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On the possibility of using alternative thermonuclear reactions for energy production, neutron generation and other applications

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Abstract. The possibilities of using alternative reactions D–D and D–3He in connection with the use of thermonuclear energy are considered. The prospects for alternative magnetic traps are discussed. The possibilities of using the D–D reaction in the neutron source of fusion–fission hybrid systems are considered. The physical limitations and possibilities for the D–3He reactor are discussed.

1. Introduction
The prospects of nuclear fusion energy produced in a high-temperature plasma is today considered practically only on the basis of the deuterium–tritium (D–T) reaction in tokamak-type magnetic confinement devices. In the D–T phase of the ITER project, it is supposed to demonstrate regimes with a plasma power gain factor $Q = 5–10$. The parameters of today’s tokamaks correspond to the conditions of working with D–T fuel at $Q \approx 1$. At the same time, there are two serious problems in this area that must be solved in order to create economically viable fusion energy. The first is the absence of materials with the radiation resistance under conditions of a neutron flux of 1 MW/m². The lifetime of today’s materials under such conditions, apparently, will not exceed 3–5 years. Another strategy for using fusion energy involves the creation of a fusion source of fast neutrons with plasma power gain of $Q \approx 1$ and a neutron flux level of $\sim 0.2$ MW/m² as part of a symbiotic fusion–fission system with a sub-critical blanket for energy generation, production of fissile materials, and closing of the nuclear fuel cycle. The second very serious problem is the tritium cycle, which is associated with insufficient opportunities for the accumulation of tritium in the quantities necessary for large-scale energy [1].

Deuterium is comparatively available, and among the products of the D–D reaction there is tritium, which burn produces fast neutrons. Such a fuel cycle can be implemented on the basis of a low-gain tokamak device of comparable size and power with ITER, but with a reduced aspect ratio and an increased ratio $\beta$ of plasma pressure to magnetic pressure [2]. In this case, unburned tritium can be used in the D–T system with smaller tokamak. An increase in the rate of the D–D reaction is possible with a powerful neutral beam injection (NBI) into the plasma, but this apparently requires $\beta \approx 0.5$ [3]. Some effect can also be achieved by adding lithium isotopes to a deuterium plasma [4].

The most important property of D–3He fuel is the low radioactivity of the energy production cycle with energy output with neutrons $\sim 5\%$. The amount of helium-3 in the atmosphere and the Earth’s interior is negligible, but the need for it today is very high, as it is used, in particular, in neutron monitoring systems [5]. Therefore, in the foreseeable future, it is possible to implement projects for its
delivery from the Moon. It is possible in principle to create a D–3He reactor based on a spherical tokamak, but the system parameters are extremely high [6].

Given the considerable time frame for developing ITER engineering solutions, physically based concepts for using alternative reactions are relevant today.

2. The power balance

In deuterium-based plasma, the following reactions are possible:

\[ D + D \rightarrow n (2.45 \text{ MeV}) + ^3\text{He} (0.817 \text{ MeV}); \]  
\[ D + D \rightarrow p (3.02 \text{ MeV}) + ^4\text{He} (1.01 \text{ MeV}); \]  
\[ D + ^3\text{He} \rightarrow p (14.68 \text{ MeV}) + ^4\text{He} (3.67 \text{ MeV}). \]

If tritium produced in the reaction (2) has time to react with deuterium, then the output of fast 14 MeV D–T neutrons (reaction (3)) becomes essential. Due to the difference in cross sections and rates of reactions (3) and (4) such regimes are possible when a considerable part of tritium (in certain conditions almost all the tritium) is burnt whereas almost all helium-3 escapes the trap without reacting with deuterium. Such a fusion cycle is commonly called semi-catalyzed. A minor quantity of tritium will not react with deuterium before leaving the trap. It is supposed that it is returned to the plasma after extracting from the gas mixture removed from the vacuum chamber. In this case, the D–D cycle is considered in our study.

The fraction of power in 14 MeV neutrons in such a cycle is \( P_{14} / P_{\text{fus}} \approx 0.56. \)

In D–3He cycle, reaction (4) is the main one. Other reactions are also considered.

The integral energy balance has the form [7, 8]:

\[ E_{\text{th}} - P_{\text{fus}} + P_{\text{ext}} + P_{b} + W_{th} / \tau_{E}. \]  

where \( P_{\text{fus}} \) is fusion power, \( P_{n} \) is the neutron power, \( P_{\text{ext}} \) is the power of external heating, \( P_{b} \) is the bremsstrahlung power, \( P_{s} \) is the synchrotron loss power, \( W_{th} = \frac{3}{2} n_{i} k_{B} T_{e} + \frac{3}{2} n_{i} k_{B} T_{i} \) is the thermal energy of the plasma, \( f \) is the proportion of charged products loss.

Total efficiency of the hybrid fusion–fission system is

\[ \eta_{\text{net}} = P_{\text{net}} / P_{\text{fus}} = \eta_{e} (\alpha_{n} M + 1 - \alpha_{n} + 1 / Q) - 1 / (\eta_{d} Q) \]

where \( P_{\text{net}} \) is outlet (electric) power, \( M \) is blanket multiplication, \( \eta_{e} \) the efficiency of heat into electricity conversion, \( \eta_{d} \) is external heating system (driver) efficiency.

For the analysis \( \eta_{e} \approx 0.35 \) is supposed. Modern systems of neutral beam injection (NBI) heating and the systems of electron cyclotron resonance (ECR) heating have an efficiency of \( \eta_{d} \approx 0.4. \)

Blanket multiplication is

\[ M = \frac{E_{\text{fis}}}{E_{\text{fus}}} \frac{k_{\text{eff}}}{\nu (1 - k_{\text{eff}})}. \]

where \( E_{\text{fis}} \) is fission energy (\( E_{\text{fis}} \approx 200 \text{ MeV} \)), \( E_{\text{fus}} \) is fusion neutron energy, \( \nu \) is the number of fission neutrons per one fusion neutron (\( \nu = 3 \)), \( k_{\text{eff}} \) is effective neutron multiplication ratio.

Electrical efficiency of a fusion–fission hybrid is

\[ \eta^{*} = \frac{P_{\text{net}}}{P_{\text{th}}} = \eta_{e} [(\eta_{d} \eta_{\text{net}})^{-1} + 1]^{-1}, \]

where \( P_{\text{th}} \) is thermal power produced in the blanket.
3. Fusion neutron source based on deuterium plasma

Table 1 shows the basic parameters of the plasma and hybrid systems: radius of the plasma column $a$, plasma length $L$ (for linear systems), aspect ratio of the tore $A = R/a$ (R is the major radius), plasma relative pressure $\beta$, vacuum magnetic field $B_0$, plasma current $I_p$ (for tokamaks), energy of injected atoms $E_0$ (for the case of NBI heating), averaged plasma temperature $T$, plasma power gain factor $Q$, energy confinement time $\tau_E$, deuterium density $n_0$, plasma volume $V_p$, volume of blanket $V_{bl}$, fusion power $P_{fus}$, neutron power $P_n$, fraction of energy in fast neutron $P_{n1d}/P_{fus}$, neutron energy flux from the plasma $J_n$, blanket heat power density $P_{bl}$, heat power of the blanket $P_{bl}$, output power of the hybrid system $P_{net}$. For comparison, version #5 shows a tokamak with 50%:50% D–T fuel. The volume of the blanket was estimated for the case of its mean thickness of 1 m.

| Version | #1 | #2 | #3 | #4 | #5 | #6 | #7 | #8 | #9 | #10 |
|---------|----|----|----|----|----|----|----|----|----|----|
| Configuration | Tokamak [9] | Mirror [10] | Mirror [11] |
| $a$, m | 2 | 1.5 | 2 | 2 | 1 | 1 | 0.5 | 1 | 2 | 2 |
| $L$, m | – | – | – | – | 10 | 10 | 4 | 4 | 4 | – |
| $A$ | 3 | 2 | 2 | 2 | 3 | – | – | – | – | – |
| $\beta$ | 0.026 | 0.16 | 0.16 | 0.16 | 0.026 | 0.5 | 0.5 | 0.5 | 0.5 | 0.5 |
| $B_0$, T | 5.3 | 4.5 | 4.5 | 4.5 | 5.3 | 3.5 | 4.9 | 4.1 | 5.8 | 11.2 |
| $I_p$, MA | 15.8 | 36 | 48 | 48 | 7.9 | – | – | – | – | – |
| $E_0$, MeV | – | – | – | – | – | 0.5 | 1 | 2 | – | – |
| $T$, keV | 10 | 9 | 12 | 12.5 | 4.5 | 25 | 30 | 37 | 72 | 113 |
| $Q$ | 0.050 | 0.25 | 0.39 | 0.25 | 1.7 | 1 | 0.084 | 0.23 | 0.53 |
| $\tau_E$, s | 2.2 | 4.1 | 6.0 | 5.7 | 1.16 | 6.3 | 3.2 | 0.21 | 0.58 | 0.74 |
| $n_0$, $10^{20}$ m$^{-3}$ | 0.7 | 3.9 | 2.9 | 2.8 | 0.85 | 3.0 | 5.0 | 2.8 | 2.8 | 5.6 |
| $V_p$, m$^3$ | 800 | 330 | 790 | 790 | 100 | 31 | 3 | 12 | 50 | 50 |
| $V_{bl}$, m$^3$ | 175 | 105 | 190 | 190 | 44 | 45 | 1.1 | 18 | 36 | 36 |
| $P_{fus}$, MW | 10 | 72 | 170 | 110 | 50 | 20 | 7.6 | 9.2 | 76 | 520 |
| $P_n$, MW | 8 | 58 | 140 | 77 | 40 | 13 | 5 | 6 | 50 | 340 |
| $P_{n1d}/P_{fus}$ | 0.56 | 0.56 | 0.56 | 0.43 | 0.8 | 0.57 | 0.57 | 0.57 | 0.57 | 0.57 |
| $J_n$, MW/m$^2$ | 0.015 | 0.16 | 0.22 | 0.14 | 0.25 | 0.2 | 0.4 | 0.25 | 0.99 | 10.0 |
| $P_{bl}$, MW/m$^3$ | – | 30 | 29 | 25 | 21 | 9.5 | 15 | 25 | 57 | 57 |
| $P_{bl}$, MW | – | 3050 | 5590 | 4750 | 930 | 430 | 400 | 910 | 3420 | 14500 |
| $P_{net}$, MW | – | 360 | 850 | 550 | 250 | 100 | 94 | 45 | 380 | 2600 |
| $\eta^*$, % | – | 12 | 15 | 12 | 27 | 23 | 23 | 23.5 | 5 | 11 | 18 |

1) Burns 50% of tritium produced in the D–D reaction.
2) D–T fuel (50%:50%).
3) NBI heating, significant non-Maxwellian population.
4) In the central cell.
5) Required value for thermal components of the plasma.

At low $\beta$, the efficiency of a tokamak system with a D–D fuel (version #1) is unacceptably low. In version #2, the neutron flux is relatively high. In versions #3 and #4, neutron fluxes are at an acceptable level, and the value $\eta^*$ is slightly higher than the value for a purely fusion reactor. Version #4 assumes that 50% of the tritium produced is not burned and, therefore, can be used in another reactor of comparable power with a neutron source on D–T fuel (version #5). The efficiency of the latter is significantly higher, and the size is smaller. In this case, for two reactors $\eta^* \approx 15\%$. Such a fuel cycle scheme may be more attractive from an economic point of view than a scheme with full use of tritium in a catalyzed D–D cycle.
The high efficiency of systems based on an open trap (versions #6 and #7) is associated with a relatively high gain $Q = 1$. For this, the plasma confinement time in an open trap must be very high ($\tau_e \sim 5$ s), therefore, it is obvious that it is necessary to complicate the system to reduce losses plasma through the open ends of the trap. We note that significant progress has been demonstrated in recent experiments on open traps [12].

In the case of NBI heating (versions #8, #9 and #10), a population of high-energy ions is formed in the plasma. Required confinement time is reduced. With an increase in the energy of the beam atoms $E_0$, plasma gain $Q$ increases. For the capture the beam, the size and density of the plasma must be relatively large. Therefore, the power of such a system and the neutron load on the first wall increase to unacceptable values.

The magnetic field of the coils is limited by the capabilities of existing superconductors. For all cases, with the possible exception of version 10, the maximum field on the surface of the superconductor does not exceed 20 T.

The reduction of neutron fluxes to the wall in an open trap is possible when the wall is distanced from the plasma. The increase in power in a linear system is relatively easily achieved by a proportional increase in its length. The heat density in the blanket allows the gas-cooled blanket, for example, with helium as a working fluid.

4. Feasibility of $D-^{3}\text{He}$ reactor

Various authors considered different confinement systems: tokamak, spherical tokamak (ST), field reversed magnetic configuration (FRC), open mirror trap, multipole traps. Table 2 shows calculated data for the operating modes of $D-^{3}\text{He}$ magnetic fusion reactors. Table 2 contain the values of the following parameters: plasma column radius $a$, plasma length $L$ (for systems with linear geometry), major radius $R$ (for toroidal systems), aspect ratio $A = R/a$, plasma cross section elongation $k$, magnetic field induction on the magnetic axis $B_0$ (vacuum value), plasma current $I_p$, average value of the $\beta$ parameter, relative helium-3 content in the fuel mixture (fuel composition) $x_{^{3}\text{He}} = n_{^{3}\text{He}}/n_D$, plasma temperature $T$, synchrotron wall reflectivity $\Gamma_s$, plasma energy confinement time $\tau_e$, fusion power $P_{\text{fus}}$, relative power of bremsstrahlung $P_{\text{br}}/P_{\text{fus}}$, relative power of cyclotron losses $P_c/P_{\text{fus}}$, neutron flux $J_n$.

| Reactor type | Tokamak | ST | FRC | FRC | FRC | Mirror | Multipole |
|-------------|---------|----|-----|-----|-----|--------|-----------|
| Reference   | [13]    | [14] | [6]  | [15] | [16] | [17]   | [18]      | [19]       |
| $a$, m      | 2.66    | 2   | 1.68 | 1.68 | 1.68 | 1.68   | 1         | < 2        |
| $L$, m      | –       | –   | –    | 22.2 | 35   | 40     | –         | –          |
| $R$, m      | 8.31    | 6   | 3.4  | –    | –    | –      | –         | –          |
| $A$         | 3.1     | 3   | 1.7  | 1    | 1    | 1      | –         | –          |
| $k$         | –       | 2.5 | 2.8  | 13.2 | 22   | 10     | –         | –          |
| $B_0$, T    | 11.1    | 11.3| 5.5  | 5.36 | 5    | 5.0    | 5.4       | 4.6–14     |
| $I_p$, MA   | 57.2    | 38  | 110  | 190  | –    | –      | –         | –          |
| $\beta$     | 0.067   | 0.09| 0.5  | 0.98 | 0.46 | 0.93   | 0.7       | > 0.5      |
| $x_{^{3}\text{He}} = n_{^{3}\text{He}}/n_D$ | 1      | 0.2 | 1    | 1.35 | 1    | 1      | 1         | 0.3        |
| $T$, keV    | 57      | 42  | 44   | 83.5 | 28   | 64     | 65        | 50         |
| $\Gamma_s$  | –       | 0.92| 0.65 | –    | 0.5  | 0.65   | 0.65      | –          |
| $\tau_e$, s | 17.4    | 14  | 9.4  | 6.87 | 4    | 6.3    | 7.1       | > 6        |
| $P_{\text{fus}}$, MW | 2602 | 2500 | 2225 | 1790 | 1000 | 1214 | 675 | ~ 1000 |
| $P_{\text{br}}/P_{\text{fus}}$ | 0.33 | 0.40 | 0.57 | –    | 0.53 | 0.52 | 0.25 | 0.27 |
| $P_c/P_{\text{fus}}$ | 0.48 | 0.33 | 0.14 | –    | 0.06 | 0.01 | 0.10 | 0.3 |
| $J_n$, MW/m² | 0.1   | 0.14 | 0.16 | 0.3  | 0.15 | 0.26 | –        | –          |

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Table 2. Parameters of $D-^{3}\text{He}$ fuelled reactors with $Q \sim 10$
For tokamaks $\tau_E$ can be determined by ITER scaling IPB98y2. In spherical tokamaks, the energy confinement is described by a somewhat different scaling showing better confinement. For the FRC, open mirror trap and multipole configuration, the required values of $\tau_E$ are presented. For these systems consider the possibility of fulfillment of these requirements.

For an open ambipolar mirror trap, table 2 shows the values of the parameters of the central cell. In end-plug cells, so-called thermal barriers are assumed to be created by heating the plasma to temperatures $T_p \approx 3T$ ($T$ is the temperature in the central cell) at relatively low density. The required confinement time is less than the time of longitudinal losses and it is determined by the transverse transport. The latter is apparently associated with turbulence and drift-type instabilities.

For multipole traps, there are no valid theories describing plasma transport across a magnetic field. We can only say that the drift instabilities typical for hot magnetized plasma are not excluded for such magnetic traps.

The magnetic field of the coils is limited by the capabilities of existing superconductors. For all the versions considered in table 2, the maximum field on the surface of the superconductor does not exceed 20 T. The level of neutron flux on the first wall of the reactor is 0.1–0.3 MW/m$^2$. This value corresponds to a resource up to 30 years. Therefore, from a technical point of view, the idea of a reactor with a D–$^3$He mixture does not contradict the level of today’s technologies.

Magnetic field of $B_0 \approx 11$ T is still very high for tokamaks. According to today’s understanding, it is problematic to get the elongation of the plasma cross section $k = 2.5$ with an aspect ratio of $A = 3$ in classical tokamaks. For a spherical tokamak, the field $B_0 \approx 5$ T is high. Technically available value is $B_0 \approx 3$ T. The elongation $k = 2.8$ is also high for a spherical tokamak. If to lower the considered parameters, the power of spherical tokamak reactor becomes unacceptably low. In principle, the requirements for tokamaks are at the limit of technical capabilities. It probably means low competitiveness of such reactors.

Alternative to tokamak systems are technically simpler then tokamak. In multipole traps, a certain technical problem is the maintenance of the temperature regime of superconductors in levitating coils surrounded by hot plasma.

An open trap and FRC have a linear geometry of the magnetic system, which distinguishes them from other systems. In an open trap, it is necessary to create high potential barriers at the ends of the system to reduce the intensity of plasma losses along open magnetic field lines. The need for powerful heating of the terminal barriers, apparently, does not allow the achievement the total plasma gain $Q > 10$ with compact sizes of the reactor.

5. Conclusions

The potential use of the D–D reaction is important from the point of view of prospects for large-scale energy based on hybrid systems. A significant advantage of a hybrid system with a neutron source based on the D–D reaction is that there is practically no need for tritium production in the blanket. Tritium is produced only in the D–D reaction in the plasma. This greatly simplifies a number of problems associated with the technologies of the tritium cycle of the hybrid reactor.

The use of a gas-cooled blanket with a solid raw material makes it possible to use a single-circuit with a closed gas-turbine installation, similar to high-temperature gas-cooled reactors with spherical fuel elements.

In the future, open traps may become prototypes of $Q \sim 1$ neutron sources on D–T mixtures. But from the point of view of the D–$^3$He reactor their prospects are not obvious today.

Apparently, FRC today looks like the most promising system for the concept of the D–$^3$He reactor. In FRC, the plasma is confined in the region of closed magnetic field lines. Therefore, this system is devoid of the disadvantages associated with longitudinal losses. The creation of an FRC is also not associated with technical difficulties. Perhaps the risks may be related to the fact that today’s FRC facilities have scales several times smaller than those required to create a reactor, both in terms of size and in terms of total energy. Moreover, if we consider D–$^3$He fuel, the difference is at least an order of magnitude.
As helium-3 is necessary for a number of today’s critical technologies, the question of its delivery from the Moon will probably be solved, in the future. Therefore, exploratory studies in the use of helium-3 in promising energy are not only relevant, but also timely.

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