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HYBRID FUSION-FISSION SYSTEM WITH NEUTRON SOURCE BASED ON DEUTERIUM PLASMA

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Abstract

Development of hybrid fusion-fission systems appears today as a promising area in practical use of thermonuclear fusion energy. Thermonuclear plasma in such systems is the source of fast neutrons with the power gain factor $Q < 1$ power amplification factor in plasma. Hybrid system high amplification is generally achieved through nuclear reactions in the subcritical blanket surrounding plasma. Not only power could be produced in such a blanket, but also nuclear fuel, and waste of the nuclear fuel cycle could be disposed. The problem of systems using the thermonuclear reaction between deuterium and tritium lies in the lack of tritium reserves and in the limited possibilities of its production. Therefore, the work considers organization of a fuel cycle based on the deuterium-deuterium reaction. Options of a neutron source based on tokamak and linear system of the open trap type were examined. Magnitude of the so-called hybrid system electrical efficiency (ratio of the system electrical output to the blanket thermal power) was estimated. Calculations demonstrated fundamental possibility of realizing substantial neutron yield from deuterium plasma. To achieve acceptable performance, power gain in thermonuclear plasma should be Q = 0.5–1. In a tokamak of reasonable scale and when working on deuterium, the gain should be $Q \sim 0.3$. Potential advantages of linear systems associated with the possibility of high-pressure plasma confinement are presented

Keywords

Deuterium, fusion neutrons, hybrid reactor, magnetic confinement, plasma, energy balance

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Introduction. It would seem that work on thermonuclear fusion came close to creating an industrial reactor. There is no doubt that ITER (International Thermonuclear Experimental Reactor) tokamak [1] under construction operating with a mixture of deuterium and tritium (D–T) would demonstrate modes with a plasma power gain factor of Q = 5–10 ($Q = P_{fus}/P_{ext}$, where P_{fus}

is the released fusion power and P_{ext} is the absorbed power of external heating). Despite this, realization of purely thermonuclear energy today arises certain pessimism, which is quite justified in the opinion, in particular, of the ITER project initiator, academician E.P. Velikhov [2]. Apparently, a more promising and, most importantly, faster way to practical use of thermonuclear fusion is connected not to a pure thermonuclear reactor, but to a symbiotic fusion-fission system combining thermonuclear and nuclear fission fuel cycles. Moreover, on the basis of such a hybrid system, it is possible to solve the problem of closed nuclear fuel cycle, which potentially is more efficient than involving fast neutron reactors. It should be noted that the idea of such use of thermonuclear reactions is not that new. The first US-Soviet symposium on hybrid reactors was held in Livermore (USA) in 1976 [3], then the second symposium was organized in Moscow in 1977 [4].

There are at least two most serious problems without solving which it is hardly possible to create an economically feasible commercial thermonuclear reactor. Firstly, the problem of materials, and it is unlikely to be resolved in the foreseeable future. Under conditions of neutron fluxes in the energy producing D–T reactor with Q = 5–10, lifetime of plasma facing components (the so-called first wall) will not apparently exceed 3–5 years even for the best existing radiation resistant materials.

Parameters of modern tokamaks are such that Q ~ 1 could be achieved by burning D–T fuel. Therefore, along with continuation of the ITER line in the form of a demonstration reactor (DEMO), today a strategy is being considered of using thermonuclear energy not as a separate area, but as part of the fusion-fission fuel cycle. Thermonuclear plasma with Q ~ 1 in hybrid systems is a source of fast neutrons for a subcritical blanket surrounding plasma. Neutrons are used in the blanket for energy generation, production of fissile isotopes (plutonium-239 and uranium-233) and transmutation of nuclear fuel cycle waste. Neutron fluxes' in a thermonuclear neutron source with Q ~ 1 is significantly lower than in the reactor with Q ~ 10. Despite the fact that the hybrid system concept is based on existing technologies, combination of many functions in one device presents considerable difficulties from the point of view of its designing. Nevertheless, even this complex system appears more suitable for power engineering than a pure thermonuclear reactor using D–T fuel [5].

Secondly, there is an acute problem of tritium reserves and possibilities of its accumulation in the amount required to launch large-scale energy projects [6, 7]. Scenarios were considered, where stages of operation on the D–T mixture are preceded by tritium production in the D–D reaction, i.e., the so-called D–D start [8]. But such modes do not completely remove the need for the initial tritium stockpile even for a demonstration reactor (DEMO).

Deuterium is attractive primarily due to its relative availability. However, the D–D reaction low rate does not make it possible to $Q \sim 10$ in the tokamak reactor at reasonable parameters [9]. Conditions for sufficiently high efficiency of a thermonuclear system with D–D reaction require plasma confinement with high β value (β is the ratio of plasma pressure to magnetic pressure). In principle, this could be performed in an open trap, as well as in the field reversed configuration (FRC) and in a number of other systems [10–12].

This paper presents estimation of main parameters of hybrid systems with a neutron source based on the D–D reaction. Options of a thermonuclear plasma confinement magnetic system based on tokamak [13, 14] and linear magnetic configuration of the open trap type [10, 15–17] are considered. Cases of Maxwellian plasma [10, 15] and nonequilibrium plasma heated by a powerful beam of neutral atoms (neutral beam injection or NBI) [16, 17] are examined. In the latter case, NBI maintains a significant population of fast particles in plasma, which provides an increase in the reaction rate and, consequently, a decrease in plasma confinement requirements.

In a tokamak with the aspect ratio $A = R/a \sim 3$ (R is the torus major radius, a is the minor radius) due to the low β values, it is problematic to achieve $Q \sim 1$ with the D–D fuel. Therefore, case of the reduced aspect ratio $A = 2$ is considered accepting $\beta = 0.16$. Examining tokamak makes it possible to estimate with a high degree of reliability, since they are based on the ITER Physics Basis (IPB) laws [1].

From the technical point of view, linear systems, such as open trap or FRC, are most attractive due to their design simplicity. From the point of view of required thermonuclear regimes realizability, reliable physical justification of the concept based on open trap or FRC is hardly possible now, since parameters of today experimental facilities are relatively far from the $Q \sim 1$ regimes. Nevertheless, considering simple geometry is useful for estimating plasma parameters, sizes, power, level of neutron fluxes and a number of other important parameters of a hybrid system.

Hybrid system parameters. The following reactions are possible in deuterium plasma:

$$\mathrm{D} + \mathrm{D} \to n\ (2.45\ \mathrm{MeV}) + {}^{3}\mathrm{He}\ (0.817\ \mathrm{MeV}) \tag{1}$$

$$\mathrm{D} + \mathrm{D} \to p\ (3.02\ \mathrm{MeV}) + \mathrm{T}\ (1.01\ \mathrm{MeV}) \tag{2}$$

$$\mathrm{D} + \mathrm{T} \to n\ (14.1\ \mathrm{MeV}) + {}^{4}\mathrm{He}\ (3.5\ \mathrm{MeV}) \tag{3}$$

$$\mathrm{D} + {}^{3}\mathrm{He} \to p\ (14.68\ \mathrm{MeV}) + {}^{4}\mathrm{He}\ (3.67\ \mathrm{MeV}) \tag{4}$$

If tritium produced in reaction (2) can to react with deuterium, then the fast D–T neutrons with the energy of 14.1 MeV (reaction (3)) becomes noticeable. Due to differences in cross sections and reaction rates (3) and (4), a situation could occur, when significant part of tritium (possibly all tritium under certain conditions) has time to burn, and helium-3, on the contrary, has time to almost completely leave the trap not having time to react with deuterium. Such thermonuclear fuel cycle is called semi-catalyzed. This case of the D–D cycle is being considered further. Power fraction in fast neutrons with energy of 14.1 MeV in such a cycle is $P_{n14}/P_{fus} \approx 0.56$.

Total efficiency of the hybrid fusion-fission system is [18]:

$$\eta_{net} = P_{net} / P_{fus} = \eta_e(\alpha_n M + 1 - \alpha_n + 1/Q) - 1/(\eta_d Q), \tag{5}$$

where P_{net} is the output (electric) power; P_{fus} is the fusion power; M is the blanket multiplication; η_e is the efficiency of heat into electricity conversion; η_d is the external heating system (driver) efficiency.

Expression (5) takes into account that the blanket thermal power is:

$$P_{th} = P_{fus}(\alpha_n M + 1 - \alpha_n + 1/Q), \tag{6}$$

and the power consumed by external heating systems is:

$$P_d = P_{fus} / (Q\eta_d). \tag{7}$$

Let us assume $\eta_e \approx 0.35$ for the analysis. Modern neutral beam injection (NBI) heating systems and electron cyclotron resonance (ECR) heating systems have the $\eta_d \approx 0.4$ efficiency. Blanket multiplication is connected to the k_{eff} effective neutron multiplication factor by the following relation:

$$M = \frac{E_{fis}}{E_{fus}} \frac{k_{eff}}{\nu(1 - k_{eff})}, \tag{8}$$

where E_{fis} is the fission energy ($E_{fis} \approx 200$ MeV); E_{fus} is the fusion neutron energy; ν is the number of fission neutrons per one fusion neutron ($\nu \approx 3$).

Electrical efficiency of the hybrid system is

$$\eta^* = P_{net} / P_{th} = 1/[1/\eta_e + 1/(\eta_e\, \eta_d\, \eta_{net}\, Q)]. \tag{9}$$

Figure 1 presents the electric efficiency values depending on the plasma power gain at different blanket multiplication values for systems with fusion neutron sources based on the D–T cycle and on the semi-catalyzed D–D cycle. Differences are due to the fact that the neutrons' energy fraction in D–T cycle is

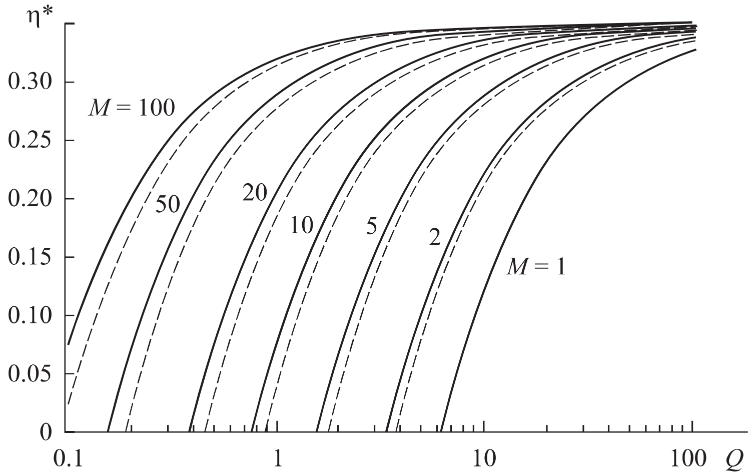

Fig. 1. Hybrid system electric efficiency depending on plasma power gain at various blanket multiplication values: solid lines is D–T fusion fuel cycle (α_n = 0.8), dashed lines is D–D cycle (α_n = 0.66)

higher. From the safety standpoint, the k_{eff} < 0.9–0.95 values are acceptable. The M ~ 50 blanket multiplication values satisfy this condition. The Q = 0.5–1 range could be considered acceptable. In particular, at Q = 1, $k_{eff} \approx 0.9$, $M \approx 35$, system efficiency $\eta_{net} \approx 5$ and electrical efficiency $\eta^* \approx 0.23$.

It should be noted that in the case of a purely thermonuclear reactor ($M \approx 1$) with $Q = 10$, $\eta_e \approx 0.35$ and $\eta_d \approx 0.4$, efficiency is $\eta_{net} \approx 0.135$, and electrical efficiency is only $\eta^* \approx 0.12$. Low efficiency is connected to the fact that a feature of thermonuclear power plants is significant internal energy consumption by the plasma heating systems [19]. According to estimates [20], the promising thermonuclear power plants efficiency could reach $\eta^* \approx 0.23$, and the cost of electricity production could be comparable to the level of today's renewable energy. Hybrid systems look more attractive from the economic point of view, while the reactor price is increasing approximately linearly with the size of the magnetic system [21]. Magnetic system accounts for up to 40 % of capital costs in construction of a power plant, which significantly exceeds other types of costs [20]. Therefore, it is important from the economic point of view to most efficiently use the magnetic field to confine plasma at the maximum acceptable pressure, which could be provided in systems alternative to tokamak with high β values.

Let us compare the main parameters of hybrid fusion-fission systems based on tokamaks and linear magnetic configuration systems. Figure 2 presents a scheme of the hybrid reactor based on an open trap. In principle, this schemealso corresponds to other linear magnetic traps, for example, FRC. Let us use the

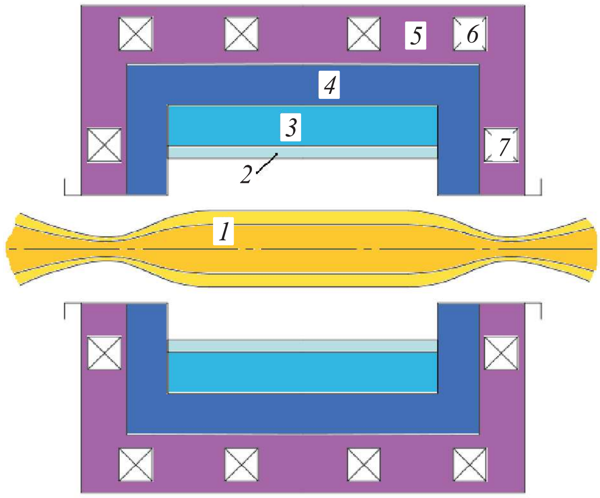

Fig. 2. Scheme of a hybrid reactor based on the open trap (longitudinal section):
1 is plasma; *2* is first wall; *3* is blanket; *4* is radiation shield; *5* is casing structural elements; *6* is central section magnetic coils; *7* is magnetic mirror coils

previously developed energy balance models to calculate the plasma parameters [10–17], which detailed description would be omitted here.

The table shows values of plasma and hybrid systems main parameters: a is plasma columm radius; L is plasma columm length (linear systems); $A = R/a$ is torus aspect ratio (tokamaks); β is relative plasma pressure; B_0 is coils' magnetic field induction in vacuum; I_p is plasma current (tokamaks); E_0 is injected atoms energy (NBI heating); T is volume average plasma temperature; Q is plasma power gain factor; τ_E is energy confinement time; n_D is deuterium average density; V_p is plasma volume; V_{bl} is approximate blanket volume; P_{fus} is fusion power; P_n is neutron power; P_{n14}/P_{fus} is energy fraction in fast neutrons; P_b/P_{fus} is relative losses due to bremsstrahlung; P_s/P_{fus} is relative losses due to synchrotron radiation; J_n is neutron energy flux from plasma; P_{bl} is heat release volume density in blanket; P_{th} is blanket thermal power; P_{net} is output electrical power of the hybrid system. Tokamak on equal component D–T fuel was presented as an example in Option 5.

Efficiency of a system based on a tokamak with D–D fuel (Option 1) is unacceptably low at low β values. Neutron fluxes are at the acceptable level in Options 2–4. In Option 3, the η^* value slightly exceeds the value for a purely thermonuclear reactor. It is supposed in Option 4 that 50 % of the produced tritium is not burned, and therefore, could be used in another reactor of comparable power with the D–T fuel neutron source (Option 5). Efficiency of the latter is significantly higher, and the size is smaller. Such fuel cycle scheme could be more attractive from the economic point of view, than a scheme with full tritium utilization in the catalyzed D–D cycle.

Parameters of deuterium hybrid systems

Parameters	Tokamak					Open trap (linear)				
	[13, 14]					[15]		[16]		
	Option									
	1	2	3	4[1]	5[2]	6	7	8[3]	9[3]	10[3]
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	–	–	–	–	–
β	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]	4.9[4]	4.1[4]	5.8[4]	11.2[4]
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	1	0.084	0.23	0.53
τ_E, s	2.2	4.1	6.0	5.7	1.16	6.3[5]	3.2[5]	0.21[5]	0.58[5]	0.74[5]
n_D, 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	8	12	50	50
P_{fus}, MW	10	72	170	110	50	20	19	9	76	520
P_n, MW	6.6	48	112	65	40	13	12.5	6	50	340
P_{n14}/P_{fus}	0.56	0.56	0.56	0.44	0.8	0.57	0.57	0.57	0.57	0.57
P_b/P_{fus}	0.015	0.16	0.22	0.14	0.25	0.43	0.36	0.1	0.085	0.076
P_s/P_{fus}	0.38	0.019	0.029	0.049	0.003	0.014	0.027	0.037	0.18	0.68
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	–	15	15	12	10	9.5	15	–	–	–
P_{th}, MW	–	3050	5590	4750	930	430	400	910	3420	14 500
P_{net}, MW	–	360	850	550	250	100	94	45	380	2600
η^*, %	–	12	15	12	27	23	23.5	5	11	18

[1] 50 % of the tritium produced in the D–D reaction burns.
[2] D–T fuel (50 % : 50 %).
[3] Beam heating (NBI), significant non-Maxwellian population.
[4] Central section.
[5] Plasma thermal components required value.

High efficiency of systems based on the open trap (Options 6 and 7) is associated with a relatively high $Q = 1$. For this purpose, plasma confinement time in the open trap should be really high ($\tau_E \sim 5$ s); therefore, the need for complicating the system to reduce losses in plasma through the ends of an open trap is obvious. Let us note that significant progress was achieved in recent experiments on open traps [22].

Population of high-energy ions is generated in plasma when heated by the NBI (Options 8–10). Requirements to confinement time are reduced. With an increase in the energy of the injection energy E_0, Q also increases. On the other hand, plasma size and density should stay significant to ensure efficient capture of the beam. Therefore, power of such a system and neutron load on the walls could turn out to be unacceptably high.

The coils' limiting magnetic field is restricted by capabilities of existing superconductors. Maximum field on the superconductor surface does not exceed 20 T in all Options.

Parameters of a system based on the open trap (Options 6, 7) appear to be the most attractive. Decrease in neutron fluxes to the wall in the open trap is possible, when the wall is moved away from plasma. Power growth in a linear system is relatively easily achieved by proportional increase in its length. Heat release density level in the blanket makes it possible to consider cooling of the blanket with a gas coolant, for example, helium [23].

Single-circuit scheme based on a closed gas turbine installation could be used in heat conversion [15], its parameters are as follows: helium temperature at the compressor inlet $T_1 = 320$ K, compressor outlet temperature $T_2 = 422$ K, pressure increase level $p_2/p_1 = 2$, turbine inlet temperature $T_3 = 1000$ K, turbine outlet temperature $T_4 = 758$ K, heat recuperation degree 0.95, temperature difference in the heat exchanger-recuperator $\Delta T = 15$ K, specific heat supplied per cycle $q_{in} = 134.5$ kJ/kg, recuperated heat $q_r = 165.3$ kJ/kg, cycle work $l_c =$ $= 462.8$ kJ/kg. For given compressor efficiency $\eta_{comp} = 0.8$ and turbine efficiency $\eta_{turb} = 0.9$, the power installation efficiency is $\eta_e = 0.35$.

Conclusion. Potential possibility to use the D–D reaction is important from the prospects for large-scale energy production based on hybrid systems. Essential advantage of a hybrid system with neutron source based on the D–D reaction is that there is no need to produce tritium in the blanket. Tritium is produced only in the D–D reaction in plasma. This significantly simplifies a number of problems associated with the tritium cycle technology in hybrid reactors.

Using a gas-cooled blanket with solid components allows implementation of a single-circuit scheme with closed gas turbine installation similar to high-temperature gas-cooled reactors with spherical heat-generating elements.

It should be noted that high power in plasma heating is a characteristic feature of thermonuclear systems. Therefore, it is advisable to create a neutron source with $Q \approx 1$ or more; otherwise the hybrid system efficiency would be lower compared to nuclear reactors, for example.

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