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Optimal configuration of a tokamak fusion system with breeding blanket based on ITER TBM

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System parameters and the optimal radial build of a tokamak fusion system with a normal aspect ratio were found through the coupled analysis of a tokamak system and neutron transport. Neutron impact on shielding and tritium breeding capability are self-consistently incorporated together with plasma physics and engineering constraints in determining the radial builds. The plasma physics and engineering constraints moderately extrapolated from the constraints adopted in the design of International Thermonuclear Experimental Reactor (ITER) were used. In a tokamak fusion system with a normal aspect ratio, configuration with only an outboard breeding blanket does not satisfy requirement on tritium self-sufficiency. The optimum system size to produce a given fusion power was determined by the requirements on the shielding, tritium breeding and the magnetic field at the toroidal field (TF) coil. Thickness of the outboard and inboard blanket were determined to allow optimal system size. With a confinement enhancement factor $H = 1.3$, $Q > 30$ was possible for fusion power greater than 2,000 MW with an aspect ratio of $A = 3.0$; however, $Q > 30$ was impossible with an aspect ratio of $A = 4.0$. The tritium breeding capability of blanket concepts proposed for testing in the International Thermonuclear Experimental Reactor (ITER) was evaluated by varying the outboard/inboard blanket thickness and the degree of lithium-6 (Li-6) enrichment. Cases with a smaller aspect ratio exhibited better performance since the number of fusion neutrons that contributed to tritium breeding were larger than the case with a larger aspect ratio. Among the blanket concepts, a helium (He)-cooled solid breeder (HCSB) concept showed the best tritium breeding capability and thus allowed for a smaller system size.

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