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Abstract

It has long been recognised that the shortage of external tritium sources for fusion reactors using D-T, the most promising fusion fuel, requires all such fusion power plants (FPP) to breed their own tritium. It is also recognised that the initial start-up of a fusion reactor will require several kilograms of tritium within a scenario in which radioactive decay, ITER and subsequent demonstrator reactors are expected to have consumed most of the known tritium stockpile. To circumvent this tritium fuel shortage and ultimately achieve steady-state operation for a FPP, it is essential to first accumulate sufficient tritium to compensate for loss due to decay and significant retention in the materials in order to start a new FPP. In this work, we propose to accumulate tritium starting from D-D fusion reactions, since D exists naturally in water, and to gradually build up the D-T plasma targeted in fusion reactor designs. There are two likely D-D fusion reaction channels, 1) D+D→T+p, and 2) D+D→He3+n. The tritium can be generated via the reaction channel '1)' and the 2.4MeV neutrons from '2)' react with lithium-6 in the breeding blanket to produce more tritium to be fed back into plasma fuel. Quantitative evaluations are conducted for two blanket concepts to assess the feasibility and suitability of this approach to FPP reactors. The preliminary results suggest that initial operation in D-D with continual feedback into the plasma of the tritium produced enables a fusion reactor designed solely for D-T operation to start-up in an acceptably short time-scale without the need for any external tritium source.

1 Introduction

Due to the high reaction cross-section of the D-T fusion reaction, so far, the vast majority of fusion power plant studies have employed the D-T fuel cycle – the easiest way to reach ignition. Deuterium exists naturally and can be extracted from water; tritium is unstable because of its radioactive decay ($T_{1/2}$ ~ 12.3year) and occurs naturally only in trace amounts, formed principally by the interaction of cosmic radiation with oxygen and nitrogen atoms in the upper atmosphere. Tritium may be produced in civil nuclear fission reactors by the following five mechanisms: (a) fissioning of uranium, (b) neutron capture reactions with boron and lithium added to the reactor coolant, (c) neutron capture reactions with boron in control rods, (d) activation of deuterium in water and (e) high energy neutron capture reactions with structural materials [1]. In the civil tritium market, the principle source of tritium is fission reactors with heavy water cooling and moderation, which in total, world-wide, produce only a few kg per year from neutron capture by the deuterium in the heavy water. However this is a very small quantity of tritium generated as a by-product compared to the actual need of a fusion power plant (FPP). The annual tritium consumption of a fusion power plant operating at 1GW fusion power is ~55.6 kg per full power year (FPY) or ~152g per full power day (FPD). As there is generally a lack of external tritium sources, all FPPs must breed their own tritium needed for fuelling the D-T fusion plasma unless a purpose built facility supplies the tritium, meeting the fusion need. This does not address the issue of reactor start-up however, where the required inventory is estimated to be of order 10kg against an expected world inventory of order 7-32kg in 2050 [2] when there will be likely multiple DEMO fusion reactors and/or FPPs built in different nations. Thus, it would be essential to accumulate sufficient tritium to start a new FPP operating at the targeted fusion power, to compensate the loss due to the decay and any inventory retention in the plant. It is generally assumed that regulatory authorities would not sanction the stockpiling of any excess tritium that may be produced by precursor demonstration plants [3]. Thus, even though we expect a FPP will maintain a self-sufficient supply of tritium, the issue remains where to obtain sufficient tritium for the initial fuel to start a D-T fusion reactor under the condition that there is no external tritium supply.

There have been some previous studies addressing this issue. Reference [4] proposed increasing the percentage of deuterium into the D-T mixture to enhance the D-D reactions so as to reduce the need for tritium. The consequent plasma performance was studied for its consistency although a simple plasma energy balance was used for the study. There were some discussions [5,6,7] on commissioning of a D-T fusion reactor without external supply of tritium, based on a Japanese conceptual FPP design - CREST (Compact Reversed Shear Tokamak). In references [8,9,10], the investigations have focused on applying a small amount of tritium as a catalyst for other fusion reactions using natural fuels in order to avoid demanding a large quantity of external tritium. Reference [11] discussed the plasma physics feasibility of using D-He3 and D-D fusion fuels in appreciation of the significantly more demanding plasma conditions, including energy confinement time, density, temperature and beta, than those required by D-T fusion reactions and in preference of minimising neutron generation in fusion reactions. In this paper we investigate alternative ways to meet the tritium requirements for D-T fusion reactors by accumulating tritium from the D-D fusion operation and exploiting the capabilities and functions of the blanket in the fusion reactor. In Section 2, the mechanism is described and two different blanket concepts are introduced to extend this study in later sections; Section 3 discusses the results of preliminary simulations demonstrating the neutronics and plasma physics feasibility of the method; the economics is discussed in Section 4; the conclusion is given in the last Section 5.

2 Mechanism

In this work, we propose to accumulate tritium starting from D-D fusion reactions, and to gradually build up the D-T fusion plasma towards the targeted design operating point for a fusion reactor. It is assumed that the following two deuterium fusion reaction channels have essentially equal probabilities:

$a)\ D + D \rightarrow T(1.01MeV) + p(3.03MeV)$

$b)\ D + D \rightarrow {}^{3}_{2}He(0.82MeV) + n(2.45MeV)$

Although a D-D fusion reactor will possess a much smaller power density in comparison with D-T fusion at equal particle densities and ion temperatures below ~200keV, the production of tritium from the reaction channel ‘a)’ can be part of the fuel for D-T fusion reactions and the production of neutrons from the reaction channel ‘b)’ can induce the lithium compounded in the tritium breeding blanket to generate more tritium (as shown in Figure 1).The neutron energy is not very high relative to the energy threshold of the $^{7}Li(n,n')T$ reaction and the neutron multiplication reactions, mainly Pb(n, 2n) and Be(n, 2n), thus, the tritium will be produced mainly via the $^{6}Li(n, T)^{4}He$ reaction which has a high cross-section for thermal neutrons. The tritium produced in the blanket is directly returned into the plasma, allowing for a suitable transfer period, thus increasing the tritium concentration in the plasma.

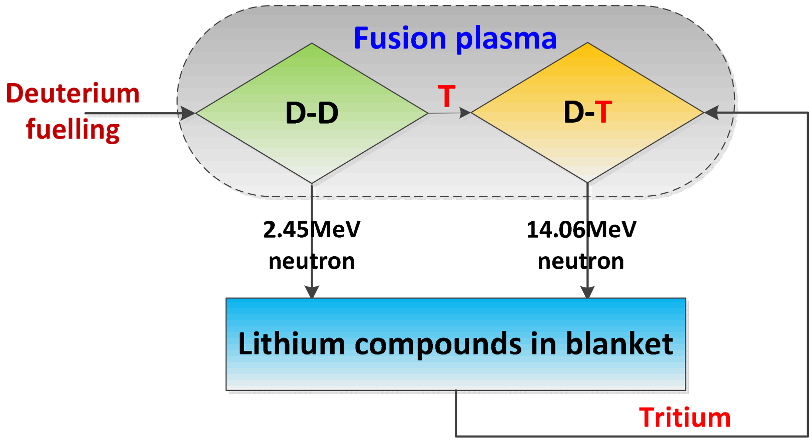


Figure 1.Mechanism of fusion plasma evolvement from D-D to D-T.

Meanwhile, there is an approximately 50% probability that D-D fusion reactions will produce tritium that will be directly diffused in the plasma mixture and result in D-T fusion reactions due to the higher reaction cross-section. The high energy neutrons from the D-T fusion reactions will further react with ^{6}Li or ^{7}Li in the blanket to produce more tritium with the benefit of more usable neutrons via the multiplication reactions.

The blanket tritium breeding capability is design dependent and evolves with time due to the depletion of the lithium isotopes and the blanket material transmutation, primarily depending on the choice of breeding materials and neutron multipliers. For the purposes of this study we adopted a basis configuration of the EU DEMO plant [12] validated by PROCESS code [13] as part of the EUROfusion Power Plant Physics & Technology (PPPT) working group on DEMO.To assess the practicality of accumulating sufficient tritium from D-D fusion operation to progress towards D-T fusion operation, two different breeding blanket designs, HCLL (Helium Cooled Lithium Lead) and HCPB (Helium Cooled Pebble Bed), which have been adopted as the EU test blanket modules (TBMs) for ITER [14], were employed. The evolution of tritium production through the different operational phases, from the D-D to the D-T phase, was modelled assuming the blanket is the design optimised for the full power D-T operation of the machine throughout.

3 Results and discussions

3.1 Tritium breeding capability

The evaluation of the tritium breeding in the selected reactors is carried out using the neutronics tool MCNP5 [15] and the IAEA fusion nuclear data FENDL2.1 [16]. In the HCLL reactor, the materials composition in the breeding blanket is homogenised with the tritium breeding material being LiPb, where the Lead acts as the neutron multiplier in a high-energy neutron environment and Eurofer as the structural material. In order to achieve a self-sufficient tritium fuel cycle for the D-T operation, the ^{6}Li is enriched to 90%. In the HCPB reactor, Li_4SiO_4 is the tritium breeder and beryllium is employed as the neutron multiplier to enhance the effectiveness of the neutrons; the structural material is again Eurofer. Each D-T fusion reaction consumes one triton and produces one high energy neutron (14.06MeV) that travels through the breeding blanket to generate both tritium and additional neutrons that also produce tritium. The ratio of generated tritium to consumed tritium is referred to as the tritium breeding ratio (TBR). From D-D fusion reactions, neutrons at 2.45MeV are generated that react with the lithium in the breeding blanket where additional tritium may be produced. The tritium production rate (TPR), the amount of tritium generated from each D-D fusion neutron, is used to measure the tritium breeding capability in a reactor which operates in the D-D fusion plasma phase. The TBR at the D-T operation and the TPR at the D-D operation for HCLL and HCPB reactors are presented in Figure 2. The variation of Li-6 enrichment is evaluated to explore the optional potential to produce tritium.

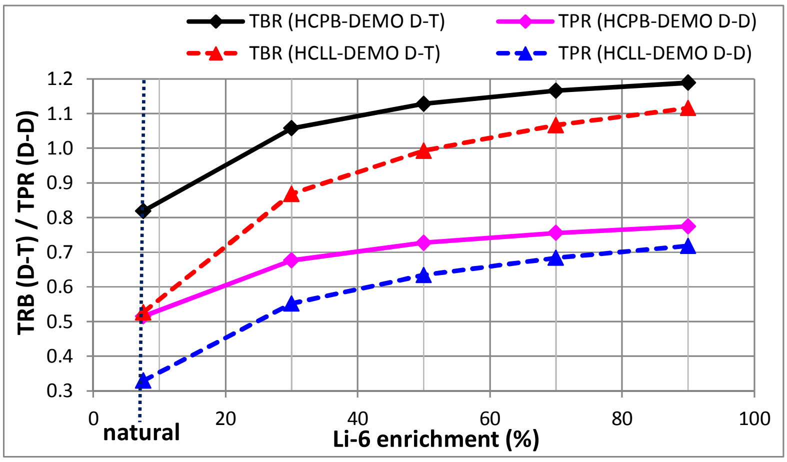


Figure 2.Tritium production ratio/rate in HCLL and HCPB reactors.

3.2 Feasibility of the D-D plasma operation

It is recognised that maintaining pure D-D fusion operation is much more difficult and less efficient in producing energy than D-T fusion operation, and hence the latter is commonly pursued for ultimate commercial realisation of fusion energy. In order to determine whether operating the selected reference basis reactor in D-D is feasible, a number of comprehensive predictive plasma physics simulations using the JINTRAC [17] suite, combining JETTO [18] and SANCO [19], have been performed. JINTRAC is the main tool used for integrated modelling of JET pulses; JETTO is a 1.5D fluid plasma code for simulating core plasma behaviour while SANCO controls plasma impurities and radiation. The results of the simulations are summarised here and will be presented in more detail in a subsequent paper.

The simulations have focussed on the stationary phases of the scenario to confirm the existence of a stable working point in a time dependent evolution and have been based on work in a previous DEMO study [20]. The amount of additional heating required for a stable H-mode D-D plasma has been calculated through a parametric study as this is of particular importance to the feasibility of this approach. If too much additional heating is required then the method would be uneconomic. The particle and heat transport coefficients were specified analytically and varied until the resulting temperature pedestal matched results from TGLF [21] simulations. It is assumed that additional heating of the plasma has been provided solely by a 1MeV NBI system; this has been modelled using the PENCIL module included in JINTRAC. When using PENCIL, a proposed beam geometry for an EU DEMO design [22] was used, no optimisation of the details of the heating system for the q-profile of the plasma was conducted for this study.

A tokamak, characterised by relatively undemanding physics and technology assumptions, as represented by the recent EU DEMO design [12,20] working point produced by the PROCESS code was used as a starting point with the tritium removed from the simulation, i.e. pure D-D plasma. Simulations were performed for a conservative H-factor of 1.0 and a Z-effective of 2.0. Impurities present in the plasma were Argon, Helium (produced from fusion reactions self-consistently) and Tungsten.

For a tokamak with the DEMO parameters the ITER physics basis scaling[23] gives an L-H threshold power of ~155MW. While a stable D-D plasma temperature was achieved at 135MW the results presented here were performed with 170MW of NBI power so that the threshold power was exceeded. Figure **3** shows the time evolution of the electron temperature and density while Figure 4 shows the profiles of the temperature, density and transport coefficients. The simulation reached a steady state with $\langle n_e\rangle=9.8\times10^{19}$ m^{-3}, $\langle T_e\rangle=7.75$ keV, $\langle T_i\rangle=8.1$ keV.

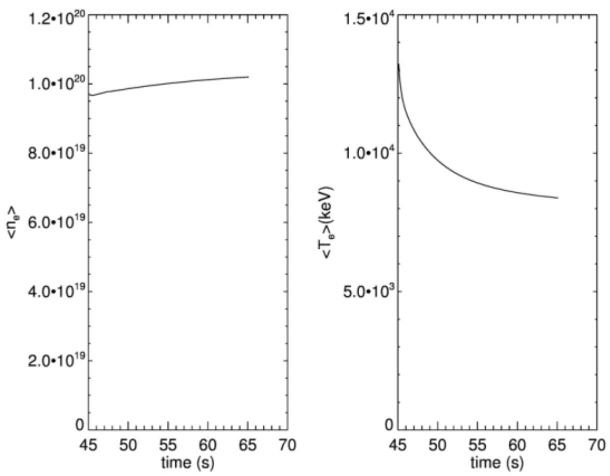


Figure 3.Density and temperatures evolution for the DEMO reactor.

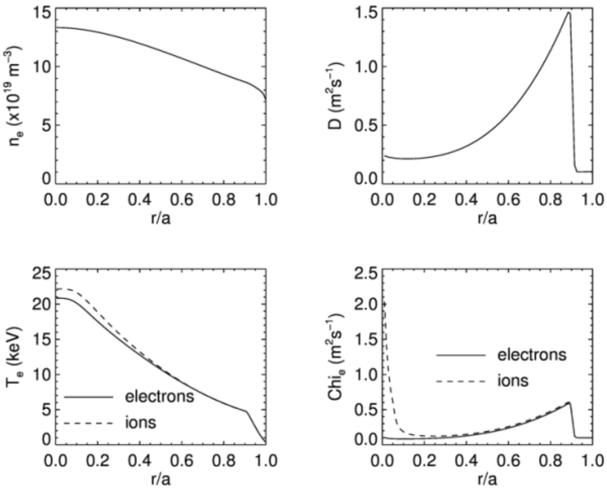


Figure 4. Density, temperature and transport coefficient profiles for the DEMO reactor

The total plasma thermal energy is ~566 MJ and the fusion power is ~ 9.3MW when stability is reached. The radiated power can be summarised as follows: 20 MW are radiated through bremsstrahlung, another 20MW are radiated through synchrotron radiation and 45 MW are radiated through line radiation from the impurities. In this simulation the synchrotron radiation was calculated using the formula provided by Trubnikov in [24]. Thermal and beam-target neutron rates are calculated in the simulation giving a total rate of $6.7x10^{18}$ s^{-1}. For full D-D simulations the beam-target neutrons make up a significant fraction of the total (26%), but as tritium is introduced then this fraction becomes negligible due to the high thermal rates. It can be observed that the plasma physics behaviour in such conditions does not match the theoretically optimistic assumption that the neutron intensities are also two orders of magnitude lower for the D-D operation in the same tokamak machine because the cross section of D-T fusion reaction is about 100 times higher than the D-D fusion reaction at the same temperature. The plasma operation simulations show that there is only partial secondary tritium further fusing with deuterium after it is generated from the D-D reaction channel ‘a)’; the ratio of tritium consumed in D-T reactions to the deuterium consumed in D-D reactions is ~15%. However, the D-D reaction rates are higher in pure D-D fuelling plasma than in a 50-50% D-T plasma.

3.3 Proposal for a dynamic process to accumulate tritium for D-T fusion

Although preliminary simulations show that the D-D plasma fusion operation is achievable in the basis reactor originally designed for operating at D-T, and it is observed that the D-D reaction rate is higher in the pure D-D fusion operation than in a 50-50% D-T plasma, it seems somewhat too slow to accumulate sufficient tritium to enable the transition from pure deuterium fuel to the targeted full-power D-T fusion operation especially when the fuel efficiency including fuel burn-up fraction in the fuel cycle is taken into account. In order to shorten the tritium building-up time prior to the full D-T fusion power operation, one possible solution is to accumulate the tritium in a dynamic process, e.g. by adding some small fraction of tritium (developing from zero or near-zero to 50% as the tritium inventory increases) in the deuterium-dominant plasma to enhance the neutron yield and the corresponding fusion power. A series of JINTRAC simulations have been done for plasmas with increasing fractions of tritium to correctly capture the corresponding plasma physics performance and relevant fusion neutron generation rate. As shown in Figure 5, by adding small amount of tritium, the neutrons generated from D-T fusion reactions increase significantly due to the high reaction cross-section in comparison with the D-D fusion reactions under the same operating conditions. This indicates a feasibility to take a few steps to gradually upgrade the fusion operation performance towards the targeted basis reactor (operating at D-T with 50% D and 50% T in the plasma fuel) where the initial step is still pure D-D fusion operation without external tritium. Indeed, in this process, surplus tritium will be essential for accumulating extra tritium fuel as well as maintaining the fuel balance for the tritium consumption in the fusion plasma despite the tritium being a minority in the fuel to react with deuterium to produce more power than pure D-D operation.

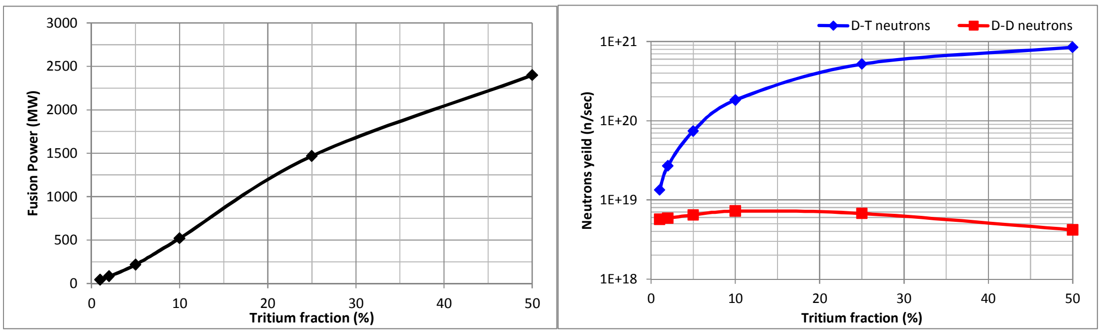


Figure 5.Fusion power and neutron yields as function of tritium fraction (up to 50%) in plasma fuel

The D-D and D-T neutron rates determined by the JINTRAC modelling as functions of tritium fraction were used to calculate the tritium production rate from TBR and TPR and the tritium accumulation as a function of time.

In this study, several factors have been applied to accommodate relevant technology and engineering requirements for near-term and more advanced fusion reactors designs.

1) The burn-up fraction of tritium in the plasma is assumed to be 5% and 20%. The impact of low tritium burn up fraction on the quantity of tritium required for start-up in full power D-T operation has already been addressed. The two values selected in this study are derived from discussions in [25] and [3], in which the results suggest low tritium burn-up fraction (<5%)

may lead to difficulties to sustain the tritium self-sufficiency in future D-T fusion reactors and higher fraction (>20%) was selected for advanced fusion reactors, e.g. ARIES-AT. For a 2.4GW D-T fusion reactor, the daily tritium inventory required for start-up is ~7.3kg for the 5% burn-up fraction and ~1.8kg for the 20% burn-up fraction.

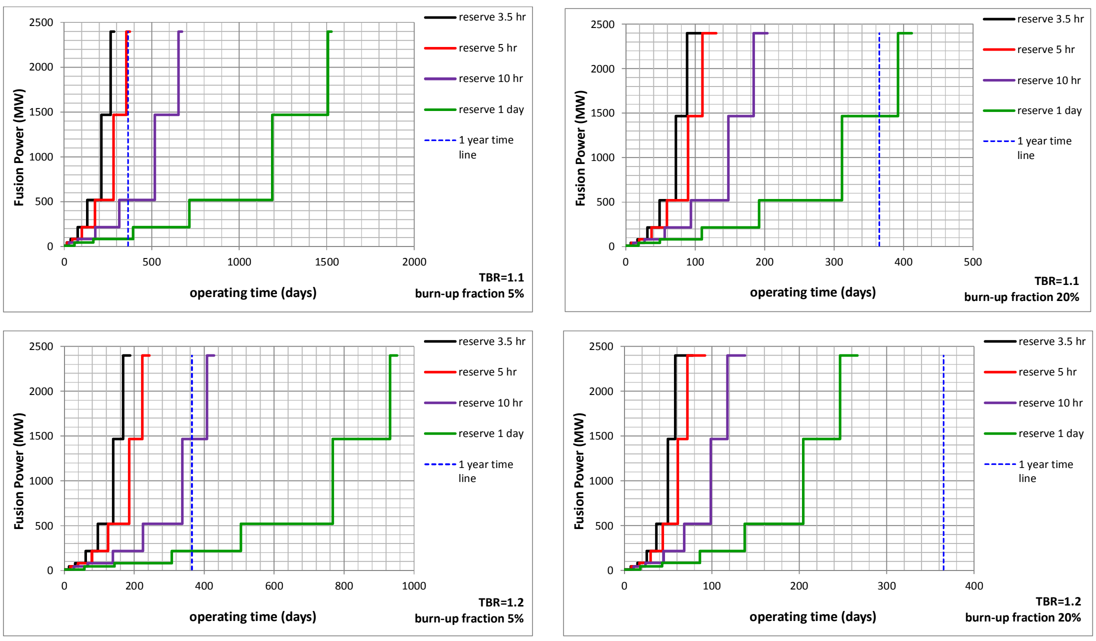
2) The typical regeneration time for the cryopump is about 30 minutes [26]. The relevant inventory is calculated to be consistent with the tritium fuel needed for corresponding plasma operations, e.g. the tritium inventory on the cryopump is ~152g for a 2.4GW D-T fusion reactor with 5% fractional burn-up.
3) The tritium residence time in the purification system is about 3 hours [26]. The inventory involved is determined by the tritium generated in the blanket system at different stages, e.g. the tritium inventory in the purification system is ~50g for a 2.4GW D-T fusion reactor if the tritium breeding ratio is 1.1.

   Considering '2)' and '3)', it is assumed the shortest time needed to feed the tritium generated in the blanket to the plasma as fuel is 3.5 hours. Therefore, there has to be sufficient tritium held in storage to maintain plasma operation for at least 3.5 hours this is referred to as the "reserve time". To provide some margin, the analysis uses 5 hours, 10 hours and 1 day as optional 'reserve time'.
4) The tritium extraction efficiency from the blanket is assumed to be 90% [27] for all operation stages.
5) The tritium is radioactive and its half-life time is ~12.33 years. The decay rate is taken into account according to the duration from its generation in the breeding blanket and its consumption in the fusion plasma.
6) The tritium breeding capability is dependent on the breeding blanket design. Due to the diversities of blanket designs in the past and on-going, three TBR options, i.e. 1.1, 1.2 and 1.3, are employed for a relatively generic discussion to avoid being locked onto a specific blanket design. This selection also covers a reasonable range for tritium breeding capability predicted for demonstration reactors and future power plants [3, 28].

Preliminary results are summarised in Figure 6 where each step (of the level of fusion power) in the curves corresponds to a tritium concentration used in the JINTRAC modelling – 1%, 2%, 5%, 10%, 25% and 50%. Following observations can be made:

- The machine has to operate at pure D-D for 1-2 weeks to accumulate sufficient tritium to start operation with 1% tritium in the plasma. The length of time taken to reach this point can vary with the tritium producing rate in the blanket induced by the D-D fusion neutrons and the burn-up fraction of the tritium in the fusion plasma.

- TBR=1.1, starting from pure D-D operation, it takes ~1 year to accumulate 1FPD of tritium supply for its full-power (2.4GW) operation if the burn-up fraction can be 20% while it would take ~4 years to achieve the same goal if the burn-up fraction is only 5%. If the fuel cycle time can be shorten to 5 hours and the fuelling process is continuous for all transition steps, it will take ~3.7 months for the case with 20% burn-up fraction and ~1 year for the case with 5% burn-up fraction to reach a full-power D-T operation, respectively.
- TBR=1.2, after the first step at a pure D-D operation, it would take from half year to ~2.5 years to reserve 1-day tritium supply for the D-T operation at 2.4GW fusion power in the range of burn-up fraction from 5% to 20%. If the burn-up fraction is no higher than 5%, it will still take more than 1 year to upgrade the plasma operation from pure D-D to full-power D-T at 2.4 GW in order to reserve enough tritium for 10-hour operation without expecting immediate self-feeding of the tritium fuel.
- TBR=1.3, because the tritium breeding ratio is high, if the machine can run at advanced burn-up fraction, i.e. 20%, the building up time is relatively short, ~half year to reserve enough tritium for 1-day need by the D-T operation at full power. If one considers the option of reserving tritium for 10-hour need by the targeted D-T fusion performance, it would only take ~3 months to achieve this goal.



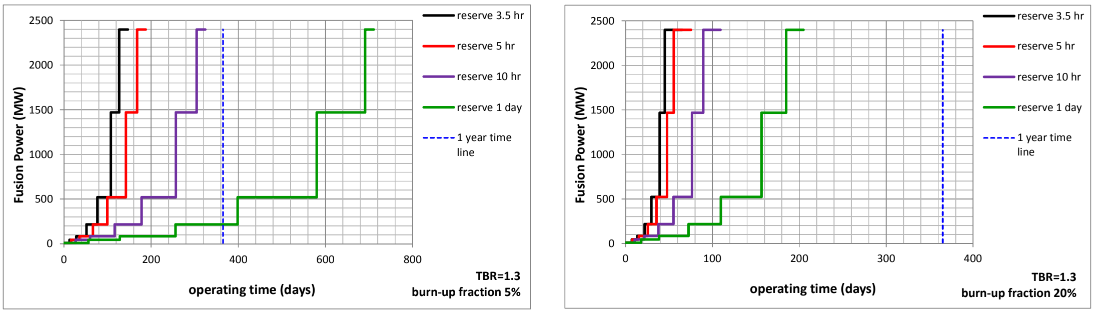


Figure 6.Fusion power building up from D-D to D-T at targeted power (2.4GW)

4 Economic considerations

The results of Section 3.4 show a wide variation in the time taken to achieve full power D-T operation from a D-D start-up depending upon the selected characteristics of the tritium cycle. The economics of this approach depends upon the assumed length of time of a commissioning phase for a FPP designed to start at full D-T operation, the cost of additional Heating and Current Drive (HCD) over the D-D to D-T phase and the equivalent cost of providing an initial tritium fill. Assuming a one year commissioning phase for any FPP, which will most likely start in deuterium (ignoring any hydrogen phase common to both approaches), the most optimistic scenario in Section 3.4 would fit within that timescale. As is shown in Section 3.4 for high burn rate advanced plasmas, the tritium accumulation can be completed in less than one year. Assuming that the FPP is a steady state device requiring continuous current drive power, there are no additional operating costs incurred in operating the HCD systems during the tritium accumulation phase than under normal generating conditions. Thus the additional costs relate only to the length of time taken to accumulate sufficient tritium beyond the normal commissioning phase and the lost income from selling the electricity generated in that time.

It is reasonable to assume that in a FPP the issue of thermal efficiency, related to operating temperature and coolant choice, will have been resolved, in which case the recirculating power will be predominantly required to drive the HCD systems. Assuming a value of 50% for the wall plug efficiency of the NBI system (slightly less than the 60% usually used as a benchmark) the approximate cost of the tritium accumulation phase can be estimated. The cost includes the operating cost of the electricity to power the HCD system (assumed to be purchased at the US industrial rate of US$0.07/kWhr[29]) and the lost income from the delay in selling electricity to the grid (assumed to be at US$0.03/kWhr [30]).

The cost of tritium generally used by authors is US$30M/kg obtained from CANDU reactors [30], so a single operational charge of ~7.5kg would cost US$230M (2.5GW fusion reactor with 5%

burn up). For operational purposes additional tritium would be needed to offset the reserve time, during which the un-burnt tritium from the initial charge is being processed, and the tritium losses (assumed to be 10%). Thus the mass of tritium required to start a FPP depends on the fusion power, the burn up rate, losses and reserve time during which the un-burned tritium is not available to the plasma and is given by:

$$M_T = \frac{1}{F_R}\frac{0.152}{f_b}P_{fus}[GW]\left(1+\frac{t[h]}{24}\right)$$

where F_R is the tritium recovery fraction (90%), f_b is the burn up fraction, P_{fus} is the fusion power in GW and t is the reserve time in hours.

The systemic trends are shown in Figure 7 for 5% and 20% tritium burn up fractions, assuming that the tritium generating phase occurs separately after commissioning. The additional cost is shown to be a strong function of the reserve time and the tritium burn up fraction.

It can be seen that in this case and for the cost of tritium at US$30M/kg, the economic case for accumulating tritium in the method described here can only be supported for low burn up fractions and high TBR. However, predictions of the likely future cost of tritium range up to US$200M/kg [31], at which value the process becomes is easily economically viable for the scenario with TBR of 1.3, 20% burn up fraction and 24 hour reserve time. (The actual price of tritium for break even in this case is ~US$125M/kg). If the cost of the tritium accumulation phase can be partially offset against commissioning, the economics becomes more favourable, as shown in Figure 8 where six months of the commissioning period is used in the tritium accumulation phase. In this case only the scenario with 5% burn up fraction and TBR of 1.1 fails to become economic at any reserve time. Obviously there are many assumptions made in performing this analysis regarding TBR, tritium retention, costs of electricity and tritium, etc. but all are consistent with the actual requirements of a fusion power plant. Furthermore, the economic arguments above ignore the possibility that sufficient tritium may not be available, in which case the process may be essential.

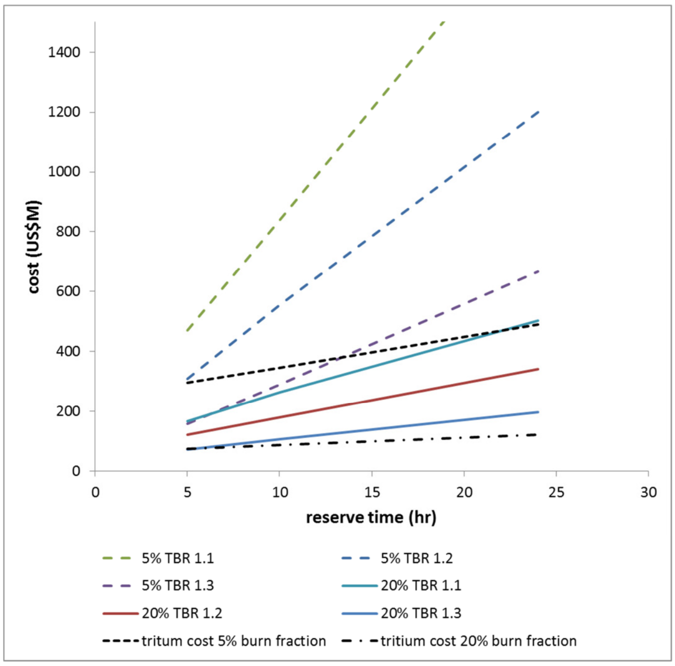


Figure 7. Estimation of additional cost of tritium accumulation compared with cost of tritium required to start D-T operation.

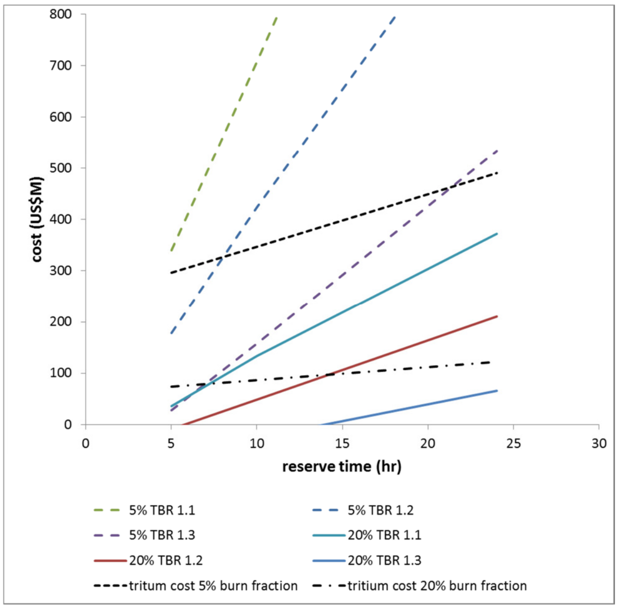


Figure 8. Estimation of additional cost of tritium accumulation compared with cost of tritium required to start D-T operation for tritium accumulation off set against 6 months of the commissioning time.

5 Conclusion and discussion

Although the D-T fusion is the most promising fusion reaction and is therefore widely promoted by the fusion community, the necessary amounts of tritium do not exist naturally to provide a sustainable fuel supply. Therefore, tritium self-sufficiency once the fusion reactor is operating stably has long been proposed. However, the initial fuel loading to start up any large-scale D-T fusion reactor has remained a significant issue.

This work aims to pursue an alternative solution to accumulate sufficient tritium in order to start a D-T fusion reactor. We propose to start a fusion reactor from the D-D fusion reactions. Tritium will be produced through 2 pathways. On one hand, the 2.45MeV neutrons produced by one of the two (roughly equal rate) D-D reactions are able to react with the Li-6 of the breeder in the blanket to produce tritium to be accumulated in preparation for the D-T fusion fuel requirements for the next step; on the other hand, there will be some tritium generated inside the plasma mixture through the other D-D fusion reaction channel which, either immediately or after fuel separation and recycling, will fuse with the deuterium to produce 14.06MeV neutrons to enhance the tritium production in the breeding blanket. Because there is no external tritium required by this process, all tritium generated in the blanket can be accumulated for the fuel supply to support the further D-T fusion operation.

To model this process realistically, several factors have been taken into account, 1) burn-up fraction for tritium in plasma, assumed to be 5% and 20%; 2) a few different options for the tritium reserve time are studied to provide a range from 3.5 hours to 1 day; 3) the tritium extraction efficiency is assumed to be 90% in the blanket; 4) the tritium loss due to its decay is included in the study.

Given that the tritium breeding capability is dependent on the breeding blanket design, three TBR options, i.e. 1.1, 1.2 and 1.3, have been employed. We have analysed a new methodology, whereby tritium is accumulated for only short periods and then injected into the plasma fuelling mixture, gradually building up the D-T fraction of the fusion reactions in order to reach full power D-T operation more efficiently. Plasma simulations at periodic tritium concentrations were conducted to obtain self-consistent fusion parameters for the breeding calculations.

The results suggest that D-D operation may be required for ~1 week to 2 months to accumulate sufficient start-up tritium to launch a tritium seeded D-D dominant plasma operation. The fusion power increases from 9.3MW (~2.53MW is from secondary D-T fusion reactions), for pure D-D fusion, to 44MW (~37.5MW is from D-T fusion reactions), for the mixed fuel fusion operation with 1% tritium injected into the plasma core.

As soon as the operation starts to consume tritium in fusion reactions, the tritium accumulation will be largely dependent to the surplus tritium bred in the breeding blanket. If we assume the tritium fractional burn-up will be 20% for a FPP, accumulation of sufficient tritium to operate at 50:50% D-T corresponding to designed TBR values for D-T operation of the blanket of 1.1, 1.2

and 1.3, it takes respectively ~13 months, ~8 months and ~6 months to reserve the tritium enough for 1-day need to achieve operating the designed D-T at 2.4 GW fusion power including less than 20 days D-D operation as the first stage. The accumulation period is a strong function of the tritium reserve time, if this is shorter than 5 hours, even at TBR ~1.1, it may take less than 3.7 months to upgrade the plasma performance to the targeted 2.4 GW D-T fusion operation because the reserve time can be as short as 5 hours for the D-T operation to start up.

There are some positive effects that might accelerate the T production: shortening the recycling time as far as realistically possible, arranging for some fraction of the He^3 to be converted into tritium, reducing the core radiative power losses by optimising the density and temperature (while preserving adequate current drive efficiency), and reducing fuel dilution by radiation-inducing impurities in the low power phases when such losses are not necessary to protect the divertor. A more complete accounting for the many subsidiary fusion reactions that can take place amongst the fuel species and the daughter products that remain in the plasma is also required for a full analysis and may alter the overall findings to some extent. In any case, it seems likely that the nuclear safety regulator would not favour the immediate start-up at full power of a first-of-a-kind high power nuclear fusion reactor and accordingly a time-scale of some months to develop the output progressively towards the full power design point may well turn out to be reasonable from a safety viewpoint.

The auxiliary heating required for performing D-D operation needs closer assessment although preliminary studies on the D-D plasma physics feasibility suggest that this may not be a big issue, due to the strong negative power dependence of ITER confinement time scaling and the potential to reduce considerably the core radiative losses of the plasma. The preliminary plasma modelling shows the heating required for D-D operation to be ~170MW. However, more detailed analysis using a full transport model is desirable. Other effects such as MHD stability and fuel retention in the wall could also be included in a further analysis. The optimisation of the D-D and D-T reactions in the plasma will be necessary by adjusting the fuel fraction of D and T. It should be noted that this alternative approach is very design dependant and this requires the integration of the blanket design to consider the desirable D-D operation phase. These require closer study for further realistic analyses. Moreover, the practicalities of the time-dependence of the extraction and recycling generation of the tritium, especially at the beginning when the amounts of tritium are very small, requires more detailed further study.

This work presents preliminary evaluations of the feasibility of starting to accumulate the tritium from operating at the D-D phase and using a tritium feedback system to evolve to full D-T operation as a power generating fusion reactor. The neutron irradiation damage to the structural material (Eurofer is employed in this study) is around 100 times lower during the D-D phase compared to the full D-T phase in a same reactor. A rudimentary economic analysis of this approach, based on 2013 electricity prices, is presented based on a tritium cost of US$30M/kg which shows viability only for high TBR, low burn-up fraction, assuming that the tritium accumulation phase is not part of the commissioning phase. The economics become more favourable if the accumulation phase can be combined with part of the commissioning phase of the reactor. It is worth noting that future tritium price is in the range of $100,000~$200,000 per gram and this would have a strong impact on the economics of this approach, assuming that an adequate supply of tritium was available in the future.

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