Contributed Paper

Development of Supercritical Pressure Water Cooled Solid Breeder Blanket in JAERI

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Abstract
This paper describes the design and experimental results regarding a solid breeder blanket for a tokamak-based fusion power plant, which is cooled with supercritical pressure water. The analysis of the system shows that a thermal efficiency of 41% can be achievable with cooling conditions of 25 MPa, 510°C. Thermo mechanical analysis of the first wall shows that the present design can withstand supercritical pressure and a high heat load from plasma. The 3D neutronics analyses show that a local TBR of more than 1.4 can be obtained. Elementary R&D concerning manufacturing techniques for the blanket structure and breeder/neutron multiplier pebbles have also been developed in parallel with the blanket design.

Keywords:
fusion power plant, blanket, solid breeder, water cooled blanket, super critical pressure water, thermal efficiency, reduced activation ferritic steel, Li₂O, Li₂TiO₃, Be

1. Introduction
Solid breeding blankets have been considered one of the most promising blanket concepts for fusion power plants. JAERI has intensively developed the design of tokamak-based fusion power plants which utilizes the solid breeding blanket cooled with water. To make fusion power more attractive, it has been strongly envisaged to achieve higher thermal efficiency of fusion power plants. Along this line, a new concept of the solid breeding blanket cooled with supercritical pressure water has been proposed. This paper describes the design and experimental results on a solid breeder blanket regarding a fusion power plant, which is cooled with supercritical pressure water.

2. Design of Fusion Power Plant “DEMO”
In JAERI, design activities have been conducted on the fusion power plant, which is called as “DEMO”. The concept of DEMO is one of the fusion power plants which will be constructed as the next step of the fusion experimental reactors such as ITER. The schematic of DEMO is shown in Fig. 1, and the major design parameters are summarized in Table 1 [1]. One of the significant features of DEMO is the utilization of a supercritical pressure water system to achieve higher thermal efficiency. The coolant conditions are an inlet temperature of 280°C, and an outlet temperature of 510°C at the pressure of 25 MPa. The system flow diagram is shown in Fig. 2, and it should be noted that the heat load to the divertor is recovered by sub-critical pressure water of 4 – 10 MPa, 200°C, and is utilized to

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heat up supercritical pressure water, which flows into
the blanket. It was found that an overall thermal
efficiency of 41.4% could be achieved using the direct
turbine cycle.

Based on this DEMO reactor design, a detailed
blanket design has been developed as shown in Fig. 3
[2,3]. The DEMO blanket has a modular structure,
which can be replaced by in-situ remote handling tools.
The typical dimension of the blanket module is 1 m high × 2 m long × 0.7 m wide. The blanket module consists
of solid breeder pebble layers, neutron multiplier pebble
layers, and cooling panels. The blanket structure is made
of reduced activation ferritic steel (RAFS), F82H, which
has been developed by JAERI. The candidates of the
breeder pebbles are Li2O or Li2TiO3, and Be or Be12Ti is
applied as the neutron multiplier. The size of the pebbles
will range from 0.2 mm to 2 mm in diameter [4]. The
breeding layer is purged with Helium gas in order to
recover Tritium.

The first wall of the blanket consists of the cooling
panel in which the rectangular tubes are embedded. In
order to remove the high heat load of up to 1 MW/m²
from plasma, the first wall is cooled with sub-critical
temperature water of 280°C at a pressure of 25 MPa.
The first walls of the 2 – 4 blanket modules are cooled
in series to achieve an exit temperature of around
380°C, and the coolant then returns to cool the breeding
areas of the blankets. Finally at the exit of the blankets
which are connected in series, an exit temperature of
510°C can be obtained as shown in Fig. 4. The results of
thermo mechanical analyses show that the maximum
temperature of the first wall reaches 535°C, and the
maximum stress of 428 MPa appears at the corner of the

Table 1 Major Design Parameters of DEMO.

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
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<tbody>
<tr>
<td>Plasma major, minor radius</td>
<td>5.8 m, 1.45m</td>
</tr>
<tr>
<td>Plasma current</td>
<td>12 MA</td>
</tr>
<tr>
<td>Fusion power</td>
<td>2,300 MW</td>
</tr>
<tr>
<td>Neutron wall load</td>
<td>average 3.5, peak 5 MW/m²</td>
</tr>
<tr>
<td>FW surface heat flux</td>
<td>average 0.5, peak 1 MW/m²</td>
</tr>
<tr>
<td>Neutron fluence</td>
<td>7.5 Mwa/m² in 2 y</td>
</tr>
<tr>
<td>Tritium breeder</td>
<td>Li₂TiO₃ or Li₂O</td>
</tr>
<tr>
<td>Neutron multiplier</td>
<td>Be or Be₁₂Ti</td>
</tr>
<tr>
<td>6-Li enrichment</td>
<td>Natural to 90%</td>
</tr>
<tr>
<td>Structural material</td>
<td>Reduced activation ferritic steel (RAFS)</td>
</tr>
<tr>
<td>Coolant</td>
<td>Supercritical pressure water (25 MPa, 280–510°C)</td>
</tr>
<tr>
<td>Blanket structure</td>
<td>Modular structure, front access by remote handling</td>
</tr>
<tr>
<td>Dimension</td>
<td>1 m x 2 m, &lt; 4 ton</td>
</tr>
</tbody>
</table>

Fig. 1 Schematic of DEMO.

Fig. 2 Flow diagram of the supercritical pressure water
cooled fusion power plant.

Fig. 3 Schematic of the DEMO Blanket Module.

Max. Surface Heat Flux: 1 MW/m²
Max. Neutron Wall Load: 5 MW/m² (1.5 x 10¹⁵ n/cm²s)
cooling tube as shown in Fig. 5, which is below the upper temperature limit of 550°C of F82H.

The local tritium breeding ratio (TBR) has been evaluated using a one dimensional model of the blanket. The results show that a local TBR of 1.4 can be achievable, which will be sufficiently high to obtain a net TBR of more than 1.05 with Li$_2$TiO$_3$/Be. It was also found that the volumetric ratio of Li$_2$TiO$_3$ to Be should be more than 3 to 4.

Sensitivity studies on plasma facing material are also conducted with a three dimensional model. The results show that the net TBR of more than 1.05 can be obtained with tungsten armor as shown in Fig. 6. For the present DEMO blanket design, it can be concluded that tungsten armor of several millimeters thick can be applicable as the plasma facing material [5].

3. Results of R&Ds

Water cooled solid breeder blankets with reduced activation ferritic steel have been under development for the DEMO blankets.

For manufacturing techniques, the hot isostatic press (HIP) method was selected as the primary fabrication method of the blanket and the first wall, as the structure is too complicated to fabricate using nominal welding techniques. HIP conditions are selected for the reduced activation ferritic steel, F82H, and a small panel of the first wall has been fabricated. The first wall panel was tested in a high heat flux test facility, and it was confirmed that the panel could withstand a relevant heat load of 1 MW/m$^2$ for more than 5,000 cycles as shown in Fig. 7.

Regarding the development of the fabrication technology for the tritium breeder pebbles, the direct wet process, by which Li$_2$TiO$_3$ pebbles were directly fabricated from the Li$_2$TiO$_3$ solution, was proposed. The direct wet process can offer low cost and high efficiency processing technology for lithium recycling, because the adjustment process of the liquid mixture is not necessary.
and the lithium recovery rate is high. To develop the direct wet process, a solution of used tritium breeders, fabrication of gel-spheres, and improvement of sintering density, was examined. It became clear that the Li$_2$TiO$_3$ powder dissolved by using the liquid mixture of 30%-H$_2$O$_2$ and 5g-C$_6$H$_8$O$_7$. Droplet in the liquid mixture became gel-spheres with a high degree of sphericity in acetone. The grain size of Li$_2$TiO$_3$ pebbles fabricated by this process was less than 5 µm which was the target value.

For characterization of the tritium breeder, the thermal properties of TiO$_2$-doped Li$_2$TiO$_3$ were evaluated. The thermal diffusivity of Li$_2$TiO$_3$ and TiO$_2$-doped Li$_2$TiO$_3$ decreased with increasing measuring temperature and decreased with increasing TiO$_2$ content. The thermal conductivity of the TiO$_2$-doped Li$_2$TiO$_3$ is estimated to be 15% smaller than that of Li$_2$TiO$_3$ at 400°C.

Regarding the pebble bed thermo-mechanical performance of the breeder, its effective thermal conductivity has been measured with a pebble bed, consisting of a mixture of two sizes of the Li$_2$TiO$_3$ pebbles, 0.3 mmΦ pebbles and 2 mmΦ pebbles, for increasing the total volume of the breeder pebbles. The packing ratio, which is the ratio of the total volume of pebbles to the volume of a pebble container, is about 60% using only 2 mmΦ pebbles, while it can be increased to about 80% by using a mixture of 0.3 mmΦ pebbles and 2 mmΦ pebbles. Experimental data and the new correlation based on the Schleuder-Zehner-Bauer (SZB) model are shown in Fig. 8. It should be noted that the SZB model is applicable for this complicated configuration.

A new test facility was built to measure the effective thermal conductivity of a Li$_2$TiO$_3$ pebble bed under a compressive load [6]. The pebbles were installed in a load cell, which could be compressed by a piston in one direction, and the thermal conductivity was measured using the hot wire method. The results are shown in Fig. 9. In this experiment, the strain of the pebble bed was estimated based on the displacement of the loading piston. It can be seen that the effective thermal conductivity increases with the strain in a temperature range of 600 – 700°C.

For development of the neutron multiplier, a neutron irradiation test of Be$_{12}$Ti as a beryllium intermetallic compound was carried out up to 0.5 dpa and 70 appmHe at 500°C in Japan Materials Testing Reactor, JMTR. The tritium release behavior and swelling behavior were evaluated after neutron irradiation.

From the results of the tritium release test, it is obvious that the tritium inventory in Be$_{12}$Ti is much smaller than that in Be. Swelling of the Be$_{12}$Ti disks was
less than 3%. On the other hand, swelling of the Be disks was ~60%. From these results, swelling of Be$_{12}$Ti under the high temperature neutron irradiation can be expected to be smaller than that of Be as shown in Fig. 10. These results show that beryllium intermetallic compounds are excellent candidates for neutron multipliers in the fusion blanket.

For the development of irradiation technology for in-pile functional tests, an evaluation of the effective thermal diffusivity of a Li$_2$TiO$_3$ pebble bed under neutron irradiation, and irradiation tests involving a small motor were carried out with JMTR. The effective thermal diffusivity of Li$_2$TiO$_3$ pebble bed under neutron irradiation was evaluated by the constant raising temperature method, which employs calculations based on the temperature difference between two locations in the pebble bed. It became clear that the effective thermal diffusivity decreased with increasing temperature and was almost constant during the reactor operation periods.

A fabrication technique for a Cr$_2$O$_3$-SiO$_2$ coating as a tritium permeation barrier has been developed using the chemical densified coating (CDC) method. The CDC method has some advantages compared with other coating methods. This method is capable of forming a densified coating on either the outer or the inner surface of a tube or a container. Figure 11 shows a cross sectional view of the coating layer. However, the Cr$_2$O$_3$-SiO$_2$ coating had open pores in the coating. For filling these open pores, the densification treatment by CrPO$_4$ was examined. It was found that the permeation reduction factor (PRF: the ratio of deuterium permeation from the SS316 substrate without coating and that with coating) of deuterium for the Cr$_2$O$_3$-SiO$_2$ coating with CrPO$_4$ reached about 1,000 at 600°C as shown in Fig. 12.

4. Conclusion

The DEMO blanket design has been proposed, which is cooled with supercritical pressure water. Thermal efficiency of more than 41% can be achieved. Thermo mechanical analyses show that the present design can withstand supercritical pressure and high heat load from plasma. 3D neutronics analyses show that a local TBR of more than 1.4 can be obtained. Elementary R&Ds on manufacturing techniques for the blanket structure and breeder/neutron multiplier pebbles have been developed in parallel with the blanket design. It has been shown that the concept of the DEMO blanket cooled by supercritical pressure water is a promising way of making the DEMO plant more attractive.
References


