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Development of the IFMIF Tritium Release Test Module in the EVEDA phase

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ABSTRACT

This paper presents the engineering design of the IFMIF (International Fusion Materials Irradiation Facility) Tritium Release Test Module (TRTM). The objectives of the TRTM are: (i) in-situ measurements of the tritium released from lithium ceramics and beryllium pebble beds during irradiation, (ii) studying the chemical compatibility between lithium ceramics and structural materials under irradiation, and (iii) performing post irradiation examinations for the irradiated materials. The TRTM has eight rigs which are arranged in two rows (2×4) perpendicular to the beam axis and enclosed by a structural container. Each rig includes one capsule that contains lithium ceramic or beryllium pebbles for irradiation. Neutrons reflectors are implemented at different locations to reflect the scattered neutrons back to the active region aiming to improve the tritium production. The TRTM is required to provide irradiation temperature range of 400–900 °C for the capsules filled with lithium ceramics and 300–700 °C for the ones packed with beryllium. The engineering design of the TRTM components such as container, rigs, capsules, pebble beds, neutrons reflectors, and purge gas and coolant tubes are presented. In addition a test matrix for the irradiation campaign is proposed.

1. Introduction

The mission of the International Fusion Materials Irradiation Facility (IFMIF) is to test the fusion materials in irradiation conditions similar to those of the DEMO fusion reactor by providing an intense neutron source to produce neutrons with relevant energy spectrum. Next to the neutron source in the high flux region, the irradiation of the fusion structural materials will be performed in the High Flux Test Module (HFTM) [1]. In addition, the Creep Fatigue Test Module (CFTM) and the Tritium Release Test Module (TRTM) are located in the medium flux region. The TRTM is a device that contains solid breeders (lithium ceramics) and beryllium in its capsules under controlled conditions, and with connections to a purge gas system and a tritium measurement system, to measure their tritium release during irradiation. At the TRTM position, the low energy part of the neutron spectrum should have adequate flux intensity to reach tritium breeding rates comparable to those of the DEMO fusion reactor. The HFTM and TRTM have been developed at the Karlsruhe Institute of Technology (KIT) during the Engineering Validation and Engineering Design Activities (EVEDA) phase. Production, inventory, transport, and permeation of tritium are important design issues for the fusion breeding blankets because they are connected to the tritium self-sufficiency and safety of the fusion reactor. In this sense, the TRTM has a special importance for the development of the fusion breeding blankets. This paper presents the engineering design of the TRTM main components such as container, rigs, capsules, pebble beds, neutrons reflectors, and purge gas and coolant tubes. In addition a test matrix for the irradiation campaign is proposed. The objectives of the TRTM are: (i) in-situ measurements of tritium released from lithium ceramics and beryllium pebble beds during irradiation, (ii) studying the chemical compatibility between lithium ceramics and structural materials under irradiation, and (iii) performing post irradiation examinations (e.g. mechanical tests, metallurgical analyses, and assessment of residual tritium) for the irradiated materials. The requirements of the TRTM are presented in the following section.

2. Requirements of the TRTM

2.1. Scientific mission and irradiation

The TRTM will be positioned behind the CFTM in the medium flux region. In order to transmit the neutrons flux and spectrum effectively to the specimens, the TRTM is required to exclude any relevant amounts of neutron-absorbing materials, and reduce its structural mass in the path between the neutron source and the specimens. The TRTM may install neutron reflectors on its boundaries (not facing the neutron source) to minimize the neutron losses and flux gradients in the capsules. Additionally, moderator and/or spectrum shifter will be attached (if necessary) on the boundary facing the neutron source to initiate a neutron energy spectrum suitable for the TRTM needs. The extent of the

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irradiation field is mainly imposed by the beam footprint (20 cm width and 5 cm height) and intensity (energy and current) of the neutron source. The capsule's volume was selected to allow meaningful measurements of tritium with the available measurement methods. Two solid breeders (lithium orthosilicate and lithium metatitanate) and beryllium are foreseen for irradiation in the TRTM.

The irradiation temperatures will comply with the temperature ranges proposed for the Helium Cooled Pebble Bed (HCPB) blanket [2]. Hence, the TRTM is dedicated for irradiation temperature range of 300–900 °C (details are given later in the test matrix). Due to the strong temperature dependence of the tritium diffusion coefficient, the capsule temperature gradient must be minimized, accordingly the diffusion coefficient at the lowest temperature should not differ by more than 50% of its value at the highest temperature. When the irradiation stops, the capsules should be rapidly cooled to a temperature (<200 °C in 30 min), where the irradiation defects can be considered frozen and the diffusion is limited. The chemical environment should not influence the properties or behaviour of the tested materials which will be in direct contact with other materials such as capsule structure and thermocouples during irradiation. These contacting materials should have good chemical compatibility with the tested materials under the irradiation conditions and high temperatures. The TRTM should enable online measuring of the tritium released to the purge gas, sweeping the capsules with the purge gas, and transporting it to the analysis system. The TRTM must provide sufficient temperature measurements in the capsules to facilitate a meaningful analysis. In addition, each irradiation capsule will be equipped with an electrical heater.

2.2. Facility integration

The TRTM lifetime must encompass the time of the irradiation campaign that is foreseen to be eleven months long. Also, the TRTM should cope with running some continuous campaigns to reach accumulated damage and lithium burn-up relevant to the fusion DEMO reactor. The irradiation time and temperatures will be a big challenge for the TRTM components and instruments (e.g. TCs, heaters). Accurate functioning of these instruments with acceptable uncertainty must be maintained during the irradiation period. The TRTM design should allow its assembly and disassembly in hot cells under the condition that some or all parts will be emitting radiation hazardous to the working staff, therefore, complicated procedures and extensive equipments should be avoided wherever possible. Transportation of the TRTM between the hot cells and the test cell by means of remote handling should be feasible. The TRTM should feature provisions for attachments and withstand certain loads exerted during its transportation.

2.3. Safety measures

The TRTM must be able to safely contain and handle its contents, especially the materials that may introduce chemical or radioactive hazards, up to a temperature above the irradiation temperature by a safety margin of 100 °C. After being detached from the supply lines, the remote handling should close the coolant and purge gas lines. Allowable leakage rate of tritium from the TRTM to the cooling system must be specified.

3. TRTM engineering design

3.1. Rigs and capsules

The main component of the TRTM (shown in Fig. 1) is the container which encloses, in its middle compartment, eight rigs arranged in two rows (2×4) perpendicular to the beam axis. The rigs occupy a zone of 175 mm in width horizontally while the total width of the TRTM is 1800 mm. Each rig contains one capsule that is packed with pebbles of the tested material forming a pebble bed. At the capsule top and bottom, upper and lower filters are used to constrain the pebbles while allowing the purge gas to flow. The purge gas is fed through the inlet tube (along the capsule axis) from the top to the bottom, and then it flows upward throughout the pebble bed and exits through the outlet tube, see Fig. 2. Each capsule has separate purge gas inlet/outlet tubes with inner diameter of 3 mm and outer diameter of 5 mm. The tritium produced by the irradiated materials in the capsules will be swept by the purge gas flowing through the capsules and will be transferred to the analyzing system located in the test facility utility room. The pebble bed has an annular volume with inner diameter of 5 mm, outer diameter of 19 mm, and height of 73 mm. For an optimal usage of the irradiation space, square cross section for the capsule was first proposed because it allows a perfect alignment of the capsules leaving no intermediate spaces. However, a circular cross section was selected for the capsules because: (i) less stresses and more reliability compared to the square cross section, (ii) uniform heat distribution, (iii) easier modelling and analysis, and (iv) easy manufacturing from a tube.

The capsule is surrounded by a cylinder (heater-can) which hosts the heater. The inner gap between the capsule and heater-can is filled with stagnant purge gas to create a temperature gradient between the capsule and the heater-can, and to avoid any differential pressure across the capsule wall. There are radial ribs on the outer surface of the capsule and heater-can to guide their axial alignment. A spiral electrical heater is wound around the heater-can to balance the nuclear heating aiming to achieve uniform temperature distribution in the pebble bed. It is required to minimize the radial and axial temperature gradients across the pebble bed because they affect the tritium diffusion coefficient. Mineral-insulated metal-sheathed heater wires fabricated by Thermocoax were selected. The heater-can is surrounded by an outer gap filled with the purge gas through a so-called breathing hole located at its bottom; therefore a temperature gradient is introduced between the heater-can and the rig cylinder. The capsule thermal management is required to enable the wide range of irradiation temperatures (300–900 °C), which is very challenging for materials especially at the upper end, and to keep the temperature spread over the pebble bed within an acceptable range, in order to attribute the measured tritium release properties as good as possible to a certain temperature.

The rig has a cylindrical structure formed by three parts (rig cylinder, head, and foot) welded together. The rig cylinder surrounds the heater-can at the middle while the rig head encloses the upper neutron reflector and the rig foot encloses the lower neutron reflector. The volumes directly above and below the capsules are used to place the upper and lower neutron reflectors. In addition, three internal neutron reflectors (having cylindrical shape with a diameter of 17 mm and a length of 450 mm, see Fig. 3) are placed between the rigs in order to smooth out the neutron spectrum. As the tritium release is dependent on the temperature, it is important to closely monitor the temperature field inside the capsules. Accordingly, each capsule is instrumented with three thermocouples (TCs), Type-K, to measure its internal temperatures. The TCs are inserted into the pebble bed at equal azimuthal angels and in the axial direction of the capsule. The TCs' sheath material should be compatible with the tested pebbles under the irradiation conditions. The TCs readings will be the input to the control of the capsule heater. The TCs and heater wires are brazed to a feed-through (FT) plug that is welded to the sealing cap which connects the rig cylinder with the rig head, see Fig. 2. The purge