Fuel Source Isotopic Tailoring
and Its
Impact on ITER Design, Operation and Safety

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Abstract

Isotopic tailoring of the deuterium and tritium density profiles in fusion reactors can lead to reduced tritium inventory in plasma facing components and, therefore, improved safety considerations. The isotopic tailoring concept consists of utilizing a tritium-rich pellet source for core fueling and a deuterium-rich gas source for edge fueling. Because of the improved particle confinement associated with the deeper tritium core fueling component, comparable core densities of deuterium and tritium can be maintained even when the edge deuterium fuel source is much larger than the core tritium fuel source. The fuel composition of the edge and scrape-off plasmas as well as the isotope fractions in plasma facing components reflect the total throughput of all makeup fuel and are therefore deuterium-rich. This innovative fueling concept results in about a factor of two reduction in tritium inventory of the plasma facing components. The higher tritium burn fraction allows a significant reduction in tritium gas flows into and out of the vacuum vessel and, for fusion reactors, implies lower required tritium breeding ratios.

I. Introduction

The purpose of the International Thermonuclear Experimental Reactor (ITER) project is to develop a fusion engineering test facility which will demonstrate the scientific and technological feasibility of fusion power. One of the key feasibility issues is the safety and environmental impact of this technology. Given the near-term size and projected power cost penalties of fusion technology (due to low engineering power density) relative to light water fission reactors (1), it is imperative that the fusion power option be shown to have quantitative safety and environmental advantages over current and improved fission power plants. It is generally recognized that the worst-case fusion accident is less severe in terms of public impact than the worst-case fission reactor accident (2). For more probable design basis accidents and events, the impact to on-site workers and the public can be comparable for fusion and fission power plants (2) and could result in similar licensing procedures and engineered safety systems. There is, therefore, much to be gained by reduction of intrinsic source terms such as tritium and volatile activated dust inventories. This paper will investigate the potential to reduce tritium inventories and throughputs by innovative use of baseline ITER systems including the plasma fueling, fuel processing and isotope separation systems to control the hydrogen isotope content of separate fueling sources. This process is called isotopic fuel tailoring.

Isotopic tailoring of the deuterium and tritium density profiles in fusion reactors can lead to reduced tritium inventory in plasma facing components and, therefore, improved safety considerations. The isotopic tailoring concept consists of utilizing a tritium-rich pellet source for core fueling and a deuterium-rich gas source for edge fueling (3). Because of the improved particle

confinement associated with the deeper tritium core fueling component, comparable core densities of deuterium and tritium can be maintained even when the edge deuterium fuel source is much larger than the core tritium fuel source. The fuel composition of the edge and scrape-off plasmas reflect the total throughput of all makeup fuel and would therefore be deuterium-rich. The composition of the fuel particles retained in the plasma facing components reflects the composition of the edge plasma and would also be deuterium-rich. We explore the ability to maintain isotopically tailored deuterium and tritium profiles with pellet injection from centrifugal and pneumatic pellet injectors supplemented by deuterium gas injection, while keeping within the limitations of the ITER Engineering Design Activity (EDA) pumping constraint of 1 bar-l/s total throughput (~5×10^{22} particles/s). Sensitivity of the results to particle transport models and fuel composition are evaluated. The hardware needs, fractional tritium inventory in the plasma facing components, ex-vessel fuel processing requirements and safety benefits are also discussed.

II. Plasma Particle Transport and Isotopic Tailoring

The WHIST 1-1/2-D, time-dependent transport code (4) is used to examine the variation in the deuterium and tritium core and edge densities in response to various combinations of makeup gas and pellet fueling sources. A brief overview of the WHIST code and its application to pellet fueling of reactor-scale devices is given in Ref. 4. The radial transport model used in this study is assumed to be neoclassical with an anomalous contribution as described in Ref. 4. The resulting ITER radial profiles of deuterium, tritium and helium diffusivity and convective (pinch) velocity indicate good agreement with recent measurement of these quantities in TFTR DT plasmas (5). In particular, the particle diffusivities of

D, T and He are comparable to the deuterium thermal conductivity and have a value of ~0.5-2 m^2/s over most of the plasma minor radius; the convective velocities are small except near the plasma edge where there is an inward pinch. The ablation model used in WHIST is the standard neutral gas shielding model with a Maxwellian electron energy distribution which has been benchmarked to an extensive JET pellet ablation database. A pellet fueling efficiency of 100% is assumed. The pellet fueling component, consisting of 90% tritium/10% deuterium is varied between 0.27 and 0.37 bar-l/s, while the edge deuterium gas fueling component is varied from 0.75 to 0.65 bar-l/s under a constant fusion power of ~1500 MW. Since the fueling sources and transport are specified, one of the variable outputs is the D-T mix in the plasma core. Operation with a core fuel mix of ~60-40 D-T does not significantly increase operating density but provides an immediate reduction in the relative tritium fraction in both the core and edge. Results from WHIST code runs are given in Table 1 with the caption describing the parameters in each column. Tritium burn fraction given in the last column is the ratio of the amount of tritium burned (which is directly related to the fusion power) to the amount of tritium provided by the fueling system. In cases 469 to 471 (shown in Fig. 1), the 90-10 D-T pellet is sized for an equivalent spherical radius of 0.425 cm (about $2x10^{22}$ particles, about a 10% density perturbation) and has a speed of 1.5 km/s; this performance is typical of existing single-stage pneumatic pellet injectors (see Section III).

For case 469, the deuterium gas fueling is 0.65 bar-l/s. This results in a 63% tritium fraction on axis and a 34% tritium fraction at the edge. This tritium fraction on axis is too large, so in cases 470 and 471 the deuterium edge fueling is increased to 0.7 and 0.75 bar-l/s, respectively. Given the ITER constraint of 1.02 bar-l/s total fueling, this results in a corresponding reduction in the pellet fueling

source. It can be seen in Table 1 and also in Fig. 1 that the tritium fraction on axis and at the edge decreases at a constant fusion power of 1500 MW and fueling/pumping throughputs of ~1 bar-l/s as the edge deuterium gas source is increased.

Next, for cases 472-474, a particle pinch term is added consisting of a neoclassical Ware pinch term and an anomalous pinch term given by:

$$V = -2\left(\frac{r}{a}\right)^6 \frac{D}{a} \qquad \text{m/s}$$

where D is the anomalous particle diffusivity. This results in more effective core fueling by the edge deuterium source and reduces the core tritium fraction relative to case 471. For cases 472 to 474, the edge fueling by deuterium gas was held constant at 0.75 bar-l/s and the pellet fueling source was constant at 0.27 bar-l/s. For case 473, the pellet size was increased to an equivalent spherical radius of 0.535 cm (a 15% density perturbation) and the speed increased to 3 km/s. This performance is typical of two-stage pneumatic injectors without sabots (3) or, alternatively, may be viewed as a single-stage pneumatic injector technology with enhanced shielding mechanisms such as magnetic shielding (6). For case 474, the pellet size was decreased to an equivalent spherical radius of 0.3 cm (a 1% density perturbation) and the speed decreased to 1 km/s. This parameter range is typical of centrifuge injectors. It can be seen from Table 1 that the penetration relative to the minor radius spanned the range from 25 to 37% for cases 472 to 474. The effect of penetration on the edge and core tritium concentrations is shown in Fig. 2. It can be seen that edge tritium concentration is nearly constant with increasing pellet penetration while the core tritium concentration increases from 30% to 43%. This is expected and allows

the flexibility to decrease the tritium-rich pellet fueling fraction as the pellet penetration increases.

A key conclusion from the above results is that it is possible to maintain a tritium concentration gradient in going from the core to the edge of a D-T plasma. It is possible to maintain core tritium concentrations in the 40 to 50% range while maintaining edge tritium concentrations in the 20 to 30% range. The edge D-T ratios are typical of the vacuum vessel exhaust stream isotopic fractions and have a direct impact on lower tritium wall inventory as will be shown in Section IV.

Recent experimental results from TFTR (7) provide convincing support to this proposed fueling concept for ITER. Figure 3 (from Ref. 7) shows radial profiles of the electron, tritium, deuterium and hydrogen densities. In these innovative experiments, tritium neutral beams provided a tritium core fueling source and deuterium edge fueling was provided by recycling (the small hydrogen source is also from recycling). Over a distance of about 70 cm, the T-D mixture changes from ~3-1 in the plasma center to ~1-3 at the plasma edge. This gradient is maintained due to the good core particle confinement even though the tritium beam particle source is much lower than the D edge recycle source. The calculations for ITER presented above indicate density gradients over a similar distance (about the outer third of the plasma minor radius) and indicate that central fueling is not a requirement for isotopic tailoring.

III. Hardware Description

The baseline ITER plasma fueling system consists of a standard gas injection system and a continuous pellet fueling system using centrifuge and single-stage pneumatic injectors (3). All hardware is tritium-compatible and those components inside the ITER cryostat are qualified for the anticipated radiation environment. The centrifuge pellet injector has the advantage of no propellant gas load while the single-stage pneumatic injector can accelerate larger pellets to somewhat higher speeds. Present centrifuge injectors operate at speeds in the range 0.8 to 1.2 km/s while single-stage pneumatic injectors accelerate D-T pellets up to about 1.5 km/s. Centrifuge injectors accelerate pellets of typical equivalent spherical radius ~2 mm, while light gas guns, with a confined acceleration surface, can accelerate pellets up to 4 mm radius. The pneumatic pellet injector design is similar to the JET (now DIII-D) pneumatic pellet injector (8) and consists of two, single-stage, light gas guns in a common vacuum enclosure.

IV. Hydrogen Isotope Retention in Plasma Facing Components

A major benefit of isotopic fueling is the potential reduction in the tritium inventory of the plasma facing components. These inventories depend strongly on the materials selected. The design under consideration for ITER includes a beryllium limiter, a beryllium-coated first wall, and divertor plates also of beryllium. The tritium inventory in this material is very uncertain with estimates ranging from ~1 to over 10 kg-T.

To estimate the potential inventory reduction, we assume a deuterium fueling rate of 0.75 bar-l/s and a pellet (90% tritium - 10% deuterium) fueling rate of 0.25 bar-l/s. The resulting implantation flux for the first wall and limiter is about 25% tritium and 75% deuterium atoms. The scrape-off plasma entering the divertor is also at this ratio.

The DIFFUSE code (9) has been used to quantify the tritium inventory benefits of a deuterium-rich edge fueling source. For a base case of a 50-50 D-T flux, the wall inventory is as follows:

Species	% Flux	Wall inventory (atom %)
D	50%	49
T	50%	51

This case corresponds to a 50-50 D-T mix to all fueling sources. It would result in about a 50-50 D-T ratio in the core, assuming there is no strong isotope effect on the inward particle pinch velocity.

In a second DIFFUSE run, the total flux was held constant relative to the baseline while the deuterium flux was increased by 50% and the tritium flux was decreased by a factor of two. This case is representative of the potential wall tritium inventory reductions possible with a deuterium-rich edge fueling source:

Species	% Flux	Wall inventory (atom %)
D	75%	73.5%
T	25%	26.5%

In all runs the results were recombination-limited. It can be seen that the species wall inventory closely tracks the species flux to the wall so isotopic fueling does have the potential to reduce tritium wall inventories. With plasma facing component tritium inventories estimated to be at least a few kilograms, reductions of order one half would result in sizable tritium inventory reductions.

V. Inventories in ITER Tritium Components

Assessment of the impact of isotopic tailoring of the fuel sources must consider the tritium containing components: the cryopumps, the fueling systems, and the processing systems The inventory in the processing system is primarily in the Isotope Separation System (ISS). The ITER design, without isotopic tailoring, envisions direct recycle of about 90% of the fuel with only a 10% side stream removed for processing (mainly for protium removal) by the tritium plant. For comparison, we assume for isotopic fueling that all the fuel is processed by the ISS. This is a realistic assumption given the requirements for tritium-rich pellets and fairly pure deuterium gas. We also assume that, without tailoring, fueling would be predominately by gas puffing. Table 2 contains a comparison of estimates of tritium inventories in the fueling system, pumping system and ISS for the 90% direct recycle (protium removal side stream) and the isotopic fueling options. These estimates are primarily based on earlier studies but should be applicable to ITER.

The size and cost of the ISS are, for the reference ITER design, determined largely by the need to recover tritium from cooling and waste water. Based on the additional requirements of isotopic fueling identified above, the incremental increase in ISS capability for isotopic tailoring, in terms of hardware, energy and

cost, represents an increase within the 5% range and consequently it is not a strong cost driver. The penalty in tritium plant inventory for this added separation capability is significant (of order 100 g-T), but still within the range of an order of magnitude lower than the potential tritium inventory reduction in the first wall. Furthermore, the ISS inventory is readily recoverable during each facility shut down as opposed to the inventory contained in the first wall material which is not. It is also noted that the added inventory will apply only during the isotopic tailoring mode of operation when the last distillation column is in operation.

VI. Safety Implications

From the results presented in Sections IV and V, it can be seen that the tritium inventory in the torus can be decreased by about a factor of two by isotopic tailoring of the fuel sources. Depending of the materials and design of the plasma facing components, this could result in an inventory reduction of a few kilograms of tritium. This reduction is significant because the torus inventory is the largest single inventory and will therefore strongly influence siting considerations. Also, the confinement of torus inventories is complicated by the presence of nearby energy and pressure sources. Primarily these include the nuclear decay heat and the high pressure steam that could result from a loss of coolant event. Such events could also produce hydrogen by chemical reactions with high-temperature plasma facing materials. Additional complicating factors in torus tritium confinement is the presence of numerous vacuum vessel and cryostat penetrations and the necessity for components such as windows and bellows as part of the confinement boundary.

By contrast, the increase in the inventory in the ex-vessel tritium components is considerably smaller. Tritium in components such as the ISS can be separated from energy and pressure sources. Also, in contrast to the torus, confinement is readily accomplished by multiple and relatively simple confinement barriers such as piping and pressure vessels.

Reduction in tritium flows is an additional benefit in that it reduces the potential for operational losses into the environment and could reduce the potential for occupational exposure. Considering all these factors, the safety benefit of this concept is potentially significant.

Another implicit benefit of isotopic fueling is the reduction of the required tritium breeding ratio due to the increase in tritium burn fraction (10). The base case of 50%-50% D-T fueling of ITER represents a tritium burn fraction of 2.1% given a specified power of 1500 MW and a total fueling rate of 1.02 bar-l/s. With isotopic fueling of 0.24 bar-l/s (tritium) via pellets at the same fusion power, the tritium burn fraction is increased to 4.5%. As shown in Ref. 10, this factor of two increase in the tritium burn fraction can have a significant effect on the tritium breeding ratio, reducing it from values of ~1.15 to about 1.05. This will benefit an engineering test facility like ITER by reducing costly tritium startup inventories and it also has significant implications for an electric grid based on fusion power reactors.

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Table 1
Summary of WHIST Code Runs

Case	Gas (bar-l/s)	rp(cm)	vp(km/s)	Pen/a	ft(0)	ft(a)	BF(%)
469	0.65	0.425	1.5		0.63	0.34	3.3
470	0.70	0.425	1.5		0.47	0.29	3.8
471	0.75	0.425	1.5		0.38	0.23	4.5
472	0.75	0.425	1.5	0.3	0.34	0.24	4.5
473	0.75	0.535	3	0.37	0.43	0.24	4.5
474	0.75	0.3	1	0.25	0.30	0.23	4.5

Caption for Table I: Column 1 is the Whist code run number, column 2 is the deuterium gas fuel source (the pellet source is found by subtracting this value from the constant total fuel source term of 1.02 bar-l/s). Columns 3 and 4 are the pellet equivalent spherical radius and speed, respectively. Column 5 is the pellet penetration normalized to the ITER minor radius. Columns 6 and 7 are the tritium concentration on axis and at the edge, respectively. The last column is the tritium burn fraction expressed as a percent.

Table 2
Tritium Inventory (g)

Component	Protium Removal Only	Isotopic Fueling
Cryopumps	120	60
Fueling	120	60
ISS (90% recycle)	18*	n/a
ISS (no recycle)	30*	125

** Regardless of the degree of direct recirculation the ISS may need some minimum capability to produce at least trace quantity of 98% tritium which is required for volume reduction for storage during shutdown and similar operating states.*

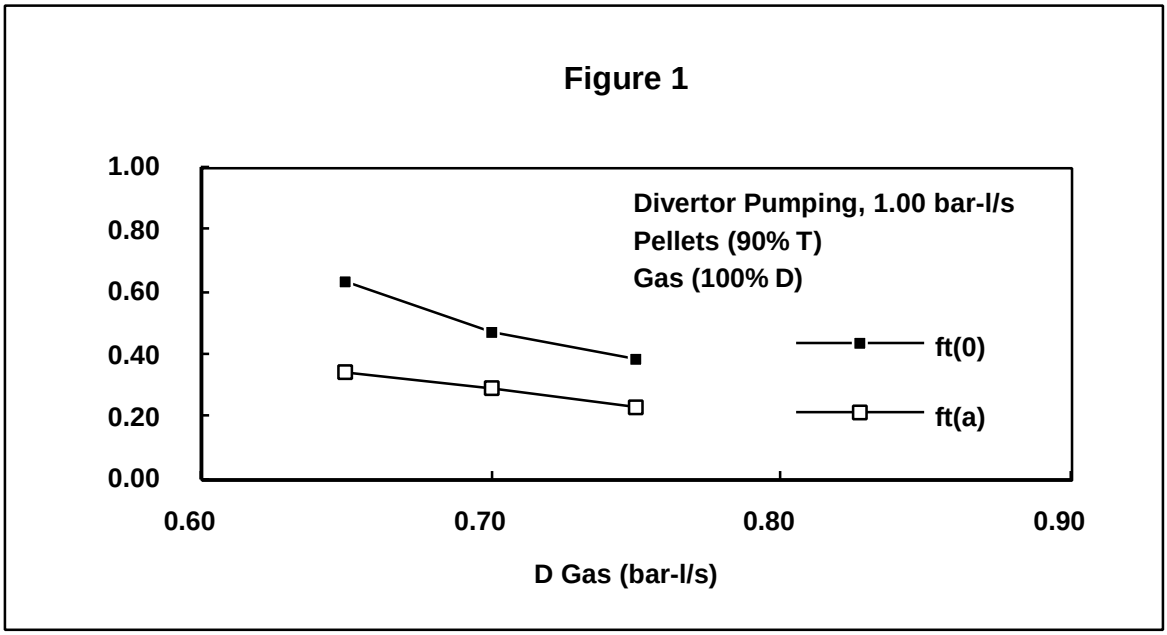
Figure 1
1.00
0.80
0.60
0.40
0.20
0.00
0.60
0.70
0.80
0.90
D Gas (bar-l/s)
Divertor Pumping, 1.00 bar-l/s
Pellets (90% T)
Gas (100% D)
ft(0)
ft(a)

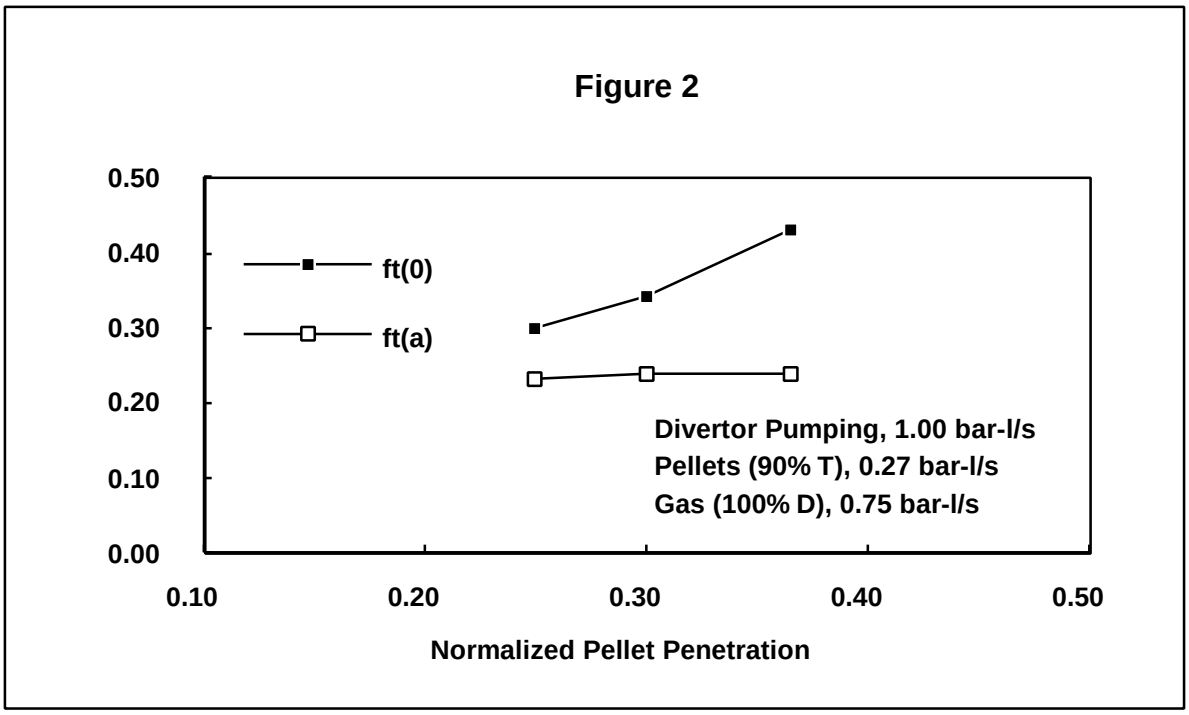
Figure 2
0.50
0.40
0.30
0.20
0.10
0.00
0.10
0.20
0.30
0.40
0.50
Normalized Pellet Penetration
ft(0)
ft(a)
Divertor Pumping, 1.00 bar-l/s
Pellets (90% T), 0.27 bar-l/s
Gas (100% D), 0.75 bar-l/s

Figure Captions

Figure 1 Tritium fraction on-axis and at the plasma edge as a function of the deuterium gas fueling level.

Figure 2 Tritium fraction on-axis and at the plasma edge as a function of the normalized pellet penetration.

Figure 3 Hydrogen, deuterium, tritium, tritium beam and electron densities in a TFTR tritium neutral-beam-heated plasma. Each region represents the density of the indicated species.

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