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Optimization Study of Normal Conductor Tokamak for Commercial Neutron Source

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The fast neutrons produced by DT fusion are able to burn the long-lived biologically hazardous transuranics (TRUs) in the spent fuel discharged from fission light water reactors more efficiently than other sources like fast fission reactors. Although a concept design of such a system employing a conventional tokamak (like ITER) with super-conducting coils as the fusion core was proposed, the long pulse operation for more than several months is highly challenging. If a tokamak with demountable copper toroidal field coils is used, replacement of in-vessel components would be possible more frequently without reducing the plant availability and then technical difficulties would be mitigated. The resistive loss in the copper coils, however, is a great concern for its feasibility.
The optimum conceptual design of tokamak with normal conductor coils was studied for minimizing the circulating power and the cost for producing a given neutron flux (cost of neutrons, CON) in the range of plasma aspect ratio A = 1.75-3 by using a system code, PEC. The plasma performance was assumed to be moderate ones; normalized beta ˜ 3-4 in A = 2-3 and H_98y2 = 1. It is also assumed that q* ≥ 2.5, considering the operation regimes of ST and of the conventional tokamak. We fix the nominal fusion power to 180 MW, the thermal power output, mainly generated in the blanket, to 3 GW, and the surface area of the blanket located on the low-field-side to ˜ 126 m^2. The fusion power is ramped up from 100 MW to 180 MW during the burn cycle.
The results are as follows. The circulating power decreases with A up to A ˜ 2.5. This is due mainly to reduction of the toroidal field coil resistance by increase in the center post radius. On the other hand, the capital cost (construction cost) increases with A. As a result, CON has its minimum around A = 2.25, namely, between ST and the conventional tokamak. At A = 2.25, the circulating power is 55% of the gross power in average during the cycle. The plasma major radius is 2.44 m, the toroidal field is 3.1 T, the plasma current is 9.4 MA, and the plasma energy gain is 1.04 at the end of the cycle.

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