Feasibility of D-D start-up under realistic technological assumptions for EU-DEMO

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One of the main issues in view of the realization of a DEMOnstration fusion reactor is the availability of a sufficient external supply of tritium (T) to start operation. T is an unstable nuclide, which is almost absent in nature and is currently available as by-product in e.g. CANDU, whose operation in the next decades (both in terms of life extension of existing reactors and construction of new ones) is at the moment under debate. During DEMO operation, T will be generated on-site by breeding blanket, employing the neutrons originating from D-T reaction. However, it is considered that a certain initial amount of T is needed to start operation, the so-called start-up inventory. An alternative approach consists of obtaining the start-up inventory exploiting reactions occurring in a D-D plasma, which generate T both directly in the plasma and via breeding in the breeding blanket. In the present paper, the conditions under which D-D start-up becomes a favorable option for a power plant are discussed. The analysis mainly focuses on the EU-DEMO reactor concept, for which design data are sufficient for a fairly quantitative evaluation of the relevant parameters. It is found that the unavoidable presence of elements requiring saturation before they are able to release significant amounts of T clamps the T production rate to the same order of magnitude as D-D reaction rate. Thus, under very optimistic assumptions, several hundreds of full-current D-D discharges are necessary for T to be available for plasma fueling, but more realistic estimates let this number raise up to several thousands.

Keywords: Tritium, DD-start-up, EU-DEMO, fuel cycle

1. Introduction

The most efficient fusion reaction for nuclear power plants has been identified to be deuterium-tritium (D-T), in view of its higher cross-section at relatively lower temperature in comparison to other fusion reactions. An obvious drawback of this (to a fair extent forced) choice is that one of the two atomic species involved in the process, namely T, is an unstable nuclide with a relatively short half-life of 12,3 years, and is thus not to be found in nature in meaningful quantities (~3.5 kg in the upper atmosphere from cosmic rays spallation [1]). During full-power operation, T is produced on-site by the breeding blanket, but a certain amount from external sources is required to start operation. The main sources of T at the moment are the CANDU heavy water fission reactors (HWRs), where T is a by-product of neutron irradiation of heavy water (T production of ~230 g per GWe and full power year [1]). CANDUs and other HWRs, however, may no longer be in operation when DEMO reactor(s) are entering the nuclear phase, and in view of the short half-life of T, the risk that not enough T is available to start operation is not negligible (for a complete and extensive analysis of the T inventory in the world in principle available to the civil nuclear market, and its availability in the next decades, the reader is referred to [2-5] – the T currently in military holdings is here neglected, since its availability for civil use is deeply uncertain). For these reasons, different reactor start-up schemes requiring smaller amounts, or even no T, are particularly attractive. One of the possibilities is the so-called D-D start-up, i.e. the on-site breeding of an initial amount of T by means of deuterium-deuterium reactions, enabling access to full power operation without any need of T from external sources. This idea has been already analyzed in the past in different publications [6-10]. An important role in determining the feasibility of this approach is played by the T absorption by the plant components and/or hold-up of tritium as gas or liquid in the subsystems to enable steady-state operation, which represents a sink for the T available for burning until a saturation (at least in some key component) is achieved, see e.g. [9]. This is the purpose of the present paper, where the feasibility of a D-D start-up for EU-DEMO is discussed with respect to the interplay between T breeding and the saturation of the T-facing plant components.

The paper is structured as follows: in section 2, a general discussion about the time evolution of the T content in the plasma in case of pure D-D start-up and in presence of components requiring saturation is presented. The analysis is then focused on the European DEMO reactor [11], and in section 3 its fuel cycle is briefly described. In section 4 and 5 the evolution of saturation for the components during D-D start-up is evaluated, distinguishing between fuel cycle and breeding blanket

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(BB). In section 6, the number of necessary EU-DEMO D-D discharges to achieve saturation is estimated. In section 7, the assumptions adopted throughout the calculation are discussed, while conclusions are drawn in section 8.

2. Dynamic evolution of T content in the plasma

In a fusion reactor able to breed, there are three different sources of tritium, namely:

1) Tritium generated directly in the plasma by D-D reactions.

$$D + D \rightarrow T + p \qquad (1)$$

2) Tritium generated by interactions of neutrons from D-D reactions (2) and Li-containing materials ($^{6}Li + n \rightarrow T + {}^{4}He$) in the breeding region.

$$D + D \rightarrow {}^{3}He + n \qquad (2)$$

3) Tritium generated by neutrons originating from D-T reaction (3) and Li-containing materials in the breeding region.

$$D + T \rightarrow {}^{4}He + n \qquad (3)$$

The two different D-D reactions have the same probability to occur. Reaction (3) is the most efficient for the T production once the full-power operation is reached, since the D-T cross-section is about two orders of magnitude larger than the D-D for the same plasma temperature and density, thus the associated neutron flux is significantly higher.

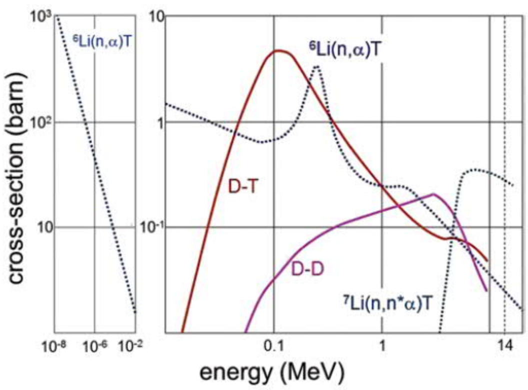

Fig. 1: Cross sections of the ^{6}Li and ^{7}Li T breeding reactions as a function of the impacting neutron energy. D-D and D-T cross-sections as a function of the plasma energy are reported as well. Figure is taken from [13].

The relevant parameter is the so-called T production rate b, which is defined as the "excess" T generated on average by every fusion-born neutron. This parameter is strongly dependent on the machine design, but this discussion goes beyond the scope of this paper (for more details on this topic see [12]). So, for each D-T reaction (3), $(1 + b)$ T atoms are produced in the breeding zone and 1 is burned, leading to a net gain of b T atoms per reaction. Instead, neutrons originating from reaction (2) possess lower energy than the corresponding D-T ones (2.45 MeV versus 14.1 MeV). This translates into a lower T production yield in the breeding zone ψ_{DD}, which in general depends on ^{6}Li enrichment, but typically $\psi_{DD} < 1$ T/neutron. Fig.1 shows the cross-sections for ^{6}Li and ^{7}Li T breeding reactions as a function of the impacting neutron energy. In the simplified model presented here, continuous tokamak operation is assumed. In a purely D-D start-up, the tritium produced in the breeding zone (both from D-D and, later, from D-T) is partially available to be re-injected in the plasma and partially retained in the components until saturation (at least in some relevant components) is reached. The parameter φ_B indicates the fraction of T which can be immediately re-injected in the plasma (incidentally, the delay between the T production in the breeding zone and its actual re-injection in the plasma is here neglected, since it has an impact on the dynamics but not on the conclusions of the present analysis). A thorough discussion on the determination of φ_B in EU-DEMO is found in section 5. In parallel, T is removed from the plasma via pumping, and then re-injected following a direct or indirect fuel cycle (for a detailed review of the DEMO fuel cycle concept, the reader is referred to [10]). During D-D start-up, it is in general not possible to prevent part of the exhaust T from the plasma being absorbed by the first wall (i.e. T implantation) and by the fuel cycle components (pipes, isotope separation systems and so on). It is here indicated with φ_P the fraction of exhaust T which is re-injected in the plasma. Again, the delay between the T pumping and its re-injection in the plasma is here neglected for the same reasons elucidated above. A thorough discussion of the determination of φ_P in EU-DEMO is found in section 4. In general, φ_B and φ_P can be a function of time, or better a function of the T content already absorbed by the structures and of the T enrichment of the exhaust gas flow. However, in the initial phases, when T content is much below the final target value, this dependency can be neglected. This is justified *a posteriori* in the following sections. Once the saturation of the components is reached, then by definition $\varphi_B = \varphi_P = 1$, and a dynamic equilibrium is established.

In view of the assumptions discussed, the balance equation for the number of T particles in the plasma $N_{T,p}$ can be written as:

$$\frac{d}{dt}N_{T,p} = (\varphi_P - 1)\Phi_{p,T} + \frac{(1 + \psi_{DD}\varphi_B)}{2}\int_V dV < \sigma\, v >_{DD} n_D^2 + [\varphi_B(1 + b) - 1]\int_V dV < \sigma\, v >_{DT} n_D\, n_T \qquad (4)$$

The first term at the right-hand side represents the losses of T due to absorption of the components in the fuel cycle (here, $\Phi_{p,T}$ indicates the rate at which T is pumped, which is, of course, a function of the plasma T content, and is equal to zero at $t = 0$, when no T is present in the plasma). It is always negative, except when saturation is reached, i.e. $\varphi_P = 1$, becoming null – meaning that what is pumped is again re-injected without affecting the T inventory in the plasma (recall that delays in re-injection are here neglected). With "saturation" it is here indicated that there is no net transfer of T from the components to the exhaust gas, which requires that a certain amount of T, generally increasing with increasing T concentration in the gas, is retained in the structures, until the mutual exchanges equilibrate. Thus, it represents a net sink for T in the plasma.

The second term represents the generation of T from D-D reactions, both directly in the plasma and from breeding. Part of what generated in the breeding zone is retained by the structures or by the functional materials (e.g. armour, breeder and neutron multiplier), and this is reflected by the presence of the parameter φ_B. This term is always positive, although it tends to become subdominant by high T concentrations (here, $<\sigma v>_{DD}$ represents the energy-averaged cross-section of the D-D reaction, whereas n_D is the local deuterium density, the integral being carried out on the entire plasma volume and thus taking into account the spatial dependence of both quantities).

Finally, the third term takes into account the T generation in the breeding zone due to D-T reactions in the plasma. Clearly, this takes place only when T is actually present in the plasma. Each D-T reaction obviously involves a T atom (here $<\sigma v>_{DT}$ represents the energy-averaged cross-section of the D-T reaction, whereas n_T is the local tritium density).

The parameter

$$\beta = 1 - \varphi_B(1 + b) \tag{5}$$

is of particular relevance in determining the behavior of the solution of Eq.(4). If negative, the plant produces more T from D-T reactions (i.e. via breeding) than is burned and absorbed in the breeding zone for saturation. When sufficiently large in absolute value to exceed the T retained in the fuel cycle, this term allows a fast growth of the T content in the main plasma – up to a 50% concentration, above which a further increase in the T content is inconvenient. This point is not captured by Eq.(4) which mainly deals with the initial phases of D-D start-up. Note that the minimum value of β as $\varphi_B \to 1$ is $-b$, i.e. the breeding factor fixes the maximum achievable T generation rate. If β is positive, or negative but not sufficiently large to compensate the first term, Eq.(4) predicts an initial growth of the T content in the plasma (when the D-D reactions dominate since no T is present) up to a point where the right-hand side of Eq.4 becomes zero, and thus a stationary state is reached ($dN_{T,C}/dt = 0$). If, for simplicity, the T absorbed in the fuel cycle is neglected ($\varphi_P \to 1$) and $\beta > 0$, an estimate of the T fraction in the plasma when stationarity is reached $f_T = n_T/n_D$ is approximated by

$$f_T \approx \hat{\alpha}\frac{(1 + \psi_{DD}\varphi_B)}{2\beta}, \tag{6}$$

where $\hat{\alpha} \approx O(1e-2)$ is the volume-averaged ratio between the D-D and D-T reaction cross-sections (the exact value depends on the kinetic profiles of the plasma scenario, which are not calculated here in order not to stick to a particular design concept). Note that, if $\varphi_p < 1$, the value of f_T would be smaller. In case the breeding zone needs complete saturation of components before releasing any T (i.e. $\varphi_B \to 0$ during D-D start-up), Eq.(6) reads:

$$f_T \approx \frac{\hat{\alpha}}{2} \ll 1. \tag{7}$$

This can be interpreted as follows:

- In presence of components which need to retain a certain amount of T before any T can be extracted and employed in the plasma, the equilibrium concentration of T in the plasma remains very low – of the order of $\hat{\alpha}$.

- This means that, until then, the T generation in the plant has a rate of the same order of magnitude as the D-D reaction rate. In other words, D-D reactions set the speed at which saturation is achieved.

From a physical point of view, Eq.(7) means that equilibrium T concentration is reached when the generation of T from reaction (1) equals the rate at which T is burned in D-T reactions. The D-T reaction and the D-D reaction (2) cannot enhance the T content in the plasma until the bred T is retained either in the breeding region or by other components in the fuel cycle. Note that assuming no delay between T exhaust pumping, re-injection and burn also implies a burn-up fraction of 100%. A lower burn-up fraction, which would be closer to reality, would correspond to a higher T concentration in the plasma to achieve the same equilibrium reaction rate, and, hence would increase the number of discharges needed to achieve the start-up.

If a small fraction of the bred T is instead not retained, but is released and re-injected in the burning plasma, then Eq.(6) applies – and the T concentration equilibrates to a higher value. When most T is instead released by structures, then β becomes negative and the T content in the plant grows indefinitely (at least in the limit of this simple model). In the following sections, the parameters φ_B, φ_P as well as the D-D reaction rate are discussed for the current EU-DEMO configuration. In particular, it will be shown that the limit $\varphi_B \to 0$ is a realistic approximation. For this reason, Eq.(7) can be considered as a valid estimate. In the conclusions, in view of these results, the feasibility of a D-D start-up in EU-DEMO is discussed.

3. Description of EU-DEMO fuel cycle

The DEMO fuel cycle is based on a three-loop architecture [10,14] consisting of the direct internal recycling loop (DIRL), the inner tritium plant loop (INTL) and the outer tritium plant loop (OUTL). The torus exhaust gas pumped at the divertor enters the metal foil pumps (MFP), where large fraction of pure hydrogen equal to 80% (permeate) of the exhausted hydrogen stream is separated by means of plasma driven super permeation, leaving the residual other gases as retentate [15]. Both effluent streams are compressed using serial combinations of linear diffusion and liquid ring pumps. A detailed description of the DEMO pump train can be found in [16]. The permeate -pure hydrogen, all isotopes - is directly routed back to the gas distribution control and monitoring (GDCM) unit and available for refuelling (DIRL). The retentate is sent to the exhaust purification system (EPS), where the majority of the remaining hydrogen isotopes are separated by two serial pressure-driven permeators designed for a combined efficiency of 99.75%. The further retentate is sent to the OUTL for a successive treatment, and, of this second permeate, 4% is processed in the isotope rebalancing and protium removal system (IRPR), whilst the remainder is also routed to the GDCM. The IRPR employs a temperature swing absorption process [17] to separate the incoming gas into two streams. The first stream is virtually protium-free with an elevated tritium content which is also sent to the GDCM. The second contains instead elevated protium and deuterium fractions, and is sent to the OUTL for further processing. Here several systems are employed that serve to extract any remaining tritium in these streams, most importantly the isotope separation system (ISS) which produces hydrogen in fuel quality – i.e. with a reasonably low value of protium (< 1%). Fig.2 depicts schematically the fuel cycle and the associated mass flows just described.

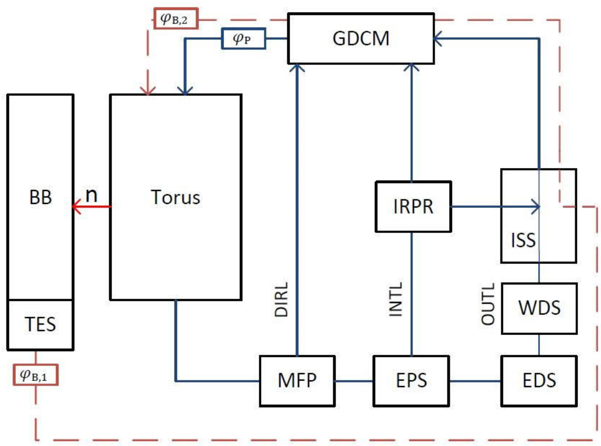

Fig. 2: Simplified layout of the DEMO fuel cycle including tritium pathways from the reactor exhaust (straight blue line) and breeding blanket (dashed red line).

There, the loops determining the parameters φ_B and φ_P introduced in the previous section have been highlighted in red and blue, respectively. In order to separate the different role of breeding blanket system - including the tritium extraction system (TES) – from the ISS, which is common to the exhaust loop just described, the parameter φ_B has been decomposed in the product of two parameters $\varphi_{B,1}$ and $\varphi_{B,2}$, as in Fig.2.

$$\varphi_B = \varphi_{B,1}\varphi_{B,2} \quad (8)$$

In particular, $\varphi_{B,1}$ is related to the breeding blanket system, (breeding zone and TES), whereas $\varphi_{B,2}$ refers to the part of the fuel cycle which is in common between the BB loop and the exhaust. The evaluation of factors φ_P, $\varphi_{B,1}$ and $\varphi_{B,2}$ is the subject of next sections.

4. Saturation of the fuel cycle

The main functions of the OUTL in Fig.2 are the purification of liquid and gaseous exhaust streams for reuse or release to the environment as well as the final separation and purification of hydrogen isotopes for fuelling and fuel balancing. Depending on the employed BB concept, streams from the TES and coolant purification system (CPS) also require further treatment in the OUTL. For water detritiation (WDS), the Combined Electrolysis and Catalytic Exchange (CECE) process is foreseen. Non-hydrogen gases to the stack are detritiated in the exhaust detritiation system (EDS) via the use of wet scrubber columns and the final hydrogen isotope separation and fuel recovery is achieved by a series of cryogenic distillation (CD) columns in the ISS.

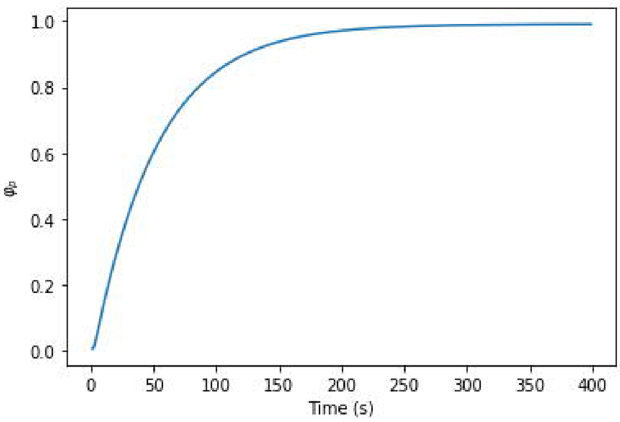

Fig. 3: Step change response of the DIRL and INTL starting from D_2 operation to the nominal exhaust tritium content of z_T=0.485 reaching φ_P= 0.99 after 350s.

These fuel cycle systems constitute a tritium sink that is filled under normal operation with tritium being present either as an operational inventory (i.e. tritium in gaseous form or in water as a result of the system operation e.g. in the form of liquid hold-ups in columns) or as sequestered inventory that has permeated into structural material. As the fuel cycle is operated at or below ambient temperatures in the majority of systems and piping, and tritium partial pressures are very low during the D-D start-up, no significant permeation is expected to occur. Tritium entering the fuel cycle therefore only becomes unavailable for fuelling if it remains in the systems for the build-up of operational inventories, until their nominal operation point is reached (neglecting possibly low surface loadings).

To determine the tritium retention of the torus exhaust φ_P, the Fuel Cycle Simulator developed at KIT [17,18] is used. Fig.3 shows the step change response of the DIRL and INTL from pure D_2 operation to the composition expected during normal operation with an output molar tritium fraction of z_T=0.485. As can be seen, more than 99% of the exhausted tritium content is available for reinjection in less than 350 seconds. The comparably fast response time is achieved by the continuous operating principle of the employed systems, only featuring gaseous inventories that do not require the buildup of concentration profiles. This also means that this effect is independent of the initial gas composition and that every change thereof propagates through these loops in a fraction of a discharge. If looking at the tritium content in the torus exhaust the overall limit of these two loops is hereby given by the fraction of tritium which is directly routed to the GDCM by either the DIRL or the IRPR bypass in the INTL. With the MFP designed to recycle 80% of the torus exhaust hydrogens, an efficiency of the EPS of 99.75% and a bypass fraction of the IRPR of 96% a theoretical design maximum of φ_P = 0.99 for the reactor exhaust is reached. For this model it is therefore assumed that φ_P = 1 and time delays between pumping and fuelling are judged negligible compared to the overall time scale for the D-D start-up. Eq.(6) therefore applies.

The fuel cycle is also tasked with processing the effluent of the Tritium Extraction System (TES). In both blanket concepts, significant amounts of protium are used as doping agent in the recovery of the bred tritium. To avoid the contamination of the fueled hydrogens, the designated operation point of the CD columns in the ISS must be first achieved. Usually, this is done in a dedicated commissioning phase in order to ensure nominal operation and optimal performance of the system. If this process is incorporated into the D-D start-up phase, the final column (producing hydrogen in fuel quality) has to be operated in total reflux at the bottom until the steady state operation point is reached. This is not achieved until an amount of tritium equal to the steady state operational inventory has been routed to the ISS. Even with the use of the Direct Internal Recycling (DIR) [14] alleviating a majority of the load on these systems the operational inventories are expected to be considerably higher than e.g. in ITER [10]. Instead, for a saturation of the TES components themselves, a short discussion is provided in the following.

As a lower bound (being conservative in this case) the ITER ISS maximum tritium inventory of 220 g is used as a reference value [19]. $\varphi_{B,2}$ is therefore zero until at least 220 g of tritium have been extracted from the blankets or routed towards the ISS from the DIRL and INTL.

5. Saturation of the breeding blanket system

The breeding blanket (BB) is one of the components, together with divertor and limiters, that directly faces and envelopes the plasma, and it is by far the largest in terms of surface (about a factor 10 above the divertor). However, it is the only one designed to breed T in order to achieve the fuel self-sufficiency [20]. Currently, two concepts have been identified as possible candidates for the European DEMO reactor. These are the Helium Cooled Pebble Bed (HCPB) [21,22] and Water Cooled Lithium Lead (WCLL) BB [23]. These concepts differ for the coolant used, helium (@ 8 MPa, 300-520°C) or water (@15.5 MPa, 295-328°C), and for the breeder, neutron multiplier and T carrier (Pb-15.7Li for the WCLL and Li_4SiO_4 + 35mol% Li_2TiO_3 ceramic breeder pebble bed, $Be_{12}Ti$ prismatic blocks and He purge gas for the HCPB). In the present analysis, focus is given to WCLL concept, but this choice has no major impact on the conclusions. Note that, at the moment, no final decision has been taken on which of the two blanket configurations will be employed in EU-DEMO. Also, such discussion goes beyond the objective of the present paper, which does not purport to compare the two concepts in any way, nor suggesting any preference. The tractation is limited to one only for the sake of brevity.

The breeding blanket offers big volumes (about ~1500 m^3), where the T can be retained, and large surfaces, where the T can permeate (e.g. the ~1400 m^2 of the first wall, FW, through which part of the T injected in the plasma permeates into the BB). However, also other parameters play an important role in the saturation of structural and functional materials of the BB. This is, for instance, the operating temperatures of the BB and ancillary systems that strongly affect the T residence times.

Preliminary results [24] on T permeation analyses have shown that the T retained in BB and in the ancillary systems (e.g. TES and primary system) may change considerably according to (i) the system performances (e.g. CPS by-pass flow rate, TER system efficiency, and permeation reduction factor), and (ii) the operational parameters (leaks from the steam generator/heat exchanger/piping, doping hydrogen pressure in the coolant and T-carrier, etc.), see Table 1. It is worth to note that these inventories are calculated assuming the operating temperatures of the BB system, therefore they may represent a very optimistic assumption if extrapolated to the D-D plasma where the power released in the blanket can be at least one order of magnitude lower than the one with a D-T plasma (a discussion on this point is found in section 7).

Table 1: T quantities to saturate the BB system under different assumptions [24].

		WCLL	
		Min	**Max**
In-VV	Coolant [g]	5.4	58.8
	Steel [g]	2.9	3.5
	Breeder/ Multipl./ T-Carrier [g]	30.0	36.0
Out-VV	Coolant [g]	9.1	99.2
	Steel [g]	~4e-3	0.2
	Breeder/ Multipl./ T-Carrier [g]	0.4	1.0
Total [g]		**47.9**	**198.7**

These represent the quantity of T for the saturation of the structures and functional materials (e.g. breeder and multiplier). However, in a a liquid-metal based BB like the WCLL and the T produced in the T-Carrier (i.e. PbLi) does not permeate in the structures and coolant and it is immediately available for the extraction, one can derive that, in terms of T release, $\varphi_{B,1}$ scales roughly linear with the T content normalised to the saturation value corresponding to the overall amount of T at equilibrium (here rounded to 40 g). At low T content though, there is almost no release [24]. The dependence of $\varphi_{B,1}$ on the T content in the BB is illustrated in Fig.4.

6. Calculation of saturation rate in EU-DEMO

To calculate the D-D reaction rate in EU-DEMO, the code PLASMOD has been employed [25,26]. PLASMOD is a simplified 1-D transport model, solving the transport equations for all ions species and electrons, both for density and for temperature, with imposed transport coefficients. Currently, PLASMOD has been implemented as an advanced transport solver in the systems code PROCESS [27,28]. The calculation has been performed by assuming geometry, field and current from the EU-DEMO 2018 baseline [29], with a pure D-D plasma instead of a D-T mixture as for the full-power phase. For the same operational parameters, a D-T plasma (50%-50%) provides 2000 MW of fusion power, with 7.2e20 D-T reactions per second. Differently than the indicated operational point though, the auxiliary heating power was increased up to 130 MW, in order to maintain an H-mode operation (in a D-T discharge, the fusion alpha power is in reality largely sufficient for this purpose). The value of 130 MW has been chosen since it is thought to be the maximum auxiliary power which can be installed with the available ports in EU-DEMO [30]. With the assumptions discussed, the D-D reaction rate (both branches) Γ_{DD} is found to amount to

$$\Gamma_{DD} = 1.2e19 \text{ sec}^{-1} \tag{9}$$

Under the very optimistic assumption that all T generated in the plasma by reaction (1) is immediately burned (which is equivalent to $\varphi_P = 1$ since we ignore the time delays, this being again optimistic), the generation of T in the breeding region amounts then to

$$\Gamma_{T,BB} = \frac{1}{2}\Gamma_{DD}(1 + b + \psi_{DD}) \tag{10}$$

which corresponds to $\Gamma_{T,BB}$= 0.99e19 sec^{-1} for $b = 0.05$, which is the target EU-DEMO value as in [12], and $\psi_{DD} = 0.6$ following [7, 31] and references therein.

Table 2: T quantities to saturate each of the components.

Component	T for saturation [g]
BB (TES)	40 (* saturation inventory only in PbLi)
ISS	>220
DIRL & INTL	Negl.

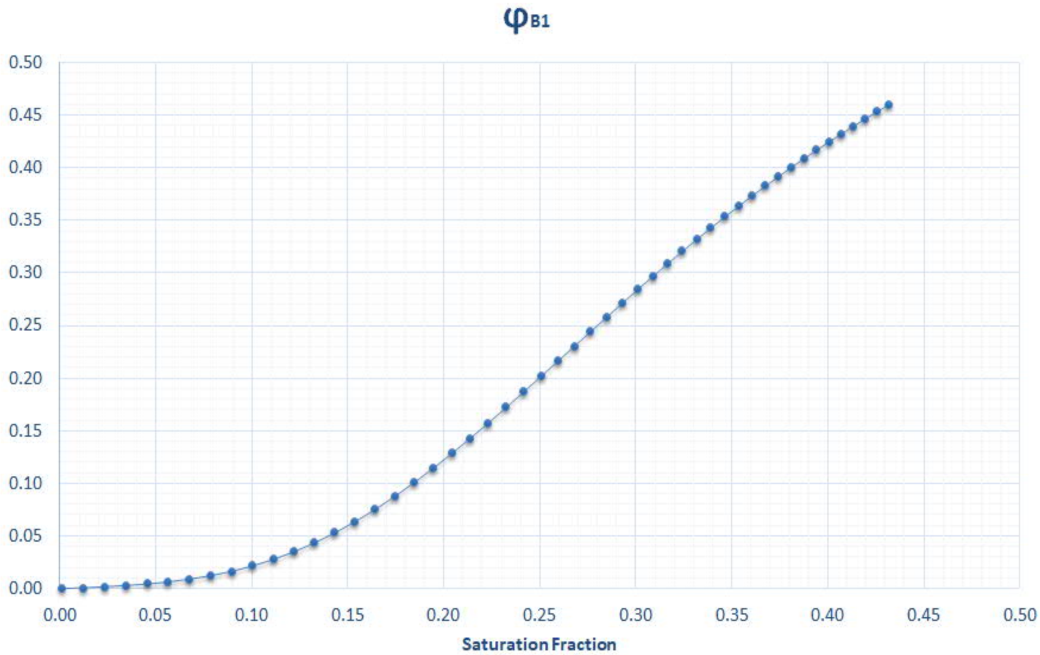

Fig. 4: Dependence of the factor $\varphi_{B,1}$ on the T content in the breeding blanket, normalized to the saturation value (rounded to 40 g).

At this point, one can calculate how much time is needed in DEMO to reach a tritium-saturation level in the breeding blanket and the ISS. It is assumed that the T flow from the blanket to the ISS follows the curve in Fig.4. Since for blanket saturation higher than 45% the numerical calculations started to exhibit some oscillations in the T flow, due to the different timescales among the involved volumes, the curve in Fig.4 has been simplified by a linear function for saturation fractions between 45% and 100%, (with $\varphi_B = 1$ at 100% and $\varphi_B = 0.45$ at 45%). Table 2 summarizes the T quantities needed for saturations assumed in this analysis.

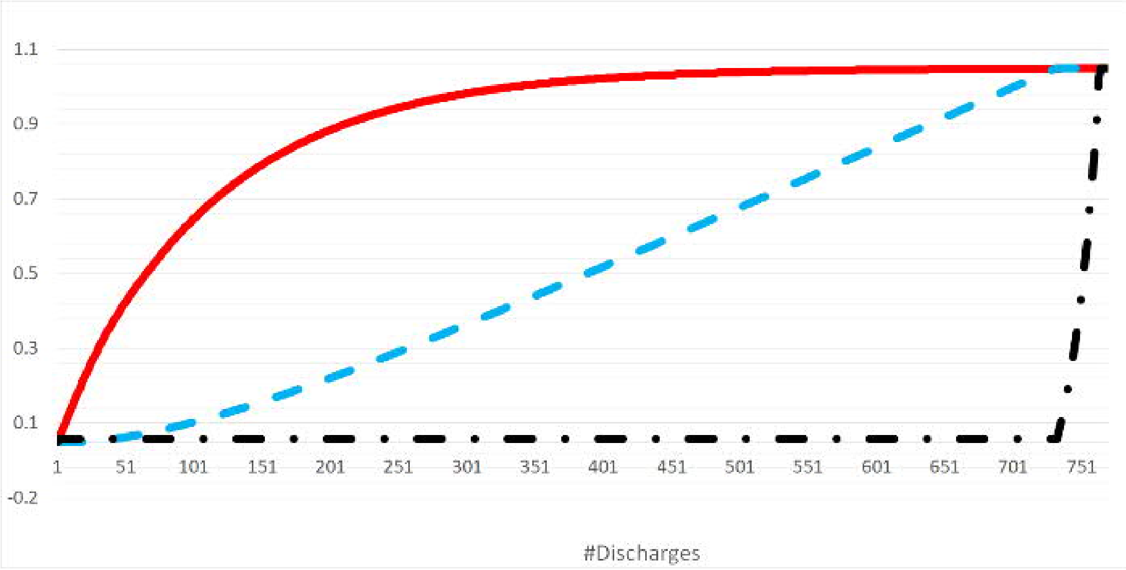

Fig. 5: Number of discharges needed to achieve the target state, i.e. the saturation of BB and ISS as well as a 50/50 D-T plasma under the assumption $\varphi_B = 0$. All quantities are normalized to the final value. BB is plotted in continuous red, ISS in dashed light blue, plasma in dashed-dotted black.

In the current assumption of $\varphi_B = 0$, no T is available to be re-injected in the plasma before both components are fully saturated. Thereafter, all T goes to the plasma, and the quantity of T burned per discharge grows rapidly up to 25.83 g, which corresponds to a full-power EU-DEMO discharge with 2 GW fusion power and 50/50 D-T. It is here indicated as "target state" a condition where i) BB is saturated, ii) ISS is saturated and iii) the quantity of T burned per discharge equals 25.83 g, or equivalently a 50%-50% D-T plasma is burned in the whole discharge. This represents in some sense the final goal of a D-D start-up process. Fig.5 shows the necessary number of DEMO discharges to reach the target state following Table 2 (it is here recalled that a DEMO discharge lasts 2 hours [11]).

Incidentally, an interesting speculation can be the following: in case D is employed as purge gas and the ISS is endowed with a system able to store the excess D in the incoming flows, there might be the possibility of establishing an enrichment equilibrium between the ISS itself and the gas flow heading to the GDCM. In that case, at least part of the T entering the ISS will be left for the plasma to be burned, increasing the efficiency of the overall T production. With simple calculations (not illustrated here), one can show that, in principle, a factor ~2 can be gained with respect to the previous case. At this stage however, this solution appears as highly speculative (e.g. it is unclear how efficiently the enrichment can in fact be equilibrated), and requires dedicated design modifications, which are currently not under consideration. Furthermore, other factors discussed in the next section still largely overcome this reduction in the number of discharges. This is however mentioned in this paper as an interesting research path, in case D-D start-up will be chosen in the future as a reasonable path by virtue of the possible scarcity of T.

The practical consequences on the plant operation of these results are here summarized. EU-DEMO aims at an availability of 30% in the full power operation phase [32]. However, in the initial phase, this value will be lower. Assuming an availability of 15%, the number of discharges per year (2 hours burn plus 10 minutes dwell [11]) amounts to ~607. This means the saturation of the components would require 1.15 years if ISS is saturated with 220 g. In principle, the low availability could be advantageous if the operation stops were employed to extract the T from the structures, in order to restart with a higher T content. The feasibility of this approach is however speculative at this stage, and especially at low concentrations, it might be very challenging, with high T losses associated. Also, an evaluation in terms of costs can be provided. Ignoring for the sake of simplicity other parasitic electrical loads, and assuming auxiliaries possess a 50 % wall-plug efficiency (which is a very optimistic, but potentially achievable value for DEMO [33]), a year at a load factor of 15% would require 315.6 GWh of electricity from the grid. With a price of 100 €/MWh (which is a reasonable approximation of the average EU industrial purchase price of electricity, with however some differences across the countries), this corresponds to 31.6 M€. At the rate calculated above, about 215 g of T are produced per year, leading to a cost of around 147 k€/g. This is about 6 times higher than the present commercial cost of 25,000 $/g [2]. It is once more stressed that the cost estimate is quite optimistic, as for example the interest rates on the construction loan have been ignored – although these charges, as well as the T decay discussed below, apply also when T is bought from an external source.

7. Discussion on the assumptions

The calculation illustrated relies on a number of highly optimistic assumptions, which are hereafter summarized.

The breeding blanket is optimized to work in a quite narrow window in terms of neutron flux, energy and power, corresponding to the full-power operation (i.e. 7.2e20 neutrons per second at 14.1 MeV with a deposited power in BB of ~2 GW). During D-D start-up, these conditions change drastically (about 0.6e19 neutrons per second at 2.45 MeV and the same amount at 14.1 MeV, following Eq.(9), and an estimated deposited power correspondingly lower). This has strong negative effects on the (i) breeding ratio – i.e. on the parameters b and ψ_{DD} and (ii) on the T residence time in the BB.

Concerning point (i), the value of ψ_{DD} has been set to 0.6 following the mentioned references. Regarding point (ii) instead, it has to be highlighted that the power released in the BB during a D-D plasma is at best ~20 times lower than the one used to design the BB assuming a D-T plasma – in case the 130 MW of auxiliary power were fully radiated onto the wall, and the power is able to reach somehow the blanket, which is an unrealistically favourable case (the D-D fusion power plays a negligible role). Therefore, the temperature field that will arise in the D-D start-up will be much lower and, moreover, the T release will be strongly reduced (usually the T residence time is directly proportional to $e^{1/T}$, see [34]). This is particularly true for the solid (pebble beds) BBs, as experimentally demonstrated in [35], where the T residence time increases from the order of minutes at 635ºC to hundred hours at ~300ºC for the orthosilicate material (see Fig.3 of [35]). A reduction of the coolant flow may allow increasing the temperature again to an optimal level. However, in view of the large difference of power w.r.t. the nominal case, the compatibility with the pumping system has to be verified. Also, the use of heaters – which have to be foreseen in order to verify the component at relevant temperature conditions before the nuclear phase is entered – can possibly be foreseen, although this will negatively impact the costs. These evaluations are left for future work.

Furthermore, it has been assumed that, from day one, full power D-D operation with 130 MW auxiliary power is feasible. This is clearly impossible, since of course DEMO has to foresee a commissioning phase where density, current and heating power are ramped up progressively. Whether this phase can already provide some T breeding is presently uncertain – since the commissioning phase of EU-DEMO has not yet been defined in detail.

Table 3: T permeation and retention in the FW from impinging ion flux [21].

	Wall concept	Permeation [mg T day $^{-1}$]	Retention [g T]
2 mm W-armor	WCLL	0	800 – 1350

However, this can easily reduce the D-D reaction rate of factor 10, or more – remember e.g. that the reaction rate scales as the square of the deuterium density, which in turn scales like the plasma current. Thus, at half current, one can expect a reaction rate of 4-8 times lower (since the plasma current also impacts the confinement time).

Finally, the tritium permeation rates from the plasma into the FW play a pivotal role in the T retention and, therefore, in the T needed to saturate the BB structures. Indeed, assuming gas- and ion-driven tritium permeation from plasma through the FW for a range of impinging tritium particle fluxes comprised between 10^{18} and 10^{20} T $m^{-2}s^{-1}$, with impinging particle energy of 500 eV (based on current SOLPS DEMO plasma models [36,37]), a tritium partial pressure of 1 Pa and material trap concentrations as a function of neutron-induced damage (from 10^{-4} to 0.4%), FW temperature, and hydrogen isotope content, the T permeation and retention in the FW is determined as shown in Table 3 (for more details on this argument see [22]). In the model presented here, this may reflect in $\varphi_P < 1$ until saturation is achieved, since the first wall absorbs T leaving the plasma. This means, more precisely, that every time the T concentration in the plasma is increased, the wall equilibrates to a higher T content, thus removing T from the fuel cycle. When the target, 50%-50% D-T plasma is reached, the wall will equilibrate at the values in Table 3. This ~1 kg of T is therefore progressively cumulated during the entire D-D start-up, always subtracting part of the available T which could have potentially been burned or employed to saturate the structures, thus slowing down the achievement of the final state – or, equivalently, slowing down the rate at which the T content in the plasma may increase. In view of the large quantities of T needed for saturation, this limitation is going to be present for an extremely large number of discharges, negatively impacting on T accumulation where it can be made available for fuelling.

Although more detailed calculations would be needed at this point, it is not too incautious to state that the real number of discharges to achieve saturation may be close to 10,000, with an impact in terms of time and costs scaling correspondingly.

Other, less impacting considerations are:

- It has been assumed in the model that the T produced in the BB and transported by the T-Carrier is completely removed by the TES and injected in the plasma through the fuel cycle. This assumption is optimistic, because the T extraction efficiency currently used as an operating point in the design of the T extraction units is about 82% (meaning that the remaining 18% cannot be immediately extracted, or, equivalently, the equilibrium is not immediately reached). Also, during the fuelling phases (i.e. during pellet formation and injection in the plasma) the efficiency is not unitary - at best ~90%.

- The decay of T has been ignored. This causes the loss of ~5.5% of the stored T per year.

- Finally, all delays in the fuel cycle have been ignored. This element alone would however only lead to a shift of a few discharges.

The analysis presented here is quite simple, and, in order to determine the real cost of a DD start-up, more sophisticated analyses would be required. However, the strategy employed consists of choosing quite optimistic assumptions every time an uncertainty was present. For this reason, it is clear that more detailed evaluations performed with more realistic assumptions or more powerful software can only provide a larger number of discharges to reach saturation, making the D-D start-up solution even less attractive. In other words, the conclusion of this work can be considered valid in spite of the simplicity of the approach.

8. Conclusions

The present piece of work analyses the possibility of accessing EU-DEMO operation by D-D start-up, thus eliminating the need of an external T source. What essentially limits the possibility of this convenient solution is the necessity to saturate a number of plant components, which cannot make the whole bred T available to fusion reactions before saturation is achieved. Incidentally, this study provides a clear indication to designers of systems like TES, where the maturity level is still low: technologies, processes and architectures should be chosen to minimize T retention. By means of a very simple model corroborated by some numerical simulations, it has been shown that the number of two hour discharges to achieve saturation - i.e. to start producing T effectively available for the plasma, allowing an efficient multiplication in the blanket – cannot be lower than about 700. This result is however only achievable under a number of very optimistic assumptions. With more realistic assumptions – listed in section 6 – a number of ~10,000 D-D discharges at full current and full power is estimated. This number constitutes a significant fraction of the total number of discharges DEMO is supposed to perform (about 30,000 [11] and references therein). Also, in terms of T cost, it has been shown that, even under the mentioned optimistic assumptions, the T produced via D-D start-up is not convenient to be pursued. For this reason, we conclude that, assuming the current configuration and technologies for the TES, D-D start-up does not deserve high priority as an option, and only under very unfavorable conditions for external supply, and with the help of dedicated design solutions (at the moment highly speculative), it may become attractive. It remains nevertheless conceivable that, if commercially available T becomes too costly or scarce, some form of D-D start-up requiring only a few hundred g of T to prime the processing plant of a DEMO-scale fusion reactor may be feasible. This point is left for future analyses.

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