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ON THE STRATEGY AND REQUIREMENTS FOR NEUTRONICS TESTING IN ITER

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Neutronics testing is among the several types of fusion technology testing scheduled to be performed in ITER. The three ports assigned for testing will test several blanket concepts proposed by the various parties with test blanket modules (TBM) that utilize different breeders and coolants. Nevertheless, neutronics issues to be resolved in ITER-TBM are generic in nature and are important to each TBM type. Dedicated neutronics tests specifically address the accuracy involved in predicting key neutronics parameters such as tritium production rate, TPR, volumetric heating rate, induced activation and decay heat, and radiation damage to the reactor components. In this paper, we address some strategies for performing the neutronics tests. Tritium self-sufficiency cannot be confirmed by testing in ITER, however, the testing can provide valuable information regarding the main parameters needed to assess the feasibility of achieving tritium self-sufficiency. The paper also addresses the operational requirement (i.e. flux and fluence) as well as the geometrical requirement of the test module (i.e. minimum size) in order to have meaningful and useful tests. Measured neutronics data require high spatial resolution. This necessitates that the measured quantity be as flat as possible in the innermost locations inside the test module. This requirement has been confirmed in the present work based on results from two-dimensional calculations. The US and Japan solid breeder test blanket modules are placed inside half a port in ITER. The R-θ model used accounts for the presence of the ITER shielding blanket and the surrounding frame of the port.

I. INTRODUCTION

The Thermonuclear Experimental Reactor[1] (ITER) will be the first plasma burning machine that will offer a volumetric D-T neutron source (plasma) representative of the one found in a realistic fusion reactor. ITER will not breed its own tritium during its first phase mission envisioned to be 10 years. However, it will provide three equatorial ports in which ITER partners could perform various types of testing on their selected tritium-producing blanket concepts to be deployed in a DEMO or power producing reactor. The operating conditions of ITER, however, are not prototypical to DEMO conditions. Testing conditions are limited to a neutron wall load of 0.78 MW/m^2, a surface heat flux of 0.25 MW/m^2, and a pulse length of 400 s with a duty cycle of 25%. ITER schedule is envisioned as follows: the first 3 years for H-H operation, 4th year for D-D operation, years 5, 6, and 7 for low duty DT operation, and years 8, 9, and 10 for high duty DT operation. The number of burn pulses/year starts as 1 in the 4th year, 750, 1000, 1500 in years 5, 6, and 7, and 2500, 3000, and 3000 in years 8, 9, and 10. As such, the cumulative fluence is limited to ~0.09 MWa/m^2 and may not be adequate to undertake certain class of tests. Since ITER testing parameters are substantially below those of DEMO, engineering scaling and/or alternate strategy are required to obtain meaningful results from testing in ITER. Engineering scaling is a process to develop meaningful tests at experimental conditions and parameters less than those in a reactor. Since ITER has a factor of 3-4 lower power density than a DEMO, it is necessary to change the dimensions of the test blanket module (TBM) to “act-alike” rather than “look-alike”. Act-alike” TBMs are needed, for example, to perform tests for thermal stresses, tritium transport and inventory, corrosion rates, and fluid flow. This approach is discussed in detail in Ref. 2. All these tests should meet, as possible, their objectives of validating the design tools and the database, and evaluating blanket performance under DEMO operational conditions. The goal of scaling is to ensure that changes in structural response and performance caused by changes in size and operating conditions do not reduce the usefulness of the tests.

The six ITER parties (EU, Japan, US, RU, China, and Korea) will share the 3 ITER test ports. The TBMs will be mounted on a 10 cm-thick water cooled frame. They are recessed by a 5 cm far from the plasma in the radial direction. The design of all the TBMs should satisfy all interface requirements of the ITER machine. The TBMs will represent several blanket concepts, including the helium- cooled pebble bed (HCPB) ceramic concept[3,4], the dual-coolant molten salt (DCMS) blanket concept[5-7], and the dual-coolant liquid lithium (DCLL) blanket concept[8-10]. The coordination and interface between the test blanket program of each party is undertaken by the test blanket working group[11-13] (TBWG). Strategies for performing various tests on these TBMs have been reported elswhere[14-15]

In this paper, we discuss some strategies and requirements placed on the machine operation (fluence, wall load, etc.) and on the TBM geometrical arrangement (size, dimension, spatial locations) in order to have a meaningful neutronics testing in ITER. These topics are covered in section II and III. Neutronics tests in fusion devices and neutronics scaling have previously been discussed in the literature[16-18]. In particular, the geometrical requirements on the TBM size is a premium factor in determining the usefulness of the neutronics tests. In this regard, we investigated, through performing two-dimensional transport calculations, the size and locations inside the TBM where neutronics tests can be best performed, based on the criterion of being less disturbed by neighboring heterogeneities and material boundaries. The results of this investigation are given in Section IV. In Section V we summarize the main findings of the present work.

II. KEY NEUTRONICS ISSUES

Neutronics issues to be resolved in ITER TBM are generic in nature and are important to each TBM type. The main issues include the demonstration of tritium self-sufficiency for a particular FW/Blanket/Shield concept, verification of the adequacy of available transport codes and nuclear data bases in predicting key neutronics parameters, and the verification of having an adequate radiation protection to the machine components and personnel

II.A. Tests for Confirming Tritium Self-Sufficiency

This does not appear to be possible in ITER since the basic shielding blanket does not produce tritium. Additionally, there is no interface with the tritium processing system of the TBM, which has a partial coverage. Direct demonstration of tritium self-sufficiency requires a fully integrated reactor system, including the plasma and all reactor prototypic nuclear components.

Unlike the case of using engineering scaling to reproduce DEMO-relevant parameters in an “act-alike” test module, dedicated neutronics tests require a “look-alike” test module for a given blanket concept. Each blanket concept should have its own tritium fuel cycle that is isolated from ITER basic machine. Tests for tritium production rates, tritium permeation, transport, and isotope separation will only demonstrate the potential of each blanket concept to generate and control tritium flow. It is by no means meant to confirm tritium self-sufficiency but only to provide valuable information regarding the main parameters needed to assess the feasibility of achieving tritium self-sufficiency. A fundamental question that needs to be answered (in ITER at least under reduced reactor parameters and low fluence) is how to reliably measure tritium production. This needs to be investigated by diagnostics’ people as well as experts’ opinion in this field.

II.B. Classification of Neutronics Tests

II.B.1 Dedicated Neutronics Tests

They aim at examining the present state-of-the-art neutron cross-section data, various methodologies implemented in transport codes, and system geometrical modeling as to the accuracy in predicting key neutronics parameters. These tests require a look-alike TBM.

It is strongly desired to perform neutronics tests on as “cold” module as possible to minimize problems associated with elevated breeding temperature that lead to tritium permeation. It is therefore recommended to perform these tests as early as possible during the D-D operation phase (year 4) or in low duties cycle DT operation phase (years 5&6).

Measurements to be performed under this class of tests can be further categorized as (a) in-pile-measurements, and (b) out-of-pile-measurements. Measurements of neutron and gamma heating rates and profiles, local (and if possible zonal) tritium production rates (TPR) and profiles, neutron and gamma spectra at various locations, and multi-foil activation measurements (e.g. ^{27}Al(n,2n), ^{27}Al(n,α), and ^{197}Au(n,γ), etc) fall within the former category. Measurements of dose behind test module and at cryostat, neutron yield from plasma, and source characterization (part of plasma diagnostics) fall within the latter category.

II.B.2. Supplementary Neutronics Tests

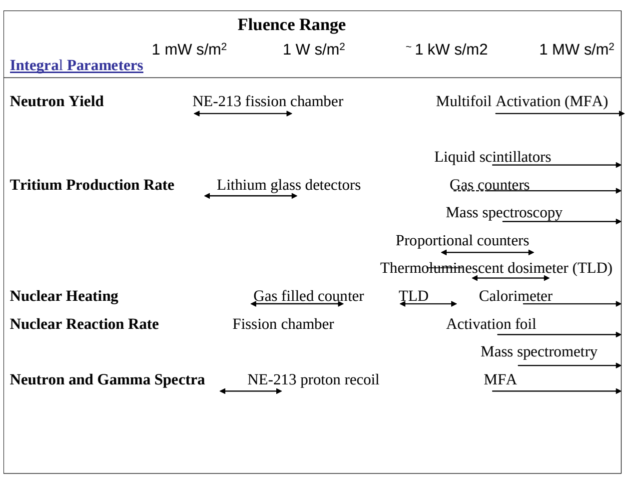
These tests provide the source term (e.g. heat generation and tritium production rates) for other non-neutronics tests devoted to predictive behavior and engineering performance verification (e.g. tritium permeation and recovery tests, thermo-mechanics tests, afterheat removal tests, etc). These tests can be scheduled during the high duty DT operation phase (year 7-10) devoted to the integrated tests. The TBM for these non-neutronics tests is of the act-alike type.

III. REQUIREMENTS FOR NEUTRONICS TESTS

III.A. Machine Operating Condition Requirements

The optimum operating condition of a test device in which dedicated neutronics measurements are intended is governed by the inherent limitations for a particular measured parameter. These limitations are the counting rate, the counting statistics, the detector size, the operation environment (e.g. temperature, magnetic field, etc.), and accuracy. The counting statistics and rates are the dominant factors in determining the wall load requirements. In Table I, we give the fluence requirements for measuring the key neutronics parameters by the listed measuring techniques. All neutronics parameters, except activation and damage parameters (e.g. dpa, hydrogen and helium production rates), can be measured at one of two fluence levels, namely, low fluence level (~1 MW.s/m^2) and very low fluence level (~1 W.s/ m^2). The low fluence level could be achieved, for example, with a wall load of 1 MW/ m^2 and 1-s pulse, or alternatively, a wall load of 0.0025 MW/ m^2 and 400 s pulses. In ITER, the envisioned pulse duration is ~400 s and the wall load at the TBM is ~0.78 MW/ m^2. This wall load is much larger than the 0.0025 MW/ m^2 needed to achieve the required low fluence level for neutronics tests. Therefore, one pulse or two is adequate for these tests. Thus, neutronics tests impose only modest requirements on the product of the wall load and plasma burn time with no stringent requirements on the magnitude of either parameter. The neutronics parameters, except induced activation, vary linearly with both the wall load and operation time.

TABLE I. Fluence Requirements for Various Experimental Techniques*



* For counter methods, the measuring time is to be 10~100 s.(from Ref. 16). NE213, fission chambers, and proton recoil counters are different detectors.

Other than the in-module neutronics measurements listed in Table I, out-of-module tests will include characterizing the neutron source impinging on the TBM and dose rate measurements at various locations in ITER (e.g. behind the shield and at the cryostat). These tests can also be used for calculation tools and data bases verification. Verification of the assessed induced activation and the radiation damage parameters will require higher fluences than those listed in Table I. The tests for these parameters may be best performed in facilities other than ITER (i.e. IFMIF).

III.B. Geometrical Requirements

The requirements on the TBM size and geometry are mainly governed by the objectives of the particular test under consideration. If a local measurement of TPR is intended, for example, then the only useful information that can be derived from such a test is the validation of the analytical tools and modeling used in the prediction of the TPR at these locations. Validation of the adequacy of present nuclear data can be best obtained from simple benchmark experiments and not from neutronics test in a complex machine such as in ITER. On the other hand, if the objective of the test is to verify an integal parameter such as tritium breeding ratio (TBR) for a given blanket concept, the test module in this case should duplicate in great detail the actual blanket module and its surrounding. Verification of the TBR requires that factors which affect the global TBR (e.g. penetrations, full coverage arrangement, closed tritium fuel cycle, etc) to be included in the design of the TBM. As mensioned earlier, a full sector is required for confirmation of tritium self-sufficiency condition, which cannot be performed in ITER. Even for local TPR measurements, a look-alike TBM is needed.

Neutronics parameters are, in general, sensitive to the TBM size and measurement locations. These locations should be selected away from boundaries as possible to minimize the influence of the ITER basic blanket (SS/H_2O) or the neighboring TBMs placed in the same port. Furthermore, parameters to be tested should not change much over a short distance inside the TBM. Steepness in the profile of these parameters can be found at locations adjacent to coolant channels or multiplier zones. It is preferable, therefore, to have flat values for these parameters over a large distance to eliminate uncertainties arising from defining the exact measuring locations, as discussed below.

IV. NUCLEAR PERFORMACE OF THE US TEST BLANKET MODULE PLACED IN ITER PORT

The U.S. TBM considered in the present analysis is based on the helium cooled pebble bed (HCPB) ceramic breeder (Li_2TiO_3-75% Li-6 enrichment) blanket concept which is placed inside half of an ITER port next to Japan TBM which is based on the HCPB (Li_4SiO_4) blanket concept and occupies a quarter of the same port. The objective of the present analysis is to examine the distances over which the nuclear heating rates and tritium production rates do not change much inside the U.S. TBM. The dimensions and arrangement inside the US TBM is shown in Fig.1 Two configurations are considered in the design of this module, namely, an act-alike configuration (left) and a look-alike configuration (right). The criterion on which the act-alike configuration is designed is discussed in a companion paper[2]. The thicknesses of the FW, the vertical coolant panels (VCP) and the horizontal coolant panels (HCP) are 2.8, 1.2 and 0.6 cm, respectively. They have a low activation ferritic steel (F82H) content of 53% by volume. The radial thickness of the ceramic breeder varies between 1.7 and 2.6 cm. in the left configuration and between 0.9 and 1.8 cm in the right configuration. The beryllium multiplier pebbles fill larger space behind the FW and around the breeder beds, as shown in Fig. 1 (see also Fig. 4 for locations of the FW, beryllium, and breeder). The TBM has a toroidal width of 67.4 cm and a radial depth of 60 cm. The Japan HCPB TBM is shown in Fig. 2 where Li_4SiO_4 (90% Li-6) is assumed as the breeder in this work. The influence of the breeder material difference of neighboring module is expected to be negligibly small in this model. The breeder and beryllium multiplier layers alternate in location where the breeder is placed first behind the FW which has a structure (F82H) content of 68%. The toroidal and radial dimensions are 63 and 60 cm, respectively. The breeder and multiplier in both TBMs have a 60% packing factor.

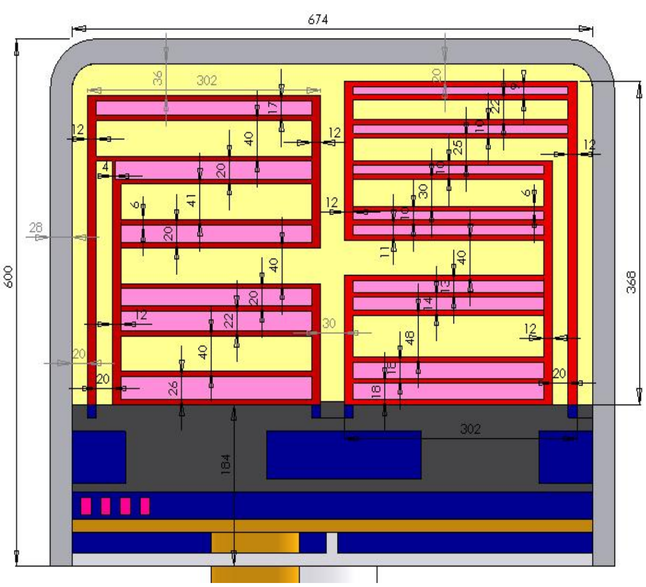


Fig. 1. The US Test Blanket Module with act-alike Configuration (left) and look-alike Configuration (right).

IV.A. Calculation Procedures and Modeling

An R-θ model was used to describe the geometrical arrangement of the ITER basic shielding blanket, the vacuum vessel (VV), the magnet, the US and Japan TBMs, and the detail of the test port at the mid plane of ITER machine. An isometric view of the material assignment of this model is shown in Fig. 3. The thickness of the VV at the OB is 75 cm (33.7 cm IB) with 3 cm-thick SS316ln walls. The thickness of the Be tile, Cu-alloy zone (79% SS316ln, 21% H_2O), 2nd FW (81% SS316ln, 19% H_2O) and ITER shielding blanket (72% SS316ln, 28% H_2O) is 1, 2.2, 4.9, 36.9 cm, respectively. In the calculation model, the Be tile in the inboard (IB) is placed at a radial distance of 356 cm from the center of the torus (850 cm for the outboard, OB).

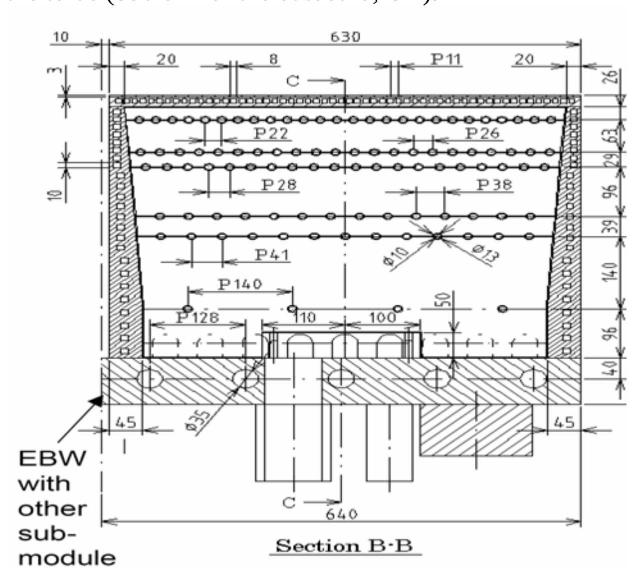


Fig. 2. Japan Test Blanket Module.

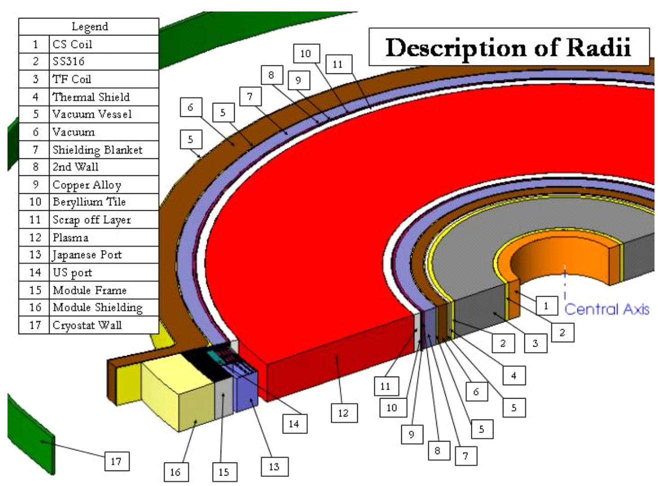


Fig. 3. Isometric View Showing Material Assignments in the Calculation Model.

Figure 4 shows the details of the model at the port where the two TBMs are placed and mounted on a 10 cm-thick steel frame (60% SS316ln, 40% H_2O). There is a 10 cm-thick gap behind the manifolds of the two TBMs. A port shield (72% SS316ln, 28% H_2O) of a thickness 90 cm is placed behind the back of the frame. The two TBMs are placed at a distance of 855 cm from the center of the torus. The recess of 5 cm depth is prescribed in ITER design for the test ports.

The discrete ordinates transport code DORT[19] was used in the calculation of the R-θ model with P_5S_8 approximation. The 46 neutron-21 gamma group library used is based on FENDL-1 data base[20-21]. In the model, the entire machine in the θ direction was considered (θ varies between 0 and 360 degree).

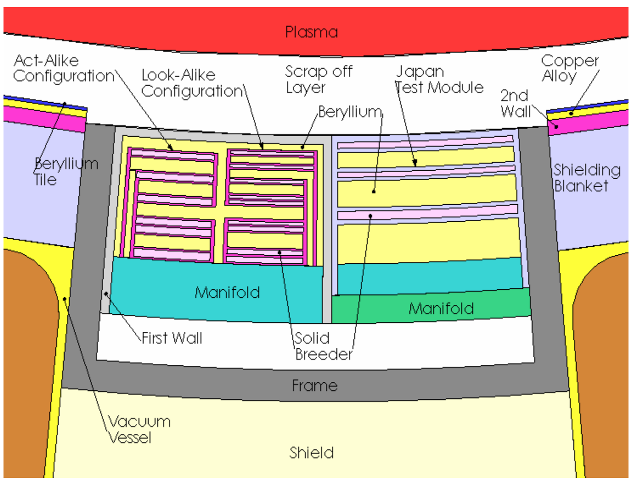


Fig. 4. Top View Showing the US TBM and Japan TBM Placed in the Port with the Surrounding ITER Basic Shielding Blanket.

IV.B. Calculation Results

The nuclear heating rates in the FW of the US TBM are shown in Fig. 5. The profiles in the toroidal direction are nearly flat over a distance of ~10-14 cm (left configuration) and ~16-20 cm (right configuration) This flatness decreases by depth. The values in the right configuration at these flat regions are slightly larger than those in the left configuration. The heating rate measurements could be performed over these flat regions with no concern for error due to uncertainty in location definition. The profiles are steep near the left side wall of the TBM over a toroidal length of ~16 cm due to the presence of the 10 cm-thick frame which is cooled with water. The heating rates there are a factor of 1.13-1.27 larger than those at the inner regions.

The nuclear heating rates in the beryllium layer located just behind the FW in the toroidal direction are shown in Fig. 6. The profiles are flat over the entire layer. As shown, the heating rates in structure content of the adjacent side wall of the TBM are much higher (with a factor of ~2.7) than those in the beryllium layer. Notice that the beryllium layer at this location extends over the entire width of the TBM that covers both the left and right configurations.

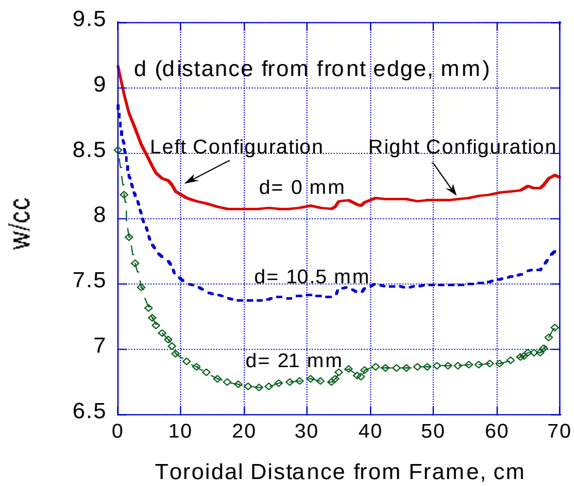


Fig. 5. Nuclear Heating in the First Wall of the U.S. Test Blanket Module in the Toroidal Direction.

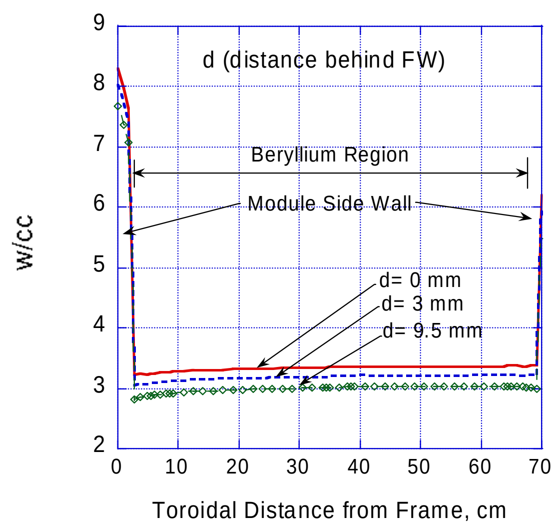


Fig. 6. Nuclear Heating in the Beryllium Region behind the FW in the U.S. Test Blanket Module in the Toroidal.

The nuclear heating rates across the module at a depth 42 mm behind FW are shown in Fig. 7. The values vary noticeably as we move in the toroidal direction and away from the frame. As shown, the largest heating rates are inside the breeding layer while, as shown in Fig. 6, the heating profile is flat over the entire beryllium layer. This figure shows the effect of heterogeneity on the heating profiles which can not be estimated from 1-D calculation model. The heating rates at the flat region in the breeder of the left configuration are a factor of ~4 larger than those in the Be of the right configuration. Their values are almost the same over a toroidal distance of ~10 cm. The heating rate in the breeder peaks near the vertical coolant panels surrounded by beryllium (see Fig. 4). As also shown in Fig. 6, the flatness of the heating rate in beryllium in Fig. 7 has been noticed in other beryllium layers (not shown).

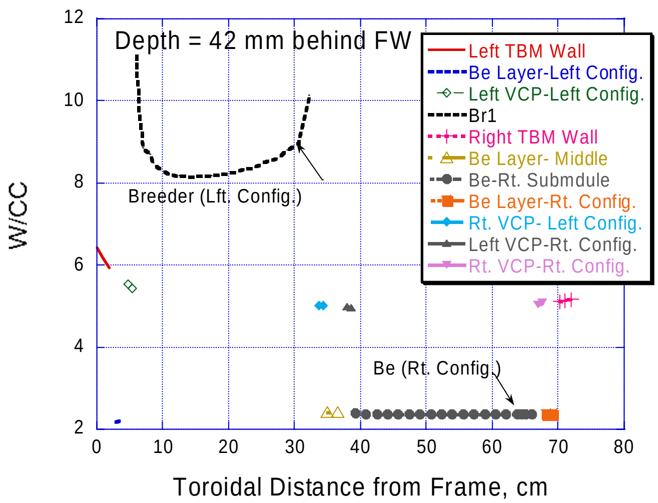


Fig. 7. Nuclear Heating across the U.S. Test Blanket Module at a Depth 42 mm behind the First Wall.

Figure 8 shows the radial nuclear heating rates at a toroidal distance of 28 cm from the left frame. The nuclear heating rates in the breeder layers are the largest while heating rates in beryllium is the lowest. The heating rates in the structure contents of the FW and horizontal cooling panels (HCP) assume intermediate values.

As for the assessment of tritium production rate, we show in Fig. 9 the toroidal profile of the TPR in each breeder layer of the two configurations of the US TBM. The profiles are nearly flat over a reasonable distance in the toroidal direction where measurements can be performed (10-16 cm in the left configuration and 10-20 cm in the right configuration). The steepness in the profiles near the ends of layers is due to presence of the beryllium layers (see Fig.4) and to the neutrons reflected by the structure presents in the surrounding coolant panels. The TPR values at these locations are larger by a peaking factor of 1.4-1.5 compared to the values in the inner zones. Notice that the breeder layers of the left configuration are placed at different radial depth than those of the right configuration (see Fig. 4).

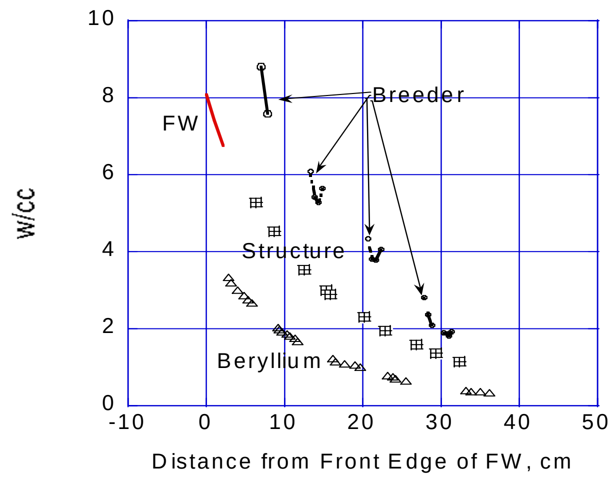


Fig. 8. Radial Nuclear Heating at a Toroidal Distance of 28 cm from the Frame.

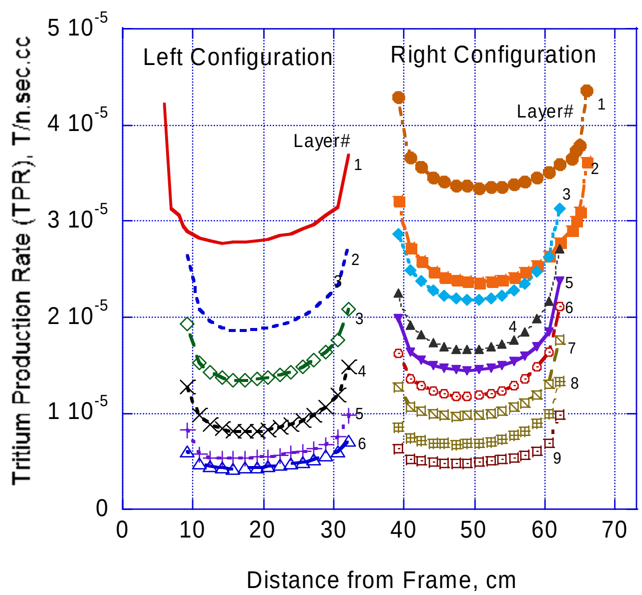


Fig. 9. Toroidal Profile of Tritium Production Rate in each Breeder Layer of the Two Configurations in the US Test Blanket Module.

Since the breeder layers have smaller thickness (between 1.7-2.6 cm. in the left configuration and between 0.9-1.8 in the right configuration), one would expect that the flatness in the TPR shown in Fig. 9 would have much smaller range in the radial direction. These radial TPR profiles are shown in Fig. 10 at a toroidal distance of 28 cm from the frame inside the 2nd and 3rd breeder layer. The steepness of these profiles is apparent near the horizontal cooling panels for the reasons mentioned above. The distance over which TPR changes by 5% from its lowest value is limited to ~1 cm. Therefore, to achieve high resolution, the TPR measurements should be performed within this 1 cm range in the radial direction.

V. SUMMARY

Neutronics tests planned to be performed in ITER aim at resolving key neutronics issues such as the demonstration of tritium self-sufficiency and verification of the adequacy of calculation tools and nuclear data bases in accurately assessing the nuclear environment. Although each ITER partners will perform these tests inside their designated test blanket module (TBM) that may utilize different breeders, coolants, and structure, neutronics issues to be resolved are generic in nature and are important to each TBM type. Dedicated neutronics tests specifically address the accuracy involved in predicting key neutronics parameters such as tritium production rate (TPR), volumetric heating rate, induced activation and decay heat, and radiation damage to the reactor components. In addition, neutronics analyses are required to provide input support for other tests (e.g. heating rates for thermo-mechanics tests). While tritium self-sufficiency for a given blanket concept can only be demonstrated in a full sector, as envisioned in a DEMO, including a closed tritium fuel cycle, testing in ITER TBM can provide valuable information regarding the main parameters needed to assess the feasibility of achieving tritium self-sufficiency.

From instrumental consideration, all neutronics parameters, except induced activation, can be performed in either low fluence (~1 MWs/m^2) or very low fluence (~1 w.s/m^2) operation mode. The low fluence level could be achieved, for example, with a wall load of 0.0025 MW/ m^2 and 400 s pulse. In ITER, the wall load at the TBM is ~0.78 MW/ m^2, which is much larger than the 0.0025 MW/ m^2 value. One pulse or two is adequate for these tests.

Unlike the case of using engineering scaling to reproduce DEMO-relevant parameters in an "act-alike" test module, dedicated neutronics tests require a "look-alike" test module for a given blanket concept. Furthermore, measured neutronics data requires a high spatial resolution. This necessitates the measured quantity to be as flat as possible in the innermost locations inside the test module. In the present work, we confirmed this requirement based on results from two-dimensional calculations in an R-θ model that accounts for the presence of the ITER shielding blanket and the surrounding frame of the port where the US and Japan TBMs are placed. It is shown that the profiles of the TPR and heating rates have flat values over a range of 10-20 cm in the toroidal direction and a range of ~1 cm in the radial direction inside the breeding layers of the HCPB TBM.

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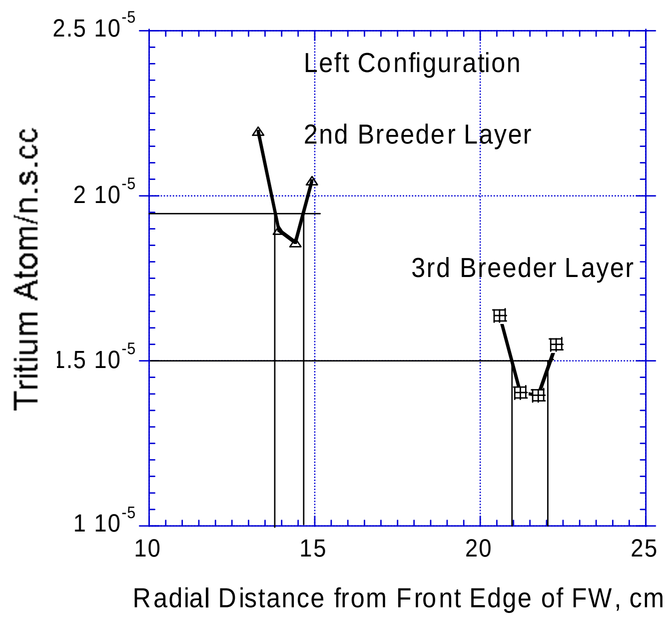

Fig. 10. Radial Tritium Production Rate at a Toroidal Distance of 28 cm from the Frame (2nd and 3rd Breeder Layers.

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