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Design description and performance analyses of the European HCPB test blanket system in ITER feat

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Abstract

The helium cooled pebble bed (HCPB) blanket is one of the two European DEMO blanket concepts proposed for testing in international thermonuclear experimental reactor (ITER). The purpose of the tests is to validate the design principles and the operational feasibility for the demonstration blanket system. This includes the basic support functions like tritium extraction, helium cooling and heat transport, and helium purification. In addition, the basic properties and operating characteristics of the system's materials will be validated. Safety, reliability, maintenance and dismantling will be equally addressed. To assess these qualities, the ITER horizontal ports will be used to provide a relevant fusion plasma and the appropriate nuclear environment. At conclusion of the ITER feat EDA phase (July 2001), a revised design of the HCPB test blanket system have been completed to adapt the previous design (for ITER FDR) to the new operational conditions of ITER and to a new strategy for the blanket testing in this reactor. Design description, performance and safety analyses are presented in this paper.

Keywords: Helium cooled pebble bed (HCPB) blanket; DEMO; ITER

1. Introduction

This paper presents an overview on the work performed on the design and performance analyses of the helium cooled pebble bed (HCPB) test blanket system (TBS) during the international thermonuclear experimental reactor (ITER) feat EDA phase in the years 2000 and 2001. A first design of a HCPB TBS was completed in 1998 [1,2] for the previous version of ITER (ITER FDR). With the modification of the basic machine [3] was necessary to revise the previous design (DDD-98); the work performed by our team in collaboration with the ITER test blanket working group became immediately more than an adaptation to the new boundary conditions of ITER feat. The new operation parameters of the fusion reactor required a basic revision of the objectives and of the test strategy of the ITER blanket test programme [4].

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This revision dictated the most important change introduced in the new design (DDD-01); the HCPB blanket testing programme for ITER-feat is now based on four test objects—the so called test blanket modules (TBMs)—that should be irradiated in the ITER equatorial port. Each TBM is devoted to a family of objectives; an electromagnetic (EM) TBM for measurements of eddy currents and electromagnetic forces in normal operation and during disruption should operate in the first 3 years during the H plasma. The neutronics and tritium production (NT) TBM should be irradiated in the next 3 years (D plasma and low-duty D–T plasma) with the aim to check the tritium breeding capability and to validate neutronic codes and data. The investigation of pebble bed behaviour, thermal creep, cyclic power operation, purged gas flow conditions will be carried out with a third module, namely the thermomechanics (TM) TBM (2 years of tests, low and high-duty D-T plasma). The plant integration (PI) TBM (2 years of test in high-duty D-T plasma) will be dedicated to Integration tests on thermo-mechanics, thermo-hydraulics, tritium and activation products handling and cooling system operation.

An upgraded design description document (Status December 2001) of the HCPB TBM which will include all the work (test strategy, conceptual design, performance analyses including safety) performed up to end 2001, is under preparation [5].

2. Design description

The TBS reproduce a portion of the HCPB breeding blanket with DEMO relevant characteristics connected to the related external systems for heat removal and tritium extraction. Its principal components are: (1) the test blanket module (first wall, breeding zone and structure); (2) the helium coolant system (HCS) (heat transfer and transport); (3) the tritium extraction system (tritium removal, handling and processing) and (4) the coolant purification system (CPS) (helium purification and conditioning).

The envisaged location of the HCPB-TBMs is in the horizontal port number 1 of ITER. This port is shared with the Japanese helium-cooled solid breeder TBM, which will be inserted in the lower compartment of a water-cooled frame serving as support and interface between the TBMs and the ITER shielding blanket. The given port size allows outer dimensions of the modules of 740 mm in poloidal, 1268 mm in toroidal and about 800 mm in radial direction.

The new test strategy calls for four test articles with different design (e.g. bed orientation and thickness, thermohydraulic layout, instrumentation, etc.) in order to achieve the different objectives of the test programme foreseen for each module and to cope with different boundary conditions of the related ITER operation phase. Notwithstanding these necessary differences, the general design, the fabrication technologies and the interface with the fusion device are similar for the four TBMs. The HCPB TBM consists essentially of horizontal layers (with exception of the TM-TBM, in which also vertical beds are foreseen) of ceramic breeder and beryllium pebble beds, separated by cooling plates. The whole stack is encapsulated by a box (see Fig. 1) that is formed by a First/Side Wall shell with integrated U-shaped cooling channels, a robust backplate containing the manifolds and the mechanical attachment system, and two massive caps at the upper and lower ends of the box. The caps are able to withstand the internal mechanical pressure of the beds during normal operation and the over-pressurisation transient in case of an accidental loos of coolant in the box (2 MPa is the max. design value). The stability of the back region of the TBM has recently been improved by two toroidal headers at the back wall of the TBM. The reduced activation ferritic–martensitic steel EUROFER is the structural material.

Helium coolant is fed from the rear side via manifolds, flowing first through the FW shell, and subsequently through the cooling plates in the internal breeder zone. The pebble beds are purged by a forced flow of low-pressure helium provided by the tritium extraction subsystem (TES).

The HCS for the DDD-01 has been strong revised in comparison to DDD-98. The changes