IOP Publishing and International Atomic Energy Agency Nuclear Fusion
Nucl. Fusion **51** (2011) 113013 (6pp) doi:10.1088/0029-5515/51/11/113013

Conceptual design study of a superconducting spherical tokamak reactor with a self-consistent system analysis code

B.G. Hong[1], Y.S. Hwang[2,4], J.S. Kang[2], D.W. Lee[2], H.G. Joo[2] and M. Ono[3]

[1] Chonbuk National University, 587 Baekje-daero, Duckjin-gu, Jeonju-si, 561-756, Republic of Korea
[2] Seoul National University, 599 Gwanak-ro, Gwanak-gu, 151-744 Seoul, Republic of Korea
[3] PPPL, Princeton, NJ 08543, USA

E-mail: yhwang@snu.ac.kr

Received 5 January 2011, accepted for publication 19 September 2011
Published 24 October 2011
Online at stacks.iop.org/NF/51/113013

Abstract
In a spherical tokamak (ST) reactor, the radial build of toroidal field coil and the shield play a key role in determining the size of the reactor. For self-consistent determination of the reactor components and physics parameters, a system analysis code is coupled with a one-dimensional radiation transport code. A conceptual design study of a compact superconducting ST reactor with an aspect ratio of up to 2.0 is conducted and the optimum radial build is identified. It is shown that the use of an improved shielding material and high-temperature superconducting magnets with high critical current density opens up the possibility of a fusion power plant with compact size and small re-circulating power simultaneously at a low aspect ratio, and that by using an inboard neutron reflector instead of a breeding blanket, tritium self-sufficiency is possible with an outboard blanket only and thus a compact-sized all superconducting coil ST reactor is viable.

(Some figures in this article are in colour only in the electronic version)

1. Introduction

Spherical tokamak (ST) plasmas have 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 $A = 1.6$, showing a conceptual projection of a 1000 MWe electric power plant with copper magnets. However, the resistive losses in the copper toroidal field (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 (SC) magnet and improved shielding material technology, we present here a conceptual design study of a 1000 MWe-class superconductor based ST power plant with minimum re-circulating power. In previous studies, SC tokamak reactors with tight aspect ratios such as VECTOR [2] were proposed with a limited aspect ratio down to only 2.3. In a ST reactor, the radial build of the SC TF coil and the shield play a key role in determining the size of the reactor and they should be determined by the self-consistent system analysis where the system analysis code [3] is coupled with the radiation transport analysis code.

To find space for the radiation shielding of the SC 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 SC material [4, 5], promising much higher engineering critical current density of 100 kA cm^{-2} for high magnetic fields beyond 20 T by operating below liquid nitrogen temperature, led us to investigate the possibility of employing the SC 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 SC TF coil. For constraints on the high-temperature SC magnet, the fast neutron fluence limit to the superconductor is set to 10^{23} n m^{-2}, the peak nuclear heating limit in the winding pack is set to 2 mW cm^{-3} and the radiation dose limit absorbed by the insulator is set to 2×10^{10} rad after 40 full power years operation. In addition

[4] Author to whom any correspondence should be addressed.

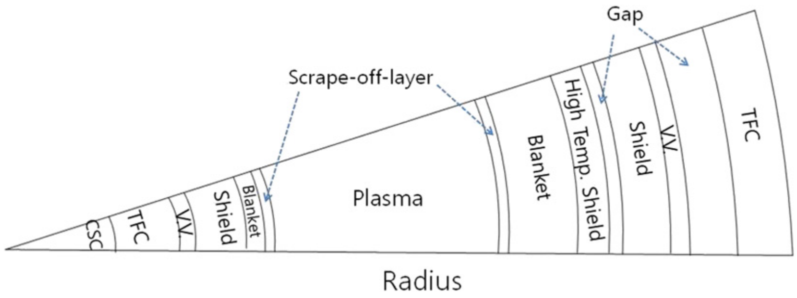


Figure 1. Calculation model of a ST reactor.

to tungsten carbide, which has been considered as a shielding material in many reactor studies, metal hydrides and borohydrides, which are reported [6, 7] to have superior neutron shielding capability due to their high density of hydrogen, are considered. It was also shown [8] that a mixture of tungsten and metal hydride gives improved performance with respect to activation parameters of importance to activated waste management. Shielding characteristics and tritium breeding capability with various shield materials need to be investigated.

For a compact-sized ST reactor, a reactor with an outboard blanket is only considered where tritium self-sufficiency is possible using a simple-structured inboard neutron reflector instead of a breeding blanket. The reflecting shield should not only provide protection for the SC TF coil but also improved neutron economy for tritium breeding in the outboard blanket. Be, graphite, ZrH_2, TiC, Pb, etc are known as good neutron reflectors in fission reactors and we investigate their characteristics in the fusion neutron spectrum.

The system analysis coupled with the radiation transport analysis is explained in section 2. The conceptual design study and an optimum radial build of a ST reactor are presented in section 3. Section 4 summarizes the result of this work.

2. Tokamak reactor system analysis coupled with radiation 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 density limit. The operating parameters of ST plasmas have been estimated previously using observed tokamak scaling and MHD calculations over a range of aspect ratios and limits on key physics parameters [9–11]. Reference [9] incorporates the experimental data from NSTX, which is a MA-class ST with an aspect ratio of ~1.3–1.7. The maximum elongation κ depends on the aspect ratio and we use the average of the expressions derived in [9, 10]. The plasma current is calculated according to

$$I_p = \frac{5a^2 B_0}{R_0 q_a}\frac{(1+\kappa^2)}{2} \tag{1}$$

with $q_{a,\min} = 1.21 + 1.3A - 0.25A^2$ calculated following the formulation of [10]. And the β_N limit is calculated according to the following expression [11]:

$$\beta_{N,\max} = \frac{-0.7748 + 1.2869\kappa - 0.2921\kappa^2 + 0.0197\kappa^3}{\tanh[(1.8524 + 0.2319\kappa)/A^{0.6163}]A^{0.5523}/10}. \tag{2}$$

The main mathematical model to capture the physics and technologies is the plasma power balance equation, which is represented as

$$P_{con} + P_{rad} = P_{OH} + P_\alpha + P_{aux} \tag{3}$$

where the conduction (P_{con}) and radiation losses (P_{rad}) are balanced by α particle heating (P_α), auxiliary heating (P_{aux}) and ohmic heating (P_{OH}). For the confinement scaling, the H-mode IPB98y2 scaling law [12] is used:

$$\tau_E = H\tau_E^{\mathrm{IPB98}(y,2)} \tag{4}$$

$$\tau_E^{\mathrm{IPB98}(y,2)} = 0.0562 I_p^{0.93} B_T^{0.15} (P_{con}\mathrm{vol})^{-0.69} \overline{n}_{19}^{0.41} M^{0.19} R_0^{1.97} \times \left(\frac{a}{R_0}\right)^{0.58} \kappa^{0.78} \tag{5}$$

where I_p is the plasma current (MA), P_{con} is the power loss (MW), n_{19} is the line-averaged density (10^{19} m^{-3}), B_0 is the toroidal magnetic field (T) at the magnetic axis, M is the fuel mass number (amu), R_0 is the major radius (m), a is the minor radius (m) and κ is the plasma elongation. In equation (2), H represents the confinement enhancement factor. For other physics constraints, we refer to [3].

We consider an ST reactor where the blanket and shield are installed inside the vacuum vessel surrounding the plasma. With the inboard blanket discarded, the radial build of a reactor consists of a central solenoid (CS) coil, TF coil, vacuum vessel, shield and plasma, as shown in figure 1. There are various engineering constraints, such as the critical current density in the SC 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.

The radial build of the CS coil is largely limited by the difference between major and minor radii at a given aspect ratio, and determines its capacity in providing the magnetic flux required for a plasma current ramp-up which can be expressed as

$$\Delta\Psi = L_p I_p + C_{\mathrm{Ejima}}\mu_0 R_0 I_p \tag{6}$$

where L_p is the plasma inductance, μ_0 is the vacuum permeability and C_{Ejima} is the Ejima coefficient.

Toroidal magnetic field 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 the winding pack. The operating current density is limited

Table 1. Shielding characteristics and tritium breeding capability when $A = 2.0$ and $B_0 = 3.0$ T.

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}
TBR	1.321	1.302	1.124	1.147	1.101	1.317	1.479

Table 2. Shielding characteristics and tritium breeding capability when $A = 1.5$ and $B_0 = 1.5$ T.

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}
TBR	1.399	1.386	1.259	1.276	1.242	1.396	1.503

by the critical current density of the superconductor. For the conductor of the TF coil, a high-temperature SC material, Bi2212, is assumed and its critical current density is obtained by interpolation from the experimental curve for a Bi2212 wire at 4 K shown in figure 1 of [5]. The SC filament operation current density is assumed to be 0.8 times the critical current density. The number of TF coils is assumed to be 16. The stress in the winding pack has to be within the allowable stress limit and it determines the thickness of the coil case. With the case made of SS316LN, the design stress limit of the TF coil case is set to 550 MPa.

The vacuum vessel is 10 cm 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. For the first wall, 2 cm thick reduced activation ferritic steel is used. Tritium breeding is mainly produced by the outboard blankets consisting of He-cooled lithium lead (LiPb) as the breeding and neutron multiplying material, and the reduced activation ferritic steel as the structural material. Sufficient space for the blankets should be maintained to maximize the tritium breeding ratio (TBR) and energy multiplication. The shield thickness is determined to protect the TF coil against radiation damage.

For neutronic optimization of the blanket and shield, quantities such as the TBR and the radiation effects on the TF coil have to be calculated. For self-consistent determination of the blanket and the shield thickness, a system analysis code [3] is coupled with a one-dimensional radiation transport code, ANISN [13]. ANISN calculates the neutronic response of the reactor components, with 30 neutron group cross section library based on FENDL-2.1 [14]. Cross section library and activity tables were prepared using the NJOY99 program [15]. The TBR is calculated by the following equation:

$$\mathrm{TBR} = T_6 + T_7 \tag{7}$$

$$T_6 = \int \mathrm{d}V\,\mathrm{d}E\phi N_6 \sigma_{\mathrm{Li-6}(n,\alpha)T} \tag{8}$$

$$T_7 = \int \mathrm{d}V\,\mathrm{d}E\phi N_7 \sigma_{\mathrm{Li-7}(n,n',\alpha)T} \tag{9}$$

For the calculation of $\sigma_{\mathrm{Li-6}(n,\alpha)T}$ and $\sigma_{\mathrm{Li-7}(n,n',\alpha)T}$, the JENDL dosimetry file [16] was used.

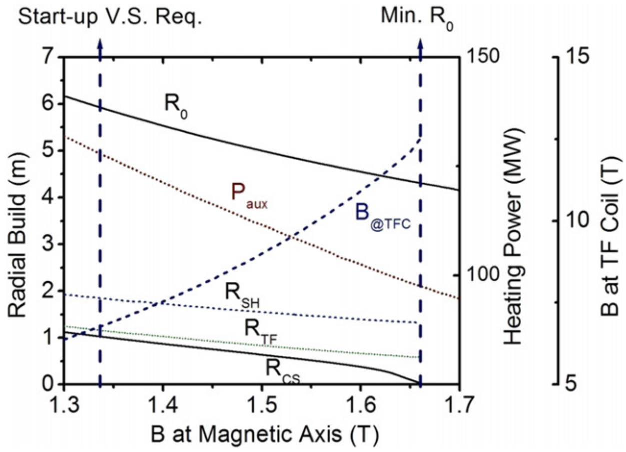


Figure 2. Radial build of a ST reactor with aspect ratio $A = 1.5$.

3. Design study of a ST reactor

3.1. Characteristics of various shielding materials

We investigate the impact of various shield materials on the design of a ST reactor. The calculation model is a cylindrical torus geometry of a ST reactor, as shown in figure 1.

Shielding characteristics and tritium breeding capability with various inboard shield materials are summarized in table 1 when $A = 2.0$ and $B_0 = 3.0$ T, and in table 2 when $A = 1.5$ and $B_0 = 1.5$ T. The inboard shield thickness was set to 1.0 m and the major radius R_0 was determined to give $P_{\mathrm{fusion}} = 3.0$ GW. The shielding capability is the best for WC. W, $Mg(BH_4)_2$ and TiH_2 also show good shielding performance. TBR of the outboard blanket increases with Pb, Be and W in the inboard shield due to the low-energy neutrons produced in the inboard shield material.

3.2. Radial build of a ST reactor

The coupled system analysis code is used to find the radial build of a ST reactor with the aspect ratio in the range 1.5–2.0. The plasma performance is assumed to give a fusion power of 3.0 GW with $q_a = q_{a,\min}$, $\beta_N = \beta_{N,\max}$, $H = 1.2$ and $n/n_G = 1.0$.

Figures 2–4 show the radial build of a ST reactor as the magnetic field at the magnetic axis, B_0, varies. For a given

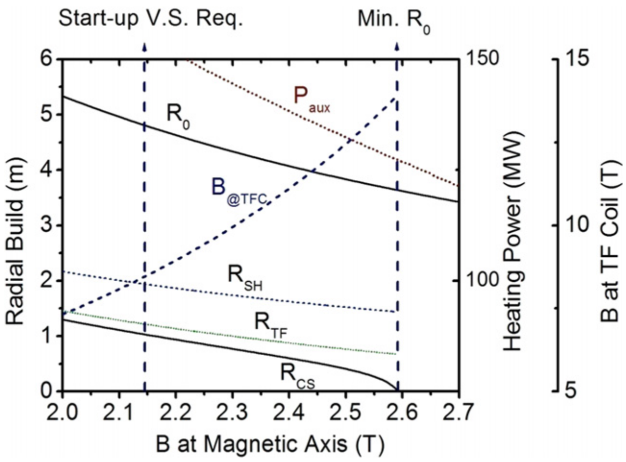


Figure 3. Radial build of a ST reactor with aspect ratio $A = 1.75$.

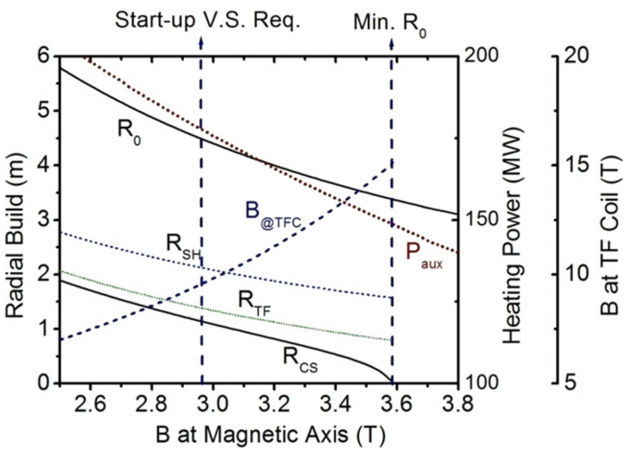


Figure 4. Radial build of a ST reactor with aspect ratio $A = 2.0$.

fusion power, larger B_0 is preferable for smaller major radius R_0 and smaller auxiliary heating power. For the inboard shield material, WC is chosen since WC has a superior shielding capability, as shown in tables 1 and 2. The required inboard shield thickness is determined mainly by the requirement on the protection of the TF coil against an insulator radiation dose limit of 2×10^{10} rad. As B_0 increases, the major radius R_0 decreases with decreased minor radius for a given fusion power at a fixed aspect ratio while the TF coil thickness (R_{TF}–R_{CS}) increases with the magnetic field at the TF coil and the shield thickness (R_{SH}–R_{TF}) increases with the increased neutron wall loading. Then the space available for the CS coil (R_{CS}) continuously decreases as B_0 increases. Thus with the given aspect ratio and plasma performance, the minimum major radius where $R_{CS} \sim 0$, which is indicated 'Min. R_0' in the figures, is determined mainly by the constraints of the shielding and the magnetic field at the TF coil.

In figures 2–4, the R_0 value which allows the space for the CS coil to provide one-tenth of the magnetic flux required for plasma current ramp-up is indicated as 'Start-up V.S. req.'. At least 1 m is needed for the CS coil thickness and it leads to bigger reactor size with larger major radius than the 'Min. R_0'. Thus for the range of $A = 1.5$–2.0, CS-less configuration is preferable although it requires the development of efficient

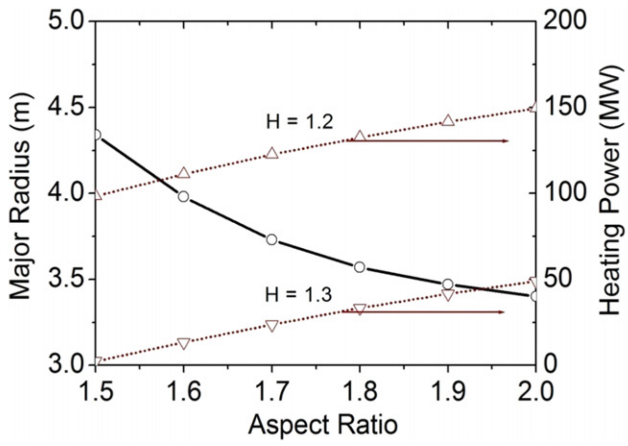


Figure 5. Minimum major radius and required auxiliary heating power as a function of aspect ratio.

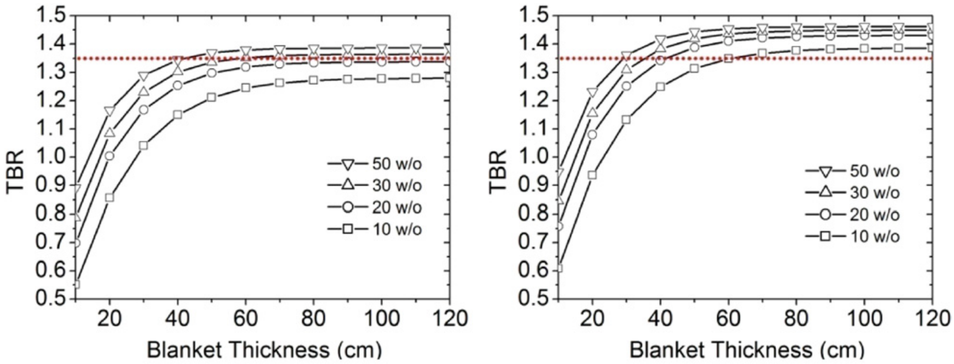
non-inductive current drive technology for current ramp-up and control.

Figure 5 shows the dependence of the minimum major radius and the required auxiliary heating power on the aspect ratio when $P_{\text{fusion}} = 3.0$ GW. The minimum major radius decreases as the aspect ratio increases but the required auxiliary heating power increases with the aspect ratio. This is due to the fact that confinement characteristics are favourable in the low aspect ratio case and less auxiliary heating power is required. With the enhanced confinement factor, $H = 1.2$, the aspect ratio less than 1.6 allows for $Q > 30$. As the aspect ratio increases, enhanced confinement is required to have $Q > 30$. In the figure, the required auxiliary heating power becomes small with $H = 1.3$.

3.3. *Consideration of tritium breeding capability*

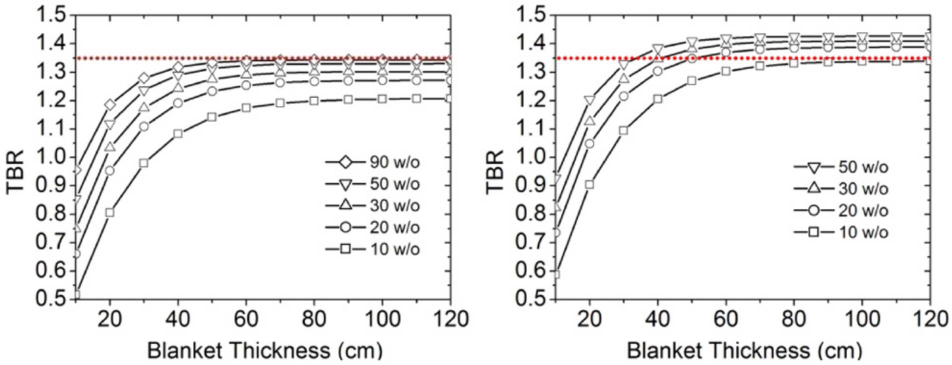
It is shown in tables 1 and 2 that the TBR of the outboard blanket increases with Pb in the inboard shield. We investigate the impact of inboard materials on the TBR with the inboard shield materials of WC and WC–Pb (20 cm thick Pb layer added to the WC shield). The system parameters were determined to have the minimum major radius when $P_{\text{fusion}} = 3.0$ GW. Pb is selected as an extra inboard reflector material since it has a higher (n,2n) cross section for high-energy neutrons compared with Be, as shown in table 1. These cross section data are included in the coupled system analysis code. TBR of at least 1.35 (with 80% coverage of the breeding region, net TBR is 1.08.) in the one-dimensional model is required for tritium self-sufficiency in the fusion reactor.

Figures 6–8 show the TBR as a function of the outboard blanket thickness when the aspect ratios are $A = 1.5$, 1.75 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, since the neutron flux which contributes to tritium breeding decreases sharply as the outboard blanket thickness increases. The TBR of the large A case is smaller compared with that of the small A case. This is due to the fact that the large A case has less neutron wall loading at the outboard side than the small A 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



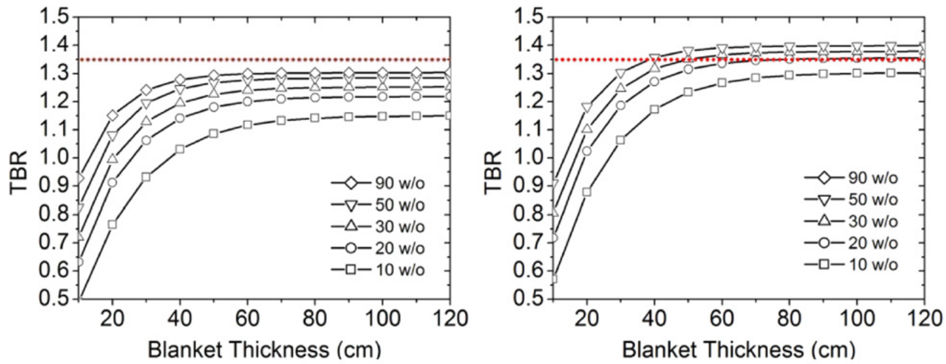
(a) WC *(b) WC-Pb*

Figure 6. TBR as a function of outboard blanket thickness when $A = 1.5$.



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

Figure 7. TBR as a function of outboard blanket thickness when $A = 1.75$.



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

Figure 8. TBR as a function of outboard blanket thickness when $A = 2.0$.

indicates that the neutron reflection and neutron multiplication effects by Pb are very effective. When $A = 1.5$, for the TBR to be bigger than 1.35, more than 50% Li-6 enrichment and an outboard blanket thickness of more than 40 cm are required for WC and $\sim$ 20% Li-6 enrichment is enough for WC-Pb. Figures 7 and 8 show that when $A > 1.75$, it is difficult for the TBR to be bigger than 1.35 with WC inboard shield material. As shown in figure 8, when $A = 2.0$, higher Li-6 enrichment and a thicker outboard blanket are required for the TBR to be bigger than 1.35.

These results tell us that material for the inboard shield should be taken into account in designing the ST reactor since it has an influence on the TBR and the outboard blanket thickness.

4. Conclusion

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

Using the coupled system analysis, the optimum radial build of a ST reactor with an aspect ratio in the range 1.5–2.0

was found. It is noted that to access a design point where $Q > 30$, enhanced confinement is required.

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

Regarding the reactor size, it is found that as A increases the minimum major radius decreases and a small-sized reactor is foreseen, although the required auxiliary heating power increases and enhanced confinement characteristics are required. When the tritium breeding capability is taken into account, a thicker outboard blanket is required as A increases. Thus, the optimum reactor size has to be found by considering not only the inboard radial build but also the tritium breeding capability of the specific blanket design.

The use of an improved shielding material and high-temperature superconducting magnets with high critical current opens up the possibility of a compact superconducting ST reactor and smaller auxiliary heating power simultaneously at a low aspect ratio.

Acknowledgment

This work was supported by the Korea Science and Engineering Foundation (KOSEF) grant funded by the Korea government (MEST) under the contract no. 2010-0001839.

References

[1] Najmabadi F. and The ARIES Team 2003 *Fusion Eng. Des.* **65** 143
[2] Nishitani T., Yamauchi M., Nishio S. and Wada M. 2006 *Fusion Eng. Des.* **81** 1245
[3] Hong B.G., Lee D.W. and In S.R. 2008 *Nucl. Eng. Technol.* **40** 1
[4] Shen T. *et al* 2010 *Supercond. Sci. Technol.* **23** 025009
[5] Dahlgreen F., Brown T., Heitzenroeder P., Bromberg L. and The ARIES Team 2006 *Fusion Eng. Des.* **80** 139
[6] Kim Y., Hong B.G. and Kim C.H. 2005 *Fusion Eng. Des.* **75–79** 1067
[7] Hayashia T., Tobita K., Nishio S., Ikeda K., Nakamori Y. and Orimo S. 2006 *Fusion Eng. Des.* **81** 1285
[8] Taylor N.P. 2002 Safety and environmental considerations in the selection of fusion power plant design options *Proc. 19th Symp. on Fusion Engineering (Atlantic City, NJ, USA, 2002)* p 107 http://www.pppl.gov/sofe01/
[9] Menard J. *et al* 2003 *Princeton Plasma Physics Laboratory Report* PPPL-3779
[10] Wong C. *et al* 2002 *Nucl. Fusion* **42** 547
[11] Lin-Liu Y. and Stambaugh R. 2002 *Optimum General Atomics Report* GA-A23980
[12] Shimada M. *et al* 2007 ITER Physics Basis *Nucl. Fusion* **47** S1
[13] 1994 ANISN-PC, CCC-514(ANISN-PC), Oak Ridge National Laboratory
[14] Aldama D.L. and Trkov A. 2004 FENDL-2.1 Update of an evaluated nuclear data library for fusion applications *IAEA Report* INDC(NDS)-467
[15] MacFarlane R.E. and Muir D.W. 1994 The NJOY Nuclear Data Processing System, Version 91 *LANL Report* LA-12740-M
[16] Nakagawa T., *et al* 1995 Japanese evaluated nuclear data library version 3 reversion-2: JENDL-3.2 *J. Nucl. Sci. Technol.* **32** 1259