A feasible DEMO blanket concept based on water cooled solid breeder

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Abstract. This paper presents the conceptual design of a blanket with simplified structure whose interior consists of the mixture of breeder and multiplier pebble bed, cooling tubes and support for them only. Neutronics calculation indicated that the blanket satisfies a self-sufficient production of tritium. An important finding is that decrease in TBR is small when the gap between neighboring blanket modules is as wide as 0.03 m. This means that blanket modules can be arranged with such a significant clearance gap without sacrifice of tritium production, which will facilitate the access of remote handling equipment for replacement of the blanket modules and improve the access of diagnostics. On the other hand, facility for handling of blanket after shutdown is important for the feasibility improvement. However, the dominant material determining decay heat is RAFM in first walls, which the RAFM is most feasible material for DEMO. It may be necessary to decrease neutron wall load in considering a plant system because the decay heat was proportional to neutron wall load.

1. Introduction

The conceptual design study of blanket has been conducted for a fusion DEMO reactor SlimCS [1]. Considering DEMO specific requirements, we place emphasis on a blanket concept with durability to severe irradiation, facility of fabrication for mass production, operation temperature of blanket materials, and maintainability using remote handling (RH) equipment. This paper presents a promising concept satisfying these requirements, which is characterized by minimized welding lines near the front, a simplified blanket interior consisting of cooling tubes and a mixed pebble bed of breeder and neutron multiplier, approximately the same outlet temperature for all blanket modules, and an arrangement of blanket modules leaving sufficient space for RH equipment access. In addition, overall tritium breeding ratio (TBR) should be higher than 1.05 because the self-sufficient production of tritium for sustaining operation and initial tritium for next fusion reactor. On the other hand, the decay heat from the activated materials of blanket is important in maintenance because the situation is same as Loss Of Coolant/Flow Accidents (LOCA/LOFA) during maintenance. In this sense, the nuclear characteristics of blanket need to be understood to define criteria for the maintenance.

2. Simplification of blanket design concept

The original concept of the DEMO blanket was based on a multilayered interior [1], in which a breeding layer with lithium ceramic pebbles and a neutron multiplier layer with beryllium plate were alternately arranged to the back of the blanket, and these neighboring layers were separated with a partition with cooling tubes. The concept was designed as an extension of the ITER-TBM (Test Blanket Module) of Japan. The role of the partition made of F82H was to separate the lithium ceramic breeder from beryllium which is chemically active and deprives the breeder of oxygen. However, the multilayered concept has engineering difficulties in welding of cooling tubes and the F82H casing for beryllium plate, pebbles packing in narrow regions and inspection after fabrication. In order to resolve the problem, we proposed a blanket concept with a simplified interior without using any partitions as shown in Fig. 1 [2]. A key point of the concept is to use chemically stable $Be_{12}Ti$ pebbles as neutron multiplier. As a result, the interior is composed of a mixed bed of Li_2TiO_3 and $Be_{12}Ti$ pebbles, and cooling tubes only. Since the cooling tubes are welded to coolant headers located in the back of the blanket modules, irradiation degradation of welding lines is alleviated.

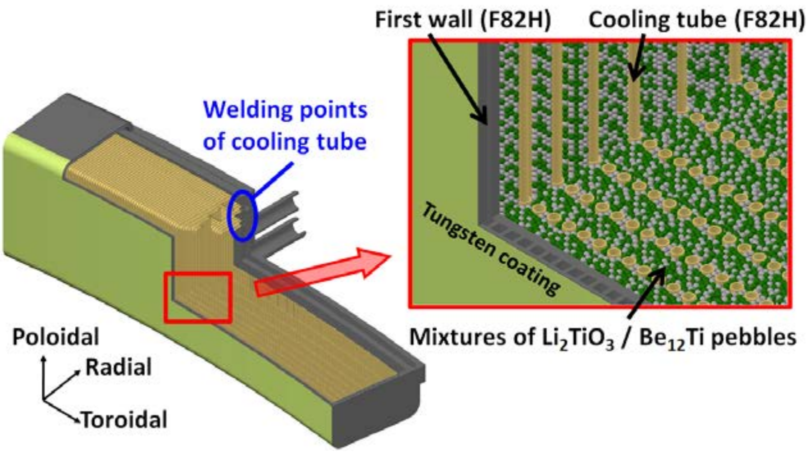

FIG. 1. Interior design of the mixed breeder blanket.

In Figure 1, the plasma facing surface of the blanket is covered with 0.2 mm-thick tungsten (W) to suppress erosion by physical sputtering. A typical size of the modules is 2 m in width and 0.6 m in height, and the structural material is reduced-activation ferritic martensitic steel (F82H). The distance of each cooling tube is designed to satisfy the operation temperature of the materials; $Li_2TiO_3 \leqq 900$ °C, $Be_{12}Ti \leqq 900$ °C and F82H $\leqq 550$ °C. When we consider water cooling for DEMO, the Pressurized Water Reactor (PWR) water conditions of 15.5 MPa and 290-330 ºC seems appropriate in terms of the compatibility with F82H. The PWR water has an advantage that matured technologies in nuclear power plants will be likely to reduce development risks in fusion plant engineering, as well. The problem is its narrow operation temperature. From the viewpoint of the effective cooling of the first wall and blanket, the inlet temperature of coolant is desired to be as low as possible. The water coolant with low temperature, however, may erode the structural material due to the residual hydrogen peroxide produced (H_2O_2) by radiolysis water decomposition. H_2O_2 starts to pyrolytically decompose at 240 ºC. Therefore, the inlet temperature of the first wall coolant is selected to be 290 ºC with sufficient margin. On the other hand, use of the PWR water to the blanket requires a reduction of coolant plumbing length because of a reduced temperature range. The direction of coolant plumbing in the blanket is changed from toroidal to poloidal direction as this solution for the problem. As a result, all length of coolant plumbing can be decreased without changing the configuration of the blanket module as shown in Fig.1.

Figure 2 shows the local TBR when the blanket is assumed to be homogeneous. The x-axis is the ratio of breeder, defined as follows; BMR = breeder / (breeder + multiplier), where the breeder and multiplier are assumed to be Li_2TiO_3 and $Be_{12}Ti$, respectively. The calculation conditions are as follows: ^{6}Li enrichment of 90%, packing fraction of 65%, 70% and 80%, and the blanket for thickness of 0.80 m. In addition to the breeder and the multiplier, the blanket model contains F82H and water, the fraction of F82H and water is 20%, and the ratio of F82H to water is assumed to be 70-30%. The result indicates that the local TBR becomes a maximum value when the BMR is between 15% and 25%. In the case of the proposed blankets, the maximum TBR is obtained at the BMR of about 20%. Under this condition for packing factor of 80%, ratio of breeder, multiplier and He-gas are 12%, 68% and 20%, respectively. The effective thermal conductivity of the mixed pebbles is defined by the combination of the conductivities of breeder and multiplier pebble beds [3-5] in accordance with the BMR as shown in Fig. 2.

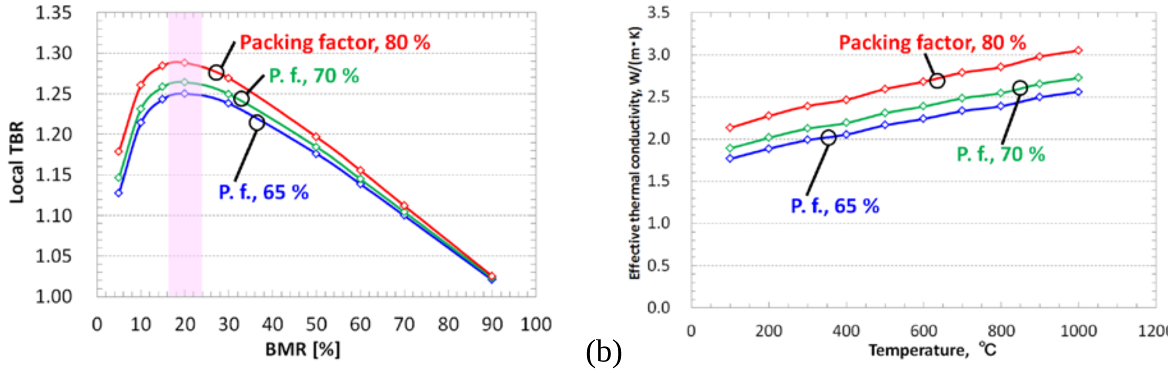

FIG. 2.Local TBR with ratio of mixed breeder (a), and effective thermal conductivities (b).

3. Study on arrangement of blanket modules

3.1. Poloidal distribution of neutron wall load

The Neutron Wall Load (NWL) varies in the poloidal direction. This means that the power generated in a single blanket module is dependent on the poloidal location. Despite that, for efficient use of heat, the outlet temperature of coolant should be almost the same independent of where the blanket module is located. Therefore, the poloidal distribution of NWL can be important background of NWL for blanket design. In this context, the poloidal direction is calculated with the 3-D Monte Carlo N-Particle transport code MCNP-5 [6] with the nuclear data library ENDF/B-VII [7]. Figure 3 shows the 3-D MCNP calculation model based on SlimCS. The model includes the geometrical arrangement of the blanket, the divertor, the central solenoid (CS) and toroidal field (TF) coils. By assuming toroidal axisymmetry, a 1/24 sector of the reactor is modeled with reflecting boundaries. The neutron with the energy of 14.06 MeV is emitted from plasma volume. The blanket coverage loss by the divertor is 11.8% and loss by the ports is 1%. In addition, rims, ribs and gap of the structure in the blanket coverage are 7.3%, 3% and 1%, respectively. Therefore, the total coverage of blanket is 75.9%. Figure 4 shows the poloidal distribution of the NWL. The peak NWL in the inboard (IB) and outboard (OB) blanket are 2.77 and 3.62 MW/m^2, respectively. The average NWL in the IB and OB blanket are 2.31 and 3.32 MW/m^2, respectively.

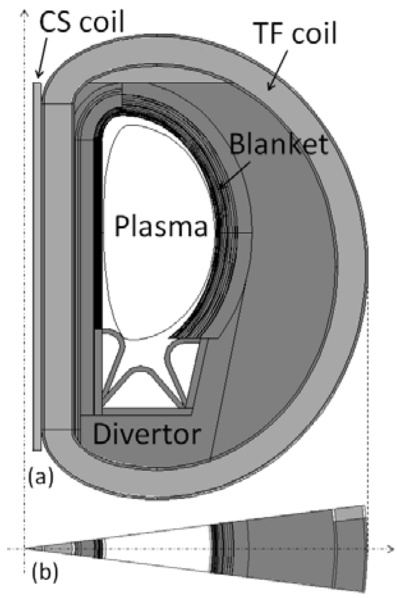

FIG. 3. MCNP calculation model for DEMO reactor

a: vertical cross section in the 3-D model and b: horizontal cross section at mid-plane

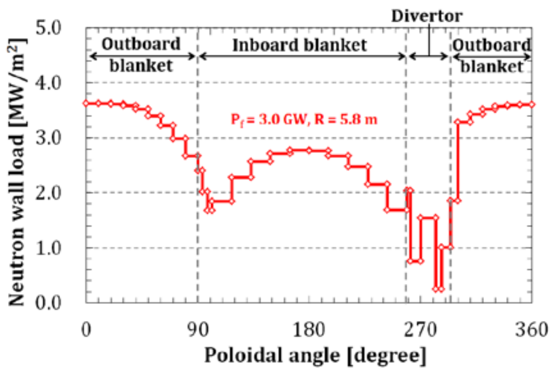

FIG. 4. Poloidal distribution of neutron wall load for fusion DEMO reactor

3.2. Dependence of TBR on neutron wall load

In the case of DEMO reactor with a fusion output of 3 GW, the NWL varies between 1.6 and 3.6 MW/m^2 in the poloidal direction. If we supply the same quantity of water to the blanket modules with different NWL, the outlet temperature from every module is different, which is inefficient in terms of heat utilization. In order to resolve the problem, the number of cooling tube or the spacing between the neighboring cooling tubes should be determined to adjust the outlet temperature from each blanket module. Here, the sizes and arrangement of the cooling tubes for the proposed blanket are changed in accordance with the NWL. Actually, the blanket was approximated by a slab model for the calculations. The cooling tubes are replaced with slabs having the equivalent cross section. In the 1-D calculations of the neutronic and thermal analysis for the blanket, the ANIHEAT code with the nuclear library FENDL-2.0 [8] is used. The calculation conditions are as follows: ^{6}Li enrichment of 90%, packing fractions of 65% and 80 %, and blanket thickness of 0.6 m. Moreover, the upper coolant velocity is limited to 5 m/s and the outlet temperature is less than 330 ºC. The heat load from plasma such as impurity radiation is assumed to be 0.5 MW/m^2 on all blanket surfaces. The neutronics is calculated on local TBR and nuclear heating in the blanket. The temperature of blanket is evaluated by the 1-D thermal conduction equation. The thickness of each layer is determined to satisfy the operation temperature of materials. Figure 5 shows the dependence of local TBR on NWL. As a whole, this design also contributes to increasing TBR due to a relative increase of the breeding material fraction for blanket modules with low NWL as shown in Fig. 5.

A neutronics analysis ensures the blanket concept meets a self-sufficient supply of tritium when the blanket thickness is about 0.6 m, which the overall TBR is calculated by considering the poloidal distribution of NWL in Fig. 4. In figure 6, it is found that the overall TBR is attained to be 1.09 and 1.15, which the packing factors are 65 % and 80 %, respectively.

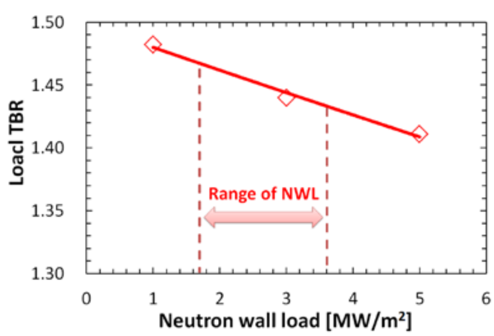

FIG. 5. Dependence of local TBR on neutron wall load

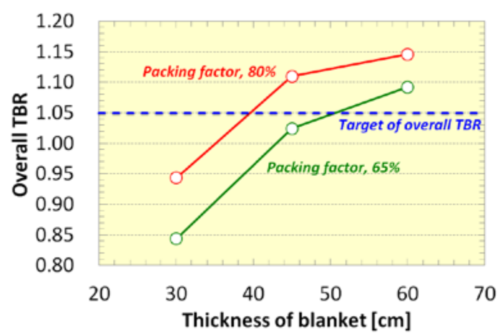

FIG. 6. Dependence of net TBR on blanket thickness

3.3. Tendency of TBR on the gap between neighboring blanket modules

In order to achieve the self-sufficient production of tritium, non-breeding zones need to be minimized. In this context, the gap between neighboring blanket modules should be as small as possible. On the other hand, the gap is necessary to cope with a thermal distortion of blanket in the operation and an allowance for blanket replacement. To withstand vapor pressure of the coolant water, moreover, sufficient thickness of module frame is necessary from viewpoint of safety apparatus. Thus, it is an important trade-off problem to determine an appropriate gap from the points of view of blanket coverage, maintenance, robust design against the vapor pressure, etc.

In the previous blanket design for SlimCS, the clearance gap between neighboring blanket modules was determined to be as small as 0.005 m to meet a self-sufficient tritium supply on the basis of a simple assessment that the net TBR was reduced from a local TBR by multiplying the coverage of the breeding zone. Here, the local TBR stands for the TBR from a 1-D calculation for the breeding zone. In the estimation, the gap and the frames of the modules were regarded as non-breeding zones, not contributing to tritium production. However, part of neutrons scattered in the non-breeding zones have a possibility of producing tritium. Certainly, the estimation leads to a pessimistic result because significant parts of neutrons scattered in non-breeding zones have a possibility of producing tritium in breeding zones. On the other hand, there is a concern that TBR may not increase as expected because of neutron absorption by Fe of F82H used in the module casing. In order to assess the scattering effect quantitatively, we carried out a three-dimension neutron transport calculation. In this study, a water cooled mixed breeder concept with reduced activation ferritic martensitic steel (F82H) are estimated by the 3-D Monte Carlo N-particle transport code MCNP-5 [6] with the nuclear data library ENDF/B-VII [7]. Figure 7 shows the 3-D MCNP calculation model. The major parameters of the reactor are a plasma major radius of 5.5 m and aspect ratio of 2.6. By assuming toroidal axisymmetry, 1/12 sector of the reactor is modeled with reflecting boundaries. The neutron volume source for plasma emitted neutron with the energy of 14.06 MeV. Tungsten coating is required on first wall (FW) surface to suppress erosion by physical sputtering. For the purpose, the surface of the blanket is covered with 0.2 mm-thick tungsten (W). The calculation model includes the geometrical arrangement of blanket modules of 0.3, 0.5 or 0.7 m in thickness around the plasma without divertor and ports for simplicity. A typical size of the modules is assumed to be 2 m in width and 0.6 cm in height, and a module frame of 0.02, 0.04 or 0.06 m in thickness composes the frame of the modules. Here, the breeding zones are modeled to be homogeneous as shown in Fig. 7. The blanket concept is assumed to be composed of structural material with F82H, tritium breeder with Li_2TiO_3 pebbles and neutron multiplier with $Be_{12}Ti$ pebbles. In these materials in the breeding zones, ratio of tritium breeder, multiplier, He for purge-gas and structural material are 10%, 42%, 28% and 20%, respectively. The tritium breeder in blanket selected to be 90 % ^6Li enrichment.

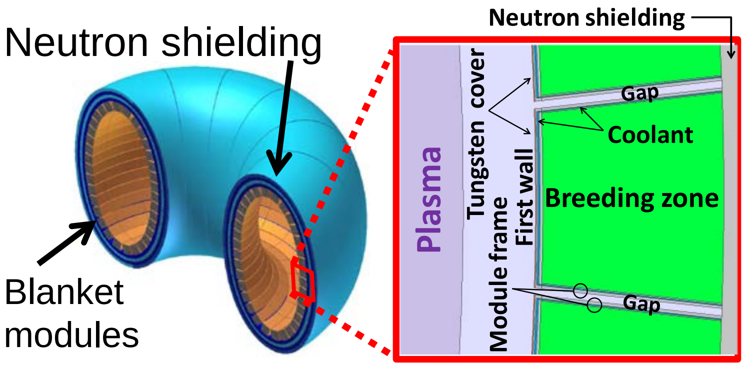

FIG. 7. Image of MCNP calculation model, and enlarged view of square area.

For dealing with the problem properly, we carried out a 3-D neutron transport calculation with the actual blanket geometry. Figure 8 show a comparison of the TBR between the previous method (PM) and the actually blanket geometry (AG). In these results, the casing thickness is fixed to be 0.02 m. Notice that the calculation for AG shows higher TBR than the PM, indicating little decrease in TBR for the gap of less than 0.03 m. The result indicates that such an allowance of the gap will facilitate access of remote handling systems for replacement of the blanket modules. Figure 9 shows a comparison of TBR when the gap or the casing thickness is changed. This result indicates that an increase in the blanket casing thickness leads to a serious decrease in TBR. On the other hand, the blanket should be designed to withstand vapor pressure of the coolant water for safety apparatus. Therefore, the decision of the blanket casing thickness should be made carefully considering such a TBR reduction effect.

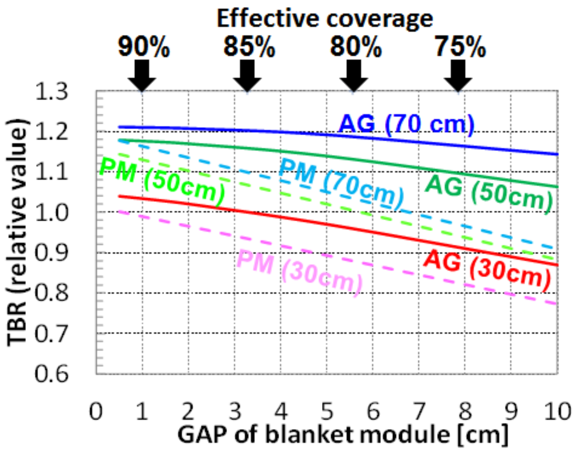

FIG. 8. TBR on the gap of blanket module

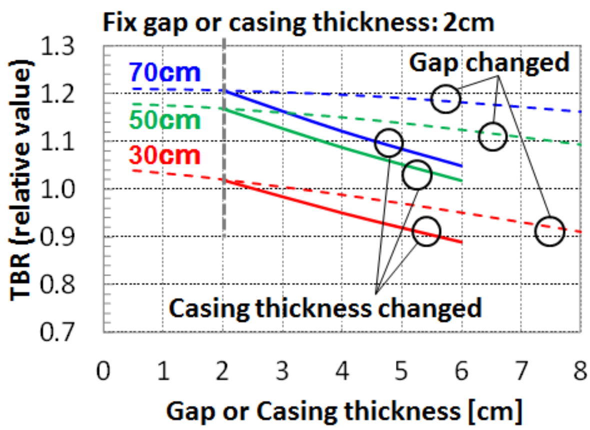

FIG. 9. TBR on the casing thickness of blanket module

4. Decay heat of water cooled solid breeder blanket

The decay heat with fusion output of 3.0 GW is calculated by a THIDA-3 code with the nuclear data library FENDL-2.0 [8]. It is found that decay heats of OB blanket, IB blanket, divertor and radiation shield would be as higher as 30.9 MW, 8.6 MW, 13.5 MW and 1.7 MW, respectively, immediately after the shutdown of operation. It is assumed that the replaceable blanket modules and divertor are changed every two years during operation. The blanket has reduced-activation martensitic steel (F82H), Li_2TiO_3 tritium breeder and $Be_{12}Ti$ neutron multiplier. The divertor is made of W mono-block and F82H cooling tubes and substrates. The total decay heat is as high as 54.6 MW and decreases to 3.1 MW one month later. Blanket produces the largest portion (72%) in the decay heat. Thus, the estimation of the decay heat for blanket is important for cooling the waste. Ratio of the decay heat in total decay heat for OB blanket is shown in Figure 10. The dominant component determining the decay heat of blanket was FW of F82H and H_2O, and breeder of Li_2TiO_3 and $Be_{12}Ti$. Here, the decay heat of W is lower than FW and breeder. This is because the total volume is lower than those of the other components. In the Li_2TiO_3 employed as tritium breeder, the activity of ^{16}N and ^{6}He produced through sequential ^{16}O (n, p) and ^{6}Li (n, p) reactions becomes prominent 10 sec after shutdown. Thus, contributory the dominant decay heat of breeder decreases in one minute as shown in Fig. 10. The ^{48}Sc activity from ^{48}Ti in breeder is prominent during cooling times between 1 minute and 3 day after shutdown. Three days after shutdown, the dominant nuclides determining the decay heat density was ^{46}Sc during the prominent time from 3 days to 1 year which originates from ^{46}Ti. In the F82H employed as blanket structural, the activity of ^{56}Mn produced through sequential ^{56}Fe (n, p) reaction becomes prominent in 3 hours after shutdown. Thus, contributory the dominant decay heat of breeder increases as one days as shown in Fig. 10. Three hours after shutdown, the dominant nuclides determining the decay heat density was ^{54}Mn during the prominent time from 3 days to 1 year which originated from ^{54}Fe. Here, the F82H is most feasible material in fusion DEMO reactor. However, there is a concern that the dominant material determining decay heat is F82H in first wall. One year after shutdown, on the other hand, ^{3}H becomes increasingly a major part of decay heat in breeder as shown in Fig. 10. However, tritium activity can be ignored from the calculated results although it is produced especially in breeder because the recovery of tritium would be possible in the breeding zone. Moreover, reducing the neutron wall load is effective in decreasing the decay heat from threshold reactions of ^{56}Fe (n, p), ^{54}Fe (n, p) and ^{48}Ti (n, p) reactions.

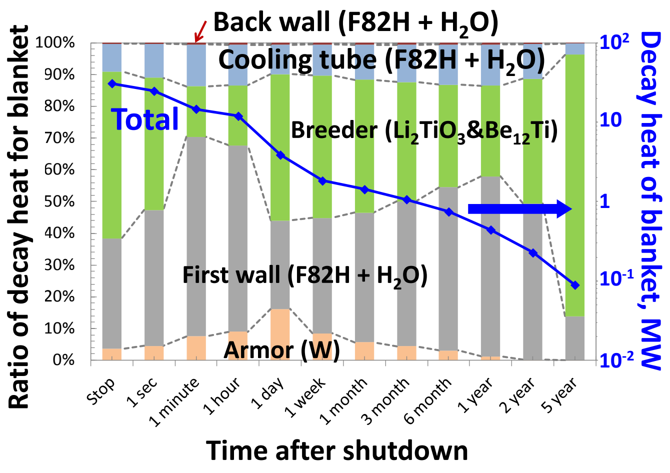

FIG. 10. Ratio of decay heat in OB blanket structure.

5. Discussion of a feasible DEMO blanket concept based on water cooled solid breeder

Neutronics calculation indicated that the mixed breeder blanket satisfies a self-sufficient production of tritium as shown in fig. 6. In the section 3.3, moreover, the result indicates that such a wide allowance of the gap will facilitate access of remote handling equipment for replacement of the blanket modules and improve access of diagnostics. However, this proposed blanket concept is not designed from the viewpoint of safety apparatus. There is a concern that the first or side-walls in the blanket may rupture when the cooling pipe in the blanket is tearing, because thickness of structure materials for blanket is thin. The internal mixed pebbles may spill in a vacuum vessel. This event makes restoration for the re-operation of the reactor difficult. Therefore, the thickness of blanket structure and ribs, and aspect ratio of blanket dimension from the view point of safety is important for fusion DEMO blanket. The structure is necessary to endure the pressure of coolant water (8 MPa) till a rapture disk is broken because the vapor pressure of the coolant water (300℃) is assumed to be 8 MPa. However, there is a concern that the TBR may decrease because of increasing the non-breeding zones for casing structure. The non-breeding zones may not contribute to tritium production as shown in fig. 9. Therefore, the width of the blanket structure was determined to be 22 mm. In the preliminary analysis, thickness of casing structure is assumed to be 37 mm for withstand vapor pressure of the coolant water. In this condition, the overall TBR decrease 1.15 to 1.02 at the blanket thickness of 0.60 m. The TBR is assessed by changing fusion power for decrease NWL and use materials to make up for the tritium shortage. For the feasibility improvement of DEMO blanket, facility of the handling after the replacement is important, too. The dominant material determining decay heat is RAFM (reduced activation ferritic/martensitic steel) in first walls shown in Fig. 10, which the F82H is most feasible material in fusion DEMO reactor. In the preliminary analysis, the maximum temperature of the blanket surface reaches 1,067 ºC with natural convection cooling, after about 3 month later in the hot cell. Here, it is reasonable to accept the temperature of blanket structure up to 550 ºC in which the structural strength of F82H is maintained. Here, the decay heat was proportional to NWL [10]. In this sense, the fusion power need to be decreased of 1.5 GW from 3.0 GW, or the storage term in reactor need to be increased of 1 year from 1 month. For high plant availability, however, it is necessary to decrease NWL in considering a plant system. With decreasing the NWL, moreover, the local tritium breeding ratio is improved because of a reduction of the coolant area in the blanket.

6. Conclusion

Although the target of tritium breeding was to ensure a self-sufficient production of tritium, we put emphasis on understanding various aspects of the relationship between the blanket and the torus configuration in the systemic design point of view, rather than assessing precise TBR based on detailed three-dimensional nuclear calculations. Considering the continuity with the ITER-TBM option of Japan and the engineering feasibility of fabrication, our design study focused on a water-cooled solid breeding blanket using a mixed bed of Li_2TiO_3 and $Be_{12}Ti$ pebbles. Important findings obtained from the recent study are that 1) higher TBR was anticipated for lower NWL because of a relative increase of the breeding zone as a result of reduced cooling channels at lower NWL, and that 2) when the gap between the neighboring blanket modules was changed from 0.005 m to 0.1 m, little decrease in the calculated TBR is seen for the gap of less than 0.03 m. Such an allowance of the gap will facilitate access of remote handling equipment for replacement of the blanket modules and improve access of diagnostics. On the other hand, the facility for the handling of blanket after shutdown was important for the feasibility improvement. However, the dominant material determining decay heat was RAFM in first walls, which the RAFM was most feasible material for DEMO. It may be necessary to decrease NWL in considering a plant system because the decay heat was proportional to NWL.

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