Blanket and Shield Design Options for ASRA6C – A Stellarator Power Reactor

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Abstract

A major task of the ASRA6C study was to reduce the dimensions of the reactor by using a thin blanket and efficient shield. Four blanket options have been considered for ASRA6C. The main differences between the different options lie in the blanket geometry and in the way the blanket inner surface tracks the plasma contour. All blankets utilize liquid $Li_{17}Pb_{83}$ as the breeder, Be as the multiplier/moderator (except one option requires no Be), He gas as the coolant, and HT-9 as the structure. A comparison study between the four options was carried out on the basis of neutronics performance, configuration, maintainability, mass utilization, and economics. One option was selected on overall merits and is considered as the reference blanket design for ASRA6C. This blanket has a constant elliptical cross section and uniform thickness. The Be is in pebble bed form at 55 vol% and is surrounded by 14 vol% LiPb. The He coolant is at 8 MPa pressure and is contained within 1 cm diameter tubes immersed in the LiPb/Be mixture. The blanket is 21 cm thick and yields an overall tritium breeding ratio and energy multiplication of 1.05 and 1.2, respectively. The tritium is recovered by slowly circulating the LiPb, resulting in a total blanket inventory of < 6 g.

Introduction

The ASRA6C study [1] is a joint effort between IPP (Garching,Germany), KfK (Karlsruhe,Germany), and Fusion Power Associates (USA). The study is directed toward the clarification of critical issues of advanced modular stellarator reactors. The ASRA6C reactor has a major radius of 20 m. There are 6 coils in each of the 5 field periods. The 30 coils have identical elliptical inner bores. The fusion power is ~ 4 GW and 20 full power years (FPY) of operation are expected. An overall tritium breeding ratio (TBR) > 1.05 is a design goal for the ASRA6C blanket and as large an energy multiplication (M) as possible is highly desirable to improve the reactor economics.

Since the plasma in a stellarator has a helical twist in the toroidal direction, the question arises as to whether the blanket shape should conform to the plasma or be made independent of the plasma shape. For example, since the plasma in ASRA6C can be contained in an elliptical chamber of uniform cross section when viewed in the toroidal direction, a blanket with a uniform elliptical cross section would avoid complicated shapes and be the simplest to fabricate. On the other hand, a blanket which conforms to the shape of the plasma may reduce the size of the magnets, achieving a more attractive reactor from the cost standpoint. For this reason, it was decided to do a trade study comparing suitable blanket options on the basis of neutronics, configuration, maintainability, mass utilization and economics. The four blanket options are described in the next section. This is followed by a comparison study and then, a detailed design of the selected blanket is presented.

Description of Blanket Options

Figure 1 shows the four blanket options.

Option I

The blanket, reflector, and shield are uniformly elliptical in the toroidal direction. Dimensions are selected to insure that the plasma, which changes shape in the toroidal direction, always fits within the provided envelope. The blanket is thin and is composed of He cooled LiPb/Be/HT-9. The reflector is made of HT-9 and is cooled in series with the blanket using the same He gas. It is followed by a shield composed of layers of B_4C and Pb and cooled with water.

Option II

This design is a variation of Option I with the difference being in the geometry only. Here the blanket follows the contour of the plasma, and thus has a helical twist in the toroidal direction, repeating the same shape every field period. The blanket has a non-uniform cross section when viewed in the toroidal direction. The reflector and shield also conform to the shape of the plasma but are made of the same materials as in Option I and are cooled in the same way.

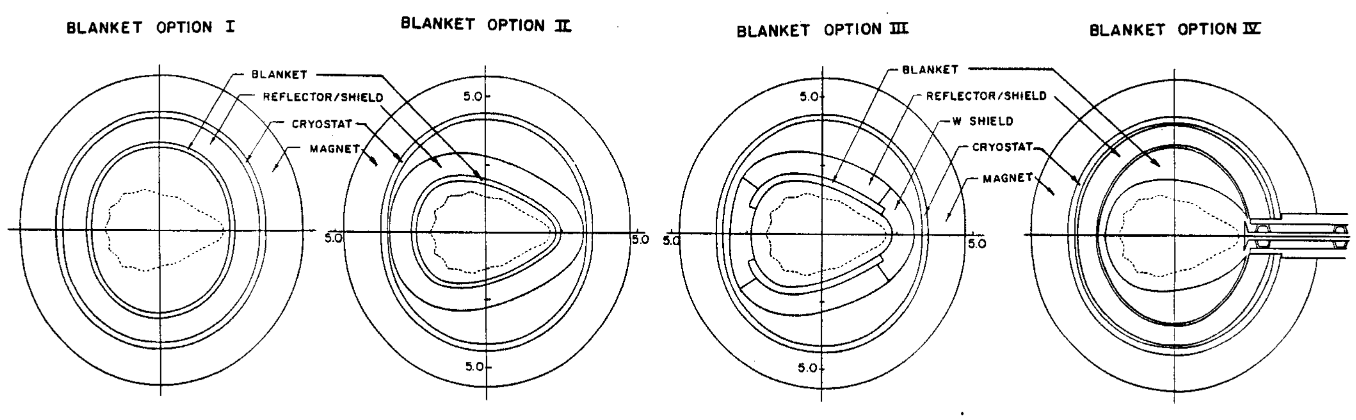

Fig. 1. Schematic representation of the four blanket options.

Option III

This design is a variation of Option II. Here it was decided to use a denser shield at the points where the plasma makes its closest approach to the coils. To do this, it was necessary to segment the blanket in order to take maximum advantage of the denser shield at these locations. The reflector and shield behind the blanket are the same as in the previous two options, but are segmented. The denser shield is composed of a He cooled tungsten layer, followed by water cooled layers of B_4C and Pb.

Option IV

In this option the first wall, which is integral with the blanket, follows the contour of the plasma, but the back surface of the blanket is uniformly elliptical. For this reason, the blanket thickness varies both in the poloidal as well as the toroidal direction. Because in places the blanket thickness was as high as 110 cm, it was decided not to use Be and instead adopt a He gas cooled $Li_{17}Pb_{83}$ blanket. The reflector and shield are of constant thickness and uniformly elliptical. They are composed of the same materials as in Option I, and are cooled in the same way.

Blanket Comparison

Neutronics

The common design criteria for all the options are as follows:

Overall TBR	1.05 - 1.10
Average nuclear heating in front layer of winding pack	~ 0.1 mW/cm^3
Peak fast neutron fluence (E_n > 0.1 MeV) in Nb_3Sn	1.5 x 10^{19} n/cm^2

The neutronics comparison is summarized in Table 1. The comparison was carried out for an early design of ASRA6C where the fraction of the penetration area amounted to 3-4% of the first wall surface area. In Options I, II and III, the blanket thickness is determined such that it yields an overall TBR in the range of 1.05-1.1. Blanket III is thicker than blanket I and II to compensate for the decrease in breeding due to the loss in blanket coverage. The energy multiplication is based on the energy recovered from both blanket and reflector. According to this neutronics analysis, all blankets will be self-sufficient in tritium and Option I gives the highest energy multiplication which may lead to the lowest cost of electricity. The peak radiation effects in the superconducting (S/C) magnets are listed in the table. In all cases, the magnet radiation limits are met. In Options II, III, and IV, the S/C magnets are overprotected. Therefore, the shield thickness and thus, the inner bore dimensions of the magnets can be reduced by 10, 28, and 10 cm, respectively, and all radiation limits are still satisfied.

Configuration and Maintainability

All four blankets are cooled with He gas at 80 atm. The containment structures are in the form of small elliptical cells joined together to form a complete removable blanket unit (RBU). In Options I, II and III, all the cells are of constant thickness and circumvent the plasma poloidally. In Option IV, the individual cells are of constant thickness and circumvent the plasma in a helical spiral. There are four RBUs in each field period and each RBU has three supply and three return coolant connections. Table 2 gives a relative judgmental evaluation of the four options. Configuration is judged on design and construction complexity and the ability of the design to accommodate penetrations. Maintainability is judged on the ease of extracting an RBU from the reflector and on the mass of the drained RBU.

As far as design and construction complexity is concerned, Option I is superior due to its simpler geometry. It is equally difficult to accommodate large penetrations in all four options. In Option III, a penetration falling between blanket segments can be accommodated easier. With respect to maintenance, Options I and IV are ahead. Because they have a uniformly elliptical interface between the RBU and the reflector, they can be extracted from each other.

Table 2. Configuration and Maintenance

	I	II	III	IV
Design/construction complexity	M*	D*	D	D
Penetration accommodation	D	D	D	D
Mass of drained RBU (tonnes)	38	27	32	58
Ease of extraction	M	VD*	VD	M

* M = Moderate, D = Difficult, VD = Very Difficult

Table 1. Neutronics Comparison of the Four Options

	Units	I	II	III	IV
Peak/Average Neutron Wall Loading	MW/m^2	2.4/1.41	2.55/1.84	2.55/1.84	~ 2.5/~ 1.6
Blanket Thickness	cm	17.5	17.5	24	15 - 110
Local/Overall TBR		1.14/1.1	1.14/1.09	1.5/1.1	1.17/1.12
Local/Overall M		1.42/1.38	1.42/1.36	1.38/1.33	1.18/1.13
HT-9 Reflector Thickness	cm	46.4	46.4	81.4; 58.1 W; HT-9	45.3
B_4C/Pb Shield Thickness	cm	33.6	33.6	18.6; 41.9	32.7
Radiation Effects in S/C Magnet:					
Peak Nuclear Heating (in innermost layer)	mW/cm^3	0.5*	0.1	0.0042	0.11
Average Nuclear Heating	mW/cm^3	0.1	0.08	< 0.001	0.03
Peak Fast Neutron Fluence to Nb_3Sn	n/cm^2	1 E19*	2.3 E18	1 E17	2.0 E18
Peak Dose in GFF Polyimide	rad	1 E10*	2.1 E9	8 E7	1.88 E9
Peak dpa in Cu Stabilizer	dpa/FPY	4.5 E-4*	1 E-4	4.5 E-6	8.3 E-5
Potential for Magnet IR Reduction	cm	-	10	28	10

*For cross section through 10 cm thick He manifolds.

Economics and Mass Utilization

To allow for the more complex construction in Options II, III and IV, the fabricated cost of the structure was taken higher than in Option I. Filler material such as Be, LiPb, B_4C, Pb and W has the same unit cost in all four options. Table 3 summarizes the economics and mass utilization. The table lists the thermal power using the energy multiplication obtained in the neutronics analysis and the net electric power using a net efficiency of 38%.

Comparison of Results

We can now list the first and second choices in each one of the comparison categories:

Neutronics:	All four options will perform
Configuration:	Option I, Option IV
Maintainability:	Option I, Option IV
Economics:	Option I, Option II
Mass Utilization:	Option II, Option IV

Given that the four options perform neutronically, the choice must be made on the basis of the remaining categories. Option I wins three of the four categories. It is, therefore, concluded that the thin blanket of constant thickness and of uniformly elliptical cross section is the best choice under these circumstances. A detailed description of the selected blanket and the related neutronic analysis, mechanical design, thermal hydraulics, and tritium removal scheme is given in the following section.

Thin Blanket Design

In ASRA6C, the blanket is not viewed as an entity in itself, but rather as a constituent in a series of components, namely blanket, reflector, and shield which together breed tritium, convert nuclear energy, and protect the S/C magnets against radiation. Since the blanket has less shielding performance, keeping its thickness to a minimum reduces the sizes of the reflector, shield, and magnet, and thus, decreases the overall cost. Other advantages for thin blankets include reduced tritium and Li inventories, and light modules which greatly ease the replacement and maintenance process of the blanket.

Neutronics Analysis

In ASRA6C, the heating and pumping ports subtend ~ 10% of the first wall area. To meet the design goal of an overall TBR of 1.05, a local TBR of 1.4 must be achieved. Many iterations were performed to determine the blanket thickness and content that satisfy all requirements. The results indicate that a 21 cm thick blanket yield a local TBR of 1.4. To fulfill the thermal hydraulics and mechanical design requirements, 10 vol% of the space is needed for the HT-9 structure and 20 vol% for the He coolant. The rest of the space is occupied by LiPb and Be balls at an optimum content of 14 vol% and 56 vol%, respectively.

The energy multiplication depends on the reflector thickness as the energy is recovered from both blanket and reflector. The reflector acts as the first layer of the magnet shield. An optimization study was performed to determine the optimum shield composition that adequately protects the magnets. The optimal shield consists of 44 cm thick HT-9 reflector, 25 cm thick B_4C-shield, and 6 cm thick Pb-shield. At some poloidal locations, reflector cutouts are used to accommodate the He manifolds. Taking this reduction in the reflector thickness and the effect of the 10% penetration into account, the overall energy multiplication in ASRA6C amounts to 1.2.

The shield in ASRA6C is designed to minimize the nuclear heating in the magnet. The radiation effects in the magnet vary poloidally and toroidally according to the variation in the neutron wall loading which peaks at the outside midplane at a value of 2.4 MW/m^2 and has an average value of 1.4 MW/m^2. The peak radiation effects occur behind the He manifolds. At these locations, the fast neutron fluence, dose to the GFF polyimide, and dpa rate in the Cu stabilizer peak at 1.4×10^{19} n/cm^2, 10^{10} rads, and 5×10^{-4} dpa/FPY, respectively. The dpa rate implies that the first magnet annealing is needed after 4 FPY. The average nuclear heating in the front layers of the magnets is 0.13 mW/cm^3. The total nuclear heating in the magnets amounts to 24 kW. This requires 7 MW cryoplant power and that corresponds to only 0.4% of the gross electric power of ASRA6C.

Mechanical Design and Thermal Hydraulics

The reaction chamber in ASRA6C is toroidal with a major radius of 20 m. The cross section of the vacuum chamber is uniformly elliptical in the toroidal direction with a vertical dimension of 6.4 m and a horizontal dimension of 4.6 m. Each field period extends 25.12 m toroidally along the axis of the ellipse. There are four blanket modules in each field period with a length on axis of 6.28 m. Each blanket

Table 3. Economics and Mass Utilization

	I	II	III	IV
Blanket mass/cost (tonnes/$\$x10^6$)	1259/175	996/151	1064/183	8937/215
Reflector mass/cost (tonnes/$\$x10^6$)	8713/174	5756/230	3761/150	6121/122
W Shield mass/cost (tonnes/$\$x10^6$)			6095/427	
B_4C Shield mass/cost (tonnes/$\$x10^6$)	1809/74	1361/64	1365/55.7	1412/57.5
Pb Shield mass/cost (tonnes/$\$x10^6$)	1746/9	1374/9.4	1046/8.6	1436/7.5
Total mass/cost (tonnes/$\$x10^6$)	13527/432	9487/454	13331/824	17906/402
Drained Mass (tonnes)	13087	9080	12906	10129
Thermal Power (MW)	4271	4180	4089	3542
Net Electric Power (MWe)	1623	1588	1554	1346
Total Direct Costs ($/kWe)*	266	286	530	299
Mass Utilization (kWe/tonne)**	124	175	117	133

* Based on direct cost of only the blanket, reflector and shield.
** Based on the drained mass of only the blanket, reflector and shield.

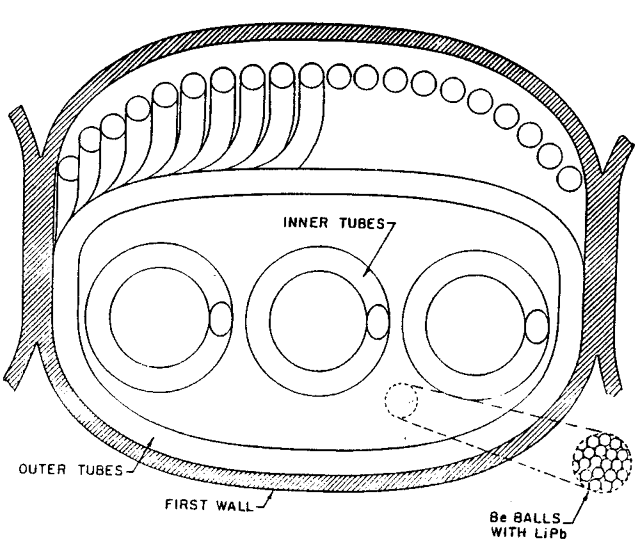

Fig. 2. Cross section of a blanket cell.

module is divided into 28 cells which are oriented circumferentially. The cells are wider (24.0 cm) at the outer perimeter of the module than on the inner perimeter (19.5 cm), thus creating the toroidal shape needed. A cross section of a cell is shown in Fig. 2. The outer walls of the cells are shaped semi-ellipsoidally to make them capable of withstanding a He gas leak. Each cell has individual cooling tubes and only communicates with the adjacent cells through the common manifolds.

The cooling tubes are 1.0 cm internal diameter and have a wall thickness of 0.5 mm. There are two types of cooling tubes in this blanket, the outer and inner tubes. The outer tubes provide cooling for the first wall and absorb a large fraction of the nuclear heating in the front zone of the blanket. These tubes spiral around the inside of the cell, coming in contact with the first wall. Each outer tube makes only seven loops inside the cell before returning to the exit manifold. There are also inner tubes which cool the central part of the cell. These tubes, of which there are three, perform a small radius large pitch spiral, travelling the full length of a cell quadrant. All the inner and outer tubes are the same length and thus present an equal impedance to the flow of He gas.

Since the blanket is only 21 cm thick, 22% of the thermal energy is deposited in the steel reflector. By routing the He gas from the blanket to the reflector, this energy is recovered very efficiently. Helium gas at 275°C enters the blanket and comes out from the reflector at 575°C. The average nuclear heating in the first wall is 11 W/cm^3 and the peak is 18.7 W/cm^3. Surface heat flux can vary between 10 and 20 W/cm^2 depending on impurity control assumption. Table 4 gives the important mechanical and thermal hydraulic parameters of the blanket.

Tritium Removal from Blanket

Tritium is removed by slowly circulating the liquid breeder from the reactor blanket to an external Tritium Removal System (TRS). The low-velocity flow reduces the corrosive effects of the liquid metal on the containment structure and, also, reduces the MHD pressure caused by the liquid metal flowing in the complex magnetic field of the stellarator. Because

Table 4. Mechanical and Thermal Hydraulic Parameters of ASRA6C Blanket

First wall radius (m)	2.32-3.22
Blanket thickness (m)	0.21
First wall thickness (cm)	0.6
Reflector thickness (m)	0.44
Cooling tube ID (cm)	1.0
Cooling tube OD (cm)	1.10
Mass of LiPb (tonnes)	635
Mass of Be (tonnes)	436
Thermal power in blanket (MW)	3266
Thermal power in reflector (MW)	876
He gas pressure (atm)	80
He gas inlet to blanket temperature (°C)	275
He gas inlet to reflector temperature (°C)	510
He gas outlet from reflector temperature (°C)	575
He gas mass throughput (kg/s)	2660
Avg. nuclear heat in FW (W/cm^3)	11
Peak nuclear heat in FW (W/cm^3)	18.7
Avg. FW temp. at nuclear heating peak (°C)	558
Max. temperature of coolant tube (°C)	550
Avg. He gas velocity (m/s)	76
Max. stress in coolant tubes (MPa)	85
Pressure drop in blanket and reflector (MPa)	0.25
Total pumping power (MW)	122

some tritium permeates from the liquid breeder into the helium coolant, an additional TRS is installed to service a small sidestream of the coolant flow. This TRS removes tritium by oxidation followed by adsorption on a desiccant. A desirable operational point is indicated at a liquid metal flow rate of 28 mm/s into the breeder TRS and the diversion of 1% of the helium flow into the coolant TRS. For this condition, the total tritium inventory is 1.9 in the liquid metal and 3.2 g in the entire He circuit.

Conclusions

Among the four blanket options proposed for ASRA6C, one blanket was selected based on neutronics performance, configuration, maintainability, mass utilization, and economics. This blanket has the advantages of being thin and uniform in thickness. It has a uniformly elliptic inner surface when viewed in the toroidal direction. It provides an overall TBR of 1.05 and M of 1.2. The tritium is recovered by slowly circulating the $Li_{17}Pb_{83}$ breeder. The He gas flows at a pressure of 80 atm in HT-9 tubes embedded in the LiPb/Be mixture. The blanket complements the shield in providing adequate protection for the magnets.

It is hoped that the solution of the blanket configuration problem found for ASRA6C can also be applied to other stellarator reactors. As long as the last closed magnetic surfaces can be surrounded by a toroidally uniform elliptical surface, a simple blanket and shield design can be envisaged for stellarators. This, of course, places a premium on the design of a high performance blanket and shield since the space between the first wall and the coil is constrained.

Acknowledgement

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Reference

[1] G. Böhme et al., "Studies of a Modular Advanced Stellarator Reactor ASRA6C," Fusion Power Associates Report, FPA-87-2, May 1987.