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Requirements for helium cooled pebble bed blanket and R&D activities

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ABSTRACT

This work aims to give an outline of the design requirements of the helium cooled pebble bed (HCPB) blanket and its associated R&D activities. In DEMO fusion reactor the plasma facing components have to fulfill several requirements dictated by safety and process sustainability criteria. In particular the blanket of a fusion reactor shall transfer the heat load coming from the plasma to the cooling system and also provide tritium breeding for the fuel cycle of the machine. KIT has been investigating and developed a helium-cooled blanket for more than three decades: the concept is based on the adoption of separated small lithium orthosilicate (tritium breeder) and beryllium (neutron multiplier) pebble beds, i.e. the HCPB blanket. One of the test blanket modules of ITER will be a HCPB type, aiming to demonstrate the soundness of the concept for the exploitation in future fusion power plants. A discussion is reported also on the development of the design criteria for the blanket to meet the requirements, such as tritium environmental release, also with reference to the TBM.

The selection of materials and components to be used in a unique environment as the Tokamak of a fusion reactor requires dedicated several R&D activities. For instance, the performance of the coolant and the tritium self-sufficiency are key elements for the realization of the HCPB concept. Experimental campaigns have been conducted to select the materials to be used inside the solid breeder blanket and R&D activities have been carried out to support the design. The paper discusses also the program of future developments for the realization of the HCPB concept, also focusing to the specific campaigns necessary to qualify the TBM for its implementation in the ITER machine.

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1. Introduction

The HCPB blanket concept is considered in EU a near term solutions for a first DEMO reactor since 1995 and it is considered in the EU TBM Programme for testing in ITER [1]. In KIT (formerly KfK and FZK) helium concepts have been studied for more than three decades with the selection of a solid breeder and beryllium as neutron multiplier; in early 1990s Dalle Donne proposed a Breeder Out of Tube (BOT) concept for NET. This concept became some years later (1995 during a selection exercise in EU) a reference concept in the EU Breeding Blanket Programme with the name of helium cooled pebble bed (HCPB) concept for DEMONET and TBM [2]. In 2000 this blanket concept was selected as part of the PPCS as reference component for PPCS Model B [3]. The main features of the HCPB concept are the employment of lithiated ceramic breeders (CB) and beryllium as neutron multiplier in form of a flat pebble beds. The coolant is helium at a pressure of 8 MPa with temperatures in the range of 300–500 °C. The velocities of helium up to 80 m/s in the FW are required to obtain the right heat transfer coefficient. An independent helium loop at relatively low pressure, in the range of few bar, provides the purging of the tritium from the pebble beds. Since the purge flow velocity is very low practically no heat removal is provided by the helium circulating in the loop.

2. Design requirements

2.1. Geometry

The blanket system shall be adapted on a reactor design subdivided toroidally on 16 sectors (16 TF coils). The Vertical Maintenance System (VMS) is based on a reactor design characterized by large vertical ports for extraction of large blanket portions (segments). In fact each of the blanket sectors is made by 5 segments: three for the Outboard (OB) and two for the Inboard (IB). The primary option for the design of the segment is the multi module segment (MMS) concept; a common back supporting

structure (BBS, which contains the main manifolds) with 10 modules attached for each segment. The number, i.e. the dimensions of the module, is determined by a trade-off among different considerations: small modules are favorable to reduce thermal stresses and to reduce EM forces at the single attachments. On the contrary drawback could be possible in the Tritium Breeder Ratio (TBR) as the relative amount of steel can increase increasing the number of modules. The VMS foreseen that the single module are attached to a manifold segment that ensure the support of all the modules. The poloidal manifold system ensures the routing of the Helium coolant and of the purge lines to the upper part where collectors go in the vertical port. Weight and dimension of segments are also dictating the design of the Remote Handling (RH) system.

2.2. Thermal loads

Three main heat sources have to be considered in the plasma design:

(1) the volumetric nuclear heating caused by neutrons;
(2) the surface heating from plasma core radiation and charged particles in SOL and
(3) the plasma wall interaction.

The nuclear heating coming from neutrons constitutes the main source of heat for the blanket: we find less than 2 MW/m^2 [4], with a peak in the outboard equatorial region for it. The volumetric power distribution is strongly depending on the materials and usually it is larger in ceramic breeder (with max value of ~20 MW/m^3 [5]) and W, than in steel and Be. Plasma core radiation (mainly synchrotron radiation and Bremsstrahlung) constitutes a surface source for the blanket FW with typical design values of ~0.5 MW/m^2 [6]. The third load related to the FW interaction with the plasma flux particles depends largely on the plasma control system (in DEMO based on future ITER experience), and therefore not yet well-defined in design requirements. The complete definition of the heat loads for the blanket requires a future more accurate definition of the requirements for each different heat source, taking into account also transient and local spots.

2.3. Tritium production

The achievement of a sufficient tritium breeding ratio (TBR) is essential for the T self-sufficiency. T is produced mainly in the breeder material, extracted and delivered to the fuel cycle. Furthermore, tritium is also produced in Be pebbles neutrons multiplier but with significantly lower generation rate. This tritium source should be considered mainly for safety reason, checking weather allowed inventories are not exceeded in the long term due to trapping issues in Be pebbles. To assess the Tritium Self-Sufficiency a 3D Neutron Analysis (3D-NA) of the reactor is required with a sufficient detail in the blanket zone. Examples of this assessment can be found in [7] for the PPCS, and in [8] for DEMO-2007. Important output value of this 3D-NA is the TBR considered as target of the calculation. In general this is a number >1 that has to account for possible reduction mainly due to uncertainties in 3D-NA calculations and reduction of blanket coverage brought by ports for diagnostic and plasma control. A target value of 1.12 is proposed for the present assessment and 3D-NA analyses campaign are currently ongoing in KIT for validating the present design.

2.4. Shielding

Some of the components of the DEMO reactor required to be strong shielded by high neutron fluxes to ensure the integrity

during the lifecycle (e.g. magnet system). The blanket system contributes to the level of neutron shielding required. A reviewed list [9] has been issued on the requirements which are applicable for the magnet system to be followed for the verification of the neutron shielding capability in DEMO. The most stringent criteria is the peak nuclear heating in magnets winding pack that has to be lower than 50 W/m^3. Additional requirements of blanket shielding capability could be concern the neutron damage and thermal protection of in-vessel devices as attachments or supporting structures.

2.5. Safety

2.5.1. Tritium transport

The determination of the T transport in blanket components is a crucial issue for the safety performance of all blanket concepts. In helium cooled systems the T can permeate in the Blanket from the breeding side (PbLi loop in the HCLL or helium purging system in the HCPB) to the high pressure helium coolant side. The permeation can be considerable as driven by the large interface surface, the high temperature of steel at the interface and the T partial pressure that can be formed in the breeding part. The tritium permeating in the coolant systems necessitates to be reduced in order to avoid large amount of tritium to reach the steam generator and permeate to the power generation loop, which uses water as working fluid. To keep the permeation rate in the steam generator lower than the safety limits, a strict control of the partial pressure in the primary helium loop is required. A coolant purification system (CPS) shall be foreseen to treat in by-pass configuration some small fraction of the coolant (~0.1% [10]) in order to control the permeation rate. Parasitic permeation through the cooling plates into the helium coolant system can drive safety relevant quantities of T to the environment; operational release should be minimized and lower than the allowable limits, i.e. 20 Ci/day [11] or 1 g/year [12]. In particular these limits are considered for the release (permeation and/or leakages) in the steam generator between the helium primary loop and the Rankine cycle for the power generation system.

2.5.2. Tritium inventory

A T inventory limit for the whole vacuum vessel lower than few kg shall be ensured to avoid critical releases in case of accidents (in ITER this limit is 1 kg [13]). Following this requirement, the scope of the design is to minimize the inventory in different part of the blanket systems, mainly in Be and Eurofer material.

2.5.3. LOCA

The blanket boxes are the primary confinement of the tritium and breeding material, therefore their integrity is paramount from the safety point of view, both during normal operation and accidental condition. The coolant used for the blanket cooling system is helium at 8 MPa, while the boxes operating pressure is 0.4 MPa. In case of internal coolant leak, the box shall be designed to withstand the over-pressurization. A rupture disk would limit the peak of pressure in the box during a LOCA, allowing to a significant lower design pressure for the box itself. This option is considered in the current design activities of the blanket.

2.5.4. Cooling circuit redundancy

It is requested that the blanket is cooled by two independent coolant loops in order

• to remove the after-heat in case of a failure of one of the two loops;
• to mitigate the transient during shut-down in case a failure of one of the two loops.