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Status and evolution of the STEP nuclear fusion power plant programme

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Within the STEP (Spherical Tokamak for Energy Production) programme, the UK is investing significant funds into the design and subsequent building of an electricity producing fusion power plant. A site has been selected and early preparation of this site is commencing. Meanwhile the conceptual design of the power plant is advancing at pace. A Spherical Tokamak has been chosen as the central component, as it allows for a more compact design, due to the ability to operate at larger normalized plasma pressure. This opens the way to increased autogenerated 'bootstrap' current and reduced current drive power requirement for steady state operation. In this presentation the main design driving tensions between plasma performance and engineering feasibility will be described. As the conceptual design is maturing, engineering considerations have enforced certain modifications to the plasma equilibrium, while plasma stability considerations have enforced reevaluation of elements of the machine design. Fine tuning of the machine design is continuing as the plasma modelling capability is evolving.

To be able to produce net electricity from a fusion plant, the fusion gain Q (Fusion power divided by power added from external sources) has to be high enough. A main driver for Q is the power required to maintain the plasma in steady state. In steady state, this power serves two distinct purposes. It must drive the plasma current not generated as bootstrap current and it has to maintain the plasma stored energy. The current should be driven at the appropriate minor radius, with significant current drive at large minor radii. This current drive requirement dominates in terms of Q and hence, the design has focused on maximising the current drive. Electron Cyclotron Current Drive (ECCD) has been chosen for this purpose which in turn leads to the requirement of a sufficiently high toroidal field (~3-3.5T at the plasma centre) and associated ECCD frequency to allow the electron cyclotron wave to propagate at the high densities needed for efficient fusion power production.

Generating the large toroidal field requires very significant total current (in ampere*turns) to flow in the toroidal field coils. The losses associated with this current has led to the choice of superconducting toroidal field coils. The STEP design includes a limited central solenoid to assure plasma initiation and early plasma ramp up, which also has to be accommodated withing the central solenoid. As the design of the superconducting toroidal field coils has progressed, neutronic heating calculations have shown that added neutron shielding is required to prevent loss of superconductivity. This shielding requirement has led to the exploration of design solutions with a larger central column. Assuming the aspect ratio is maintained, this leads to a proportionally larger machine. Alternatively, an increase in aspect ratio may be accepted, with its consequence on allowable elongation, normalized beta and bootstrap current.

One of the significant STEP challenges is associated with the need for tritium self-sufficiency during steady state operation. This requires a breeding ratio larger than unity. If the size of the central column becomes too big, an excessive number of neutrons will be intercepted by this column, leading to the need for breeding blankets in the column. Given the low shielding efficiency of breeding blankets, such inboard breeding would result in a prohibitively wide central column. Avoiding the need for inboard breeding drives the maximum central column width, while the shielding of the coils and the required toroidal field drives the minimum width. The latest iteration of the design includes a modest increase in column width to accommodate more shielding and a modest decrease in toroidal field. A more compact design of the coils in the central column or a better tolerance to neutron irradiation, could lead to the possibility of designing a smaller machine.

All the plasma additional heating and alpha power must be exhausted from the machine. To distribute this power as evenly as possible over the plasma facing components impurities need to be injected into the core plasma and the divertor region has to operate in a detached regime. To improve the power handling a double null design with advanced divertor leg configurations has been chosen. The power which the divertor can handle, while remaining detached, determines the required core radiation and efforts are being undertaken to improve this power handling to reduce the needed core radiation.

The noninductive operation and the associated high pressure and high elongation operation means that active stabilisation of the vertical instability and of the resistive wall mode must be possible. This leads to requirements on conductivity of structures surrounding the plasma, plasma elongation, plasma current profile and plasma pressure. The conductive structures and the other equipment needed to operate the machine will impact the volume available for the tritium breeding blankets. If this volume becomes too small the requirement of a design which is self-sufficient in tritium production becomes compromised.

A design that resolves these tensions will require a plasma that resides in a sufficiently large 'safe' parameter region. A robust control system is needed to guide the plasma to the desired operational point, with a more performant control system allowing operation at higher performance facilitating the resolution of the tensions described.

References

[1] Philosophical Transactions, volume 382, issue 2280 – Special issue dedicated to the STEP project and references therein.