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Plasma Heating and Current Drive for Fusion Reactors

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ITER (in Latin "the way") is designed to demonstrate the scientific and technological feasibility of fusion energy. Fusion is the process by which two light atomic nuclei combine to form a heavier one and thus release energy. In the fusion process two isotopes of hydrogen - deuterium and tritium - fuse together to form a helium atom and a neutron. Thus fusion could provide large scale energy production without greenhouse effects; essentially limitless fuel would be available all over the world. The principal goals of ITER are to generate 500 megawatts of fusion power for periods of 300 to 500 seconds with a fusion power multiplication factor, Q, of at least 10. $Q \geq 10$ (input power 50 MW / output power 500 MW). In a Tokamak the definition of the functionalities and requirements for the Plasma Heating and Current Drive are relevant in the determination of the overall plant efficiency, the operation cost of the plant and the plant availability. This paper summarise these functionalities and requirements in perspective of the systems under construction in ITER. It discusses the further steps necessary to meet those requirements. Approximately one half of the total heating will be provided by two Neutral Beam injection systems at with energy of 1 MeV and a beam power of 16 MW into the plasma. For ITER specific test facility is being build in order to develop and test the Neutral Beam injectors. Remote handling maintenance scheme for the NB systems, critical during the nuclear phase of the project, will be developed. In addition the paper will give an overview over the general status of ITER.