TOKAMAK as a nuclear fuel breeder

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Due to the limited natural reserves of fissile isotopes of uranium and thorium ores (0.7% in uranium ore) with the rise of the world energy consumption the need for "artificial" production of nuclear fuel due to 14.1 MeV fusion neutrons impact [1,2] may be arisen by the end of the 21st century. The level of 1GW i.e. 4.4·1020 n/s for industrial production of fuel by estimations of authors of [1] is required. An important characteristic is the full neutron yield, the dimensions of the source are not essential. It can be a pulsed operation of the neutron source, but the important thing is to keep this figure in average during the year. Now it is assumed the fast reactors be the main path for fuel breeder system development. However, there are certain difficulties on this way [2].

If we consider the existing fusion system, currently a large size tokamak (R = 10 m, a = 2.5 m), working only in the ohmic heating, it is capable to generate continuously with 0.8 duty cycle neutron flux 2·1017 n/s on a 50% mixture of deuterium and tritium. The neutron pulse duration of such a device is determined by the volt-seconds supply in the magnetic circuit and is estimated to be equal to 30 seconds. It is not necessary to achieve ignition in the plasma. It is possible to work in the subcritical regime.

We consider the first step should use maximally the proven and working experience: circle plasma cross section, iron magnetic core (for the minimal energy consumption to the plasma heating), the plasma current, electron density and equilibrium control only, graphite limiters. The tokamak device should not be the subject of research in the path to the first nuclear fuel breeder. We should use the results of more than a half of a century work. This device will yield 1 – 1.5 kg of U235 during the year. The steady state plasma current is not required for this tokamak-breeder – it will work as a transformer with the frequency scale of 0.01 Hz. Calculations show that in the second stage 100 MW of additional power to the ion component would result in an increase of neutron flux by more than 200 times, accordingly increasing the fuel yield. The use of ion or electron cyclotron heating will not lead to a radical modernization of the device: additional heat sources are located outside the tritium area. The wall thermal load (auxiliary heating, Joule heating and alpha particles heat transport) will not exceed 0.15 MW/m2 and can be utilized by the radiation without the camera design changing with the camera wall temperature about 400oC.

References

1. Velikhov E.P., Kovalchuk M.V., Azizov E.A. et al., Atomnaya energiya [in Russian], 1996, 114, 160-165.
2. Tsibulskij V.F., Andrianova E.A., Davidenko V.D. et al., VANT, Termoyadernya sintez [in Russian] 2016, 4, in print.