Neutronics Assessment of Molten Salt Breeding Blanket Design Options

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Molten salts have been considered as a breeding material and coolant candidates in fusion systems. Flibe consisting of LiF and BeF_2 with 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. A low melting point Flibe (380°C) was also considered but it has much 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. The breeding capability of the molten salt is limited, requiring separate neutron multiplier.

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, the conventional ferritic steel alloy F82H with a temperature limit of 550°C is considered as the structural material. Three molten salt 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 outlet temperature. 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. If the high melting point Flibe is used in the DC concepts, the FW should be plated with ODS ferritic steel to allow higher temperatures. Otherwise, the low melting point Flibe or Flinabe should be used.

In this paper, the neutronics assessment of the three concepts is presented. The ARIES-RS configuration with a peak neutron wall loading of 5.45 MW/m^2 was used. Several iterations were made to determine the blanket radial build that achieves adequate tritium breeding ratio (TBR). Larger margins were considered to account for uncertainties resulting from approximations in modeling. The same TBR can be achieved with a thinner SC OB blanket (50 cm) compared to the DC blanket (65 cm). A thicker Be zone is required in designs with Flinabe. The overall TBR will be ~1.17 excluding breeding in the divertor region. We conclude that the three design concepts have the potential for achieving tritium self-sufficiency. Several design parameters can be adjusted (e.g., multiplier thickness and blanket thickness) to ensure tritium self-sufficiency if necessary. Using Be yields higher blanket energy multiplication (1.27 for SC, 1.21 for DC-Be, and 1.13 for DC-Pb). Modest amount of tritium is produced in the Be (<3 kg) over the blanket lifetime of ~2.4 FPY. The tritium inventory will be much smaller depending on the Be temperature. Using He gas in the DC 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.