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The EU Power Plant Conceptual Study – Neutronic Design Analyses for Near Term and Advanced Reactor Models

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Abstract

A Power Plant Conceptual Study (PPCS) has been conducted in the framework of the European fusion programme with the main objective to demonstrate the safety and environmental advantages and the economic viability of fusion power. Power plant models with limited ("near term concepts") and advanced plasma physics and technological extrapolations ("advanced concepts") were considered. Two near term plant models were selected, one employing a water cooled lithium-lead (WCLL), and the other one a helium cooled pebble bed (HCPB) blanket. Two variants were also considered for the advanced power plant models, one adopting a liquid metal blanket with a self-cooled lithium-lead breeder zone and a helium cooled steel structure ("dual coolant lithium lead", DCLL), and the other one a self-cooled lithium-lead (SCLL) blanket with SiC_f/SiC composite as structural material.

This report provides a detailed documentation of the neutronics design analyses performed as part of the PPCS study for both the near term and advanced power plant models. Main issues are the assessment of the tritium breeding capability, the evaluation of the nuclear power generation and its spatial distribution, and the assessment and optimisation of the shielding performance. The analyses were based on three-dimensional Monte Carlo calculations with the MCNP code using suitable torus sector models developed for the different PPCS plant variants.

Die europäische Leistungsreaktorstudie – Neutronenphysikalische Designanalysen für Reaktoren des nächsten Schritts und fortgeschrittene Varianten.

Zusammenfassung

Im Rahmen des europäischen Fusionstechnologieprogramms wird eine Leistungsreaktorstudie mit dem Ziel durchgeführt, die Vorteile der „Energiequelle Kernfusion“ in Bezug auf Sicherheit und Umweltfreundlichkeit nachzuweisen sowie ihre ökonomische Konkurrenzfähigkeit zu demonstrieren. Es werden Leistungsreaktorvarianten betrachtet, die sowohl kleine („nächster Schritt“) als auch große Extrapolationsschritte („fortgeschritten“) gegenüber der abgesicherten Plasmaphysik und der bereits verfügbaren Technologie erfordern. Für die Reaktoren des nächsten Schritts wurden zwei Blanketkonzepte ausgewählt: das wassergekühlte Lithium-Blei-Blanket („water cooled lithium-lead“, WCLL) und das heliumgekühlte Feststoffblanket mit Partikelbettschüttung („helium cooled pebble bed“, HCPB). Auch für die fortgeschrittenen Leistungsreaktoren wurden zwei Blanketvarianten untersucht: ein selbstgekühltes Lithium-Blei-Blanket mit heliumgekühlter Stahlstruktur (“dual coolant lithium lead”, DCLL) sowie ein selbstgekühltes Lithium-Blei-Blanket mit SiC_f/SiC-Faserverbundwerkstoff als Strukturmaterial („self-cooled lithium-lead“, SCLL).

Dieser Bericht dokumentiert die neutronenphysikalischen Designanalysen, die als Teil der Studie sowohl für die Reaktoren des nächsten Schritts als auch für die fortgeschrittenen Varianten durchgeführt wurden. Schwerpunkte sind die Bestimmung des Tritiumbrutvermögens, die Berechnung der Leistungserzeugung und ihrer räumlichen Verteilung sowie die Optimierung des Abschirmvermögens. Die Analysen basieren auf drei-dimensionalen Monte-Carlo Rechnungen mit Torussektormodellen, die mit dem MCNP–Code für die vier Reaktorvarianten entwickelt wurden.

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1 Introduction

A Power Plant Conceptual Study (PPCS) is being conducted in the frame of the European fusion programme with the main objective to demonstrate the safety and environmental advantages and the economic viability of fusion power [Mar02]. This includes the conceptual design of models of a commercial fusion power plant, their safety, environmental and economic assessment and the demonstration of their credibility and viability. In the first step of the study conducted in 2001/02, plant models with limited physics and technology extrapolations ("near term concepts") were analysed. Two near term plant models, A and B, were considered: one with a water cooled lithium-lead (WCLL) and the other one with a helium cooled pebble bed (HCPB) blanket. Both blanket concepts use the low activation ferritic steel Eurofer as structural material. Power plant variants with advanced plasma physics and technological extrapolations ("advanced concepts") were considered in the second step of the study conducted in 2002. Two variants of advanced power plant concepts, models C and D, were considered: one adopting a liquid metal blanket with a self-cooled lithium-lead breeder zone and a helium cooled steel structure ("dual coolant lithium lead", DCLL), and the other one employing a self-cooled lithium-lead (SCLL) blanket with SiC_f/SiC composite as structural material.

This report provides a detailed documentation of the neutronics design analyses performed as part of the PPCS study for both the near term and advanced power plant models. Main issues are the assessment of the tritium breeding capability, the evaluation of the nuclear power generation and its spatial distribution, and the assessment and optimisation of the shielding performance. The analyses were based on three-dimensional Monte Carlo calculations using suitable torus sector models developed for the different PPCS plant variants.

2 Near Term and Advanced Power Plant Models

PPCS model A is based on a Water Cooled Lead Lithium (WCLL) blanket employing a quasi stagnant pool of the liquid metal breeder Pb-17Li cooled by pressurised water at conditions similar to those of a fission pressurized water reactor [Sar02]. The divertor concept is similar to the one developed in the frame of ITER, employing a water-cooled heat sink with cooper structure. Both the blanket and the divertor concept are based on materials and technologies, which are either already in hand or can be developed with very limited extrapolation of the present status of technology such as the low activation steel Eurofer assumed as structural material. Power conversion system is a saturated steam turbine plant more or less identical to the one used in fission pressurized water reactors (PWR). Plasma physics models are very similar to the ones employed in the ITER design and therefore represent a limited extrapolation of the present state of the art.

PPCS model B is based on a Helium Cooled Pebble Bed (HCPB) blanket employing the lithium ceramics Li_4SiO_4 as breeder material, beryllium as neutron multiplier and high-pressure helium as coolant gas [Her02], [Her03]. Breeder material and multiplier are arranged in pebble beds between flat cooling plates running in poloidal direction. The reduced activation ferritic steel Eurofer is used as structural material. The helium coolant is also employed for the divertor targets where refractory metals are anticipated as structural material. A helium exit temperature of 480 °C allows for the use of a superheated steam cycle in the power conversion system, leading to a higher efficiency than the one of model A. Slightly more advanced plasma physics models are employed to compensate for the lower load capabilities of the divertor targets compared to PPCS model A.

PPCS model C is based on a Dual Coolant Lithium-Lead (DCLL) blanket employing the liquid metal alloy Pb-17Li both as breeder and coolant in the breeding zone and helium gas for the cooling of the Eurofer structure including the first wall [Nor02],[Nor03]. Thin SiC_f/SiC flow channel inserts in the large coolant channels serve as thermal and electrical insulators in order to minimize magneto-hydrodynamic (MHD) problems and to obtain high coolant exit temperatures suitable for highly efficient power conversion systems. Helium cooled divertor targets are designed for high coolant exit temperatures in order to use this coolant to increase the efficiency of the BRAYTON cycle (closed cycle helium gas turbine) power conversion system. These technologies as well as the plasma physics models employed for PPCS model C are larger extrapolations from present knowledge as assumed for models A and B.

PPCS model D is based on a Self-Cooled Lead Lithium (SCLL) blanket with SiC-composite as structural material [Gia02]. The divertor targets are also cooled by lithium-lead, and the targets are fabricated with a combination of refractory metals with SiC-composites. This blanket design allows a coolant exit temperature up to 1100 °C leading to efficiencies > 55 % in the closed cycle helium turbine power conversion system. The surface heat fluxes at the divertor targets are reduced to values < 10 MW/m² by assuming very advanced plasma physics models. An other characteristic of the large extrapolation in technologies for PPCS model D is the use of high temperature super-conducting magnets.

The basic reactor parameters of the PPCS models have been evaluated by UKAEA Culham using the PROCESS system code [Hen96] assuming a net electric power of 1500 MW. The system code evaluations were based on preliminary assessments of the blanket and shield thicknesses, the thermal power production and the pumping power in accordance with the plasma physics assumptions for the near term and the advanced reactor models. Table 2-1 shows the main reactor parameters for the four models. Note that the requirement of a unit net electric power results in power plants of different sizes.

	Model A	Model B	Model C	Model D
Basic Parameters				
Major Radius (m)	9.8	8.7	7.5	6.1
Aspect Ratio	3.0	3.0	3.0	3.0
Plasma Current (MA)	33.5	28.1	19	14.1
Toroidal Field on axis (T)	7.3	6.9	6.0	5.6
TF on TF Coil Conductor (T)	12.9	13.1	13.6	13.4
Elongation (95% and separatrix)	1.7, 1.9	1.7, 1.9	1.9, 2.1	1.9, 2.1
Triangularity (95% and separatrix)	0.27,0.4	0.27, 0.4	0.47, 0.7	0.47, 0.7
Q	21	15	34	35
Engineering Parameters				
Fusion Power (GW)	5.5	3.4	3.45	2.5
P_{add} (MW)	265	234	100	71
Avge. Neutron Wall Load (MW/m^2)	2.3	1.8	2.25	2.4
Max. Divertor Heat Load (MW/m^2)	15	10	10	5
Net Reactor Efficiency	27%	43%	43%	61%

Tab. 2-1: Main parameters of the PPCS plant models for a net electric power of 1500 MW

3 Reactor modelling for neutronics calculations

The neutronic design analyses are based on neutron and photon transport calculations using the Monte Carlo approach. Suitable three-dimensional torus sector models were developed for the four different PPCS plant variants to enable proper design calculations with the MCN4C Monte Carlo code [Bri00]. The models include the plasma chamber, poloidally arranged blanket and shield modules, a bottom divertor port with integrated divertor, the vacuum vessel and the toroidal field coil. The torus sector models have been devised on the basis of the PPCS reactor parameters and neutron source distribution data as provided by UKAEA Culham Lou01], [Lou02], see Table 4-1 below. As far as available, technical design drawings have been used for the blanket modelling.

PPCS reactor model A assumes 18 toroidal sectors [Sar03] including poloidally arranged WCLL blanket modules. Accordingly, a MCNP torus sector model of 20° has been constructed including 3 inboard and 9 outboard blanket modules with their full toroidal extension. Vertical and horizontal cross-sections of the MCNP model are shown in Figs. 3-1 and 3-2, respectively. Table 3-1 shows the radial build assumed for the torus mid-plane at the inboard and outboard side of the reactor.

PPCS reactor model B is divided into 9 toroidal sectors of 40° each containing 1 x 4 inboard and 2 x 7 outboard HCPB blanket modules [Her03]. Assuming toroidal symmetry, a MCNP torus sector model of 20° has been constructed including 4 inboard modules with half their toroidal extension and 7 outboard modules with their full toroidal extension. Vertical and horizontal cross-sections of the MCNP model are shown in Figs. 3-3 and 3-4, respectively. Fig. 3-4 shows the modelling of the poloidal-radial pebble bed layers. Table 3-3 shows the radial build assumed for the torus mid-plane at the inboard and outboard side of the reactor.

The initial design of **PPCS reactor model C** was based on a segmentation of 18 toroidal sectors (20°) with large banana-type DCLL blanket sectors [Nor03]. This variant has been used as basis for the neutronics design analyses. It is noted that the blanket segmentation scheme for model C has been significantly changed in the course of the PPCS study by switching to 16 toroidal sectors (22.5°) with 11 poloidally arranged DCLL blanket modules each [Nor03]. Based on the initial PPCS model C design, a MCNP torus sector model of 10° has been constructed including one inboard and 1 ½ outboard WCLL blanket segments of the banana-type. Vertical and horizontal cross-sections of the MCNP model are shown in Fig. 3-6. Table 3-4 shows the radial build assumed for the torus mid-plane at the inboard and outboard side of the reactor.

PPCS reactor model D assumes a segmentation of 16 toroidal sectors with large SCLL blanket segments [Gia02]. Accordingly, a MCNP torus sector model of 11.25° has been constructed including, for symmetry reasons, two halves of an inboard segment (with a gap in between), and 1 ½ outboard segments. Vertical and horizontal cross-sections of the MCNP model are shown in Fig. 3-7. Table 3-5 shows the radial build assumed for the torus mid-plane at the inboard and outboard side of the reactor.

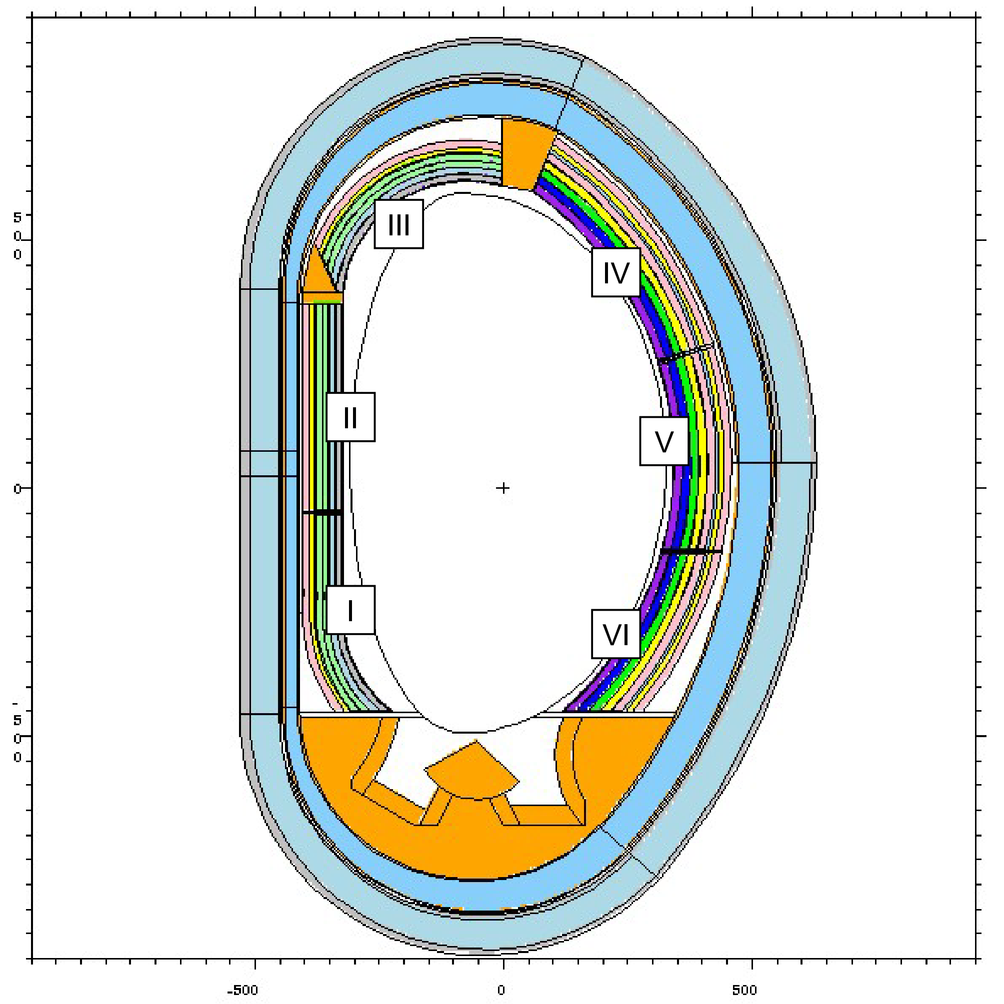


Fig. 3-1 PPCS model A (WCLL): vertical cut through MCNP torus sector model.

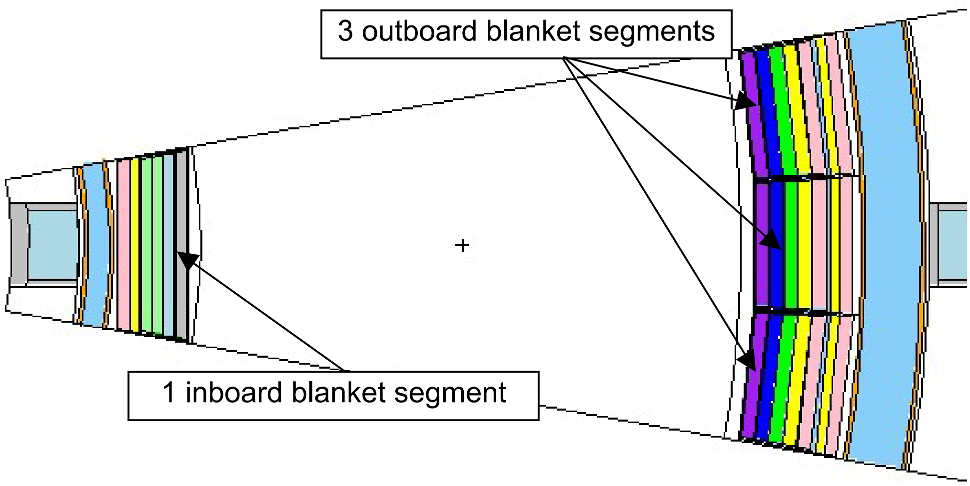


Fig. 3-2 PPCS model A (WCLL): horizontal cut through MCNP torus sector model (20°)

Inboard Thickness [cm]	cumulative	material	**Outboard** Thickness [cm]	cumulative	material	component
0.67	0.67	Eurofer (44.4%)	0.67	0.67	Eurofer (44.4%)	first wall
0.3	0.97	Eurofer (35%) H_2O (65%)	0.3	0.97	Eurofer(35%) H_2O (65%)	first wall
1.13	2.10	Eurofer	1.13	2.1	Eurofer	first wall
12.6	14.70	Pb-17Li (85.4%) H_2O (6.9%) Eurofer (7.7%)	15.94	18.04	Pb-17Li (83%) H_2O (8%) Eurofer (9%)	breeder/ coolant, rad. stiffener
0.8	15.50	Eurofer	0.8	18.84	Eurofer	Tor. stiffener
12.6	28.10	Pb-17Li (88.6%) H_2O (5.4%) Eurofer (6%)	15.94	37.78	Pb-17Li (93%) H_2O (3.3%) Eurofer (3.7%)	breeder/ coolant, rad. stiffener
0.8	28.90	Eurofer	0.8	35.58	Eurofer	Tor. stiffener
12.6	41.50	Pb-17Li (92.1%) H_2O (3.7%) Eurofer (4.2%)	15.94	51.52	Pb-17Li (93%) H_2O (3.3%) Eurofer (3.7%)	breeder/ coolant, rad. stiffener
0.8	42.30	Eurofer	0.8	52.32	Eurofer	Tor. stiffener
12.6	54.90	Pb-17Li (92.1%) H_2O (3.7%) Eurofer (4.2%)	15.94	68.26	Pb-17Li (92%) H_2O (3.8%) Eurofer (4.2%)	breeder/ coolant, tor. stiffener
			0.8	69.06	Eurofer	Toroidal stif-fener
			15.94	85.00	Pb-17Li (91%) H_2O (4.2%) Eurofer (4.8%)	breeder/ coolant, rad. stiffener
3.0	57.90	Eurofer (99.46%) H_2O (0.54%)	6.0	91.00	Eurofer (99.75%) H_2O (0.25%)	Back Plate
9.1	67.00	Eurofer (56%) H_2O (40%)	10.9	101.90	Eurofer (56%) H_2O (40%)	SB manifolds + shield
15.0	82.00	Eurofer (70%) H_2O (30%)	15.0	116.90	Eurofer (70%) H_2O (30%)	shield

Tab. 3-1: PPCS model A (WCLL): radial build of blanket and shield at torus mid-plane.

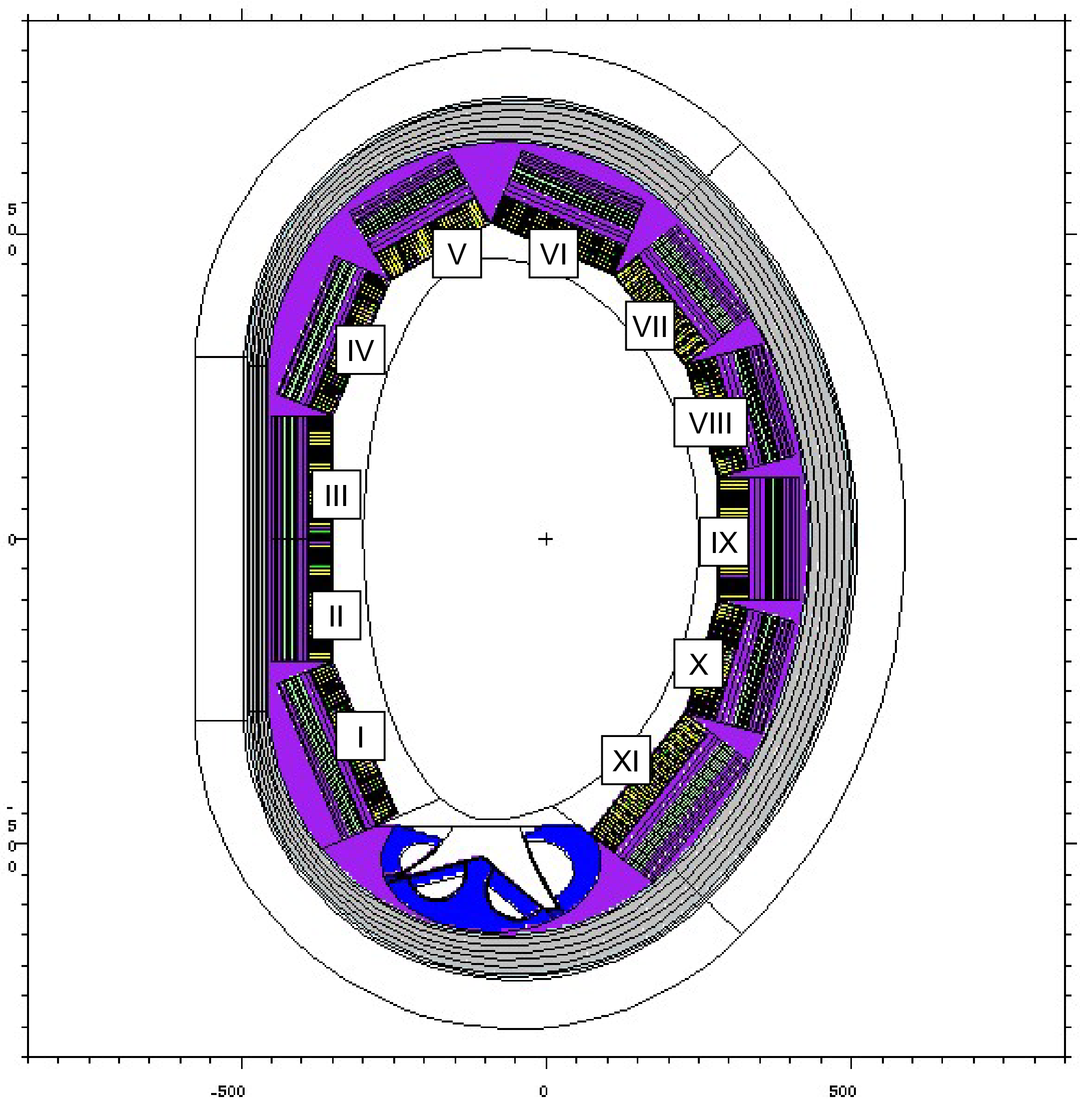
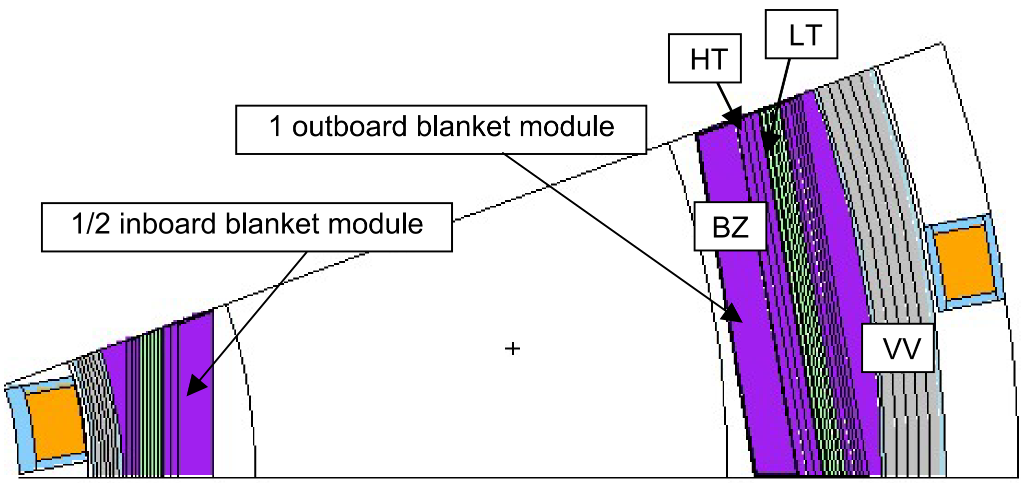


Fig. 3-3 PPCS model B (HCPB): vertical cut through MCNP torus sector model



HT = high temperature shield, LT = low temperature shield, VV = vacuum vessel, BZ = breeding zone

Fig. 3-4 PPCS model B (HCPB): vertical cut through MCNP torus sector model (20°)

Inboard		Outboard			
thickness [cm]	cumulative	thickness [cm]	cumulative	material (volume fractions)	component
0.4	0.4	0.4	0.4	1.0 Eurofer	first wall
1.4	1.8	1.4	1.8	0.27 Eurofer/0.73 He	first wall
0.5	2.3	0.5	2.3	1.0 Eurofer	first wall
36.5	38.8	46.5	48.8	0.154Breeder/0.692 Be/0.098Eurofer/0.055 He-coolant[a)]	blanket breeding zone
2	40.8	2	50.8	1.0 Eurofer	blanket back wall
17	57.8	27	77.8	0.6 Eurofer/0.4He/void	HT shield
2	59.8	2	79.8	void	gap
25	84.8	25	104.8	0.9 (0.6 Eurofer+0.4 ZrH)/0.1He	LT shield
15	99.8	25	129.8	0.15 Eurofer	manifolds
5	104.8	44	173.8	void	gap
5	109.8	5	178.8	SS-316	vacuum vessel
25	134.8	65	243.8	0.6 SS316/ 0.4 water	vacuum vessel
5	139.8	5	248.8	SS316	vacuum vessel
5	144.8	variable		void	gap
1.2	146	10		SS-316	inner TF-coil case
58.5	204.5	60		magnet mixture	TF coil
20	224.5	10		SS-316	outer TF-coil case

a) The material volume fractions shown in the table were obtained by homogenising over the breeder zone. Actually the breeder zone is a heterogeneous array consisting of a cooling plate (0.5 cm thickness; 64 vol% Eurofer), a Beryllium pebble bed (4.5 cm thickness), another cooling plate (0.5 cm thickness; 64 vol% Eurofer) , and a breeder pebble bed (1.0 cm), and is described in this way in the MCNP model, see Fig. 3.5 below. For both the Beryllium and the breeder ceramics, a mono-disperse pebble bed with 63 % volume packing fraction is assumed. The breeder is Li_4SiO_4 with 2.15w% (4.15at%) SiO_4 and 20 at% Li-6 enrichment in the reference case.

Tab. 3-2: PPCS model B (HCPB): radial build of blanket, shield, vacuum vessel and TF-coil (torus mid-plane).

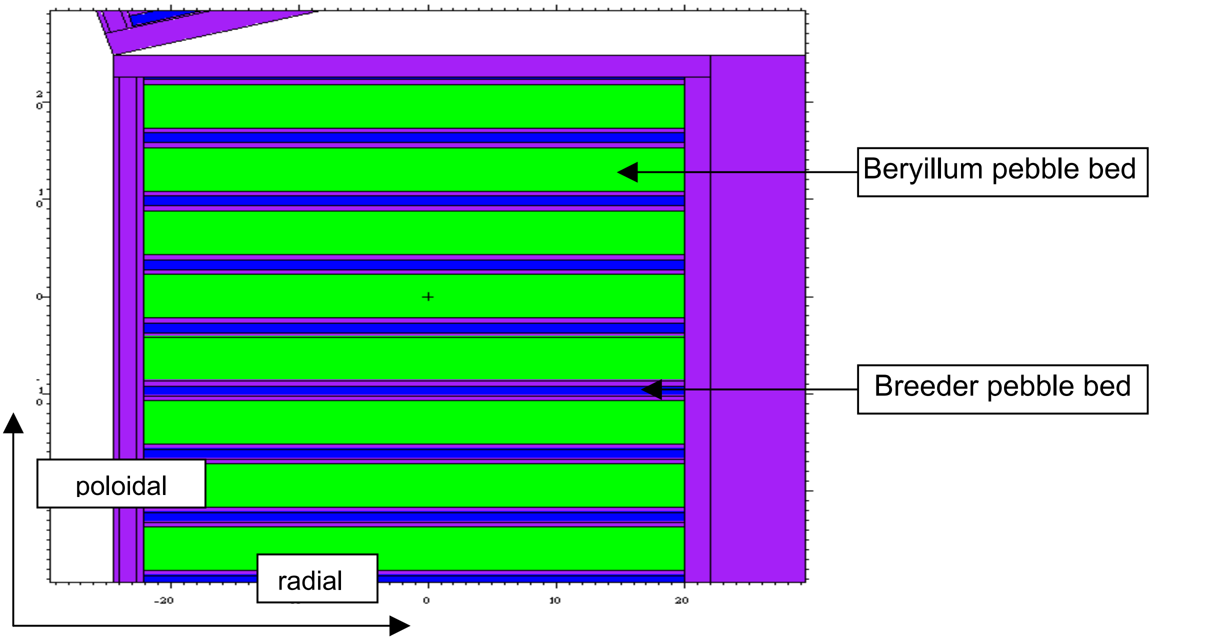


Fig. 3-5 PPCS model B (HCPB): Poloidal-radial arrangement of pebble beds (central outboard HCPB blanket module, cut through MCNP model)

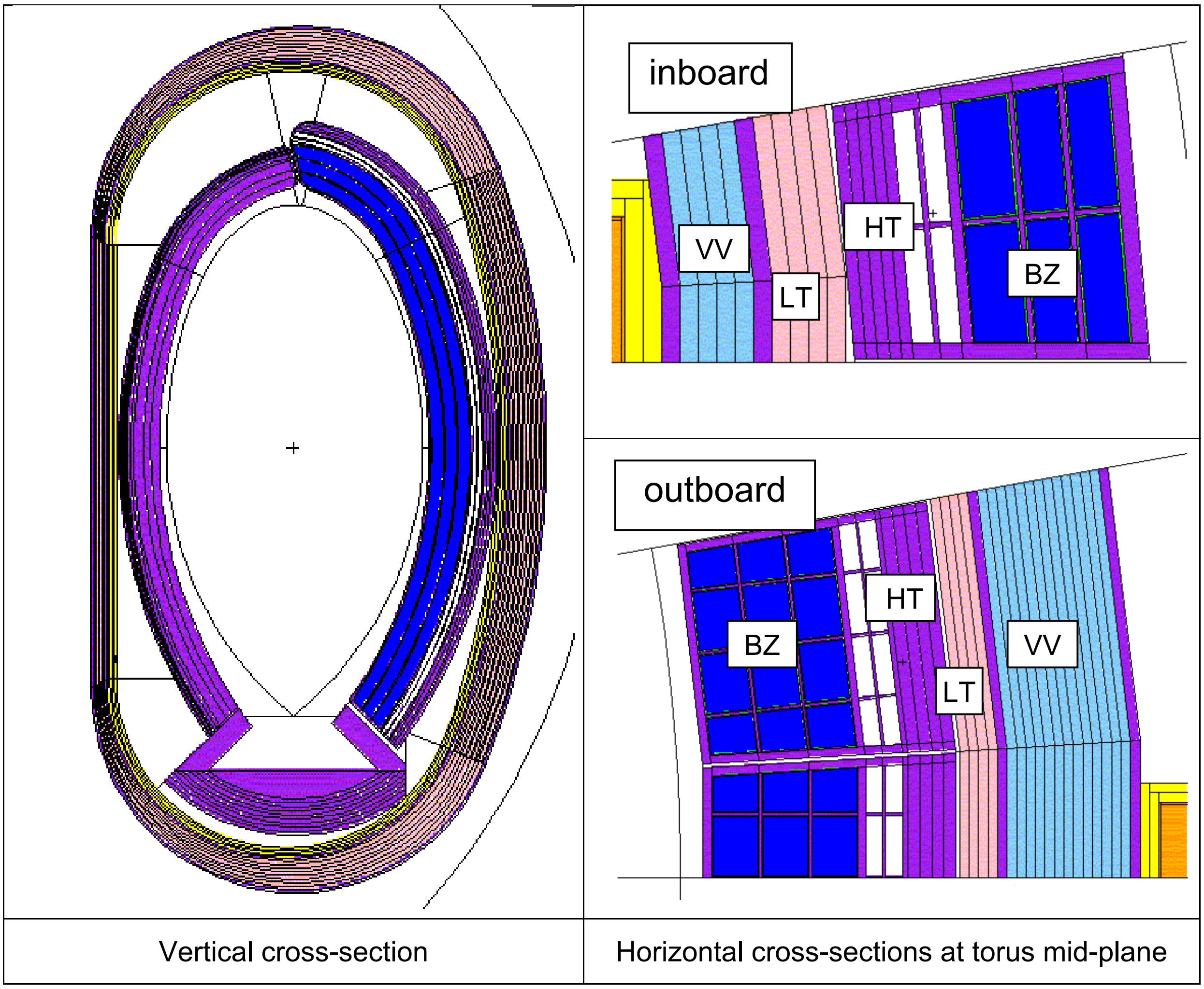
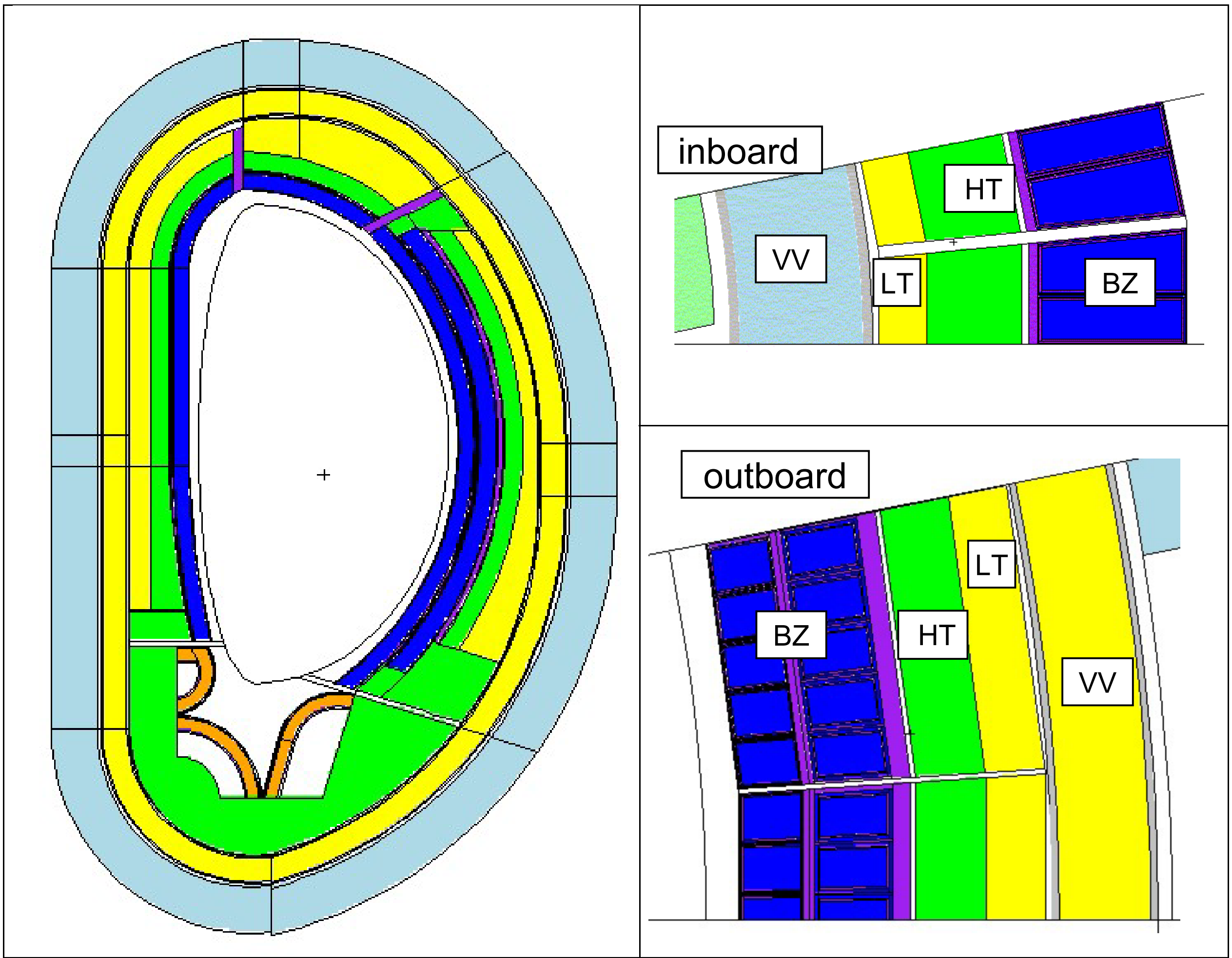


Fig. 3-6: PPCS model C (DCLL): MCNP torus sector model (10°)

inboard		outboard			
thickness [cm]		thickness [cm]		material	component
	cumulative		cumulative		
4.4	4.4	4.4	4.4	Eurofer (0.45)	first wall
0.5	4.9	0.5	4.9	SiC/SiC	FCI
11.5	16.4	24.1	29	Pb-17Li	breeder/coolant
0.5	16.9	0.5	29.5	SiC/SiC	FCI
1.5	18.4	1.5	31	Eurofer	structure
0.5	18.9	0.5	31.5	SiC/SiC	FCI
11.3	30.2	23.5	55	Pb-17Li	breeder/coolant
0.5	30.7	0.5	55.5	SiC/SiC	FCI
1.5	32.2	1.5	57	Eurofer	structure
0.5	32.7	0.5	57.5	SiC/SiC	FCI
14.3	47	23.5	81	Pb-17Li	breeder/coolant
0.5	47.5	0.5	81.5	SiC/SiC	FCI
3	50.5	4	85.5	Eurofer	structure
6	56.5	9	94.5	He	He in/out
1.5	58	1.5	96	Eurofer	structure
6	64	9	105	He	He in/out
3	67	3	108	Eurofer	structure
13	80	25	133	Eurofer	HT shield
30	110	30	163	0.6 Eurofer/0.4 water	LT shield
35	145	75	238	steel/borated water	vacuum vessel

Tab. 3-3: PPCS model C (DCLL): radial build of blanket, shield and vacuum vessel (torus mid-plane).



Vertical cross-section	Horizontal cross-sections at torus mid-plane

HT = high temperature shield, LT = low temperature shield, VV = vacuum vessel, BZ = breeding zone

Fig. 3-7: PPCS model D (SCLL): MCNP torus sector model (11.25°)

Inboard thickness [cm]		Outboard thickness [cm]			
	cumulative		cumulative	material	component
0.2	0.2	0.2	0.2	W	FW protection layer
0.5	0.7	0.5	0.7	SiC/SiC	FW
0.4	1.1	0.4	1.1	Pb-17Li	breeder/coolant
0.7	1.8	0.7	1.8	SiC/SiC	SW
24.3	26.1	24.3	26.1	Pb-17Li	breeder/coolant
0.7	26.8	0.7	26.8	SiC/SiC	back SW
0.4	27.2	0.4	27.2	Pb-17Li	breeder/coolant
3.0	30.2	3.0	30.2	SiC/SiC	back plate
-		2.0	32.2	SiC/SiC	FW 2nd box
-		1.5	33.7	Pb-17Li	breeder/coolant
-		0.7	34.4	SiC/SiC	SW 2nd box
-		28.6	63.0	Pb-17Li	breeder/coolant
-		0.7	63.7	SiC/SiC	back 2nd wall
-		1.5	65.2	Pb-17Li	breeder/coolant
-		5.0	70.2	C/SiC	2nd back plate
30	60.2	33	103.2	Pb-17Li (10%), C/SiC (10%)WC (80%)	HT (hot) shield
35	95.2	35	138.2	Borated steel (20%), WC (60%) He-coolant (20%)	LT (cold) shield
42	137.2	42	180.2	Borated steel (20%), WC (60%), He-coolant (20%)	Vacuum vessel

Tab. 3-4: PPCS model D (SCLL): radial build of blanket, shield and vacuum vessel (torus mid-plane).

4 Neutron source modelling

The neutron source distribution is described by a parametric representation using a subroutine linked to the MCNP code as provided by UKAEA Culham Lou01], [Lou02]. Source neutrons are sampled from a probability distribution of the 14 MeV neutron source density *s(a)* according to:

$$s(a) = \left[1 - \left(\frac{a}{A} \right)^2 \right]^4 \qquad 0 \le a \le A, \qquad A = \text{minor plasma radius}$$

The parameter a fixes a contour line at constant source density. It corresponds to a magnetic flux line that can be described in a parametric representation according to:

$$R = R_0 + a \cdot \cos(t + \delta \cdot \sin t) + e \cdot \left[1 - \left(\frac{a}{A} \right)^2 \right] \qquad z = E \cdot a \cdot \sin t$$

R gives the radial distance to the torus axis and *z* the poloidal distance to the torus midplane. R_0 denotes the major plasma radius. δ is the plasma triangularity and δ_0 it's maximum value. The parameters assumed for the four PPCS power plant models are shown in Table 4-1 Lou01],[Lou02]. The neutron source intensity shown in the table corresponds to the nominal fusion powers of 5500, 3300, 3410 and 2460 MW of PPCS models A, B, C and D, respectively. The sampling of the D-T source neutrons is performed in a Fortran subroutine linked to the MCNP-code.

	Model A	Model B	Model C	Model D
Mean Neutron Energy [MeV]	14.1	14.1	14.1	14.1
Spectrum Width [keV]	1285	1200	1067	924
Major radius [m]	10.6	8.6	7.5	6.1
Minor radius [m]	3.5	2.8	2.5	2.03
Plasma elongation	1.7	1.7	1.9	1.9
Maximum triangularity	0.27	0.27	0.47	0.47
Radial plasma shift [m]	0	0	0	0
Vertical plasma shift [m]	0	0	0	0
Source intensity [n/sec]	1.88×10^{21}	1.17×10^{21}	1.21×10^{21}	8.72×10^{20}
Source Peaking Factor	1.7	1.7	2.5	2.5

Tab. 4-1: Parameters for describing the neutron source distribution and the normalisation.

5 Neutronic analyses, results and discussion

The neutronics analyses are based on Monte Carlo calculations with the MCNP4C code running on a HPC Linux cluster machine in the parallel mode under the Parallel Virtual Machine (PVM).

5.1 Neutron wall loading distribution

The neutron wall loading distributions were calculated with MCNP for the voided torus sector models by scoring the number of (virgin) 14 MeV neutrons crossing the first wall and normalising the related current densities to the fusion powers of the four PPCS power plant models. The spatial distribution of D-T source neutrons as described above in section 4 was assumed in these calculations. Table 5-1 shows the resulting neutron wall loading data along with the associated surface areas. About 90 % of the fusion neutron power released in the plasma chamber is loaded to the first wall of the blanket modules/segments. The remainder flows through the divertor opening.

	Model A	Model B	Model C	Model D
First wall surface area [m^2][a)]	1720	1365	1210	746
Fusion neutron power [MW]				
Released in plasma chamber	4400	2640	2728	1968
Loaded to the first wall	3810	2320	2557	1818
Neutron wall loading [MW/m²]				
Inboard peak value	2.69	1.99	2.69	2.94
Outboard peak value	3.05	2.41	3.10	3.44
Average value	2.56	1.94	2.23	2.59

[a)] Includes areas covering blanket modules/segments and divertor opening

Tab. 5-1: Neutron wall loading and first wall surface areas of the blanket modules/segments of the PPCS models.

For the PPCS model B, a detailed distribution of the neutron wall loading of the single blanket modules is given in Table 5-2 It is recalled that inboard and outboard blanket modules have different toroidal extensions (40 and 20 °, respectively) for model B. In total, there are 36 inboard and 126 outboard HCPB blanket modules. Only 25% of the fusion neutron power released to the first wall of the blanket modules is flowing to the inboard modules. Figs. 5-1 – 5-4 show plots of the poloidal neutron wall loading distribution for all the PPCS models A, B, C and D.

Module #	First wall area [cm^2] Module [a]	Reactor [b]	Neutron wall loading [MW/m^2]	Fusion neutron power [MW] Modules [a]	Reactor [b]
1	1.19E+05	1.07E+06	1.48	1.76E+01	1.59E+02
2	7.84E+04	7.06E+05	1.78	1.39E+01	1.26E+02
3	7.84E+04	7.06E+05	1.78	1.39E+01	1.26E+02
4	1.11E+05	1.00E+06	1.50	1.67E+01	1.50E+02
5	4.03E+04	7.26E+05	1.44	5.81E+00	1.05E+02
6	6.71E+04	1.21E+06	1.71	1.15E+01	2.06E+02
7	6.96E+04	1.25E+06	1.92	1.33E+01	2.40E+02
8	7.64E+04	1.38E+06	2.12	1.62E+01	2.91E+02
9	8.15E+04	1.47E+06	2.24	1.83E+01	3.29E+02
10	7.67E+04	1.38E+06	2.11	1.62E+01	2.92E+02
11	9.10E+04	1.64E+06	1.82	1.66E+01	2.99E+02
total		1.25E+07	1.85		2.32E+03

[a] *Data refer to single modules. Inboard modules 1-4 and outboard modules 5-11 have toroidal extensions of 40 ° and 20°, respectively.*

[b] *Data refer to complete reactor with 4 x 9 = 36 inboard and 7 x 18 = 126 outboard modules.*

Tab. 5-2: PPCS model B (HCPB): First wall areas and neutron wall loading for single blanket modules and complete reactor.

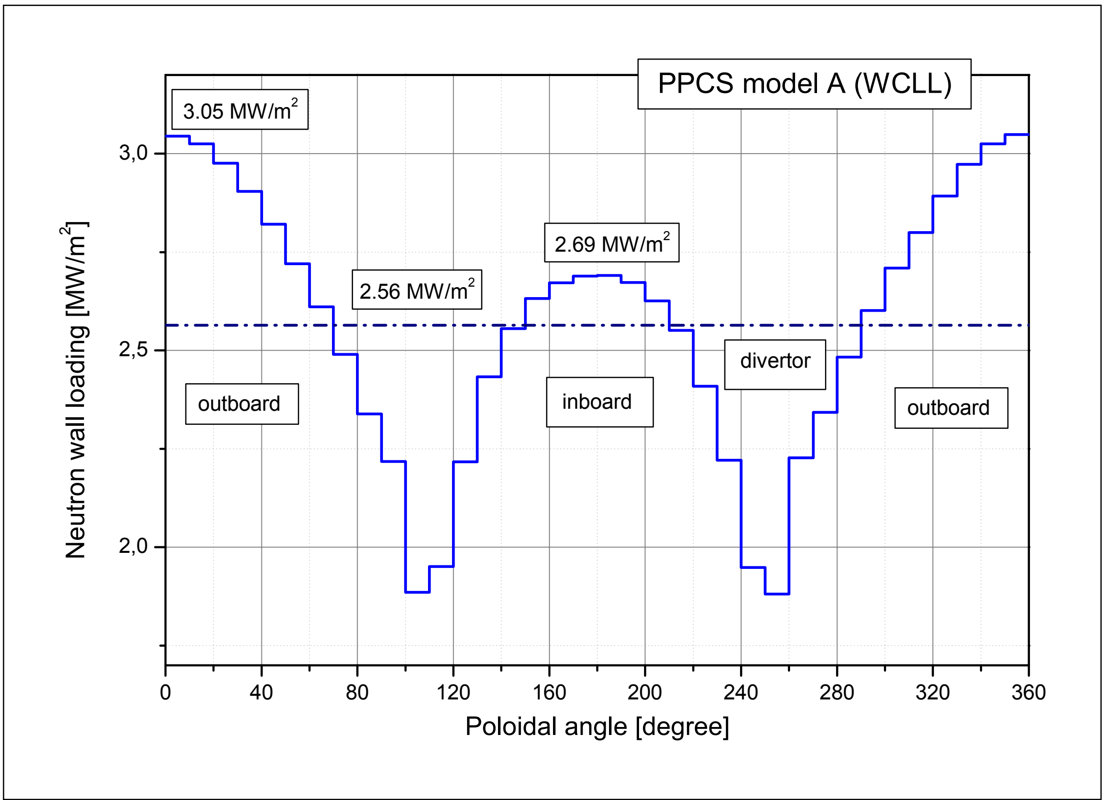


Fig. 5-1: PPCS model A (WCLL): Poloidal neutron wall loading distribution

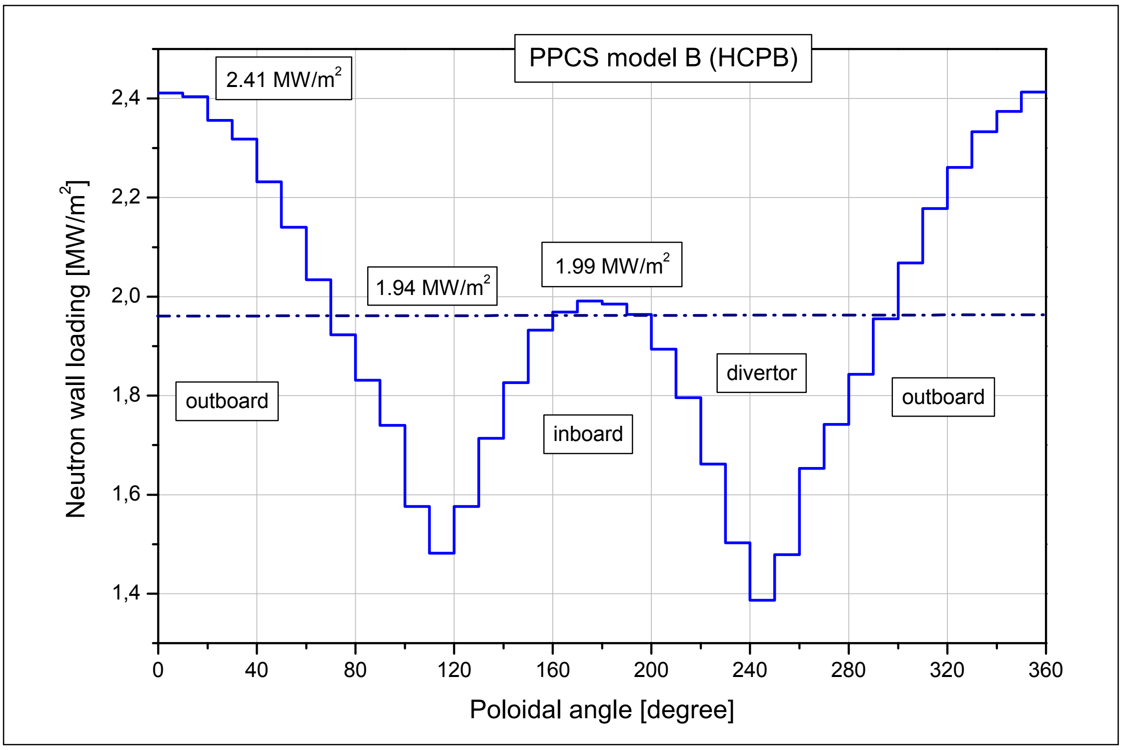


Fig. 5-2: PPCS model B (HCPB): Poloidal neutron wall loading distribution.

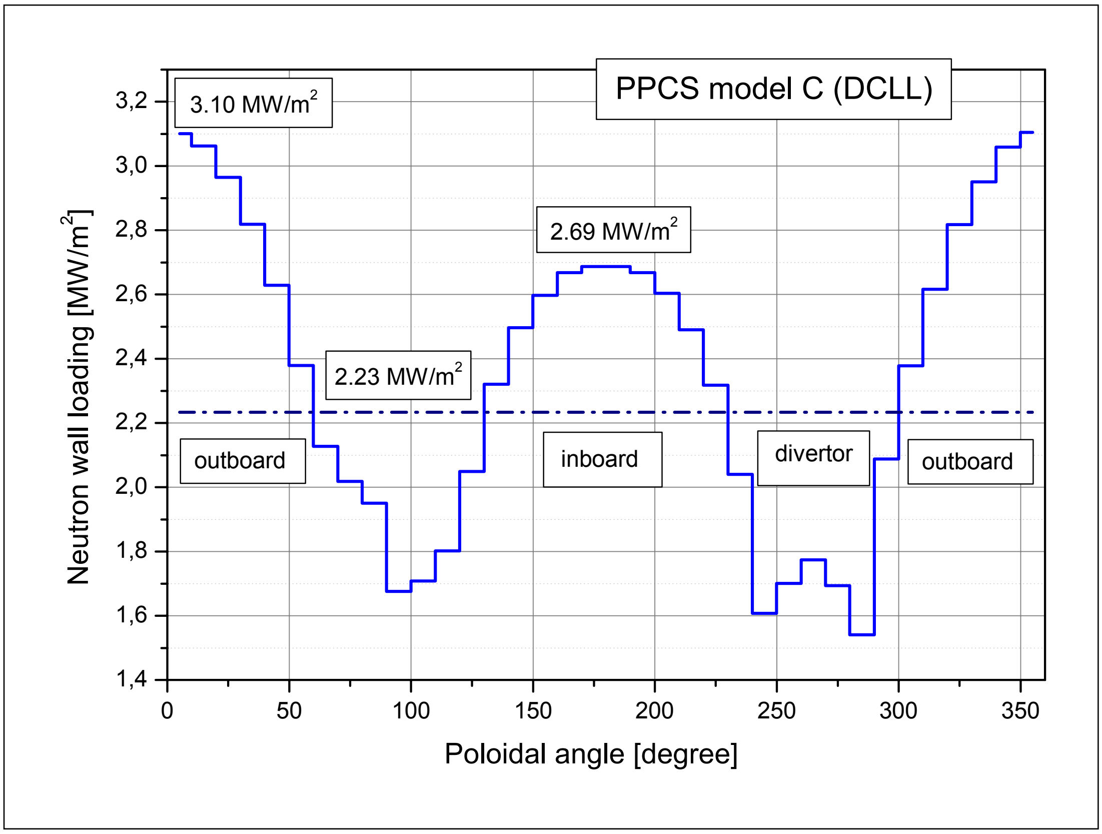


Fig. 5-3: PPCS model C (DCLL): Poloidal neutron wall loading distribution.

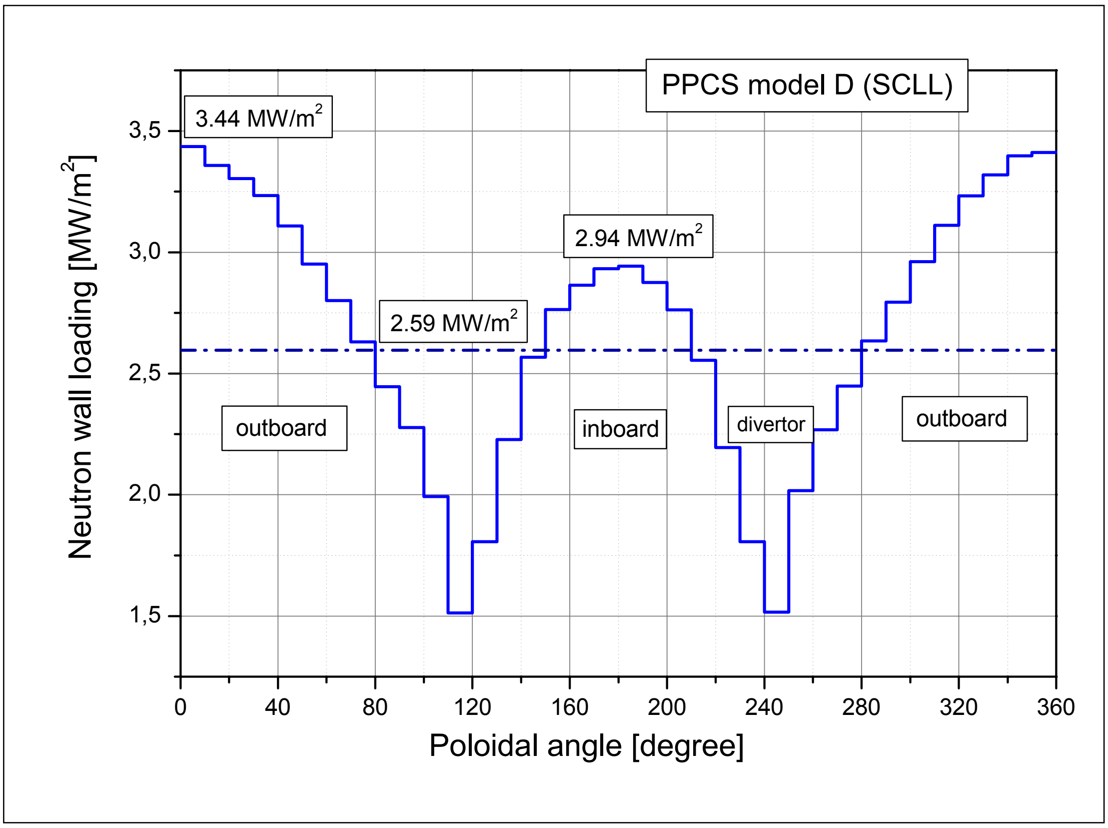


Fig. 5-4: PPCS model D (SCLL): Poloidal neutron wall loading distribution

5.2 Tritium breeding capability

Table 5-4. shows an overview of the Tritium Breeding Ration (TBR) as calculated for the four PPCS models assuming the reference blanket designs.

	Model A (WCLL)	Model B (HCPB)	Model C (DCLL)	Model D (SCLL)
Breeder material	Pb-17Li	Li_4SiO_4	Pb-17Li	Pb-17Li
Neutron multiplier	Pb-17Li	Be	Pb-17Li	Pb-17Li
Li-6 enrichment [at%]	90	30	90	90
Blanket thickness				
Inboard mid-plane	57.9	40.8	50.5	30.2
Outboard mid-plane	91.0	50.8	85.5	70.2
Neutron multiplication factor	1.55	1.78	1.57	1.58
Tritium Breeding Ratio (TBR)	1.06	1.12	1.15	1.12

Tab. 5-3: Tritium breeding, neutron multiplication and related characteristic features of the four PPCS models.

The WCLL blanket (model A), in principle, provides a good tritium breeding potential by using Pb-17Li at a ^{6}Li enrichment of 90 at% as breeder material in the large-sized liquid metal channels. With a total radial blanket thickness of 57.9 and 91.0 cm, inboard and outboard, respectively (shielding/ manifolds not included), a global TBR of 1.06 is achieved for the PPCS model A. This is sufficient to compensate for potential tritium losses and account for uncertainties of the TBR-calculation. It is, however, not sufficient to account for a reduction of the TBR due to the presence of blanket ports equipped with heating systems and diagnostics. A global TBR ≥ 1.10 is required to compensate for the port effect. The TBR of model A could be increased, however, by reducing the size of the segment gaps between the top inboard and outboard modules (see Fig. 3-1). These large segment gaps were introduced to allow an independent removal of the top inboard and outboard modules [Sar03]. Actually, this may not be required to ensure a sufficiently high plant availability.

The HCPB blanket (model B) provides a high tritium breeding potential through the use of the efficient Beryllium neutron multiplier. Optimisation is required for the geometrical configuration of the pebble bed layers and the ^{6}Li enrichment of the breeder. The reference design for the PPCS reactor assumes a pebble bed height of 45 and 10 mm for Beryllium and the Li_4SiO_4 breeder ceramics, respectively, with 5 mm thick cooling plates in between. A mono-disperse pebble bed is assumed both for the breeder ceramics and the Beryllium. With a total radial blanket thickness of 40.8 and 50.8 cm, inboard and outboard, respectively (shielding/manifolds not included), the global TBR amounts to 1.07 and 1.12 at a ^{6}Li enrichment of 20 and 30 at%, respectively. A ^{6}Li enrichment of 30 at% is thus required to achieve the target TBR ≥ 1.10. Tritium is also produced in the Beryllium multiplier where it accumulates to a large extent during irradiation. At a total beryllium inventory of 390 tons, the tritium production in Beryllium sums up to 23.8 kg after an irradiation time of 40,000 hours (Table 5-3).

		Single modules		Total reactor	
Module #	Neutron wall loading [MW/m^2]	Beryllium inventory [grams]	Tritium inventory [grams]	Beryllium inventory [grams]	Tritium inventory [grams]
1	1.48	1.70E+06	9.75E+01	1.53E+07	8.78E+02
2	1.78	1.09E+06	7.53E+01	9.85E+06	6.77E+02
3	1.78	1.09E+06	7.53E+01	9.85E+06	6.77E+02
4	1.50	1.56E+06	9.06E+01	1.41E+07	8.15E+02
5	1.44	1.45E+06	6.53E+01	2.61E+07	1.17E+03
6	1.71	2.49E+06	1.33E+02	4.48E+07	2.39E+03
7	1.92	2.61E+06	1.56E+02	4.70E+07	2.81E+03
8	2.12	2.90E+06	1.92E+02	5.23E+07	3.46E+03
9	2.25	3.10E+06	2.18E+02	5.59E+07	3.92E+03
10	2.11	2.91E+06	1.92E+02	5.24E+07	3.46E+03
11	1.83	3.45E+06	1.97E+02	6.21E+07	3.54E+03
Total				3.90E+08	2.38E+04

Tab. 5-4: PPCS model B (HCPB blanket): Tritium generation in Beryllium assuming an 40000h full power operation.

Both **advanced blanket concepts (models C and D)** provide a good tritium breeding potential due the large-sized liquid metal channels with Pb-17Li as breeder material and a ^{6}Li enrichment of 90 at%. The DCLL blanket requires a total radial blanket thickness of 50.5 and 85.5 cm, inboard and outboard, respectively (shielding/ manifolds not included), to achieve a global TBR of 1.15. For the SCLL blanket with SiC_f/SiC structure the corresponding total radial blanket thickness amounts to 30.2 and 70.2 cm, inboard and outboard, respectively. The resulting global TBR is 1.12 when using WC as shielding material in the HT shield (see Fig. 3-7) and including a breeder zone behind the divertor. A divertor breeder zone could be avoided by increasing the breeder zone thickness for compensation. Note that the TBR is very sensitive to the material composition of the HT shield acting as neutron reflector to the breeder zone. The use of the efficient WC shield material is also favourable for the tritium breeding performance. Due to the strong inelastic scattering reactions on tungsten, the low energy neutron flux component in both the HT shield and the back of the breeder zone is increased. This results in enhanced neutron absorption reactions in Li-6 leading to the generation of tritium.

	WC in HT shield	B_4C in HT shield
Inboard blanket	0.202	0.177
Topboard blanket	0.089	0.078
Outboard blanket	0.690	0.668
Sub-total (blanket)	0.981	0.923
Divertor breeder zone	0.125	0.108
HT shield (Pb-17 Li cooled)	0.014	0.004
Grand total	1.120	1.035

Tab. 5-5: PPCS model D (SCLL blanket): Tritium breeding ratio for two shield variants.

5.3 Nuclear power generation

Table 5-6. shows an overview of the nuclear power generation as calculated for the four PPCS models assuming a unit net electrical power of 1500 MW. In either case, a major fraction of ≅ 80% of the nuclear power is generated in the blanket modules/segments including the first wall. The remainder is produced in the shields, the manifolds, the divertor and the vacuum vessel.

The global energy multiplication, defined by the ratio of the total nuclear power generated in all reactor components and the underlying fusion neutron power, is comparatively high for the HCPB reactor. This is mainly due to the high neutron multiplication because of the high Beryllium mass inventory as noted above. The low energy multiplication factor of the SCLL reactor, on the other hand, is due to the use of SiC as structural material. (Parasitic neutron absorption reactions in iron, the main steel constituent, show a comparatively high energy release).

Detailed data on the nuclear power generation and the spatial distribution are given for the four PPCS models in the following sub-sections 5.3.1- 5.3.4.

	Model A (WCLL)	Model B (HCPB)	Model C (DCLL)	Model D (SCLL)
Blanket & first wall	4258	3002	2785	1748
Manifolds & shielding	251	293	194.3	182.3
Vacuum vessel	1.4	18	7.6	0.3
Divertor	884	323	346	271
Total	5394	3636	3333	2202
Energy multiplication factor	1.23	1.38	1.22	1.13

Tab. 5-6: Overview of nuclear power generation as calculated for the four PPCS plant models assuming a unit net electrical power of 1500 MW.

5.3.1 PPCS model A (WCLL)

Table 5-7 shows an overview of the nuclear power generation in the single poloidal modules divided into first wall (FW), blanket, manifolds and shielding. It is recalled that the MCNP model assumes a toroidal segmentation of 20° for the inboard modules (I-III) and of 6.75, 6.5 and 6.75° for the outboard modules (IV-VI) as shown in Figs. 3-1 and 3-2 above. Results are given for the 20° sector (FW, blanket, manifolds, shielding, blanket modules) and the complete WCLL reactor. The comparatively high power generation in the WCLL divertor is due to the use of a steel/water mixture which efficiently absorbs neutrons streaming into the divertor opening. It is noted that supplementary steel plugs have been employed between the blanket modules to avoid neutron streaming. These shield plugs have not been taken into account in the technical design of PPCS model A based on the WCLL blanket concept. As seen in Table 5-7, there is a considerable power generation in the plugs.

Module	FW	Blanket	Manifolds	Shielding	Modules	Reactor
I	2.68	$2.04\cdot10^{1}$	$4.42\cdot10^{-1}$	$4.95\cdot10^{-1}$	$2.40\cdot10^{1}$	$4.32\cdot10^{2}$
II	2.77	$2.15\cdot10^{1}$	$4.78\cdot10^{-1}$	$5.32\cdot10^{-1}$	$2.53\cdot10^{1}$	$4.55\cdot10^{2}$
III	3.15	$2.57\cdot10^{1}$	$5.50\cdot10^{-1}$	$5.78\cdot10^{-1}$	$2.99\cdot10^{1}$	$5.38\cdot10^{2}$
IV	5.31	$4.91\cdot10^{1}$	$1.57\cdot10^{-1}$	$1.41\cdot10^{-1}$	$5.47\cdot10^{1}$	$9.85\cdot10^{2}$
V	5.58	$5.10\cdot10^{1}$	$1.75\cdot10^{-1}$	$1.62\cdot10^{-1}$	$5.69\cdot10^{1}$	$1.02\cdot10^{3}$
VI	4.85	$4.46\cdot10^{1}$	$9.85\cdot10^{-2}$	$1.42\cdot10^{-1}$	$4.97\cdot10^{1}$	$8.95\cdot10^{2}$
Vacuum vessel						$1.41\cdot10^{0}$
Divertor						$8.84\cdot10^{2}$
Suppl. shielding(*)						$1.80\cdot10^{2}$
Total						$5.39\cdot10^{3}$

(*) Supplementary shielding plugs in gaps between modules II/III, III/IV and behind the divertor.

Tab. 5-7: PPCS model A: Nuclear power production [MW] in WCLL blanket modules (20°), vacuum vessel, shield plugs and divertor.

Table 5-8 compares the nuclear power production of the single blanket modules with the fusion neutron power flowing into the modules. The multiplication factor given there is defined by the ratio of the nuclear power generated in the module and the fusion power loaded to the first wall of the module. This ratio does not take into account the fusion neutron power flowing into the blanket modules through the lateral walls. Note that the power multiplication factor averaged over the blanket modules is significant lower than the global energy multiplication factor (Tab. 5-6). This is mainly due to the significant power production in the divertor region. The partition of the power generation among the materials is shown in Table 5-9.

Module	First wall area [cm^2]	Neutron wall loading [MW/m^2]	Fusion neutron power [MW]	Nuclear power [MW]	Multiplication factor(*)
I	$1.03 \cdot 10^{05}$	2.06	$2.12 \cdot 10^{01}$	$2.40 \cdot 10^{01}$	1.13
II	$9.69 \cdot 10^{04}$	2.39	$2.32 \cdot 10^{01}$	$2.53 \cdot 10^{01}$	1.09
III	$1.22 \cdot 10^{05}$	2.02	$2.46 \cdot 10^{01}$	$2.99 \cdot 10^{01}$	1.21
IV	$1.78 \cdot 10^{05}$	2.63	$4.68 \cdot 10^{01}$	$5.47 \cdot 10^{01}$	1.17
V	$1.75 \cdot 10^{05}$	2.96	$5.18 \cdot 10^{01}$	$5.69 \cdot 10^{01}$	1.10
VI	$1.63 \cdot 10^{05}$	2.69	$4.38 \cdot 10^{01}$	$4.97 \cdot 10^{01}$	1.13
Average		2.52			1.14

(*) *Ratio of nuclear power generated in the module(s) and the fusion neutron power loaded to the first wall of the module(s).*

Tab. 5-8: PPCS model A: Nuclear power generation and multiplication factor for WCLL blanket modules (20° sector).

	Eurofer	Pb-17Li	H_2O		Eurofer	Pb-17Li	H_2O
Module I					Module IV		
First wall	$2.41\cdot10^{0}$	-	$2.70\cdot10^{-1}$		$4.79\cdot10^{0}$	-	$5.20\cdot10^{-1}$
Blanket	$1.20\cdot10^{0}$	$1.84\cdot10^{1}$	$8.11\cdot10^{-1}$		$3.72\cdot10^{0}$	$43.5\cdot10^{1}$	$1.92\cdot10^{0}$
Manifold/shielding	$7.63\cdot10^{-1}$	-	$1.74\cdot10^{-1}$		$2.49\cdot10^{-1}$	-	$4.59\cdot10^{-2}$
Total	$4.37\cdot10^{0}$	$1.84\cdot10^{1}$	$1.35\cdot10^{0}$		$7.28\cdot10^{0}$	$4.49\cdot10^{1}$	$2.54\cdot10^{0}$
Module II					Module V		
First wall	$2.50\cdot10^{0}$	-	$2.70\cdot10^{-1}$		$5.04\cdot10^{0}$	-	$5.43\cdot10^{-1}$
Blanket	$1.32\cdot10^{0}$	$1.93\cdot10^{1}$	$8.68\cdot10^{-1}$		$3.92\cdot10^{0}$	$4.53\cdot10^{1}$	$1.82\cdot10^{0}$
Manifold/shielding	$8.11\cdot10^{-1}$	-	$1.99\cdot10^{-1}$		$2.82\cdot10^{-1}$	-	$5.35\cdot10^{-2}$
Total	$4.63\cdot10^{0}$	$1.93\cdot10^{1}$	$1.34\cdot10^{0}$		$7.51\cdot10^{0}$	$4.70\cdot10^{1}$	$2.46\cdot10^{0}$
Module III					Module VI		
First wall	$2.84\cdot10^{0}$	-	$3.14\cdot10^{-1}$		$4.43\cdot10^{0}$	-	$4.20\cdot10^{-1}$
Blanket	$1.34\cdot10^{0}$	$2.33\cdot10^{1}$	$1.01\cdot10^{0}$		$3.58\cdot10^{0}$	$3.94\cdot10^{1}$	$1.66\cdot10^{0}$
Manifold/shielding	$9.03\cdot10^{-1}$	-	$2.20\cdot10^{-1}$		$1.95\cdot10^{-1}$	-	$4.55\cdot10^{-2}$
Total	$5.08\cdot10^{0}$	$2.33\cdot10^{1}$	$1.54\cdot10^{0}$		$7.09\cdot10^{0}$	$40.5\cdot10^{1}$	$2.15\cdot10^{0}$

Tab. 5-9: PPCS model A: Nuclear power generation [MW] in the different materials of the WCLL blanket modules (20° sector).

Power density profiles are shown in Fig. 5-5 for the Pb-17Li breeder, the water coolant and the Eurofer structure all of the poloidal WCLL blanket modules. The corresponding numerical data are given in Table 5-10 and 5-11. Associated volumes and masses as calculated for the MCNP torus sector model are given in Table 5-12.

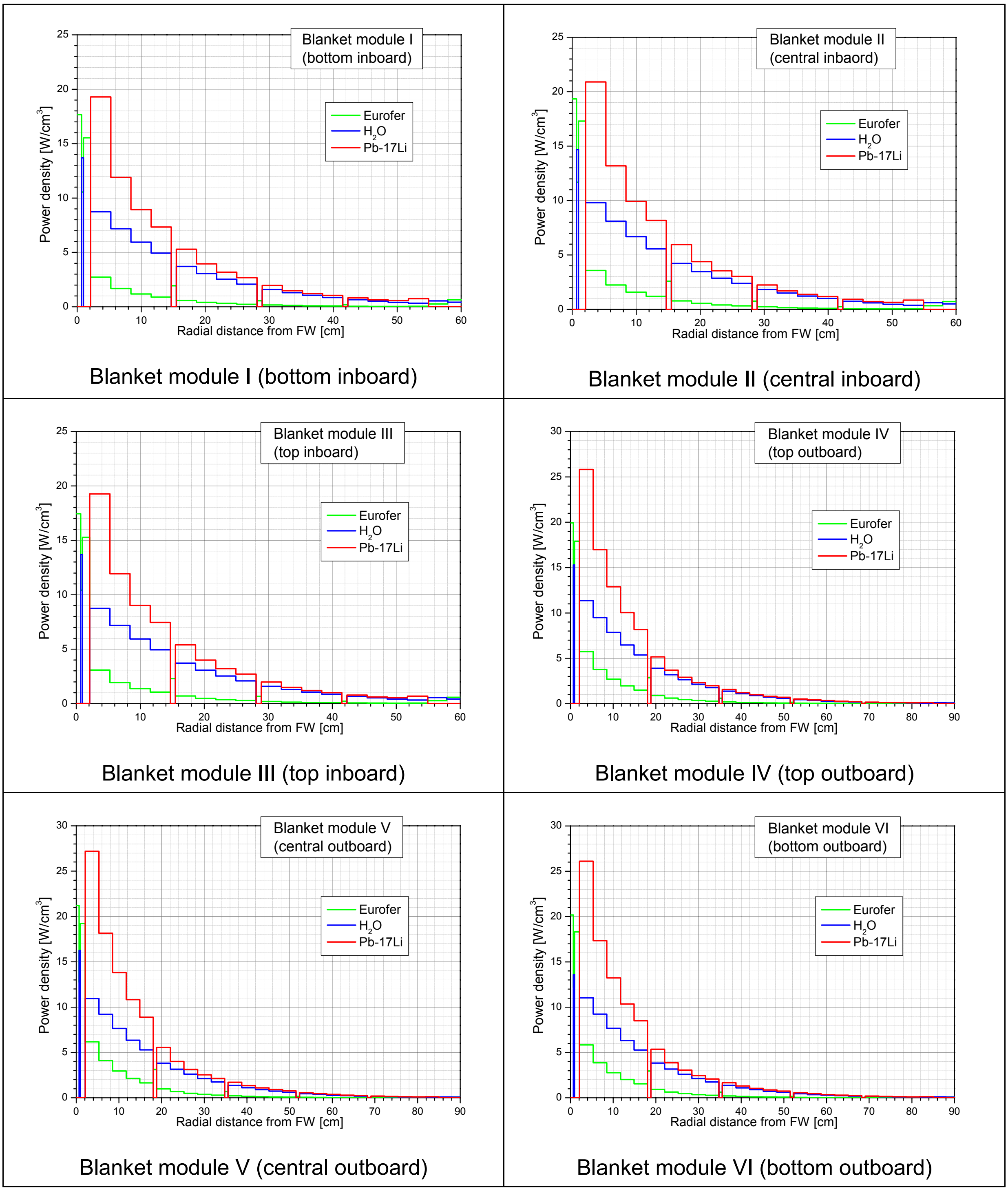


Fig. 5-5: PPCS model A: Radial power density profiles for the different materials in the WCLL blanket modules I-VI

	Inboard module I			Inboard module II			Inboard module III		
R [cm]	Eurofer	H_2O	Pb-17Li	Eurofer	H_2O	Pb-17Li	Eurofer	H_2O	Pb-17Li
0.67	17.66			19.34			17.44		
0.97	10.55	13.71		11.68	14.69		10.43	13.37	
2.10	15.54			17.30			15.28		
5.25	2.73	8.74	19.28	4.05	9.82	20.89	2.88	8.59	19.27
8.40	1.68	7.19	11.90	2.54	8.11	13.19	1.80	7.12	11.93
11.55	1.19	5.94	8.93	1.79	6.69	9.92	1.29	5.90	9.02
14.70	0.90	4.93	7.33	1.36	5.56	8.18	0.99	4.90	7.46
15.50	1.93			2.93			2.14		
18.65	0.58	3.72	5.30	0.89	4.23	5.95	0.65	3.70	5.40
21.80	0.41	3.07	3.95	0.62	3.46	4.39	0.45	3.03	3.99
24.95	0.31	2.53	3.18	0.47	2.87	3.56	0.34	2.50	3.21
28.10	0.24	2.08	2.68	0.37	2.38	3.03	0.27	2.06	2.71
28.90	0.57			0.86			0.63		
32.05	0.17	1.58	1.96	0.26	1.83	2.24	0.19	1.56	1.98
35.20	0.12	1.29	1.49	0.19	1.50	1.70	0.13	1.27	1.48
38.35	0.09	1.06	1.22	0.14	1.23	1.39	0.10	1.03	1.20
41.50	0.08	0.86	1.04	0.12	1.00	1.17	0.08	0.84	1.01
42.30	0.19			0.28			0.20		
45.45	0.06	0.65	0.82	0.08	0.75	0.92	0.06	0.62	0.78
48.16	0.04	0.52	0.64	0.06	0.60	0.73	0.04	0.49	0.61
51.75	0.03	0.41	0.57	0.05	0.47	0.65	0.04	0.39	0.53
54.90	0.05	0.32	0.75	0.07	0.37	0.85	0.05	0.30	0.69
57.90	0.26	0.54		0.38	0.62		0.25	0.50	
60.90	0.64	0.43		0.72	0.51		0.57	0.41	
63.90	0.72	0.34		0.81	0.40		0.65	0.32	
67.00	0.71	0.27		0.79	0.32		0.63	0.25	
70.00	0.70	0.24		0.79	0.29		0.61	0.22	
73.00	0.60	0.19		0.65	0.22		0.50	0.17	
76.00	0.50	0.16		0.53	0.18		0.40	0.13	
78.00	0.40	0.13		0.42	0.14		0.30	0.10	
82.00	0.32	0.11		0.31	0.11		0.21	0.07	

Tab. 5-10: PPCS model A: Radial power density distributions [W/cm^3] of the materials in the blanket modules I-III.

	Outboard module IV			Outboard module V			Outboard module VI		
R [cm]	Eurofer	H_2O	Pb-17Li	Eurofer	H_2O	Pb-17Li	Eurofer	H_2O	Pb-17Li
0.67	19.93			21.22			20.18		
0.97	11.98	15.29		12.75	16.24		12.19	13.60	
2.10	17.91			19.23			18.29		
5.30	5.73	11.36	25.82	6.19	10.96	27.19	5.83	11.03	26.11
8.50	3.77	9.49	16.98	4.12	9.21	18.15	3.87	9.23	17.34
11.70	2.69	7.84	12.88	2.95	7.65	13.82	2.77	7.66	13.24
14.90	1.96	6.46	10.04	2.16	6.35	10.84	2.02	6.33	10.35
18.04	1.49	5.37	8.18	1.65	5.28	8.89	1.55	5.28	8.50
18.84	2.85			3.14			2.96		
22.04	0.89	3.89	5.16	0.98	3.81	5.56	0.93	3.83	5.36
25.24	0.61	3.19	3.70	0.68	3.16	4.01	0.64	3.17	3.86
28.44	0.45	2.63	2.90	0.50	2.60	3.14	0.48	2.60	3.05
31.64	0.34	2.15	2.31	0.38	2.13	2.54	0.36	2.14	2.45
34.78	0.27	1.77	1.97	0.30	1.75	2.16	0.28	1.76	2.08
35.58	0.60			0.68			0.63		
38.78	0.19	1.37	1.57	0.22	1.37	1.72	0.20	1.37	1.66
41.98	0.14	1.10	1.21	0.16	1.11	1.34	0.15	1.11	1.30
45.18	0.10	0.89	0.99	0.12	0.90	1.09	0.11	0.90	1.05
48.38	0.08	0.72	0.81	0.09	0.73	0.90	0.09	0.73	0.87
51.52	0.06	0.59	0.68	0.07	0.59	0.76	0.07	0.60	0.75
52.32	0.14			0.16			0.16		
55.52	0.05	0.44	0.51	0.05	0.44	0.57	0.05	0.45	0.57
58.72	0.03	0.35	0.40	0.04	0.36	0.45	0.04	0.36	0.44
61.92	0.03	0.28	0.32	0.03	0.29	0.37	0.03	0.29	0.36
65.12	0.02	0.23	0.26	0.02	0.23	0.30	0.02	0.23	0.29
68.26	0.02	0.19	0.23	0.02	0.19	0.26	0.02	0.19	0.25
69.06	0.04			0.05			0.04		
72.26	0.01	0.14	0.17	0.01	0.14	0.19	0.01	0.14	0.19
75.46	0.01	0.11	0.13	0.01	0.12	0.15	0.01	0.12	0.15
78.66	0.01	0.09	0.11	0.01	0.09	0.13	0.01	0.09	0.12
81.86	0.01	0.07	0.09	0.01	0.07	0.11	0.01	0.07	0.11
85.00	0.01	0.05	0.10	0.01	0.06	0.13	0.01	0.06	0.12
88.00	0.04	0.09		0.06	0.10		0.06	0.10	
91.00	0.07	0.07		0.09	0.07		0.09	0.08	
94.60	0.08	0.04		0.10	0.05		0.09	0.05	
98.20	0.10	0.03		0.12	0.04		0.09	0.04	
101.90	0.09	0.03		0.11	0.03		0.09	0.03	
104.90	0.08	0.02		0.10	0.03		0.08	0.03	
107.90	0.06	0.02		0.08	0.02		0.07	0.02	
110.90	0.05	0.01		0.07	0.02		0.06	0.02	
113.90	0.04	0.01		0.06	0.02		0.05	0.02	
116.90	0.03	0.01		0.04	0.01		0.04	0.02	

Tab. 5-11: PPCS model A: Radial power density distributions [W/cm^3] of the materials in the blanket modules IV-VI.

	Volumes [cm^3]			Masses [kg]		
	Eurofer	H_2O	Pb-17Li	Eurofer	H_2O	Pb-17Li
Module I						
First wall	$1.548 \cdot 10^5$	$1.972 \cdot 10^4$		$1.207 \cdot 10^3$	$1.972 \cdot 10^1$	
Blanket	$8.994 \cdot 10^5$	$2.317 \cdot 10^5$	$4.131 \cdot 10^6$	$7.015 \cdot 10^3$	$2.317 \cdot 10^2$	$3.925 \cdot 10^4$
Manifold/shielding	$1.345 \cdot 10^6$	$7.039 \cdot 10^5$		$1.049 \cdot 10^4$	$7.039 \cdot 10^2$	
Total	$2.399 \cdot 10^6$	$9.553 \cdot 10^5$	$4.131 \cdot 10^6$	$1.871 \cdot 10^4$	$9.553 \cdot 10^2$	$3.925 \cdot 10^4$
Module II						
First wall	$1.458 \cdot 10^5$	$1.855 \cdot 10^4$		$1.137 \cdot 10^3$	$1.855 \cdot 10^1$	
Blanket	$8.602 \cdot 10^5$	$2.230 \cdot 10^5$	$3.988 \cdot 10^6$	$6.709 \cdot 10^3$	$2.230 \cdot 10^2$	$3.789 \cdot 10^4$
Manifold/shielding	$1.323 \cdot 10^6$	$6.924 \cdot 10^5$		$1.032 \cdot 10^4$	$6.924 \cdot 10^2$	
Total	$2.329 \cdot 10^6$	$9.340 \cdot 10^5$	$3.998 \cdot 10^6$	$1.817 \cdot 10^4$	$9.339 \cdot 10^2$	$3.789 \cdot 10^4$
Module III						
First wall	$1.848 \cdot 10^5$	$2.351 \cdot 10^4$		$1.442 \cdot 10^3$	$2.351 \cdot 10^1$	
Blanket	$1.165 \cdot 10^6$	$2.999 \cdot 10^5$	$5.369 \cdot 10^6$	$9.087 \cdot 10^3$	$2.999 \cdot 10^2$	$5.128 \cdot 10^4$
Manifold/shielding	$1.911 \cdot 10^6$	$9.973 \cdot 10^5$		$1.491 \cdot 10^4$	$9.973 \cdot 10^2$	
Total	$3.261 \cdot 10^6$	$1.321 \cdot 10^6$	$5.369 \cdot 10^6$	$2.543 \cdot 10^4$	$1.321 \cdot 10^3$	$5.128 \cdot 10^4$
Module IV						
First wall	$2.685 \cdot 10^5$	$3.411 \cdot 10^4$		$2.094 \cdot 10^3$	$3.411 \cdot 10^1$	
Blanket	$3.908 \cdot 10^6$	$4.308 \cdot 10^5$	$1.313 \cdot 10^7$	$3.048 \cdot 10^4$	$4.308 \cdot 10^2$	$1.248 \cdot 10^5$
Manifold/shielding	$3.371 \cdot 10^6$	$1.981 \cdot 10^6$		$2.629 \cdot 10^4$	$1,981 \cdot 10^3$	
Total	$7.546 \cdot 10^6$	$2.446 \cdot 10^6$	$1.313 \cdot 10^7$	$5.886 \cdot 10^4$	$2.446 \cdot 10^3$	$1.248 \cdot 10^5$
Module V						
First wall	$2.635 \cdot 10^5$	$3.351 \cdot 10^4$		$2.055 \cdot 10^3$	$3.351 \cdot 10^1$	
Blanket	$3.626 \cdot 10^6$	$4.120 \cdot 10^5$	$1.237 \cdot 10^7$	$2.828 \cdot 10^4$	$4.120 \cdot 10^2$	$1.176 \cdot 10^5$
Manifold/shielding	$3.339 \cdot 10^6$	$1.778 \cdot 10^6$		$2.604 \cdot 10^4$	$1.778 \cdot 10^3$	
Total	$7.229 \cdot 10^6$	$2.224 \cdot 10^6$	$1.237 \cdot 10^7$	$5.638 \cdot 10^4$	$2.224 \cdot 10^3$	$1.176 \cdot 10^5$
Module VI						
First wall	$2.447 \cdot 10^5$	$3.113 \cdot 10^4$		$1.909 \cdot 10^3$	$3.113 \cdot 10^1$	
Blanket	$3.215 \cdot 10^6$	$3.685 \cdot 10^5$	$1.086 \cdot 10^7$	$2.508 \cdot 10^4$	$3.685 \cdot 10^2$	$1.032 \cdot 10^5$
Manifold/shielding	$2.767 \cdot 10^6$	$1.476 \cdot 10^6$		$2.158 \cdot 10^4$	$1.476 \cdot 10^3$	
Total	$6.227 \cdot 10^6$	$1.876 \cdot 10^6$	$1.086 \cdot 10^7$	$4.857 \cdot 10^4$	$1.876 \cdot 10^3$	$1.032 \cdot 10^5$

Tab. 5-12: PPCS model A: Volumes and masses of materials in WCLL blanket modules (20° torus sector).

5.3.2 PPCS model B (HCPB)

Table 5-13 shows an overview of the nuclear power generation in the single poloidal modules divided into first wall (FW), blanket, manifolds and shielding. It is recalled that the MCNP model assumes a toroidal segmentation of 40° for the inboard modules I-IV and 20° for the outboard modules V-XI. The power generation data are thus given for inboard and outboard blanket modules with the corresponding toroidal segmentation. Due to the comparatively small blanket thickness, the power generation in the manifold/shield region is relatively high for the HCPB blanket. The high and low temperature shields contribute to the total nuclear power generation by 3.2% and 1.7%, respectively. Supplementary shield plugs have been employed between the poloidal blanket modules to avoid neutron streaming. As seen in Table 5.13, there is a considerable power generation in these plugs. This has to be taken into accounted in the thermal-hydraulic lay-out of the reactor.

Module	FW	Blanket	HT	LT	Manifold	Modules[a)]	Reactor
I	2.05E+00	2.13E+01	1.26E+00	9.85E-01	1.32E-02	2.56E+01	2.30E+02
II	1.46E+00	1.55E+01	8.74E-01	6.12E-01	4.49E-03	1.84E+01	1.66E+02
III	1.47E+00	1.55E+01	8.83E-01	6.21E-01	4.29E-03	1.84E+01	1.66E+02
IV	1.94E+00	1.98E+01	1.10E+00	7.76E-01	6.18E-03	2.36E+01	2.13E+02
V	6.80E-01	7.55E+00	2.57E-01	1.10E-01	1.43E-03	8.59E+00	1.55E+02
VI	1.23E+00	1.44E+01	5.33E-01	2.28E-01	2.71E-03	1.64E+01	2.95E+02
VII	1.35E+00	1.61E+01	6.01E-01	2.57E-01	3.13E-03	1.83E+01	3.30E+02
VIII	1.57E+00	1.90E+01	7.28E-01	3.07E-01	3.72E-03	2.16E+01	3.89E+02
IX	1.72E+00	2.10E+01	8.08E-01	3.43E-01	3.65E-03	2.38E+01	4.29E+02
X	1.57E+00	1.89E+01	7.17E-01	3.05E-01	3.44E-03	2.15E+01	3.86E+02
XI	1.73E+00	2.06E+01	8.00E-01	3.68E-01	6.24E-03	2.35E+01	4.22E+02
VV					5.35E-03	1.00E+00	1.80E+01
Divertor					9.58E-02	1.79E+01	3.23E+02
Shield plugs					3.39E-02	6.35E+00	1.14E+02
Total							3.64E+03

[a)] Data refer to single modules. Inboard modules 1-4 and outboard modules 5-11 have toroidal extensions of 40 ° and 20°, respectively.

Tab. 5-13: PPCS model B: Nuclear power production [MW] in HCPB blanket modules, vacuum vessel (VV), shield plugs and divertor.

Table 5-14 compares the nuclear power production of the single blanket modules with the fusion neutron power flowing into the modules. The (local) power multiplication factor is lowest for the central outboard blanket module and highest for the top outboard blanket module, i. e. the modules showing the highest and lowest, respectively, neutron wall loading and power production. Note that the power multiplication factor averaged over the blanket modules is very close to the global energy multiplication factor (Tab. 5-6) in case of the HCPB blanket with the Beryllium neutron multiplier.

Module	First wall area [cm^2]	Neutron wall loading [MW/m^2]	Fusion neutron power [MW]	Nuclear heating power [MW]	Multiplication factor(*)
I	1.19E+05	1.48	1.76E+01	2.56E+01	1.45E+00
II	7.84E+04	1.78	1.39E+01	1.84E+01	1.32E+00
III	7.84E+04	1.78	1.39E+01	1.84E+01	1.32E+00
IV	1.11E+05	1.50	1.67E+01	2.36E+01	1.42E+00
V	4.03E+04	1.44	5.81E+00	8.59E+00	1.48E+00
VI	6.71E+04	1.71	1.15E+01	1.64E+01	1.43E+00
VII	6.96E+04	1.92	1.33E+01	1.83E+01	1.38E+00
VIII	7.64E+04	2.12	1.62E+01	2.16E+01	1.33E+00
IX	8.15E+04	2.24	1.83E+01	2.38E+01	1.30E+00
X	7.67E+04	2.11	1.62E+01	2.15E+01	1.33E+00
XI	9.10E+04	1.82	1.66E+01	2.35E+01	1.41E+00
Average		1.85			1.37E+00

(*) Ratio of nuclear power generated in the module(s) and the fusion neutron power loaded to the first wall of the module(s).

Tab. 5-14: PPCS model B: Nuclear power production and multiplication factor for the HCPB blanket modules.

Radial power density profiles of the Li_4SiO_4 breeder ceramics, the Beryllium neutron multiplier and the Eurofer structure are shown in Fig. 5-6 for the central inboard and outboard blanket modules. The corresponding numerical data are given in Table 5-15. Associated volumes and masses as calculated for the MCNP torus sector model are given in Table 5-16. Note that the maximum power density of the Li_4SiO_4 breeder ceramics amounts to no more than 26 W/cm^3 assuming a ^6Li-enrichment of 20 at%.

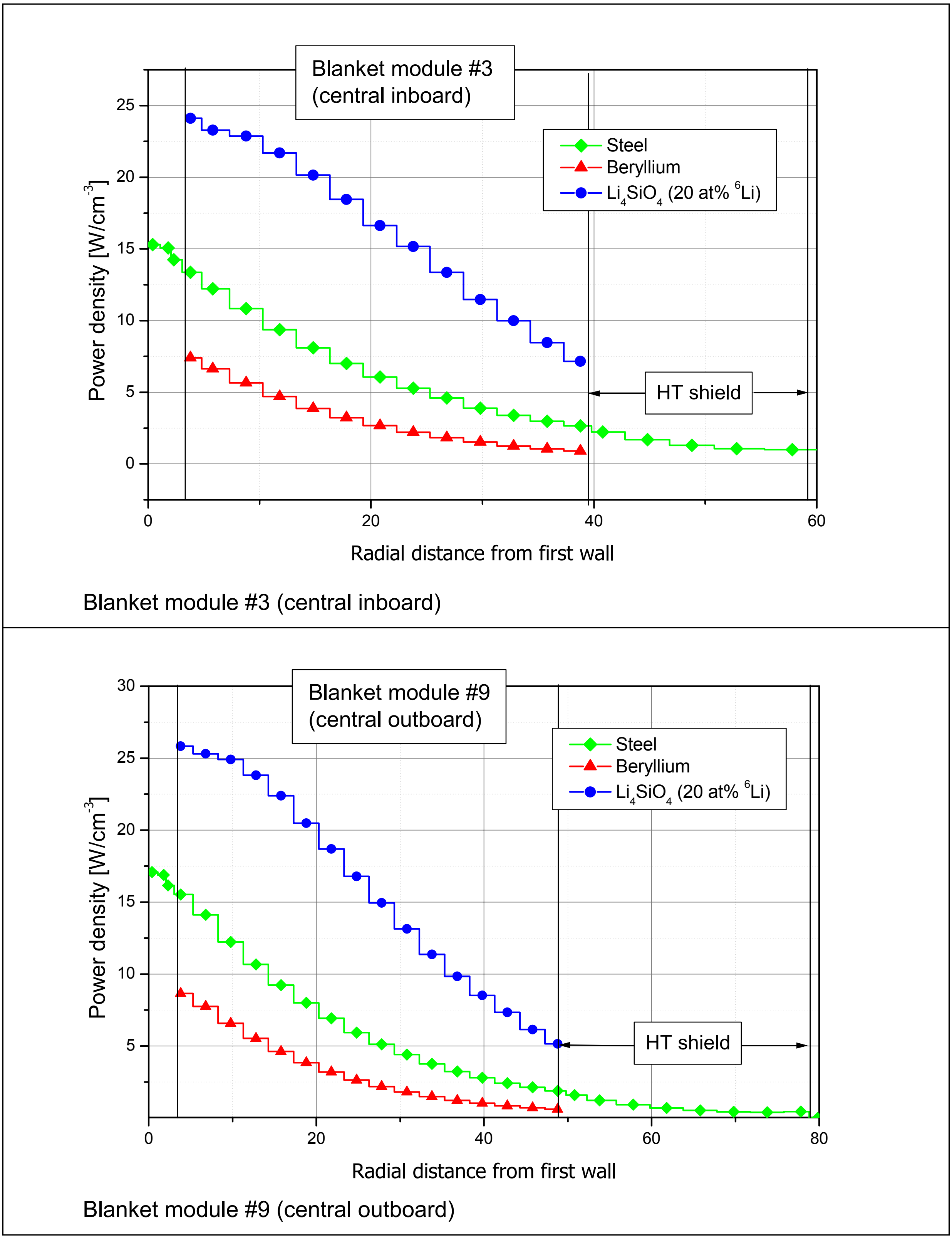


Fig. 5-6: PCS model B: Radial power density profiles of the materials in the central inboard and outboard HCPB blanket modules

Central outboard module (#9)				Central inboard module (#3)			
r[cm]	Eurofer	Be	Breeder	r[cm]	Eurofer	Be	Breeder
0.40	17.09			0.40	15.31		
1.80	16.88			1.80	15.07		
2.30	16.16			2.30	14.24		
3.80	15.53	8.65	25.84	3.80	13.37	7.41	24.11
6.80	14.12	7.76	25.30	5.80	12.22	6.64	23.28
9.80	12.23	6.57	24.92	8.80	10.85	5.66	22.88
12.80	10.68	5.53	23.82	11.80	9.36	4.70	21.70
15.80	9.23	4.62	22.39	14.80	8.10	3.87	20.14
18.80	8.00	3.85	20.48	17.80	7.01	3.22	18.45
21.80	6.93	3.20	18.70	20.80	6.06	2.67	16.63
24.80	5.94	2.64	16.79	23.80	5.28	2.22	15.17
27.80	5.12	2.18	14.94	26.80	4.60	1.84	13.37
30.80	4.40	1.80	13.15	29.80	3.89	1.52	11.47
33.80	3.75	1.49	11.36	32.80	3.38	1.26	9.99
36.80	3.24	1.23	9.84	35.80	2.97	1.05	8.47
39.80	2.80	1.02	8.52	38.80	2.66	0.91	7.16
42.80	2.42	0.84	7.34	40.80	2.23		
45.80	2.13	0.71	6.15	44.80	1.70		
48.80	1.89	0.61	5.15	48.80	1.30		
50.80	1.59			52.80	1.06		
53.80	1.22			57.80	1.00		
57.80	0.93			59.80			
61.80	0.69			63.97	0.85		
65.80	0.53			68.13	0.61		
69.80	0.44			72.30	0.35		
73.80	0.40			76.47	0.21		
77.80	0.44			80.63	0.14		
79.80	0.00			84.80	0.11		
83.37	0.50			89.80	0.12		
86.94	0.41			94.80	0.11		
90.51	0.28			99.80	0.11		
94.09	0.19						
97.66	0.15						
101.23	0.14						
104.80	0.15						
109.80	0.14						
114.80	0.14						
119.80	0.14						
124.80	0.14						
129.80	0.14						

Tab. 5-15: PPCS model B: Radial power density distributions [W/cm^3] of the materials in the central inboard and outboard HCPB blanket modules

	Blanket + HT shield		LT shield		Manifold		Total	
Module	volume [cm^3]	mass (g)	volume [cm^3]	mass (g)	volume [cm^3]	mass (g)	volume [cm^3]	mass (g)
1	6.55E+06	2.01E+07	2.63E+06	1.66E+07	1.52E+06	1.80E+06	1.07E+07	3.84E+07
2	4.27E+06	1.31E+07	1.69E+06	1.05E+07	9.70E+05	1.13E+06	6.93E+06	2.47E+07
3	4.27E+06	1.31E+07	1.69E+06	1.05E+07	9.70E+05	1.13E+06	6.93E+06	2.47E+07
4	6.07E+06	1.85E+07	2.43E+06	1.51E+07	1.40E+06	1.64E+06	9.90E+06	3.52E+07
5	3.04E+06	9.70E+06	9.43E+05	5.88E+06	9.27E+05	1.09E+06	4.91E+06	1.67E+07
6	5.28E+06	1.67E+07	1.74E+06	1.08E+07	1.76E+06	2.05E+06	8.78E+06	2.95E+07
7	5.55E+06	1.76E+07	1.85E+06	1.15E+07	1.88E+06	2.20E+06	9.28E+06	3.13E+07
8	6.12E+06	1.94E+07	2.05E+06	1.28E+07	2.09E+06	2.45E+06	1.03E+07	3.47E+07
9	6.52E+06	2.07E+07	2.19E+06	1.36E+07	2.23E+06	2.61E+06	1.09E+07	3.69E+07
10	6.12E+06	1.95E+07	2.05E+06	1.28E+07	2.09E+06	2.45E+06	1.03E+07	3.47E+07
11	7.26E+06	2.30E+07	2.43E+06	1.51E+07	2.47E+06	2.89E+06	1.22E+07	4.10E+07

NB: Inboard modules 1-4 and outboard modules 5-11 have toroidal extensions of 40 ° and 20°, respectively.

Tab. 5-16: PPCS model B: Volumes and masses of materials of HCPB blanket modules.

5.3.3 PPCS model C (DCLL)

Table 5-17 shows the nuclear power generation as calculated for PPCS model C on the basis of the 10° sector model (MCNP) with banana-type DCLL blanket segments. With the DCLL reference design, ≈ 4% of the nuclear power is generated in the water cooled low (LT) temperature shield (cf. Fig. 3-6). The heating power of the LT shield, however, may be not utilised for the electricity production and, therefore, must be minimised. This requires to improve the shielding performance of the HT shield. As a consequence, hydrogenous material such as ZrH_2 must be introduced in the HT shield, or a more efficient shielding material (e. g. WC) has to be used. In either case, the nuclear power production in the LT shield would be reduced to less than 2% of the total nuclear heat without affecting the tritium breeding capability.

Table 5-18 shows the nuclear power generation for the DCLL reference design with a toroidal segmentation of 22.5° assuming 11 poloidal blanket modules. These data have been assessed on the basis of the calculations for the 10° sector model with banana-type blanket segments. Fig. 5-7 shows the radial profiles of the power density as calculated for the inboard and the outboard torus mid-plane. Numerical data are given in Tables 5-19 and 5-20.

	Power [MW]	Shares [%]
First wall	333	9.99
Blanket	2452	73.6
HT shield	65.6	1.97
LT shield	128.7	3.9
Vacuum vessel	7.6	0.2
Divertor	346	10.4
Total	3333	

Tab. 5-17: PPCS model C: Nuclear power generation calculated for 10° sector model with large banana-type DCLL blanket segments.

Module #	Relative fraction	Power [MW]	Module #	Relative fraction	Power [MW]
Inboard			Outboard		
I	0.058	10.39	VI	0.106	18.88
II	0.065	11.56	VII	0.125	22.28
III	0.063	11.28	VIII	0.110	19.56
IV	0.063	11.16	IX	0.110	19.57
V	0.050	8.89	X	0.133	23.66
			XI	0.117	20.93
Total	0.299	53.27		0.701	124.89
Grand Total	1.000	178.17			

Tab. 5-18: PPCS model C: Nuclear power generation assessed for 22.5° torus sector including 11 DCLL blanket modules.

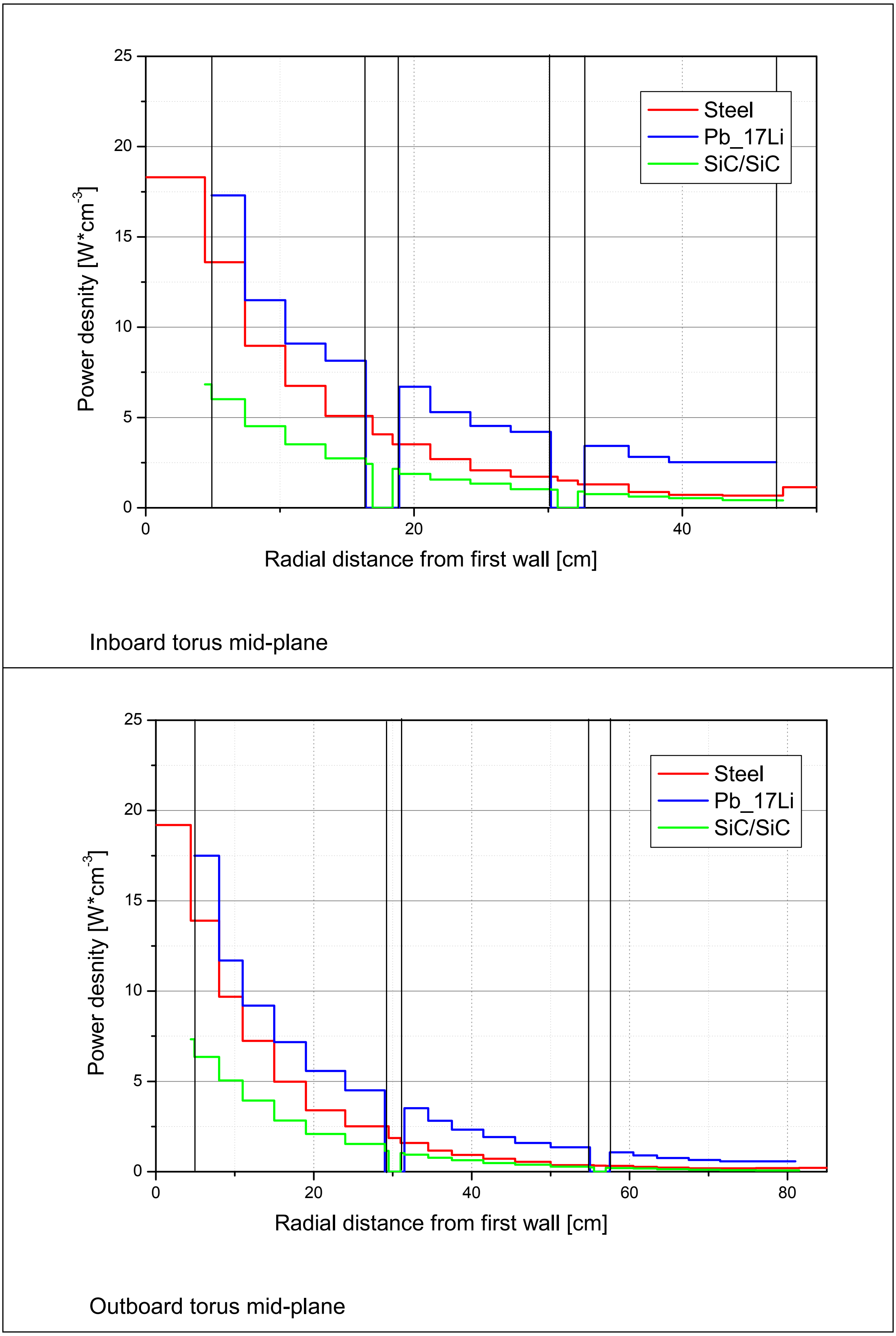


Fig. 5-7: PPCS model C: Radial power density profiles of the materials in the in-board and outboard DCLL blanket segments at torus mid-pane

Steel		Pb-17Li		SiC/SiC	
R_i[cm]	P [W/cm**3]	R_i[cm]	P [W/cm**3]	R_i[cm]	P [W/cm**3]
0.000E+00		4.900E+00		4.400E+00	
4.400E+00	1.923E+01	8.000E+00	1.748E+01	4.900E+00	7.325E+00
8.000E+00	1.385E+01	1.100E+01	1.174E+01	8.000E+00	6.357E+00
1.100E+01	9.690E+00	1.500E+01	9.203E+00	1.100E+01	5.050E+00
1.500E+01	7.235E+00	1.900E+01	7.166E+00	1.500E+01	3.940E+00
1.900E+01	4.978E+00	2.400E+01	5.583E+00	1.900E+01	2.826E+00
2.400E+01	3.387E+00	2.900E+01	4.506E+00	2.400E+01	2.088E+00
2.950E+01	2.505E+00	3.150E+01		2.900E+01	1.533E+00
3.100E+01	1.859E+00	3.450E+01	3.520E+00	2.950E+01	1.154E+00
3.450E+01	1.593E+00	3.750E+01	2.805E+00	3.100E+01	
3.750E+01	1.163E+00	4.150E+01	2.328E+00	3.150E+01	1.029E+00
4.150E+01	9.262E-01	4.550E+01	1.924E+00	3.450E+01	9.360E-01
4.550E+01	7.200E-01	5.000E+01	1.591E+00	3.750E+01	7.671E-01
5.000E+01	5.467E-01	5.500E+01	1.354E+00	4.150E+01	6.389E-01
5.550E+01	3.561E-01	5.750E+01		4.550E+01	4.818E-01
5.700E+01	3.285E-01	6.050E+01	1.067E+00	5.000E+01	3.774E-01
6.050E+01	3.214E-01	6.350E+01	8.926E-01	5.500E+01	2.768E-01
6.350E+01	2.631E-01	6.750E+01	7.541E-01	5.550E+01	2.406E-01
6.750E+01	2.206E-01	7.150E+01	6.455E-01	5.700E+01	
7.150E+01	1.794E-01	7.600E+01	5.702E-01	5.750E+01	2.171E-01
7.600E+01	1.937E-01	8.100E+01	5.733E-01	6.050E+01	1.986E-01
8.150E+01	1.982E-01			6.350E+01	1.688E-01
8.550E+01	2.134E-01			6.750E+01	1.378E-01
				7.150E+01	1.117E-01
				7.600E+01	8.348E-02
				8.100E+01	6.990E-02
				8.150E+01	6.680E-02

Tab. 5-19: PPCS model C: Radial power density distributions [W/cm^3] of the materials in the outboard mid-plane of the DCLL blanket.

Steel		**Pb-17Li**		**SiC/SiC**	
R_i[cm]	P [W/cm**3]	R_i[cm]	P [W/cm**3]	R_i[cm]	P [W/cm**3]
0.000E+00		4.900E+00		4.400E+00	
4.400E+00	1.834E+01	7.400E+00	1.728E+01	4.900E+00	6.827E+00
7.400E+00	1.363E+01	1.040E+01	1.149E+01	7.400E+00	6.012E+00
1.040E+01	8.966E+00	1.340E+01	9.088E+00	1.040E+01	4.523E+00
1.340E+01	6.748E+00	1.640E+01	8.142E+00	1.340E+01	3.505E+00
1.690E+01	5.087E+00	1.890E+01		1.640E+01	2.738E+00
1.840E+01	4.065E+00	2.120E+01	6.695E+00	1.690E+01	2.433E+00
2.120E+01	3.509E+00	2.420E+01	5.302E+00	1.840E+01	
2.420E+01	2.690E+00	2.720E+01	4.532E+00	1.890E+01	2.157E+00
2.720E+01	2.080E+00	3.020E+01	4.197E+00	2.120E+01	1.872E+00
3.070E+01	1.720E+00	3.270E+01		2.420E+01	1.562E+00
3.220E+01	1.510E+00	3.600E+01	3.420E+00	2.720E+01	1.328E+00
3.600E+01	1.287E+00	3.900E+01	2.822E+00	3.020E+01	1.032E+00
3.900E+01	8.670E-01	4.300E+01	2.526E+00	3.070E+01	1.001E+00
4.300E+01	7.123E-01	4.700E+01	2.533E+00	3.220E+01	
4.750E+01	6.716E-01			3.270E+01	8.997E-01
5.050E+01	1.132E+00			3.600E+01	7.483E-01
				3.900E+01	6.206E-01
				4.300E+01	5.229E-01
				4.700E+01	4.250E-01
				4.750E+01	4.152E-01

Tab. 5-20: PPCS model C: Radial power density distributions [W/cm^3] of the materials in the inboard mid-plane of the DCLL blanket.

5.3.4 PPCS model D (SCLL)

Table 5-21 shows the nuclear power generation of the inboard, topboard and outboard SCLL blanket segments divided into first wall (FW), blanket and shielding. There is a comparatively high power production in the HT shield (8% of the total nuclear power) and, accordingly, a low power production in the LT shield of the SCLL blanket (less than 0.1% of the total nuclear power). This is due to the fact the SCLL concept assumes the efficient WC shielding material for both the HT and the LT shield resulting in a strong radiation absorption already in the HT shield attached to the SCLL blanket. Such a power partition is very favourable in terms of the thermal plant efficiency since the thermal power produced in the LT shield cannot be utilized for the electricity production. Table 5-22 compares the nuclear power production of the single SCLL blanket segments with the fusion neutron power flowing into the segments. The partition of the power generation among the materials is shown in Table 5-23.

Fig. 5-8 shows the radial profiles of the power density as calculated for the inboard and the outboard torus mid-plane. The corresponding numerical data are given in Table 5-24.

Segment	FW	Blanket		Shield		Sector	Reactor
		front	back	HT	LT	(11.25°)	
Inboard	1.95	10.69	-	2.87	0.02	15.53	497
Topboard	0.76	4.63	-	1.41	0.02	6.82	218
Outboard	4.89	26.83	4.87	1.38	0.001	37.97	1215
Divertor	2.10	4.43		1.94		8.47	271
Vacuum vessel						0.01	0.32
Total						68.80	2202

Tab. 5-21: PPCS model D: Nuclear power production [MW] in SCLL blanket segments, vacuum vessel and divertor (11.25° torus sector).

Module	First wall area [cm^2]		Neutron wall loading [MW/m^2]	Fusion neutron power [MW]		Nuclear heating power [MW]	
	segments	reactor		segments	reactor	segments	reactor
Inboard	$5.84 \cdot 10^4$	$1.87 \cdot 10^6$	2.17	$1.27 \cdot 10^1$	$4.05 \cdot 10^2$	$1.55 \cdot 10^1$	$4.97 \cdot 10^2$
Topboard	$2.25 \cdot 10^4$	$7.22 \cdot 10^5$	2.24	$5.05 \cdot 10^0$	$1.62 \cdot 10^2$	$6.82 \cdot 10^0$	$2.18 \cdot 10^2$
Outboard	$1.28 \cdot 10^5$	$4.10 \cdot 10^6$	2.98	$3.82 \cdot 10^1$	$1.22 \cdot 10^3$	$3.80 \cdot 10^1$	$1.215 \cdot 10^3$
Divertor	$2.37 \cdot 10^4$	$7.70 \cdot 10^5$	1.94	4.68	$1.50 \cdot 10^2$	$8.47 \cdot 10^0$	$2.71 \cdot 10^2$
Total		$7.46 \cdot 10^6$	2.60		$1.94 \cdot 10^3$		$2.20 \cdot 10^3$

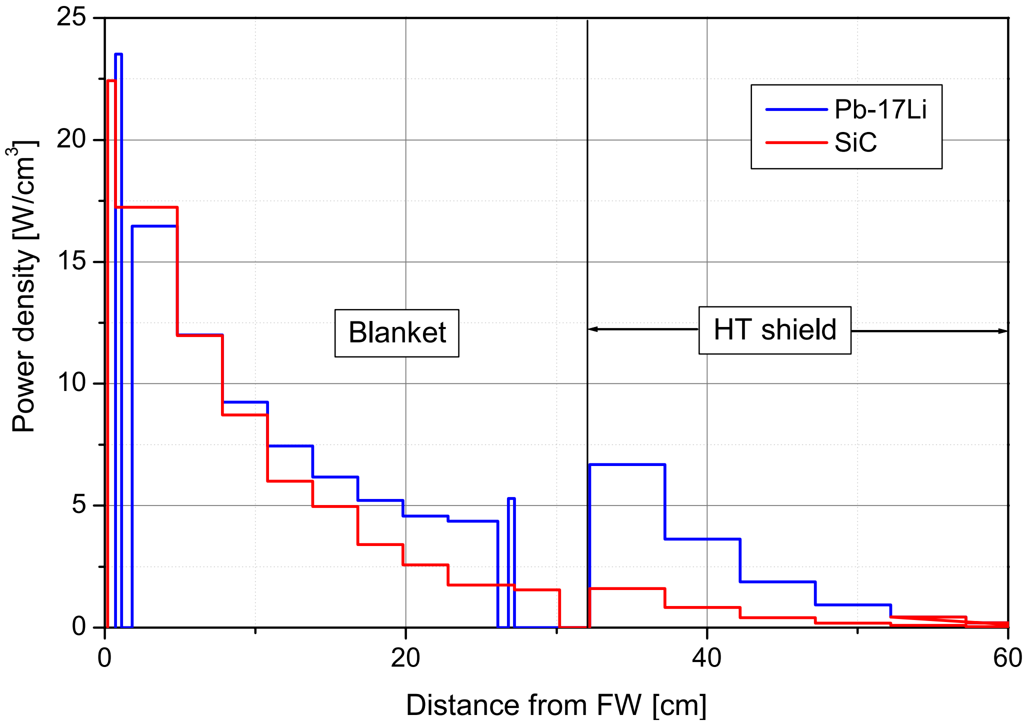
Tab. 5-22: PPCS model D: First wall area, fusion neutron and nuclear power of SCLL blanket segments (11.25° torus sector).

	W	Pb-17Li	SiC	WC	Borated steel
Inboard blanket segment					
First wall	0.16	0.45	1.34	-	-
Blanket	-	8.86	1.83	-	-
HT shield	-	0.29	0.12	2.46	
LT shield	-	-	-	0.018	0.003
Topboard blanket segment					
First wall	0.04	0.18	0.54	-	-
Blanket	-	3.96	0.67	-	-
HT shield	-	0.11	0.03	1.27	-
LT shield	-	-	-	0.016	0.003
Outboard blanket segment					
First wall	1.57	1.11	2.21	-	-
Blanket (front)	-	23.36	3.47	-	-
Blanket (back)	-	4.39	0.48	-	-
HT shield	-	0.10	0.02	1.26	
LT shield	-	-	-		
Divertor					
	W	Pb-17Li	SiC	WC	
First wall	0.50	1.34	0.26	-	
Blanket	-	3.67	0.76	-	
Shielding	-	0.19	0.03	1.72	
Total	0.50	5.20	1.05	1.72	

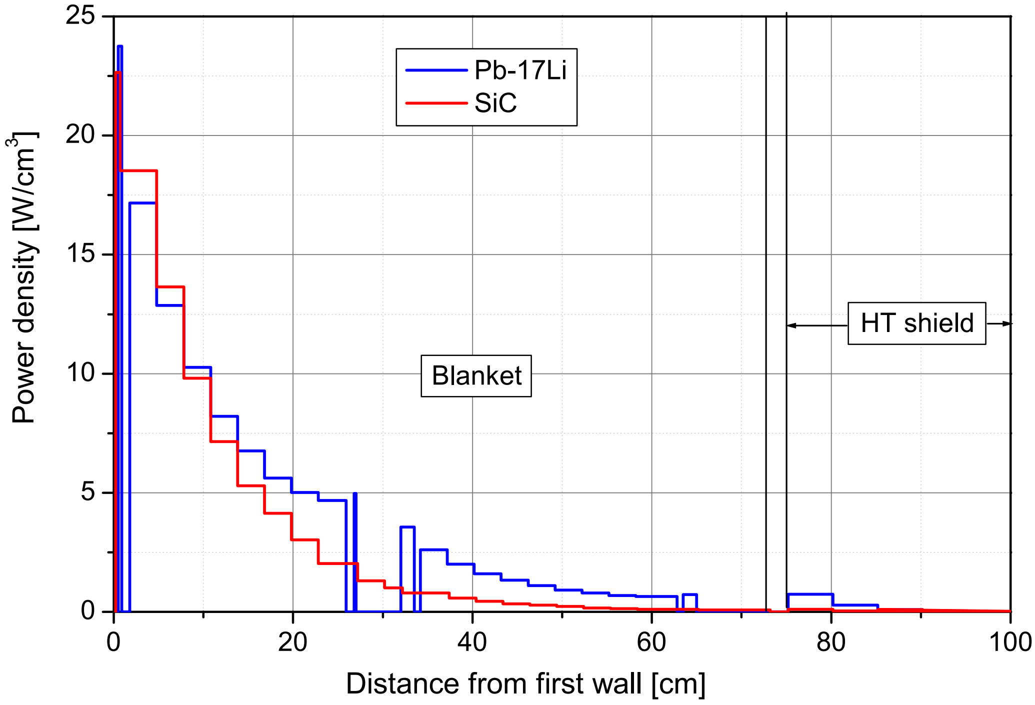
Tab. 5-23: PPCS model D: Nuclear power generation [MW] in the different materials of the SCLL blanket segments (11.25° torus sector)

Outboard blanket segment					Inboard blanket segment			
Pb-17Li		SiC			Pb-17Li		SiC	
R [cm]	P [W/cm^3]	R [cm]	P [W/cm^3]		R [cm]	P [W/cm^3]	R [cm]	P [W/cm^3]
0.7		0.2			0.7	-	0.2	-
1.1	23.7	0.7	22.6		1.1	23.5	0.7	22.4
1.8		4.8	18.5		1.8	-	4.8	17.2
4.8	17.2	7.8	13.7		4.8	16.5	7.8	12.0
7.8	12.9	10.8	9.8		7.8	12.0	10.8	8.7
10.8	10.3	13.8	7.1		10.8	9.2	13.8	6.0
13.8	8.2	16.8	5.3		13.8	7.4	16.8	5.0
16.8	6.8	19.8	4.1		16.8	6.2	19.8	3.4
19.8	5.6	22.8	3.0		19.8	5.2	22.8	2.6
22.8	5.0	27.2	2.0		22.8	4.6	27.2	1.7
26.1	4.7	30.2	1.3		26.1	4.4	30.2	1.5
26.8		32.2	1.0		26.8	-	32.2	-
27.2	5.0	37.4	0.8		27.2	5.3	37.2	1.6
32.2	-	40.4	0.6		32.2	-	42.2	0.8
33.7	3.6	43.4	0.4		37.2	6.7	47.2	0.4
34.4		46.4	0.3		42.2	3.6	52.2	0.2
37.4	2.6	49.4	0.3		47.2	1.9	57.2	0.1
40.4	2.0	52.4	0.2		52.2	0.9	62.2	0.0
43.4	1.6	55.4	0.2		57.2	0.4		
46.4	1.3	58.4	0.1		62.2	0.2		
49.4	1.1	65.2	0.1					
52.4	0.9	69.2	0.1					
55.4	0.8	73.2	0.1					
58.4	0.7	75.2	-					
63.0	0.6	80.2	0.1					
63.7	-	85.2	0.0					
65.2	0.7	90.2	0.0					
75.2	-	95.2	0.0					
80.2	0.7	100.2	0.0					
85.2	0.3	105.2	0.0					
90.2	0.1							
95.2	0.0							
100.2	0.0							
105.2	0.0							

Tab. 5-24: PPCS model D: Radial power density distributions of the materials in the torus mid-plane of the inboard and outboard SCLL blanket segments.



Inboard torus mid-plane



Outboard torus mid-plane

Fig. 5-8: PPCS model D: Radial power density profiles of the materials in the inboard and outboard SCLL blanket segments at torus mid-pane

5.4 Shielding performance

There are two essential shielding requirements that must be fulfilled: first, the re-weldability of life-time components made of steel such as the vacuum vessel, and, possibly, the low temperature (LT) shield, and, second, the sufficient protection of the super-conducting toroidal field (TF) coils from the radiation penetrating the vacuum vessel. A lifetime of 40 full power years (fpy) is considered necessary for the operation of the PPCS fusion power plants.

While the build of the blanket is dictated by the breeding requirements, the material composition of the shields and the vacuum vessel can be optimised for shielding. To this end, an efficient neutron moderator (i. e. a hydrogenous material) is required combined with a good neutron absorber (steel, tungsten, etc.). The build of the HT and LT shields is specific to the blanket concepts and are discussed in the respective sections. As for the vacuum vessel, the ITER FDR design has been assumed for all of the four PPCS models. It consists of two 5 cm thick SS-316 steel plates with, at the inboard side, a 25 cm thick shielding mixture of water (40%) and borated steel (60%) in between. The total thickness of the vacuum vessel thus amounts to 35 cm at the inboard side.

The shielding calculations were performed for the inboard mid-plane of the PPCS reactors where the shielding is most crucial due to the restricted space available for the blanket and shield system (in general about 100 cm) while at the same time the neutron wall loading shows a peaking value.

5.4.1 Helium production in Eurofer (Models A, B and C)

Based on existing data, the current assumption is that re-welding of stainless steel should be successful at He concentrations below 1 appm [GDR95]. Radial profiles of the Helium production in the Eurofer structure are shown in Figs. 5-9 – 5-11 for the PCCS models A, B and C. The re-weldability criterion for a plant lifetime of 40 full power years can be fulfilled at penetration depths of 85, 90 and 95 cm for the models A (WCLL), B (HCPB) and C (WCLL) blankets, respectively. In the case of model A, the required penetration depth of 85 cm is at the back of the shield when using the reference WCLL blanket/shield design. As shown in section 5.4.3 below, however, it is required to enlarge the shield thickness of the model A to satisfy the shielding requirements for the super-conducting TF-coil. As for models C and D, the required penetration depths are at the back of the LT shields when using again the reference shield designs of the respective blankets (see sections 5.4.3.4, 5.4.3.5 below). As a result, weld joints have to be placed only at the back of the HT shields if designed as lifetime components.

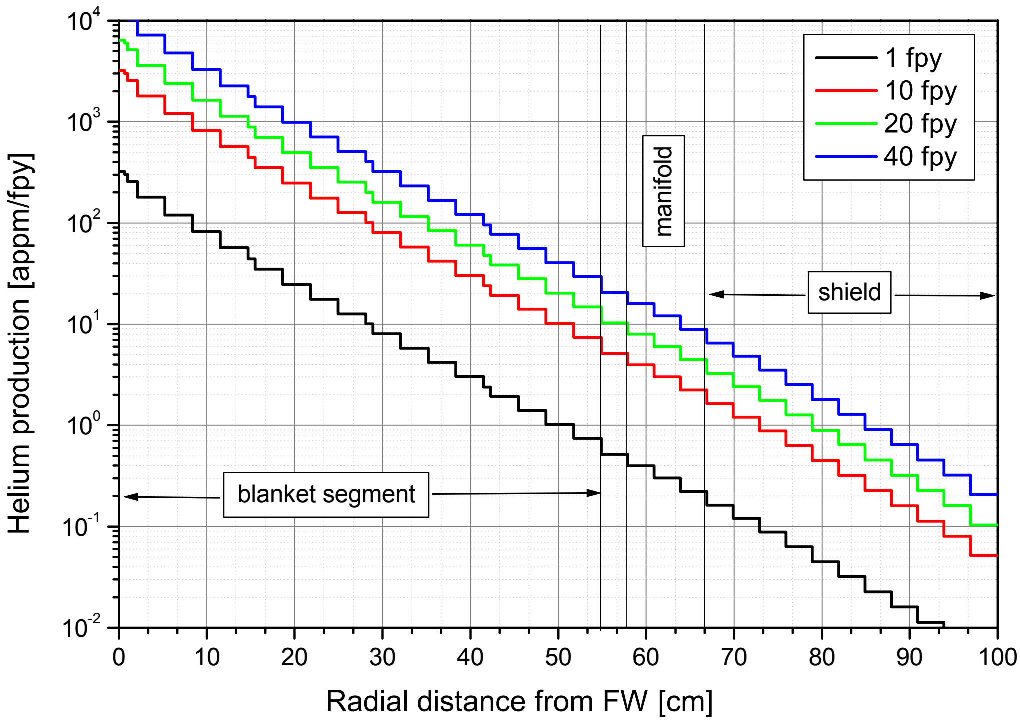


Fig.. 5-9: PPCS model A: Radial profiles of the Helium produced in the Eurofer structure of the inboard WCLL blanket segment at torus mid-plane.

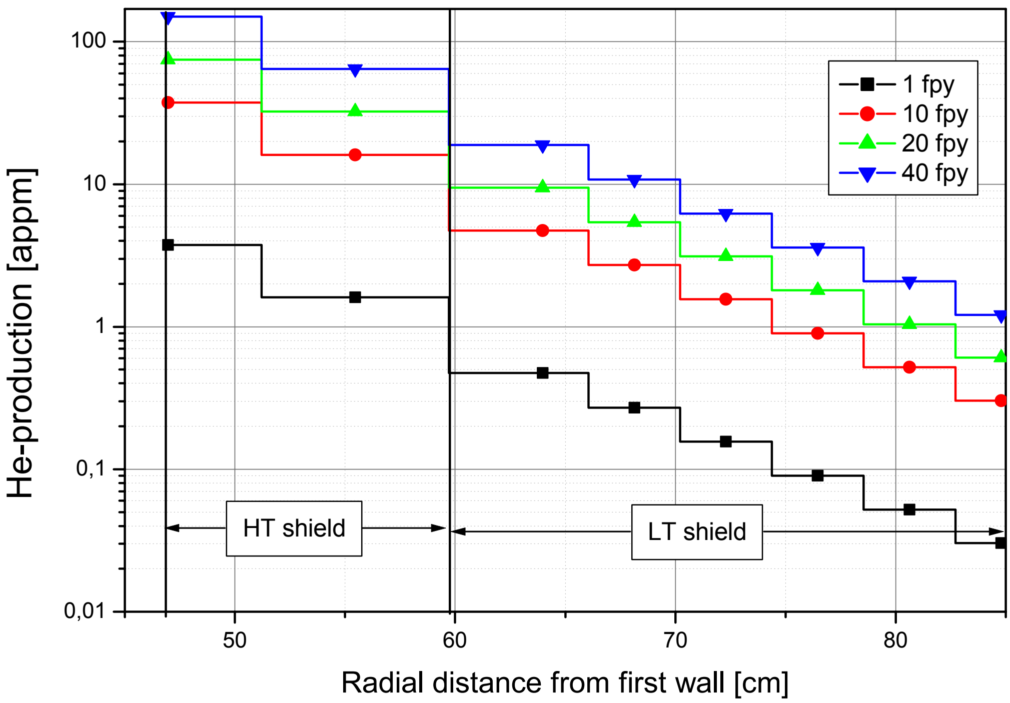


Fig. 5-10: PPCS model B: Radial profiles of the Helium produced in the Eurofer structure of the HT and LT shields of the central inboard HCPB blanket module.

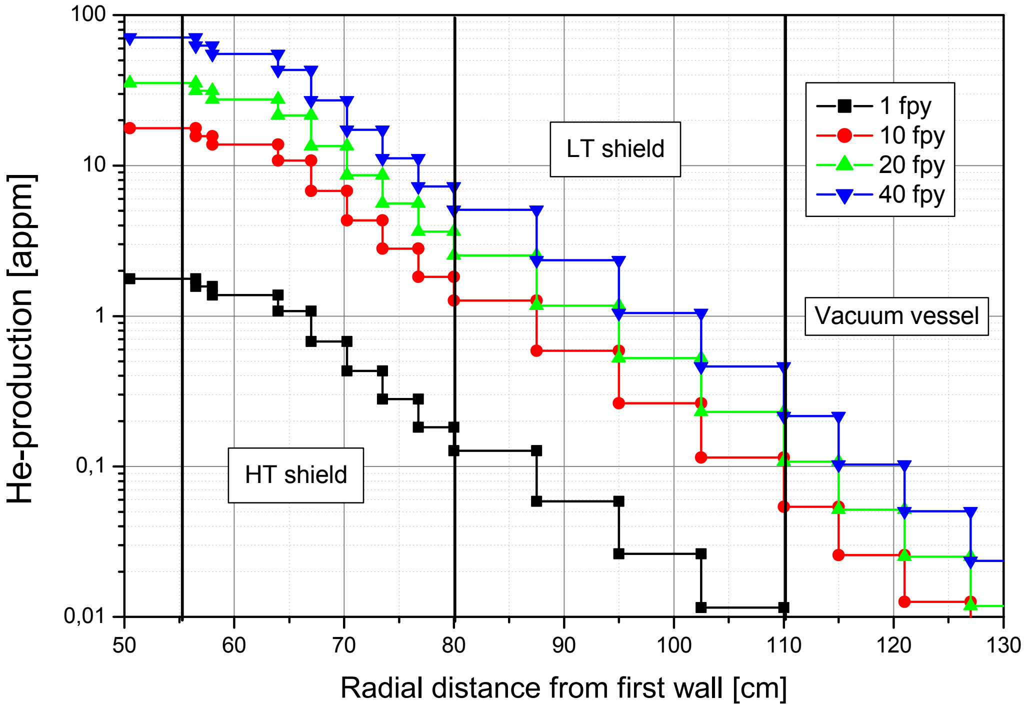


Fig. 5-11: PPCS model D: Radial profiles of the Helium produced in the Eurofer structure of the HT and LT shields of the inboard DCLL blanket segment at torus mid-plane.

5.4.2 SiC burn-up (model D)

As a result of various transmutation reactions on Si and C such as C-12 (n,n'α)He-4, C-12(n, α)Be-9, Si-28(n,p)Al-28, Si-28(n,d)Al-27 and Si-28(n,n' α)Mg-24, there is a significant burn-up of SiC in a fusion power reactor. This may be a lifetime limitation for the SiC structure of the SCLL blanket in addition to the radiation damage accumulation which deteriorates the mechanical properties of the SiC/SiC composite. Fig. 5-12 shows the radial profile of the SiC burn-up calculated for one full power year for the SCLL outboard blanket segment at torus mid-plane. It is revealed that the SiC based SCLL blanket could be operated for 3.5 full power years assuming a SiC burn-up of 3at% as upper limit and assuming, at the same time, no failure or severe degradation of the SiC/SiC composite due to the radiation induced damage.

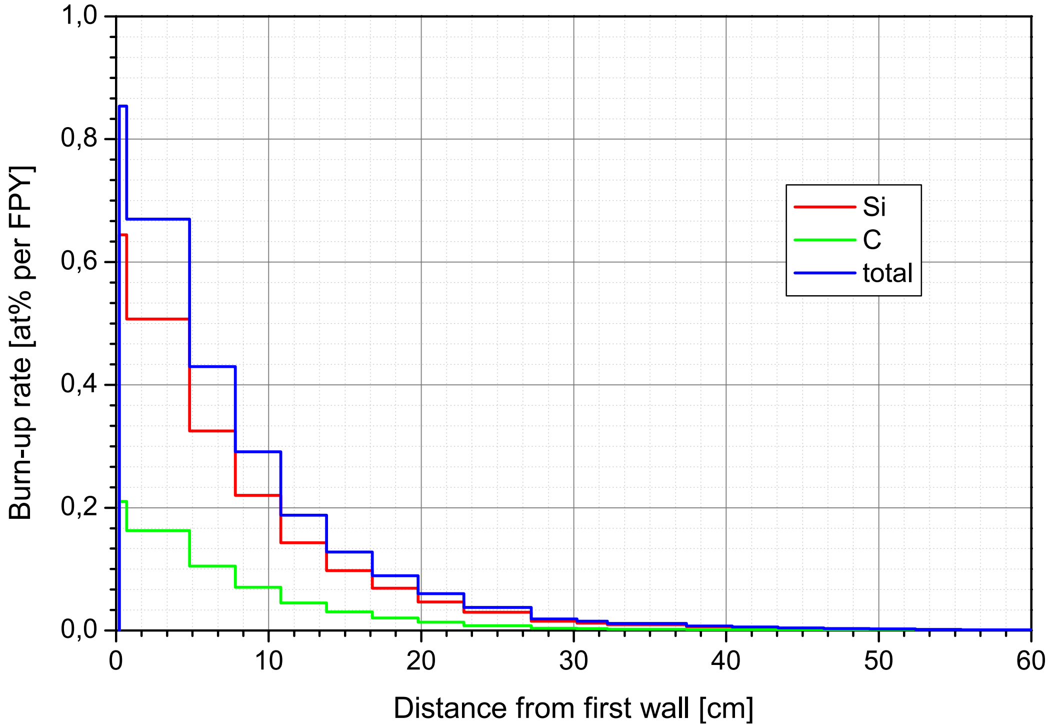


Fig. 5-12: PPCS model D: Radial profile of SiC burn-up in SCLL outboard blanket segment at torus mid-plane.

5.4.3 Radiation loads to the super-conducting TF-coil.

5.4.3.1 Overview

The most crucial radiation loads to the TF-coil are the fast neutron fluence to the superconductor, the peak nuclear heating in the winding pack, the radiation damage to the copper insulator and the radiation dose absorbed by the Epoxy resin insulator. Suitable design radiation limits for the super-conducting TF-coils were defined by ITER [GDR95]. It is recalled that both the shield and the vacuum vessel must be optimised for protecting the TF-coil from the penetrating radiation. This refers to both the material composition and the total thickness of the shield system including the vacuum vessel. A good shielding performance can be achieved by combining an efficient neutron moderator (i. e. a hydrogenous material such as water or a hydride) combined with a good neutron absorber (steel, tungsten). In this way fast neutrons are slowed down and captured by the resonances of the absorber material.

A steel water mixture as used for the ITER shielding blanket modules has proven to be a simple and efficient shield composition. Typically, the steel fraction is between 60 and 80 vol% steel and the water fraction, correspondingly, between 20 to 40 vol%. A shield of the water/steel type is considered for PPCS models A and C employing the WCLL and the DCLL blankets with Pb-17Li as breeder material. In the case of model A, water is used as coolant of the breeding zone. Thus it is straight forward to make use of water in the shield as well. In the case of model C, this means, however, to consider a second coolant in addition to Pb-17Li which is used for cooling the breeding zone. This is a reasonable solution as far as water is also used in the vacuum vessel. Analogous to models B and D, water can be avoided in the shielding system by using hydrides and/or efficient neutron absorbers such as tungsten carbide (WC). It is noted, however, that the use of water does not impose any technical or safety related problem for model C. Water must not be used for model B (HCPB blanket), however, to avoid any potential reaction with the hot beryllium during a possible accident. The HCPB reference design therefore employs ZrH_2 as neutron moderator in the 25 cm thick LT shield attached to the vacuum vessel.

The SCLL blanket, used for model D, shows an inherent low shielding performance due to the use of the SiC structure which does not significantly absorb neutrons. In addition, the SCLL blanket concept assumes a rather wide and straight gap assumed between the toroidal blanket segments (cf. Fig. 3-7).The segment gap, however, is subject to design optimization. The bulk shield option with no segment gap was therefore considered in the first place since it represents the best case one could achieve for an optimal shielding efficiency. The worst case would be the straight gap with a width of 4 cm as assumed in the current design of the SCLL reactor.

Table 5-25 compares the radiation loads calculated for the four PPCS models at the inboard mid-plane with the radiation design limits as specified for ITER. It is noted that the design limits can be met for all of the four PPCS plant models assuming the material build shown in Tables 3-1 through 3-4, and no segment gap in case of model D (SCLL) and an increased shield thickness in case of model A (WCLL), see the following sub-sections.

	Design limits	WCLL (model A)	HCPB (model B)	DCLL (model C)	SCLL (model D) 4 cm gap	SCLL (model D) bulk shield
Integral Epoxy radiation dose [Gy]	$1.0{\cdot}10^{7}$	$9.85{\cdot}10^{6}$	$1.36{\cdot}10^{7}$	$1.15{\cdot}10^{7}$	$1.32{\cdot}10^{8}$	$1.73{\cdot}10^{4}$
Peak fast neutron fluence [cm^{-2}]	$1.0{\cdot}10^{19}$	$5.84{\cdot}10^{17}$	$1.86{\cdot}10^{18}$	$7.54{\cdot}10^{17}$	$8.72{\cdot}10^{18}$	$1.28{\cdot}10^{15}$
Peak damage to Cu stabiliser [dpa]	$5.00{\cdot}10^{-4}$	$2.94{\cdot}10^{-4}$	$5.72{\cdot}10^{-4}$	$4.22{\cdot}10^{-4}$	$5.80{\cdot}10^{-3}$	$8.32{\cdot}10^{-7}$
Peak nuclear heating [Wcm^{-3}]	$1.0{\cdot}10^{-3}$	$2.66{\cdot}10^{-5}$	$2.99{\cdot}10^{-4}$	$1.64{\cdot}10^{-5}$	$1.67{\cdot}10^{-4}$	$2.64{\cdot}10^{-8}$

Tab. 5-25: Radiation loads to the inboard TF-coil as calculated for the four PPCS models assuming an operation time of 40 full power years.

The radial profiles of the total and fast neutron fluxes as calculated for the four PPCS models at the inboard mid-plane are shown in Figs. 5-13 and 5-14. Although the shape of the profiles is rather different due to the different material composition and radial builds, the fast neutron flux is at the same level at the back of the vacuum vessel except for model D including the 4 cm segment gap mentioned above. As a result, the inboard blanket/shield system of all four power plant variants requires a total thickness of about 100 cm to ensure the operation of the super-conducting TF-coils over the full anticipated plant lifetime. More specific results for the four PPCS power plant variants are given in the following sub-sections 5.4.3.2-5.4.3.5.

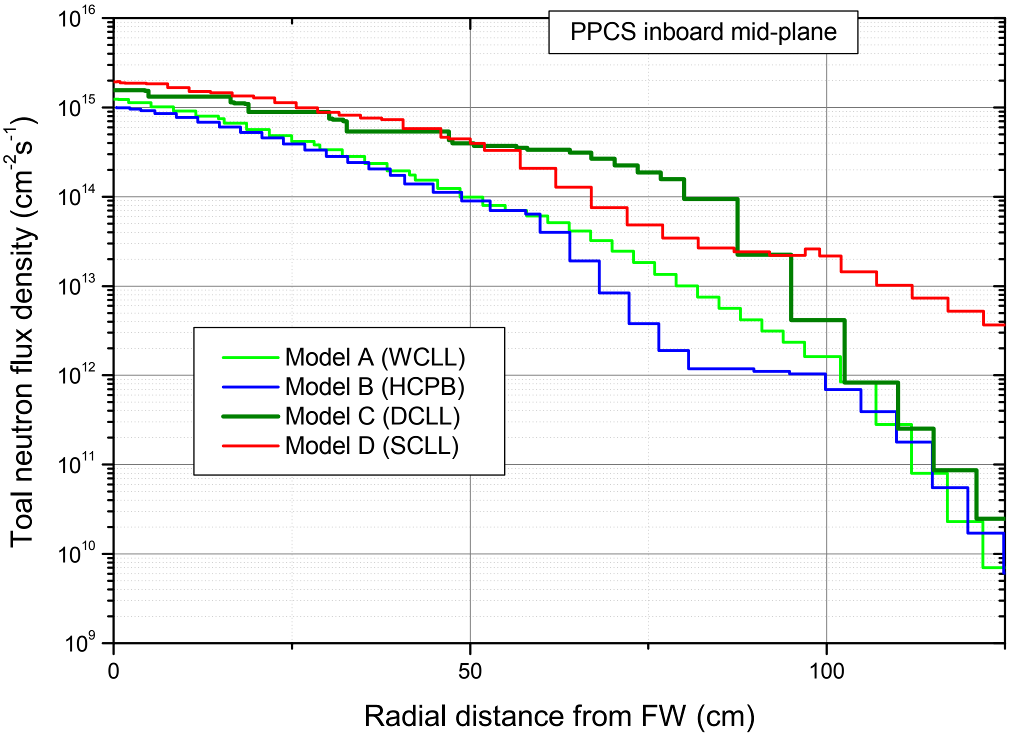


Fig. 5-13: Total neutron flux densities: Radial profiles at the inboard mid-plane of the four PPCS power plants

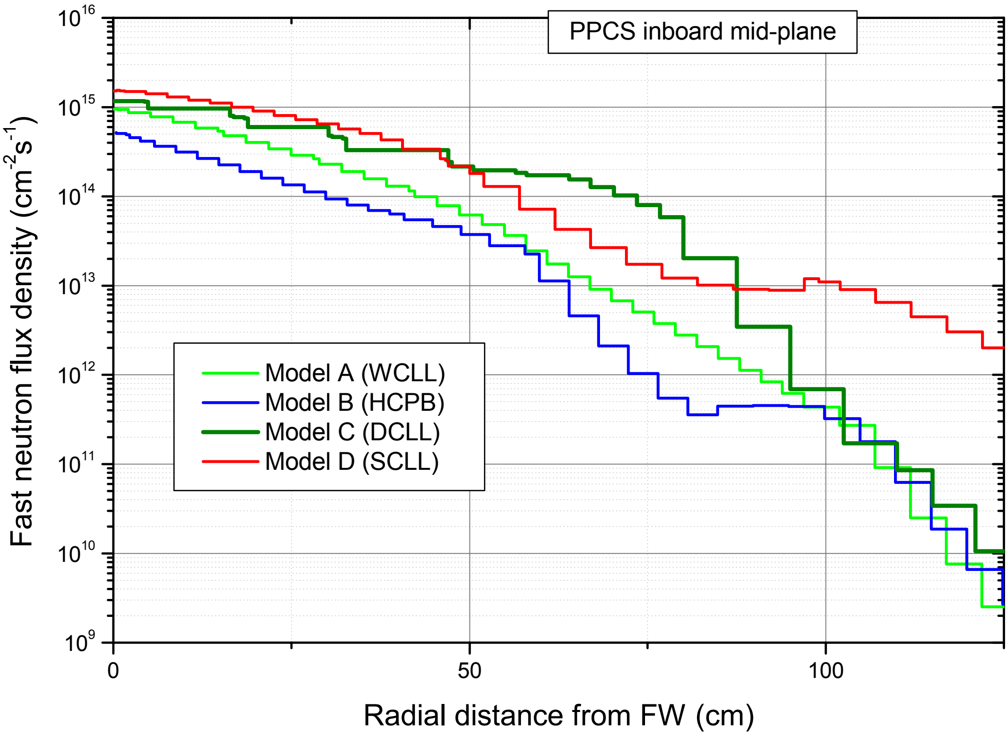


Fig. 5-14: Fast (E>0.1 MeV) neutron flux densities: Radial profiles at the inboard mid-plane of the four PPCS power plants

5.4.3.2 Model A: WCLL blanket

The WCLL basic design [Sar03] assumes a water/steel (Eurofer) shield with a thickness of only 15 cm following the 9 cm thick manifold/shield region, see Table 3-1. The total thickness of the WCLL blanket/shield system thus amounts to 82 cm at the inboard side of the reactor. As noted above, this configuration does not provide sufficient shielding of the TF-coil. Assuming the same material composition as specified for the basic design, about 20 cm shield have to be added to satisfy the shielding requirements, see Table 5-26. The total thickness of the WCLL blanket/shield system at the inboard side then sums up to ≅ 100 cm, analogous to the other blanket variants.

	Design limits	WCLL basic design (cf. Table. 3-1)	WCLL shield thickness increased by 20 cm
Integral radiation dose in the insulator (Epoxy) [Gy]	$1.0{\cdot}10^{7}$	$1.10{\cdot}10^{8}$	$9.85{\cdot}10^{6}$
Peak fast neutron fluence in the Nb_3Sn super conductor [cm^{-2}]	$1.0 \cdot 10^{19}$	$1.02{\cdot}10^{19}$	$5.84{\cdot}10^{17}$
Peak displacement damage to Cu stabiliser [dpa]	$5.00{\cdot}10^{-4}$	$4.86{\cdot}10^{-3}$	$2.94{\cdot}10^{-4}$
Peak nuclear heating in winding pack [Wcm^{-3}]	$1.0{\cdot}10^{-3}$	$1.38{\cdot}10^{-4}$	$2.66{\cdot}10^{-5}$

Tab. 5-26: PPCS model A: Radiation loads to the inboard TF-coil calculated for the WCLL reference design and a variant with increased shield thickness (40 full power years operation time).

5.4.3.3 Model B: HCPB blanket

The HCPB design employs a 17 cm thick high temperature (HT) shield attached to the back of the blanket, cf. Table 3-2. The HT shield is made of Eurofer steel and will be replaced together with the blanket module. The following low temperature (LT) shield is designed as lifetime component attached to the vacuum vessel. The thickness of the LT shield amounts to 25 cm at the inboard side. The reference design as shown in Table 3-2 assumes a LT shield made of Eurofer steel with a large volume fraction of ZrH_2 for the neutron moderation. The LT shield is backed by the manifold region with a rather low steel fraction. In total, the thickness of blanket/shield system at the inboard side sums up to 100 cm (see Table 3-2). As shown in Table 5-25, the required radiation design limits can be met with the material build of Table 3-2. Thus the use of water can be avoided for PPCS model B employing the HCPB blanket. The large ZrH_2 volume fraction, however, can be reduced to 20% without affecting the shielding efficiency significantly, see Fig. 5-15. As an attractive alternative, WC could be employed as shielding material thus avoiding the use of any hydrogenous material for model B (Fig. 5-15).

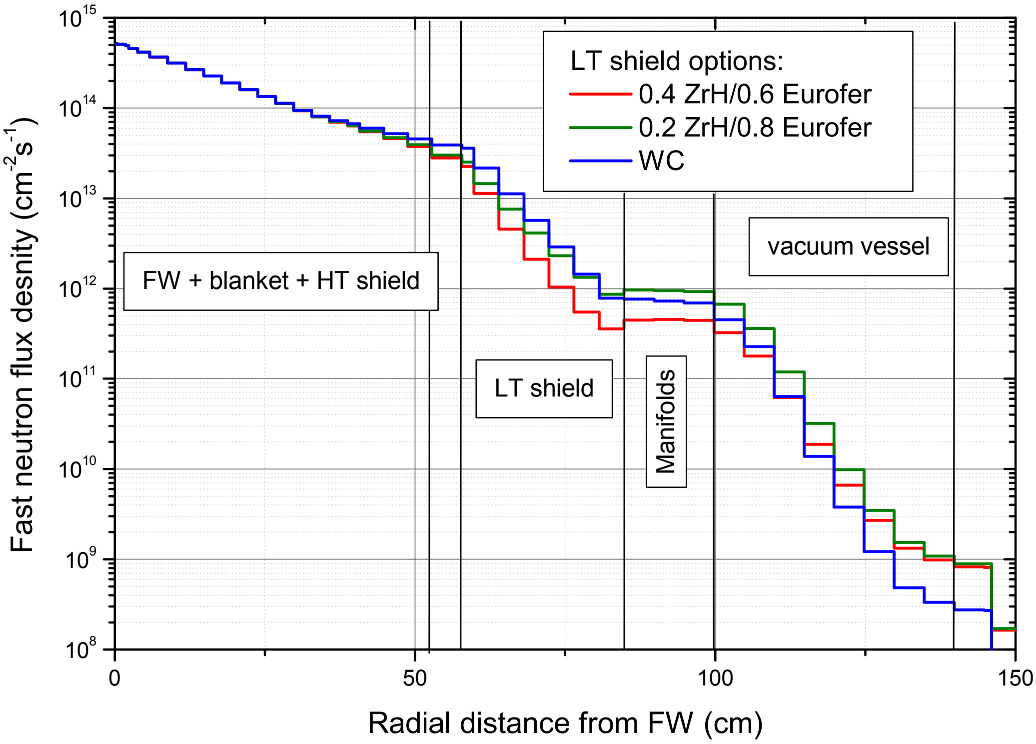


Fig. 5-15: PPCS model B: Fast (E>0.1 MeV) neutron flux profiles at the inboard mid-plane for different shield variants of the HCPB blanket.

5.4.3.4 Model C: DCLL blanket

The DCLL reference design assumes a 30 cm thick LT shield consisting of 40% water and 60% Eurofer. The 13 cm thick HT shield is again made of pure Eurofer and is part of the replaceable blanket module. This results in a total thickness of the DCLL blanket/shield system of 110 cm which is necessary to meet the design radiation requirements at the inboard side of the reactor. The use of water can be avoided in the DCLL blanket/shield system by employing tungsten carbide as shielding material in the HT and the LT shields as well as filler in the vacuum vessel, see Table 5-27 for the radiation loads. Fig. 5-16 shows the radial profiles of the total and fast neutron fluxes as calculated at the inboard mid-plane of the reactor. for the two considered shield options. It is noted that the WC shield option provides a stronger attenuation of the neutron radiation although a thickness of only 20 cm was assumed for the LT shield.

	Design limits	DCLL reference design[a)]	WC shield option[b)]
Integral radiation dose in insulator (Epoxy) [Gy]	$1.0{\cdot}10^{7}$	$1.15{\cdot}10^{7}$	$8.45{\cdot}10^{6}$
Peak fast neutron fluence (E>0.1 MeV) in the NB_3Sn superconductor [cm^{-2}]	$1.0\cdot10^{19}$	$7.54{\cdot}10^{17}$	$2.04{\cdot}10^{17}$
Peak displacement damage to copper stabiliser [dpa]	$5.00{\cdot}10^{-4}$	$4.22{\cdot}10^{-4}$	$9.77{\cdot}10^{-5}$
Peak nuclear heating in winding pack [Wcm^{-3}]	$1.0{\cdot}10^{-3}$	$1.64{\cdot}10^{-5}$	$8.27{\cdot}10^{-6}$

[a)] *Water in LT shield and VV, see Table 3-3*

[b)] *0.1 He/0.2Eurofer/0.7 WC in HT shield, 0.1 He/0.2 borated steel/0.7 WC in LT shield and VV*

Tab. 5-27: PPCS model C (DCLL): Radiation loads to the inboard TF-coil calculated for the DCLL reference design and a WC shield variant

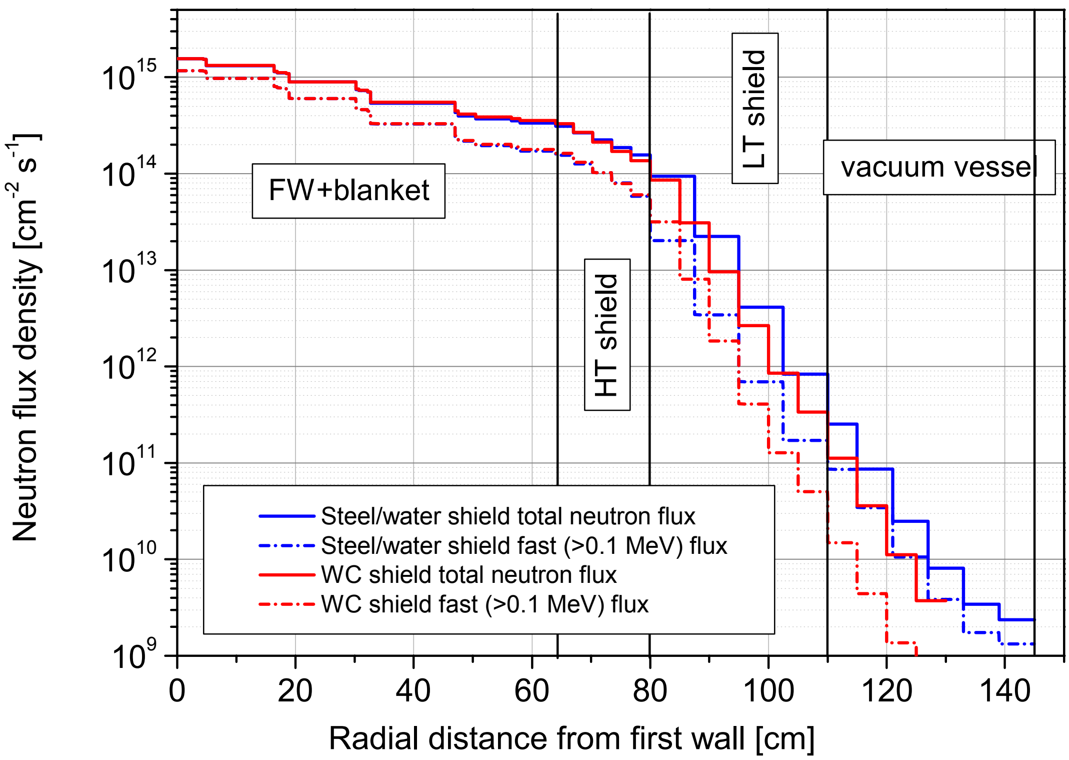


Fig. 5-16: PPCS model C: Radial neutron flux profiles at the inboard mid-plane for different shield variants of the DCLL blanket

5.4.3.5 Model D: SCLL blanket

The low shielding performance of the SiC based SCLL blanket necessitated to investigate a number of different shield options. Table 5-28 shows an overview of the material compositions considered for the HT and LT shields and the vacuum vessel. The total thickness assumed for the SCLL blanket/shield system amounts to 95 and 138 cm, inboard and outboard, respectively. The thickness of the HT and the LT shield at the inboard side of the reactor is at 30 and 35 cm, respectively. The 4 cm wide segment gap was shown to enhance the radiation penetrating the vacuum vessel by about four orders of magnitude (Fig. 5-17). The same is true for the radiation loads to the TF-coil (Table 5-29). It is noted that sufficient shielding cannot be provided for the case of a 4 cm wide segment gap with the assumed shield thickness. This is even true for the most efficient material combination, i. e. WC with a hydride such as ZrH_2. Nevertheless the WC option (with no water nor hydrides) was adopted for the SCLL reference design as there is enough room to optimize the segment gap design and any use of hydrogenous material can be avoided.

Radial profiles of the fast and low energy neutron fluxes are shown in Figs. 5- 18, 5-19 for the considered WC and B_4C HT shield variants: It is revealed that WC is the more efficient shielding material whilst it also acts as a better neutron reflector in the HT–shield. Due to strong inelastic scattering reactions on W, the low energy (W<0.1 MeV) flux in the HT-shield and the blanket back is greater than for the B_4C variant. This also affects the tritium breeding significantly (see Table 5-5 above). In addition, the power generation in the HT shield is larger for WC than for B_4C.

Variant	HT shield	LT shield	Vacuum vessel
WC + ZrH_2	0.1 Pb-17Li, 0.1 SiC, 0.8WC	0.1 He, 0.1 steel, 0.8×(1/3ZrH_2+2/3 WC)	0.1 He, 0.1 steel, 0.8×(1/3ZrH_2+2/3 WC)
WC + H_2O	0.1 Pb-17Li, 0.1 SiC, 0.8WC	0.4 H_2O, 0.6 borated steel	0.4 H_2O, 0.6 borated steel
B_4C + ZrH_2	0.2 He, 0.8 B_4C	0.1 He, 0.9×(0.4 ZrH_2 + 0.6 borated steel)	0.1 He, 0.9×(0.4 ZrH_2 + 0.6 borated steel)
B_4C + WC (no water, no hydrides)	0.1 Pb-17Li, 0.1 SiC, 0.8 B_4C	0.2 He, 0.2 borated steel, 0.6 WC	0.2 He, 0.2 borated steel, 0.6 WC
WC (no water, no hydrides)	0.1 Pb-17Li, 0.1 SiC, 0.8WC	0.2 He, 0.2 borated steel, 0.6 WC	0.2 He, 0.2 borated steel, 0.6 WC

Tab. 5-28: Model D (SCLL): Material compositions considered for the shield system

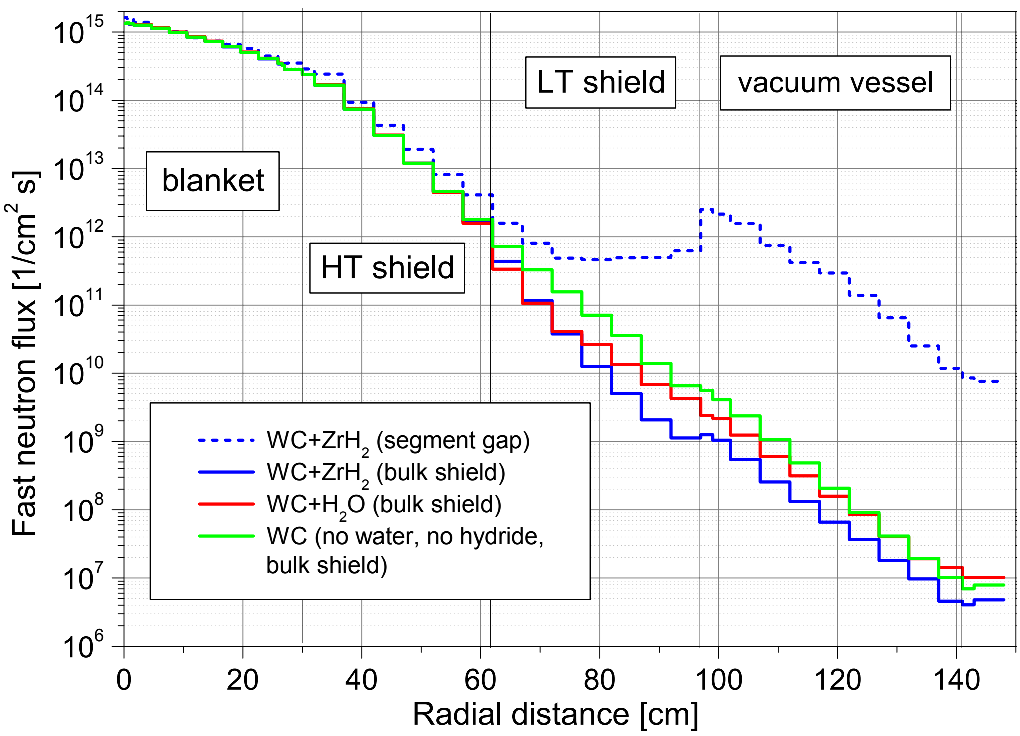


Fig. 5-17: PPCS model D (SCLL): Fast (E>0.1 MeV) neutron flux profiles at the inboard mid-plane for different WC shield variants with and without segment gap.

	Design limits	$WC+ZrH_2$	Bulk shield (no segment gaps)				
			$WC+ZrH_2$	$WC+H_2O$	B_4C+ZrH_2	B_4C+WC	WC
Integral radiation dose in insulator [Gy]	$1.0 \cdot 10^7$	$1.32 \cdot 10^8$	$1.73 \cdot 10^4$	$1.48 \cdot 10^5$	$7.78 \cdot 10^5$	$2.32 \cdot 10^6$	$3.28 \cdot 10^5$
Peak fast neutron fluence in superconductor [cm^{-2}]	$1.0 \cdot 10^{19}$	$8.72 \cdot 10^{18}$	$1.28 \cdot 10^{15}$	$1.25 \cdot 10^{16}$	$4.13 \cdot 10^{16}$	$9.60 \cdot 10^{16}$	$1.37 \cdot 10^{16}$
Peak displacement damage to copper stabiliser [dpa]	$5.00 \cdot 10^{-4}$	$5.80 \cdot 10^{-3}$	$8.32 \cdot 10^{-7}$	$7.94 \cdot 10^{-6}$	$4.15 \cdot 10^{-5}$	$4.82 \cdot 10^{-5}$	$8.67 \cdot 10^{-6}$
Peak nuclear heating in winding pack [Wcm^{-3}]	$1.0 \cdot 10^{-3}$	$1.67 \cdot 10^{-4}$	$2.64 \cdot 10^{-8}$	$1.79 \cdot 10^{-7}$	$6.24 \cdot 10^{-6}$	$3.02 \cdot 10^{-6}$	$3.70 \cdot 10^{-7}$

Tab. 5-29: PPCS model D: Radiation loads to the inboard TF-coil calculated for the shield variants of the SCLL blanket.

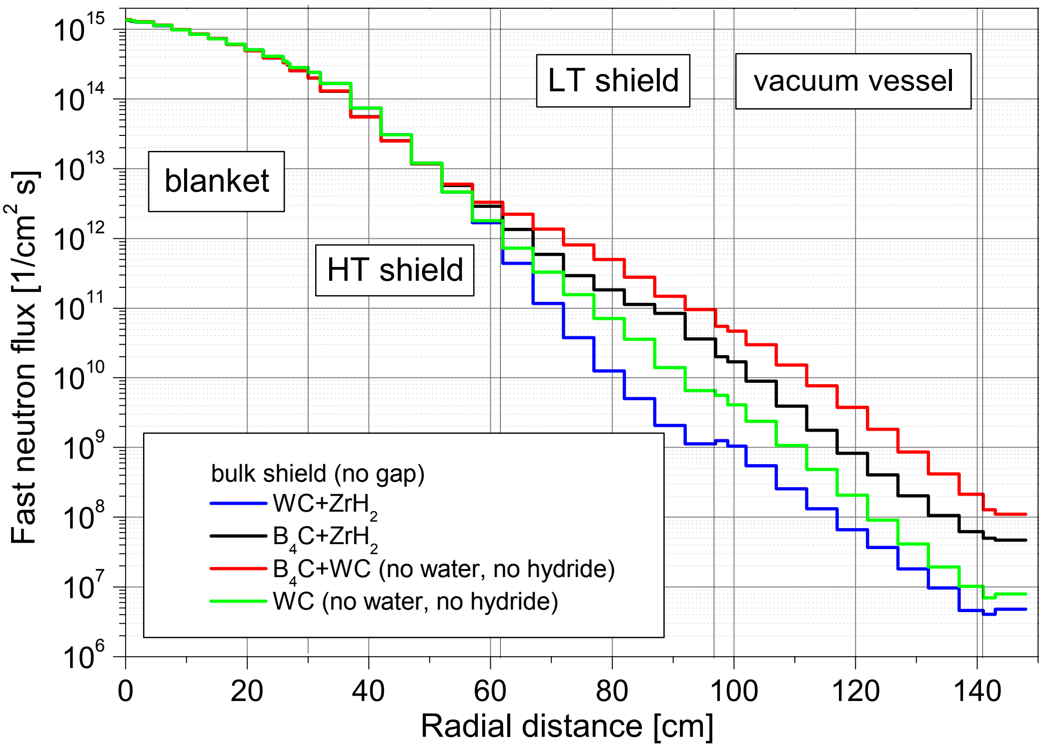


Fig. 5-18: PPCS model D (SCLL): Fast (E>0.1 MeV) neutron flux profiles at the inboard mid-plane for WC and B_4C shield variants without segment gap.

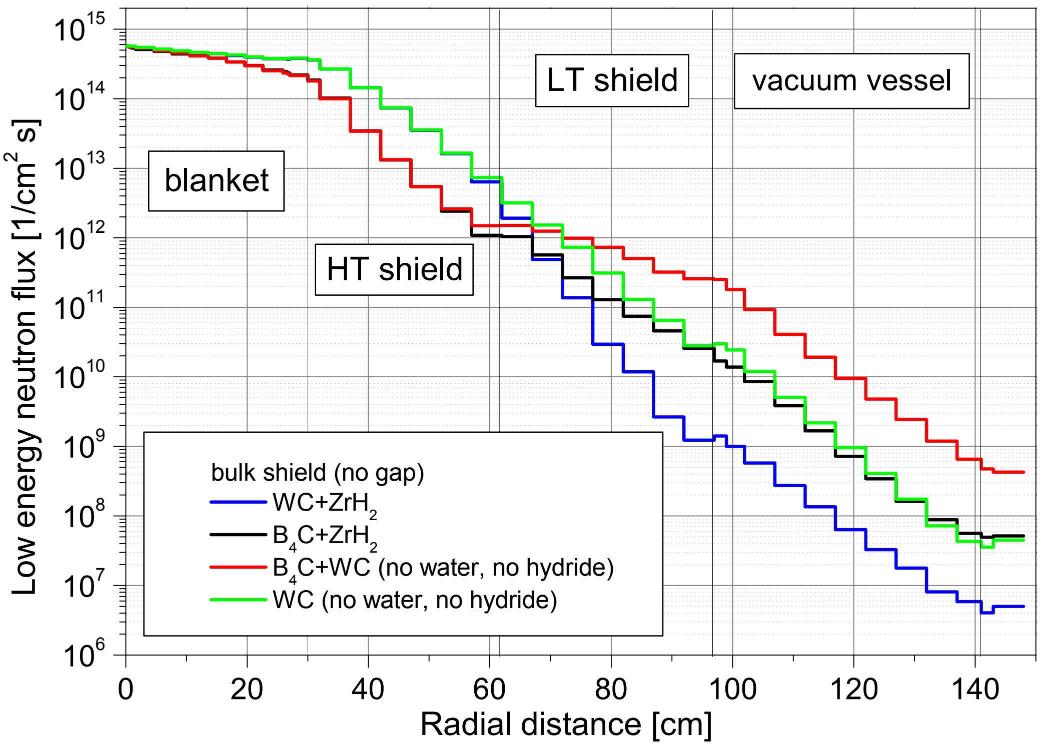


Fig. 5-19: PPCS model D (SCLL): Low energy (E<0.1 MeV) neutron flux profiles at the inboard mid-plane for WC and B_4C shield variants without segment gap

6 Conclusions

As part of the European Power Plant Conceptual Study (PPCS), comprehensive neutronic design analyses have been performed for both the near term and advanced power plant models. A water cooled lithium-lead and a helium cooled pebble bed blanket were employed with the near term models, while a dual coolant lithium lead and a self-cooled lithium-lead blanket with SiC_f/SiC composite as structural material were considered for the advanced power plant models. Main issues of the neutronics analyses were the assessment of the tritium breeding capability, the evaluation of the nuclear power generation and its spatial distribution, and the assessment and optimisation of the shielding performance. The analyses were based on three-dimensional Monte Carlo calculations using suitable torus sector models developed for the different PPCS power plant variants.

Tritium self-sufficiency, including a sufficient high uncertainty margin, can be provided for PPCS models B, C and D employing the HCPB, DCLL and SCLL blanket concepts, respectively. As for PPCS model A, based on the water cooled lithium-lead (WCLL) blanket, the uncertainty margin of the current design is too low to compensate for TBR-losses due to blanket ports not accounted for in the TBR-calculation. An optimised adaptation of the WCLL blanket concept to the PPCS boundary conditions should result, however, in a sufficiently high TBR uncertainty margin. This could be achieved, for instance, by reducing the size of the large segment gaps between the top inboard and outboard WCLL blanket modules.

With regard to the nuclear power generation, the HCPB blanket was shown to provide the highest energy multiplication. This is mainly due to the high neutron multiplication because of the high mass inventory of the Beryllium neutron multiplier. The SCLL blanket concept, on the other hand, shows the lowest energy multiplication due to the use of SiC as structural material. Nevertheless the thermal efficiency of the PPCS plant model D, employing the SCLL blanket, is highest due to the high coolant exit temperatures up to 1100 °C and the power conversion system considered. Accordingly, PPCS model A (WCLL blanket) shows the lowest thermal efficiency resulting in a rather large plant size required for the same net electric power as the other plants.

The shielding performance of the blanket concepts was assessed for the inboard mid-plane of the PPCS power plants where shielding is most crucial. A total thickness of about 100 cm is required for the inboard blanket/shielding system to ensure the operation of the superconducting TF-coils over the full anticipated plant lifetime of 40 full power years. As for model A, this means an increase of the shield thickness by about 20 cm with regard to the present design. A simple shield composition based on steel and water is sufficient for the WCLL and DCLL blanket concepts. A more efficient shielding material such as tungsten carbide (WC) is required for the SCLL blanket showing an inherent low shield efficiency. WC was also shown to provide sufficient shielding for the HCPB blanket where the use of water is to be avoided to prevent any possible chemical reaction with the hot Beryllium. WC, in principle, can be applied with all blanket concepts considered in the PPCS study. When cooled by helium gas, the use of any hydrogenous material thus can be avoided for the power plant models B, C and D.

7 References

[Bri00] J.F. Briesmeister (ed.), MCNP - A General Monte Carlo N-Particle Transport Code, Version 4C, Los Alamos National Laboratory, Report LA-13709-M, April 2000

[FDR01] ITER Final Design Report, Materials Assessment Report, G74 MA 1001-07-11 W0.2, July 2001

[Gia02] L. Giancarli et al, In-vessel Component Designs for a Self-Cooled Lithium-Lead Fusion Reactor, 22nd Symposium Fusion Technology (SOFT-22), Helsinki, September 9-13, 2002

[GDR95] ITER General Design Requirements Document (GDRD), S10 GDRD 2 95-02-10 F1.0, June 1995

[Hen96] T. C. Hender, P. J. Knight, I. Cook, The PROCESS code, UKAEA FUS 333 (1996)

[Her02] S. Hermsmeyer et al, Lay-out of the He-cooled solid breeder plant model in the European power plant conceptual study, 22nd Symposium Fusion Technology (SOFT-22), Helsinki, September 9-13, 2002

[Her03] S. Hermsmeyer (Comp.), Conceptual Design of the Helium Cooled Pebble Bed Blanket Plant Model in the Frame of the EU Power Plant Conceptual Study (TW2-TRP-PPCS2), Final Report, FZKA-6814 (in preparation).

[Lou01] M. Loughlin, UKAEA Culham, Neutron source for initial design phase of the Power Plant Conceptual Study, personal communication, September 2001

[Lou02] M. Loughlin, UKAEA Culham, Neutron source for intermediate and advanced models of Power Plant Conceptual Study, personal communication, April 2002.

[Mar02] G. Marbach, I. Cook, D. Maisonnier, The EU power plant conceptual study, Fusion Engineering and Design 63-64(2002), 1-9.

[Nor02] P. Norajitra et al, Conceptual Design of the Dual-Coolant Blanket in the Frame of the EU Power Plant Conceptual Study, 22nd Symposium Fusion Technology (SOFT-22), Helsinki, September 9-13, 2002

[Nor03] P. Norajitra et al, Conceptual Design of the Dual-Coolant Blanket in the Frame of the EU Power Plant Conceptual Study (TW2-TRP-PPCS12D9), Final Report, FZKA-6780 (in preparation).

[Sar02] P. Sardain et al, Power Plant Conceptual Study - WCLL Concept, 22nd Symposium on Fusion Technology (SOFT-22), Helsinki, September 9-13, 2002

[Sar03] P. Sardain et al, Power Plant Conceptual Study - Model A: Water Cooled Lithium Lead Concept, CEA report (to appear)