Seminar

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Institute for Plasma Research

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Title: An Engineering Study of Concepts for Heat Extraction and Power Conversion from Moderate Sized Tokamak Fusion Reactors

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Venue: Board Room, IPR

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Abstract

Controlled thermonuclear fusion is a futuristic source of energy. Early experiments using a magnetically confined hot and dense deuterium (D) - tritium (T) plasma in the tokamak devices have yielded positive results with fusion power of the order of 10-20 MW for a few seconds. To extract the thermal power produced by fusion, one must trap the 14 MeV neutrons produced by the D-T reaction in a blanket surrounding the plasma. The extraction of heat is to be done by a coolant that runs in a primary loop, which is used to transfer the heat to secondary loop for power conversion. Recent technological developments are enabling design of moderate sized ($R \sim$3-4 m) tokamak fusion reactors. Such reactors will be the most important step for building full scale fusion power plants ($R \sim 7$ m). However, very little attention has been paid to the considerations of the blanket and the problem of heat extraction and power conversion for moderate sized reactors. Such reactors are important for providing a solid technical basis for building large net-electricity producing reactors. However, they have unique challenges: (a) the space available for handling the blanket modules is highly constrained, but design must allow ease in replacement/ repair, (b) power deposition profile and small module size make it challenging to extract heat at high temperatures (c) the reactor operates in the pulsed mode but the electricity generation needs to be continuous with $Q_E \sim 0.5$ (ratio of gross electric power produced to the electric power consumed).

Using standard equations from tokamak physics, engineering constraints, 2D and 3D graphics tools, a consistent radial and vertical build of a 200 MW reactor with major radius $R_0 = 3$ m and minor radius a = 1.1 m has been developed. Three horizontal ports and one vertical port have been considered for PF coil locations optimization. It is shown that the inboard radial build of the reactor with TF(CS) current density 33(25) A/mm^2 allows a shield thickness of 310 mm, resulting in a reactor lifetime of 0.74 full power years (considering the dose to magnet insulation). Feasibility of horizontal withdrawal and vertical lift of the module through the vertical port are discussed.

A helium-cooled solid breeder blanket concept that has radially stacked breeder/multiplier zones has been developed. The zones that are away from the plasma receive lesser heat. Thus, the challenge is to design the helium flow paths so that each of them has nearly uniform outlet

temperature. This has been accomplished by creating 4-circuits using the 7 cooling plates and the first wall. The 'C-shaped' path, connecting 7 cooling plates and the first-wall in a paring fashion shows a nearly uniform outlet temperature of about 500°C from each pair. The temperature evolution of the blanket is calculated by solving 1D multi-region heat diffusion equation using an 'Effective Loss Channel Model' (ELCM). The results show that for a pulse length of 3000 s and a dwell time of 1000 s, the blanket reaches steady state within the temperature limits of the materials.

For steady state power generation, the thermal power (290 MW) generated by the fusion reactor is transferred to an energy storage system of HITEC molten salt (tank volume ∼7500 m^3) with 95% efficiency. This is then used to drive a power conversion cycle of either a reheat Rankine cycle or a supercritical CO_2 Brayron cycle. For the Rankine cycle, the best efficiency found is 41.7% (electrical power ∼90 MW) for a steam generator pressure of 69 bar and super-heated steam temperature of 460°C. For the case of supercritical-CO_2 Brayton cycle, the temperature at the inlet of high pressure turbine (HPT) is 450°C and the pressure is 200 bar. This gives an efficiency of 42.75%