Forschungszentrum Karlsruhe

Technik und Umwelt

Wissenschaftliche Berichte

FZKA 6402

Advanced Helium Cooled Pebble Bed Blanket

Task PPA 2.6 - Final Report

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Forschungszentrum Karlsruhe GmbH, Karlsruhe
2000

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Abstract

The Helium Cooled Pebble Bed blanket concept developed in the frame of the EPB-programme is based on the use of low activation ferritic/martensitic steel (EUROFER-97) as structural material. As the maximum allowable temperature of this steel is 550°C, the coolant helium temperature can not exceed about 500°C, resulting in a relatively low thermal efficiency of the power generation system. The use of a ceramic structural material like SiC_f/SiC with a maximum allowable temperature of about 1200°C allows to increase the maximum helium temperatures in the blanket, with the possibility to adopt more efficient power conversion systems. SIC_f/SiC provides some other attractive features from the neutronic point of view (low neutron absorber in comparison to EUROFER) and safety (low afterheat).

To take full advantage of the potential of this structural material, a new blanket design has been proposed. The pebble beds have been arranged in parallel to the first wall; by this configuration it was possible to reduce the required amount of beryllium, to improve the tritium breeding capability and to increase the allowable neutron wall loading. Finally, the adopted flow scheme results in a decisive reduction of the coolant pressure drop.

On the basis of this design thermo-mechanical, thermo-hydraulic and neutronic calculations have been performed to optimise the design parameters (number and thickness of the beds, ^{6}Li enrichment, helium temperatures and pressure, etc). An assessment of the limitation of this concept in terms of maximum neutron wall loading, surface heating, achievable tritium breeding ratio, thermal efficiency in the power conversion system and pumping power for the blanket cooling loops have been performed.

Fortgeschrittenes Heliumgekühltes Festoffblanket

Kurzfassung

Das im Rahmen des Europäischen Blanketprogrammes enwickelte Heliumgekühlte Festoffblanketkonzept basiert auf der Benutzung eines ferritisch-martensitischen Stahls (EUROFER-97) als Strukturmaterial. Da die maximal zulässige Temperatur dieses Stahls 550°C beträgt, kann die Temperatur des Helium Kühlmittels circa 500°C nicht überschreiten, was zu einem relativ niedrigen Wirkungsgrad des Energiekonversionssystems führt. Die Verwendung eines keramischen Strukturmaterials wie SiC_f/SiC mit einer maximal zulässigen Temperatur von circa 1200°C erlaubt einen Anstieg der maximalen Heliumtemperatur im Blanket mit der Möglichkeit effizientere Energiekonversionssysteme zu benutzen. Außerdem hat SiC_f/SiC einige andere attraktive Eigenschaften in Bezug auf die Neutronik (niedrigere Neutronenabsorbtion als EUROFER) und die Sicherheit (niedrigere Nachwärme).

Um alle Vorteile dieses Strukturmaterials nutzen zu können, wurde ein neues Blanketdesign vorgeschlagen. Die Kugelschüttungen wurden parallel zur ersten Wand angeordnet. Durch diese Konfiguration war es möglich, die erforderliche Menge an Beryllium zu reduzieren, die Fähigkeit, Tritium zu erzeugen, zu steigern und die zulässige Belastung an der Wand durch Neutronen zu erhöhen. Schließlich führt das angewandte Flußschema zu einer deutlichen Reduktion der Druckverluste im Kühlmittel.

Auf der Grundlage dieses Entwurfs wurden themohydraulische, thermomechanische und neutronische Berechnungen durchgeführt, um Parameter wie Anzahl und Dicke der Kugelschüttungen, ^{6}Li-Anreicherung, Temperaturen und Drücke des Kühlmittels, etc., zu optimieren. Eine Untersuchung zur Ermittlung der Leistungsgrenze dieses Konzepts bezüglich der maximal zulässigen Belastung der ersten Wand durch Neutronen und durch Erhitzung der Oberfläche, Tritiumerzeugung, thermischem Wirkungsgrad des Energiekonversionssystems und Pumpleistung des Blanketkühlkreises wurde durchgeführt.

List of Abbreviations and Acronyms

A-HCPB	Advanced Helium Cooled Pebble Bed
BZ	Breeding Zone
CEA	Commissariat a l'Énergie Atomique
DEMO	Demonstration (reactor)
FW	First Wall
HCPB	Helium Cooled Pebble Bed
MCNP	Monte Carlo Neutron Photon
OD	outer diameter
PPA	Power Plant Availability
R&D	Research and Development
SEAFP	Safety and Environmental Aspects of Fusion Power
SiC_f/SiC	Silicon Carbide Composite
TBR	Tritium Breeding Ratio
UKAEA	United Kingdom Atomic Energy Authority

Table of Contents

1 Introduction

In the framework of the European Power Plant Study planned to start in 2000, preparatory work based on an Advanced Helium Cooled Pebble Bed (A-HCPB) Blanket concept has been carried out at the Forschungszentrum Karlsruhe in cooperation with CEA (Subtask PPA2.6.2, " SiC_f/SiC related issues of the Advanced HCPB concept") as a proposal for further blanket development.

The Helium Cooled Pebble Bed (HCPB) Blanket [1] is one of the blanket concepts considered in the European Programme as a DEMO relevant blanket. Starting point of this work was the analysis of the technological design limitations of this concept and the identification of the necessary improvements in view of its inclusion in the European Reactor Study. One of the most important limitations of the DEMO-HCPB concept is the relative low temperature of the coolant helium (250°C inlet and 450°C outlet) leading to gross thermal efficiency of the conversion cycle of about 35%. The coolant temperatures are dictated by the maximum allowable temperature (about 550°C) of the ferritic-martensitic steel (EUROFER-97) used as structural material. The development of advanced materials with higher temperature limits will result in distinct advantages in terms of helium outlet temperature and gross thermal efficiency.

In particular, SiC_f/SiC is a promising candidate [2]. It presents a high strength at temperatures greater than 1000°C, leading to potentially very high temperatures of the coolant with the possibility to adopt power conversion cycles at a high thermodynamic efficiency. Furthermore, its inherently low afterheat generation results in advantages in the area of safety and maintenance. On the other side despite of some improvements reached in this last years important questions such as the degradation of thermal conductivity during irradiation, joining technology with low activation materials and the hermetic sealing remain severe issues for using this material as structural material in fusion applications.

To take full advantage of the potential of this structural material, a new blanket design has been proposed. The pebble beds have been arranged in parallel to the first wall; by this configuration a variation of the ^6Li enrichment and the ratio breeder/multiplier in radial direction is facilitated, reducing the required amount of beryllium, improving the tritium breeding capability and increasing the allowable neutron wall loading by reducing the maximum ^6Li burn-up and the tritium inventory in the beryllium multiplier. Finally, the adopted flow scheme based on tubes arranged in a shell in the first wall and in "meanders" in the breeding zone results in a decisive reduction of the coolant pressure drop.

Helium cooled ceramic breeder blanket concepts with SiC_f/SiC as structural material have been investigated in the last years in US and Japan in the frame of the reactor studies ARIES-I [3] and DREAMS [4] , respectively.

2 Design Description

The Advanced Helium Cooled Pebble Bed Blanket (see Fig.1) exhibits the same basic design features as the European DEMO HCPB blanket, which is based on the use of separate ceramic breeder and beryllium pebble beds placed between cooling plates. The separation between breeder and multiplier in form of a pebble bed is necessary to avoid chemical reactions between Be and ceramic breeder [5] and high tritium inventory in beryllium [6] as it has been observed in irradiation experiments carried out for mixed pebble beds.

The concept of a segment box typical of DEMO with components height about 12 m, is here replaced by applying a poloidal and toroidal segmentation of the blankets, that leads to separated boxes with dimensions of about 1.4 x 0.6 x 0.6 m (toroidal x poloidal x radial).

The boxes in a segment are supported by a common back plate, which allows to remove all the boxes in the segment together (see Fig.2).

As in the HCPB blanket, the advanced concept is cooled by helium at high pressure flowing in tubes first in the first wall (FW) and then in the breeding zone (BZ). Fig. 3 shows the flow scheme used in this concept. For safety reasons the helium flows in two completely separated loops. The FW is constituted by an U-shaped tube shell. This shell is formed by radial-poloidal tubes (16 mm internal diameter in the reference design as shown in Fig.4) tied together for the whole width of the box. The coolant is flowing alternately in opposite directions to make the temperature distribution more uniform.

In the BZ, the flat beryllium and ceramic breeder pebble beds, toroidally-poloidally oriented, are confined and cooled by long tubes in the form of a meander; these meanders are bounded together to form cooling plates (see Fig.5). The coolant helium flows inside these tubes through the BZ from the plasma to the vessel side.

As the BZ tubes have an internal diameter that is the half of the FW tubes (see Fig.4), the outlet helium of each tube of the FW must feed 4 tubes of the "meander" region. The connection between large (in FW) and small (in BZ) tubes is realised in the back region with low neutron flux adopting a brazing joint technique [7]. As the section of a FW tube and the joint 4 BZ tubes have the same flow area, the helium velocity magnitude increases without discontinuity in the tubes from the FW to the BZ. The adopted flow scheme results in relatively low pressure drops.

An independent low pressure helium purge flow is used to extract the produced tritium from the beds. Hence, during normal operation the space inside the box is at the low purge gas pressure. A graphite reflector is used behind the BZ to improve the neutronic performance, to shield more effectively the magnets and to reduce stresses that arise inside the beds due to thermal expansion and swelling.

3 Material properties and design limits

The performances of the proposed concept are very strongly dependent on the properties of the materials and on the design limits used in the assessment. Some of these properties and limits are presently not well known. Table 1 summarises the most important values assumed in this assessment.

The present design considers lithium orthosilicate (Li_4SiO_4) as ceramic breeder. The thermal conductivity of Li_4SiO_4 pebble beds has been measured and correlations are available for the design [8]; the maximum design temperature has been chosen 100 K lower than the phase transition at 1024°C, taking into account uncertainties due to hot spots.

The maximum allowable temperature of beryllium is one of the most important constraints of this design; in fact it limits the maximum temperature of the coolant helium. Swelling, degradation of mechanical properties considerations and safety concerns in case of presence of water in the power generation loops suggest to adopt a design limit of 700°C [9]. The thermal conductivity of binary beryllium beds has been measured in dedicated experiments carried out in Forschungszentrum Karlsruhe leading to maximal values of 25 W m^{-1} K^{-1} [10]. However, these experiments have been carried out at average temperatures in the beryllium bed up to 300°C and at thermal expansion differences between pebble bed and structure below 0.4 %, a value much lower than the one anticipated in the blanket with SiC_f/SiC as structural material (much lower thermal expansion coefficient than steel). It is expected that the thermal conductivity will be considerably higher in the present concept.

As far as SiC_f/SiC is concerned, the values of the thermomechanical properties used in the present assessment have been taken from design value proposed in the TAURO project [11]. Some of these values must be interpreted as minimum requirements for application in fusion technology; typical example of this is the assumed thermal conductivity of 15 W m^{-1} K^{-1} in the thickness without degradation under irradiation. At present, this ambitious goal is far-away from presently achieved values.

The anisotropy of this composite material depending on the orientation of the fibres requires new methods in the design optimisation of the mechanical and thermal behaviour of components [7]. Criteria such as the von Mises stresses are not satisfactory for materials which present different properties along each orthotropic axis. However, conventional stress criteria such as those suggested in the ARIES-I study [3] have been adopted in this assessment for a preliminary dimensioning. A more sophisticated approach has been proposed by CEA [12]; the stress levels are separately evaluated along each orthotrophy axis for each of the basic parts of the blanket and the maximum values of the obtained stresses are then compared with the corresponding rupture limits. Based on this approach, thermomechanical analyses have been carried out for the FW of this concept.

As far as the chemical compatibility of SiC_f/SiC with beryllium and Li_4SiO_4 is concerned, a recent investigation at Forschungszentrum Karlsruhe [13] has shown that the unstable Be-SiC couple is kinetically hindered to form Be_2C and Si at temperature up to 700°C and that SiC and Li_4SiO_4 are compatible, provided fibres are used with a low oxygen content. In fact SiO_2 reacts with Li_4SiO_4 producing lithium-metasilicate.

4 Calculations for the reference design.

On the basis of the core design described in Section 2, neutronic and thermo-hydraulic calculations have been performed to optimise the design parameters to reach the self-sufficient condition for the tritium breeding ratio with temperatures in the bed below the design limits. The necessary boundary conditions (reactor dimensions, neutron source distribution, surface heating, etc.) have been taken from the reactor model provided by UKAEA Culham (see Appendix A). However, the power level has been increased by about 25% to have an average neutron wall loading of 2.76 MW/m^2 leading to a peak neutron wall loading of 3.5 MW/m^2 like that assumed in the DEMO reactor.

The results of these calculations are shown in Table 2, where the major parameters of the reference design of the A-HCPB blanket are summarised. Table 3 shows the results of the core optimisation in terms of bed thickness and ^{6}Li-enrichment. The power density and the maximum temperature in the bed and at the interface with the structural material have been calculated for the equatorial region of the outboard blanket, where the neutron wall loading and surface heating have their peak values.

The operational conditions of the coolant are: pressure 8 MPa, inlet and outlet temperature of 350°C and 700°C, respectively. This seems a good compromise (for this level of power density) to achieve the design goals, namely to keep the helium pressure as low as possible, to have acceptable pressure drops in the loops and to assure high thermal efficiency for the conversion power cycle.

4.1 Neutronic design

Neutronic calculations have been performed with the MCNP (Monte Carlo Neutron

Photon) code and nuclear data from the European Fusion File to assess and optimise the breeding performance of the A-HCPB blanket concept and provide the nuclear heating input data for the subsequent thermal-hydraulic calculations.

Based on the reactor parameters and the neutron source distribution of the PPA reactor, a generic 7.5 degree torus sector model has been developed. This model includes the plasma chamber, four poloidal blanket/shield segments and a bottom divertor port with an integrated divertor of the SEAFP-type. The first wall profile has been adapted in an arbitrary way to the plasma boundary contour shape assuming a scrape-off layer of 15 cm at torus mid-plane. According to the MCNP reactor models, the blanket coverage amounts to 82 % with a resulting FW blanket surface of 1187 m^2 (1467 m^2 including the divertor port). The calculational results for the A-HCPB Blanket Concept yield a tritium breeding ratio of 1.09 and an energy multiplication factor of 1.22 for the blankets (1.24 for the whole reactor).

Details of the neutronic calculations are presented in Appendix A. The power density of the beds in the outboard equatorial region are shown in Table 3. These values have been extrapolated from those given in Appendix A to account for the peak neutron wall loading of 3.5 MW/m^2 and for some modifications (e.g. the decrease of the thickness of the Li_4SiO_4 beds) when optimising the design.

4.2 Thermohydraulic design

The temperature distribution in the core, the coolant temperatures, the helium mass flow and the pressure drops in the tubes have been calculated with a steady-state flow calculation (that take into account the helium that flow first in the FW and then in the two different meander paths in the BZ) combined with a 1D conduction calculation scheme for the beds and the plates. The results in terms of maximum bed temperatures and interface temperatures are presented in Table 3. Mass flow and pump losses for the whole blanket system are shown in Table 2. As far the pressure drops are concerned, a value of 59 kPa has been calculated for the pressure drop in the blanket and a value of 110 kPa has been assumed for the remaining loop (main pipes, heat exchanger, etc.).

Furthermore, the influence of the blanket inlet temperature on the blanket design has been investigated. In fact, higher temperatures than 350°C can be required to increase the thermal efficiency of the power conversion system (especially if a gas Brayton cycle is adopted). The results of this assessment are presented in Table 4. The main conclusion is that the inlet temperature can be increased up to 430°C without exceeding the design temperature limits. However, the related pressure drops increase strongly with that temperature. By design modifications the pumping power could be reduced, e.g. by an increase of the BZ tubes diameter of 1mm. A detailed optimisation of this variant was not carried out in this study, but a reduction of the pumping power under 3% is envisaged.

4.3 Thermo-mechanical design

2D thermomechanical calculations have been performed for the FW with the structural program ABAQUS. Fig.6 shows the thermal boundary conditions considered in this assessment and Fig.7 the resulting temperature distribution; a maximum temperature of 913°C in the SiC_f/SiC structure at the plasma side has been calculated. The combined thermal and mechanical von Mises stresses are presented in Fig.8.

As mentioned in Section 3, thermomechanical calculations for the FW have been carried out by CEA using a more suitable approach for composite materials. The results obtained with the CASTEM200 code are presented in [12]. The calculated stresses are lower than the allowable limits of Table 1.

4.4 Power conversion system

The outlet conditions of the blanket coolant influence decisively the attractiveness of the overall efficiency and ,therefore, the economy of electrical power generation. The outlet conditions are mainly restricted by physical, mechanical thermodynamic and hydraulic processes and design conditions. A certain margin for modifications concerning efficiency is possible by selection of the power conversion process and of the properties of the heat transfer medium.

Two power conversion systems are alternatively considered. Both systems are characterised by a complete separation of the blanket cooling circuit and the power generation circuit. The He operating conditions for the blanket cooling circuit are in both cases 700 °C outlet temperature on a pressure level of 8 MPa.

The first system is based on a Rankine process with a He/Water steam generator connecting the two circuits. The resulting overheated live steam conditions are 670 °C and 11 MPa. The main design parameters of the steam generator are listed in Table 5. The Q-T-Diagram for the steam generators is shown in Fig. 9. Supposing one intermediate overheating in the steam power conversion circuit would result in a thermal efficiency of 44.2 %. Taking into account also the main blower power and the generator efficiency this will lead to an overall electrical efficiency of 41.1 %. The principle of the power generation circuit is given in Fig. 10.

A potential for improvements of the plant efficiency exist by transition to an overcritical Rankine process. Calculations show that an electrical overall efficiency of 44.3 % and 47.7 % for the thermal efficiency can be reached.

The second power conversion system is based on a high pressure Brayton process as proposed in [14] . The intermediate heat transfer components are He/He heat exchangers which deliver the fusion energy to the closed Brayton circuit (see Fig.11). A direct Brayton process, without an intermediate heat transfer system, was not taken into account due to the high operation pressure of 18 MPa of the considered cycle. The operating temperature of the secondary cooling He would be 670 °C assuming a constant pinch of 30°C in the He/He heat exchanger. The design data of the heat exchanger are also listed in Table 5.

The pressure drop in the Brayton process has a strong effect on the overall efficiency. Presently it is difficult to give definite numbers. Therefore, two values are considered for the pressure drop ratio ($\Delta p/p$) 0.05 and 0.08. These assumptions would lead to an overall efficiency of 46.5 % and 44.5 %, respectively. The secondary pressure drop for the present example was estimated to be 0.4 MPa. If necessary, also half of the value could be reached.

As the high thermal efficiency of the proposed Brayton cycle is based on a compression ratio of 2, a parametric investigation of the dependence of the thermal efficiency on the compression ratio was performed. As compression ratio and heat exchanger outlet temperature are strictly correlated, an optimisation of the thermal efficiency of the Brayton cycle requires a variation of the blanket inlet temperature. Some considerations are presented in Appendix B. To reach power conversion efficiency of about 46% it will be necessary to reduce the compression ratio to about 2.4, that means to increase the blanket inlet temperature up to 430°C (see Section 4.2).

As far as the whole reactor is concerned, helium cooled divertors harmonise well with this blanket concept, working in the same temperature range [15]. The heat produced in the divertors (~18% of the whole thermal heat) can be removed and integrated in the same heat removal system of the blankets at least with the same thermal efficiency, increasing the net efficiency of the whole plant.

4.5 Maintainability and reliability

At the present stage of the reactor study a detailed reliability analysis can not be performed. A sufficiently extended reliability data collection for components made by SiC_f/SiC is not yet available and will be not presumably available in the near future. However, in the proposed design we have applied some design criteria that should manufacture components with high reliability. In fact, the number of connections between the high pressure coolant tubes have been minimised. The FW shell should be realised in one piece with the connections (brazing joint) only in back part of the box blanket where the neutron flux is lower; also each “meander” should be realised in one or only few pieces with the connections sufficiently far from the FW region.

Furthermore, if a gas cycle in the power conversion system is adopted, the presence of water as coolant can be completely eliminated in the reactor avoiding the possibility of water reactions.

4.6 Dimensions of the breeding zone

Table 6 show a comparison of the thickness of the BZ layers between the HCPB for the DEMO reactor (with 9mm-thick Li_4SiO_4 beds) and the A-HCPB. As the HCPB Blanket has a toroidal-radial arrangement of the beds, the values presented in the table have been calculated for a geometrical equivalent parallel arrangement of the beds.

Due to the parallel arrangement of the beds, the optimised ^{6}Li-enrichment of the ceramic beds and the relatively low neutron absorption of the structural material, the equivalent thickness of the Beryllium and Li_4SiO_4 pebble beds can be reduced considerably. In comparison with the HCPB concept, the advanced one will require about 60% less amount of Beryllium and about 17% less of Li_4SiO_4.

The total thickness of the BZ does not differ very much between the two concepts. In fact, a graphite layer of some 30 cm thickness is necessary in the A-HCPB blanket to accomplish the double function of neutron reflector and neutron shield; because of its reflector function, this layer has been included in the BZ. As for the HCPB concept, an additional steel shield must be added at the back of the BZ to protect sufficiently vacuum vessel and magnets.

5 Performance limitations, lifetime and necessary R&D.

Starting from the reference design, parametric analyses have been carried out to investigate the performance limitations of this concept in terms of helium temperatures, neutron wall load, surface heating and thermal efficiency. The results are summarised in Table 7.

A limit of 700°C for the outlet temperature of the helium is dictated by the design limit assumed for the maximum temperature in the beryllium pebble bed. This dictates also the maximum net electrical efficiency of about 45% (with reference to the power removed by the helium in the blanket) achievable in this concept. Higher temperatures in beryllium pebble beds could be allowed if both water in the power generation system is avoided and can be demonstrated that the proposed design is capable to withstand the high swelling rate in beryllium.

As far as the peak neutron wall loading is concerned, it is possible to find design combinations (with 16 beds and with 300-600°C helium temperatures) able to fulfil the design requirements up to 6 MW/m^2. The margin to increase the surface heating is much less, a value of 0.7 MW/m^2 seems a limitation due to the lower thermal stress parameter of SiC_f/SiC in comparison to the ferritic-martensitic steels.

Lifetime limitations of this concept are not well known; all three materials (beryllium, Li_4SiO_4 and SiC_f/SiC) might limit the lifetime, namely beryllium due to swelling at high temperature, Li_4SiO_4 due to Li-burnup or dpa damages and SiC_f/SiC due to the degradation of its properties under irradiation.

A large amount of R&D work is required especially for SiC_f/SiC in order to have a qualified material for fusion application (see Section 1). Tightness to the coolants (particularly to helium at high pressure) seems the most severe issue. Finally, fabrication issues specific for this design must be addressed, such as the manufactory of long tubes and shells made by tube used in this concept as FW and cooling structure in the BZ. Further R&D work is required also for the breeder ceramic and beryllium to determinate its lifetime limitation during irradiation.

6 Conclusions

An advanced version of the HCPB blanket with SiC_f/SiC as structural material has been presented; the reference design has been optimised for a peak neutron wall loading of 3.5 MW/m^2 and surface heating of 0.6 MW/m^2. A potential advantage of this concept is the possibility to increase the helium temperatures (700°C outlet) achieving a gross thermal efficiency of the power generation system up to 46%. With these helium temperatures, it is also possible to use an high pressure/high efficiency closed gas Brayton cycle as power conversion system avoiding water in the secondary cycle and increasing the safety of the concept. Furthermore, the pebble beds have been arranged in parallel to the first wall with advantages in reducing the required amount of beryllium and improving the neutronic performances. Finally, the adopted flow scheme results in a decisive reduction of the coolant pressure drop.

In principle, the combination of an inert and non activatable coolant such as helium with SiC_f/SiC structures, permits to reach the highest safety and environmental standards in term of accidental release and maintenance operation issues for a fusion power reactor. However, the success of this particular blanket concept and in general for all the blanket concepts using SiC_f/SiC as structural material depends entirely on the future possibility to produce materials with the necessary requirements for fusion application (good thermal conductivity, hermetic sealing, joining technology, stability under irradiation).

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Tables

Table 1: Material properties and design limits used in the assessment of the A-HCPB Blanket

Lithium Orthosilicate Pebble Bed	
Thermal conductivity (T= 500-900°C)	1.0-1.2 W m^{-1} K^{-1}
Max allowable temperature	924 °C
Max. allow. temperature at SiC_f/SiC interface	924 °C
Beryllium Pebble Bed	
Thermal conductivity (T= 500-700°C)	25 W m^{-1} K^{-1}
Max allowable temperature	700 °C
Max. allow. temperature at SiC_f/SiC interface	700 °C
SiC_f/SiC	
Thermal conductivity	15 W m^{-1} K^{-1}
Max allowable temperature	1300 °C
Young Module	200 GPa
Thermal expansion coefficient	$4.0 \cdot 10^{-6}$ K^{-1}
Poisson ratio	0.18
ARIES-I structural criteria: Max. allowable primary stress Max. allowable secondary stress	 140 MPa 190 MPa
CEA structural criteria: Max. allowable tensile stress (in plane) Max. allowable compressive stress (in plane) Max. allowable tensile stress (through the thickness) Max. allowable shear stress (through the thickness)	 145 MPa 500 MPa 110 MPa 45 MPa

Table 2: Main Plant and Blanket Design Data

Overall plant		
Fusion power	[MW]	4500
Neutron power	[MW]	3600
Alpha-particle power	[MW]	900
Energy multiplication	[-]	1.24
Thermal power	[MW]	5364
Blanket system		
Neutron power	[MW]	3276
Alpha-particle power	[MW]	558
Energy multiplication	[-]	1.22
Thermal power	[MW]	4555
Blanket surface	[m^2]	1187
Average neutron wall load	[MW/m^2]	2.76
Max. neutron wall load	[MW/m^2]	3.5
Average surface heat load	[MW/m^2]	0.47
Max. surface heat load	[MW/m^2]	0.61
Coolant		He
- Inlet temperature	[°C]	350
- Outlet temperature	[°C]	700
- Pressure	[MPa]	8
- Mass flow rate	[kg/s]	2503
- Pumping power (η= 0,8)	[MW]	85
Net efficiency of power conversion system (with a thermal efficiency of 0.46)	[-]	44.8
Electrical output	[MW]	2041

Table 3: Results of the neutronic and thermo-hydraulic calculations for the A-HCPB blanket reference case for the equatorial outboard region **.**

Bed No.	**Material (^{6}Li-enrich)**	**Equivalent thickness**	**Power density**	**Max. bed temp.**	**Max. interface temperature (*)**
		[mm]	**[MW/m^3]**	**[°C]**	**[°C]**
1	Be	20	15.5	646	628
2	Li_4SiO_4 (nat.)	8	38.1	893	607
3	Be	30	11.7	657	610
4	Li_4SiO_4 (nat.)	8	30.4	819	592
5	Be	30	8.2	661	630
6	Li_4SiO_4 (15%)	8	31.7	910	675
7	Be	30	5.7	669	650
8	Li_4SiO_4 (35%)	8	28.8	882	671
9	Be	30	3.9	678	665
10	Li_4SiO_4 (70%)	9	19.3	885	707
11	Li_4SiO_4 (70%)	9	14.2	827	696
12	Li_4SiO_4 (70%)	9	21.2	913	717

(*) At the interface between pebble bed and structural material (SiC_f/SiC)

Table 4: Influence of the blanket inlet temperature on the design parameters of the reference design.

Blanket inlet temp.	He mass flow rate	max. Li_4SiO_4 temp.	max. Be temp.	max. interf. Li_4SiO_4 temp.	max. interf. Be temp.	pressure drop
[°C]	[Kg/s]	[°C]	[°C]	[°C]	[°C]	[%]
350 °C	2503	913	678	717	665	1.87
400 °C	2920	918	686	717	672	2.68
430 °C	3249	921	690	718	676	3.45[a)]

[a)] This value can be reduced with an optimisation of the design.

Table 5: Heat exchanger design calculation for a Brayton Process and a corresponding Rankine Process

	Rankine	Brayton
Medium	He/H_2O	He/He
Power [MW]	743	743
No. of units	6	6
primary side inlet temperature [°C]	700	700
primary side outlet temperature [°C]	350	350
primary side pressure [MPa]	8	8
secondary side outlet temperature [°C]	670	670
secondary side inlet temperature [°C]	195	320
secondary side pressure [MPa]	18	18
Tube dimensions OD × s [mm]	25×2	22×2
Heat transfer area [m^2]	4230	18500
Bundle type [(1)]	helical	helical
OD×H [m]	3.1 × 9.4	5.4 × 12.5
Δp primary side [kPa]	95	52
Δp secondary side [kPa]	580	400

(1) Helical tube bundles contain a 0.6 m OD inner cylinder

Table 6: BZ thickness: comparison between HCPB for DEMO and A-HCPB Blanket.

	DEMO-HCPB	**A-HCPB**
Li_4SiO_4 pebble bed thickness [mm]	71	59
Beryllium pebble bed thickness [mm]	357	140
Structural material thickness [mm]	127 (EUROFER)	54 (SiC_f/SiC)
Graphite reflector thickness [mm]	-	300
Total BZ thickness [mm]	**555**	**553 (253)** (*)

(*) The value in bracket give the BZ thickness without the graphite layer.

Table 7: Performance limits for the A-HCPB Blanket.

	Reference design	**Limits**
Max. He temperature	700°C	700°C
Gross thermal efficiency	44.8 %	45 %
Surface heating (peak)	0.6 MWm^{-2}	0.7 MWm^{-2}
Neutron wall load (peak)	3.5 MWm^{-2}	6.0 MWm^{-2}

Figures

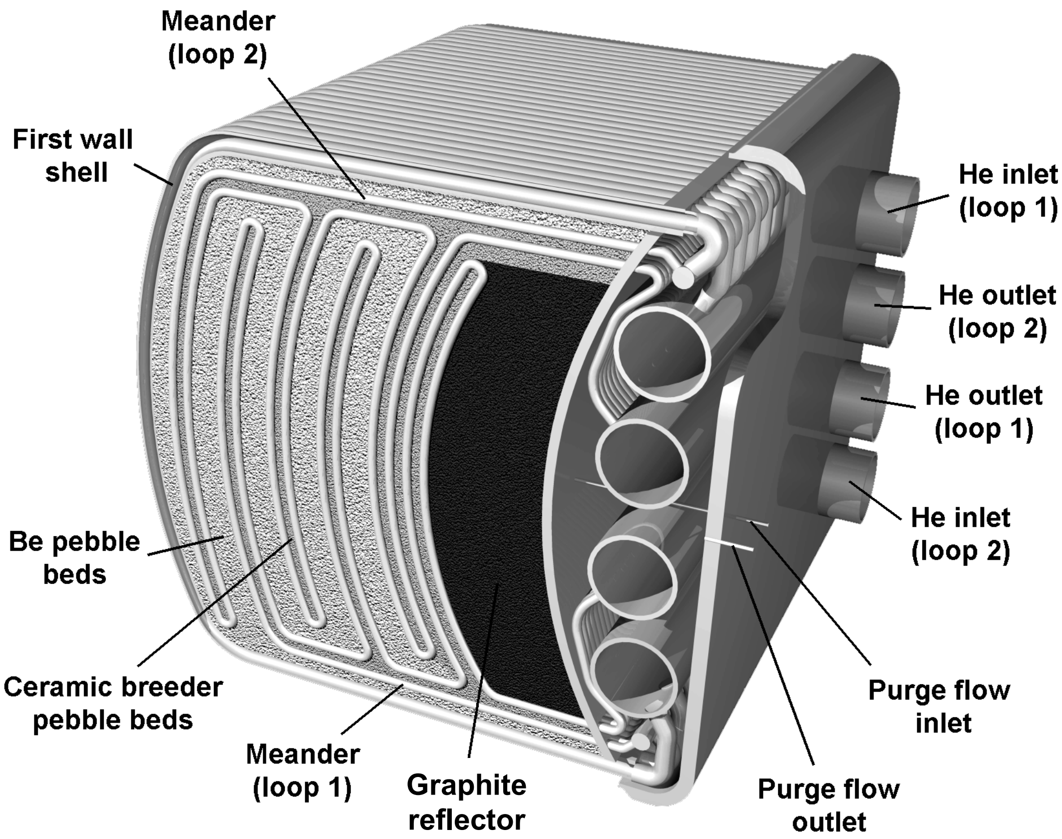


Fig.1: Isometric view of the A-HCPB blanket box.

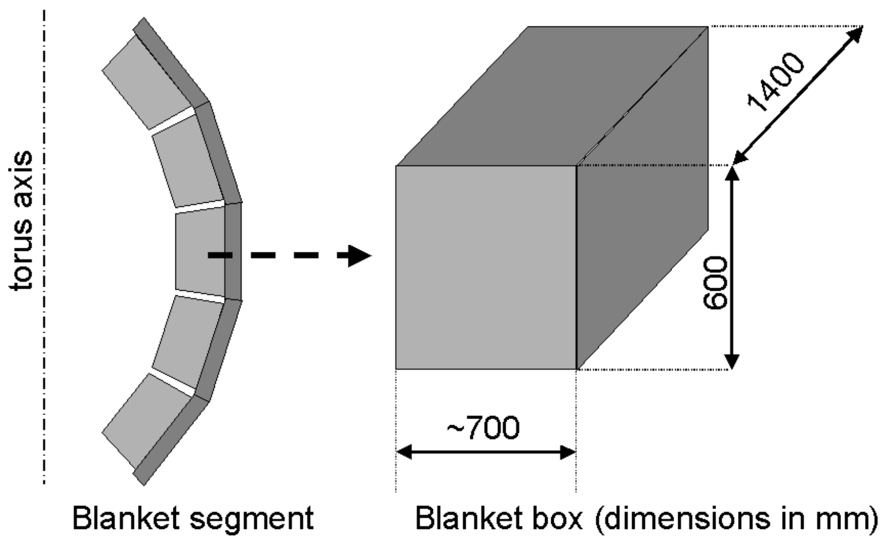


Fig.2: Blanket segment schematic picture and blanket box dimensions

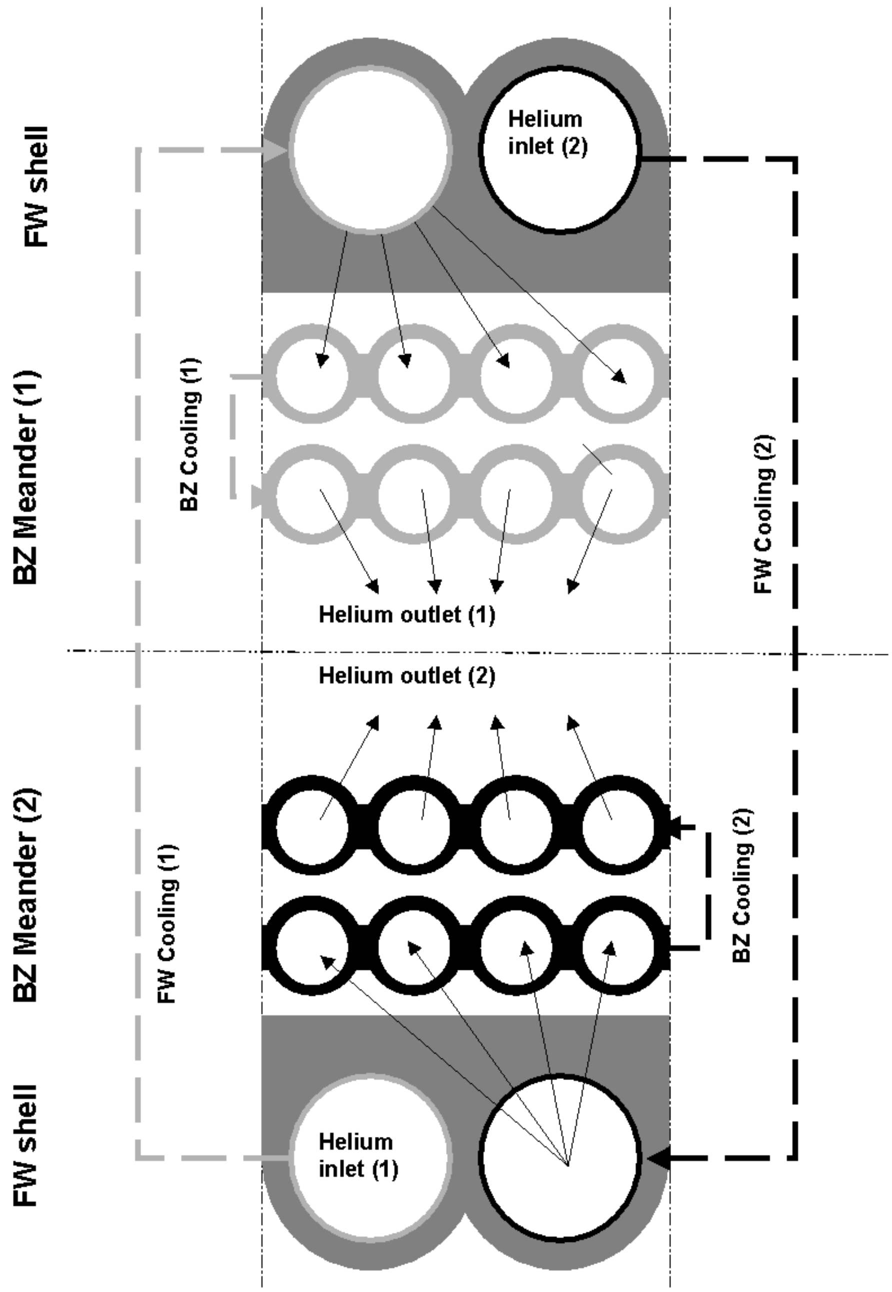


Fig.3: Helium coolant flow scheme.

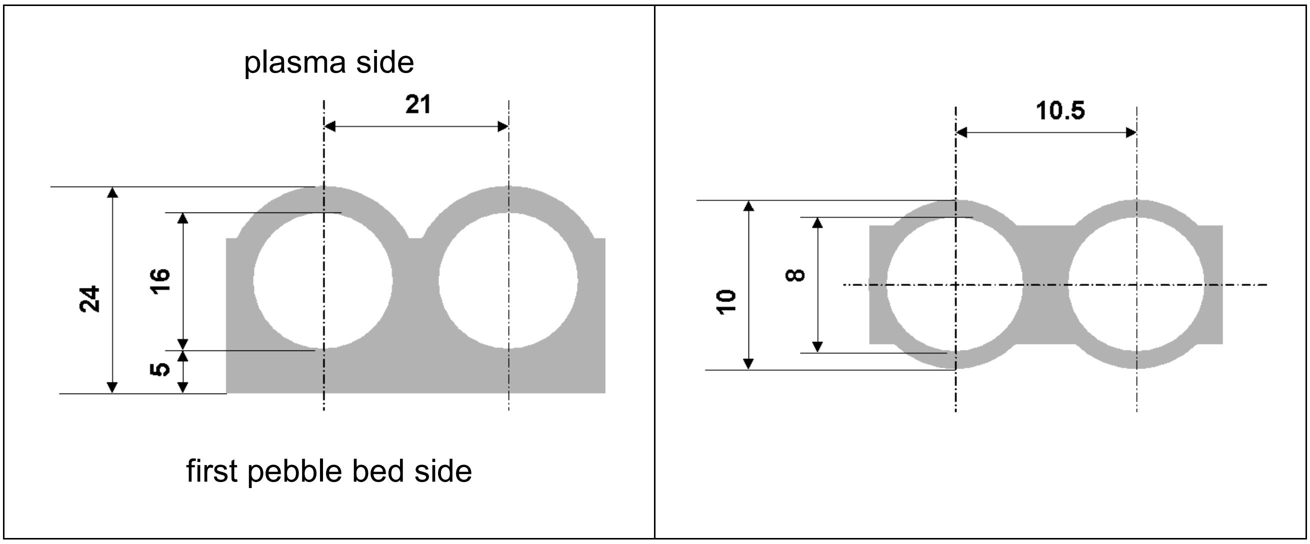


Fig.4: Section (with dimensions in mm) of the FW shell (left) and of the cooling plates in form of a meander (right).

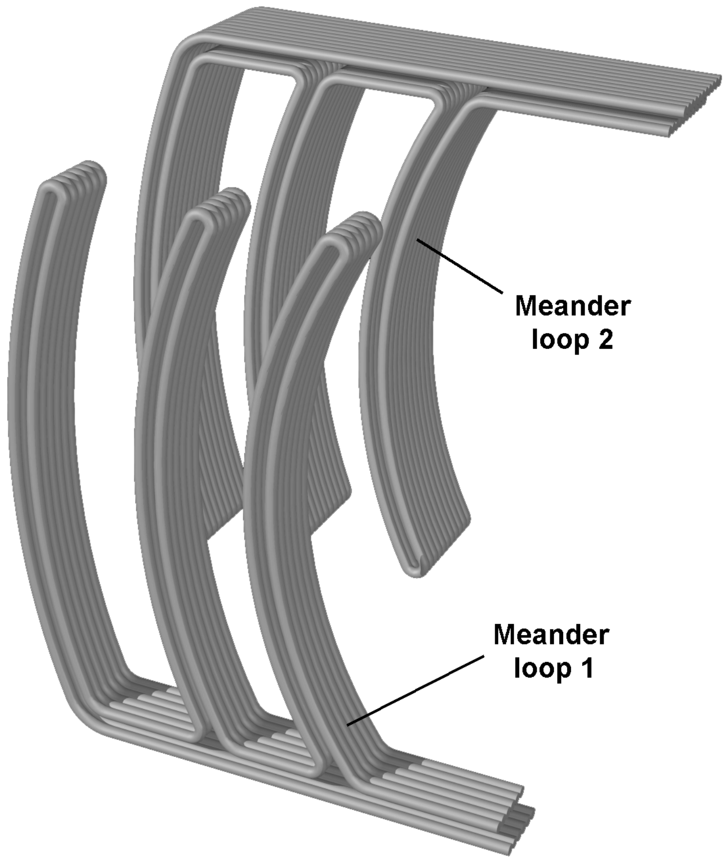


Fig. 5: Details of the tubes in form a meander in the BZ.

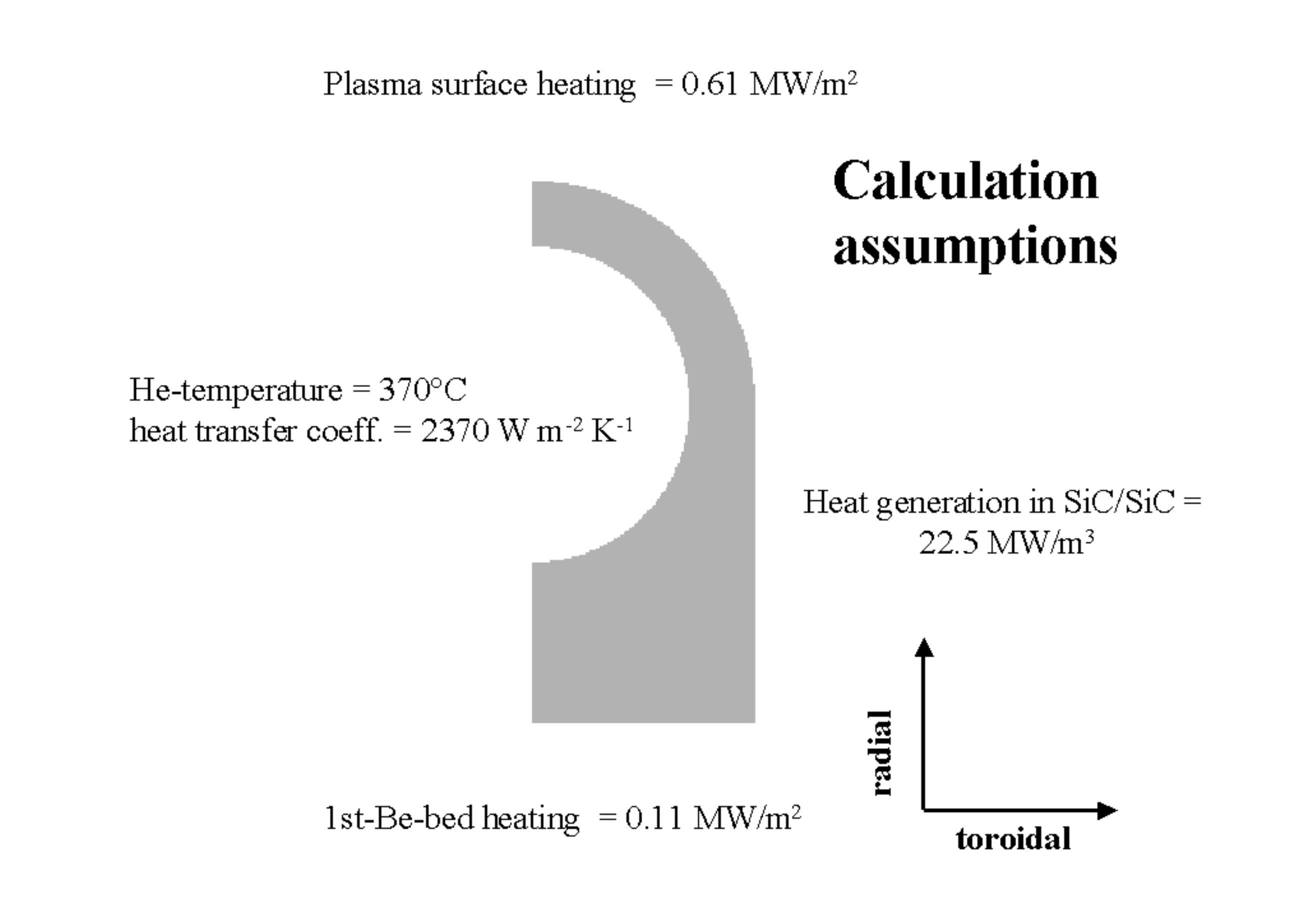


Fig. 6: Boundary conditions used in the temperature calculation of the FW.

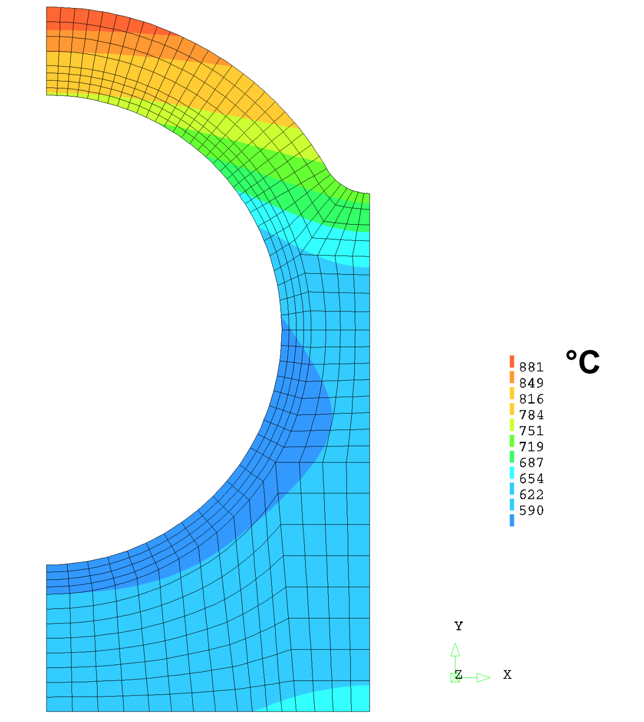


Fig. 7: Temperature distribution in the FW. A maximum temperature of 913°C has been calculated at the plasma side.

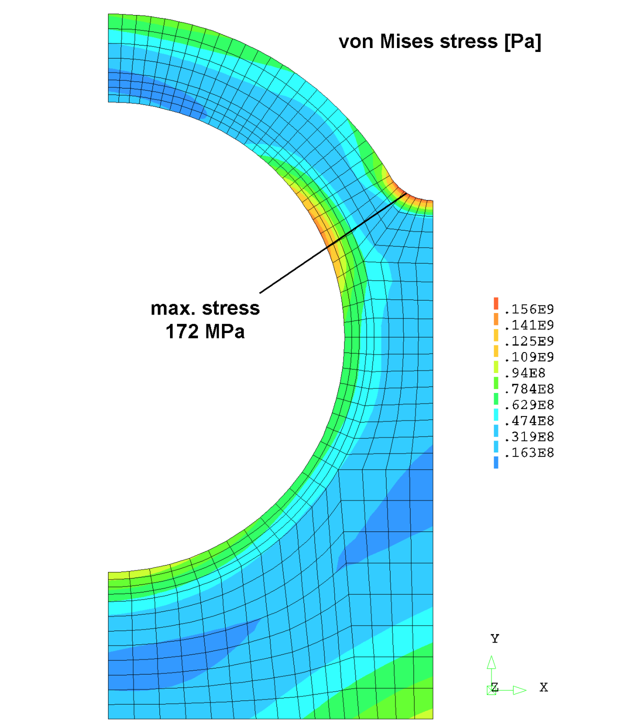


Fig. 8: Von Mises stress distribution in the FW.

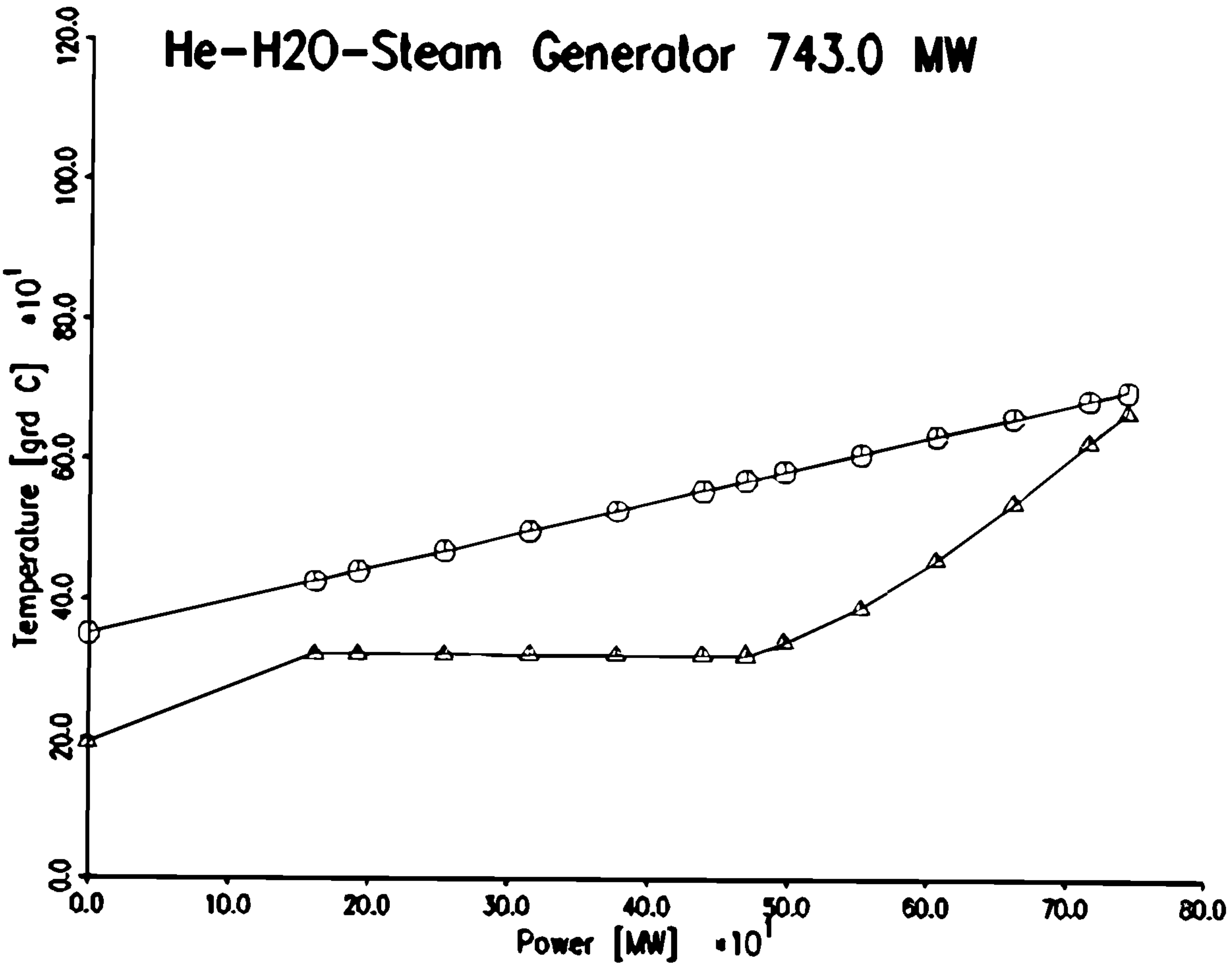
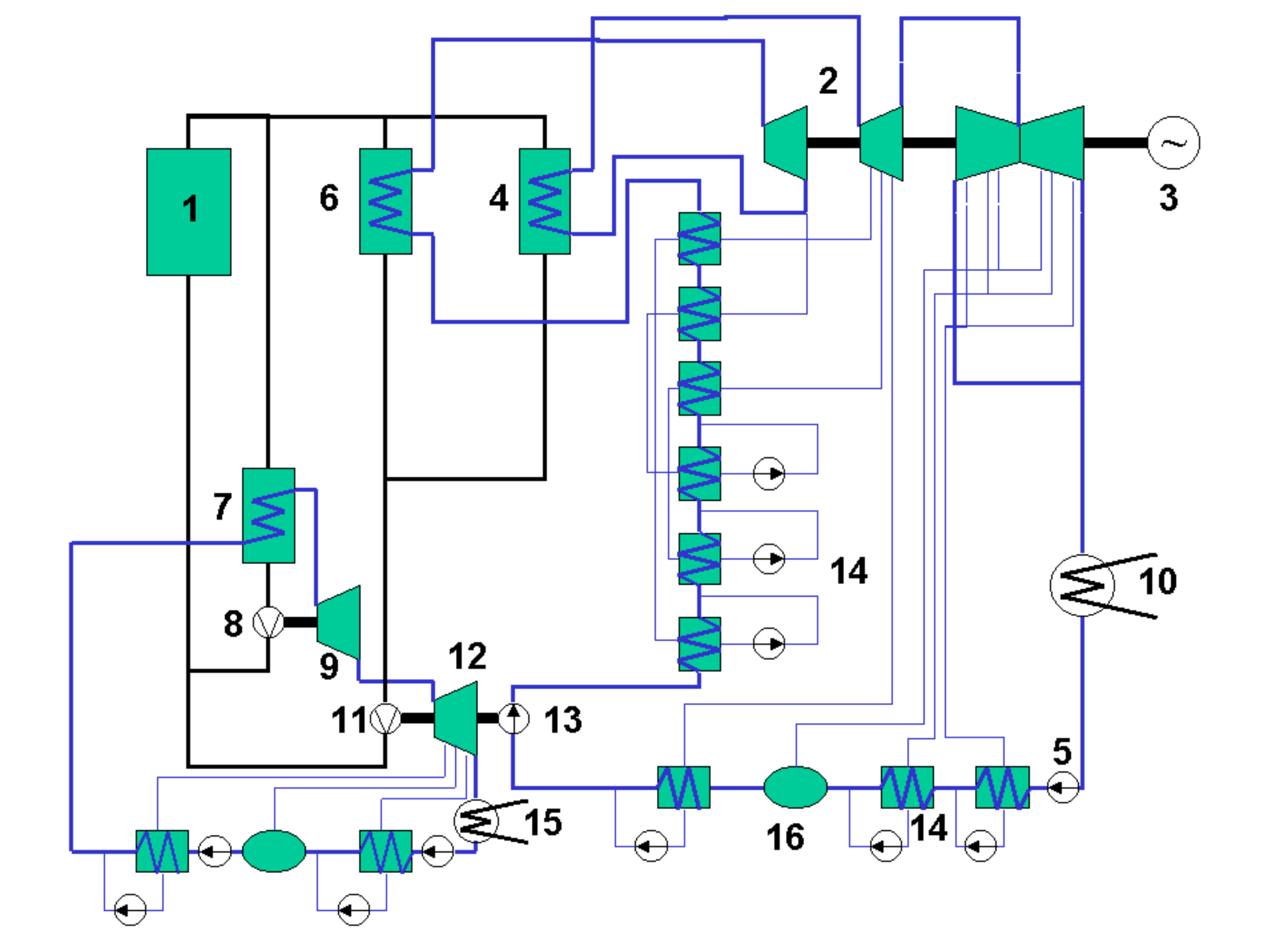
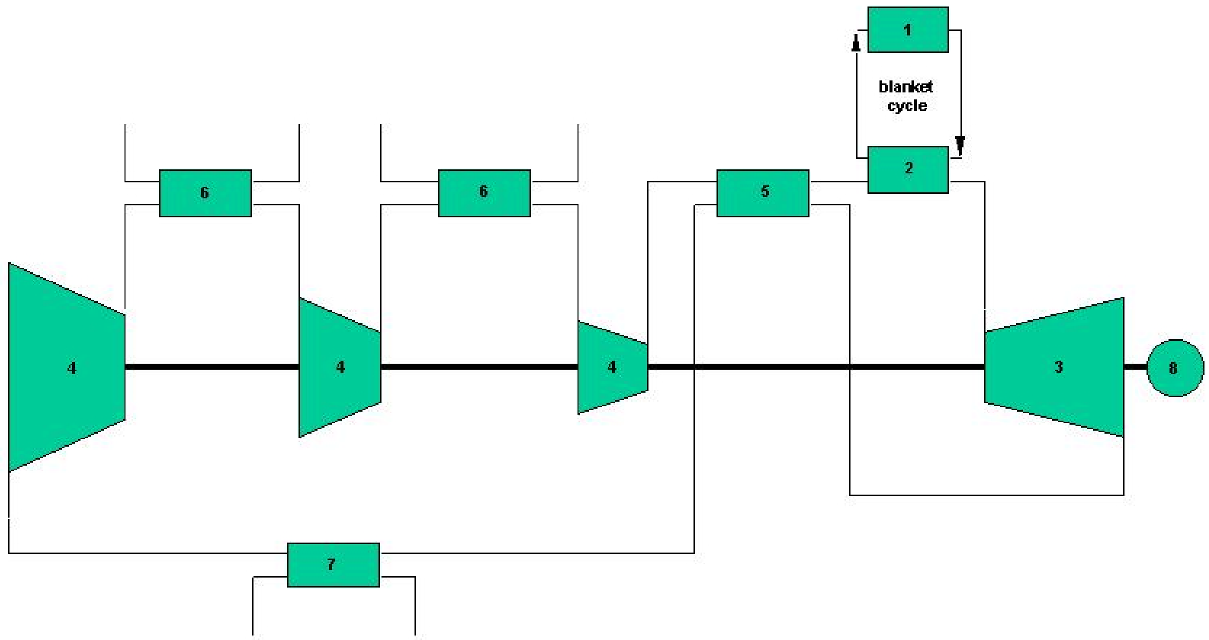


Fig. 9: Q-T-Diagram for the Rankine process.



1 A-HCPB Blanket	7 Aux. steam generator	13 Main feed-water pump
2 Main steam turbine	8 Aux. circuit blower	14 Pre-heater
3 Generator	9 Auxiliary turbine	15 Auxiliary condenser
4 Interm. heat exchanger	10 Main Condenser	16 De-aerating feed-water heater
5 Condensate pump	11 Main blower	
6 Steam generator	12 Main blower turbine	

Fig. 10: Rankine plant scheme.



1 A-HCPB Blanket 2 intermed.heat exchanger 3 turbine	4 compressor 5 recuperator 6 intercooler	7 heat rejection heat exchanger 8 generator

Fig. 11: Brayton plant scheme.

Appendix A: Neutronic calculations for the A-HCPB blanket concept

Neutronic calculations have been performed with the MCNP (Monte Carlo Neutron Photon) code [A1] and nuclear data from the European Fusion File [A2] to assess and optimise the breeding performance of the A-HCPB blanket concept and provide the nuclear heating input data for the subsequent thermal-hydraulic calculations.

A.1 PPA reactor models

Based on the reactor parameters and the neutron source distribution provided by UKAEA Culham [A3,A4], (see Table A1), a generic 7.5 degree torus sector model has been developed for the reactor variant PPA1. This model includes the plasma chamber, four poloidal blanket/shield segments, labelled I-IV, and a bottom divertor port with an integrated divertor of the SEAFP-type, see Fig. A1 for a vertical cross-section. The first wall profile has been adapted in an arbitrary way to the plasma boundary contour shape assuming a scrape-off layer of 15 cm at torus mid-plane. A suitable model of the considered A-HCPB blanket concept was integrated into the generic PPA1 reactor model when investigating the nuclear performance. The radial dimensions are given in Table A2 for the two reactor variants PPA1 [A3] and PPA2 [A5].

Table A1: Main reactor parameters for PPA reactor models.

	PPA1 [A3]	PPA2 [A5]
Plasma major radius [m]	6.73	8.10
Plasma minor radius [m]	2.24	2.7
Plasma aspect ratio	3.0	3.0
Plasma elongation	2.0	1.9
Plasma triangularity	0.36	0.4
Fusion power [MW]	2418	3607

Table A2: Radial blanket dimensions for PPA reactor models.

	PPA1	PPA2
Inboard		
First wall radius [cm]	434	525
Thickness blanket + shield [cm]	90	90
Outboard		
First wall radius [cm]	912	1095
Thickness blanket + shield [cm]	170	170

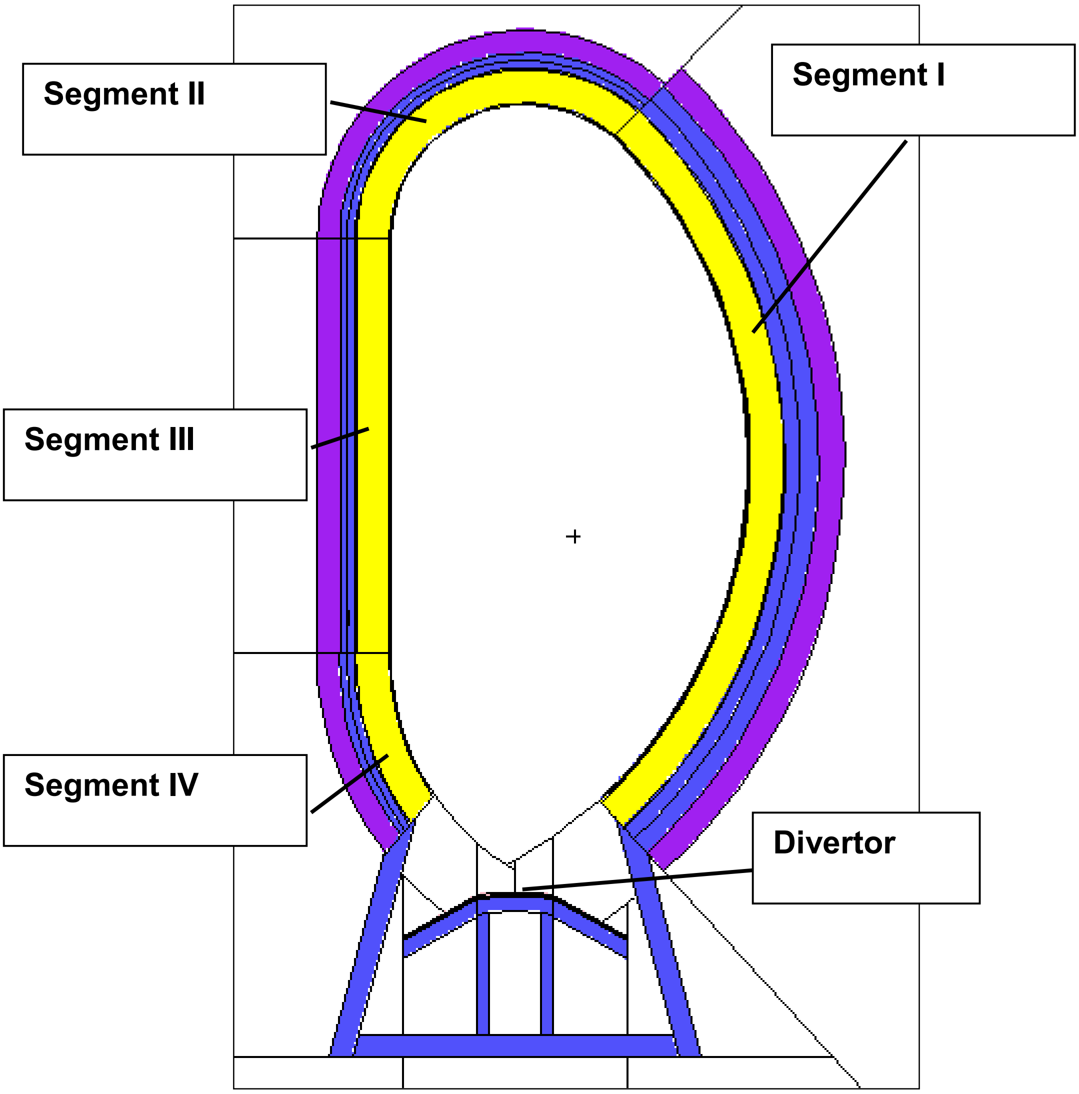


Fig. A1: Vertical cross-section of MCNP torus sector model (PPA1 reactor with A-HCPB blanket segments included)

A.2 Neutron source and wall loading distribution

The neutron source distribution was provided by UKAEA Culham in the form of a numerical data array for a normalised source intensity on a 25 x 40 (r, z) regular mesh [A4, A5]. These data were transformed into a cumulative probability distribution which is being used in a FORTRAN subroutine called by MCNP to sample the source neutrons.

The neutron wall loading distribution was calculated with MCNP for the voided torus sector model by scoring the number of (virgin) 14 MeV neutrons crossing the first wall. Both of the PPA reactor models were considered to compare the respective loadings and enable the extrapolation of results as described below. For the PPA2 reactor, only a simplified skeleton model was developed including plasma chamber, first wall and blanket back wall contour surfaces, and the divertor port opening. This is sufficient when calculating the neutron wall loading distribution with the proper PPA2 neutron source distribution. Table A3 shows the resulting average and peak values for the two PPA reactor models. The normalisation has been performed on the basis of the total fusion power as indicated for

PPA1 and PPA2 above (Table A1). Note that the poloidal profiles of the neutron wall loading are comparatively flat: the poloidal form factor (peak/average) amounts to no more than 1.12 for the outboard segment. In Table A3, there are also given the surface areas as calculated with MCNP for the four poloidal segments. According to the MCNP reactor models, the blanket coverage amounts to 81 and 82 %, for PPA1 and PPA2, respectively.

Table A3: Neutron wall loadings and first wall surface areas for the PPA reactor models.

	PPA1		**PPA2**	
	Neutron wall loading [MW/m²]	Surface area [m²]	Neutron wall loading [MW/m²]	Surface area [m²]
Pol. Segment I (outboard)				
Average	2.27	530	2.51	716
Peak	2.57	-	2.79	-
Pol. Segment II (top)	1.56	143	1.62	182
Pol. Segment III (inboard)				
Average	1.92	152	2.05	215
Peak value	2.28	-	2.55	-
Pol. Segment IV (bottom inboard)	1.26	58	1.37	73.6
Total of blanket segments	2.03	882	2.22	1187
Divertor port	0.755	190	0.903	280
Total including divertor port	1.80	1072	1.97	1467

A.3 Approach for nuclear heating calculations

The nuclear calculations have been performed by making use of the generic PPA1 reactor model as noted above. To allow the assessment for the PPA2 reactor model, the following extrapolation rules have been established:

Nuclear power density at outboard torus mid-plane

$$P_{PPA2}\ [\mathrm{W/cm^3}] = P_{PPA1}\ [\mathrm{W/cm^3}] \cdot WL_{PPA2,max}/WL_{PPA1,max} \cong P_{PPA1}\ [\mathrm{W/cm^3}] \cdot 1.10$$

with WL [MW/m²] = neutron wall loading.

Nuclear power generation in the blanket

$P\ [MW] = WL\ [MW/m^2] \cdot S_{FW}\ [m^2] \cdot M_E$

with S_{FW} = first wall area and M_E = energy multiplication of the blanket.

We have e. g. for the outboard blanket segment (poloidal segment I, 7.5° torus sector):

$P_{PPA1,\,I} = WL_{PPA1,I} \cdot S_{FW,\,PPA1,\,I} \cdot M_{E\,=} = 2.27\ MW/m^2 \cdot 14.92\ m^2 \cdot M_E = 25.1\ MW \cdot M_E$

$P_{PPA2,\,I} = WL_{PPA1,I} \cdot S_{FW,\,PPA1,\,I} \cdot WL_{PPA2,I} / WL_{PPA1,I} \cdot S_{FW,\,PPA2,\,I} / S_{FW,\,PPA1,\,I} \cdot M_E$
$= 25.1\ MW \cdot 2.51\ MW/m^2 / 2.27\ MW/m^2 \cdot 14.92\ m^2 / 11.04\ m^2 \cdot M_E$

$P_{PPA2,\,I} = 25.1\ MW \cdot 1.10 \cdot 1.35 \cdot M_E$

Table A4 shows the multiplication factors M_E that have been derived for the A-HCPB blanket in the PPA1 reactor. The specified rules have also been used to extrapolate to higher neutron wall loading when assessing the design limits of the A-HCPB blanket.

Table A4: energy multiplication factors in the A-HCPB blanket

	energy multiplication factor
outboard blanket segment	1.19
blanket system	1.22
whole reactor	1.24

A.4 Nuclear calculations for the A-HCPB Blanket concept

A.4.1 Blanket lay-out

A technical description of the A-HCPB blanket concept is given in Section 2 of this report. Li_4SiO_4 pebbles are used as breeder material and Beryllium pebbles as neutron multiplier. The SiC_f/SiC fibre reinforced composite is applied as structural material. From the neutronic viewpoint, one of the outstanding features of the A-HCPB blanket concept is the arrangement of the pebble beds parallel to the first wall, see Fig. A2 for a horizontal blanket cross-section. This configuration enables to vary easily the ^6Li-enrichment and the Beryllium/breeder volume ratio in radial direction. As a result, the A-HCPB blanket concept can achieve a minimum Beryllium mass inventory at a minimum breeder zone thickness.

A.4.2 Tritium breeding performance

With the A-HCPB blanket concept, the required neutron multiplication is provided by 5 Beryllium pebble beds with a total radial thickness of 14 cm. The radial thickness of the single beds is at 2 cm (front bed) and 3 cm (remaining beds) to keep the Beryllium temperature below 600°C. Tritium breeding is provided by 7 Li_4SiO_4 pebble beds with a thickness of 9 mm (first four beds) and 10 mm (remaining beds). The total blanket thickness amounts to 35 cm. The ^6Li enrichment is varied between 7.5 at% (natural enrichment) in the front beds and 60 at% in the back beds. A low enrichment is required in the front beds to keep the breeder temperatures below the tolerable maximum temperatures. The high enrichment in the back beds along with a graphite reflector attached to the back of the blanket ensures a sufficient Tritium breeding performance, see Table A5.

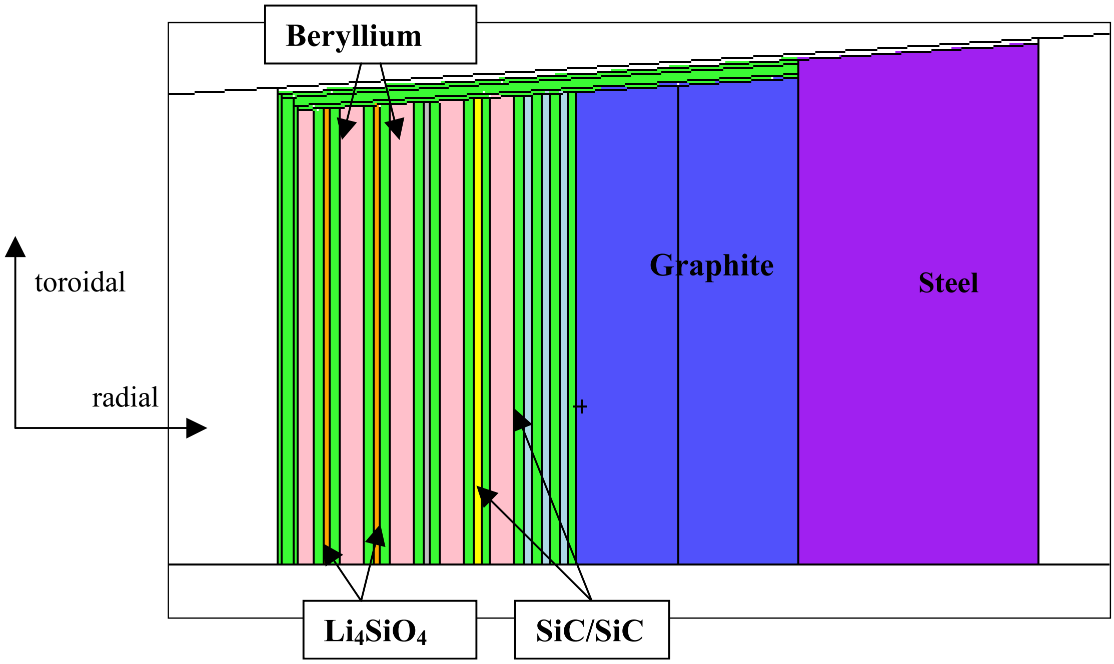


Fig. A2: A-HCPB blanket: horizontal cut through outboard blanket segment (PPA1 reactor, torus mid-plane)

Table A5: Tritium breeding ratio of the A-HCPB blanket

Breeder pebble bed	I	II	III	IV	V	VI	VII	Total
^{6}Li-enrichment [at%]	7.5	7.5	15	30	60	60	60	
TBR	0.19	0.16	0.20	0.17	0.13	0.09	0.15	1.09

A.2.1 Nuclear heating

Nuclear heating calculations include Monte Carlo calculations of the nuclear power generated in the various reactor components (first wall, blanket, shield, divertor etc.) and of the nuclear power density distribution. For the purpose of the PPA study, the radial power density profile at torus mid-plane of the outboard blanket segment has been considered where there are the highest nuclear and thermal loadings.

The nuclear power generation is given in Table A6 as calculated for a 7.5 ° toroidal sector of the PPA reactor with an fusion power assumed as specified above (Table A1). The total nuclear power generated in the reactor amounts to 2410 and 3600 MW, PPA1 and PPA2, respectively. This corresponds to a global energy multiplication factor of 1.24 when taking into account all reactor components, i. e. including blanket, shield and divertor. Fig. A3 shows the radial power density profile of the outboard blanket at torus mid-plane for the reference solution with 5 Beryllium and 7 orthosilicate pebble beds. For illustration purposes, Fig. A4 displays the power density profiles for a variant with 5 Beryllium and 6 orthosilicate pebble beds with slightly varied enrichments. Note the large impact on the power density when varying the enrichment in one of the pebble beds only.

Table A6: Power generation [MW] in a 7.5 ° toroidal sector of the PPA reactor with A-HCPB blanket segments.

PPA1 **Poloidal segment**	**I**	**II**	**III**	**IV**	**Divertor**	**Total**
Li_4SiO_4	**11.7**	**2.65**	**2.74**	**0.92**	**-**	**18.0**
Beryllium	**8.71**	**1.70**	**1.91**	**0.53**	**-**	**12.8**
SiC/SiC	**6.00**	**1.30**	**1.60**	**0.45**	**-**	**9.36**
Graphite	**3.53**	**0.75**	**0.66**	**0.26**	**-**	**5.19**
Total sector	**29.9**	**6.39**	**6.92**	**2.16**	**4.74**	**50.1**

PPA2 **Poloidal segment**	**I**	**II**	**III**	**IV**	**Divertor**	**Total**
Li_4SiO_4	17.4	3.53	4.16	1.28	**-**	**26.4**
Beryllium	13.0	2.26	2.90	0.74	**-**	**18.9**
SiC/SiC	8.94	1.73	2.43	0.63	**-**	**13.7**
Graphite	5.3	0.99	1.0	0.36	**-**	**7.6**
Total sector	44.6	8.51	10.5	3.0	**8.37**	**75.0**

References

[A1] F. Briesmeister (ed.): MCNP - A General Monte Carlo N-Particle Transport Code, Version 4A, LA-12625-M, November 1993, Version 4B March 1997

[A2] P. Vontobel: A NJOY Generated Neutron Data Library Based on EFF-1 for the Continuous Energy Monte Carlo Code MCNP, PSI-Bericht Nr. 107, September 1991

[A3] I. Cook, , UKAEA Culham, PPA geometry etc, e-mail of 21 January, 1999.

[A4] N. Taylor, UKAEA Culham, Neutron source distribution, e-mail of 16 February, 1999.

[A5] I. Cook, UKAEA Culham, PPA parameters, e-mail of 12 March, 1999.

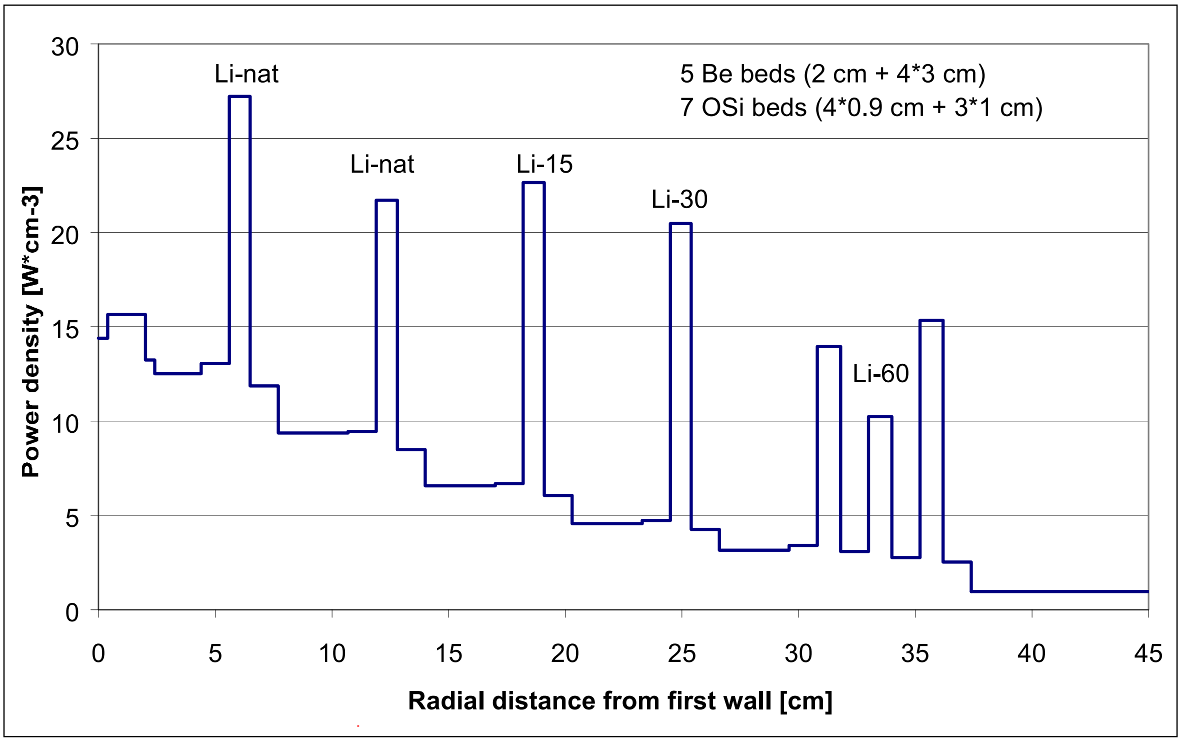


Fig. A3: A-HCPB blanket: Radial power density profile in the outboard blanket segment (PPA2 reactor, torus mid-plane, reference solution).

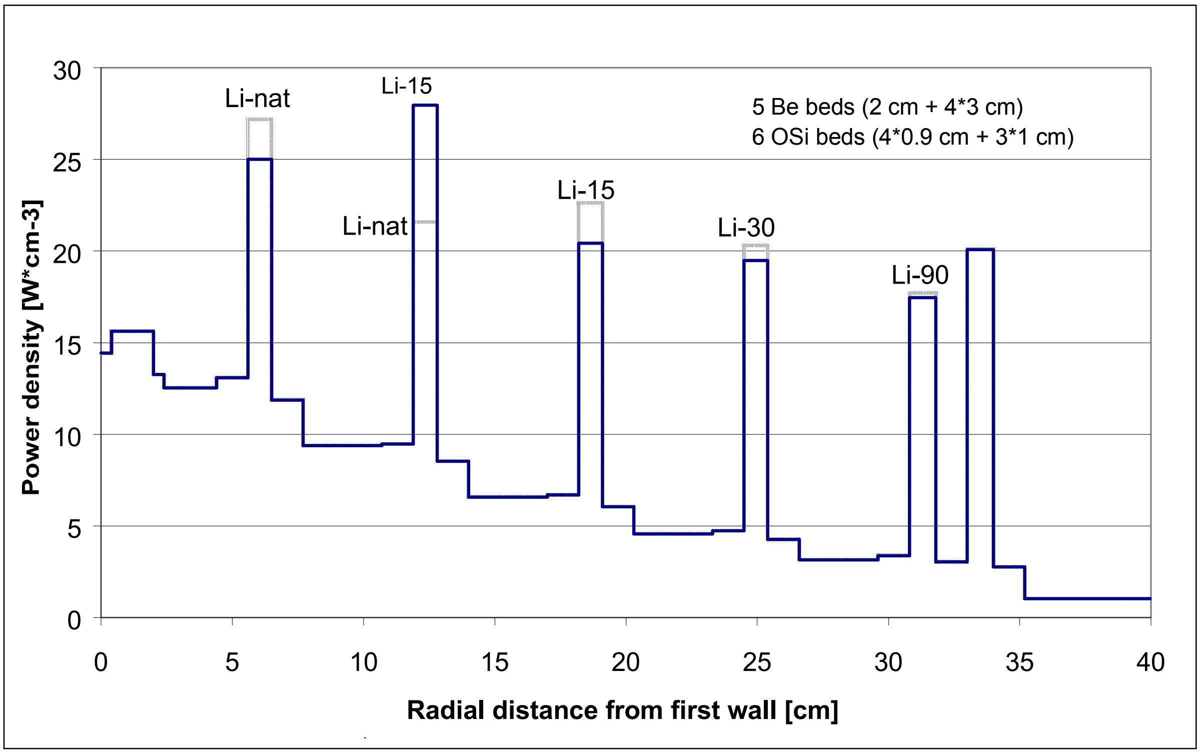


Fig. A4: A-HCPB blanket: Radial power density profile in the outboard blanket segment (PPA2 reactor, torus mid-plane, 11 beds variants).

Appendix B: Optimisation of the 3 compression stage Brayton cycle vs. blanket inlet temperature

The proposed Brayton cycle [14] has to be optimised with respect to the blanket inlet temperature T_{in}. In fact, this temperature - that is directly correlated with the compression ratio r - has a great influence on the internal efficiency of the cycle. The reference cycle was designed for a compression ratio of 2, that means for an inlet blanket temperature of about 482°C in the A-HCPB design. The range of temperature that can be adopted for the blanket inlet temperature is between 350°C and 430°C as it was discussed in Section 4.3.

The following study extrapolates this cycle to higher values of the compression ratio to investigate the influence of the blanket inlet temperature on the efficiency. Table B1 shows the values of the cycle parameters used in this assessment. This values have been taken from reference [14]; a recuperator pinch of about 15K was evaluated on the basis of a the given effectiveness of the recuperator of 0.96 at the compression ratio of 2. In this extrapolation the effectiveness of the recuperator decrease with the increase of the compression ratio because a constant value of the recuperator pinch is assumed. Other parameters are supposed to be not affected by the variation of the compression ratio.

Table B1: reference parameters used in the assessment.

reference parameters	
blanket outlet temperature	700°C
heat rejection temperature	35°C
heat exchanger pinch	30K
recuperator pinch	15K
pressure drop	5%
compressor efficiency	0.92
turbine efficiency	0.92

The calculation of the efficiency of the cycle is based on the correlation presented in [14]. Fig B1,B2 and B3 show the results of the calculation in term of internal efficiency, recuperator effectiveness and blanket inlet temperature as function of the compression ratio. Fig B4 shows the efficiency as function of the blanket inlet temperature. From this picture is clear that efficiency of 46% can be reached for a blanket inlet temperature at least of 430°C.

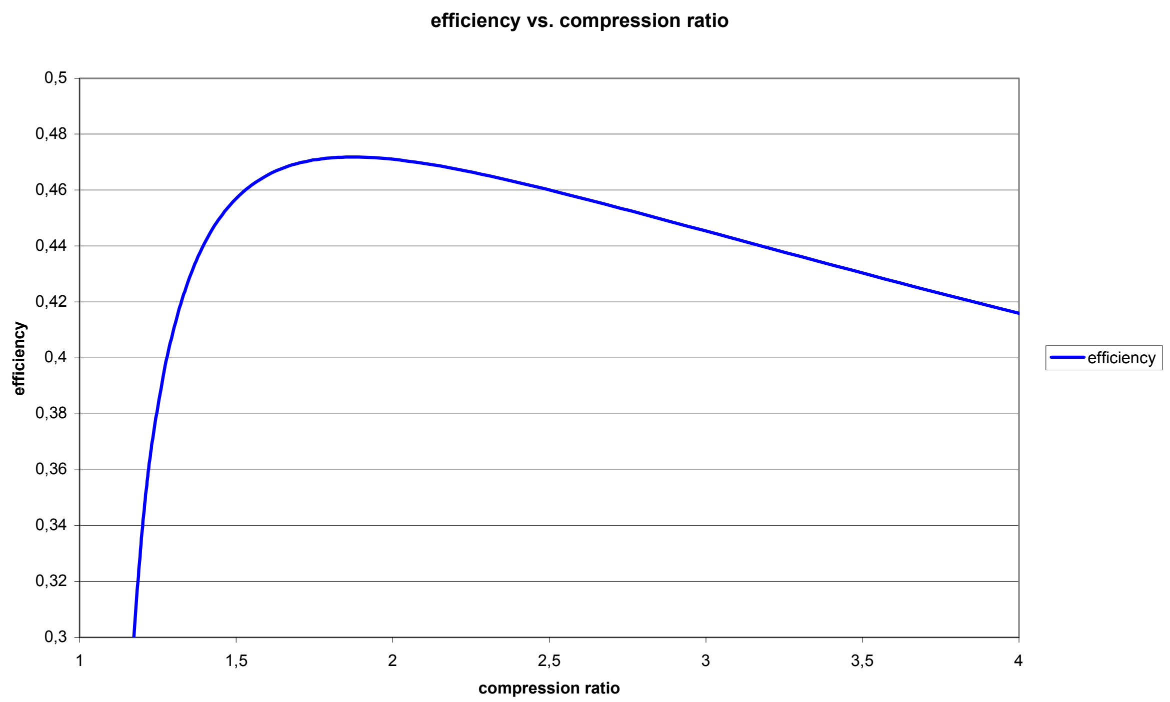


Fig. B1: internal efficiency as function of the compression ratio

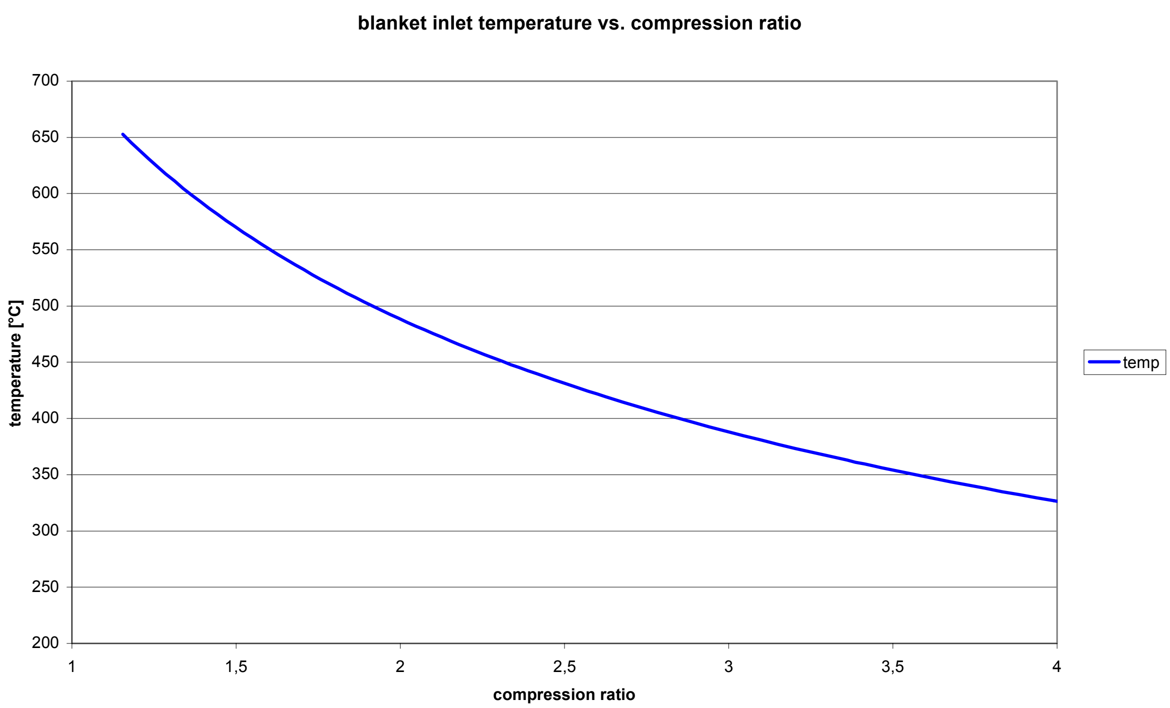


Fig. B2: Blanket inlet temperature as function of the compression ratio

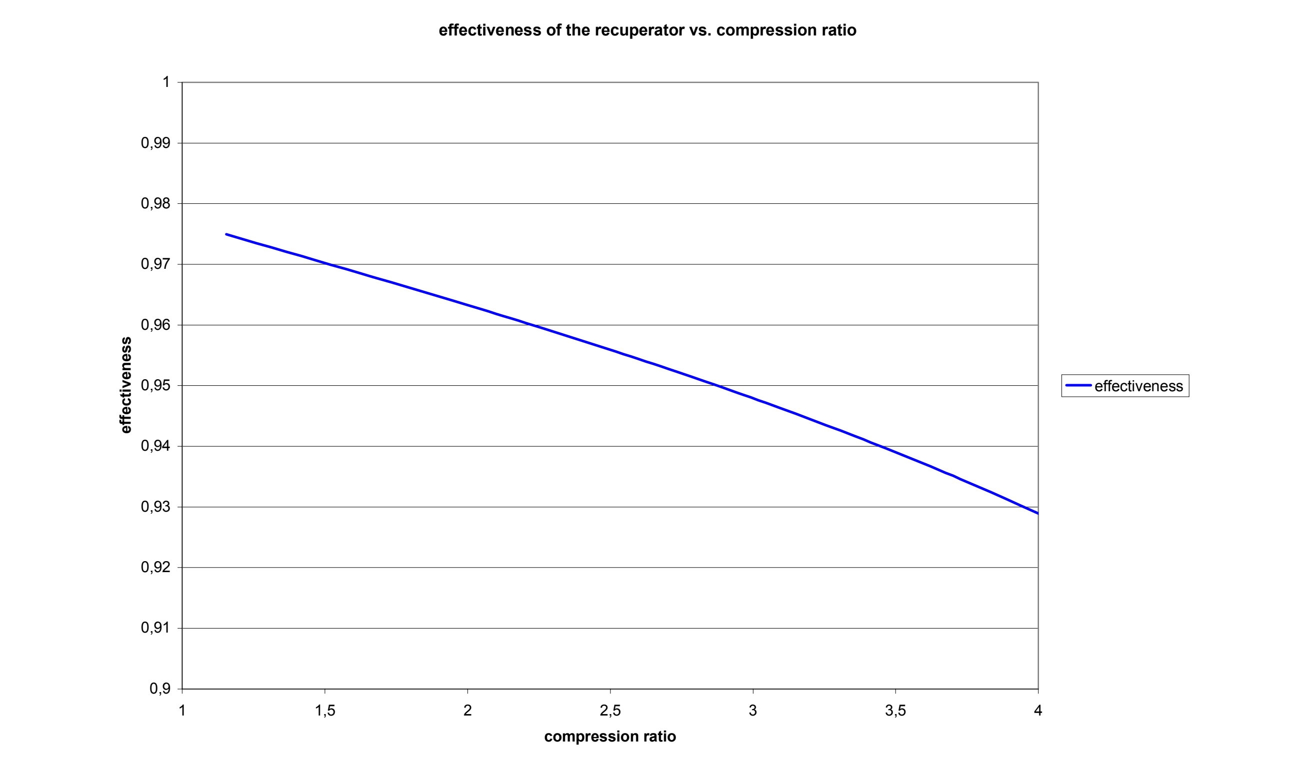


Fig. B3: recuperator effectiveness as function of the compression ratio

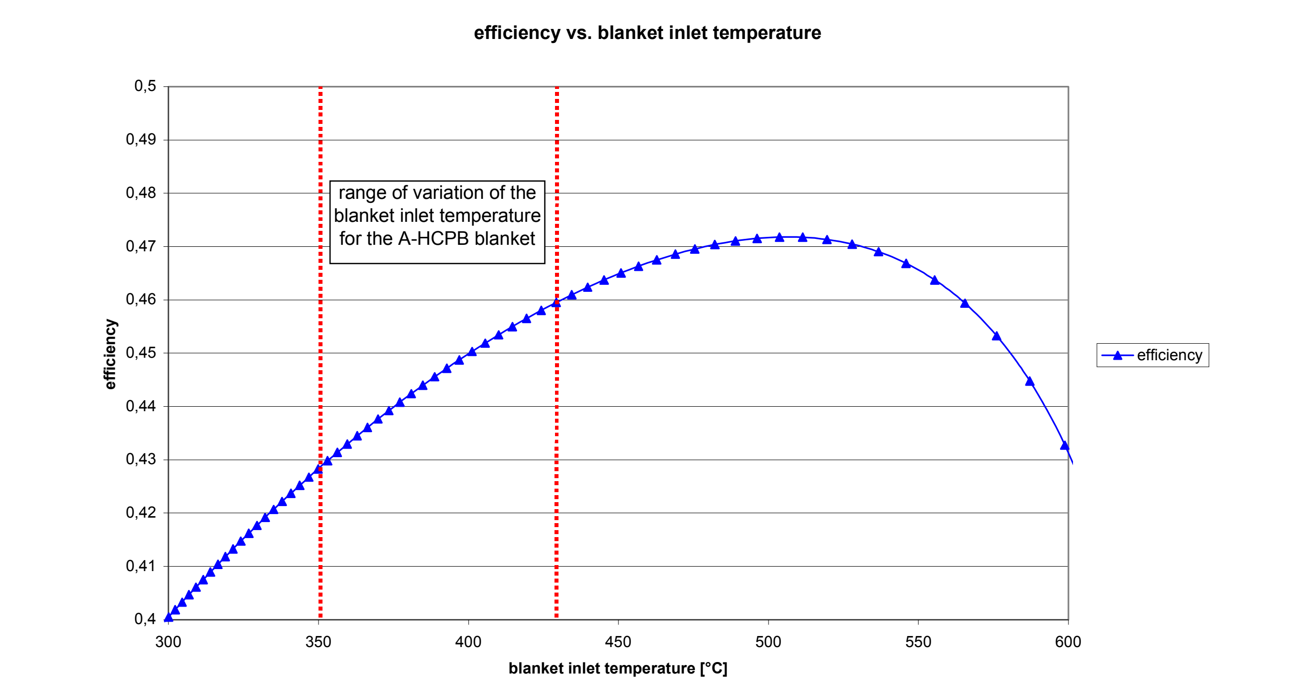


Fig. B4: internal efficiency as function of the blanket inlet temperature