§9. Neutronics Investigation for Reduction of Radiation Shield Thickness

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In the conceptual design of the helical reactor FFHR2, a combination of JLF-1 (Low activation ferritic/martensitic steel) and B_4C has been proposed for an efficient radiation shielding. Neutron transport calculations indicate that the total blanket space of ~100 cm would be achieved by adopting the JLF-1 (70 vol.%) + B_4C (30 vol.%) shield for the back side of the Flibe cooled tritium breeder blanket [1]. Reinforcement of the shielding performance by using WC has also been investigated in the FFHR2 design [2].

In the present study, roles of the materials in radiation shielding have been investigated by transport calculations in a two layered shield configuration. Shielding performances of hydride materials which have been proposed as a neutron moderator in fast breeder reactors [3] and have been introduced to recent neutronics studies of fusion reactors [4, 5].

To avoid critical damages in a superconducting magnet system for longer than 30 years, a fast neutron flux of >0.1 MeV would have to be suppressed to $<1.0 \times 10^{10}$ $n/cm^2/s$ at the back side of the radiation shield. Figure 1 shows a neutron spectrum at the front surface of a radiation shield. The neutron spectrum of >0.1 MeV consists of two components, i.e. (1) 14 MeV neutron peak originated from DT reactions in a plasma and (2) lower energy neutrons decelerated by collisions and nuclear reactions. To understand the effects of the shielding materials on these components, the 60 cm thick radiation shield was divided into two 30 cm thick layers and the neutron transport calculations were performed for the combinations of (i) B_4C/B_4C, (ii) JLF-1/B_4C, (iii) $ZrH_{1.65}/ZrH_{1.65}$, (iv) JLF-1/$ZrH_{1.65}$ and (v) WC/WC. The MCNP5 neutron transport code and JENDL-3.3 library were used in the calculation and a simple torus model was assumed as the calculation geometry.

Figures 2 (a) and (b) show the neutron flux distributions of >10 MeV and >0.1 MeV, respectively. Figure 2 (a) indicates that the WC layer is the most effective for the suppression of the 14 MeV peak and followed by JLF-1, $ZrH_{1.65}$ and B_4C. Also in the comparison including lower energy neutrons (Fig. 2 (b)), the shielding ability of WC is highest and followed by $ZrH_{1.65}$ and B_4C. In the case of the JLF-1 layer, the low energy neutron component is significantly high compared with the other materials despite its superior shielding performance for >10 MeV neutrons. A JLF-1 shielding layer is required to be used with B_4C, $ZrH_{1.65}$ etc. which are effective for suppression of neutrons of 0.1-10 MeV.

Although the applicable positions in the reactor might be limited due to the heavy weight, usage of WC would reduce the shield thickness by ~20 cm compared with the original shield of JLF-1 and B_4C. The shielding by a $ZrH_{1.65}$ layer would also have a superior shielding performance and reduce the shield thickness by ~15 cm. However, the combination of JLF-1 and $ZrH_{1.65}$ which could achieve the similar shielding performance would be preferable in a reactor. The irradiation damage at the front surface of the radiation shield will reach 10-20 dpa after 30 years operation and the temperature would be close to the melting point of liquid molten salt or metal coolants. Considering the chemical stability and mechanical property of hydrides and insufficient database of the irradiation effects, the $ZrH_{1.65}$ layer should be placed at the backside of the thick JLF-1 layer. The tritium breeding ratio in the breeder layer was ~5 % higher for the JLF-1/$ZrH_{1.65}$ configuration compared with the $ZrH_{1.65}/ZrH_{1.65}$ configuration, since the JLF-1 layer has an effect as a neutron reflector. The B_4C layer with the stable chemical property could also be placed at the backside of the JLF-1.

Chemical and mechanical properties of the above materials under the conditions of the radiation shield region, a structure and cooling method of the radiation shield are under investigation.

[1] A. Sagara et al., Fusion Enginnering and Design 29 (1995) 51-56.
[2] A. Sagara et al., Fusion Engineering and Design 83 (2008) 1690-1695.
[3] J. B. Vetrano, Nuclear Engineering and Design 14 (1970) 390-412.
[4] T. Hayashi et al., Fusion Engineering and Design 81 (2006) 1285–1290.
[5] Y. Chen, “The EU Power Plant Conceptual Study- Neutronic Design Analyses for Near Term and Advanced Reactor Models”, Forschungszentrum Karlsruhe, FZKA 6763, 2003

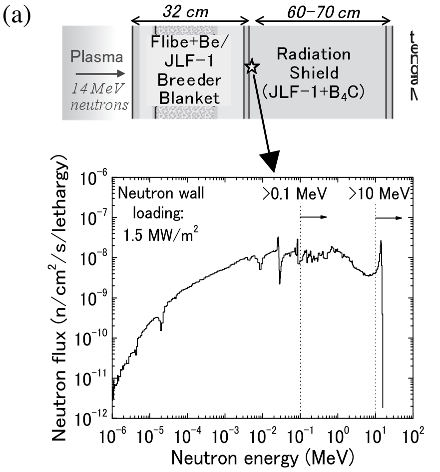

Fig. 1 (a) Structure of a blanket and (b) neutron spectrum at a front surface of a radiation shield.

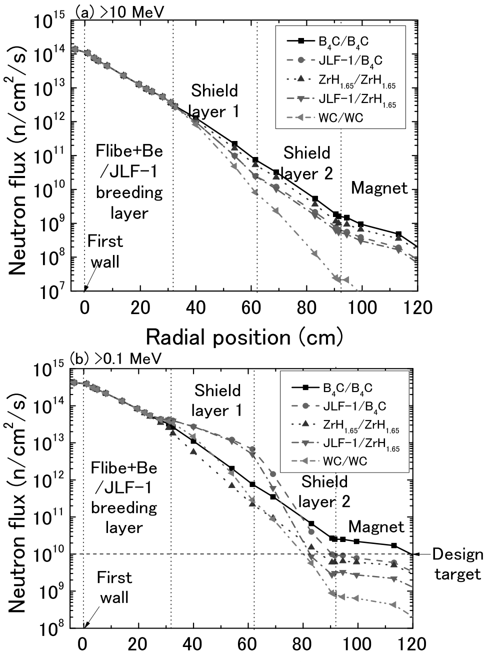

Fig. 2 Distribution of neutron flux of (a) >10 MeV and (b) >0.1 MeV.