Integration issues on tritium management of the European DEMO Breeding Blanket and ancillary systems

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Within the EUROfusion framework, two reactor architectures are being developed as possible candidates for the EU DEMO. One is based on the Water Cooled Lithium Lead (WCLL) Breeding Blanket (BB) concept and the other on the Helium Cooled Pebble Bed (HCPB) BB concept. The two DEMO variants result in completely different reactor and ancillary system designs due to the use of different coolants (i.e water or helium) as well as different neutron multipliers, tritium breeders, and carriers (i.e. PbLi for the WCLL and solid ceramic and beryllide with helium as a purge gas for the HCPB). As a consequence, the definition of the tritium management systems for the extraction and the control of inventories and environmental releases also needs to be tuned for each variant. Furthermore, dedicated technologies for tritium extraction and removal must be developed and specific working points for the two architectures must be defined. It is, therefore, necessary to rely on holistic approaches for the analysis of tritium management issues of the BB, its ancillary systems and fueling systems. In this work, an overview of the tritium permeation issues and the applied mitigation measures (e.g. permeation barriers) are described. Furthermore, design limits given by tritium extraction technologies as well as experimental campaigns (e.g. determination of the Sieverts' constant of tritium in PbLi at DEMO relevant temperatures) to substantiate the modelling approach are also reported. Finally, the impact on the fueling system due to the tritium inventories sequestered in the BB and other systems is described, providing information on the doubling time and initial tritium inventories for a given tritium breeding ratio.

Keywords: Breeding Blanket; tritium permeation; Tritium Extraction and Removal; tritium fueling.

1. Introduction

In the EU DEMOstration Fusion Reactor DEMO, one of the key components is the Breeding Blanket (BB). It will accomplish three main functions such as (i) the removal and transfer of heat, (ii) the shielding of Vacuum Vessel (VV) and Magnet System and (iii) the breeding of tritium to guarantee the self-sufficiency [1].

Currently, two main BB concepts are studied within the Work Package Breeding Blanket (WPBB), namely the Water Cooled Lithium Lead (WCLL) and the Helium Cooled Pebble Bed (HCPB) [2, 3]. They differ, respectively, for (i) the type of coolant (water or He), (ii) type of breeder, neutron multiplier and tritium carriers (i.e. PbLi for the WCLL and solid ceramic and beryllide with helium as a purge gas for the HCPB), and technologies used for the extraction of tritium and purification of coolant as well as tritium control and interfaces to fueling systems [4, 5]. Therefore, the design of the BB affects also the design of the ancillary systems as well as the R&D program implemented in EUROfusion. Among the BB interfacing systems, the most notables in the management of tritium are: the Primary Heat Transfer System (PHTS) with the Coolant Purification Systems (CPS), the Tritium Extraction and Removal Systems (TER) and, to less extent, the Tokamak Fueling System that provides the VV with the tritium produced in BB and extracted in the TER. Both R&D and modelling activities are linked to the design of these systems that all together with the BB constitute the DEMO architecture. For this reason, it is worth to tackle at an early stage the coordination of different studies among the area involved in order to assure that the developed solutions are consistent and that the major issues are addressed. This coordination becomes vital in case systems are involved in the management of tritium. As a consequence, the definition of the tritium management systems for the extraction and the control of inventories and releases also needs to be tuned for each variant. Furthermore, dedicated technologies for tritium extraction and removal must be developed and specific working points for the two architectures must be defined.

As a result of the activities conducted in the Key Design Integration Issue 2 (KDII2) [1, 6], in this work, the integration aspects related to the choice of the BB concept are addressed with particular focus on the management of tritium in the DEMO Plant. However, the presented topics are not meant to be exhaustive on R&D and modelling activities or of the integration efforts but rather summarize the most recent and relevant achievements.

2. R&D in support of tritium modelling

2.1 Permeation barriers

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The release of tritium into the BB coolant and in the environment through the pipes/components walls has to be considered since the phenomenon affects the economy of the fuel management and raises issues related to the radio-toxicity. As a consequence, the application of a protective coating on the blanket walls is mandatory to minimize in particular the permeation effect from the breeder to the coolant and, in case of PbLi, also the corrosive effect. In this context, Al-based coatings are considered as the reference for barriers thanks to their good chemical compatibility with the PbLi alloy and their capability to reduce permeation. Moreover, Al_2O_3 coating can work also as an electrical insulator and hence limiting Magneto Hydrodynamic effects in the PbLi flow. Three coating technologies were selected:

- Electrochemical ECX process [7] developed by KIT;
- Atomic Layer Deposition (ALD) coating developed by Italian Institute of Technology (IIT) [8];
- Pulsed Laser Deposition (PLD) coating developed by IIT [8].

These permeation barriers have been developed at laboratory scale and a preliminary scale-up of the technologies has been performed.

ECX coating has shown a Permeation Reduction Factor (PRF) (i.e. the ratio between the permeation flux with and without the permeation barrier) of up to 100-200, good compatibility with PbLi (corrosion test carried out in flowing PbLi up to 10.000h), no modification of the coating after 2000 thermal cycles, and good insulation properties have been verified after the tests. The coating can be manufactured on all surfaces (inside and outside BB or pipes), but the process requires high temperatures and a heat treatment of the BB structural material EUROFER97.

The main advantage of ALD coating is the possibility to obtain continuous and pinhole-free films on substrates with complex geometries (differently from PLD, which is a line of sight technique). The experiments have shown PRFs of up to 900-1000, good compatibility with stagnant PbLi and good insulation properties also after 500 thermal cycles. The coating can be manufactured at a temperature below 500°C on all surfaces.

For the PLD process, the characterization has shown a PRF up to 1000-10000, good compatibility with PbLi (corrosion test carried out in stagnant PbLi) and no modification of the coating after thermal cycling (1000). Furthermore, good insulation properties have been also verified after the previously mentioned tests. The coating can be manufactured at low temperature only on the external surface of the components, therefore it is possible to coat WCLL water tubes and the First Wall (FW).

A preliminary characterization under neutron irradiation in presence of PbLi of PLD and ECX has been carried out in the LVR-15 reactor in CVREZ in 2020. A test section with 4 specimens (2 PLD and 1 ECX coated EUROFER samples and 1 capsule without coating) has been exposed to flowing PbLi. The temperature gradient generated inside the capsule containing the samples allowed a PbLi natural circulation with a velocity profile in the range between 1-2.7 mm/s, i.e. relevant for WCLL conditions. The preliminary results have shown a PRF for PLD techniques higher than 250 and no corrosion attack of PbLi. The experiments for ECX are still in progress.

2.2 Tritium permeation experiments

2.2.1 Tritium permeation experimental set-ups

The quantification of the tritium permeation in the BB systems is of high importance from the safety point of view and in the definition of the interfaces between various subsystems. Besides the development of tritium transport-modelling tool, the experimental validation must be considered during the development and the design of the BB subsystems. For this purpose, two experimental set-ups have been constructed to characterize the permeation parameters for both WCLL and HCPB BB concepts. These two experimental rigs are under construction and commissioning at KIT in view of validation of the modelling tool.

As far as benchmarking of tritium permeation rates for WCLL is concerned, two experiments have been set-up.

For the first test, it has been decided to not use the PbLi because the same behaviour can be easily reproduced using the same tritium partial pressure as expected in the liquid metal. Indeed, the tritium permeates in atomic form through the metal due to the fact that at the interface gas-metal molecular tritium dissociate in atomic form. The dissociation coefficient is well known and the process of dissociation can be easily quantified. The main purpose of the experiment is to investigate the dynamic of the isotopic exchange at the opposite interface between the atomic tritium and water. Therefore, He containing tritium is used on one side of the rig. On the other side, instead, water is used to reproduce WCLL relevant conditions.

For the second experiment at WCLL relevant conditions, measurements having tritiated water on one side of the set-up and non-tritiated water on the other side have been foreseen to reproduce the permeation issues in the primary Heat eXchangers (HXs) or the Steam Generator (SG), for instance.

This experiment will also provide information on possible tritiated water dissociation on one side, followed by tritium atomic permeation through the separation plate and the isotopic exchange on the opposite face of the plate with water. The water is sampled at certain time intervals to provide information on the dynamic of the process.

Concerning the first test campaign, the experimental set-up is shown in Figure 1. In the inner vessel, marked in green, tritiated gas has been fed and the amount of tritium after 10 days has been measured in the inner vessel and also in the outer vessel where water is present (blue region). The operation temperature was typically 350ºC.

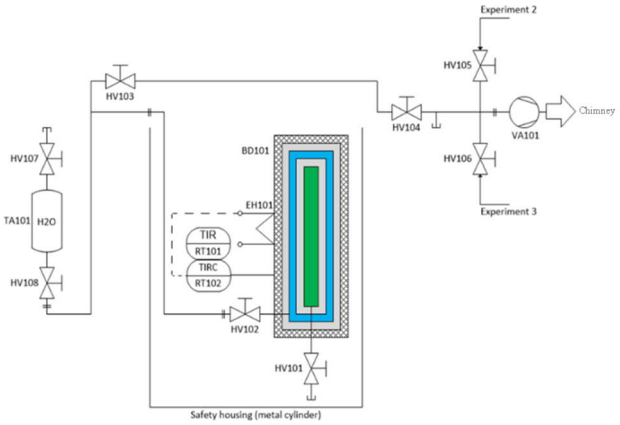

Figure 1. The experimental set-up for tritium permeation from gas to water.

The preliminary experimental results obtained are reported in Table 1. The second experiment to reproduce the permeation through HX is planned for the end of 2020.

The first measurements show the tritium content in the inner vessel and outer vessel after 10 days. It can be concluded that even after 10 days the equilibrium as far as the isotopic exchange between tritium gas and water was not reached. The experimental results will be used for the calibration of the modelling tool.

Table 1. The amount of tritium in various locations of the experimental set-ups.

	Inner vessel (start)	**Inner vessel (after 10 days)**	**Outer vessel (water)**
Amount of tritium [Bq]	1,20E+10	1,54E+09	3,40E+09

For the HCPB case, tritium will permeate from purge gas to He coolant, and from He coolant to water. The experimental set-up is operated with He and tritium on one side and He with hydrogen/deuterium at various partial pressure on the other side. This will allow simulating the tritium permeation from purge gas to cooling gas. The experimental set-up has incorporated enough sampling points and the required instrumentation in order to quantify the dynamic of tritium permeation from one side to another. The measuring cell is designed to allow the characterization of the permeation surface that may influence the dissociation and recombination coefficients. The experiments will be continued for long term operation with hydrogen and deuterium in order to understand the dynamic of the surface and to give references concerning the possibilities for chemical treatment of the surface. The experiments will be carried out at temperatures and tritium partial pressures that are relevant for DEMO applications. In Figure 2, the configuration of the permeation cell is shown. The high-temperature part of the cell is manufactured by the glass in order to mitigate tritium permeation outside of the experiment and providing the permeation only through the stainless steel tube located in the inner part. Laser Raman measurements are implemented in order to provide data related to the dynamic of the process. The experiments are planned in 2021.

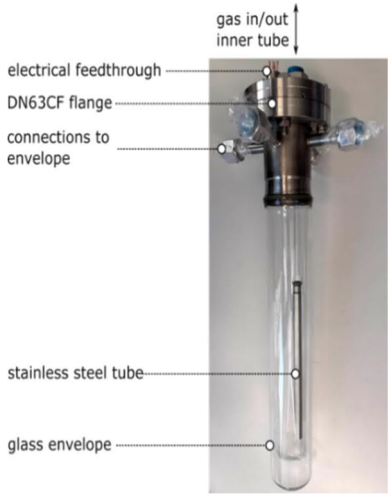

Figure 2. Mockup of the permeation cell.

2.2.2 PbLi tritium extraction technologies and experiments

The recovery of bred tritium is a crucial step in the path to achieve the self-sufficiency of the reactor and, therefore, to reach the production of energy by fusion power plants. Within the WCLL, the system in charge of tritium removal from the BB is the TER System. The aim of this system is to extract tritium at the highest rate to minimize the inventory in the liquid metal and, thus, to reduce possible leakages to other systems. Tritium can be removed from the PbLi alloy employing different technologies which are still under development [9]. During the last years, the program has been focused on the Permeation Against Vacuum (PAV) in which a permeable membrane (V, Nb) contains the liquid metal and is subjected to vacuum on the outer side, thus promoting tritium extraction due to the existing pressure gradient. However, PAV designs remain in a conceptual stage and no experimental validation has been performed yet. New PAV concepts are being developed within the EUROfusion program to cover the specifications of the liquid-based breeding blankets [10, 11]. The design and fabrication for the experimental validation of the concept based on a V membrane have been performed at CIEMAT [10, 12]. Complementary, ENEA has been working on a PAV design based on a Nb membrane [11]. Two installations have been constructed/upgraded, CLIPPER in CIEMAT [13] and TRIEX in ENEA [14], where the designed permeators have been or will be tested.

Concerning the experimental activity performed at CIEMAT, dedicated tritium transport models for the V-membrane PAV have been used to set the dimensions of the components for an 80% target removal efficiency . Thus, the selected configurations consist of a squared multi-channel [10] (Figure 3) component. The membrane must have a high permeability to tritium, must withstand high temperatures and be compatible with PbLi. Pumping speed requirements for the vacuum system have also been evaluated, finding that values of up to 370 m^3/s could be required [15].

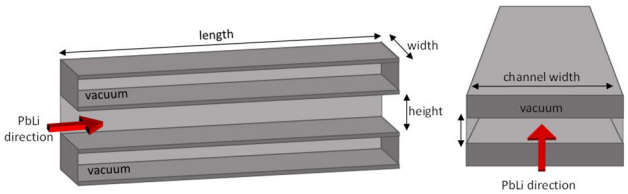

Figure 3. Squared-PAV design principle.

Commercial linear diffusion pumps could be employed up to a limit of 20 m^3/s of pumping speed.

Diffusion pumps present some advantages such as the absence of moving parts and good pumping speed for light molecules, however the use of mercury as working fluid gives rise to incompatibilities with tritium [16]. Nevertheless, currently, there is a strong R&D line for the development of this kind of pumps to update its compatibility with tritium and be applicable to the fusion related necessities [17].

A review of the literature revealed that there exists a large scattering on permeability values for the selected membranes [18]. Permeability of V was measured in two different laboratories (CIEMAT [18] and ASIPP [19]) looking for cross-check validation and obtaining a good agreement, but with a deviation of 2 orders of magnitude with respect to literature. The same behaviour was obtained for Nb [18] corroborating the importance of performing this analysis. At this moment prototypes are under construction, their evaluation will be performed during 2021. CLIPPER (Figure 4) covers a wide operational range, allowing the characterization of the PAV behaviour at different blanket conditions of temperature and velocity of the liquid metal, and deuterium/hydrogen concentrations (see Table 2).

Table 2. CLIPPER operational parameters.

Parameter	Value
PbLi velocity in PAV	up to 1 m/s
H concentration in PbLi	1E-4 – 5E-2 mol/m^3
PbLi mass flow rate	2 – 39 kg/s
PbLi temperature	300 – 500 °C

The test section includes the PAV prototype and a novel gas injection system to introduce the required H/D concentration in the liquid metal to reproduce breeding blanket conditions [20]. Besides, the PbLi is heated and melted in a dedicated tank outside the loop. A glove box installed at the top of the melting tank, and operated under Argon atmosphere, allows access to the liquid metal to remove impurities during the melting process and other operations (e.g. quality verification). Once the PbLi is melted, it is routed to the storage/filling tank of the loop, and from there to the circuit. Upon the finalization of the new PAV prototype and the gas phase testing, the validation under PbLi flow will be performed. The experimental campaign is firstly aimed to demonstrate the PAV technique. Then, if the results are conclusive, experiments will be oriented to the operation at relevant BB conditions in a quite ambitious test matrix. In parallel, the results obtained will be used for the validation of tritium transport codes.

Figure 4. CLIPPER loop.

Concerning the PAV based on Nb membranes developed by ENEA in collaboration with the NEMO group of Politecnico di Torino, the design of the experimental set-up has been started. It is constituted by permeable niobium tubes inside a vacuum vessel. Each vessel is a cylinder assumed to be installed vertically in the tokamak building. The operative conditions of PAV TER system for WCLL BB are reported in Table 3.

Table 3. Operative conditions adopted in PAV – WCLL BB.

Design Parameter	WCLL BB
PbLi mass flow rate [kg/s]	~956
N. Pb-16Li Loops (IB+OB)	6
Pb-16Li mass flow rate IB/LLE loop [kg/s]	40-80
Pb-16Li mass flow rate OB/LLE loop [kg/s]	264
Tritium partial pressure in LLE [Pa]	0.19-210
Surface-to-volume ratio [1/m]	125-750
LLE pressure [MPa]	0.2-1.0
Vacuum required in the shell [Pa]	10-1 -10-2
TES operative temperature	330-500
TES efficiency min [%]	80
TES efficiency nominal [%]	90

Each vessel contains the channels for the PbLi flow: the wall of the channels is the tritium permeation membrane. The parallel channels are “U-shaped”, as reported in [21] and shown in Figure 5.

Tritium transport through permeation membrane can be determined by tritium diffusion in the material (diffusion-limited regime) or by surface limited regime where the surface phenomena are slower than the diffusion in the metal bulk and consequently they dominate the mass transport so that the concentration gradient due to the diffusion in the membrane bulk can be neglected. The new measurements on permeation property set for the Nb membrane recently performed by Malo et al. [18] showed a permeability that is orders of magnitude lower than the theoretical that determine a surface limited regime. The design has shown the possibility to design the permeator with 6,0 m diameter satisfying all the constraints on size and pressure drops derived by the integration of the system in PbLi loops and tokamak building [4]. A mock-up of PAV technology manufactured with Nb tube will be characterized in TRIEX-II facility at ENEA. The PAV is composed of a cylindrical vessel with 16 niobium U-pipes inside [22].

The vacuum is made in the vessel while the PbLi flows in the Nb pipes. The PbLi is distributed into the niobium pipes by a collector, which constitutes the lower part of the PAV. The experimental campaign will start in 2021.

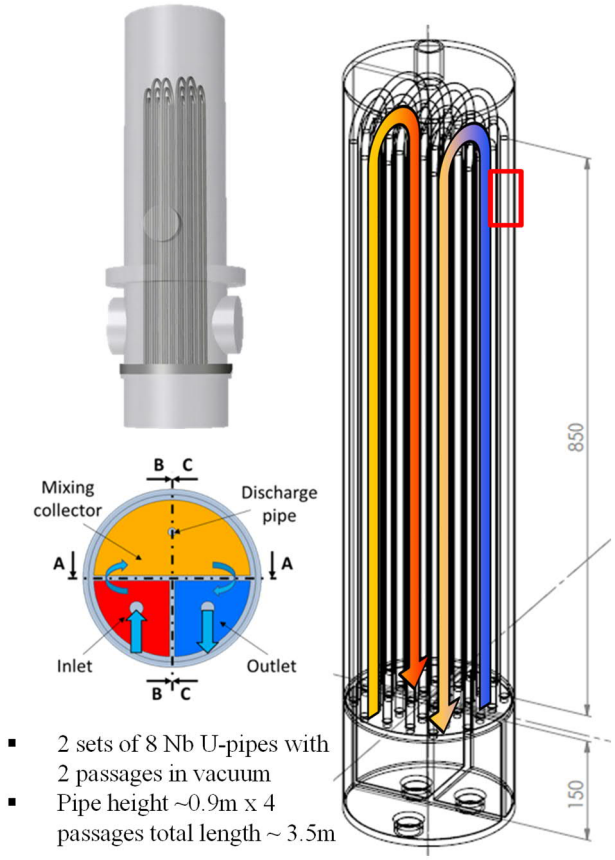

Figure 5. PAV mock-up drawing manufactured with niobium tubes and installed in TRIEX-II facility.

3. Tritium permeation and retention in the BB First Wall

The tritium permeation rates from the plasma into the coolant and the tritium amount retained in the FW have been estimated as a function of time for the DEMO BB concepts using TMAP7 [23] and TESSIM-X [24] codes. In these estimates, the WCLL [25] and HCPB [26] temperature profiles and geometries were investigated under simultaneous gas- and ion-driven tritium loading for a range of impinging tritium particle fluxes (10^{18} to 10^{20} T/m^{2}s), impinging particle energy of 500 eV and a tritium partial pressure of 1 Pa, based on current SOLPS DEMO plasma models.

The current design of the DEMO First Wall consists of a 0.8 mm W layer, a series of W-EUROFER Functionally Graded Material (FGM) interlayers with a total thickness of 1.2 mm, and a 2-3 mm EUROFER substrate. Modelling of complex layer systems such as FGMs is presently infeasible due to the lack of experimental data regarding the material properties and interface properties between layers. Therefore, three scenarios were investigated: 2 mm W on 2 – 3 mm EUROFER (FGM behaving like W), 0.8 mm W on 3.2 – 4.2 mm EUROFER (FGM behaving like EUROFER), and pure EUROFER (W layer eroded or otherwise removed).

To model the W/EUROFER interface, equilibrium in the chemical potential was assumed, and Sieverts' law was applied at the vacuum and coolant sides, fixing the surface concentration of tritium on each side. The ion driven loading was modelled as a bulk source of implanted tritium with ion range parameters computed by TRIM [27]. As ion loading leads to tritium concentrations in excess of those given by Sieverts' law, T diffuses from the source inside the bulk towards both the vacuum and coolant sides. Material properties for tritium in W and EUROFER were based on data with deuterium from [28] and [29], respectively.

Material trap concentrations were varied as a function of neutron-induced damage, temperature, and hydrogen isotope content. The data for W was based on [30], with defect concentrations ranging from 10^{-4} (no neutron damage, of the order of bulk trap concentrations in unirradiated W [31] and EUROFER [29]) to 0.4% (high neutron damage and simultaneous hydrogen isotope loading), and de-trapping energies ranging from 1.35 eV to 2.1 eV. Trap concentrations and energies for EUROFER are based on [32-34], with similar concentrations to W (10^{-4} to 0.2%), but significantly lower de-trapping energies (0.65 to 1.35 eV). Figure 6 shows the evolution of the retained tritium in a FW with a 0.8 mm W lining for the WCLL and HCPB scenarios under various flux assumptions, with high trap concentrations.

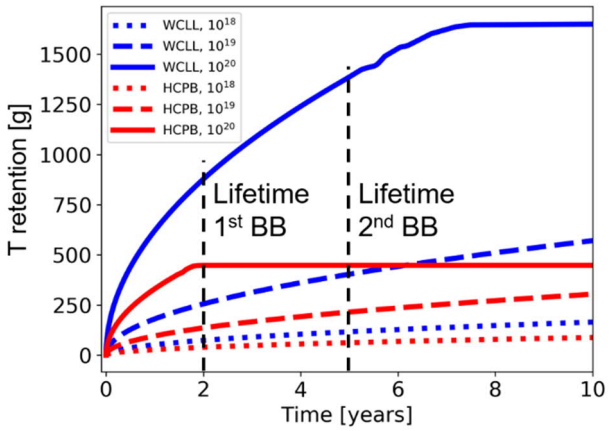

Figure 6. Tritium retention in the DEMO FW as a function of time (in full-power years) for the WCLL and HCPB designs with a 0.8 mm W cladding for varying flux scenarios, under the assumption of high trap concentrations from n-damaging and simultaneous hydrogen isotope loading.

The expected total tritium permeation rates and retained tritium amounts at the end of life of the 2nd BB (5 fpy) are reported in Table 4 for the 2 mm W, 0.8 mm W and pure EUROFER scenarios for the WCLL and HCPB concepts, assuming an impinging ion flux of 10^{20} T/m^{2}s and neutron-induced traps.

Table 4. T permeation to the BB coolant and retention in the FW at the end of life of the 2nd BB (5 fpy).

Scenario	Wall concept	Permeation [mg T day $^{-1}$]	Retention [g T]
2 mm W (no crack considered)	WCLL	0	800 – 1350
	HCPB	0 – 120	700
0.8 mm W	WCLL	0 – 300	800 – 1350

	HCPB	300	450
EUROFER only	WCLL	3250	60 - 185
	HCPB	13180	10 - 20

The ranges shown in the table correspond to results simulating trap concentrations with and without the simultaneous effect of hydrogen isotope loading, which affect the onset of permeation and tritium retention. In the cases where tritium permeation is listed as zero, the onset of permeation is delayed beyond the lifetime of the 2nd BB. Radioactive decay of tritium was not taken into account as it has a negligible impact relative to tritium loading. In the models shown here, retention is driven by trapping, so increased trap concentrations lead to increased retention. However, although increased trapping can significantly delay the onset of permeation (potentially beyond the lifetime of the 2nd BB), it does not directly affect the steady-state permeation flux, which is only a function of the impinging particle flux and energy, and the thickness of the W/EUROFER layers.

Within the lifetime of the 2nd BB, for the WCLL case, there is no difference in retention between a 0.8 and 2 mm W layer, because this time is not sufficient to fully saturate the W layer. For the HCPB case, where trap concentrations are lower due to the higher operating temperature, there is sufficient time to reach tritium saturation with a 0.8 mm W cladding, so there is a difference in retention between 0.8 mm and 2 mm W layer cases. When there is no W layer present (surface is EUROFER), the system behaves differently, displaying lower retention due to the absence of high-energy traps like in W, but permeation is increased, and the onset of permeation is much sooner because of lower trapping. Due to the difference in solubility between W and EUROFER, in the EUROFER-only case, the increase of permeation is driven by gas loading. Permeation with a EUROFER-only wall could be quite significant under these conditions, of the order of several grams per day..

Comparing T retention from this analysis with the estimates given in [35, 36], tritium losses due to trapping in the BB FW may pose a serious problem for tritium balance in the current DEMO design if a W layer is present. In particular, the trapping of tritium (i) affects the starting inventory (more tritium is required to saturate the BB structures at the start-up), (ii) poses serious concerns for the confinement of tritium during the RM operation and (iii) complicates the waste management of the plant.

However, three major assumptions are employed in these estimates:

- This work assumes there are no interface barriers between the W and EUROFER layers. If interface or surface barriers or traps are indeed present, the effective permeation rate would likely be lower than the one stated here. Retention on the other hand could be slightly higher. However, this effect could be negligible for the case of WCLL design, as the traps are not saturated by the BB end of life. For the HCPB, a slight increase in retention could be the result.
- The models assume the W layer is dense and presents a smooth surface. If the W layer presents open porosity, the tritium flux and retention could potentially be reduced, as it may increase the desorbing flux to the surface without impacting the impinging ion flux.
- In the event of a EUROFER-bare wall, it is assumed that the surface does not change over time. However, experiments have shown that, under bombardment with D ions, a spiked surface morphology may develop, e.g. [37-40]. In this event, tritium permeation could vary.

Overall, while the current modelling work relies on some assumptions that are as of yet unconfirmed, the results shown here are intended to provide a first estimate of tritium permeation and retention in the BB FW.

4. T inventories and flows

The design of the DEMO BB system and its ancillaries shall aim at minimizing the tritium inventories and the unavoidable parasitic losses due to permeation and process fluids' leakages.

The major release pathways and retention aspects of a DEMO plant shall be identified through devoted tritium permeation analyses, exploring possible design measures to reduce and mitigate parasitic losses and inventories. In this study, this exercise has been addressed with parametric analysis, following a one-at-a-time parameter variation of a set of selected operational figures of merit. The parameter variation denotes a different BB/ancillaries scenario and is performed starting from a baseline referred as to "Case 0", denoting the design features of BB, PHTS and TER of the WCLL and HCPB DEMO 2017 baselines [25, 26]. These are qualitatively and quantitatively elucidated in the next subsections, along with the outcome of this study. The main parameters used for the Case-0 of both WCLL and HCPB DEMO variants are reported in Table 5 and Table 6, respectively. Starting from these parameters, a sensitivity analysis has been performed taking into account a set of parameters both for the WCLL and HCPB DEMO variants, respectively.

Table 5. WCLL DEMO 2017 baseline.

Parameter (unit)	**WCLL Case-0**
CPS by pass flow rate [kg/h]	0
TER efficiency [%]	82
T perm. rate from plasma [mg/d]	0
PRF in BB [-]	100
H_2 concentration in water [ppm]	8
PRF in PbLi loop[-]	1
H_2O leak rate from HXs [kg/h]	0

Table 6. HCPB DEMO 2017 baseline.

Parameter (unit)	**HCPB Case-0**
CPS by pass flow rate [kg/s]	3
CPS efficiency [%]	90
H_2 part. Press. in He coolant [Pa]	0
He purge gas flow rate [kg/s]	0.497
TER efficiency [%]	80
T perm. rate from plasma [mg/d]	3.5
He leak rate [%inventory/yr]	4
PRF on IHX pipes	100
Steam conc. in purge gas [%]	0
He leak rate from HXs [kg/h]	0

4.1 Operational scenarios for BB and ancillary systems

Table 7 lists the parameters addressed within the parameter study for the WCLL blanket, along with their numerical ranges. Some of these are related to systems performances (e.g. CPS by-pass flow rate, TER system efficiency, and permeation reduction factor), whereas others denote operational figures (water leak rate on SG/HX pipes and doping hydrogen pressure in cooling water).

Table 7. Parameter scans for WCLL DEMO 2017 baseline.

Case ID	Parameter (unit)	Min	Max
1	CPS by pass flow rate [kg/h]	20	360
2	TER efficiency [%]	80	95
3	T perm. rate from plasma [mg/d]	0	20
4	PRF in BB [-]	1	1000
5	H_2 concentration in water [ppm]	8	100
6	PRF in PbLi loop[-]	100	1000
7	H_2O leak rate from HXs [kg/h] [41]	0.3	0.5

The enhancement of tritium removal capabilities in TER and CPS systems (given by processed flow rate and/or efficiency) directly impact the average tritium partial pressure in tritium carrier and water coolant's loops, hence representing a goal-oriented way to reduce the permeation rates through BB cooling structures and PHTS heat exchangers (HX and SG). In particular, 20 kg/h and 360 kg/h CPS by-pass flow rates denote the ITER and the Darlington nuclear power plant's water detritiation system (WDS) [42] capacity, respectively. The lower and upper bounds indicated for the TER systems efficiency denote the minimum efficiency required for a DEMO tritium extraction system [43] and a maximum assumed as a reasonable target for the current R&D programme on TER-WCLL system.

The permeation barriers are present on both BB cooling pipes (PbLi/EUROFER interface) and ex-vessel PbLi pipes. The quantitative figure is given by the PRF. The values indicated in Table 7 reflect the current status of the ongoing R&D programme on coating development, reported in section 2.1.

The hydrogen doping of the cooling water is primarily required to maintain a reducing chemical environment and to compensate for the effect of water radiolysis, thereby to control the corrosion potential. Furthermore, the concentration of hydrogen dissolved in cooling water affects the overall tritium permeation rate because of a larger hydrogen/tritium atom recombination flux on the Eurofer/Water interface and, simultaneously, of a wider HT dissociation flux (see [44])These two opposing phenomena must be properly quantified to study the effects on tritium permeation of hydrogen dissolved in cooling water. The min and max values reported in Table 7 indicate a reasonably wide range around the optimal hydrogen concentration needed in cooling water to keep reducing conditions [45].

In Table 8 are reported the different scenarios addressed for HCPB blanket.

Table 8: Parameter scans for HCPBB DEMO 2017 baseline

Case ID	Parameter (unit)	Min/ Inter.	Max
1	CPS by pass flow rate [kg/s]	2	4
2	CPS efficiency [%]	80	95
3	H_2 part. Press. in He coolant [Pa]	1	300
4	He purge gas flow rate [kg/s]	0.25	0.5
5	TER efficiency [%]	80	98
6	T perm. rate from plasma [mg/d]	0	60
7	He leak rate [%inventory/yr]	0	36.5
8	PRF on IHX pipes	1	1000
9	Steam conc. in purge gas [%]	0	4
10	He leak rate from HXs [kg/h] [41]	0.024	0.34

Similar considerations done for WCLL BB can be drawn for the HCPB blanket concept, regarding TER and CPS efficiencies, and by-pass fractions, driven by the ongoing DEMO R&D programme.

The He-purge gas flow rate is addressed here by halving the value adopted in the current TER system design (10000 Nm^3/h) [46], to evaluate possible and more favourable configurations, which still comply with the imposed limits. The steam content in the purge gas is envisaged as an alternative swamping gas to enhance the tritium release from the ceramic breeder and to reduce the HT-driven permeation rate by the aforementioned isotope exchange reaction.

The upper bound set for the parametric study (4 % molar) is assumed. Also, the hydrogen content in the coolant is investigated as a possible measure to counteract the permeating gas from the high-pressure side (purge gas), similarly to Case-5 of the WCLL BB variant. The maximum limit is imposed by technological limitation in the CPS.

The HCPB BB concept does not foresee by design the deployment of permeation/corrosion coatings inside the BB, because of the absence of corroding agents and also due to the low tritium partial pressures inside the purge gas (~ 1 Pa). However, to reduce the environmental parasitic losses, permeation barriers could be installed in the Intermediate Heat Exchanger (IHX) pipes. For stainless steel pipes working between 300 and 500 °C, a PRF equal to 1000 was considered an appropriate and achievable upper limit [47].

When operating with high-pressure water or helium gas, important gas leakages through valves and circulators' sealings can occur during normal operation. Therefore, the coolant leak leads to a direct and non-negligible tritium environmental loss to be accounted for. In this study, the parametric figure needed to quantify the impact of coolant leakages through piping layout is given by the fraction of inventory released per year of operation. The minimum and maximum values reported in Table 8 are inherited from the operating conditions of high-Temperature gas reactors [41]. Another important routine coolant loss is due to the leakage from the primary to the secondary side, and it occurs through the pipes inside the heat exchangers (IHX and SG). The values reported in Table 8 are both inherited from PWR, CANDU and HTGR operations [41].

4.2 Tritium transport analyses

Tritium permeation models at system levels have been developed for both the HCPB and WCLL variants [48]. The models are built using the 1D object-oriented simulation platform EcosimPro. They include the main processes affecting tritium transport such as advection by the fluids, diffusion in solids, recombination and dissociation in the surfaces and chemical equilibrium between species. A complete description of the mathematical equations can be found in [49, 50]. The model diagrams cover the breeder circuit, the solid structures of the blanket and coolant circuits. The fluid circuits include (i) the in-vessel part, where tritium normally permeates from breeder to coolant and (ii) the ex-vessel part formed by piping, extraction systems, IHX and SG. Tritium transport sensitivity analyses have been performed varying the parameters already presented in Section 4.1.

4.2.1 Tritium transport analyses in WCLL

Concerning the WCLL variant, the current configuration named "Case-0" is based on the most recent WCLL design (see [25] for more details). It considers a PRF of 100 and an off-line CPS configuration in the coolant circuit. With this configuration, the permeation rates from breeder to coolant are moderate due to the permeation barriers (~ 400 mg/day).

The water circuits never reach the steady-state since there is no tritium sink in this sub-system (e.g. no CPS). Besides, tritium in water is mostly present in form of HTO and this molecule does not oppose to permeation. As a result, tritium concentration and inventory in PbLi reach an asymptotic value after 1 day but the concentration and inventory in water grow linearly during the whole reactor lifetime (

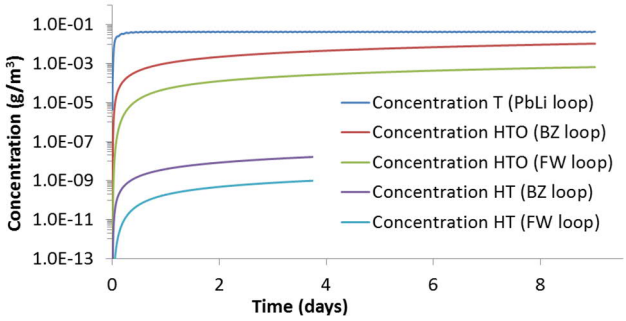

Figure 7).

The highest source of permeation losses is found to be located in the PbLi piping system (~150 mg/day). Indeed, since very small quantities of HT and T2 molecules are solved in water (most of tritium is in form of HTO), little permeation is found in the SG, the IHX and water pipes. According to the parameters reported in Table 7, the following parametric analyses have been performed.

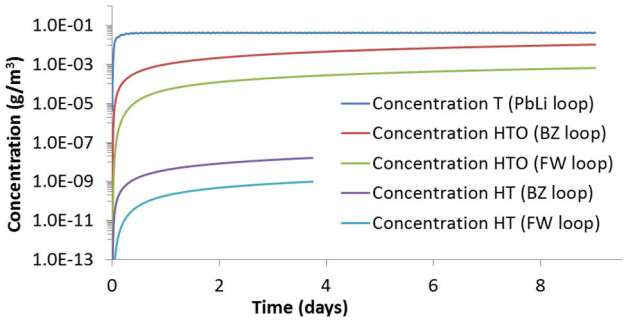

Figure 7. Evolution of tritium concentrations as a function of time in the WCLL Case-0.

Case-1 studies the possibility of including an in-line CPS that process a small amount of water. This has a major impact on tritium concentration and inventory in water since it introduces a new sink in the coolant system. The concentration does no longer grow linearly in time but saturates as depicted in Figure 8. The higher the amount of water the CPS can process, the faster the concentration reaches the steady-state value.

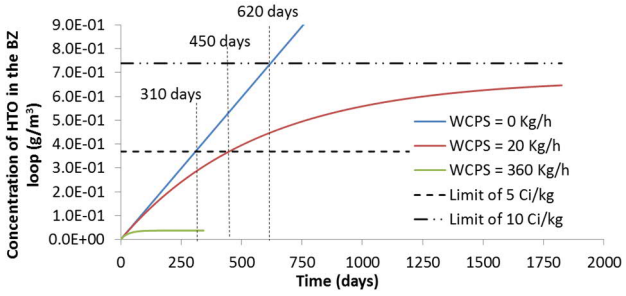

Figure 8. Evolution of HTO concentration in the breeder zone (BZ) water loop for different values of the CPS bypass.

The Case-2 of the WCLL sensitivity matrix analyzes the influence of the TER efficiency. Since this parameter directly affects the tritium concentration in PbLi, it has a direct impact on all model outcomes. For example, reducing the PbLi concentration reduces the concentration gradient in the solid materials. This is translated into lower permeation rate from breeder to coolant and less inventory both in the steel and the water. Nevertheless, considering that only high efficiencies have been considered (higher than 80%), the effect on the results is moderate Indeed, increasing the extraction efficiency from 80% to 95% reduces the permeation rate form PbLi to the coolant to approximately 17%.

Case-3 is dedicated to the influence of the permeation rate through the FW of the BB. This phenomenon is discussed in Section 3. The addition of an extra tritium source in the FW coolant loop (20 mg/day) increases the tritium concentration and inventories in this loop to approximately 66%. Logically, it barely has an impact on the rest of the systems since the breeder zone (BZ) and FW water loops are disconnected.

Case-4 analyzes the impact of the permeation barriers on tritium transport. This is done by modifying the value of the PRF in the PbLi-steel interfaces inside the blanket. As expected, the PRF is one of the most impactful variables. Indeed, increasing the PRF by a factor 10 (from 100 to 1000) decreases the tritium permeation rate and the tritium inventories in the steels and water by the same

factor. However, since the permeation losses are much smaller than the extraction rate in the TER, the tritium inventory in PbLi does not change significantly (less than 0.2%).

Case-5 studies the effect of the hydrogen concentration in water. The WCLL considers a certain amount of hydrogen dissolved in water due to corrosion issues. This has an impact on tritium transport in two ways. On the one hand, it affects the chemical equilibrium between the H_2, HT, T_2, HTO and H_2O molecules. On the other hand, it favours permeation via the recombination of HT molecules in the steel-water interfaces. In the WCLL conditions, the model results predict that, increasing the hydrogen molar fraction (from 8 ppm to 100 ppm), increases the permeation rate from PbLi to water in the blanket only a small amount (~0.1%). However, it increases the permeation rate in the SG and IHX significantly (~75%).

Case-6 considers the possibility of including permeation barriers in the PbLi piping. Logically, including a PRF in the PbLi pipes is an effective way of decreasing the permeation losses in this system. The effect of this parameter on the permeation losses through the PbLi piping is linear by definition.

The last case, Case-7, analyzes the effect of undesired water leaks in the SGs and IHXs from the primary to the secondary loop. The water leaked contains tritium and therefore constitute an important tritium sink for the water loops. Indeed, it is found that for the water leak rates considered (0.3 and 0.5 kg/h corresponding to 3 and 5 Gal/d taking into account the water at PWR conditions, i.e. density of ~701.3 kg/m3 @ 311.5°C and 155 bar), the tritium leaked is much more relevant than the tritium permeated in both SG and IHX. In the worst scenario considered, the tritium leak rate in the SGs after 3.5 years (6.71E-02 g/d) is almost 100 times higher than the tritium permeation rate through the SGs tubes (6.74E-04 g/d).

4.2.2 Tritium transport analyses in HCPB

According to the parameters reported in Table 8, the following parametric analyses have been performed.

In the study carried out for the HCPB concept, Case-1 explores the influence of the flow rate sent to the CPS. The concentration of HT in the coolant increases as the flow rate delivered to the CPS is lowered (2.45E-06 mol/m^3 in current configuration named "Case-0" and 3.65E-06 mol/m^3 in Case-1 with G_{CPS} = 2 kg/s**)**. A similar effect can be seen in tritium permeation due to increased tritium concentrations (~50% more than Case-0) in the coolant. The same behaviour is noted for Case-2 where the influence of the purification efficiency of the CPS is studied. An increase in both parameters, CPS bypass flowrate and efficiency, cause a reduction of the tritium concentration in the coolant (reduction of 25% and 5% when the CPS bypass flowrate and efficiency are increased, respectively).

In Case-3, the increase of the H_2 partial pressure in the He coolant reduces (~88%) the permeation from PHTS to room due to the interaction in isotopic co-permeation [51].

In Case-4, as expected, a lower purge gas mass flow rate produces larger inventories in the breeding blanket (0.69 g in Case-0 and 1.32 g in Case-4). Therefore, the permeation increases, which also affects the concentration in the coolant.

In Case-5, the study of the efficiency of tritium extraction in the TER is performed. The inventory of tritium in the purge gas can be reduced to half when increasing the TER efficiency from 80% up to 98%. The rest of the model outcomes are little affected by this parameter since the study range is short.

In Case-6, the contribution of the tritium that arises from the plasma implanted in the first wall and directly reaches the coolant is studied. It can be perceived that a variation in this parameter mainly affects the concentration of tritium in the coolant (+18%) and therefore the permeation from PHTS to the room (+26%).

In Case-7 the contribution to helium losses in the coolant circuit has been studied. The tritium losses associated with the He leaks do not affect other values of the system since the contribution to tritium leak rate is very low (1.53E-05 g/d).

In Case-8, the permeation through the IHX reduces proportionally to the increase of the applied PRF.

In Case-9, the presence of steam causes a shift in the chemical equilibrium towards the formation of HTO. Therefore, the decrease of HT directly affects permeation rates from purge gas to coolant by a factor 5, and indirectly from PHTS to room and IHX.

In Case-10, as for Case-7, the influence of the leak rate in the intermediate heat exchanger is studied and such parameter does not affect the global values of the system except for the associated tritium leak (~0.001 mg/d).

4.3 Preliminary tritium operating limits and impact on DEMO design

To evaluate the impact of the different parameters analysed in paragraphs 4.2.1 and 4.2.2, some operating and safety limits have been identified and assumed in this work. These limits are reported in Table 9.

Table 9. Tritium operating limits.

Operating limit description	Limit value
Release in Tokamak building	1 [g/y]
Threshold to put into service the DS	10^8 [Bq/m^3]
Threshold for nuclear classification of the secondary/intermediate systems according to ESPN Directive [52]	370 [GBq]
Tritium concentration in water based on CANDU experience	5 [Ci/kg]

The limit of 1 g/y from Tokamak building is posed on the environmental releases to the Stack. However, in this work, it is conservatively used to compare the results in terms of permeation rates directly from the PHTS and TER systems into the Tokamak building. It corresponds to the dose release limits in Normal Operation (NO) to comply with the maximum dose to the population of 0.1 mSv/y applying the dose conversion factor from [53]. Concerning the threshold of 10^8 Bq/m^3, it depends on the concentration of tritium per volume inside the room served by the Detritiation System (DS). In ITER the threshold to switch from Heating Ventilation and Air Conditioning (HVAC) to DS is 1x10^8 Bq/m^3 [54]. In

DEMO, the same threshold for DS intervention is assumed. Regarding the contamination of the secondary/intermediate systems, the Equipment Sous Pression Nucléaire (ESPN) Directive [52] has been selected as a preliminary assumption. In this Standard, equipment is considered nuclear equipment if:

- It is submitted to the Pressure Equipment Directive (PED);
- Provide confinement of radioactive nuclides for at least one of its NO conditions. This means that if the secondary/intermediate system is not ESPN, the activity of any component (vessels[1], not piping) should be kept under 370 MBq (10 mCi) during normal operation. The activity of tritium, ^{16}N and ^{17}N are divided by a factor 1/1000 according to ESPN rules, allowing consequently an increase of the total contamination to 370 GBq (10 Ci).

Finally, the tritium concentration limit of 5 Ci/kg [55] used in CANDU reactor for the heavy water contained in the Calandria is used for WCLL PHTS.

However, considering the early stage of the DEMO design, most of these limits are very uncertain and additional limits should be identified or current limits could be substituted by others (e.g. ESPN is valid only in France, other European countries can have different Codes&Standards). However, for the scope of the study, these limits represent a preliminary term for comparison. In the following, the results obtained by sensitivity analysis are compared with the limits reported in Table 9. However, some assumptions are taken, in particular:

- The releases within the tokamak building used to compare with the limits of 1 g/y are: (i) T permeation rates through the PHTS and TER pipes and (ii) T released due to leakages from the PHTS to the Tokamak Building.
- The activation of the DS system is evaluated in two different areas: (i) the PHTS building area (i.e. area where the PHTS piping are routed) and (ii) the TER building area (i.e. area where the TER piping are routed). To compare the results with the threshold of 10^8 Bq/m^3, the following assumptions are taken:
  - For the PHTS building area: the permeation and leak rates through the PHTS pipes are integrated over time, added and, then, divided by the overall volume.
  - For the TER building area: the permeation rates through the TER pipes are integrated over time and divided by the overall volume.
- To evaluate the contamination of the secondary/intermediate system, the permeation and leak rates through the SG/IHX are taken into account. These rates are integrated over time, summed and multiplied by the ratio between the vessel with the maximum volume and the overall volume[2] of the secondary/intermediate system.

In Figure 9, the comparison between the release limit of 1 g/y with respect to the base case (namely Case-0) and some of the most notable cases identified in the sensitivity matrix (see Table 7 and Table 8) is reported both for the WCLL and HCPB.

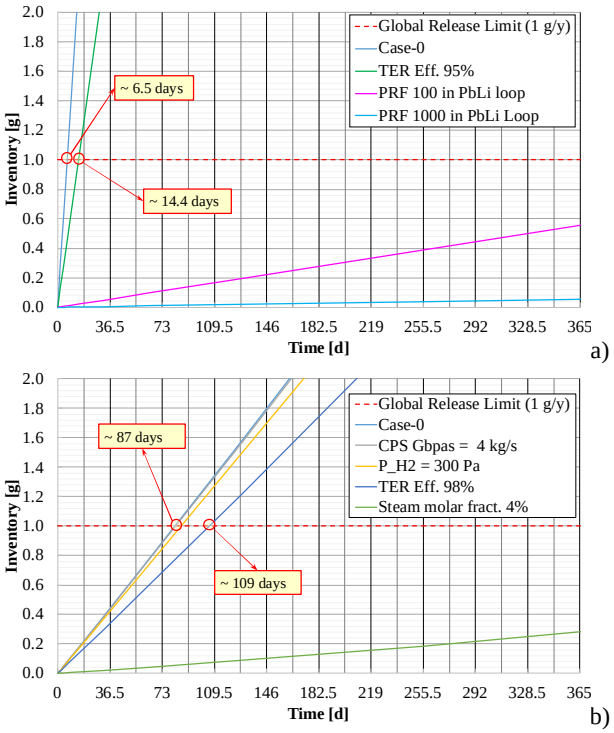

Figure 9. Tritium releases in Tokamak building for (a) WCLL and (b) HCPB.

For the WCLL Case-0, the release overcomes the limit of 1 g after 6.5 days of full-power operation. The major contribution to the permeation losses is from the PbLi piping. The high permeation surface added to the relative high tritium concentration in PbLi produces high permeation rates through the pipe walls to the room (~150 mg/d). The permeation from the water tubes of the FW and BZ circuit to the room is found to be negligible (between 10^{-12} and 10^{-14} g/d). Among the cases identified in the sensitivity matrix (see Table 7), only three have a substantial impact on the minimization of the releases in the Tokamak building. These are the increase of the TER tritium extraction efficiency and the use of permeation

[1] For a vessel: nuclear activity is the product of the volume of the equipment and the volumetric nuclear activity. For pipes, safety devices, etc.: no calculations are required, the Nuclear Pressure Equipment (NPE) level is the highest of the connected equipment (i.e. the vessels).

[2] Note that to determine if a vessel is a nuclear equipment according to ESPN, only the activity contained in the vessel has to be taken into account. The activity of the connected piping is not accounted for. It is assumed homogeneous distribution in tritium concentration for all secondary/intermediate system of WCLL and HCPB. Therefore, the specific allowable activity (A_s) can be determined as:

$$A_s = \frac{370\ Gb}{V_{max}}$$

The overall allowable activity (A_{tot}) on the secondary/intermediate system can be calculated then:

$$A_{tot} = A_s * V_{tot}$$

Therefore, the activity released in the secondary/intermediate system due to the permeation and leak rates ($r_p(t)$ and $r_l(t)$, respectively) should be always below A_{tot} in order to avoid the nuclear classification of the equipment, therefore:

$$\int_t r_p(t) + r_l(t)dt \leq A_{tot}$$

From this equation is easy to derive that:

$$370\ Gb = \frac{V_{max}}{V_{tot}} \int_t r_p(t) + r_l(t)dt$$

barriers on PbLi loop with different PRF (e.g. 100 and 1000, respectively).

Concerning the TER efficiency, the increment has a limited effect moving to overcome of the limit of 1 g after ~14.4 days of full-power operation. Therefore, the efforts to be spent to increase the TER efficiency do not provide a reliable solution to overcome this issue. A different effect is found when permeation barriers are applied to the TER loop (Case-6, Table 7). As it is possible to note in Figure 9-a), the tritium released reduces considerably according to the increment of the PRF. This means that permeation barriers will most likely be necessary for the PbLi piping. It has to be noted that the application of permeation barriers in the PbLi loop does not present the same problem in the BB because: (i) it is not subjected to neutron irradiation and (ii) it has to be deposited on more regular geometries (i.e. mainly pipes). Another solution could be the use of coaxial/guard pipes surrounding the PbLi loop with a controlled atmosphere.

For the HCPB Case-0, the release overcomes the limit of 1 g after ~87 days of full-power operation. The major contribution to the permeation losses is from the TER piping (~11.5 mg/day). Among the cases identified in the sensitivity matrix (see Table 8), only two have a substantial impact on the minimization of the releases into the Tokamak building. These are the increase of the TER T extraction efficiency to up to 98% (Case-5), and the use of steam in the purge gas (Case-9). Concerning the increment of the CPS by-pass mass flow rate (Case-1) as well as the doping of the coolant with H_2 (Case-3), these have limited effects. Therefore, the efforts to be spent on them do not provide a reliable solution to overcome the issue. From this sensitivity analysis, the most promising solution seems to be the use of steam in the purge gas. Indeed, in this way, the tritium binds to steam and not permeate trough the TER piping. Other solutions to be investigated in the future are: (i) the use of permeation barriers in the HCPB TER loop and (ii) the use of coaxial/guard pipes surrounding the purge gas loop with a controlled atmosphere.

Concerning the service of the DS for the PHTS area, for the WCLL, all the analysed cases show that the permeation rate from PHTS pipe to the building area is negligible (between 10^{-14} and 10^{-12} g/d) during NO and without considering any water leak from the PHTS piping. This means that the threshold of 10^8 Bq/m^3 will be never reached. However, due to the high inventory of T in water (between 30 and 145 g), this limit can be overpassed in case of accidental condition with a discharge of water into the building (e.g. Ex-Vessel LOCA). Regarding the HCPB, the results are shown in Figure 10.

As it is possible to see, for HCPB Case-0, the threshold for putting the DS system into service is reached after ~64 days of full power operation due to the permeation through the PHTS pipes. Very high leak rates from the PHTS to the building (e.g. 36.5 % of the coolant inventory per year, Case-7 of Table 8) do not compromise the situation. This is due to the low inventory present in the He coolant (between 0.003 and 0.03 g). As a consequence, the effect of increasing the CPS by-pass mass flow rate (Case-1) does not contribute substantially to the reduction of the releases into the PHTS building area. It is interesting to note the negative effect of the purge gas mass flow rate reduction (Case-4). Therefore, this solution should be avoided by maintaining the current purge gas mass flow rate (0.497 kg/s) or, if possible, increasing it. Some improvements can be obtained with the use of steam in the purge gas (Case-9) and with doping of the coolant with H_2 (Case-3) (the DS is put into service after 240 and 300 days of full-power operation, respectively).

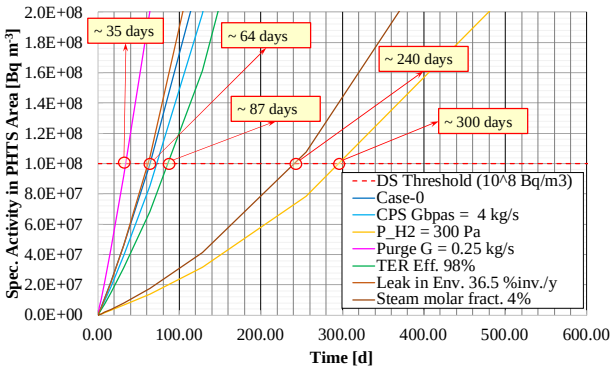

Figure 10. Sensitivity analysis results on DS activation in HCPB PHTS building area.

Concerning the service by the DS in the TER area, the comparison between the DS threshold (10^8 Bq/m^3) and the most important cases analysed both for WCLL and HCPB is reported in Figure 11.

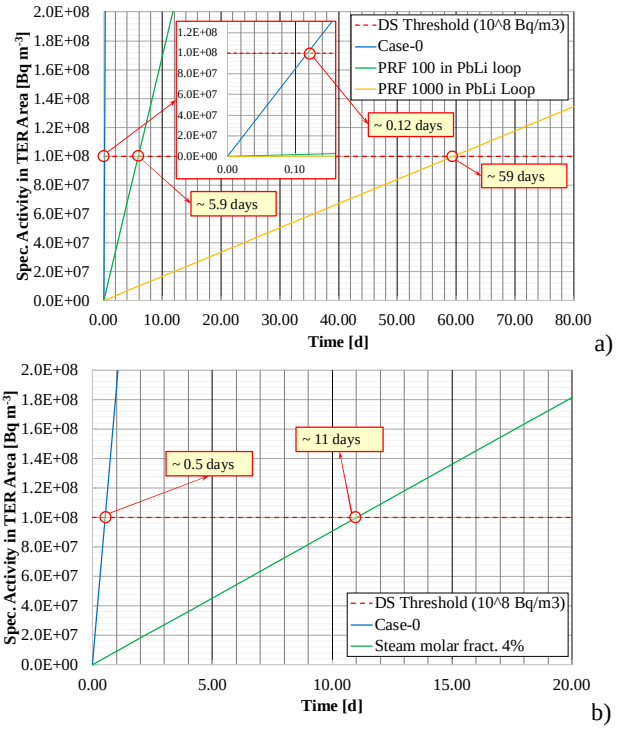

Figure 11. Sensitivity analysis results on DS activation in a) WCLL and b) HCPB area.

For the WCLL (see Figure 11-a), the permeation rates into the PbLi loop are very high in Case-0 (~150 mg/day). This means that the threshold to switch on the DS is reached immediately. The most promising solution, which minimizes the intervention of the DS, is the

introduction of T permeation barriers in the whole TER loop. However, even if a PRF of 1000 is reached, only 59 days of full power operation would be needed before to switch on the DS. This means that other solutions should be investigated to overcome the issue such as the use of coatings with high performances in terms of permeation reduction (i.e. PRF ≈ 10000) or of guard pipes with a controlled atmosphere.

For the HCPB (see Figure 11-b), the permeation rates into the purge gas loop are high in Case-0 (~11.5 mg/day). This means that the threshold to switch on the DS is reached after half a day. The unique parameter investigated that plays a role in the minimization of the T release from the TER loop is the use of steam in the purge gas. It increases the time needed to switch on the DS up to 11 days of full-power operation. However, this solution does not solve the issue at the root. Therefore, other solutions like the use of permeation barriers in the TER loop or the use of guard pipes with a controlled atmosphere should be further investigated.

Regarding the nuclear classification according to the ESPN for the secondary/intermediate systems of both WCLL and HCPB variants, the most important results are reported in Figure 12.

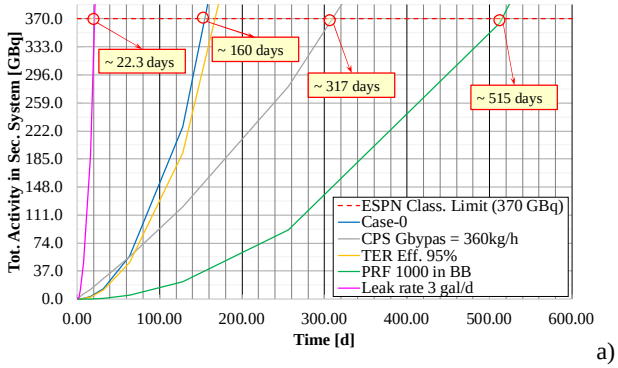

Figure 12. Sensitivity analysis results on ESPN classification of a) WCLL secondary system and b) HCPB intermediate system.

For the WCLL, in Case-0, the secondary system reaches the limit for nuclear classification in about 160 days of full-power operation. The increment to 95% of T extraction efficiency in the TER system (Case-6, Table 7) practically does not play any role. Increasing the CPS by-pass mass flow rate (Case-1) or the PRF within the BB to up to 1000 (Case-4), considerably enlarges the time (e.g. 317 and 515 days of full-power operation, respectively). However, when the operational leak rates from the PHTS to the secondary systems are taken into account (Case-7), the time required to reach the limit for the ESPN classification drops down to ~22 days of full-power operation. A possible solution to solve this issue can be: (i) strong reduction of the T concentration in the PHTS (e.g. increment of permeation reduction factor, an increase of CPS mass flow rate and efficiency) together with (ii) a strong reduction of the operational leaks at the IHX and SG, by improved design solutions for these systems. Other solutions to avoid the nuclear classifications involve the complete draining of the primary and secondary circuits before reaching the limit. Although they can be practicable only if the plant availability is not negatively affected; therefore the frequency of the draining cannot be lower than 1-2 fpy.

For the HCPB, in Case-0, the intermediate system reaches the limit for the nuclear classification in about 6 days due to the permeation through the IHX (between 0.059 and 52.2 mg/d). The reduction of the purge gas mass flow rate (Case-4, Table 8) has a detrimental effect also on the permeation through the IHX reducing, consequently, the time required to reach the limit for the ESPN classification. No significant improvements are obtained by increasing both the CPS by-pass mass flow rate (Case-1) and the TER efficiency (Case-5). Remarkable improvements are obtained with the use of steam into the purge gas (32 days of full-power operation) and the use of permeation barriers (PRF=1000) into the IHX (56 days of full-power operation). It is also interesting to note that, in the case of HCPB, the leaks from the PHTS to the intermediate system through the IHX do not play an important role like it is the case for the WCLL. This is due to the very low T concentration in the He coolant. However, these results show an issue that should be further investigated and solved by employing additional solution like (i) the use of steam into the coolant and (ii) the increase of PRFs at the IHX to above 1000.

Finally, concerning the tritium concentration in water for the WCLL, the limit of 5 Ci/kg is overcome after 310 full power days in Case-0, mainly due to the absence of the CPS and a pronounced permeation from the breeder zone to the coolant. Substantial effects on T concentration in the PHTS are obtained by having a CPS in-line and using permeation barriers within the BB with a PRF of 1000. Indeed, the presence of the CPS, although with limited by-pass mass flow rate (e.g. 20 kg/h), produces a positive effect on T concentration, increasing the time to reach the limit of 5 Ci/kg to up to 450 days of full-power operation. The CPS benefits are more visible when a by-pass mass flow rate of 360 kg/h is reached (i.e. the limit of 5 Ci/kg is never reached) (see also Figure 8

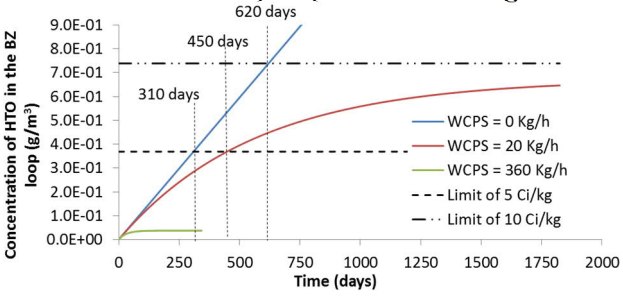

Figure 8). The same effect can be obtained with T permeation barriers within the BB. In all the other cases, after about 300 days of full-power operation, the limit of 5 Ci/kg is overcome and, therefore, the draining of the PHTS has to be carried out, replacing the inventory with fresh water.

5. T start-up, doubling time, and TBR

The Stakeholder requirement DEMO-SHRD-8 states that "The DEMO FPP shall breed sufficient tritium to (i) guarantee its planned operational life and (ii) provide adequate back-up storage in case of unforeseen losses", while the DEMO-SHRD-55 "The DEMO FPP shall breed sufficient tritium to start-up one other FPP" [56]. Several open questions remain: it is not specified when the tritium must be delivered for a future reactor, nor is a regulatory limit on the total site inventory specified, nor is any indication given as to what might constitute an "acceptable" start-up inventory. For instance, it would be trivial for a reactor to deliver tritium to another, at the end of its life. Similarly, a required Tritium Breeding Ratio (TBR) of 0 would be possible, if one is prepared to buy approximately 770 kg (~55 kg per full power year, fpy, per 1 GW fusion power multiplied 7 fpy) of tritium as a start-up inventory that becomes 2500 kg (taking into account a 5.23% of tritium decay per annum and that 7 fpy correspond to approximately 23 calendar years).

These unspecific requirements, coupled with the large uncertainties associated with reactor plasma physics (e.g. for the burn-up fraction, f_b, the design of the fuel cycle, and the performance of the various fuel cycle components), make determining a required TBR problematic.

A simplified dynamic tritium fuel cycle model, described in [57], was used to estimate the start-up inventory, m_{Tstart}, and reactor doubling time, t_d. The start-up inventory and doubling time were estimated using a Monte Carlo approach, running the fuel cycle model on multiple pseudo-randomly generated reactor operation timelines (designed to mimic the unpredictable nature of a first-of-a-kind reactor).

The performance of the fuel cycle depends on a wide range of parameters, for most of which only indicative values are available in the pre-conceptual design stage. The assumptions for these parameters are listed in Table 10; note that these assumptions differ (in some cases significantly) from the values originally presented in [56]. The rest of the parameters are kept as described in [56]. A new and improved tritium retention model was also included for the FW T sink, based on the detailed modelling of tritium retention in plasma-facing surfaces presented in Section 3.

Table 10. Fuel cycle parameter assumptions.

Parameter	Value
Maximum IVC sink inventory (HCPB)	1.11 kg
Maximum IVC sink inventory (WCLL)	1.65 kg
IVC sink parameter (HCPB)	0.154
IVC sink parameter (WCLL)	0.290
Pellet fueling efficiency	0.5
Fuel line pump efficiency	0.9
Minimum TFV sink inventory	1.8 kg [58]
Maximum TFV sink inventory	2.0 kg [58]
Maximum BB sink inventory (HCPB)	0.055 kg
Maximum BB sink inventory (WCLL)	0.185 kg
DIR pumping duration	73 s [58]

Two scenarios were investigated, see Table 11: Scenario I, which corresponds to the present set of assumptions regarding overall reactor performance, and Scenario II, which is a more pessimistic set of assumptions for the fuel cycle.

Table 11. Preliminary Monte Carlo fuel cycle model maximum results (over 200 runs).

	Scenario I (TBR=1.05, availability=0.3, f_{DIR} =0.8)		**Scenario II (TBR=1.02, availability=0.2, f_{DIR} =0.6)**	
	HCPB	**WCLL**	**HCPB**	**WCLL**
m_{Tstart} [kg]	2.84	2.99	3.50	4.15
t_d [y]	5.33	6.26	28.76	31.92

In Scenario II, the reactor doubling time is approaching the reactor lifetime, which some would consider not to equate to tritium self-sufficiency. The tritium start-up inventories in both scenarios are significantly lower than originally reported in [56], owing chiefly to the reductions in the inventory terms in the Tritium, Fuelling and Vacuum (TFV) systems and the lower T retention in the BB than previously assumed.

These results are indicative of the present understanding and should be re-visited once the conceptual design of the TFV systems has been finalised, and once more reliable estimates are available for more of the fuel cycle parameters. A range of sensitivity studies should also be carried out on the various parameters, in order to ensure that the performance of the fuel cycle is robust to uncertainties in key parameters, such as the plant availability (AV) and the burn-up fraction. However, based on these initial results, the situation is notably better than previously assumed and points to the possibility of reducing the required TBR (once the plant high-level requirements for tritium self-sufficiency have been further specified).

6. Conclusions

The experimental campaign on permeation barriers has demonstrated promising performances in terms of permeation reduction factors of about 100-10000 for the three depositing technologies (i.e. PLD, ALD and ECX). These performances have been confirmed also under thermal cycling and BB operating conditions (e.g. PbLi flow). Preliminary tests under neutron irradiation have been performed demonstrating good results in terms of PRFs (~250 for the PLD technology), and no corrosion attack by PbLi. However, experiments under neutron irradiation should be further conducted to complete their qualification. Efforts should be also dedicated to the scale-up to BB relevant geometries and the industrialization of the processes.

Regarding the tritium permeation experiments, the set-ups to reproduce WCLL and HCPB conditions have been constructed and preliminary results on the co-permeation of Q2 species in He/He or He/water have been

obtained in the two small-scale experiments realized at KIT.

Several progresses have been done on the experimental facilities for testing PAV with V and Nb permeable membranes at CIEMAT and ENEA. The facilities are now ready to enter into the testing phase providing important results on the extraction of tritium from PbLi extrapolable to DEMO scale in early 2021.

Concerning the tritium permeation and retention through/into the FW, the studies show the important role of the W armour. In particular, tritium retention increases considerably according to the thickness of the armour. Therefore, its presence should be minimised as much as possible to reduce the tritium needed to saturate the structures during the start-up or that can be mobilizable during an accident. The possibility to reduce the W thickness would be in line with the Limiter protection strategy already pursued for DEMO. However, further studies should be performed to assess the minimum thickness needed to protect the FW.

Regarding the tritium transport analyses and the verification of the parameter sensitivity studies versus the operating limits have demonstrated:

- for the WCLL, (i) the necessity to have an in-line CPS to minimise the tritium concentration in the coolant and, therefore, to reduce the contamination of the secondary system; (ii) the increment of TER extraction efficiency does not have a big effect on T permeation into the coolant and, consequently, in the tokamak building and secondary system; (iii) a detrimental effect of the increment of H_2 partial pressure in the coolant to reduce the permeation losses in the IHX and SG; (iv) the necessity of permeation barriers within the BB and in the TER loop (PRF between 100 and 1000); and (v) the necessity to minimise the water leaks (<< 3 gal/d) within the IHX/SG to reduce the T contamination of the secondary system.
- for the HCPB, (i) the tritium permeation through the IHX and PHTS piping plays an important role with respect to the leakages into the tokamak building; (ii) the necessity to have permeation barriers (PRF between 100 and 10000) in the PHTS piping and IHX as well as in the TER loop to reduce the permeation into the tokamak building; (iii) the necessity to increase the performances of CPS and TER in terms of both mass flow rates and extraction efficiencies to reduce the tritium permeation; (iv) positive effects of adding steam to the purge gas to trap tritium, reducing the permeation to the coolant; and (v) positive effects in the use of H_2 in the coolant to reduce the T permeation through the PHTS piping.

Regarding the fuel cycle, its inventory is dependent on a wide variety of fuel cycle-specific parameters, most importantly the total throughput requiring treatment in the tritium plant. This is mainly governed by the DIR-fraction, as well as the BB choice and the accompanying CPS and TERs technology selection. Direct Internal Recycling has the potential to relax key design requirements for the EU-DEMO (such as a TBR between 1.02 and 1.05) or mitigates poor performance (such as with f_b or AV). Seeing the fuel cycle development is still in its pre-conceptual design phase, no fixed design point exists. The results have shown the necessity to define a strategy to handle the excess of tritium towards the end of the lifetime (larger storage, selling, modulation of TBR in the course of the operational years).

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