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Material irradiation conditions for the IFMIF medium flux test module

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Abstract

The International Fusion Materials Irradiation Facility (IFMIF) is an accelerator driven neutron source, which is designed to perform material irradiations at conditions very close to that of future fusion reactors up to the anticipated lifetime of structural materials. Besides irradiation of material samples in the high flux test module, in situ creep-fatigue tests for structural materials in the creep-fatigue test module (CFTM) and tritium release experiments for breeder blanket materials in the tritium release module (TRM) are foreseen in the medium flux test module (MFTM) of IFMIF. The present study is devoted to the evaluation of the creep-fatigue specimens and beryllium irradiation conditions in the MFTM of IFMIF. It is demonstrated that maximum displacement damage rate in the CFTM is about 13 dpa/fpy. The analysis shows that the TRM can closely meet fusion-typical He and dpa production rates in Be, while the T-production is somewhat above fusion conditions for the present test module design. Removing the CFTM and shifting of the whole TRM upstream increases the damage rate to about 60% of that for a fusion reactor and provides tritium and helium production rates comparable with fusion. In addition, the important He/T ratio nicely fits the fusion relevant value after one year of irradiation in the IFMIF TRM.

1. Introduction

The theoretical and experimental design of the International Fusion Materials Irradiation Facility – IFMIF was started in 1995. The mission of this facility is to provide an accelerator based deuterium–lithium (d–Li) intense source to produce neutrons with a suitable energy spectrum at sufficient intensity and irradiation volume to test candidate fusion materials up to the full lifetime of their anticipated use in a fusion power reactor. Special attention during the design of IFMIF is given to reproducing fusion irradiation conditions as close as possible for adequate material testing. It is worthwhile to note that for different materials the required irradiation conditions could differ. To satisfy these needs IFMIF should have enough flexibility in its design to enable easy transformation according to the requirements for a specific material. Fortunately, the modular IFMIF design allows shifting and replacement of the parts of the test cell without significant changes in their layout.

In IFMIF, the majority of specimens made of structural materials will be irradiated in the high

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flux test module (HFTM) followed by post irradiation examination in hot cells. In addition to static material irradiation, more sophisticated in situ creep-fatigue of steel samples and tritium release tests on beryllium and various breeder materials are foreseen in the medium flux test module (MFTM) (see Fig. 1).

The suitability of irradiation conditions for structural materials in the high- and medium flux test modules was already proven and discussed on several occasions (see e.g., [1]).

However, experiments with breeder materials demanded further improvement in the irradiation conditions. Such improvement for lithium-based ceramics was performed in a previous study [2].

The present study is aimed at improvement of the MFTM design with respect to neutronics for beryllium irradiation in the tritium release module (TRM) as well as in accessing irradiation conditions of creep-fatigue samples, and comparison of the material response with that for a future fusion reactor.

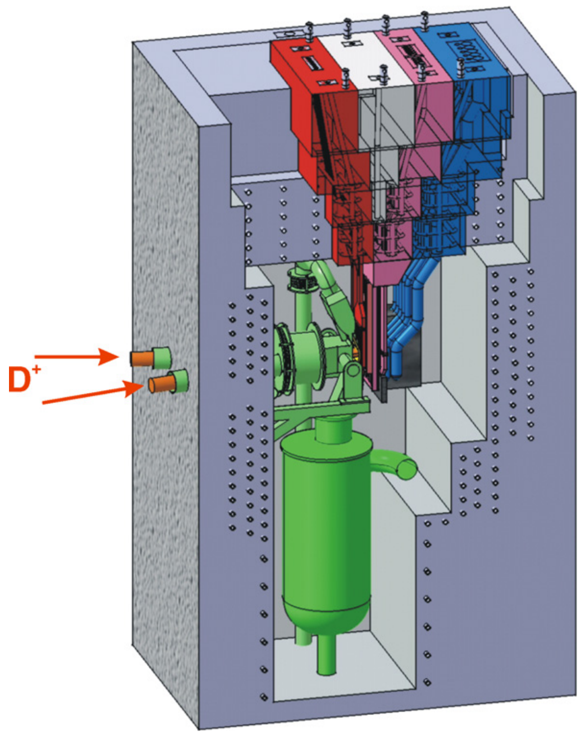

Fig. 1. Elevation view of the IFMIF test facilities with the lithium target and vertical test assemblies (from left to right): high (red), medium (pink) and low (light blue) flux test modules. The approximate size of the section shown is 4 m × 2.5 m × 8.4 m. (For interpretation of the references to colour in this figure, the reader is referred to the web version of this article.)

The description of the geometry model and the details of the neutronics analysis for the IFMIF high and medium flux test modules (HF&MFTM) can be found elsewhere [2–4].

2. Evaluation of irradiation conditions

2.1. Methodological approach

The IFMIF global 3D geometry model [5] was used as an input for the MCNP code [6] for neutral and charged particle transport calculations with an updated version of the neutron source McDelicious-05 [7]. The fusion power reactor (FPR) model of the EU power plant conceptual study (PPCS) [8] employing the helium cooled pebble bed (HCPB) breeder blanket was considered. A suitable 3D-torus sector model has been developed as a part of the PPCS study [9] to enable proper neutronics calculations with the MCNP code. The high-energy neutron cross-section library LA150 [10] was used in the present calculations. The neutron flux spectra for several variants of the MFTM geometry were calculated and processed with the FISPACT-2005 inventory code [11] for evaluation of helium and tritium production rates using the data from European Activation File (EAF-2005) [12]. The results were compared with our previous calculations using the ALARA code [13].

2.2. Assessment of changes in tritium and helium production in the IFMIF TRM due to shielding block

One of the objectives of this work was to determine tritium, helium and damage production rates in beryllium irradiated in the TRM in the presence of a shielding block. The horseshoe-type shielding block behind the test module was considered to reduce the activation of concrete walls and dose levels inside the maintenance room. After the initial screening of materials for shielding, tungsten carbide was identified as providing the highest reduction of the activation level.

During beryllium irradiation helium is produced in reactions ^{9}Be(n,α) and ^{9}Be(n,2n) with two resulting helium nuclei per reaction for the latter. Since both reactions have neutron energy thresholds above 1 MeV, helium production in beryllium depends mostly on the population in the high-energy range of the neutron spectrum. The threshold for the direct tritium production through the

$^{9}Be(n,t)^{7}Li$ reaction is even higher (about 11.6 MeV).

However, tritium can be also be generated through the two-step reaction $^{9}Be(n,\alpha)^{6}He \rightarrow ^{6}Li(n,\alpha)T$ which, in particular, gives a significant contribution for a soft neutron spectra (e.g., mixed neutron spectrum of the HFR, Petten). Therefore, the two-step reaction is effective only if the neutron spectrum shows both a pronounced population of high- (for the efficient ^{6}He production) and low-energy (for the T production on ^{6}Li) neutrons. Several design variants were considered for the irradiation of beryllium at the TRM position (see Table 1).

Table 1
Design variants of MFTM used for the optimization of beryllium irradiation conditions in the TRM

#	W spectral shifter	CFTM	Additional moderator	Shielding block
1	Yes	Yes	No	No
2	No	Yes	Yes	No
3	shifted	shifted	No	No
4	shifted	shifted	Yes	No
5	shifted	shifted	EXTENDED	No
6	Yes	Yes	Yes	Yes
7	shifted	shifted	Yes	Yes

Two additional variants with the shielding block placed behind all test modules were considered as well:

- the initial model with a tungsten spectral shifter and extended graphite reflector (6);
- without the CFTM and tungsten spectral shifter, where the TRM was shifted upstream to the HFTM to increase the neutron flux (7).

The neutron spectra in the TRM calculated for the variants from Table 1 are presented in Fig. 2. In the high-energy range the spectra differ manly due to the presence of the CFTM and tungsten spectral shifter. Variants with the CFTM and W-plate (curves 1 and 6) have the lowest high-energy part, which increases with removal of the W-plate (curve 2) and further increases with the additional removal of the CFTM and shift of the whole TRM upstream (curves 3–5, 7). The low-energy parts of the spectra differ due to the presence and size of the carbon moderator and/or shielding block. The highest population of the low-energy part of the spectra corresponds to variants with the shielding block, which also effectively slows down neutrons.

The results of helium and tritium accumulation are summarized in Table 2. As soon as the new version of FISPACT became available, we repeated some calculations to verify our results obtained with the ALARA code [13]. The results of both codes are in good agreement.

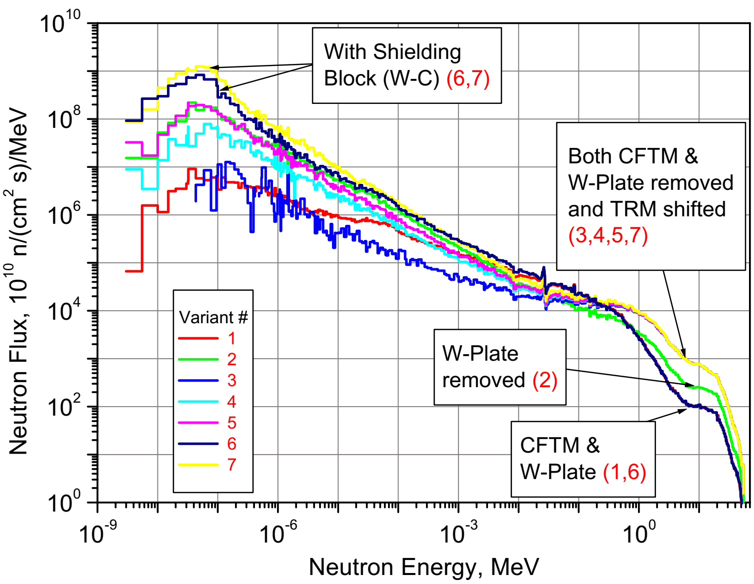

Fig. 2. Neutron spectra in the TRM calculated for several design variants. The curves are numbered according to the Table 1.

Table 2
He and T-production in Be irradiated in the TRM of the IFMIF and in the HCPB of the FPR

#	Code	n-Flux n/cm^2/s	He (appm)			T (appm)			He/T	
			1 y	2.28 y	4.57 y	1 y	2.3 y	4.6 y	1 y	4.6 y
With CFTM and W moderator										
1	ALARA	1.5×10^{14}	795	1821	3649	18.4	41.1	78.6	43.2	46.4
	FISPACT-2005		754	1719	3441	18.3	40.7	77.9	41.2	44.2
6	FISPACT-2005	1.4×10^{14}	717	1634	3272	17.4	39.0	75.7	41.2	43.2
With CFTM and without W moderator										
2	ALARA	1.7×10^{14}	1923	4407	8833	44.6	101	199	43.1	44.4
	FISPACT-2005		1827	4159	8312	44.3	99.7	195.7	41.2	42.5
No CFTM, no W moderator, TRM is shifted upstream										
3	ALARA	3.7×10^{14}	5566	12749	25524	116	260	497	48.0	51.4
	FISPACT-2005		5135	11645	23116	103	228	434	49.8	53.3
4	ALARA	3.8×10^{14}	5581	12786	25607	117	264	514	47.7	49.8
	FISPACT-2005		5300	12017	23849	116	259	500	45.7	47.7
5	ALARA	3.9×10^{14}	5584	12798	25639	118	269	534	47.3	48.0
	FISPACT-2005		5303	12027	23876	117	263	517	45.3	46.2
7	FISPACT-2005	4.1×10^{14}	5274	11974	23816	121	287	613	43.6	38.8
HCPB fusion power reactor blanket										
HCPB	ALARA	1.1×10^{16}	5656	12912	25825	85.5	247	683	66.1	37.8

A comparison of the gas production in the IFMIF MFTM and HCPB fusion power reactor reveals that helium production in the shifted TRM has practically the same level as in the HCPB, while tritium production after one to two years is slightly higher, but decreases after about five years. The implementation of a shielding block slightly decreases the helium production for both the initial and shifted geometry, while for the shifted geometry the tritium production slightly increases. This

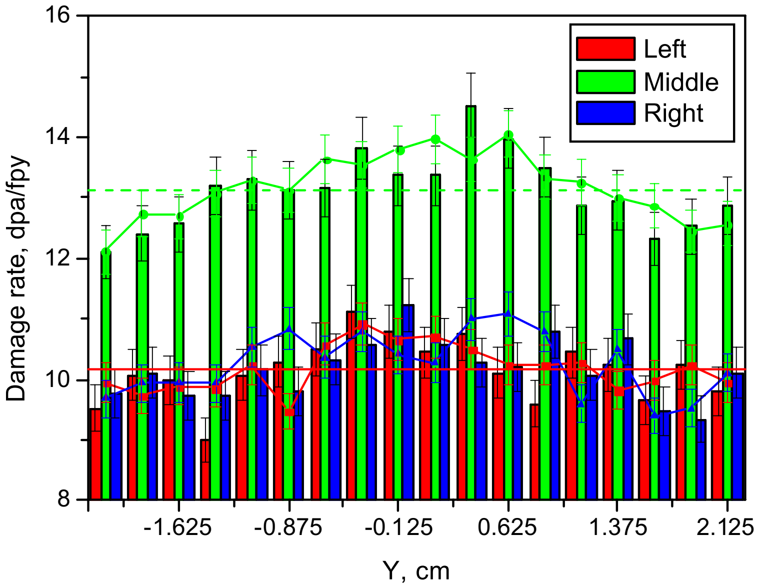

Fig. 3. Damage profile along the creep-fatigue specimen length.

results in a further reduction of the He to T-production ratio. However, after five years the ratio nicely fits that of a fusion reactor.

2.3. Irradiation damage and heat deposition at CFTM

The creep-fatigue test module (CFTM) is considered to be a part of the MFTM to perform in situ tests of structural materials. An important task of this work was to evaluate the damage rate inside creep-fatigue specimens and the heat generation rate in push–pulling rods. The later information is necessary for determination of the temperature distribution inside specimens, which is important for the design of the specimen cooling system.

The damage rate distribution along the length of the creep-fatigue specimens is shown in Fig. 3. Average damage in the central specimen is about 13 dpa/year, while for the periphery specimens it is about 10 dpa/year. The damage variation along the specimen length does not exceed ±10%, which is acceptable according to the IFMIF user requirements.

The heads of the push–pulling rods were divided into small cubes with dimensions of $0.25 \times 0.25 \times 0.25\ \mathrm{cm}^3$. The distribution of heat deposition in the lower row of heads is presented in Fig. 4. The upper part sequence of plots in Fig. 4(a) presents the spatial heat distribution in the upper slice. As expected the highest heat deposition occurs near the center of the beam. The maximum deposition in the slices does not exceed 0.8 W/g as shown in the lower sequence of plots in Fig. 4(b). The presence of the shielding block has only a minor effect on the damage and heat production rate in the CFTM.

3. Conclusions

The goals of the present study were (i) to find the optimal geometry for reproducing fusion irradiation conditions for beryllium in the TRM, (ii) to evaluate

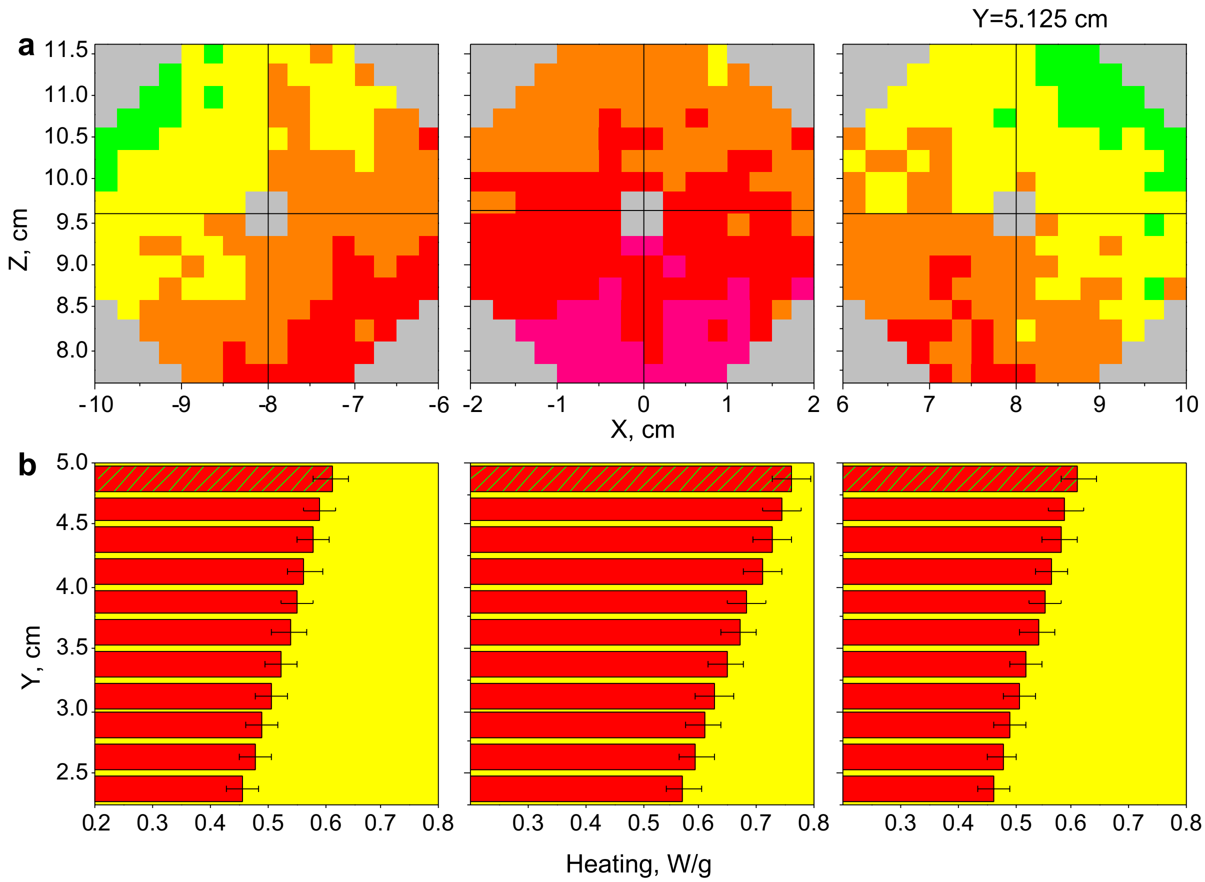

Fig. 4. Spatial heat deposition in the lower row of push–pull rods of CFTM: (a) horizontal cross section with the highest heat deposition, (b) total deposition in cross sections as a function of the rod height.

heat deposition and damage rates in the CFTM and (iii) to assess the effect of the horseshoe-type shielding block on the irradiation conditions in the MFTM. The following conclusions can be drawn:

- The results demonstrate that the irradiation of beryllium in the IFMIF TRM provides a helium to tritium production ratio, which is closer to that of a typical PPCS reactor than other irradiation facilities. At the same time it was shown that the indirect tritium production is ineffective for the IFMIF TRM due to the low ^{6}Li-production rate. It implies that an additional carbon moderator has no significant effect. Removing the tungsten spectral shifter increased the tritium production rate in the IFMIF TRM although it is still about two times lower than that of the PPCS. The tritium production even exceeds the fusion relevant level after moving the whole TRM upstream towards the HFTM while the tungsten spectral shifter module is completely removed.
- 3D distributions of heat deposition in the heads of pulling rods necessary for thermo-hydraulic calculations were obtained.
- Damage rate variations along the length of the creep-fatigue samples were found to be less than 20%.
- A horseshoe-type shielding block has a minor effect on the heat deposition and damage rate in the CFTM.

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