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Spherical tokamaks: Present status and role in the development of fusion power

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Abstract

The spherical tokamak (ST) has triggered a fast-growing activity world-wide on account of its promising potential and its strong physics overlap with conventional tokamaks, including ITER. There has long been a view that it could have a key role as a component test facility, to complement ITER, IFMIF, and DEMO, and there are also interesting possibilities as an option for the fusion power source of an electricity plant. The experimental base is now considerably advanced from the time when these ideas were first raised, with the advent of the MA scale machines MAST and NSTX, and a growing theoretical and modelling base. Here, we describe the status of development on the key engineering and physics issues of the ST, considering in particular application to a component test facility and input to an accelerated programme towards deployed fusion power plants, the so-called "fast track".
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1. Introduction

One present streamlined approach to fusion power [1,2] is to construct and operate ITER and in parallel construct and operate IFMIF [3] in order to provide information for an early decision on a DEMO. The precise objectives and scope of the DEMO stage are not yet fully defined, but the overall aim is to allow an early decision to construct the first commercial fusion power plant. This is done by demonstrating that the physics, materials, components and systems are viable and reliable. All of this can be achieved with the right programmes in the ITER, IFMIF and DEMO stages. It is, however, possible that the programme could be further accelerated, or performed with reduced risk of delay or technical problems, by incorporating a component test facility (CTF) to test items in between the small materials samples in IFMIF and the full size components of the present DEMO concepts. The experimental basis has advanced to the stage where an ST can be seriously considered for a CTF, since with the appearance of MAST [4,5] and NSTX [6], the ST is now on a par in many respects (plasma size, current and performance)

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with existing medium-sized conventional-aspect-ratio tokamaks, and there are now many STs worldwide. The design of ITER was based on these medium-sized tokamaks as well as the large devices JET, JT-60U and TFTR. As will be seen below the size step to a compact CTF is much smaller than the size step to ITER.

The spherical tokamak displays tokamak properties (most of the physics is common with ITER), yet is very compact, given the small aspect ratio ($A = R/a \sim 1.5$). There are a number of potential advantages, such as low magnetic field energy for a given performance, and potential engineering simplifications due essentially to the much smaller volume of the centre column (which although challenging is not superconducting).

This paper outlines the features of a possible compact ST-based CTF, and uses this to illustrate the physics and technology progress and issues in the spherical tokamak. A brief discussion of the ST as a power plant is included at the end.

2. Component test facility

The reliability and availability of a fusion power plant will depend on the behaviour of complex macroscopic components, such as breeding blanket modules in the combined presence of thermal, electromagnetic and neutron-induced effects. The structural materials properties under irradiation will have been tested in IFMIF, and many of the technology integration issues will have been studied in ITER, as a part of the test-blanket programme or in other areas (such as the divertor). DEMO is presently envisaged as proving full scale assemblies for functionality and reliability prior to commercial deployment. There could be a significant benefit from a smaller facility to test intermediate scale subassemblies and prototype elements with mixed materials, joints, cooling and breeding elements at high neutron flux and fluence, and also to test plasma facing components. This is the component test facility (CTF).

Most of the issues and basic requirements for a CTF were developed in the 1980s and 1990s and are summarised in [7]. The spherical tokamak has long been considered attractive for this purpose [8], as it appears that it is possible to make a compact device with a rather low consumption of tritium, yet with a test area of many m^2 (larger STs are also considered [9]). (There is an alternative with even lower tritium consumption, the mirror-based gas dynamic trap [10], which could test materials and smaller components.) For simplicity, only the compact non-breeding ST version is considered here, although there will be many issues common to all CTFs.

2.1. CTF design principles, parameters and engineering approach

Our approach has been to attain the required performance [7] by setting quite demanding targets for some aspects and allowing freedom over others. The goal is a device capable of CW operation, generating the required 14 MeV neutron flux of 1–2 MW/m^2 over an area of $\sim$10 m^2 or more, yet with modest tritium consumption ($\sim$1 kg/year). This has led us to the following design principles:

- Use standard tokamak physics where possible (to reduce extrapolation risk).
- Minimise the size (low capital cost, as well as T consumption).
- Minimise the plasma volume/test area ratio (low T consumption).
- Operate in a strongly driven mode, $Q = 1$ (minimise uncertainty when α-heating appears).
- Adopt a simple design, rapid access (availability, maintainability).

To assist this we have allowed freedom (within reason) on:

- Running costs and power consumption.
- Waste, activation (maintenance needs will constrain, and waste will be minimised).
- The need to breed T (default is no breeding).
- Component lifetime (subject to achieving adequate availability).

The target device is small, to minimise T consumption without breeding (a larger breeding CTF is studied in [9]). For small devices we expect the plasma pressure to be limited by confinement and power exhaust rather than stability. There are several ways to determine the device parameters: e.g. prescribe the neutron flux and deduce the plasma parameters based on confinement scalings, or iterate via other constraints, such as bootstrap fraction, externally driven current. Each of these is based on a number of assumptions, but all point

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