Three-Dimensional Neutronics Assessment of Dual Coolant Molten Salt Blankets with Comparison to One-Dimensional Results

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Abstract

Detailed three-dimensional (3-D) neutronics calculations have been performed for the dual coolant molten salt blanket designs. The model includes the detailed heterogeneous geometrical arrangement of the inboard (IB) and outboard (OB) blanket sectors. The total tritium breeding ratio (TBR) is ~1.07. The 1-D calculations tend to overestimate nuclear heating in the blanket by ~8% resulting in overestimating the plant thermal power. The approximate 1-D calculations overestimate damage and nuclear heating in the first wall (FW) and front zone of the blanket by factors of 1.3-1.7 but result in a steeper radial drop leading to significant underestimation (by up to a factor of 3) of nuclear parameters at the back. The 1-D calculations significantly underestimate radiation damage in the shield and vacuum vessel behind the blanket.

1. Introduction

As candidate blanket concept for a U.S. Advanced Power Plant design we assessed blanket design concepts based on the use of reduced activation ferritic steel (RAFS) as structural material and liquid breeders as the coolant and tritium breeder [1]. Molten salts (MS) have been considered as breeding material and coolant candidates in fusion systems [2,3]. Flibe, consisting of LiF and BeF_2 with a mole ratio of 2:1, has been widely considered. It has the attractive features of low activation, low chemical reactivity with air and water, low electrical conductivity, and good neutron attenuation properties. On the other hand, it has a relatively high melting point (459°C), low thermal conductivity, tritium permeation concern, and requires control of the corrosive TF and F_2 [4]. A low melting point Flibe (380°C) was also considered but it has higher viscosity. The molten salt Flinabe that consists of LiF, BeF_2 and NaF has recently been considered due to its low melting point (305°C) and vapor pressure [5]. The breeding capability of the molten salt is limited, requiring separate neutron multiplier. Due to the smaller Li content in Flinabe, a thicker separate multiplier zone is required [6,7].

An attractive design option was identified based on the dual coolant (DC) concept with helium cooling the first wall (FW) and blanket structure, Flibe breeder, and Be neutron multiplier [1]. Special attention is given to concepts that can be developed, qualified and tested in the time frame of ITER. Hence, the conventional LAFS alloy F82H [8] with a temperature limit of 550°C is considered. The low electrical conductivity of the MS implies that there is no need for separate MHD insulator in the coolant channels. In addition, the low thermal conductivity of the MS together with the suppression of turbulence by the magnetic field reduce the heat losses from the breeder to the actively cooled steel structure, allowing MS bulk temperatures higher than the structure temperature with the potential for higher power plant performance while keeping the metallic structure within its temperature limit. To avoid MS freezing, the low melting point Flibe that has a mole ratio of 1:1 is used. The MS Flinabe is also considered. In this paper, detailed 3-D neutronics analysis is presented for the DC-MS blanket concepts with the low melting point Flibe ($LiBeF_3$) and Flinabe. The results are compared to previous results based on 1-D calculations [9] to shed light on the impact of accurate 3-D modeling on the neutronics performance features.

2. Description of blanket concept

The design configuration of the DC-MS blanket is shown in Fig. 1. Behind the 3 mm thick FW of the module we have the toroidally oriented helium cooling channels. The concept has a Be pebble bed arranged at the front between two poloidal MS channels and cooled directly with MS flowing through the pebble bed in the radial direction. We then have the helium cooled poloidal channels forming the large channels for the MS to flow in the poloidal direction. Figure 2 illustrates the radial build of the design in the OB side.

3. Calculation model

The assessment is performed for an advanced tokamak power plant with 2116 MW fusion power, 5.8 m major radius, and aspect ratio of 2.6. 3-D neutronics calculations were performed using the continuous energy, coupled neutron-gamma Monte Carlo code MCNP, version 5 [10] along with nuclear data based on the FENDL-2 evaluation [11]. The model includes the detailed heterogeneous geometrical arrangement of the IB and OB blanket sectors that are 40 and 65 cm thick, respectively. Because of symmetry only 1/128 of the chamber is modeled (1/4 of a sector) with reflecting boundaries. Figure 3 gives a vertical cross section in the model showing the IB and OB blankets and Figure 4 shows a horizontal cross section at mid-plane. The plasma minor radius is 2.23 m and the IB and OB FW radii are 3.47 and 8.13 m, respectively, at mid-plane. The total FW areas in the IB and

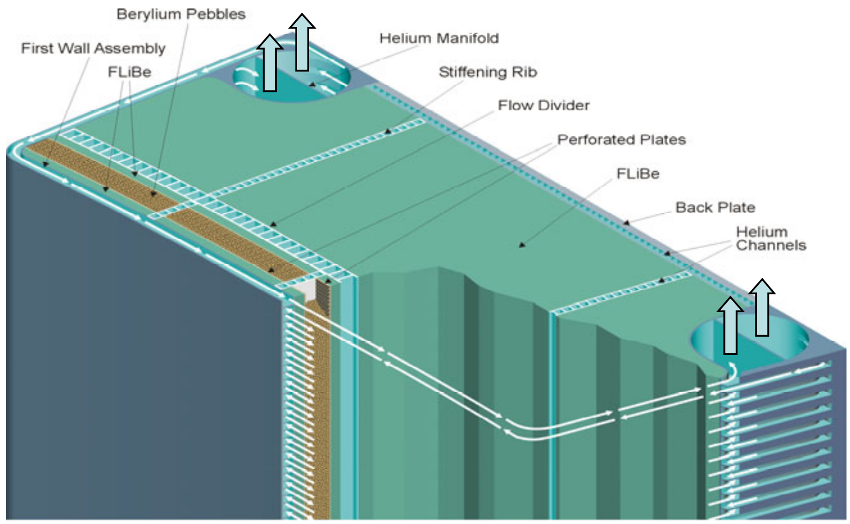

Fig. 1. Design configuration of DC-MC blanket showing He flow circuit.

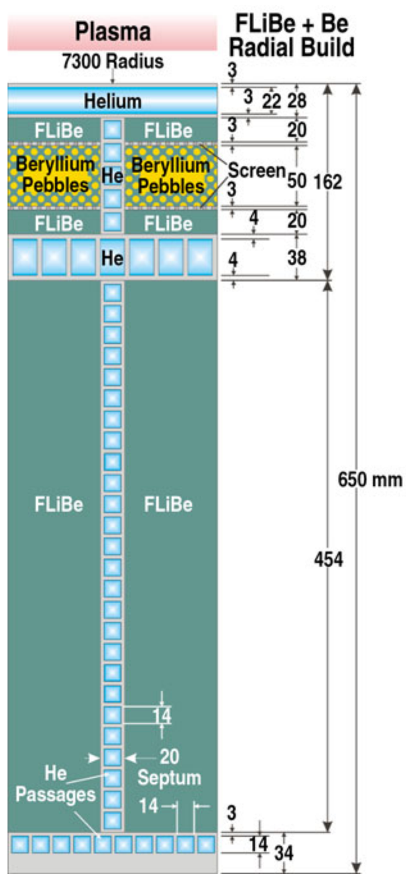

Fig. 2. Radial build dimensions for the OB DC-MS blanket design.

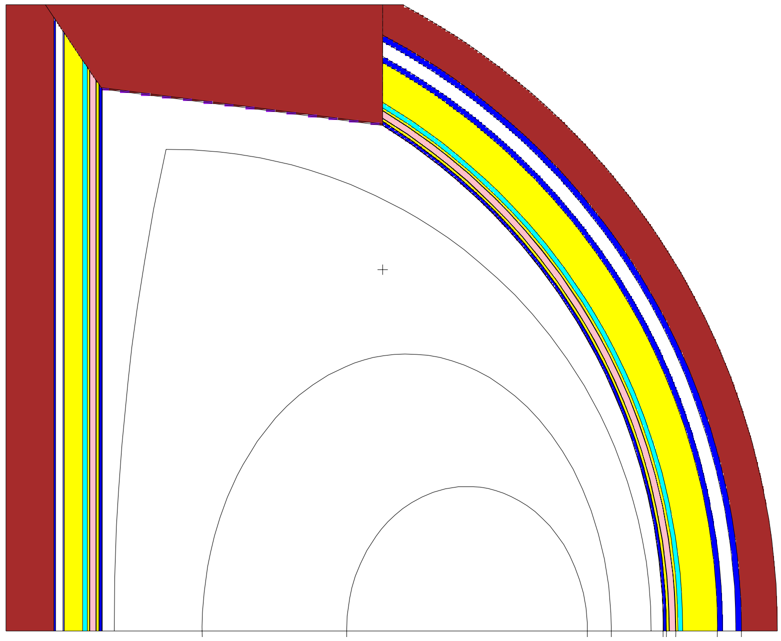

Fig. 3. Vertical cross section in the 3-D model.

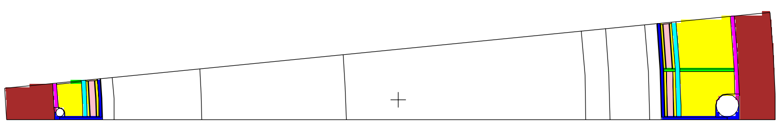

Fig. 4. Horizontal cross section at mid-plane.

OB regions are 196 and 461 m^2, respectively. The neutron source is sampled from the D-shaped plasma using a peaked distribution at the magnetic axis. One million source particles were sampled and variance reduction techniques were utilized to yield statistical uncertainties <0.1% in the calculated overall parameters and <1% in the local parameters.

Based on the 3-D calculations, the peak neutron wall loading values in the IB and OB regions are 2.14, and 3.72 MW/m^2, respectively. The poloidal variation of neutron wall loading is given in Fig. 5. The corresponding average values are 1.33 and 2.66 MW/m^2, respectively. Since we do not have a divertor design, the 3-D model used a conservative assumption by including water-cooled steel with 1 cm tungsten armor in the double null divertor region. The lithium is enriched to 50% Li-6 in Flibe and 60% Li-6 in Flinabe. The Be multiplier zone thickness is 5 cm with Flibe and 8 cm with Flinabe. A water-cooled steel shield is included behind the blanket to account for neutron reflection from the vacuum vessel and shield.

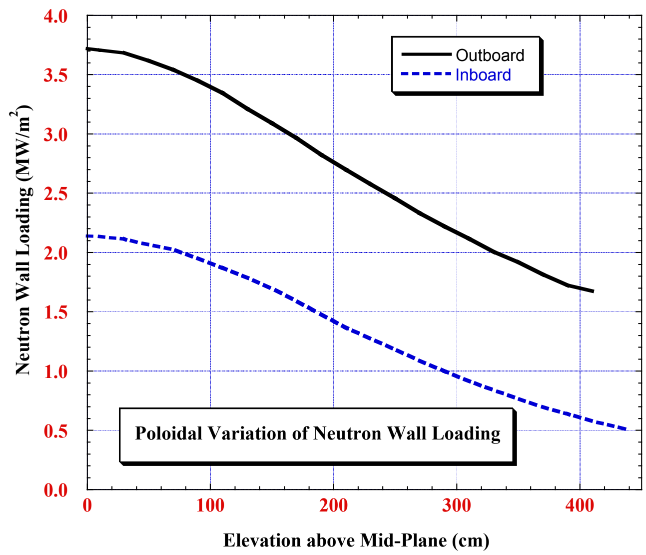

Fig. 5. Poloidal neutron wall loading distribution.

4. Tritium production

Table 1 lists the calculated tritium production in the IB and OB blanket zones. In the Flibe blanket, ~55% of tritium breeding occurs in the large Flibe breeder channels. On the other hand, ~56% of the tritium is bred in the Flinabe cooling the Be multiplier zone in the Flinabe design because of the thicker multiplier zone. The OB blanket contributes ~80% of the total TBR in both designs. The calculated TBR is a conservative estimate since it assumes no breeding in the double null divertor zones (which could be utilized for partial breeding) on which 12% of the source neutrons impinge. Minor design modifications such as increasing the Be zone and/or blanket thickness can be made to enhance the TBR if needed to ensure tritium self-sufficiency. For example, increasing the Be zone thickness from 5 to 8 cm resulted in enhancing the TBR to 1.15 for the Flibe design as illustrated in Fig. 6. Direct cooling of the Be pebble bed by the MS allows varying the multiplier zone thickness without violating the Be temperature limit.

Table 2 compares the TBR results obtained from 3-D calculations to those estimated from 1-D calculations. The 1-D calculations are based on a toroidal cylindrical geometry model (discussed in detail elsewhere [9]) in which the IB and OB blankets extend indefinitely in the vertical direction (no divertor) with a uniform neutron source extended in the vertical direction (no source peaking at mid-plane). While material composition in each radial zone used in the 1-D model was carefully determined to account for the toroidal material arrangement shown in Fig. 1 [9], the TBR based on 1-D local TBR values coupled with blanket coverage fractions (72.6% OB, 15.4% IB) is ~6.3% higher than the calulated 3-D value. Therefore, the combined effects of blanket and source 3-D configurations and detailed blanket heterogeniety modeling can lead to more than ~6% lower TBR compared to 1-D estimates.

A critical issue associated with using Be in fusion blankets is the amount of tritium produced and retained in the beryllium. Tritium is produced in the Be pebbles used in the IB and OB modules at the rate of 0.512 kg/FPY for the Flibe blanket and 0.71 kg/FPY for the Flinabe blanket that has 60% more Be. About 83% of this amount is contributed by the OB blanket. Comparing the results with the estimates based on 1-D calculations indicated that the 1-D approximation overestimates the tritium

Table 1. Tritium production (tritons per fusion)

			Dual Coolant Flibe Blanket	Dual Coolant Flinabe Blanket
Outboard Region	Multiplier Zone	Front Coolant Channel	0.1237	0.1194
		Multiplier Front Wall	0.0024	0.0024
		Be Pebble Bed Region	0.1374	0.2374
		Multiplier Back Wall	0.0020	0.0019
		Back Coolant Channel	0.0958	0.0867
		Total	0.3613	0.4478
	Breeder Zone		0.4899	0.3958
	Total Outboard		0.8512	0.8436
Inboard Region	Multiplier Zone	Front Coolant Channel	0.0445	0.0430
		Multiplier Front Wall	0.0008	0.0008
		Be Pebble Bed Region	0.0443	0.0743
		Multiplier Back Wall	0.0006	0.0006
		Back Coolant Channel	0.0289	0.0249
		Total	0.1191	0.1436
	Breeder Zone		0.1002	0.0737
	Total Inboard		0.2193	0.2173
Total Overall TBR			1.0705	1.0609

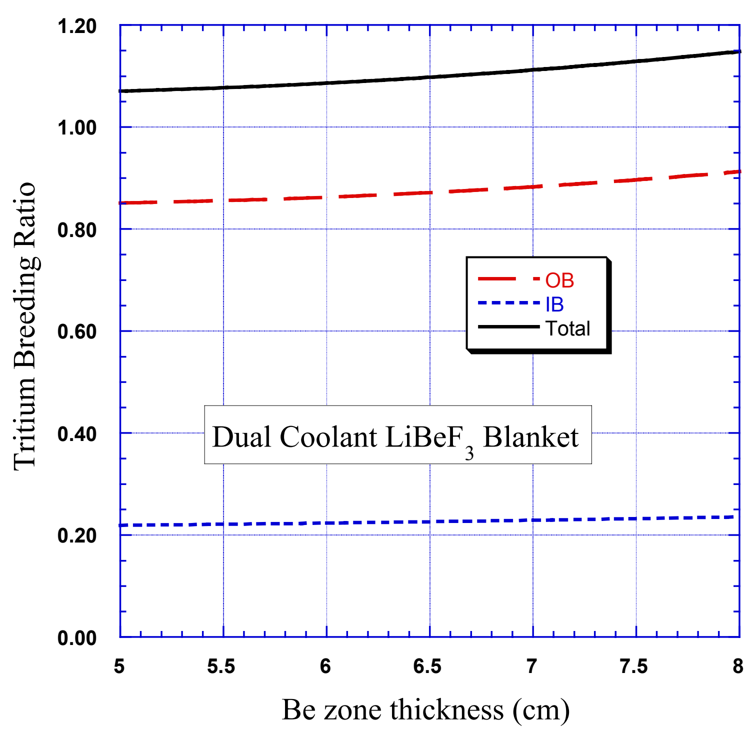

Fig. 6. Impact of multiplier zone thickness on TBR for Flibe blanket.

Table 2. Comparison between TBR results estimated from 1-D and from 3-D calculations

	Dual Coolant Flibe Blanket		Dual Coolant Flinabe Blanket	
	3-D	1-D	3-D	1-D
Outboard Region	0.8512	0.9111	0.8436	0.9104
Inboard Region	0.2193	0.2172	0.2173	0.2165
Total Overall TBR	1.0705	1.1383	1.0609	1.1269

Table 3. Blanket energy multiplication

	Dual Coolant Flibe Blanket		Dual Coolant Flinabe Blanket	
	3-D	1-D	3-D	1-D
Outboard Region	1.111	1.200	1.123	1.230
Inboard Region	1.256	1.300	1.269	1.330
Average	1.136	1.223	1.148	1.247

Table 4. Peak power density (W/cm^3) in FW structure at mid-plane

	Dual Coolant Flibe Blanket		Dual Coolant Flinabe Blanket	
	3-D	1-D	3-D	1-D
Outboard Region	25.6	37.8	26.2	37.9
Inboard Region	20.6	26.5	21.1	26.7

production rate in Be by ~13%. The tritium inventory over the life of the blanket will be much lower than the tritium production due to tritium permeation out of Be at the high Be operating temperatures and during possible frequent bake-outs.

5. Nuclear heating

Table 3 compares the calculated blanket energy multiplication values. Notice that energy multiplication in the Flinabe blanket with thicker Be zone is slightly higher than that in the Flibe blanket. The total nuclear heating in the IB and OB blankets is 1693 MW for Flibe and 1711 MW for Flinabe. The energy multiplication in the IB blanket is ~13% higher than in the OB blanket since neutrons incident on the IB FW are mostly tangential resulting in more interactions in the front multiplier zone and more gamma generation in the front structure. The 1-D calculations tend to overestimate nuclear heating in the blanket by ~8% resulting in overestimating the plant thermal power.

The average power density values in the three OB MS coolant channels are 19.3, 12.3, and 2.7 W/cm^3. Table 4 gives the peak power density values in the FW structure as calculated by the 3-D and 1-D models. The 1-D calculations result in overestimating the peak FW power density by a factor of ~1.5 in OB and ~1.3 in IB. This is primarily due to the approximate angular distribution of source neutrons incident on the FW from the infinitely extended uniform source in the 1-D model that results in more tangentially incident neutrons compared to the actual 3-D model with neutron source peaked at mid-plane. This results also in a steeper radial drop in power density predicted by the 1-D

Table 5. Radiation damage in FW structure of Flibe blanket at mid-plane

	Outboard Region		Inboard Region	
	3-D	1-D	3-D	1-D
Peak dpa/FPY	28.1	48.4	19.9	30.9
Peak He appm/FPY	356	625	243	384

Table 6. Radiation damage at front of shield behind Flibe blanket

	Outboard Region		Inboard Region	
	dpa/FPY	He appm/FPY	dpa/FPY	He appm/FPY
Peak behind Manifold at Mid-plane	0.62	4.53	1.48	11.85
Poloidal Average behind Manifold	0.50	3.61	1.17	8.82
Average behind Blanket	0.20	0.86	0.65	4.57

calculations. This is illustrated by comparing the toroidal average results in the back wall at mid-plane that indicate ~8% lower estimates from the 1-D calculations.

6. Radiation damage

Table 5 gives the peak FW damage rates at mid-plane in the Flibe blanket from both the 3-D and 1-D calculations. The 1-D calculations overestimate the peak FW radiation damage rate by factors of ~1.7 in the OB and ~1.5 in the IB. Again, this is primarily due to the more tangential source neutrons incident on the FW from the infinitely extended uniform source in the approximate 1-D model. Similar differences were observed in previous studies [12]. Assuming a lifetime radiation damage limit of 200 dpa for the RAFS structure, the blanket lifetime is expected to be ~7 full power years (FPY) based on the 3-D results.

We calculated the radiation damage rate in the front 3 cm zone of the shield at different locations behind the blanket. The results are given in Table 6. The peak cumulative end-of-life (30 FPY) dpa in the shield structure is 45 dpa and occurs in the IB region at mid-plane. The shield is therefore expected to be a lifetime component. Peaking factors of 3.1 in OB and 2.3 in IB occur for the dpa rate behind the manifolds. Higher peaking factors (5.3 in OB, 2.6 in IB) are obtained for helium production rate. The peaking factors are smaller in the IB side that has a thinner blanket. The approximate 1-D calculations underestimate the average dpa rate at the front of the shield by a factor of ~3 compared to that obtained from the 3-D calculation. When combined with peaking factors obtained due to the 3-D geometrical heterogeneity effects, it is concluded that 1-D calculations significantly underestimate radiation damage in the shield and vacuum vessel behind the blanket. Large design margins should be allowed when 1-D calculations are used in shielding assessment.

7. Summary and conclusions

Detailed 3-D neutronics calculations have been performed for the dual coolant molten salt blanket designs with the low melting point Flibe or Flinabe in a tokamak power plant configuration. The model includes the detailed heterogeneous geometrical arrangement of the blanket sectors. The Be multiplier zone thickness is 5 cm with Flibe and 8 cm with Flinabe. The 3-D model includes water-cooled steel with tungsten armor in the divertor region. The total TBR was determined to be ~1.07. This is a conservative estimate since it assumes no breeding in the double null divertor zones on which 12% of the source neutrons impinge. We demonstrated that minor design modifications such as increasing the Be zone thickness can be made to enhance the TBR if needed to ensure tritium self-sufficiency. We conclude that the DC-MS design concept has the potential for achieving tritium self-sufficiency. The calculated TBR that accounts for heterogeneity and 3-D geometrical effects is ~6% lower than estimates based on 1-D calculations. The 1-D calculations tend to overestimate nuclear heating in the blanket by ~8% resulting in overestimating the plant thermal power. The peak dpa and helium production rates in the structure are 28 dpa/FPY and 356 appm/FPY, respectively, and occur in the OB blanket at mid-plane. Comparing the 3-D results with the 1-D results indicates that the approximate 1-D calculations overestimate damage and nuclear heating in the FW and front zone of the blanket by factors of 1.3-1.7. However, the 1-D calculations result in a steeper radial drop in nuclear parameters leading in significant underestimation (by up to a factor of 3) of radiation effects at the back of the blanket. When combined with peaking factors of up to ~3 obtained due to the 3-D geometrical heterogeneity effects, it is concluded that 1-D calculations significantly underestimate radiation damage in the shield and vacuum vessel behind the blanket and large design margins should be allowed when 1-D calculations are used in shielding assessment.

Acknowledgement

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References

[1] C. Wong, S. Malang, M. Sawan, et al., Assessment of First Wall and Blanket Options with the Use of Liquid Breeder, Fusion Science & Engineering 47(3) (2005) 502-509.

[2] M.A. Abdou, A. Ying, N. Morley, et al., On the Exploration of Innovative Concepts for Fusion Chamber Technology, Fusion Engineering & Design 54 (2001) 181-247.

[3] A. Sagara, H. Yamanishi, T. Uda, et al., Studies on Flibe Blanket Designs in Helical Reactor FFHR, Fusion Technology 39 (2001) 753-757.

[4] M. Sawan and D.-K. Sze, Transmutation and Production Rates of Elements in Flibe and Flinabe with Impact on Chemistry Control, Fusion Science & Technology 44 (2003) 64-68.

[5] J. McDonald, R. Nygren, T. Lutz, et al., Measurement of the Melting Point Temperature of Several Lithium-Sodium-Beryllium Fluoride Salt (Flinabe) Mixtures, Fusion Science & Engineering 47(3) (2005) 555-559.

[6] M. Youssef, M. Sawan, and D.-K. Sze, The Breeding Potential of Flinabe and Comparison to Flibe in CLiFF High Power Density Concept, Fusion Engineering & Design 61-62 (2002) 497-503.

[7] E. Cheng and D-K. Sze, Nuclear Aspects of Molten Salt Blankets, Proceedings of 22nd Symposium on Fusion Technology, Helsinki, Finland, September 9-13, 2002.

[8] S. Zinkle, J. Robertson, and R. Klueh, Thermophysical and Mechanical Properties of Fe-(8-9)%Cr Reduced Activation Steels, Fusion Materials Semiannual Progress Report for the Period Ending June 30, 2000 (DOE/ER-0313/28), pp. 135-143 (2000).

[9] M. Sawan, S. Malang, C. Wong, and M. Youssef, Neutronics Assessment of Molten Salt Breeding Blanket Design Options, Fusion Science & Engineering 47(3) (2005) 510-517.

[10] X-5 Monte Carlo Team, MCNP-A General Monte Carlo N-Particle Transport Code, Version 5-Volume II: Users Guide, LA-CP-03-0245, Los Alamos National Laboratory (April 2003).

[11] M. Herman and H. Wienke, FENDL/MG-2.0 and FENDL/MC-2.0, The Processed Cross-Section Libraries For Neutron-Photon Transport Calculations, Report IAEA-NDS-176, Rev. 3, International Atomic Energy Agency (October 1998).

[12] L.A. El-Guebaly and the ARIES Team, Three-Dimensional Neutronics Study for ARIES-ST Power Plant, Fusion Technology 34 (1998) 1084-1088.