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Neutronics Requirements for a DEMO Fusion Power Plant

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This paper addresses the neutronic requirements a DEMO fusion power plant needs to fulfil for a reliable and safe operation. The major requirement is to ensure tritium self-sufficiency taking into account the various uncertainties and plant-internal losses that occur during DEMO operation. A further major requirement is to ensure sufficient protection of the superconducting magnets against the radiation penetrating in-vessel components and vessel. Reliable criteria for the radiation loads need to be defined and verified to ensure the reliable operation of the magnets over the lifetime of DEMO. Other issues include radiation induced effects on structural materials such as the accumulated displacement damage, the generation of gases such as helium which may deteriorate the material performance. The paper discusses these issues and their impact on design options for DEMO taking into account results obtained in the frame of European Power Plant Physics and Technology (PPPT) 2013 programme activities with DEMO models employing the helium cooled pebble bed (HCPB), the helium cooled lithium lead (HCLL), and the water-cooled (WCLL) blanket concepts.

Keywords: Neutronics, DEMO, Tritium breeding, shielding

1. Introduction

The European Power Plant Physics and Technology (PPPT) programme [1], launched initially by the European Fusion Development Agreement (EFDA) and organised now within the newly established EUROfusion Consortium, aims at developing a conceptual design of a fusion power demonstration plant (DEMO) within the “Horizon 2020” roadmap [2].

Various integrated PPPT projects are being conducted to meet this ambitious goal including e. g. Breeder Blanket (BB), Safety and Environment (SAE), Magnets (MAG), Materials (MAT), Remote Maintenance (RM), and others. Neutronics plays an important role for all of the related activities since it has to provide essential data which are required for the nuclear design of DEMO and its components, its performance assessment and verification.

This paper addresses the neutronic requirements a DEMO fusion power plant needs to fulfil for a reliable operation. The major requirement is to ensure tritium self-sufficiency taking into account the various plant-internal losses that occur during DEMO operation. A further major requirement is to ensure sufficient protection of the superconducting magnets against the radiation penetrating in-vessel components and vessel. To this end, reliable criteria for the radiation loads need to be defined and verified to ensure the reliable operation of the magnets over the lifetime of DEMO. Other issues include radiation induced effects on structural materials such as the accumulated displacement damage, the generation of gases such as helium which may deteriorate the material performance.

The paper discusses these issues and their impact on design options for DEMO taking into account results obtained in the frame of the 2013 PPPT activities with DEMO models employing the helium cooled pebble bed (HCPB), the helium cooled lithium lead (HCLL), and the water-cooled lithium lead (WCLL) blanket concepts.

2. Tritium breeding performance

Tritium self-sufficiency is a pre-condition for the operation of a fusion power plant utilizing the D-T fusion reaction as source of energy production. (This is due to the fact that no external sources are available/conceivable that could provide a sufficient Tritium production). To ensure Tritium self-sufficiency, a net Tritium Breeding Ratio (TBR) ≥ 1.0 is required, i. e. it must be assured that per D-T fusion reaction one triton, generated in the breeding blankets surrounding the plasma chamber, is finally available for injection into the plasma. In effect, a global TBR with some additional margin in excess of unity must be achieved to account for Tritium losses and uncertainties. This needs to be proven by means of realistic neutronic calculations which requires, first of all, a suitable computational method such as the Monte Carlo technique qualified for fusion neutronics applications, second, a realistic model of the fusion reactor with the blanket geometry and the materials detailed as much as needed, and, finally, high quality nuclear data which are validated against experiments as far as possible. The considered TBR margin will thus include uncertainties of the calculation

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itself - uncertainties due to the modeling assumptions when using e. g. geometrical simplifications and the neglecting of non-breeding blanket ports, and uncertainties due to the computational approach including e. g. statistical errors, uncertainties of the nuclear data, and the neglecting of effects such as the Lithium burn-up during the blanket lifetime - and uncertainties due to Tritium losses encountered in the D-T fuel cycle. Typical design targets of the global TBR to account for all of these effects are in the range of 1.05 to 1.15 [3].

The prediction of a TBR calculation, not considering design margins, needs to be checked against benchmark experiments to ensure they are reliable within the considered uncertainty margin. Such benchmark experiments have been previously performed on mock-ups of the European concepts for a Helium Cooled Lithium Lead (HCLL) and a Helium Cooled Pebble Bed (HCPB) blanket using 14 MeV neutron generator facilities at ENEA, Frascati, and TUD, Dresden, Germany [4]. The tritium production rates measured for the HCLL mock-up could be reproduced within 2 to 3% while those measured for the HCPB mock-up were underestimated by 5 to 10%. This will result in a corresponding underestimation of the TBR predicted for HCPB-type blankets with Beryllium employed as neutron multiplier. Design calculations for such blanket configurations are thus conservative providing (positive) margins for the compensation of other (negative) margins imposed on the TBR calculation.

2.1 TBR design margins for DEMO

To guarantee Tritium self-sufficiency for DEMO, the global TBR, which is provided by means of 3D Monte Carlo (MC) calculations, must exceed unity by a safety margin that accounts for the following uncertainties and effects:

- Nuclear cross-section data uncertainties depend very much on the assumed blanket concept and the materials involved. Typical uncertainties range from 1-2 % for liquid metal blankets utilizing Pb-Li as breeder material, such as the HCLL and the WCLL blankets, up to 5 – 10 % for solid breeder blankets with Beryllium as neutron multiplier [4].
- Statistical uncertainties of MC calculations are negligible for the TBR assessment. A statistical error less than 0.1% can be easily achieved without any special computation effort.
- Uncertainties due to modelling assumptions are hard to quantify. From numerous analyses conducted over the past two decades, it is inferred, however, that such uncertainties are very small due the capabilities of the MC codes in modelling the geometry in great detail – even for large and complex geometric configurations. This uncertainty, in the end, depends on the expertise and knowledge of the nuclear analyst in devising the model according to the needs for the neutronic transport simulation. With the approaches available nowadays to generate MC simulation models directly from CAD models this uncertainty is further reduced. Actually, it is just limited by the extent of simplification applied to the underlying CAD models which are mostly based on engineering designs. The related uncertainty is thus determined by the expert judgment of the nuclear analyst and can be reduced to an insignificant level, i. e. far less than 1%.
- Uncertainties due to specific engineering design assumptions are extremely difficult to quantify and predict since they usually change with the design progress. As a general rule, as the design progresses, the engineering design becomes more detailed and includes more elements with a negative impact on the TBR performance. Even with a technical mature design, one has to assume (and cope with) design changes that will impact the Tritium breeding. To account for this effect, it would be safe to include an uncertainty margin of 2 to 3% [3]. This is, however, not mandatory and might be neglected if one can be sure the design is technically mature.
- The effect of the ^{6}Li burn-up on the TBR during blanket lifetime is negligible for Pb-Li based liquid metal blankets due to i) the circulation of the liquid metal for the external Tritium extraction and ii) the high ^{6}Li enrichment of 90at% employed in such blankets. For HCPB type solid breeder blankets, a small but significant effect in the range of 1 to 2% TBR losses need to be assumed for the considered blanket lifetime of DEMO (2 to 5 fpy) and the ^{6}Li enrichment in the range of 30 to 60%.
- The effect of blanket ports without breeder material is of high importance for the TBR performance of DEMO. It depends very much on the breeder material employed in the blanket modules, the size and number of the ports, and their build with e. g. no material included in the port (void space) or some neutron absorbing and/or reflecting material included such as steel with a coolant. Typically, the TBR losses due to the port effect are larger for Pb-Li based liquid metal blankets than for solid breeder blankets. This is due to the inherent nuclear properties of lead providing a high neutron reflection but a low neutron moderation power. This results in strong out-scattering processes of high energy neutrons from the Pb-Li breeder into the blanket ports where they get lost by leakage. This effect is mitigated when some material is inserted into the port: both the leakage of neutrons out of the blanket system is reduced and neutrons are scattered back from the port into the breeding blanket modules. The port effect is smaller for HCPB type breeder blankets with a Beryllium neutron multiplier which also acts as neutron moderator and shows a smaller neutron reflection power as compared to lead. To quantify the port effect for DEMO conditions, the TBR was assessed for different port configurations using the 2013 HCPB and HCLL models of DEMO. With 16 ports of 1m x 2m size, the TBR losses for the HCLL DEMO amount to 15% and 6 %, with the port voided and plugged with a typical plug mixture (32 % SS-316, 8 % H_2O, 60% void). For the HCPB DEMO the losses are at 10%

and 4%, respectively. The related reduction of the blanket coverage ("loss of breeder area") amounts to about 3%. Thus, in the case of the HCLL DEMO, the TBR losses are considerable larger than the reduction of the blanket coverage: by a factor two when the port is plugged with a steel/coolant mixture.

- Tritium losses in the fuel cycle include the Tritium which is not available for the re-fuelling of the plasma but is retained e. g. in the materials of the in-vessel components or the Tritium recovering system, or lost by radioactive decay or leakage to the environment. Assessments of such losses are in the range of 3 to 5% depending strongly on the time window assumed between the generation of the Tritium in the blanket and its re-use in the plasma [3]. The limits are mostly determined by the Tritium decay which is at a level of about 5% per year. Thus, if we conservatively assume a one year delay for the re-use of the generated Tritium, a margin of 5% needs to be added to the calculated TBR.

2.2 TBR design target for DEMO

A meaningful TBR design target should include margins that account for the uncertainties and effects discussed in section above 2.2 when evaluating the Tritium breeding performance on the basis of the global TBR through a 3D MC calculation. The goal for the specification of such a target value is to ensure a (final) net TBR $\geq$ 1.0 for DEMO independent of the considered blanket concept and effects which are not taken into account in the TBR calculation. These effects, uncertainties and related margins depend, however, to a large extent on the considered blanket concepts as detailed above. This applies, in particular, for the uncertainties of the nuclear data, the port and the burn-effect. A pragmatic approach is to assume a very conservative value of 5% for the Tritium losses in the fuel cycle, neglect on the other hand the nuclear data related uncertainties and the burn-up effect, and assign another margin of 5% to the port effect. This results in a design target value of TBR=1.10 which needs to be achieved in a 3D MC calculation without taking into account blanket ports and the burn-up effect. This design target is actually compliant with other assessments [3] and also agrees with earlier estimations for the European blanket concepts [5].

The rationale for this approach is as follows. The decision not to include a safety margin for nuclear data related uncertainties is well justified since it is known from the benchmark experiments that the Tritium production is underestimated for the HCPB blanket and is well reproduced for the HCLL blanket, i. e. Pb-Li based liquid metal blankets. Small uncertainties in the case of Pb-Li could thus be covered with the generous 5% margin assumed for the Tritium fuel cycle losses. The same applies for the burn-up effect in case of the HCPB type blanket which must not be considered for Pb-Li based blankets. The 5 % margin assigned for the port effect is crucial since it limits the total port area to a bit less than 3% for Pb-Li based blankets and a bit more for HCPB type solid breeder blankets. Thus, if larger port areas are required, this needs to be justified and the TBR design margin needs to be increased correspondingly.

2.3 Tritium breeding performance of HCPB, HCLL and WCLL type DEMO

Assessments of the TBR performance of the DEMO power plant have been performed in the frame of the 2013 PPPT programme on the basis of the available preliminary design concepts for the HCPB, HCLL and WCLL blankets [8, 9, 10]. Tritium self-sufficiency can be achieved with all considered blanket variants. The margins obtained for the global TBR are, however, significantly affected by the current design assumptions for the DEMO blankets which are, in the case of the HCPB and the HCLL blanket, based on the related ITER TBM design with a very massive blanket box structure including a very solid internal stiffening grid, and the blanket segmentation scheme with a comparatively large number of blanket modules. To arrive at a higher margin, there is the need to improve the breeding blanket design and configuration, e.g. by decreasing the number of blanket modules (i. e. larger module size) and minimize the amount of steel structure in the breeding modules.

3. Shielding requirements for DEMO

The following shielding requirements must be fulfilled for a fusion power reactor: first, the sufficient protection of the super-conducting toroidal field (TF) coils, second the irradiation induced damage accumulation of the vessel needs to be limited to prevent degradation of the stainless steel properties, third, the re-weldability of components and connections/pipes made of steel must be ensured.

Based on existing data, the assumption is that re-welding of stainless steel is should be successful at He concentrations below 1 appm [11]. This limit is assumed for the accumulated He production of components which need to be re-welded during their life-time as for example coolant feeding pipes. Thus it must be shown by 3D neutronic calculations that He accumulations above the level of 1 appm do not occur at locations where re-welding of steel components will be required during the assumed DEMO lifetime. The DEMO design goal is actually to necessitate re-welding only at such locations where sufficient shielding can be provided, e. g. in the vessel ports or at the bottom of the blanket segments.

Another crucial value for in-vessel components is the displacement damage accumulation, which together with the operating temperature, will determine the component lifetime and also has an impact on the choice of the material used. A target limit of 50 dpa (displacements per atom) is assumed for the DEMO first wall made of Eurofer steel [1]. This is actually the limit for the operation of the blanket which translates into a blanket lifetime of 5 fpy. Radiation induced degradation of the material strength is another issue for the austenitic stainless steel assumed for the vacuum vessel. A recent evaluation concluded that the dpa level, accumulated over the full DEMO plant lifetime, should be lower than 2.75 dpa [12] to ensure that the fracture toughness is reduced by no more than 30%. Keeping this limit would allow to operate the assumed DEMO, based on the HCLL and HCPB blankets, for about 5 and 10 fpy, respectively.

The most crucial radiation loads to the TF-coil are the fast neutron fluence to the superconductor, the peak nuclear heating in the winding pack, the radiation damage to the copper insulator and the radiation dose absorbed by the Epoxy resin insulator. The related radiation design limits are the criteria for assessing the shielding efficiency which must be also met at the inboard mid-plane of the reactor where minimum space is available for shielding. Table 1 shows the radiation design limits as elaborated for ITER and DEMO based on the current state-of-the-art [13]. The limits on the neutron fluences result, with the assumed DEMO conditions, in a limit for the fast neutron flux around $1{\cdot}10^9$ $cm^{-2}s^{-1}$. Note the extremely low power density limit of 50 W/m^3 specified for the super-conducting toroidal field coil (TFC) of DEMO.

Table 1. Recommended radiation design limits for super-conducting coils in ITER and DEMO [13]

	ITER	**DEMO**
Total neutron fluence to epoxy insulator $[m^{-2}]$	$1{\cdot}10^{22}$ (equiv. to 10^7Gray)	$\mathbf{1{\cdot}10^{22}}$
Peak fast neutron fluence to the Nb_3Sn super-conductor $[m^{-2}]$	$0.5{\sim}1{\cdot}10^{22}$	$\mathbf{1{\cdot}10^{22}}$
Peak displacement damage to Cu stabilizer between TFC warm-ups $[m^{-2}]$	$1{\sim}2{\cdot}10^{21}$ (equiv. to $0.5{\sim}1{\cdot}10^{-4}$ dpa)	$\mathbf{1{\sim}2{\cdot}10^{21}}$
Peak nuclear heating in winding pack $[W/m^3]$	$1{\cdot}10^3$	$\mathbf{<0.05{\cdot}10^3}$

Shielding analyses have been performed in the frame PPPT 2013 programme for the preliminary design versions of the HCPB, HCLL and WCLL based DEMO. These analyses showed that with the assumed conditions the radiation design limits, specified for DEMO (Table 1), can be met [8-10]. The underlying radial build assumes for the inboard side 70 – 75 cm for the breeder modules with first wall and manifolds, and 55 cm for the vacuum vessel/shield. The assumed conditions include, however, the utilization of an efficient shielding material like WC or borated water which is filled in the vacuum vessel /shield. If such materials are not utilized, the radial dimensions of the shield need to be increased to provide a sufficient shielding of the super-conducting TF coils.

4. Conclusions

Neutronic requirements for DEMO have been discussed in this paper with regard to the Tritium breeding and the shielding performance. Specific requirements were elaborated for the global TBR which need to be provided by a 3D neutronics assessment with a safety margin of 10% above unity. Shield requirements concern the radiation induced material damage, the gas production and the radiation loads to the superconducting TF coils. The nuclear analyses performed in the framework of the 2013 PPPT program on a preliminary DEMO design showed that these limits can be met. Tritium self-sufficiency can be achieved with all considered blanket variants although the current design concepts need to be further optimized.

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