D-D Power Plants: the ultimate fusion goal

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(based mainly on a paper with Mohamed Sawan, University of Wisconsin-Madison [1])

I. Opportunities for Catalysed D-D fusion

Catalyzed D-D is the ultimate fusion cycle, because deuterium is essentially unlimited on earth. In this approach, the helium-3 and tritium fusion products are recycled to increase the charged particle fusion power. Numerous studies have been made of MFE power plants e.g., [2]. In the IFE area, the possibility of using a small amount of tritium has been suggested [3]. Recovered tritium produced in the burn, but not fused, might sustain this IFE-approach using only deuterium.

A difficulty with this fusion cycle is that the tritium from fusion, if left in the plasma, produces 14 MeV neutrons, leading to radiation damage comparable to that of the D-T cycle. In reference [1] it was shown that the damage problems might be alleviated by removing tritium before it could burn. Fortunately, the charged particle fusion power from burning the tritium is small compared to that from the helium-3 and removing it from the plasma makes little difference to the plasma power balance. In the MFE case, ion cyclotron power might be used to pump out tritium [4, 5, 6]. In the IFE case, the percentage of the energy produced as 14 MeV neutrons would be substantially less than in the conventional D-T case

In addition, any tritium recovered but not recycled, would decay to helium-3, which could be recycled to improve the power balance. In total, copious amounts of helium-3 would be produced, which could be used to support satellite D-He^3 power plants. With 90% tritium removal, the wall neutron-radiation damage would be comparable to that in a fission breeder reactor. This type of operation was also proposed in reference [7]. In reference [1], we determined illustrative parameters for an advanced tokamak [4] and an advanced stellarator [8, 9]. The approach has also been suggested for the levitated dipole [10].

II. Energy Spectrum of Produced Neutrons

The energy spectrum of neutrons produced in catalyzed D-D system includes components at 2.45 MeV and 14.1 MeV. The relative number of neutrons at these two energies depends on the fraction of tritium removed (f_{TR}) and the fraction of removed tritium that is recycled as He-3 (f_{Rec}). Note that, the characteristic decay time (half life) of tritium to He-3 is 12.3 years; consequently, in a single power plant it would take a time of that order before f_{Rec} gets to 0.5.

The fusion reactions are represented by the following equation that indicates the energy associated with the various neutron and charged particle products:

$$\begin{aligned}[4 + (1-f_{TR}) + (1+ f_{Rec}\, f_{TR})]D \quad \Rightarrow \quad & [T\ (1.01 MeV) + p\ (3.02 MeV)] + \\ & [^3He\ (0.82 MeV) + n\ (2.45 MeV)] + \\ & [1-f_{TR}][^4He\ (3.5 Mev) + n\ (14.1 Mev)] + \\ & [1+ f_{Rec}\, f_{Tr}][^4He\ (3.6 Mev) + p\ (14.7 Mev)]. \qquad 1.\end{aligned}$$

Based on this equation, the fraction of fusion energy carried by D-D neutrons (2.45 MeV) is given by $2.45/(43.2 - 17.6\, f_{TR} + 18.3\, f_{Rec}\, f_{TR})$, and the fraction of fusion power carried by 14.1 MeV neutrons is $14.1(1-f_{TR})/(43.2 - 17.6\, f_{TR} + 18.3\, f_{Rec}\, f_{TR})$.

III. Results of Radiation Damage

One-dimensional calculations were performed in [1] to determine the impact of removing tritium and recycling part of it as He-3 in a catalyzed D-D system [9]. The damage parameters calculated were the atomic displacement rate and the gas production rate. The total energy flux from neutrons and charged particles at the first wall (FW) was kept at $5MW/m^2$ in all the cases analyzed. The ONEDANT module of the DANTYS 3.0 discrete ordinates particle transport code system [11] was used along with the International Fusion Energy Nuclear Data Library, FENDL-2 [12] in a 175 neutron energy group structure.

Three materials considered for use as structural materials in fusion first wall/blanket/shield systems were analyzed: the ferritic steel alloy 9Cr-2WVTa, the SiC/SiC composite, and the vanadium alloy V4Cr4Ti. The results for ferritic steel are shown in the figure.

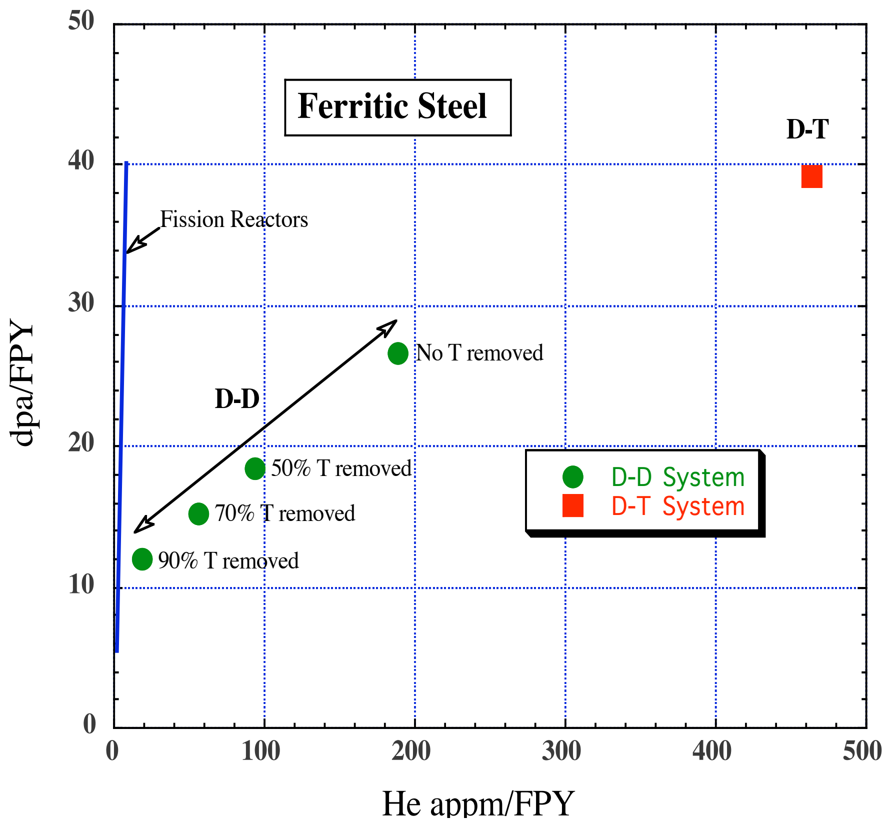

Figure I. dpa and He production rates in ferritic steel for D-T and D-D systems for the first wall of a ferritic steel/H_2O shield.

IV. Needed programs.

In the MFE case, coupled experimental and theoretical work is needed on using ICH to remove tritium (or a surrogate ion) from tokamak and stellarator plasmas. In both cases, it would be interesting to find out if magnetic ripple (natural in a stellarator) could be used to enhance tritium pumping.

In the IFE case, further computations are needed to refine capsule design and determine how much tritium would be produced by D-D reactions and how much of it would remain unburned.

Systems studies are needed to refine the understanding of what might be possible, including finding an optimum approach for storing the massive amounts of tritium that would be produced.

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