Conceptual Design Study of Superconducting Spherical Tokamak Reactor with a Self-consistent System Analysis Code

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Abstract. In the ST reactor, the radial build of TF coil and the shield play a key role in determining the size of a reactor. For self-consistent determination of these components and physics parameters, a system analysis code is coupled with the one-dimensional radiation transport code. Conceptual design of a compact superconducting ST reactor with aspect ratio of less than 2 was conducted and it is shown that the ST reactor with outboard blanket only can provide tritium self-sufficiency by using an inboard neutron reflector instead of breeding blanket. With the use of an improved shielding material and high temperature superconducting magnets with high critical current density open up the possibility of a fusion power plant with compact size and smaller auxiliary heating power simultaneously at low aspect ratio.

Key words: spherical tokamak, reactor design, system analysis

1. Introduction

Spherical tokamak (ST) plasma has the potential of high β operation with high bootstrap current fractions. For the possibility of a compact fusion reactor, the ARIES-ST study [1] investigated ST with a low aspect ratio A in the range of 1.2 ~ 2.5, showing a conceptual projection of a 1000 MWe-class electric power plant with copper magnets. However, the resistive losses in the copper TF magnet make the re- circulating power significantly large for an attractive economical fusion power plant. Taking into account the recent progress in the high temperature superconducting magnet technology, we therefore present here a conceptual design of a superconductor based ST power plant design with minimum re-circulating power. In the previous studies, superconducting tokamak reactors with tight aspect ratios such as VECTOR [2] were proposed with limited aspect ratio only down to 2.3. In the ST reactor, a radial build of a superconducting TF coil and a shield play a key role in determining the size of a reactor. To find space for the radiation shielding of the superconducting TF coil inside the torus, high critical current density at high magnetic field strength is required for the TF coil conductor.

Recent progress in the development of superconducting material [3], promising much higher engineering critical current density of 100 kA/cm^2 for high magnetic fields beyond 20T by operating below liquid nitrogen temperature, led us to investigate the possibility of employing the superconducting TF coil in the aspect ratios of 1.5 ~ 2.0.

An inboard shield requires improved performance with respect to neutron economy for enough tritium breeding and shielding capability to protect the superconducting TF coil; the fast neutron fluence to the superconductor, the peak nuclear heating in the winding pack, and the radiation dose absorbed by the insulator. In addition to tungsten carbide which has been considered as a shielding material in many reactor studies, metal hydrides and borohydrides which are reported [4,5] to provide a good shielding performance are used for improved shielding performance and thus smaller shield thickness.

The ST reactor with outboard blanket only is considered where tritium self-sufficiency is possible by using a simple-structured inboard neutron reflector instead of breeding blanket. The reflecting shield should provide not only protection for the superconducting TF coil but also improved neutron economy for the tritium breeding in outboard blanket. Be, graphite, ZrH2, TiC, Pb etc are known as good neutron reflectors in fission reactor and we investigate their characteristics in fusion neutron spectrum. Pb is selected as an extra inboard reflector material since it has a higher (n,2n) cross section for high energy neutron compared to Be as shown in Table. 1. These cross section data are included in coupled system analysis code.

This paper addresses viability of superconducting ST reactor concept with recent technologies for superconducting TF coil and shield materials.

2. Tokamak Reactor System Analysis Coupled with Neutron Transport Analysis

In the system analysis code, plasma physics properties are expressed in a zero-dimensional model. They impose a limitation to the possible plasma performance through the beta limit, the plasma current limit imposed by a limit on the safety factor q at the edge, and the plasma beta limit. The operating parameters of ST plasmas have been estimated previously utilizing observed tokamak scaling and MHD calculations over a range of the aspect ratio and limits on key physics parameters[6 – 8]. The maximum elongation κ depends on aspect ratio and the average of the expressions from Ref. 6 and 7 is used. Reference 6 incorporates the experimental data from NSTX which is a MA-class ST with aspect ratio of ~ 1.3 – 1.5. The plasma current limit is calculated according to the formulation of Ref. 7 and the β_N limit is calculated according to the formulation of Ref. 8.

In this work, we consider the case that the plasma current is fully driven by the non-inductive current drive and the bootstrap current. Neutral beam (NB) is chosen for the non-inductive current drive and for its model, we use the formulation developed in Ref. 9. Bootstrap current fraction is calculated following the formulation developed for arbitrary aspect ratio [10, 11]. For other physics constraints, we refer to Ref. 12-.

We consider an ST reactor where the blanket and shield are installed inside the vacuum vessel surrounding the plasma. Central solenoid coil and the inboard blanket are discarded. Then the radial build of a reactor consists of toroidal field coils, vacuum vessel, shield, blanket and plasma as shown in Fig. 1. There are various engineering constraints, such as the radial build, the critical current density in the superconducting coil, the maximum TF field, the stress limit, the breeding requirement and the shield requirements. The radial build of these components should be determined by the physics and engineering constraints which they should satisfy.

Toroidal magnetic field (TF) and the TF coil current density at that field have an impact on the system design. Ampere's law relates the maximum toroidal magnetic field at the inner leg of the TF coil, B_{max} to the operating current density and the width of winding pack. The operating current density is limited by the critical current density of superconductor. For the conductor of the TF coil, a high temperature super-conducting(SC) material, Bi2212 is assumed. SC filament operation current density is assumed to be 0.8 times the critical current density. The number of the TF coils is assumed to be 16. The constraint that in-plane stresses in the winding pack have to be within the allowable stress has an impact on the design of coil case. The design stress of structure material for the TF coil case is assumed to be 800 MPa.

The vacuum vessel is 100 mm thick stainless steel type SS316LN which is the same material as the ITER vacuum vessel. The thickness of the scrape off layer (SOL) is set to be 0.1 m.

The tritium production is mainly made by outboard blankets consisting of the He-cooled lithium lead (LiPb) as tritium breeding and neutron multiplying material, and the reduced activation ferritic steel as structural material. Sufficient space for the blankets should be maintained to maximize the tritium breeding ratio and the energy multiplication. Shield thickness is also closely related to the protection of the TF coil against radiation damage.

For neutronic optimization of the blanket and the shield, the quantities such as the tritium breeding ratio (TBR), radiation effects on the TF coil have to be calculated. For self-consistent determination of blanket and shield with other components of the reactor system, a system analysis code [12] is coupled with the one-dimensional radiation transport code, ANISN [13]. ANISN calculates the neutronic response of the components, with 30 neutron group cross section library based on JENDL-3.2 [14]. For the estimation of the local tritium breeding ratio (TBR), the JENDL dosimetry file is used.

3. Impact of Shield Materials in the Size of a ST Reactor

3.1. Characteristics of various shielding materials

It was shown [5] that metal hydrides and borohydrides as advanced shielding materials have superior neutron shielding capability compared to the conventional materials due to their high density of hydrogen. It was also shown [15] that a mixture of tungsten and metal hydride gives improved performance with respect to activation parameters of importance to waste management. We investigate the impact of various shield materials in the design of a ST reactor. The calculation model is a toroidal cylindrical geometry of a ST reactor as shown in Fig. 1. The blanket is located on the outboard side, and it is 80 cm in thickness.

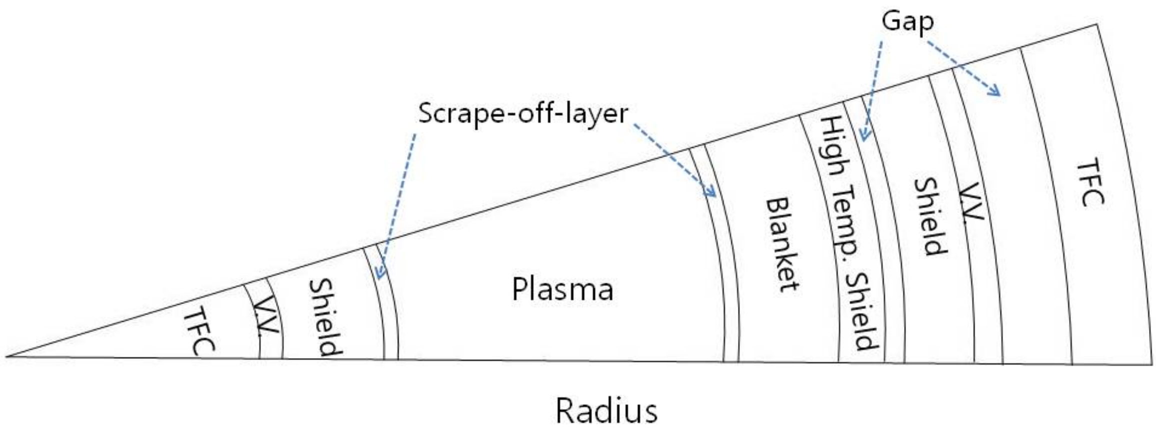


FIG.1. Calculation model of a ST reactor.

Shielding characteristics and tritium breeding capability with various inboard shield materials are summarized in Table 1 when A = 2.0 and B_0 = 3.0T, and in Table 2 when A= 1.5 and B_0 = 1.5T. Major radius R_0 and the inboard shield thickness were determined to give P_{fusion} = 3.0 GW. The shielding capability is the best for WC. W, $Mg(BH_4)_2$ and TiH_2 also shows good shielding performance. Tritium breeding ratio of the outboard blanket increase with Pb, Be and W in the inboard shield due to the low energy neutrons produced in the inboard shield material.

TABLE I: Shielding characteristics and tritium breeding capability when A = 2.0 and B_0 = 3.0T

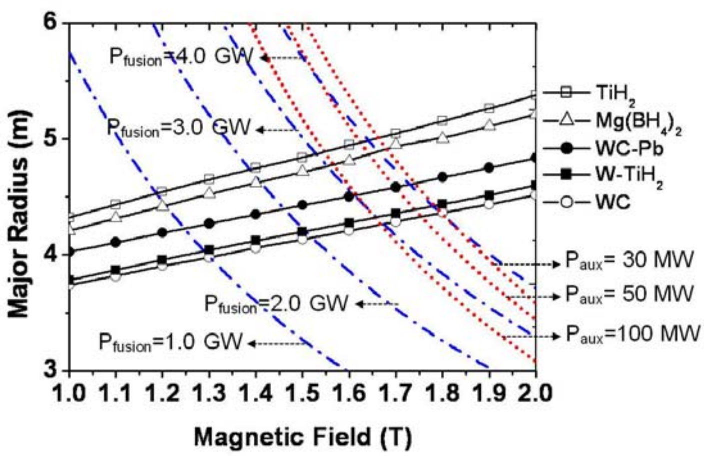
Inboard Shield	W	WC	TiH_2	ZrH_2	$Mg(BH_4)_2$	Be	Pb
Insulator dose (rad)	3.97×10^{6}	1.40×10^{6}	3.62×10^{8}	8.46×10^{8}	1.56×10^{8}	2.84×10^{10}	1.73×10^{13}
Maximum fast neutron Fluence (n/cm^2)	1.29×10^{15}	9.13×10^{13}	1.22×10^{16}	2.89×10^{16}	5.19×10^{15}	1.20×10^{18}	2.15×10^{21}
Tritium Breeding Ratio	1.321	1.302	1.124	1.147	1.101	1.317	1.479

TABLE II: Shielding characteristics and tritium breeding capability when A = 1.5 and B_0 = 1.5T

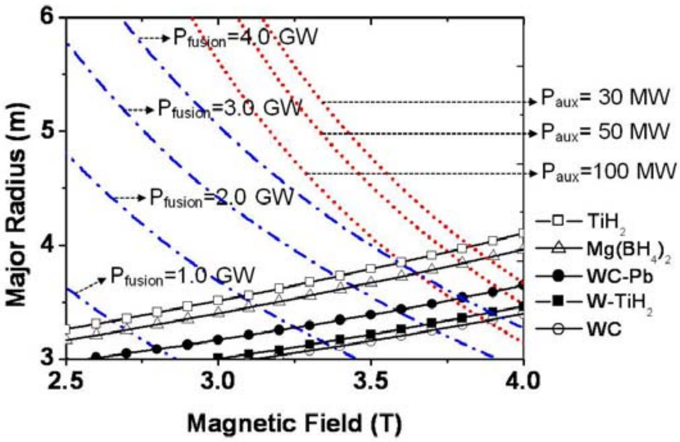
Inboard Shield	W	WC	TiH_2	ZrH_2	$Mg(BH_4)_2$	Be	Pb
Insulator dose (rad)	1.89×10^{8}	8.41×10^{7}	5.77×10^{9}	1.1×10^{10}	3.09×10^{9}	2.06×10^{11}	2.23×10^{13}
Maximum fast neutron fluence (n/cm^2)	4.70×10^{16}	5.46×10^{15}	1.95×10^{17}	3.85×10^{17}	1.03×10^{17}	8.68×10^{18}	2.51×10^{21}
Tritium Breeding Ratio	1.399	1.386	1.259	1.276	1.242	1.396	1.503

3.2. The Minimum Major Radius

The coupled system analysis code is used to find the minimum major radius R_0 of ST Reactor with the aspect ratio in the range of 1.5 to 2.0. We assume the maximum plasma performance with $q_a = q_{a,min}$, $\beta_N = \beta_{N,max}$, H = 1.2, and n/n_G =1.2. Figures 2 show a plasma performance as the major radius R_0 and the magnetic field at magnetic axis B_T vary in the case of the aspect ratio, A=1.5 and 2.0. For a given fusion power, large B_T is preferrable for small R_0 and small auxiliary heating power. For the inboard shield material, we selected 5 materials, WC, Mg(BH4)2, TiH_2, a mixture of W(80%) and TiH_2(20%), WC-Pb(20 cm thick Pb layer added to the WC shield). A required inboard shield thickness is determined by the requirement on the protection of the TF coil against insulator radiation dose limit of 10^{10} rad. The minimum major radius is mainly determined by the radial build constraint given by the shielding and the magnetic field at TFC. WC allows smaller major radius than any other shield materials due to its superior shielding capability. This is due to the fact that confinement characteristics is favorable in the low aspect ratio case and less auxiliary heating power is required.



(a) A=1.5



(b) A=2.0

FIG.2. Minimum major radius and plasma performance as a function of B_T.

Figure 3 shows the dependence of the minimum major radius and required auxiliary heating power on the aspect ratio for the inboard shield material of WC, W- TiH_2 mixture and WC-Pb when P_{fusion} = 3.0 GW. The minimum major radius decreases as the aspect ratio increases but the required auxiliary heating power increases with the aspect ratio. Thus the aspect ratio less than 1.6 is allowed when Q > 30 is required. With the confinement enhancement factor, H > 1.2, the required auxiliary heating power becomes smaller and the aspect ratio bigger than 1.6 is allowed.

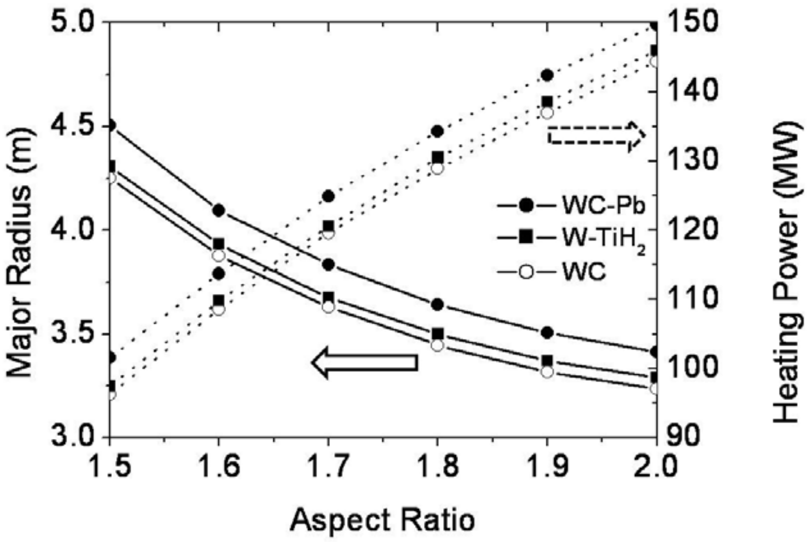
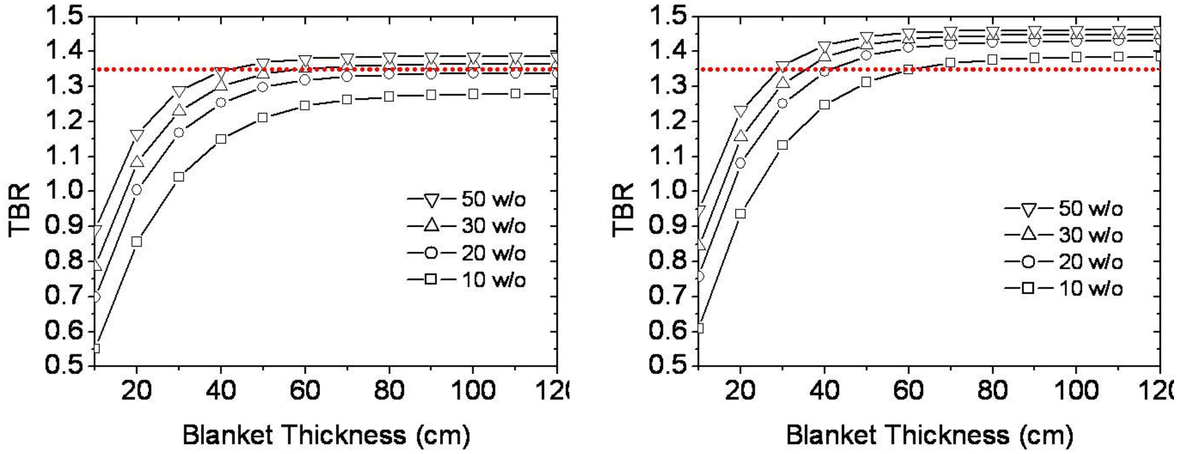


FIG.3. Minimum major radius and required auxiliary heating power as a function a aspect ratio.

3.3. The Minimum Size ST Reactor

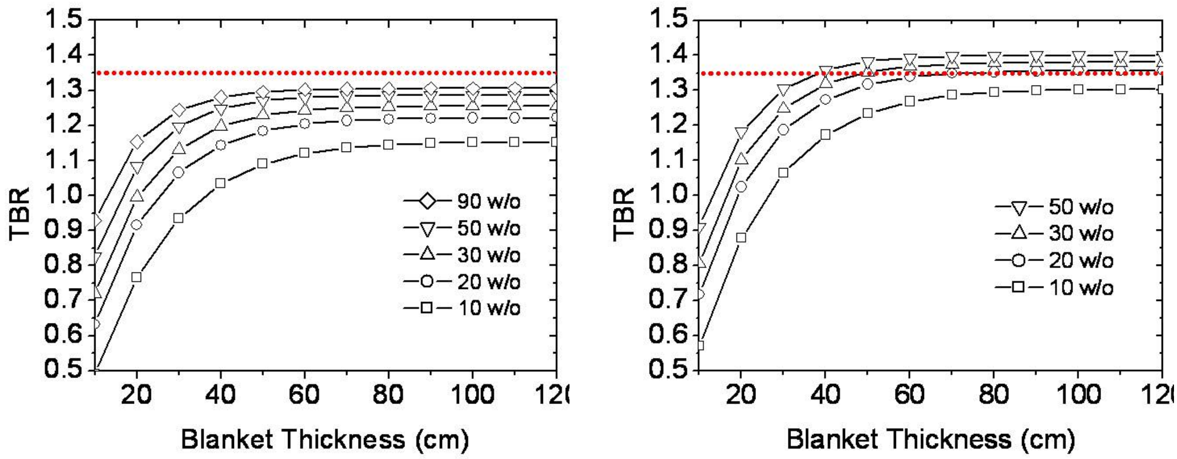
For the inboard shield materials of WC and WC-Pb, we investigate the impact of inboard materials on the TBR with system parameters determined to have the minimum major radius when P_{fusion} = 3.0 GW.

Figures 4 and 5 show the TBR as a function of the outboard blanket thickness when aspect ratios are A = 1.5 and 2.0, respectively. TBR increases sharply with the outboard blanket thickness initially but saturates when the outboard blanket thickness is bigger than 80 cm. The TBR when A = 2.0 is smaller compared to the case with A = 1.5. This is due to the fact that A = 2.0 case has less neutron wall loading at outboard side than A = 1.5 case for a given fusion power. It is also found that adding a reflector of 0.2 m thick Pb to the inboard shield effectively improves the TBR, which indicates that the neutron reflection and neutron multiplication effects by Pb is very effective.



(a) WC *(b) WC-Pb*

FIG.4. TBR as a function of outboard blanket thickness when A = 1.5.

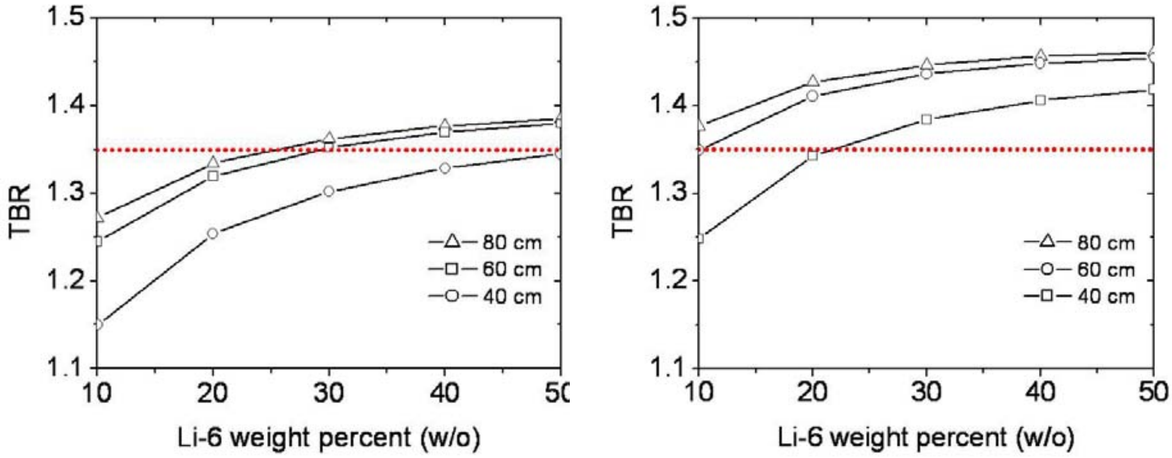


(a) WC *(b) WC-Pb*

FIG.5. TBR as a function of outboard blanket thickness when A = 2.0.

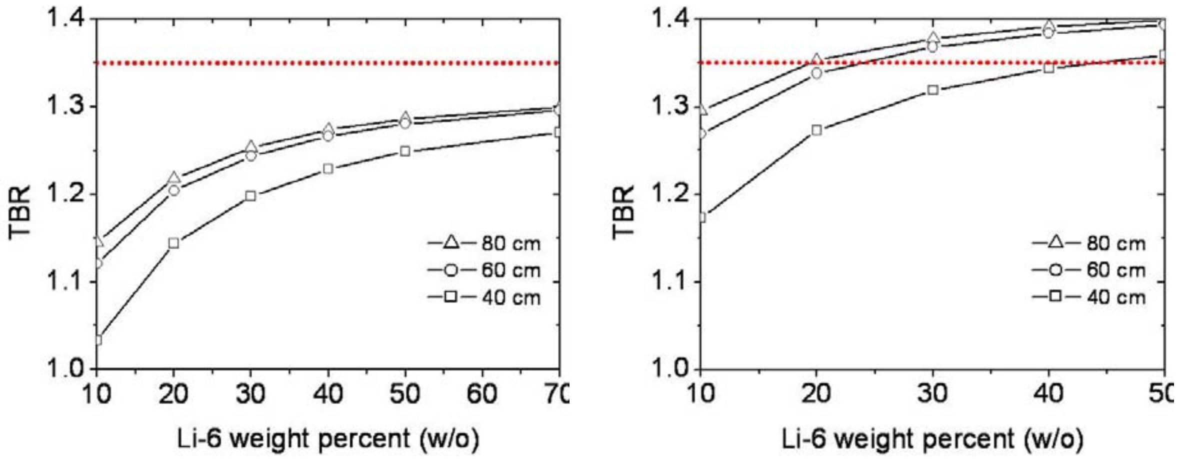
Figures 6 and 7 show that the TBR as a function of the Li-6 enrichment when the aspect ratio A = 1.5 and 2.0, respectively. When A = 1.5, for TBR to be bigger than 1.35, more than 50 % Li-6 enrichment and the outboard blanket thickness more than 40 cm are required for WC and ~ 20 % Li-6 enrichment is enough for WC-Pb. When A = 2.0, it is difficult for TBR to be bigger than 1.35 with WC inboard shield material.

These results tell us that the minimum major radius does not always lead to the minimum reactor size. Material for the inboard shield need to be taken into account in designing the ST reactor since it has influence on the TBR and the outboard blanket thickness



(a) WC *(b) WC-Pb*

FIG.6. TBR as a function of Li-6 enrichment when A = 1.5.



(a) WC *(b) WC-Pb*

FIG.7. TBR as a function of Li-6 enrichment when A = 2.0.

4. Summary

For self-consistent calculation of the physical and engineering constraints which relate the various components of a tokamak reactor, the system analysis code was coupled with the one dimensional radiation transport code, ANISN.

It was shown that the ST reactor at lower aspect ratio of 1.5 with outboard blanket only can provide tritium self-sufficiency by using an inboard neutron reflector instead of breeding blanket. The reflecting shield provides not only protection for the superconducting TF coil but also improved neutron economy for the tritium breeding in outboard blanket.

In addition to high temperature superconducting magnets with high critical current density, reflecting shield open up the possibility of a compact superconducting ST reactor and smaller auxiliary heating power simultaneously at low aspect ratio.

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