Neutronics and Shielding Results of the U.S. ITER Blanket Trade-Off Study

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Abstract

Detailed neutronics and shielding analyses have been performed for the different ITER blanket design options considered in the ITER blanket option trade-off study. The options considered included a self-cooled Li/V option, a helium cooled Li/V option, ferritic steel designs with helium cooled first wall and liquid metal (Li or LiPb) cooled blanket, the CDA design option, and a water cooled 316 SS nonbreeding shield option. Among the tritium breeding blanket design options, the self-cooled Li/V option has the highest breeding potential while the CDA blanket design option has the best shielding performance. Additional shielding improvement is achieved by using a nonbreeding SS/H_2O shield.

1. Introduction

The first formal phase of the International Thermonuclear Experimental Reactor (ITER) project was the Conceptual Design Activity (CDA) [1]. The CDA blanket utilized Li_2O solid breeder, beryllium neutron multiplier, 316 SS structure and low temperature water coolant [2]. The ITER project embarked on a new phase called the Engineering Design Activity (EDA). A variety of blanket design options have been considered since the start of the EDA. The U.S. ITER home team carried out a blanket option trade-off study (BOTS) to assess the different blanket concepts under consideration. The options considered included

1) A self-cooled Li/V option.
2) A helium cooled Li/V option.
3) Ferritic steel designs with helium cooled first wall and liquid metal (Li or LiPb) cooled blanket.
4) The CDA design option.
5) Water cooled 316 SS nonbreeding shield option.

Option 1 represents a poloidal flow self-cooled liquid lithium design with V5Cr5Ti structure and beryllium neutron multiplier. This design was proposed originally by the U.S. home team. Option 2 is based on an early helium cooled blanket design proposed by the ITER Joint Central Team (JCT). The design utilizes V5Cr5Ti structure, beryllium multiplier and liquid lithium breeder. Option 3 was proposed by S. Malang from KfK [3] and uses ferritic steel as structural material. Helium is used as first wall (FW) coolant and the rest of the blanket is cooled by liquid Li or the $Li_{17}Pb_{83}$ eutectic. Beryllium neutron multiplier is not used in this design option. Natural lithium is considered for the design options with liquid lithium while 90% ^6Li enrichment is assumed for $Li_{17}Pb_{83}$. Option 4 is based on the layered solid breeder design considered during the ITER CDA [2]. The structural material is 316 SS and water is used as coolant. Beryllium is used for neutron multiplication and temperature control and Li_2O with 95% ^6Li enrichment is used as a breeder. Option 5 represents an alternate nonbreeding shield option utilizing water cooled austenitic steel.

Detailed neutronics and shielding analyses have been performed for the different ITER blanket design options. The shielding performance was determined for these options and inboard shielding space requirements were compared. Peak FW nuclear performance parameters (nuclear heating and radiation damage) were calculated. Peak vacuum vessel nuclear performance parameters have also been determined. The overall tritium breeding ratio was determined for each of the design options.

2. Calculational Method

The neutronics calculations have been performed using a one-dimensional (1-D) toroidal cylindrical geometry model with the torus axis used as the axis of symmetry. The inboard (IB) and outboard (OB) regions are modeled simultaneously to account for toroidal effects and mutual neutronic interactions between the two regions. The 1-D discrete ordinates code ONEDANT [4] has been used along with cross section data based on the ENDF/B-V evaluation. A double wall Inconel 625 vacuum vessel (VV) is used with single size water cooled 316 SS balls. For the design options that utilize liquid metals in the blanket, the option of using double size tungsten carbide balls cooled by NaK is also considered. A 5 cm thick back shield zone is used behind the VV. An optimum combination of Pb and B_4C is used in this layer. The minimum total thickness of FW/B/S/VV is maintained at 1 m in the IB region above the reactor midplane. It is assumed that the blanket thickness does not vary poloidally with the VV thickness increasing where enough space is provided. The total OB FW/B/S/VV thickness used in the calculations is 1.15 m. The radial builds used in the calculations are given in Figures 1-5. Since the ITER EDA blanket design is continuously evolving, one should keep in mind that the results given here represent the major differences in the general neutronics and shielding features of these designs with some changes expected if the radial build is significantly changed.

The calculations have been performed for a fusion power of 3 GW and an average neutron wall loading of 2 MW/m^2. The corresponding peak IB and OB wall loadings are 2.64 and 2.78 MW/m^2, respectively. The nominal average fluence of 3 MWa/m^2 with the appropriate poloidal peaking factors has been used to determine the end-of-life neutronics and shielding parameters.

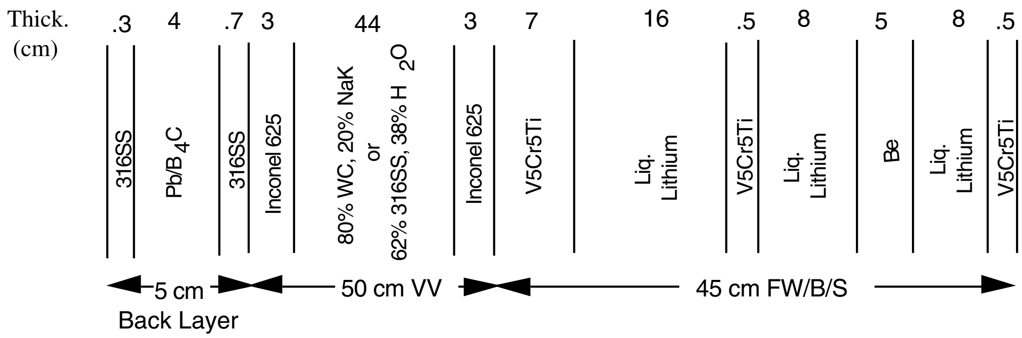

Fig. 1. Radial build of inboard FW/B/S/VV for the self-cooled Li/V design option.

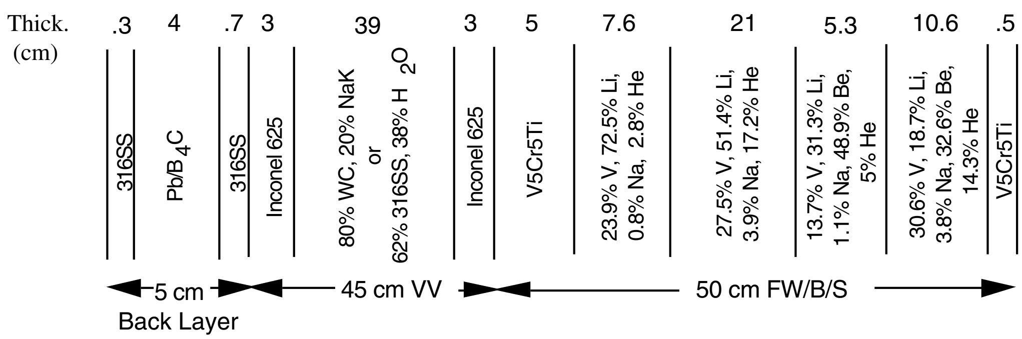

Fig. 2. Radial build of inboard FW/B/S/VV for the helium cooled Li/V design option.

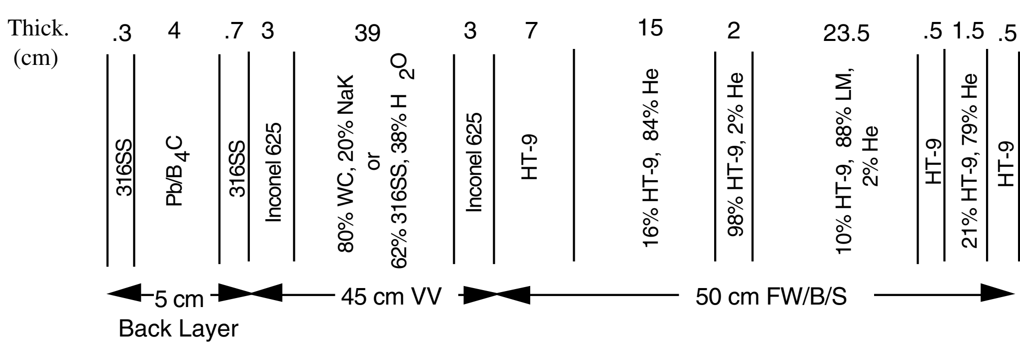

Fig. 3. Radial build of inboard FW/B/S/VV for the ferritic steel design option.

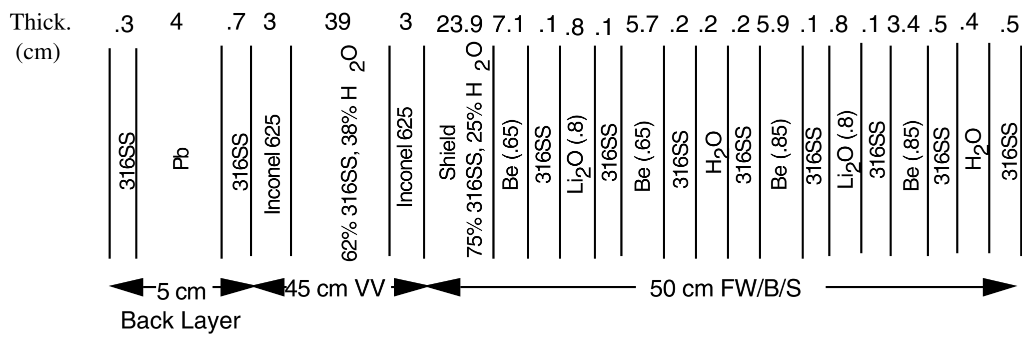

Fig. 4. Radial build of inboard FW/B/S/VV for the CDA design option.

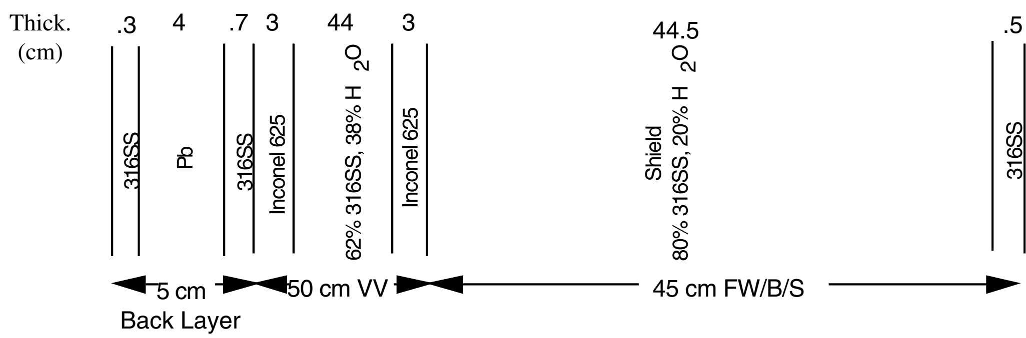

Fig. 5. Radial build of inboard FW/B/S/VV for the nonbreeding SS/H_2O design option.

3. Peak First Wall Neutronics Parameters

The peak nuclear heating, end-of-life dpa and end-of-life helium production in the vanadium structure are given in Table 1 for the self-cooled and helium cooled Li/V design options. The peak values in the IB and OB regions are listed separately. In general, the results in the IB region are higher than those in the OB region due to the toroidal effect. This results mainly from the different angular distributions of source neutrons incident on the IB and OB first walls. Nuclear heating in the FW is higher for the He-cooled design compared to the self-cooled design due to the proximity of Be to the FW and the large amount of structure behind the FW that generates more gamma photons. The FW damage and helium production values are similar for the two designs.

Table 2 gives the peak nuclear heating, end-of-life dpa and end-of-life helium production in the ferritic steel HT-9 structure used in the blanket concept utilizing ferritic steel structure,

Table 1. Peak First Wall Neutronics Parameters for the Li/V Design Options

Coolant	**Liquid Lithium**		**Helium Gas**	
Reactor Zone	**Inboard**	**Outboard**	**Inboard**	**Outboard**
Peak Nuclear Heating in FW (W/cm^3)	15.95	12.40	28.85	22.12
Peak End-of-Life dpa in FW (dpa)	60	46	62	47
Peak End-of-Life Helium Production in FW (appm)	335	269	332	267

Table 2. Peak First Wall Neutronics Parameters for the Ferritic Steel Design Options

Liquid Metal	**$Li_{17}Pb_{83}$**		**Li**	
Li Enrichment	**90% ^6Li**		**natural**	
Reactor Zone	**Inboard**	**Outboard**	**Inboard**	**Outboard**
Peak Nuclear Heating in FW (W/cm^3)	30.46	24.12	34.90	27.07
Peak End-of-Life dpa in FW (dpa)	83	62	63	49
Peak End-of-Life Helium Production in FW (appm)	769	587	780	596

Table 3. Peak First Wall Neutronics Parameters for the CDA and Nonbreeding Design Options

Design Option	**CDA**		**Nonbreeding 316 SS/H_2O**	
Reactor Zone	**Inboard**	**Outboard**	**Inboard**	**Outboard**
Peak Nuclear Heating in FW (W/cm^3)	43.2	33.6	52.5	39.1
Peak End-of-Life dpa in FW (dpa)	61	48	56	44
Peak End-of-Life Helium Production in FW (appm)	917	702	892	680

He-cooled FW and self-cooled liquid metal blanket. The calculations have been performed for two liquid metal options. These are natural lithium and $Li_{17}Pb_{83}$ with 90% ^6Li enrichment. Nuclear heating in HT-9 with LiPb is lower than that with Li due to more neutron slowing down and gamma absorption in Pb. On the other hand, FW damage is higher with LiPb than with Li due to Pb neutron multiplication. Comparing the FW nuclear performance for this design to that for the Li/V design options, one notices that while nuclear heating and atomic displacements are comparable for both V and HT-9 first walls cooled by helium with Li blankets, helium production in a ferritic steel FW is about a factor of two higher than that in a vanadium FW.

The peak nuclear heating, end-of-life dpa and end-of-life helium production in the austenitic steel 316 SS structure are given in Table 3 for the CDA water cooled solid breeder design option and the nonbreeding water cooled steel shield design. The nuclear heating in the FW of the CDA design option is larger than that in the design options considered above due to the much larger amount of Be used. First wall nuclear heating is even higher for the nonbreeding option due to the very large amount of steel (80%) in the region behind the FW that generates more gamma photons. The FW dpa level is comparable to that in designs with either V or ferritic steel structure and liquid Li breeder. On the other hand, helium production in the austenitic steel FW is higher due to the high Ni content in 316 SS.

4. Tritium Breeding

The local tritium breeding ratio (TBR) is calculated for both the IB and OB regions of the breeding design concepts. While no lithium enrichment is considered for liquid lithium, 90% ^6Li enrichment is used for $Li_{17}Pb_{83}$ and 95% ^6Li enrichment is assumed for Li_2O. The overall TBR has been estimated by coupling the 1-D results with the coverage fraction of the different blanket regions. It is assumed that 60% of the fusion neutrons impinge on the OB blanket and 20% go to the IB blanket. The coverage fractions for the radial ports and divertor zone are assumed to be 10% each. Table 4 gives the tritium breeding results for the different design options considered. Again, the toroidal effect results in higher local TBR in the IB blanket compared to the OB blanket

Table 4. Tritium Breeding Results for the Different Design Options

Structure	Breeder	FW Coolant	Blanket Coolant	Local Inboard TBR	Local Outboard TBR	Overall TBR
V5Cr5Ti	Li	Li	Li	1.78	1.46	1.23
V5Cr5Ti	Li	He	He	1.50	1.19	1.02
HT-9	Li	He	Li	1.06	0.86	0.73
HT-9	$Li_{17}Pb_{83}$	He	$Li_{17}Pb_{83}$	1.29	0.96	0.83
316 SS	Li_2O	H_2O	H_2O	1.53	1.26	1.06

of the same thickness. Comparing the results for the two Li/V design options, one notices that tritium breeding is smaller in the He-cooled concept because of the larger structure content used. Tritium breeding for the design options with ferritic steel is smaller than that for the Li/V designs which utilize Be neutron multiplier. Other reasons for the lower TBR are the larger neutron capture in steel compared to V and the larger volume fraction of He gas. The ferritic steel design utilizing LiPb yields higher TBR than that with Li due to neutron multiplication in Pb. The TBR for the CDA design option is lower than that for the self-cooled Li/V design even though much more Be is used because a thinner blanket with less breeder material is utilized. The TBR for the CDA option is comparable to that for the He-cooled Li/V design despite the smaller amount of breeder used because of the very large amount of Be and the much less structure used.

5. Magnet Shielding

The radial builds for the IB FW/B/S/VV have been used to determine magnet shielding performance for the different design options. For each of the VV designs considered, the composition of the 4 cm thick Pb/B_4C zone in the shield back layer was optimized. For the water cooled VV, Pb is used in this zone. On the other hand, 3 cm B_4C and 1 cm Pb layers are used with the NaK cooled VV. The calculations have been performed for the peak IB neutron wall loading of 2.64 MW/m^2 and the peak end-of-life fluence of 3.96 MWa/m^2. The peak magnet radiation effects reported here were determined by increasing the 1-D results by a factor of 3 to

Table 5. Peak Magnet Radiation Effects for the Li/V Blanket Design Options

VV Filling Material	**62% 316 SS, 38% H_2O**		**80% WC, 20% NaK**	
Back Shield Layer	**Pb**		**75% B_4C, 25% Pb**	
Design Option	**Self-Cooled**	**He-Cooled**	**Self-Cooled**	**He-Cooled**
Peak Nuclear Heating in Magnet (mW/cm^3)	9.5	7.7	4.1	4.2
Peak Fast Neutron Fluence in Magnet (n/cm^2)	1.6×10^{19}	1.3×10^{19}	4.7×10^{18}	4.7×10^{18}
Peak Organic Insulator Dose (Rads)	1.7×10^{10}	1.3×10^{10}	5.6×10^{9}	5.6×10^{9}
Peak Cu dpa in Magnet (dpa)	1.1×10^{-2}	8.5×10^{-3}	2.8×10^{-3}	2.8×10^{-3}
Total Magnet Nuclear Heating (kW)	185	150	72	71

account for assembly gaps between adjacent modules and uncertainties in nuclear data and modeling. The estimated total magnet nuclear heating was determined by taking into account the poloidal variations of neutron wall loading and FW/B/S/VV thickness. It includes also the estimated contributions from the divertor region and radial ports. The total nuclear heating values include a safety factor of 2.

Table 5 gives the peak magnet radiation effects for the Li/V blanket design options. It is clear that the shielding performance is improved by using WC balls in a NaK cooled VV compared to the simple water cooled steel balls. Replacing the water by NaK while keeping the steel balls results in increasing magnet radiation effects by a factor of ~50. Hence, if liquid metal coolant is to be used instead of water in the VV because of safety concerns, good shielding material such as WC should be used for the filling balls. The cost penalty is not great since this will be needed only over a height of about 4 m of the IB region. Assuming unit costs of ~\$65/kg for WC and ~\$15/kg for 316 SS, the cost penalty is ~\$30M. The results in Table 5 indicate that the shielding performance is about the same for the self-cooled and He-cooled designs due to the larger structure content in the He-cooled design that compensates for the shielding space lost by the He coolant channels. An additional order of magnitude attenuation is achieved by increasing the

Table 6. Peak Magnet Radiation Effects for the Ferritic Steel Blanket Design Options

VV Filling Material	**62% 316 SS, 38% H_2O**		**80% WC, 20% NaK**	
Back Shield Layer	**Pb**		**75% B_4C, 25% Pb**	
Breeder	**$Li_{17}Pb_{83}$**	**Li**	**$Li_{17}Pb_{83}$**	**Li**
Peak Nuclear Heating in Magnet (mW/cm^3)	5.2	13.0	4.6	7.4
Peak Fast Neutron Fluence in Magnet (n/cm^2)	5.9×10^{18}	2.1×10^{19}	3.0×10^{18}	7.9×10^{18}
Peak Organic Insulator Dose (Rads)	7.4×10^{9}	2.2×10^{10}	4.9×10^{9}	9.7×10^{9}
Peak Cu dpa in Magnet (dpa)	3.8×10^{-3}	1.4×10^{-2}	1.7×10^{-3}	4.7×10^{-3}
Total Magnet Nuclear Heating (kW)	83	248	66	125

thickness of the SS/H_2O VV by 18 cm or the WC/NaK VV by 14 cm. Based on the CDA magnet design limits of 55 kW total heating and 5×10^9 Rads insulator dose, about 2 cm thicker WC/NaK VV or ~8 cm thicker SS/H_2O VV are needed. For a total nuclear heating limit of 20 kW, the additional thicknesses needed will be 8 cm and 16 cm for WC/NaK and SS/H_2O, respectively. If the insulator dose limit is reduced significantly below the CDA limit, additional shielding space will be required.

The peak magnet radiation effects for the ferritic steel blanket options are listed in Table 6. The shielding performance with LiPb is better than that for the Li/V options due to the better shielding characteristics of enriched LiPb compared to natural Li (neutron slowing down in Pb and neutron capture in Li) and the better shielding effectiveness of steel compared to vanadium. The shielding performance with Li is worse than that for the Li/V designs since no Be is used and a larger amount of He gas is utilized. Based on the CDA magnet design limits of 55 kW total heating and 5×10^9 Rads insulator dose, about 1 cm thicker WC/NaK VV or ~3 cm thicker SS/H_2O VV are needed for the blanket design utilizing LiPb. About 5 cm thicker WC/NaK VV or ~12 cm thicker SS/H_2O VV are needed for the blanket design utilizing Li. Additional shielding space will be required for lower EDA nuclear heating and insulator dose limits.

Table 7. Peak Magnet Radiation Effects for the CDA and Nonbreeding Design Options

VV Filling Material	**62% 316 SS, 38% H_2O**	
Back Shield Layer	**Pb**	
Design Option	**CDA**	**Nonbreeding**
Peak Nuclear Heating in Magnet (mW/cm^3)	1.1	0.3
Peak Fast Neutron Fluence in Magnet (n/cm^2)	1.9×10^{18}	4.6×10^{17}
Peak Organic Insulator Dose (Rads)	2.0×10^{9}	4.8×10^{8}
Peak Cu dpa in Magnet (dpa)	1.3×10^{-3}	3.1×10^{-4}
Total Magnet Nuclear Heating (kW)	20	5

Table 7 gives the peak magnet radiation effects for the CDA blanket design concept and the nonbreeding water cooled austenitic steel shield. The results show much better shielding performance for the CDA concept compared to the Li/V options and the ferritic steel designs that utilize either Li or LiPb. The reason is that the CDA blanket/shield utilizes materials with good shielding characteristics such as SS, water, and Be. Meanwhile, the other blanket design options include materials with worse shielding characteristics such as liquid metals, V, and He. In addition, the blanket thickness (26.1 cm) and radial build used is the same as that for the US CDA layered blanket design and the blanket is followed by a 23.9 cm thick water/SS shield to yield a total FW/B/S thickness of 50 cm similar to designs considered by the JCT. Hence the thickness of the breeding zone for the CDA concept is smaller than that for the other blanket concepts considered. The best shielding performance is obtained for the nonbreeding SS/H_2O shield due to elimination of breeding blanket. Based on the CDA magnet design limits, the VV thickness can be reduced by 7 and 18 cm for the CDA and nonbreeding design options, respectively. For a total nuclear heating limit of 20 kW, the VV thickness cannot be reduced for the CDA blanket option and can be reduced by only 11 cm for the nonbreeding option.

Table 8. Peak Neutronics Parameters in Vacuum Vessel for the Li/V Blanket Options

VV Filling Material	62% 316 SS, 38% H_2O		80% WC, 20% NaK	
Back Shield Layer	Pb		75% B_4C, 25% Pb	
Design Option	Self-Cooled	He-Cooled	Self-Cooled	He-Cooled
Peak Nuclear Heating (W/cm^3)	6.5	4.1	5.2	3.3
Peak End-of-Life Damage (dpa)	4.4	2.2	5.1	2.6
Peak End-of-Life He appm	50	20.2	50	20.2

Table 9. Peak VV End-of-Life Helium Production (He appm) for the Different Blanket Designs

Self-Cooled Li/V	50
He-Cooled Li/V	20.2
Ferritic Steel/He/Li	32.7
Ferritic Steel/He/LiPb	8
CDA Concept	2.9
Nonbreeding SS/H_2O Shield	1.3

6. Vacuum Vessel Neutronics Parameters

The peak neutronics parameters in the VV are given in Table 8 for the Li/V blanket design options. No safety factors are included in these results. The peak nuclear heating and damage levels in the VV are higher with the self-cooled blanket which has less structure and is 5 cm thinner than the He-cooled blanket. The VV coolant and filling material do not impact the peak helium production in the front Inconel 625 VV wall since low energy backscattered neutrons do not contribute to the high energy helium production reactions. Table 9 lists the peak VV end-of-life helium production for the different design options considered. Again, the lowest VV damage is achieved by using a nonbreeding steel/water shield. On the other hand, the self-cooled Li/V blanket design, which has the least shielding capability among the different concepts considered in this study, yields the highest VV damage. It is clear that VV reweldability is questionable particularly for the options with breeding blankets. Thinning the VV and using additional shield will be necessary if VV reweldability is required.

7. Summary

Detailed neutronics and shielding analyses have been performed for the different ITER blanket design options considered in the U.S. ITER blanket option trade-off study. The Options considered included a self-cooled Li/V option, a helium cooled Li/V option, ferritic steel designs with helium cooled first wall and liquid metal (Li or LiPb) cooled blanket, the CDA design option, and a water cooled 316 SS nonbreeding shield option.

Nuclear heating in the FW is higher for the He-cooled Li/V design compared to the self-cooled Li/V design. While nuclear heating and atomic displacements are comparable for both V and HT-9 first walls, helium production in a ferritic steel FW is a factor of two higher than in a vanadium FW. The nuclear heating in the FW of the CDA design option is larger than that in the design options utilizing liquid metals and V or ferritic steel structure. Helium production in the austenitic steel FW is higher due to the high Ni content in 316 SS. Tritium breeding is smaller in the He-cooled Li/V concept than in the self-cooled Li/V concept. Tritium breeding for the design options with ferritic steel is smaller than that for the Li/V designs. The ferritic steel design utilizing LiPb yields higher TBR than that with Li. The TBR for the CDA design option is lower than that for the self-cooled Li/V design and is comparable to that for the He-cooled Li/V design.

The shielding performance is about the same for the self-cooled and He-cooled Li/V designs. The shielding performance for the ferritic steel blanket option with LiPb is better than that for the Li/V options. The results show much better shielding performance for the CDA concept compared to the Li/V options and the ferritic steel designs that utilize either Li or LiPb. The best shielding performance is obtained for the nonbreeding SS/H_2O shield due to the elimination of the breeding blanket. The lowest VV damage is achieved by using a nonbreeding steel/water shield. On the other hand, the self-cooled Li/V blanket design, which has the least shielding capability among the different concepts, yields the highest VV damage.

It is concluded that among the tritium breeding blanket design options, the self-cooled Li/V option has the highest breeding potential while the CDA blanket design option has the best shielding performance. Additional shielding improvement is achieved by using a nonbreeding

SS/H_2O shield. These conclusions are based on one-dimensional calculations. Although the toroidal effects are accounted for in the model, three-dimensional effects resulting from the finite vertical extent of the plasma and the influence of materials used in the divertor region are not included and the results need to be confirmed by three-dimensional calculations.

Acknowledgment

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