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NEUTRONICS ASSESSMENT OF MOLTEN SALT BREEDING BLANKET DESIGN OPTIONS

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Neutronics assessment has been performed for molten salt breeding blanket design options that can be utilized in fusion power plants. The concepts evaluated are a self-cooled Flinabe blanket with Be multiplier and dual-coolant blankets with He-cooled FW and structure. Three different molten salts were considered including the high melting point Flibe, a low melting point Flibe, and Flinabe. The same TBR can be achieved with a thinner self-cooled blanket compared to the dual-coolant blanket. A thicker Be zone is required in designs with Flinabe. The overall TBR will be ~1.07 based on 3-D calculations without breeding in the divertor region. Using Be yields higher blanket energy multiplication than obtainable with Pb. A modest amount of tritium is produced in the Be (~3 kg) over the blanket lifetime of ~3 FPY. Using He gas in the dual-coolant blanket results in about a factor of 2 lower blanket shielding effectiveness. We show that it is possible to ensure that the shield is a lifetime component, the vacuum vessel is reweldable, and the magnets are adequately shielded. We conclude that molten salt blankets can be designed for fusion power plants with neutronics requirements such as adequate tritium breeding and shielding being satisfied.

I. INTRODUCTION

Molten salts have been considered as breeding material and coolant candidates in fusion systems.[1,2] 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 tritium retention, small density change on melting, low chemical reactivity with air and water, and low electrical conductivity which alleviates the MHD problems encountered in magnetic confinement fusion systems. In addition, Flibe has good neutron attenuation properties. On the other hand, it has a relatively high melting point (459°C), low thermal conductivity resulting in reduced heat transfer capability, tritium permeation concern, and requires control of the corrosive TF and F_2.[3] A low melting point Flibe (380°C) with 1:1 mole ratio was also considered but it has much higher viscosity. The breeding capability of Flibe is limited, requiring a separate neutron multiplier such as beryllium or lead. 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.[4,5] Due to the smaller Li content, a thicker separate multiplier zone is required.[6,7]

A study is underway to identify attractive molten salt breeding blanket concepts that can be utilized in fusion power plants. Special attention is given to concepts that can be developed, qualified and tested in the time frame of ITER. For this reason, conventional reduced activation ferritic steel (RAFS) alloys like F82H with a temperature limit of 550°C are considered as the structural material. Three blanket concepts were evaluated. The first concept is a self-cooled Flinabe blanket with Be multiplier (SC). It uses an innovative re-circulating coolant scheme, which allows effective cooling of the first wall (FW) while enhancing the coolant outlet temperature.[8] The other concepts are dual-coolant options with helium cooling the FW and blanket structure, Flibe breeder, and either Be (DC-Be) or lead (DC-Pb) as neutron multiplier.[9] In this paper, a brief description of the blanket concepts is given and the neutronics assessment of the three concepts is presented. The assessment is based on the ARIES-RS[10] configuration with a peak neutron wall loading of 5.45 MW/m^2 and a maximum surface heat flux of 1 MW/m^2 as a common basis.

II. DESCRIPTION OF BLANKET CONCEPTS

II.A. Self-Cooled Flinabe Blanket Concept (SC)

The use of molten salt as breeder and coolant enables simple self-cooled blankets without the need for electrical insulators between coolant and walls since the electrical conductivity is orders of magnitude lower than that of lead lithium or lithium. Using the low melting point Flinabe allows keeping the structure temperature below the 550°C limit. The concept uses an innovative re-circulating coolant scheme that enhances the coolant outlet temperature, and thus improves the power conversion cycle efficiency. Figure 1 shows a cut in an outboard (OB) blanket module. The FW is shown as a sheet of five scallops facing the plasma with five coolant channels behind them. The FW assembly is attached to the remaining blanket consisting of a box with channels on the sides and back, surrounding a large central channel.

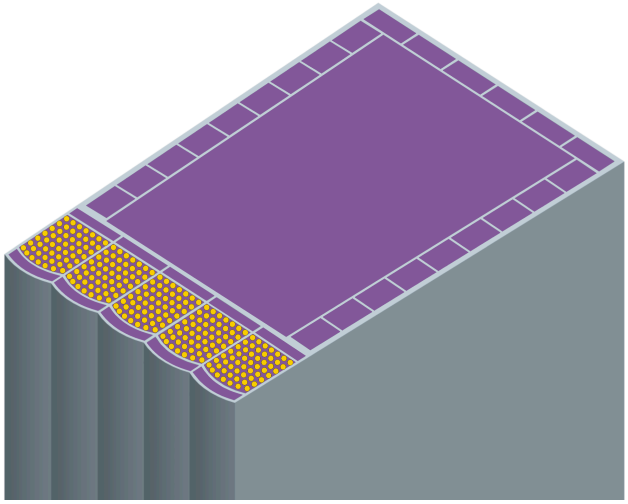

Fig. 1. Cross section in OB SC blanket concept.

The FW coolant flows radially via perforated plates through the Be pebble bed multiplier. Direct flow of Flinabe through the Be pebble bed provides for sufficient Be cooling and the contact required for chemistry control. The coolant then splits into two streams, one going through the rectangular side and back channels attached to the rear box, and the second going through the large central channel. The large central channel coolant is routed to the heat exchanger. In both cases the return flow is in the poloidal direction from top to bottom. The stream going through the side and back channels goes to a mixing chamber where it combines with the coolant returning from the heat exchanger. This mixing raises the temperature of the coolant stream from the heat exchanger, allowing it to return to the FW at a higher temperature. The re-circulating blanket combines the high flow rate in the FW/multiplier region with the low flow rate in the large central channel resulting in effective cooling of the FW and Be pebble beds and, at the same time, in maximized coolant exit temperatures.

II.B. Dual-Coolant Blanket Concepts (DC)

In these concepts, the FW and blanket structure is cooled with helium and only the breeding zone is self-cooled with Flibe. Such dual coolant blankets with lead lithium breeder have been investigated in the frame of the ARIES-ST study[11] and a European power plant study.[12] Replacing the liquid metal in the DC concept by low electrical conductivity Flibe eliminates the need for electrical insulating material between coolant and wall. In addition, no thermal insulator is required since the low thermal conductivity of Flibe together with the suppression of turbulence by the magnetic field results in extremely low heat transfer and, therefore, negligible small heat losses from the flowing breeder/coolant. As a result, the Flibe coolant can be heated up to an exit temperature about 100 degrees higher than the maximum structure temperature.

Two variations of such a DC blanket concept have been considered. The DC-Be concept has a Be pebble bed arranged at the front between two poloidal Flibe channels and cooled directly with Flibe flowing through the pebble bed in the radial direction. The DC-Pb concept utilizes a liquid lead multiplier layer placed directly behind the FW and cooled with the same high pressure helium. The helium also cools the entire steel structure. Figure 2 is a cut in the DC-Be blanket sector illustrating the typical helium flow circuit for the DC blanket. Figure 3 shows the OB DC-Pb blanket segment.

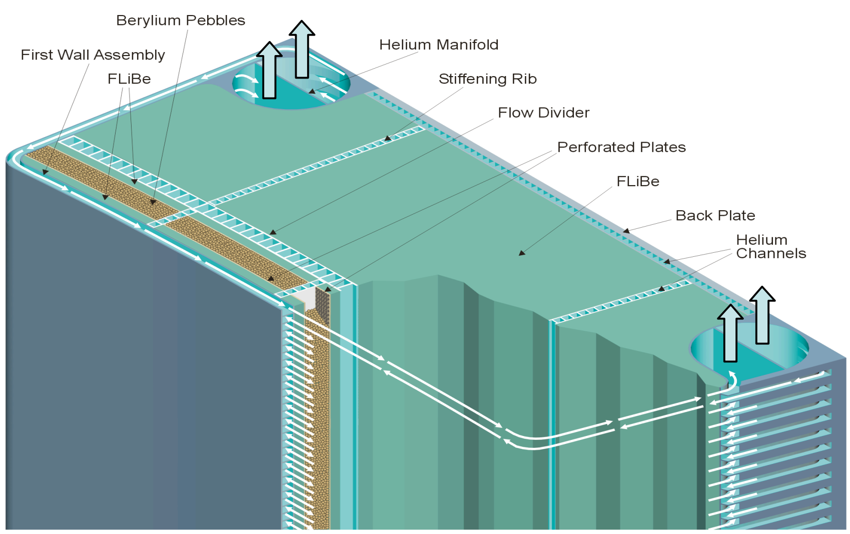

Fig. 2. Cut in DC-Be blanket showing He flow circuit.

Thermal analysis of the DC concepts indicated that if the high melting point Flibe is used, the FW should be plated with ODS ferritic steel to allow higher temperatures. We also found that with helium inlet/outlet temperatures of 300/450°C there will be local temperatures at the steel/Flibe interface below the melting point leading to formation of thin (1-2 mm) frozen Flibe layers. These concerns can be avoided by using the low melting point Flibe or Flinabe.

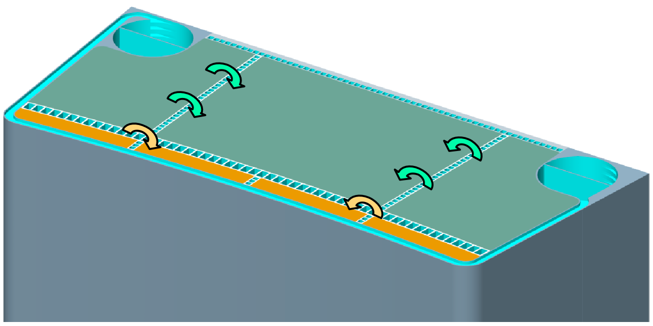

Fig. 3. Cross section in OB DC-Pb blanket segment.

III. NEUTRONICS ASSESSMENT OF MOLTEN SALT BLANKETS FOR ADVANCED POWER PLANTS

III.A. Nuclear Analysis Procedure

Neutronics calculations were performed to determine the relevant nuclear performance parameters for the three blanket concepts. These include tritium breeding, nuclear heating, radiation damage, and shielding requirements. The average reactor neutron wall loading is 3.84 MW/m^2 with peak outboard (OB) and inboard (IB) neutron wall loadings of 5.45 and 3.61 MW/m^2, respectively. The total radial build between the FW and vacuum vessel (VV) that includes the blanket and shield is 80 cm in IB and 95 cm in OB. A 25 cm thick VV is used in the calculation. Both shield and VV are water-cooled steel. Investigation of the effect of enriching Li in ^{6}Li on the tritium breeding ratio (TBR) indicates that the TBR has a flat peak in the enrichment range between 40 and 60%. An enrichment of 40% ^{6}Li is chosen and used in the calculations. The ONEDANT module of the DANTSYS 3.0 discrete ordinates particle transport code system[13] was used to perform the calculations utilizing the FENDL-2 nuclear data library.[14] Both the IB and OB regions were modeled simultaneously to account for the toroidal effects.

III.B. Blanket Radial Build

Several iterations were made to determine the radial build that achieves adequate tritium breeding and shielding for the VV and magnet. Larger margins are considered to account for uncertainties resulting from approximations in modeling. Table I gives the radial build for the OB SC blanket. Material composition in each radial zone was carefully determined to account for the toroidal material arrangement shown in Fig. 1. The total OB blanket thickness is 50 cm. The IB blanket is only 40 cm with similar radial build except for zone 8 which is reduced by 10 cm. The radial build of the OB DC-Pb blanket that corresponds to Fig. 3 is given in Table II. The OB blanket thickness is 65 cm with the IB blanket being only 40 cm. Table III gives the radial build of the OB DC-Be blanket shown in Fig. 2. Again, the IB blanket is reduced to 40 cm in thickness.

TABLE I. Radial Build of OB SC Blanket

Zone		Thickness (mm)	% Flinabe	% FS	% Be
1	FW	3		100	
2	FW Flinabe channel	10	92	8	
3	Multiplier front wall	3		100	
4	Multiplier region	70	32.2	8	59.8
5	Multiplier back wall	3		100	
6	Flinabe channel	10	92	8	
7	Flinabe channel wall	6		100	
8	Flinabe large channel	366	93.2	6.8	
9	Back channels	29	60.69	39.31	
	Total	500			

TABLE II. Radial Build of OB DC-Pb Blanket

Zone		Thickness (mm)	% Flibe	% FS	% Pb	% He
1	FW front	3		100		
2	FW cooling channel	22		13		87
3	FW back	3		100		
4	Multiplier	50		5	87	8
5	Second wall	38		32		68
6	Breeding zone	334	93	3		4
7	Breeding/manifold	166	64	9		27
8	Back wall	34		66		34
	Total	650				

TABLE III. Radial Build of OB DC-Be Blanket

Zone		Thickness (mm)	% Flibe	% FS	% Be	% He
1	FW front	3		100		
2	FW cooling channel	22		13		87
3	FW back	3		100		
4	Flibe front channel	20	93	3		4
5	Multiplier front wall	3	10	88		2
6	Be pebble bed	50	35	3	60	2
7	Multiplier back wall	3	10	88		2
8	Flibe back channel	20	93	3		4
9	Second wall	38		32		68
10	Breeding zone	288	93	3		4
11	Breeding/manifold	166	64	9		27
12	Back wall	34		66		34
	Total	650				

III.C. Tritium Breeding

These blanket radial builds were determined such that the three blanket concepts yield similar adequate TBR values. The same TBR can be achieved with a thinner SC OB blanket compared to the DC blankets (50 cm versus 65 cm) that include He coolant. A smaller Be zone thickness (5 cm) is required in the DC design with Flibe compared to the SC with Flinabe (7 cm) that has lower breeding potential. Although the multiplier zone thickness is the same in both DC blankets, a higher Pb volume fraction is used. Therefore, in the DC design more Pb is needed than Be although the Be is pushed farther from the FW by the Flibe poloidal flow channel required to cool it. This reflects the superior neutron multiplication capability of Be compared to Pb.

Table IV lists the calculated local TBR values for the three blankets. The results are comparable. If neutron coverage for the divertor (double null) is 12%, the overall TBR will be ~1.13 excluding breeding in the divertor region that could add ~0.06. Hence, the blanket concepts have the potential for achieving tritium self-sufficiency. Some design parameters can be adjusted (e.g., multiplier thickness, blanket thickness, etc.) to insure tritium self-sufficiency if needed based on detailed multi-dimensional calculations.

TABLE IV. Local TBR Values for the Blanket Concepts

	SC	DC-Be	DC-Pb
IB	0.432	0.406	0.399
OB	0.867	0.882	0.897
Total	1.299	1.288	1.296

III.D. Nuclear Heating

The total nuclear energy multiplication was determined for the three blanket concepts. The SC blanket with Be multiplier yields the highest energy multiplication of 1.27. Using Be in the DC concept results in higher energy multiplication (1.21) compared to the DC design with Pb multiplier (1.13). Nuclear heating radial profiles in the different blanket components were determined for use in the thermal hydraulics analysis. The results are shown in Figures 4, 5, and 6 for the three blanket concepts in the OB region at mid-plane. Table V compares the peak power densities in the blanket components.

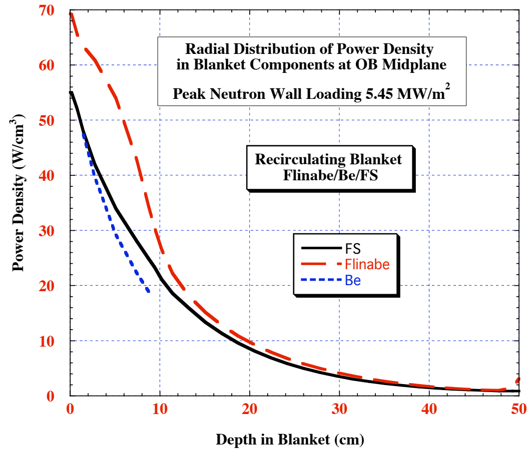

Fig. 4. Radial distribution of power density in SC blanket.

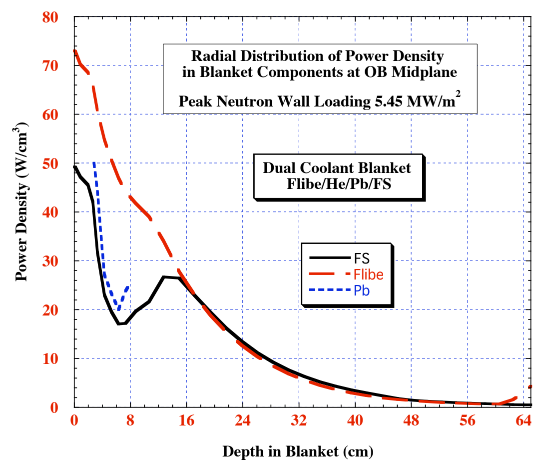

Fig. 5. Power density distribution in DC-Pb blanket.

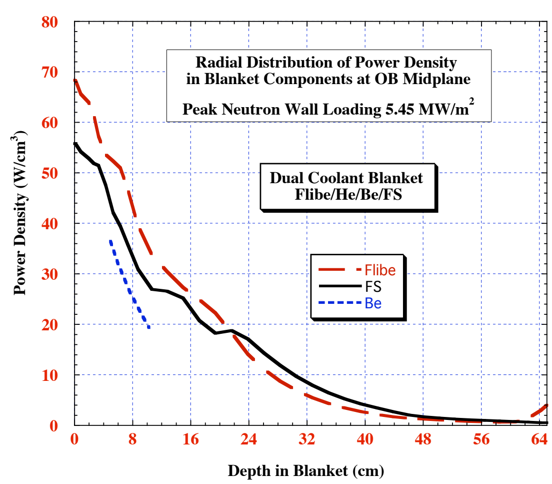

Fig. 6. Power density distribution in DC-Be blanket.

TABLE V. Peak Power Densities (W/cm^3) in Blanket Components

	SC	DC-Pb	DC-Be
Ferritic Steel	55	49	56
Molten Salt	69	73	70
Multiplier	47	50	36

III.E. Radiation Damage in Blanket and Shield

The peak OB dpa and helium production rates in the FW structure were determined for the three blanket design options. The results are given in Table VI. The peak dpa rate in the blanket option using Pb multiplier is ~10% larger than that in blanket options with Be. Assuming a lifetime radiation damage limit of 200 dpa for RAFS structure, the DC-Pb blanket lifetime is expected to be ~2.4 full power years (FPY) with slightly longer lifetime expected for the other two blankets.

TABLE VI. Peak Radiation Damage Parameters in OB FW Structure

	SC	DC-Pb	DC-Be
dpa/FPY	76.4	84.2	74.8
He appm/FPY	1005	922	983

The peak cumulative end-of-life (30 FPY) dpa in the shield structure occurs in the IB region at mid-plane. The results are given in Table VII for the three blankets. About an order of magnitude lower damage is obtained behind the thicker OB blanket. The SC blanket provides better protection for the shield compared to the DC concepts that include helium cooling. The shield is expected to be a lifetime component (based on 200 dpa limit) with a large margin that allows for uncertainties due to modeling and possible hot spots due to streaming at the module sides. Rewelding is possible only at the back of the shield.

TABLE VII. Peak End-of-life (30 FPY) Damage Parameters in IB Shield

	SC	DC-Pb	DC-Be
dpa	18	41	33
He appm	100	213	183

III.F. Tritium Production in Beryllium

A critical issue associated with using Be in fusion blankets is the amount of tritium produced and retained in the beryllium. The total amount of tritium produced in the Be pebbles used in all IB and OB modules for the two blankets that utilize Be was calculated. At end-of-life of the blanket (2.4 FPY) the total tritium production in Be is 2.97 kg for the SC blanket and 1.8 kg for the DC-Be blanket. About 75% of this amount is contributed by the OB blanket. Notice that the smaller Be inventory in the DC-Be blanket results in ~40% less tritium production. The tritium inventory 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. Based on available experimental data, the temperature at which most of tritium is released is in the range 500-700°C[15,16] depending on the density and fluence level. Tritium inventory in the Be is not expected to be an issue for these blanket options.[17]

III.G. Shielding Requirement for VV and Magnet

Calculations were performed to determine the water-cooled steel shield thickness in both the IB and OB regions required to provide adequate shielding for the VV and TF magnet coils. For the VV to be reweldable, the end-of-life helium production should be less than 1 appm. The main driver for magnet shielding is that the end-of-life dose in the polyimide insulator should not exceed 10^{10} Rads.[18] An IB shield thickness of 40 cm is required for the three blanket concepts. While a 45 cm thick OB shield is utilized behind the SC blanket, a thinner 30 cm thick OB shield is used behind the thicker DC blankets. This results in total blanket and shield thicknesses of 80 cm IB and 95 cm OB for the three blanket options. These are followed by 25 cm thick water-cooled VV. The largest VV and magnet damage parameters occur in the IB region at midplane. All VV and magnet radiation limits are satisfied with adequate margins. The peak end-of-life helium production values in the VV are 0.21, 0.45, and 0.38 appm for the SC, DC-Pb, and DC-Be blankets, respectively. The peak end-of-life magnet insulator dose values are 3.1×10^9, 6.6×10^9, and 7.6×10^9 Rads for the SC, DC-Pb, and DC-Be blankets, respectively. The peak values of end-of-life fast neutron fluence in the magnet are 1.3×10^{18}, and 2.8×10^{18} n/cm^2 for the SC and DC blankets, respectively, which are below the widely accepted limits of 10^{19} n/cm^2 for Nb_3Sn.[18] The DC blankets with Be and Pb result in comparable radiation damage parameters. On the other hand, with the same IB radial build, radiation damage parameters are a factor of ~2 lower with the SC blanket. This allows reducing the IB shield radial build by ~4 cm while maintaining the same damage levels as with the dual coolant blanket.

IV. NEUTRONICS ANALYSIS FOR DC BLANKET WITH LOW MELTING POINT MOLTEN SALT IN DEMO

The self-cooled concept was found to have limited thermal capabilities, restricting the neutron wall loading that can be handled while meeting the low temperature limit of 550°C for conventional RAFS. Advanced ODS ferritic steels, such as the nano-composited ferritic (NCF) steel,[19] with tolerance for higher temperature should be used to yield an attractive performance with high thermal efficiency. On the other hand, the dual-coolant concept allows for large exit coolant temperature while satisfying all temperature requirements for the near term RAFS, such as F82H. The performance of the dual-coolant concept with Be multiplier is investigated further in a DEMO configuration and power loading conditions. The low melting point Flibe ($LiBeF_3$) and Flinabe are used to avoid the need for ODS steel coating and eliminate any possible molten salt freezing. Detailed description and analysis of the concepts is given in a companion paper.[9]

The DEMO design used has a major radius of 5.8 m and an aspect ratio of 2.6. The peak OB and IB neutron wall loadings are 3 and 2 MW/m^2, respectively. Several neutronics calculations were performed for the concept with the two low melting point molten salts. Large margins are considered to account for uncertainties resulting from approximations in modeling. The same radial build of the DC-Be blanket given in Table III with 5 cm Be zone was used with the Flibe replaced by either the low melting point Flibe ($LiBeF_3$) or Flinabe. Figure 7 shows the effect of Li enrichment on local TBR with the different molten salts. Above 50% ^{6}Li, the TBR is flat and comparable for both low and high melting point Flibe. 50% ^{6}Li enrichment is used for the DC blanket with low melting point Flibe. For the same amount of Be, Flinabe gives lower TBR. To achieve similar TBR with Flinabe the Be zone thickness is increased to 8 cm and the enrichment increased to 60% ^{6}Li. The local TBR for both designs is 1.287. With 12% neutron coverage for the double null divertor, the overall TBR will be ~1.13 excluding breeding in the divertor region that could add ~0.06.

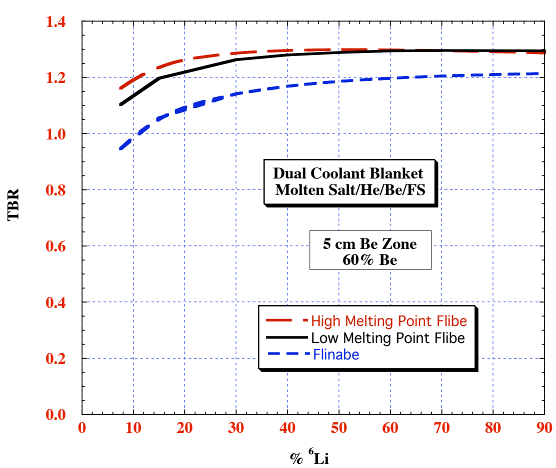

Fig. 7. Effect of enrichment on TBR in DC blanket with different molten salts.

The nuclear energy multiplication is 1.223 with $LiBeF_3$ and 1.247 with Flinabe. The power density distribution in the blanket components is shown in Figures 8 and 9 for the designs with $LiBeF_3$ and Flinabe, respectively. The peak radiation damage rate in the FW is 39 dpa/FPY in both options with the blanket lifetime expected to be ~5 FPY. The total tritium produced in the Be over the life of the blanket is 2.9 kg with $LiBeF_3$ and 3.9 kg with Flinabe. The peak cumulative end-of-life damage in the shield behind the IB DC blanket is 17.2 dpa with $LiBeF_3$ and 19.1 dpa with Flinabe implying that the shield will be a lifetime component. Using Flinabe results in a slightly lower blanket shielding capability. Using the same radial build for shield and VV given in Section III.G, all VV and magnet radiation limits were found to be satisfied with adequate margins. The peak end-of-life helium production in VV is 0.21 appm with $LiBeF_3$ and 0.23 appm with Flinabe. The peak end-of-life magnet insulator dose is 3.6×10^9 Rads with $LiBeF_3$ and 3.9×10^9 Rads with Flinabe.

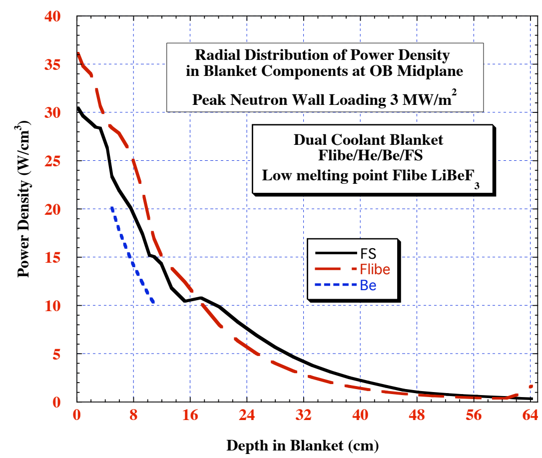

Fig. 8. Power density in components of OB blanket with $LiBeF_3$.

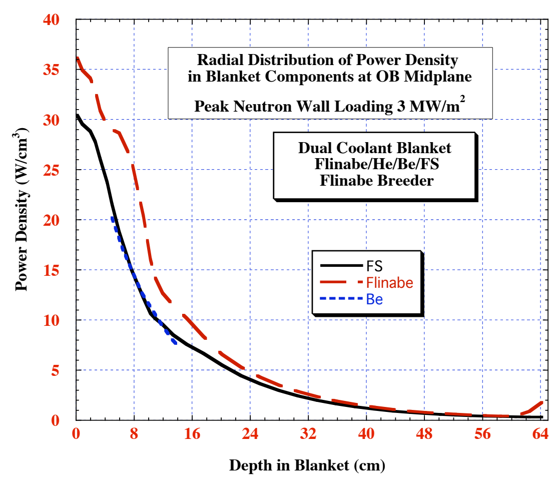

Fig. 9. Power density in components of OB blanket with Flinabe.

Three-dimensional (3-D) neutronics calculations were performed for the DC blanket with $LiBeF_3$ in the DEMO configuration using the MCNP, version 5 Monte Carlo neutronics code[20] along with nuclear data based on the FENDL-2 evaluation.[14] The aim here is to check the impact of 3-D geometrical effects and blanket heterogeneity on the overall TBR. Because of symmetry only 1/128 of the chamber is modeled (1/4 of a sector) with reflecting boundaries. The neutron source is sampled from the D-shaped plasma using a peaked distribution at the magnetic axis. The model includes the detailed heterogeneous geometrical configuration of the IB and OB blanket sectors as shown in Fig. 2. 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. Figure 10 gives a vertical cross section in the DEMO model used showing the IB and OB blankets. Fig. 11 shows a cross section in the model for the OB blanket at mid-plane. The total TBR from the 3-D calculations is 1.07 (0.85 OB, 0.22 IB). This is a conservative estimate since it assumes no breeding in the double null divertor zones 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 from 5 to 6 cm resulted in enhancing the TBR to 1.09. The 3-D modeling and heterogeneity effects resulted in ~6% lower TBR compared to an estimate based on 1-D calculations. While the OB TBR drops by only 1% the IB TBR drops by ~18% due to the shifting of the neutron source peak towards the OB region.

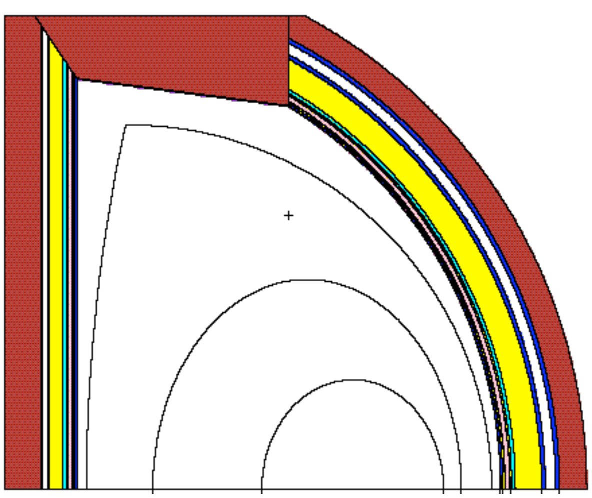

Fig. 10. Vertical cross section in 3-D model.

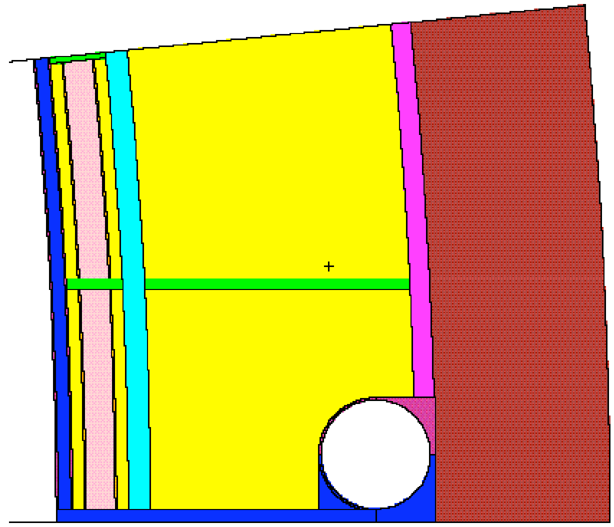

Fig. 11. Cross section in the 3-D model in OB blanket at mid-plane.

V. SUMMARY AND CONCLUSIONS

Neutronics assessment has been performed for molten salt breeding blanket concepts that can be utilized in fusion power plants. Special attention is given to concepts that can be developed, qualified and tested in the time frame of ITER. The conventional ferritic steel alloy F82H with a temperature limit of 550°C is considered as the structural material. The concepts evaluated are a self-cooled Flinabe blanket with Be multiplier and dual-coolant blankets with He-cooled FW and structure. Several options were considered for the dual-coolant concept. These include using Be or Pb multiplier. In addition, three different molten salts were considered including the high melting point Flibe, a low melting point Flibe, and Flinabe. Several iterations were made to determine the blanket radial build that achieves adequate TBR. Larger margins were considered to account for uncertainties resulting from approximations in modeling. The same TBR can be achieved with a thinner self-cooled blanket compared to the dual-coolant blanket. A thicker Be zone is required in designs with Flinabe. The overall TBR will be ~1.07 based on 3-D calculations and excluding breeding in the divertor region. Minor design modifications can be made to enhance the TBR if needed to ensure tritium self-sufficiency. We conclude that the molten salt design concepts have the potential for achieving tritium self-sufficiency. Using Be yields higher blanket energy multiplication. A modest amount of tritium is produced in the Be (~3 kg) over the blanket lifetime of ~3 FPY. Using He gas in the dual-coolant blanket results in about a factor of 2 lower blanket shielding effectiveness. With a total blanket/shield/VV radial build of 105 cm in the IB and 120 cm in the OB it is possible to ensure that the shield is a lifetime component, the VV is reweldable, and the magnets are adequately shielded. Based on this analysis we conclude that molten salt blankets can be designed for fusion power plants with neutronics requirements such as adequate tritium breeding and shielding being satisfied.

ACKNOWLEDGMENT

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REFERENCES

[1] M.A. ABDOU et al., "On the Exploration of Innovative Concepts for Fusion Chamber Technology," *Fusion Engineering & Design*, **54**, 181 (2001).

[2] A. SAGARA et al., "Studies on Flibe Blanket Designs in Helical Reactor FFHR," *Fusion Technology*, **39**, 753 (2001).

[3] 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**, 64 (2003).

[4] R. NYGREN et al., "A Fusion Reactor Design with a Liquid First Wall and Divertor," To be Published in *Fusion Engineering & Design* (2004).

[5] J. MCDONALD et al., "Measurement of the Melting Point Temperature of Several Lithium-Sodium-Beryllium Fluoride Salt (Flinabe) Mixtures," These Proceedings.

[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**, 497 (2002).

[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] C. WONG et al., "Molten Salt Self-Cooled Solid First Wall and Blanket Design Based on Advanced Ferritic Steel," To be Published in *Fusion Engineering & Design* (2004).

[9] C. WONG et al., "Assessment of Liquid Breeder First Wall and Blanket Options for the DEMO Design," These Proceedings.

[10] F. NAJMABADI et al., "Overview of ARIES-RS Tokamak Fusion Power Plant," *Fusion Engineering & Design*, **41,** 365 (1998).

[11] D-K. SZE, M. TILLACK, and L. EL-GUEBALY, "Blanket System Selection for the ARIES-ST," *Fusion Engineering & Design*, **48,** 371 (2000).

[12] P. NORAJITRA et al., "The Second Advanced Lead Lithium Blanket Concept Using ODS Steel as Structural Material and SiC/SiC Flow Channel Inserts as Electrical and Thermal Insulators," FZKA-6385, Forschungszentrum Karlsruhe (1999).

[13] R.E. ALCOUFFE et al., "DANTSYS 3.0, A Diffusion Accelerated Neutral Particle Transport Code System," LA-12969-M, Los Alamos National Laboratory (June 1995).

[14] 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).

[15] BALDWIN et al., "Tritium Release from Be," *Journal of Nuclear Materials,* **212-215**, 948 (1994).

[16] ANDREEV et al., "Tritium Release from Be," *Journal of Nuclear Materials,* **237-244**, 880 (1996).

[17] B. J. MERRILL, M. SAWAN, C.P.C. WONG et al., "Safety Assessment of Two Advanced Ferritic Steel Molten Salt Blanket Design Concepts," To be Published in *Fusion Engineering and Design* (2004).

[18] M. SAWAN AND P. WALSTROM, "Superconducting Magnet Radiation Effects in Fusion Reactors," *Fusion Technology,* **10/3**, 741 (1986).

[19] R. KLUEH et al., "Microstructure and Mechanical Properties of Oxide-Dispersion-Strengthened Steels," Fusion Materials Semiannual Progress Report for the Period Ending June 30, 2000 (DOE/ER-0313/28), pp. 123-130 (2000).

[20] X-5 MONTE CARLO TEAM, Ed., "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).