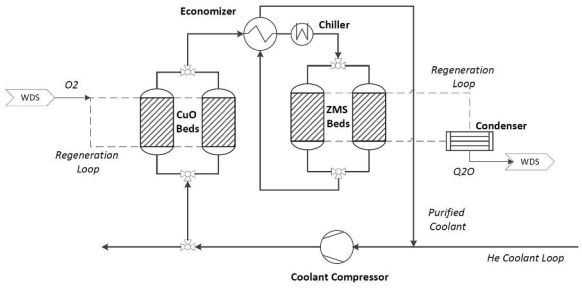

Fig. 19. Conventional helium coolant purification process.

phase. The humidifier, or re-boiler, is used to generate some HTO vapour which is fed up into the column, this promotes the isotopic exchange. In the electrolyser, the concentrated tritiated water is separated into HT and O_2 gas streams, both may contain traces of HTO gas. A portion of the HT (with traces of HTO) is fed into the permeator, which removes any HTO and feeds this back into the LPCE column. The HT gas is then fed to ISS for isotopic separation.

The sizing of the LPCE system to handle 190 kg/h water can be predicted by analogy with another LPCE plant with a similar flow requirement: Wolsong (a CANDU fission plant in South Korea) has a tritium removal facility [77] which largely consists of twin LPCE columns processing 100 kg/h of heavy water, with tritium feed concentrations of 370 to 2,220 GBq/kg. Each column is approximately 20 meters tall.

A preliminary model of WDS LPCE and CECE systems was produced. There, the use of LPCE alone, and LPCE plus electrolysis (CECE) was modelled for individual water arising from EDS, CPS and TCS. The modelling identified detritiation factors of more than 3500 for EDS and CPS, respectively. These figures respect the discharge limits for the hydrogen effluent stream from the top of the LPCE columns. In order to process the throughput of water and tritium, and achieve these good detritiation factors, the fluid must make multiple system passes which can lead to extended residence times and significant steady state tritium inventories.

4.10.3. Open issues and questions to be addressed in CD phase

The available design has to be developed further, with some focus on mixed configurations.

Wet scrubber design. Investigation into efficacy of scrubbing tritiated gas streams through CECE LPCE columns as a means of detritiation (additional to tritiated water loading).

Trade-off studies. This activity will in detail study an integrated configuration with water distillation combined with CECE, considering applicable activity limits for electrolysis.

Safety analysis to evaluate the impact of accidents on system processing and producing large amounts of highly tritiated water.

Water balance. A water balance will have to be set up and used as a basis for (in particular energetic) optimization of the WDS and its related systems.

4.11. Coolant purification

The Coolant Purification System (CPS) has the duty of recovering tritium from the primary coolant loop. The tritium concentration inside the primary coolant has to be kept below a certain value because of safety reasons and to maintain tritium balance. Tritium control inside the coolant is a complex matter and it is achieved by opportunely tuning several parameters, such as: the coolant fraction routed inside the CPS

Table 8
Main characteristics of some reviewed water detritiation facilities.

Facility	Feed stream, kg/h	Dimensions	Process
Darlington Tritium Removal Facility (DTRF)	360	The building is 35 m long, 25 m wide, 12 m height expected for the CD, 38 m height	Vapor Phase Catalytic Exchange (VPCE) + Cryogenic Distillation (CD)
Wolsong Tritium Removal Facility (WTRF)	100	2 LPCE columns with a diameter of 0.6 m, the height of about 20 m each.	Liquid Phase Catalytic Exchange (LPCE) + Cryogenic Distillation (CD)
ITER Water Detritiation System (ITER WDS)	20 (up to 60)	LPCE column of 26 m height, electrolysis cell of 50 m^3/h (120 kA)	Combined Electrolysis Catalytic Exchange (CECE)

(α_{CPS}), the use of tritium permeation barriers (the effectiveness of which is given by PRF), and the coolant chemistry (addition of H_2 and/or H_2O) [18].

The following sections describe the work specifically related to the CPS system. Since the DEMO design still considers different blanket variants with two possible blanket coolant options, helium or water, the processes utilised in the CPS are completely different.

4.11.1. Technology selection

The technology survey and ranking analysis for helium coolant purification technologies is included in Table 2. Molecular sieve and getter bed technologies received the highest score. Although the most common process used in nuclear applications for removing Q_2 from large helium streams relies on the combination of copper oxide and molecular sieve beds, see Fig. 19, getters ranked similarly high because new and promising hydrogen getter materials [40] became available recently. The potential advantage in using getter beds is that the Q2 species are directly adsorbed over the gettering materials without the need for the oxidation stage so that the formation of Q2O and its subsequent treatment is avoided.

For the case where a water CPS is used, the processes reviewed are the same as the ones considered for the water detritiation system, and similarly, CECE and water distillation ranked comparably. Also in this case, the review has been extended to the study of some existing (or under design and construction) water detritiation systems developed for fusion facilities and fission power plants. The most relevant outcomes of this review are illustrated in Table 8, and are related to the feed stream capacity and the dimensions of the facilities. In view of the magnitude of coolant flow (several 1000 kg/s) these numbers immediately suggest that the α_{CPS} for the case of water coolant has to be kept very low to avoid an unsustainable dimension, complexity and cost of the water CPS.

4.11.2. Selected R&D achievements

The pre-concept design of helium and water CPS was performed by following the established procedure: (i) survey of the state of art of tritium removal processes from helium and water, (ii) definition of CPS requirements and interfaces, (iii) identification of relevant input data and/or assumptions, (iv) evaluation of the applicability of the identified processes at DEMO scale. Having reached this point, it was clear that for a helium CPS, the identified processes can be scaled up to fulfil the requirements, especially if the H_2 partial pressure inside the coolant does not exceed the value of 300 Pa. Therefore, the activity was continued by approaching the dimensioning of the main components of the two potential processes: the "conventional" one based on copper oxide and molecular sieve beds and the "alternative" one based on novel Non-Evaporable Getter (NEG) beds. Conversely, in the case of water coolant, the high tritium permeation rate from the blanket in the coolant in case effective permeation barrier cannot provide the required Permeation Reduction Factor (PRF) (~ 37 g/d for PRF=1), does clearly result in an unfeasibly big and complex water detritiation facility. It was therefore concluded that tritium control inside the water coolant systems has to be achieved by opportunely combining a certain PRF value and CPS size. Therefore, simulations were run to identify suitable couples of PRF and α_{CPS} values able to guarantee a tritium content in water coolants below the value of 5 Ci/kg ($1.85 \cdot 10^{11}$ Bq/kg), a target value that was adopted from the heavy water contained in the Calandria of CANDU reactors [78].

The main parameter to base a CPS design on is the coolant fraction to be routed inside CPS. This is given by the tritium balance and depends on the CPS efficiency, the tritium permeation rate from the blanket into coolant, the PRF on the blanket side and the allowable tritium concentration in the primary coolant. Considering the input data and assumptions reported above, the order of magnitude for the by-pass stream in the case of helium results as about several kg/s, while for the case of water the resulting by-pass is of several hundred kg/day. Table 9 is summarizing the input data as used for the PCD activities.

Pre-Concept Design for helium. The "conventional" process chosen is similar to the ones proposed for the HCPB TBM of ITER, thus the adopted technologies are rather mature. It foresees three steps: (i) oxidation of Q2 into Q2O over CuO beds; (ii) removal of Q2O from He using Zeolite Molecular Sieves (ZMS); (iii) treatment of the desorbed Q2O in a metallic reducing bed to transform Q2O into Q2 or, alternatively, processing the desorbed Q2O in the WDS. The layout and dimensions of the main components can be found in [75]. The "alternative" process considered for helium CPS foresees the direct adsorption of the Q2 contained in helium coolant by using NEGs. Several getter metal alloys have been investigated and their Sieverts' plot was assessed at different temperatures [79]. The plots revealed that the alloy named ZAO is the most promising for application to a CPS. Also for this concept, a preliminary evaluation of the amount (kg) of getter required to fulfil DEMO CPS requirements was performed. It is worth mentioning that a parameter with a significant impact on the helium CPS design is the amount of H_2 present inside the coolant. In general, the higher the H_2 content, the bigger the system (and the amount of required packing materials) and/or alternatively, more frequent regenerations are required (i.e. resulting in reduced availability and a shorter lifetime of components).

Pre-Concept Design for water. For a water coolant CPS, due to the technological limit of water detritiation systems in processing large amounts of water and the large tritium permeation rate, the PCD Phase has been undertaken using numerical analysis to evaluate the impact of tritium mitigation strategies (i.e. PRF values from 1 to 1000) on the water by-pass fraction to be treated inside the CPS. Two water CPS configurations were considered: (i) "in-line", where a certain coolant by-pass is continuously routed inside the CPS, (ii) "off-line", after one year of DEMO operation, where the entire water coolant is discharged and treated in a dedicated on-site facility. A relevant result of the analysis is that by assuming a PRF=100, which is realistically attainable [80], both the in-line and off-line CPS would have roughly the same dimensions as the ITER WDS. The process currently considered for a water CPS is the combined electrolysis catalytic exchange (CECE), but further investigation about this aspect is required. When choosing an on-line system, the size could be reduced further by the implementation of a water distillation system for pre-concentration of the WDS feed.

Table 9
Main input data and assumptions considered for the CPS Pre-Concept Design activities.

Parameter	HCPB	WCLL	Comment
T generation rate	320 g/d		Input
T permeation rate from blanket into coolant	0.74 g/d (PRF=1)	0.04 g/d (PRF = 1000) – 0.4 g/d (PRF = 100)	Input from T permeation analysis [18]
CPS efficiency	90%	90%	Reasonable value
Allowable T concentration in coolant	$5 \cdot 10^{-9}$ (*)	$3 \cdot 10^{-6}$ (5 Ci/kg)	Assumptions made from literature review
α_{CPS}	Several 1000 kg/h	Several 10 kg/h	Order of magnitude throughput for CPS

(*) the indicated concentration value corresponding to an HT partial pressure in He-coolant of $4 \cdot 10^{-2}$ Pa.

4.11.3. Open issues and questions to be addressed in CD phase

Definition of the H_2 content inside the helium coolant. The technologies used in both helium CPS options are scalable, however in the case of a high H_2 content inside the coolant the size, the weight and the cost of all the adopted technologies will increase. For the process based on NEG, the H_2 content in helium could be a risk if it reaches the embrittlement limit of the ZAO alloy.

Confirmation of getter performance for trace tritium in a high-pressure helium stream. Regarding the "alternative" solution based on

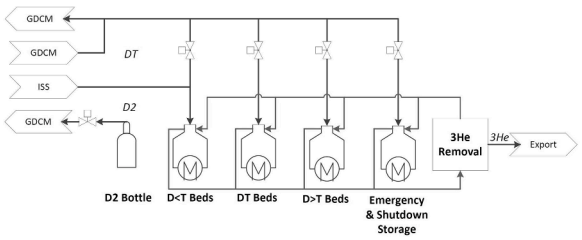

Fig. 20. Hydrogen storage scheme.

NEG, dedicated experimental activities are required to assess the applicability of this technology under DEMO CPS relevant conditions (operation with helium at 80 bar). The NEG technology is also of interest for the tritium extraction from the HCPB blanket, and therefore a joint experimental program will be defined.

Integration of dynamic loads from the conventional helium purification process. The open issues to be addressed in the next design phase are related to the integration aspects for the "conventional" process which have to be faced in agreement with the balance of plant requirements once the design will reach a more mature level. It also leads to strongly varying loads that are difficult to integrate in the OUTL.

Water distillation. For the water coolant CPS in both variants (in-line and off-line) a refined study shall be made specifically comparing CECE technology against water distillation. Concerning technical readiness, it has to be considered that the CPS designed for the ITER WCLL TBM program does not presently foresee tritium removal to be required for the case of helium coolant. This means that WCLL BB of DEMO is the first example in which tritium is removed from water coolant. A useful reference to be exploited is the experience in CANDU reactors, even if heavy water is used there.

Inclusion of divertor and first wall (water) coolant streams. The work so far has been limited to the blanket coolant water stream, and the amount of tritium that permeates from the first wall and the divertor has not yet been considered.

4.12. Storage

The primary requirement placed upon the storage system is the ability to safely store the free tritium inventory, under either accident conditions, during start-up and shut-down, or when the reactor is under maintenance. Excess tritium will also need to be buffered or stored in preparation for export to future reactors.

4.12.1. Technology selection

The most common, safe and well-established technology (also used at ITER) is not to store tritium in gaseous phase, but bound on depleted uranium beds. Tritium is stored as a tritide (UT_3), which readily forms at atmospheric temperature and pressure through an exothermic reaction with the uranium. Systems are typically constructed from stainless-steel vessels, and consist of primary containment (in which tritium gas can be passed over the uranium), secondary containment, cooling arrangements to remove heat from exothermic absorption and electric heaters to drive off the tritium from the uranium tritide at temperatures of circa 430°C.

There are however some issues with the technology that should be considered:

- The technology is not easily scalable to larger quantities of tritium, resulting in the requirement for many parallel beds.
- Uranium is a nuclear material and is therefore subject to various requirements and regulations, including regular inspections.
- Depleted uranium beds can be socio-politically sensitive; some countries and members of the populace are opposed to the use of depleted uranium, which may limit the placement options of DEMO if such beds were used.

Although there are these drawbacks to the use of uranium, they have not been deemed significant enough to warrant active research into alternative techniques. Extensive R&D would be required in order to bring any alternative storage method to the similar TRL for uranium beds. Hence, uranium getter bed technology was kept as the default solution. The final storage system is comprising many beds for different compositions and buffer vessels for operation. Fig. 20 is illustrating a generic configuration of the storage system block.

A review of European legislation revealed that the regulatory control surrounding the use of depleted uranium in the DEMO tritium plant is unlikely to add complexity to plant operations and will not preclude any EU countries from being the potential location for DEMO.

4.12.2. Selected R&D achievements

Social acceptance of depleted uranium. Research has been conducted into the public perception of the use of depleted uranium in fusion [81]. This work involved focus groups and questionnaires in both the UK and Germany.

The findings of this work indicated that many people automatically affiliate "depleted uranium" with negative thoughts and images. This negative branding would appear to have had a noticeable and detrimental impact on evaluations of nuclear fusion technology overall within both the UK and Germany when those interviewed were first informed about the use of depleted uranium in fusion power plants. However, it was also found that subsequent discussions explaining the actual nature of depleted uranium helped to dissipate concerns to some extent. This demonstrates the importance of the fusion community engaging in a continuous effort to educate the public.

4.12.3. Open issues and questions to be addressed in CD phase

Uranium bed technology has been identified as the most appropriate for tritium storage at DEMO. There is very little outstanding work required to take the hydrogen storage system to conceptual design. Once accurate tritium inventory figures are known it will be possible to approximately size the system. This should ensure that the number of U-beds required is feasible.

4.13. Tritium conditioning

Tritium conditioning is required to prepare the effluent streams of the tritium extraction systems for isotope separation. Here different tasks arise depending on the employed blanket concept and its chosen extraction technologies. These tasks are however very similar in their nature to tasks handled by other systems in the fuel cycle. It is therefore assumed that the same technologies can be employed. In the following the requirements and possible technologies are briefly described.

4.13.1. Tritium removal (HCPB)

The purge gas circulated through the HCPB for tritium extraction is treated in the HCPB tritium removal system as described in [76]. This system in turn produces a number of streams from the regeneration and purging of the reactive and cryogenic molecular sieve beds:

- 1-3 Nm^3/h of Q2 from isotope exchange of the reactive molecular sieve beds,
- 10 Nm^3/h of Q2 from regeneration of the cryogenic molecular sieve beds,
- 0.2 kg/h of excess tritiated water from regeneration of the reactive molecular sieve bed in TERS.

The first two streams of the list above are primarily hydrogen isotopologues that require isotope separation. However, since cyclic beds are used, helium contamination from the switchover of the beds is

expected. This contamination must be removed in the tritium conditioning system. The task is inherently similar to that of EPS, namely the removal of small amounts of inert gases from a hydrogen stream, therefore it is assumed that Pd-Ag permeators are suitable as well. The excess water is directly routed to the WDS.

4.13.2. Tritium removal (WCLL)

The tritium extraction system for the WCLL blanket is described in [82]. For the case of Gas Liquid Contacting (GLC) a stream of primarily helium containing protium and tritium is sent to the tritium plant where the hydrogens have to be separated and the helium stream returned to the tritium extraction system. Currently this is estimated to be in the order of 3500 Nm^3/h helium containing 0.1-0.5% of hydrogens. This separation task is very similar to that of a He-CPS and can be provided using the same technology with R&D efforts in those areas benefiting both applications. Additional T extraction technologies are under investigation, such as the Permeation Against Vacuum (PAV). This technology allows extracting T from PbLi without using a stripping gas like Helium simplifying considerably the interface with the fuel cycle system [6].

4.14. Other system blocks

Fig. 3 shows two more system blocks that are not discussed in detail above, because their functionality is very similar to other systems or can be covered by off-the shelf solutions. These systems are the Auxiliary Vacuum Systems (AVS) and the Stack.

The AVS comprise Conventional Vacuum Systems (CVS) and the Service Vacuum System (SVS). CVS include all vacuum systems that are allowed to discharge directly to the atmosphere as they will never see tritium or other radioactive (activated) gases. Examples are e.g. the insulation vacuum of cryogenic transfer lines. SVS might transport radioactive gases, but only in small quantities and/or in case of safety events, their exhaust would have to be routed to EDS. Examples here are guard vacuum systems for diagnostics or leak detection vacuum systems or the cryostat vacuum system. However, these systems use mainly standard pumping equipment and hence, their design can be shifted to the DEMO engineering design phase. It is important that these systems are considered when designing the building layout and attributing room space and space in the pipe chases because these systems, despite being relatively simple, are huge and distributed all over the DEMO plant.

The design of the stack is dependent on the total release rate of ventilation and process gases and the evaluation of accident scenarios under geographical considerations and regulations such as dose limits applicable to the building site not yet defined. Once selected, the stack height can be defined based on the geographical and (average) weather conditions on site.

Table 10
Total upper bound flowrates and their composition between systems during flat-top operation for the DIRL and INTL. z_{T2} refers to the superficial molar fraction of tritium, calculated as T2 + 0.5(DT+HT). z_{inert} refers to the total mole fraction of non Q2, meaning PEGs, impurities, ash and chemically bound Q2 (e.g. as Q2O).

From	To	Flow $\frac{Pam^3}{s}$	z_{Q2}-	z_{T2}-	z_{inert}-
PI	Torus	380.2	0.999	0.495	5e-4
GP	Torus	50.4	0.99	0.491	0.09
Torus	FS	430.7	0.98	0.49	0.017
FS	DIRL-V	338.5	1	0.5	-
FS	INTL-V	92.0	0.92	0.45	0.08
DIRL-V	GDCM	338.5	1	0.495	-
INTL-V	EPS	92.0	0.92	0.45	0.08
EPS	IRPR	11.1	1	0.495	-
EPS	GDCM	77.6	1	0.495	-
IRPR	GDCM	9.4	1	0.54	-
IRPR	ISS	1.7	1	0.28	-
EPS	PEG	3.4	0.03	0.016	0.97
PEG	EDS	2.9	0.04	0.019	0.96

Table 11
Tritium flows entering the fuel cycle in the OUTL.

From	Tritium flowg/(full power day)	Tritium form
HCPB TES	320	HT with HTO, or DT/DTO
WCLL TES	320	HT
Helium coolant	0.74 [78]	HT
Water coolant	0.4 [78]	HTO
Building areas; secondary confinement; service vacuum	11	HT/HTO

Table 12
System operational tritium inventory values during flat-top operation as evaluated in [83], not including piping between systems.

System	m_{T2}(g)
Matter Injection	421
Torus Vacuum (DIRL + INTL)	78
Exhaust Purifications	6.1
Isotope Rebalancing & Protium Removal	556
Exhaust Detritiation	11
Water Detritiation (WCLL)	35
Coolant Purification (HCPB)	2.8
Isotope Separation	613
Helium Conditioning (HCPB)	<1
Tritium Conditioning (WCLL)	67

4.15. Pre-concept design point

4.15.1. Definition of a design point

A fuel cycle design point is given by the conjecture of its architecture (Section 3), system technology selections and layout (Section 4 above), and key performance indicators achievable thereby, which we introduce as follows.

As is discussed in detail elsewhere [24], a key figure of merit is the tritium conversion rate, defined as the burn-up fraction times the effective fuelling efficiency, for which we assume 0.6%. To avoid protium build-up in the circulated fuel, an amount equal to the time averaged (over burn and dwell phases) protium source term has to be continuously removed by the IRPR system. The total amount that can be removed is limited by its concentration at that point in the fuel cycle. Here we assume an allowable fuel protium content of 1%, as well as a protium source term of 0.15 $(Pa{\cdot}m^3/s)$ (@ 573.15K). Regarding deuterium imbalances placing additional separation requirements onto IRPR, we only consider the effects caused by plasma transport processes [24], and as obtained during dwell pumping. A large additional contributor to a potential deuterium imbalance is the neutral beam heating system if based on a gas neutralizer operated with deuterium (as for ITER). DEMO will only explore ECRH-only scenarios in the CD Phase, which is welcomed from fuel cycle point of view. This is the reason why neutral beam injection will not be considered at present. In consequence, we size IRPR to continuously process a total of 2.5% of the torus exhaust hydrogens ($\sim$11 $(Pa{\cdot}m^3)/s$).

The target DIR ratio is specified to be 80% of the exhausted hydrogen flows.

For this operating point, the flat-top flowrates between systems in the DIRL and INTL (neglecting the D_2 feed to replace burnt fuel), as

Table 13
Loop equivalent residence times for the three loops of the fuel cycle.

	DIRL	INTL	OUTL
Residence time (h)	0.25	1.7	38

given in Table 10 have been evaluated and are thought to constitute an upper bound.

Flows in the OUTL span a much wider range of flowrates, composition, species, and tritium content. Furthermore, many thereof may be intermittent (e.g. purge streams of adsorption beds) or only active when required (e.g. detritiation systems for building areas) as well as exact quantities being dependent on the detailed design and optimization of each system. Nevertheless, an attempt is made to specify an integral tritium balance of the OUTL based on available boundary conditions.

Tritium may enter the OUTL in different forms via four pathways:

1) Trace tritium in gas streams that need detritiation before release to stack via the EDS, including the waste stream of the EPS and IRPR.
2) Tritium extracted from the breeding blankets.
3) Tritium extracted from the coolant via the CPS.
4) Tritium via the discharged stream from the IRPR for further isotope separation.

Table 11 lists the expected daily average streams of tritium received by the OUTL.

4.15.2. Estimation of tritium inventories in system blocks

A detailed derivation of each system's tritium inventory for the design point given above exceeds the scope of this contribution and is instead given in [83] with key results summarized in Table 12 below. Wherever the expected tritium inventory is affected by the blanket choice the maximum of both options is given.

It must be noted that for an all-encompassing calculation of tritium inventories, design information such as volumes and pipe diameters typically only available at the detailed design phase, are required. Inventories reported here, therefore, constitute "system operational tritium inventories" meaning tritium hold-up in systems that are strictly required for their operation, with conservative estimations as to the system design based on literature or their preliminary design as given in Sections 4.2-4.14. Concentration profiles have then been evaluated from the fuel cycle simulator during flat-top or steady state operation.

Out of these, systems for isotope separation are expectedly the largest contributors but also represent the area with the largest uncertainties and optimization potential. It is nevertheless concluded that the fuel cycle as presented here can be operated with a maximum system operational tritium inventory below 2 kg. It has to be noted that the inventory of the first wall and the blanket systems themselves has to be added [18].

Also not considered here are storage inventories as these are driven by requirements for reserve inventories as well as excess tritium production according to the stakeholder requirements (see Section 1.1).

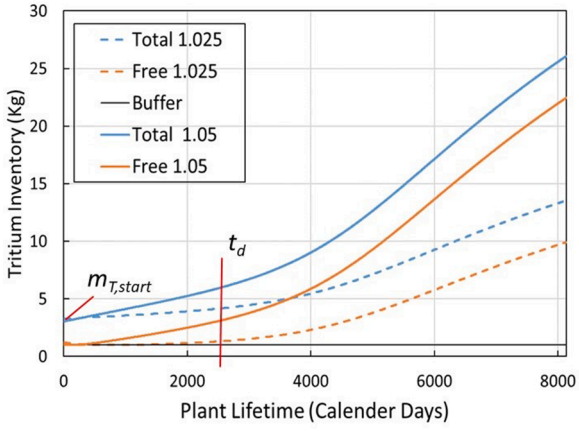

Fig. 21. Evolution of the overall DEMO tritium inventories as well as the free storage inventory for two different TBR values (1.025 and 1.05).

This will be discussed in the next section 4.14.3.

4.15.2.7. Loop equivalent residence times. Residence times for the fuel cycle are best given as integral loop equivalent residence times. They are evaluated as the ratio of tritium inventory in a loop to the total tritium flow entering the loop. Wherever systems share contributions of multiple loops (e.g. matter injection systems receiving tritium from the INTL and DIRL) the inventory is weighted by the relative throughput of this loop. Based on the inventories given in Table 12 and the flowrates given in Tables 10 and 11, the loop equivalent residence times as given in Table 13 have been evaluated.

4.15.3. Tritium lifetime analysis

As DEMO is also tasked with providing enough tritium to start up another FPP, not only the plant operational inventory is of interest, but also the overall tritium balance over the entire plant lifetime. To achieve its goal of tritium doubling, the tritium breeding ratio must exceed unity by a margin large enough to compensate for tritium losses via decay or sequestration and still produce enough tritium to achieve doubling of the start-up inventory ($m_{T,start}$) in a reasonable time (t_d). If assuming negligible losses to the environment the overall tritium balance of the plant given for the amount of tritium N_{T2} then is

$$\frac{dN_{T2}}{dt} = Av(t)\cdot(TBR-1)\cdot P_{th}\cdot e_f - \lambda N(t)_{T2},$$

where λ is the tritium decay constant of $1.7828\bullet10^{-9}$/s, Av the full power plasma availability, P_{th} the thermal power of the plasma in GW and e_f the specific energy of the fusion reaction of 1.3 GW per (Pa•m^3/s) of fused DT. By introducing the plasma availability (defined as the ratio of integral full power plasma duration over total operation time), we consider that decay is occurring all the time while burn and breeding only during plasma operation. The plasma downtime is given by the dwell phase, and further limited by maintenance phases and unplanned outages (failures). It is assumed here that the plasma availability is not constant but rather follows an S-shaped curve over the plant lifetime, starting with lower values during the early life and reaching a plateau as increased operating experience is gained. However, not all tritium on site is available for operation as some becomes sequestrated (trapped) over time e.g., in the first wall or breeding blankets and is considered unavailable. Dedicated studies on these effects are given in [78], which point to about 1.6 kg. Similarly, the circa 2 kg of operational inventories in the fuel cycle can be considered "occupied" as they are required for the normal operation of the plant and cannot be extracted. Any tritium inventory not accounted for by these two inventories can then be

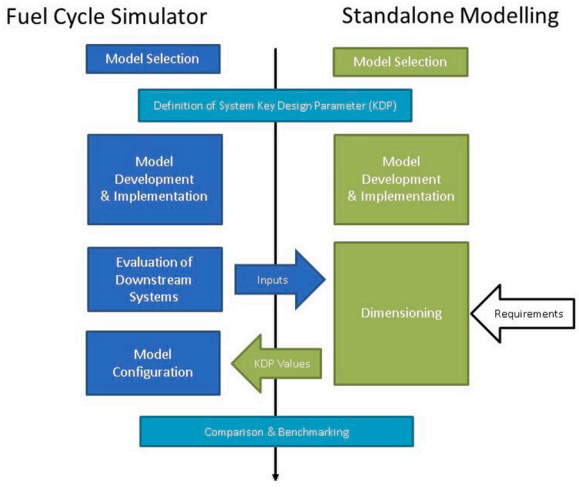

Fig. 22. Schematic illustration of the workflow between the integral simulator and its connected standalone models.

considered "free", or excess inventory that can be used elsewhere. However, from an operational point of view, it is preferably to keep a reserve buffer inventory on hand to allow continued operation or controlled shutdown in case tritium processing systems or parts thereof become unavailable, assumed to be 1 kg here. Tritium doubling is understood to be achieved once the free inventory equals the start inventory.

Fig. 21 illustrates the resulting tritium inventory evolution over DEMO plant lifetime using the WCLL blanket as example and for two cases of the TBR value; the lower value of TBR=1.025 is found to be the minimum value which just ensures that the tritium inventory provides sufficient operational inventory in spite of decay, sequestration and low availability, and never falls below the buffer inventory reserve.

During the early plant lifetime the accumulation of excess tritium proceeds slowest. For the default assumption of TBR=1.05, an average full power plasma availability of 30%, and a total of 1.6 kg of tritium being trapped over the lifetime (corresponding to the use of a WCLL BB), tritium doubling is nevertheless achieved within 6.8 years (indicated by t_d in Fig. 21) while requiring a total start inventory $m_{T,start}$ of 3.0 kg, of which 2 kg correspond to the fuel cycle operational inventory and 1 kg as reserve. As expected, a lower TBR has to be paid with higher start-up inventories, we found 3.27 kg for TBR=1.025, and a doubling time of 12.8 years.

If the tritium breeding ratio is kept constant, large amounts of excess tritium will be generated until the end of the DEMO lifetime, potentially allowing for a relaxation of the tritium breeding ratio in later stages. This could be established by introducing dummy blankets in one of the planned exchange cycles. If lower tritium breeding ratios are encountered, the early life becomes especially critical, as in this case tritium sequestration rates and decay losses may exceed the excess tritium generation rate. As shown in the example above, additional 0.27 kg are needed to offset this and maintain operability in the early plant lifetime.

5. Integration towards a concept design

Single system block technology choices cannot be the only input to the development of the DEMO fuel cycle. Gradually, the system blocks have to be integrated and, finally, a holistic view has to be established requiring a descriptive software tool.

5.1. Fuel cycle simulator development

Due to the nature of the pulsed tokamak operation, as well as different time constants in the individual loops, process modelling on a holistic fuel cycle level has been identified as indispensable. Development of such a tool was started in 2017, initially focused on the DIRL and INTL, and now is expanded to also include the OUTL.

The Fuel Cycle Simulator (FCS) is a self-built software developed with the Custom Modeller at the commercial Aspen platform. It is focused on building a mirror image of the Fuel Cycle that can be used for integral analysis and evaluations of its transient operation. For this purpose, it aims to incorporate dynamic models of all major unit operations in each subsystems with connections arranged to represent the fuel cycle architecture. Wherever possible, models shall be physics based and able to accurately reflect the system performance as a function of the models dimensioning and operating parameters. These parameters that govern the behavior and performance of the models are called System Key Design Parameters (KDP). It is therefore necessary at the start of the model development to agree on the best suited KDP set. Once the models in the FCS are configured with a given set of KDP values, comparison of the modeling results can be performed. The overall workflow is shown in Fig. 22. On one side, FCS has to provide the input from downstream systems in a pre-defined format usable by the system designers. On the other side, standalone modeling has to provide the numerical values of the chosen system key design parameters to FCS in a pre-defined format.

For the upcoming CD Phase the fuel cycle simulator will be closely coupled to the system design, using a consistent architecture and set of design parameters. The two-layer structure of FCS with a central software that is talking with detailed models for each system block will avoid at more advanced phases in the project the convergence times becoming excessive.

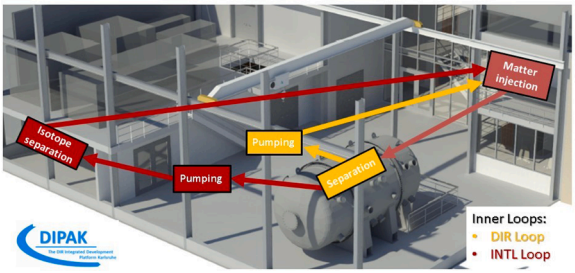

Fig. 23. DIPAK: Replication of DIRL and INTL around a central test vessel.

5.2. Tritium management & control

The purpose of the Tritium Management & Control (TMC) system is to account for tritium at various locations and at all times throughout the fuel cycle, meeting requirements regarding non-proliferation, safety, environmental protection and plasma and process control. Such requirements impose the need for tritium detection at specific locations in the fuel cycle. Moreover, plasma and process control imposes requirements on detection in several other locations in the fuel cycle.

We have performed a literature review, collating information about the main candidates for tritium detection technologies in the DEMO fuel cycle. These technologies are ß-induced x-ray spectrometry (BIXS), calorimetry, liquid scintillation, gas chromatography, Raman spectroscopy, mass spectroscopy, ionisation chambers and proportional counters. We have considered their suitability to detect tritium in the conditions that are expected in measurement locations, and gaps have been identified in particular for real-time online analysis. As each measurement technology is only applicable in specific conditions for specific measurements, with specific accuracies, not all locations in the fuel cycle can be served by this set of detection technologies. One also has to look further into the integration of plasma diagnostics in the TMC strategy. It is one of the key goals of the upcoming CD Phase to elaborate and verify a coherent TMC concept.

5.3. Key technology demonstration facilities

5.3.1. DIR Integrated development platform Karlsruhe (DIPAK)

DIPAK is a dedicated development platform (a synergistic combination of facilities and scientific infrastructure) on a pilot plant scale for integrated testing and validation of the novel inner loop architecture of the DEMO fuel cycle. It will be built at KIT and address the first-of-a-kind technologies that are necessary to limit the DEMO fuel cycle tritium inventories to an acceptable level. These new technologies (Direct Internal Recycling using superpermeation/metal foil pumps, continuous tokamak exhaust pumping using mercury diffusion and mercury ring pumps, MC-TSA for IRPR, and highly repeatable pellet injection) require a facility to increase maturity and technical readiness to levels the other more classical technologies already have. For this purpose we have established a partnership with the industry for each of the new technologies. DIPAK does also integrate PET, the European pellet engineering test bed.

The facility DIPAK is one of the large investments under EUROfusion in the coming Framework programme. It mimics the inner loops (DIRL and INTL) of the DEMO fuel cycle, as shown in Fig. 23. It can replicate the operational phases of DEMO (throughputs under burn and dwell conditions, processing of hydrogen mixtures though without tritium).

The DIPAK experimental programme will be organized around three technology lines and finally culminate in the integrated operation of them:

- Fuelling and plasma exhaust: Demonstrate continuous operation at minimum pellet scatter and highest efficiency.
- Fuel separation and clean-up: Demonstrate the requested separation ratio and purity and continuous adjustment of isotope rebalancing and protium removal.
- Continuous vacuum pumping: Demonstrate high throughput performance over the full pressure range.

5.3.2. Tritium advanced technology (H3AT) facility

This facility will be constructed and operated at UKAEA Culham Centre. It is a timely opportunity to provide the tritium development capabilities required to deliver proof of concept demonstrators of DEMO fuel cycle unit operations. There will be an on-site tritium inventory of 100 g, shared among the different subassemblies. To develop and test novel technologies for DEMO and beyond is one of the prime objectives of H3AT.

H3AT has proposed providing EUROfusion with access during the DEMO CD Phase, providing the necessary experimental opportunities as required by TFV. The H3AT facility will potentially have the capability to support all areas which require performance testing under tritium in cases where extrapolation from H/D is considered to add too much uncertainty (as the scaling is not properly understood, or the model does not come with sufficient fidelity and validation), such as for MC-TSA or launching of DT pellets. In this regard, H3AT is complementary to DIPAK.

The H3AT facility provides two parts, related to each other. One is a complete tritium plant emulating the ITER configuration, to de-risk the ITER project; the second is a complement of tritium buildings that can use the ITER part as infrastructure to perform R&D tasks.

6. Conclusions and perspective

In the 2014-2020 period the TFV team has elaborated a fuel cycle architecture that is adapted to the DEMO requirements. It is based on three loops to reduce the tritium inventory to a minimum. For the first time, the outer fuel cycle is designed to allow for tritium self-sufficiency of the DEMO plant.

In the first two inner loops, new technologies had to be included, which cannot benefit from the ITER experience. First proof-of-principle confirmation was achieved that these technologies do potentially work. However, a strong maturation programme will have to be executed in the next programme period to bring these technologies to a higher readiness level. The outer loop of the inner fuel cycle and the whole outer fuel cycle involves known technology but clearly at an unprecedented scale.

Quantitatively, the fuel cycle design depends significantly on the machine gas throughput which itself is determined by the plasma scenario that is still evolving. To deliver a fuel cycle concept design by the middle of this decade, a strong programme has been started that combines dedicated lab-scale experiments and larger scale key technology facilities with the development of system block level models that feed into a holistic fuel cycle simulator tool. This approach allows for a high fidelity description of the operational space of the fuel cycle and will lead to a well justified concept design of the DEMO fuel cycle (Fig. 9).

CRediT authorship contribution statement

Chr. Day: Conceptualization, Methodology. **K. Battes:** Investigation. **B. Butler:** Conceptualization, Methodology. **S. Davies:** Investigation. **L. Farina:** Investigation. **A. Frattolillo:** Investigation. **R. George:** Investigation. **T. Giegerich:** Conceptualization, Methodology. **S. Hanke:** Investigation. **T. Härtl:** Investigation. **Y. Igitkhanov:** Investigation. **T. Jackson:** Investigation. **N. Jayasekera:** Investigation. **Y. Kathage:** Investigation. **P.T. Lang:** Investigation. **R. Lawless:** Investigation. **X. Luo:** Investigation. **C. Neugebauer:** Investigation. **B. Ploeckl:** Conceptualization, Methodology. **A. Santucci:** Investigation. **J. Schwenzer:** Investigation. **T. Teichmann:** Investigation. **T. Tijssen:** Investigation. **S. Tosti:** Investigation. **S. Varoutis:** Investigation. **A. Vazquez Cortes:** Investigation.

Declaration of Competing Interest

The authors declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper.

Acknowledgment

This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014-2018 and 2019-2020 under grant agreement No 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission. The main author on behalf of the whole TFV team is very grateful to the valuable recommendations of the TFV Design Review Panel who followed the project since 2018 (chair: M. Glugla, members: L. Baylor, I. Hagenlocher, P. Ladd).

References

[1] T. Donne, et al., European Research Roadmap to the Realisation of Fusion Energy, EUROfusion, 2018.
[2] M. Kovari, et al., Tritium resources available for fusion reactors, Nucl. Fusion 58 (2018), 026010.
[3] B. Nie, et al., Anthropogenic tritium: Inventory, discharge, environmental behavior and health effects, Renew. Energies 135 (2021), 110188.
[4] B. Bornschein, et al., Tritium management and safety issues in ITER and DEMO breeding blankets, Fus. Eng. Des. 88 (2013) 466–471.
[5] L.V. Boccaccini, Chr. Day, Chapter "Magnetic confinement fusion – Reactor blanket technologies. Enyclopedia of Nuclear Energy, Elsevier, 2021.
[6] L.V. Boccaccini, et al., Status of Maturation of Critical Technologies and Systems Design: Breeding Blanket, submitted to, Fus. Eng. Des. (2022).
[7] G. Caruso, et al., DEMO – DEMO – The main achievements of the Pre – Concept phase of the safety and environmental work package and the development of the GSSR, Fus. Eng. Des. 176 (2022), 113025.
[8] L. Barucca, et al., Maturation of Critical Technologies for the DEMO Balance of Plant Systems, submitted to, Fus. Eng. Des. (2022).
[9] G. Federici, et al., An overview of the EU breeding blanket design strategy as an integral part of the DEMO design effort, Fus. Eng. Des. 141 (2019) 30–42.
[10] M. Siccinio, et al., DEMO physics challenges beyond ITER, Fus. Eng. Des. 156 (2020), 111603.
[11] Chr. Day, et al., Consequences of the technology survey and gap analysis on the EUDEMO R&D programme in tritium, matter injection and vacuum, Fus. Eng. Des. 109-111 (2016) 299–308.
[12] Chr. Day, et al., A smart three-loop fuel cycle architecture for DEMO, Fus. Eng. Des. 146 (2019) 2462–2468.
[13] S. Varoutis, Y. Igitkhanov, Chr. Day, Effect of neutral screening on pumping efficiency in the DEMO divertor, Fus. Eng. Des. 146 (2019) 1741–1746.
[14] S. Varoutis, et al., Simulation of neutral gas flow in the JET sub-divertor, Fus. Eng. Des. 121 (2017) 13–21.
[15] T. Haertl, et al., Design and feasibility of a pumping concept based on tritium direct recycling, Fus. Eng. Des. 174 (2022), 112969.
[16] U. Fischer, et al., TBR Required, achievable and target TBR for the European DEMO, Fus. Eng. Des. 155 (2020), 111553.
[17] F. Cismondi, et al., Progress of the conceptual design of the European DEMO breeding blanket, tritium extraction and coolant purification systems, Fus. Eng. Des. 157 (2020), 111640.
[18] G.A. Spagnuolo, et al., Integrated design of breeding blanket and ancillary systems related to the use of helium or water as a coolant and impact on the overall plant design, Fus. Eng. Des. 173 (2021), 112933.
[19] K. Battes, C. Day, V. Rohde, Basic considerations on the pump-down time in the dwell phase of a pulsed fusion DEMO, Fus. Eng. Des. 100 (2015) 431–435.
[20] D. Ricci, et al., Discharge recovery by means of EC assisted start-up, EPS, Prague, Czech Republic, July 2018.
[21] R. Arredondo, et al., Preliminary estimates of tritium permeation and retention in the first wall of DEMO due to ion bombardment, Nucl. Mat. Energy 28 (2021), 101039.
[22] Chr. Day, T. Giegerich, The Direct Internal Recycling concept to simplify the fuel cycle of a fusion power plant, Fus. Eng. Des. 88 (2013) 616–620.

[23] T. Giegerich, C. Day, The KALPUREX-process – a new vacuum pumping process for exhaust gases in fusion power plants, Fus. Eng. Des. 89 (2014) 1476–1481.
[24] M. Abdou, et al., Physics and technology considerations for the deuterium–tritium fuel cycle and conditions for tritium fuel self sufficiency, Nucl. Fusion 61 (2021), 013001.
[25] R.J. Walker, M.R. Gilbert, Neutron activation of impurity seeding gases within a DEMO environment, Fus. Eng. Des. 124 (2017) 892–895.
[26] VDI Guideline 2225-3, Technisch-wirtschaftliche Bewertung, German Engineering Society (VDI), 1998.
[27] R. Lawless, et al., Tritium Plant Technology Development for a DEMO Power Plant, Fus. Sci. Technol. 71 (2017) 679–686.
[28] A.I. Livshits, et al., Physico-chemical origin of superpermeability – large-scale effects of surface chemistry on "hot" hydrogen permeation and absorption in metals, J. Nucl. Mat. 170 (1990) 79–94.
[29] A.I. Livshits, et al., Plasma driven superpermeation of hydrogen through group Va metals, J. Appl. Phys. 84 (1998) 2558–2563.
[30] Y. Hatano, et al., Effects of bulk impurity concentration on the reactivity of metal surface: Sticking of hydrogen molecules and atoms to polycrystalline Nb containing oxygen, J. Chem. Phys. 127 (2007), 204707.
[31] A.A. Skovoroda, et al., Plasma-driven superpermeation of hydrogen through Nb membranes: bulk effects, J. Nucl. Mat. 306 (2002) 232–240.
[32] M. Bacal, et al., Plasma driven superpermeation and its applications for particle control, in: 30th EPS Conference on Contr. Fusion and Plasma Physics 27A, 2003, p. 1182.
[33] A. Mack, D. Perinic, Conceptual study of the cryocascade for pumping, separation and recycling of ITER torus exhaust, Fus. Eng. Des. 28 (1995) 319–323.
[34] C. Day, and M. Scannapiego, A multi-stage cryopump for fuel separation in the Direct Internal Recycling loop of DEMO, to be submitted.
[35] B.J. Peters, Chr. Day, Analysis of low pressure hydrogen separation from fusion exhaust gases by the means of superpermeability, Fus. Eng. Des. 124 (2017) 696–699.
[36] S. Hanke, et al., Progress of the R&D programme to develop a metal foil pump for DEMO, Fus. Eng. Des. 161 (2020), 111890.
[37] R.K. Musyaev, et al., Tritium superpermeability: experimental investigation and simulation of tritium recirculation in Prometheus setup, Fus. Sci. Technol. 48 (2005) 35–38.
[38] Y. Kathage, et al., Experimental and simulation progress of a metal foil pump for fuel separation in the Direct Internal Recycling loop of DEMO", submitted to, Fus. Eng. Des. (2022).
[39] C. Day, T. Giegerich, Development of Advanced Exhaust Pumping Technology for a DT Fusion Power Plant, Trans. Plasma Sci. 42 (2014) 1058–1071.
[40] F. Siviero, et al., Robustness of ZAO based NEG pump solutions for fusion applications, Fus. Eng. Des. 166 (2021), 112306.
[41] T. Giegerich, et al., Development of a viable route for lithium-6 supply of DEMO and future fusion power plants, Fus. Eng. Des. 149 (2019), 111339.
[42] T. Giegerich, C. Day, Conceptuation of a continuously working vacuum pump train for fusion power plants, Fus. Eng. Des 88 (9–10) (2013) 2206–2209.
[43] T. Giegerich, et al., Preliminary configuration of the torus vacuum pumping system installed in the DEMO lower port, Fus. Eng. Des 146 (B) (2019) 2180–2183.
[44] R. Lässer, D.K. Murdoch, R.-D. Penzhorn, Use of Ferrofluidic Seals in the Design of Tritium Compatible Pumps, Fus. Sci. Technol. 41 (2002) 621–625.
[45] A. Antipenkov, et al., Tritium Pumps for ITER Roughing System, Fus. Sci. Technol. 48 (2005) 47–50.
[46] T. Giegerich, et al., Conceptual Design of the Mechanical Tritium Pumping System for JET DTE2, Fus. Sci. Technol. 68 (2015) 631–634.
[47] T. Giegerich, et al., Advanced design of the Mechanical Tritium Pumping System for JET DTE2, Fus. Eng. Des. 109-111 (2016) 359–364.
[48] T. Teichmann, C. Day, Particle Simulation of Linear Diffusion Pumps for DEMO Torus Exhaust Pumping, Fus. Eng. Des. 169 (2021), 112694.
[49] The PROMETEUS Project ("PROcess of radioactive MErcury Treatment under EU Safety-Standards"), German Ministry of Education and Research, 2018–2020. Research Project 15S9266A-B.
[50] P.T. Lang, et al., Considerations on the DEMO pellet fuelling system, Fus. Eng. Des. 96 (2015) 123–128.
[51] B. Ploeckl, et al., Matter Injection in EU-DEMO: The Preconceptual Design, Fus. Sci. Technol. 77 (2021) 266–278.
[52] P.T. Lang, et al., Optimizing the EU-DEMO pellet fuelling scheme, Fus. Eng. Des. 111 (2020), 111591.
[53] A. Frattolillo, et al., Addressing the feasibility of inboard direct-line injection of high-speed pellets for core fueling of DEMO, Fus. Eng. Des. 146 (2019) 2426–2429.
[54] B. Ploeckl, et al., Comparison of different pellet injection systems for ELM pacing, Fus. Eng. Des. 86 (2011) 1022–1025.
[55] A. Frattolillo, et al., Injection of high-speed solid D2 pellets using a "Direct-Line-of-Sight" (DLS) guide tube, Fus. Eng. Des. 162 (2021), 112138.
[56] P.T. Lang, et al., Targeting a Versatile Actuator for EU-DEMO: Xenon Doping of Fueling Pellets, Fus. Sci. Technol. 77 (2021) 42–50.
[57] B. Ploeckl, et al., Targeting a Versatile Actuator for EU-DEMO: novel Control Scheme for Multisource Pellet Injector, Fus. Sci. Technol. 77 (2021) 199–205.
[58] F. Janky, et al., Demo control challenges, in: virtual SOFT, 2020.
[59] T. Jiang, et al., Manifold Concept design for ITER Gas Injection System, IEEE Trans. Plasma Sci. 40 (2012) 788–792.
[60] S. Garcia, D. Chatain, J.P. Perin, Continuous production of a thin ribbon of solid hydrogen, Laser Part. Beams 32 (2014) 569–575.
[61] Y.N. Hörstensmeyer, et al., Permeator Simulations for the EU-DEMO Fuel Cycle, Fus. Sci. Technol. 76 (2020) 232–237.
[62] D. De Meis, et al., Low pressure fusion exhaust gases separation, Fus. Eng. Des. 146 (2019) 1665–1669.
[63] L. Farina, A. Santucci, S. Tosti, Plasma Enhancement Gases separation via ceramic porous membranes for plasma exhaust processing system of DEMO, Fus. Eng. Des. 169 (2021), 112484.
[64] R.C.R. Shaw, B. Butler, Applicability of a cryogenic distillation system for D-T isotope rebalancing and protium removal in a DEMO power plant, Fus. Eng. Des. 141 (2019) 59–67.
[65] M.W. Lee, TCAP for Separation of Hydrogen Isotopes, USDOE Report DP -1660, E.I. Du Pont de Nemours & Co., Savannah River Laboratory, Aiken, 1983, p. SC29808.
[66] C. Neugebauer, Y. Hörstensmeyer, Chr. Day, Technology development for isotope rebalancing and protium removal in the EU DEMO fuel cycle, Fus. Sci. Technol. 76 (2020) 215–220.
[67] A. Busigin, S.K. Flood, ITER hydrogen isotope separation system conceptual design description, Fus. Eng. Des. 13 (1990) 77–89.
[68] N. Bainbridge, et al., Operational experience with the JET AGHS cryodistillation system during and after DTE1, Fus. Eng. Des. 47 (1999) 321–332.
[69] J.R. Bartlit, et al., Hydrogen isotope distillation for the tritium systems test assembly, in: Third Topical Meeting on Technology of Controlled Nuclear Fusion, Los Alamos Scientific Lab, NMUSA, 1978. Report Number LA-UR-78-1325.
[70] F. Sabathier, et al., Assessment of the performance of the JET exhaust detritiation system, Fus. Eng. Des. 54 (2001) 547–553.
[71] M. Rozenkevich, et al., Main Features of the Technology for Air Detritiation in Scrubber Column, Fus. Sci. Technol. 70 (2016) 435–447.
[72] J. Johnston, DEMO Plant Safety Requirements Document, (PSRD), EUROfusion technical report (2019).
[73] I. Cristescu, Enhanced configuration of a water detritiation system; impact on ITER Isotope Separation System based cryogenic distillation, Fus. Eng. Des. 109–111 (2016) 1404–1407.
[74] M. Glugla, Detritiation systems for ITER, Rayonnements Ionis. Tech. Mes. Prot. 4 (2014) 6–15.
[75] A. Santucci, et al., The issue of Tritium in DEMO coolant and mitigation strategies, Fus. Eng. Des. 158 (2020), 111759.
[76] F.A. Hernandez, et al., Consolidated design of the HCPB Breeding Blanket for the pre-Conceptual Design Phase of the EU DEMO and harmonization with the ITER HCPB TBM program, Fus. Eng. Des. 157 (2020), 111614.
[77] K.M. Song, et al., Introduction to Wolsong Tritium Removal Facility (WTRF), in: Transactions of the Korean Nuclear Society Autumn Meeting Busan, Korea, October 27-28, 2005.
[78] G.A. Spagnuolo, et al., Integration issues on tritium management of the European DEMO Breeding Blanket and ancillary systems, Fus. Eng. Des. 171 (2021), 112573.
[79] A. Santucci, et al., Novel Non-Evaporable Getter Materials and Their Possible Use in Fusion Application for Tritium Recovery, Molecules 25 (2020) 5675.
[80] M. Utili, et al., Development of anti-permeation and corrosion barrier coatings for the WCLL breeding blanket of the European DEMO, Fus. Eng. Des. 170 (2021), 112453.
[81] C.R. Jones, S. Yardley, S. Medley, The social acceptance of fusion: Critically examining public perceptions of uranium-based fuel storage for nuclear fusion in Europe, Energy Res. Soc. Sci. 52 (2019) 192–203.
[82] M. Utili, et al., Tritium Extraction From HCLL/WCLL/DCLL PbLi BBs of DEMO and HCLL TBS of ITER, IEEE Trans. Plasma Sci. 47 (2019), 2886409.
[83] J.C. Schwenzer, et al., Operational tritium inventories in the EU-DEMO fuel cycle, submitted to, Fus. Sci. Technol. (2022).