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Improved Helium Cooled Pebble Bed Blanket

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Weiterentwicklung des heliumgekühlten Schüttbett-Blankets

Zusammenfassung

Im europäischen Fusionsprogramm sind 1999 Vorarbeiten (Preparation of a Power Plant Conceptual Study - Availability, PPA) zu einer ab 2000 geplanten Reaktorstudie durchgeführt worden, bei der die Attraktivität einer Fusionsanlage unter kommerziellen Gesichtspunkten wie erzielbare Leistung, Wirkungsgrad und Verfügbarkeit, im Mittelpunkt steht. Im Rahmen dieser Arbeiten wurde im Forschungszentrum Karlsruhe das heliumgekühlte Schüttbett-Blanket (HCPB) für DEMO zum „Improved HCPB“ weiterentwickelt (Subtask PPA 2.3). Das modifizierte Konzept erlaubt eine Reduzierung der Höhe der Brutstoffschüttbetten und damit die Erhöhung der Leistungsdichte, sowie die Verwendung mono-dispersiver Beryllium-Schüttbetten. Der elektrische Wirkungsgrad des Blankets konnte um fast 7 Punkte auf etwa 37% gesteigert werden, die sich aus höherer Aufwärmspanne des Kühlmittels, Reduzierung von Strömungsverlusten im Blanket und verbesserter Energieumwandlung im vorgesehenen Dampfprozess ergeben. Die in früheren Studien nachgewiesene gute Verfügbarkeit des DEMO-HCPB dürfte dabei erhalten bleiben.

Abstract

In the European Fusion Programme of 1999 preparatory work (Preparation of a Power Plant Conceptual Study - Availability, PPA) has been carried out for a fusion power plant study that is planned to start in 2000. This study will focus on the commercial attractiveness of a fusion plant, particularly achievable power level, net efficiency and availability. Part of the activity at the Forschungszentrum Karlsruhe has been the further development of the Helium Cooled Pebble Bed (HCPB) blanket for DEMO as “Improved HCPB” (Subtask PPA 2.3). The modified concept allows for the height of breeder pebble beds to be reduced and thus for larger power densities to be accommodated. Also, mono-disperse Beryllium pebble beds can be used. The net electric efficiency of the blanket was raised by almost 7 points to about 37% due to increased coolant temperature gain, reduction of pressure losses in the blanket and enhanced energy conversion in the proposed steam process. The good availability of the DEMO-HCPB that was shown in earlier studies is expected to carry over to the I-HCPB.

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1 Introduction

The helium cooled pebble bed (HCPB) ceramic blanket [1] is one of two blankets chosen for a European DEMO reactor and has seen steady development over the past years. Key points of the development have been the design of a HCPB test module for ITER [2], continuing work on characterising physical properties of Beryllium and ceramic breeder pebble beds [3,4] and the exploration of blanket manufacturing technology [5].

Adapting the concept in the framework of a preliminary reactor study (Preparation of a Power Plant Conceptual Study - Availability, PPA, Subtask PPA 2.3) was a welcome opportunity to address design issues that are now seen as limiting the attractiveness of the HCPB blanket, i.e. (i) a low electrical efficiency of 29.8% due to coolant outlet temperature of 450°C and large pressure drop in the blanket; and (ii) minimum bed heights of 9mm in the ceramic breeder that limit the level of neutron power acceptable in the breeding zone.

Design changes described in the following can bring significant improvements for the low-activation EUROFER steel that is to be used in the DEMO HCPB. This is contrasted with a blanket based on oxide-dispersion strengthened (ODS) steel with a limiting temperature 100K higher than EUROFER to explore the scope for higher power and efficiency.

2 Design description of the Improved HCPB

Fig. 1 and 2 depict a section from an outboard segment of the proposed improved HCPB (I-HCPB). A U-shaped First Wall (FW) of 25 mm thickness together with shielding and support structure provides a stiff blanket box. The FW coolant channels of 16x16 mm^2 cross section and 22 mm pitch lie in radial-toroidal planes; alternating directions of flow support a balanced FW temperature distribution, while the use of two separate cooling circuits enhances blanket safety.

The blanket box contains the breeding region with alternating shallow pebble beds of Beryllium and ceramic breeder that are separated by 5 mm-thick steel cooling plates. The plates adjacent to the ceramic breeder beds are designed as closed containers without mechanical connection to the FW. Additional stiffening plates are welded into the blanket box dividing the Beryllium bed into equal halves; they strengthen the box and provide extra cooling for the Beryllium beds. Coolant flow through the 3x3 mm^2 channels of 5 mm toroidal pitch is in radial direction.

FW cooling channels are connected in series with those in the breeding region. On leaving the FW, helium is fed by collectors into the radial channels of both breeder containers and stiffening plates. Void channels in the stiffening plates determine the distribution of mass flow through the two types of cooling plates in such a way that the same temperature rise is attained.

Single-size Be pebble beds are proposed as neutron multiplier, while Li_4SiO_4 pebbles have been assumed as breeder material. According to neutronic calculations for the two reference cases in section 3 this choice provides Tritium breeding self-sufficiency, at 30 at% ^{6}Li-enrichment of the breeder.

Two different low-activation structural materials, EUROFER and ODS, have been considered.

The concept of cooling plate containers and cooled stiffening plates has important consequences for operating the breeding zone. (i) The manufacturing limit on the minimal height of the ceramic breeder bed known from the DEMO HCPB is done away with; hence, the power density can be increased significantly without exceeding maximum breeder temperatures. (ii) The additional cooling of the Be bed brought about by the stiffening plate is essential for increasing the power density in Be, and for using a single-size Be pebble bed that has a thermal conductivity much lower than the binary Be bed in the DEMO HCPB. The lower packing fraction increases the margin for Be swelling; in conjunction with a reduced radial extension of the breeding zone it leads to a reduction of the Be inventory by more than 30% of the DEMO HCPB.

An I-HCPB based on the available EUROFER will have all the advantages discussed above. In addition, using as structural material an ODS steel that is assumed to have operating limits 100 K above EUROFER would allow larger Helium outlet temperatures, increased electrical net efficiency and hence could be commercially more attractive.

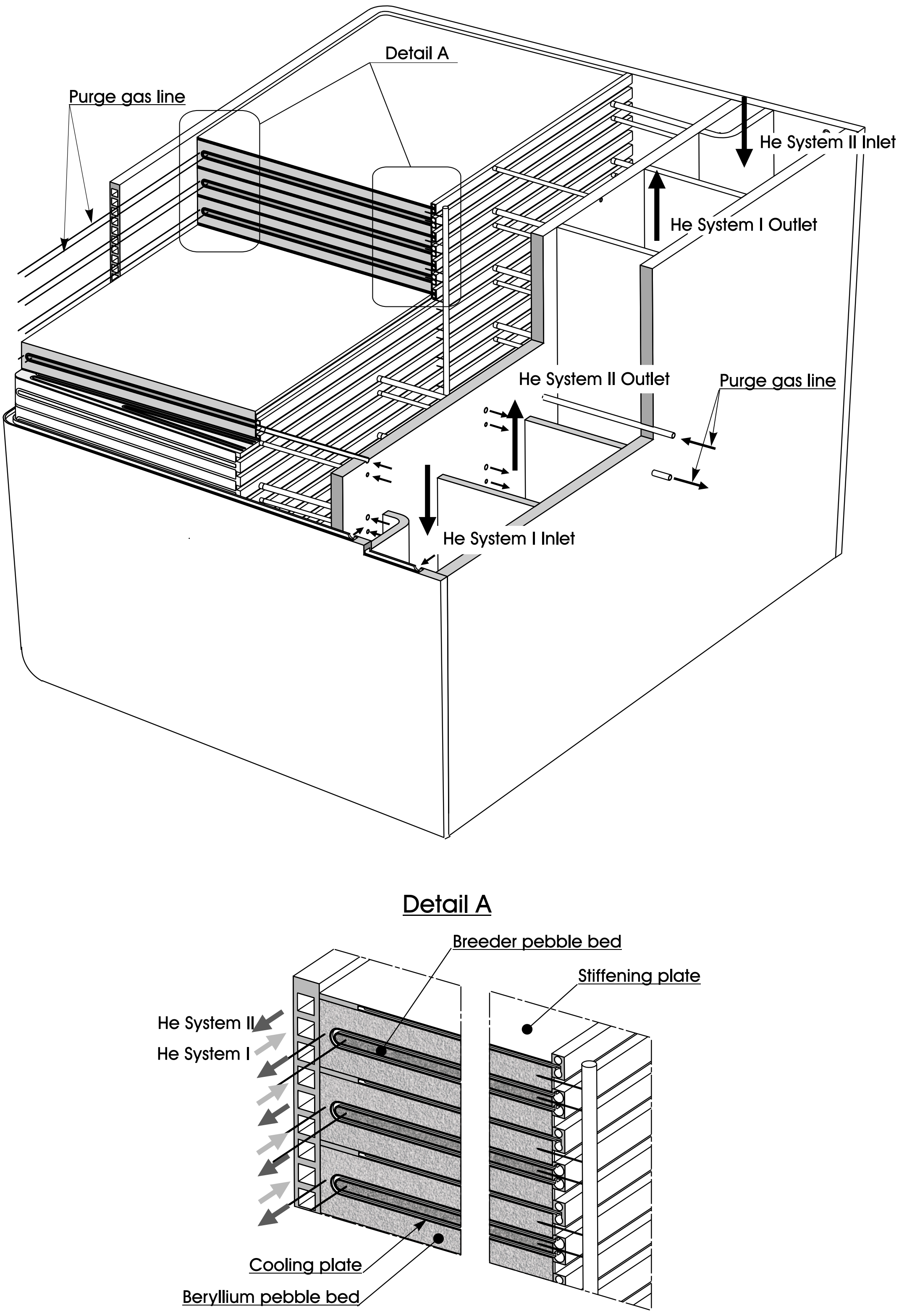


Figure 1 Improved HCPB blanket

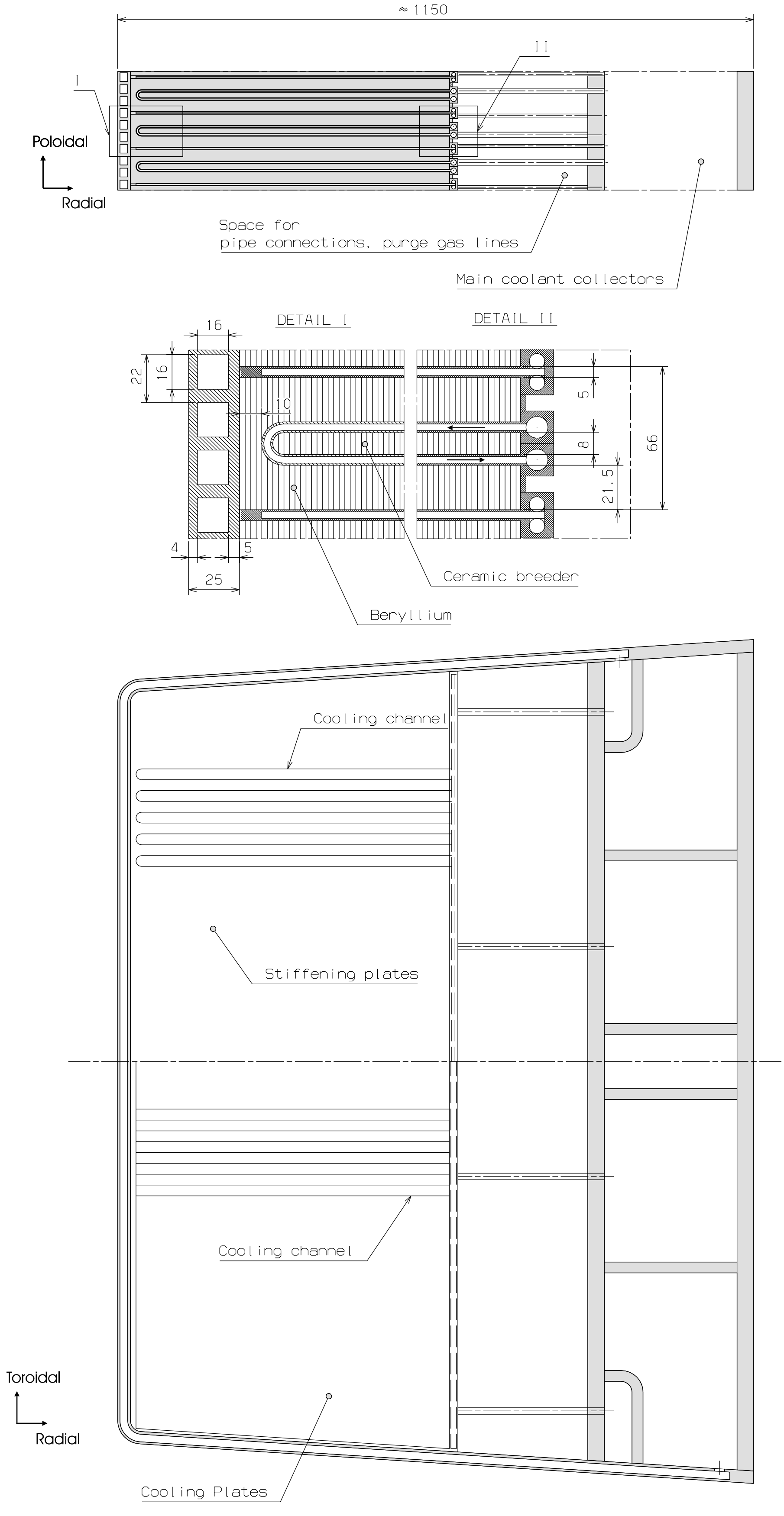


Figure 2 Improved HCPB blanket: section from an outboard blanket module (EUROFER case)

3 Calculations for EUROFER and ODS reference designs

A feasible blanket design will be a compromise of a number of goals and limitations like temperature limits, Tritium breeding self-sufficiency, efficiency considerations etc. This fact is recognised here by presenting balanced reference cases that try to push the I-HCPB design to reasonable limits. With the two alternative structural materials so different regarding their stage of development, and with EUROFER found to allow an attractive design it was decided to contrast two cases based on EUROFER and ODS. The major difference between the cases is the coolant temperature gain of 250 K for EUROFER compared to 300 K for ODS, with the inlet temperature 250°C in both cases. Properties and design limits for the materials used are given in Tab. 1. It was assumed that (i) EUROFER properties are identical to those of T91 and that (ii) ODS mechanical properties are those of T91 shifted to temperatures 100K higher.

Table 1 Properties and design limits for materials used in the I-HCPB blanket

Li_4SiO_4 pebble bed	
Pebble size [mm]	0.25-0.63
Thermal conductivity [W m^{-1} K^{-1}] (T=500-900°C)	1.0-1.2
Max. allowable temperature [°C]	920
Single-size Beryllium pebble bed	
Pebble size [mm]	2
Thermal conductivity [W m^{-1} K^{-1}] (500,600,700°C)	5.4,8.8,10.6
Max. allowable temperature [°C]	700
EUROFER low-activation steel	
Thermal conductivity [W m^{-1} K^{-1}]	29
Max. allowable temperature [°C]	550
Young's modulus [GPa] (T=400°C)	181.5
Poisson ratio	0.3
Thermal expansion coefficient [10^{-6} K^{-1}] (T=400°C)	11.9
Max. allowable primary stress $S_{m,t=10000h}$ [MPa] (400-550°C)	262-105
Max. allowable total stress $3 \cdot S_m$ [MPa] (400-550°C)	522-378
ODS low-activation steel where assumed different from EUROFER	
Max. allowable temperature [°C]	650
Max. allowable primary stress $S_{m,t=10000h}$ [MPa] (500-650°C)	262-105
Max. allowable total stress $3 \cdot S_m$ [MPa] (500-650°C)	522-378

The two reference cases are the result of an iterative process in which linearly scaled power levels, pebble bed heights and thermohydraulic quantities were brought to agreement. The proposed EUROFER design has 8 mm ceramic breeder, Be beds of 2x21.5 mm and has been tailored to reactor maximum neutron wall load of 3.5 MW/m^2 with 0.61 MW/m^2 maximum surface heat load. At 7 mm breeder bed height and 2x22 mm Be beds, the ODS reference case has been designed for values of 4.4 MW/m^2 max. neutron wall load and 0.78 MW/m^2 max. surface heat flux. These values exceed the boundary conditions of the base PPA2 model [6] provided by UKAEA Culham by 25% for EUROFER and 58% for ODS. Characteristic data of the two reference cases are provided in Tables 2 and 3; they have been calculated for the equatorial midplane of the outboard blanket with neutron wall load and surface heat flux at their peak.

Table 2 I-HCPB. Main plant and blanket design data

Reference case	EUROFER	ODS
Overall plant		
Fusion power [MW]	4500	5700
Neutron power [MW]	3600	4560
Alpha-particle power [MW]	900	1140
Energy multiplication	1.41	1.41
Thermal power [MW]	5976	7570
Blanket		
Neutron power [MW]	3285	4160
Alpha-particle power [MW]	558	707
Energy multiplication	1.34	1.34
Thermal power [MW]	4960	6280
Blanket surface [m^2]	1187	1187
Average neutron wall load [MW/m^2]	2.8	3.5
Max. neutron wall load [MW/m^2]	3.5	4.4
Average surface heat load [MW/m^2]	0.47	0.60
Max. surface heat load [MW/m^2]	0.61	0.78
Coolant	He	He
- Inlet temperature [° C]	250	250
- Outlet temperature [° C]	500	550
- Pressure [MPa]	8	8
- Mass flow rate [kg/s]	3815	4025
- Pumping power, η= 0,8 [MW]	196	219
Net efficiency of power conversion system*	36.5	37.5
Electrical output [MW]	1810	2350

*with intermediate superheating

Table 3 I-HCPB. Preliminary results of the reference case analyses

Reference case	EUROFER	ODS
Design data		
Max. neutron wall load [MW/m^2]	3.5	4.4
Breeder bed height [mm]	8	7
Beryllium bed height [mm]	2x21.5	2x22
Neutronic results		
Tritium breeding ratio	1.11	1.1
Max. power density in the breeder [MW/m^3]	49.8	63.3
Max. power density in Beryllium [MW/m^3]	14	17.8
Max. power density in structural material [MW/m^3]	29.4	37.4
Thermohydraulic results		
Max. temperature in the breeder [°C]	887[a]	878[a]
Max. temperature in Beryllium [°C]	564[a]	598[a]
Max. temperature in stiffening plate [°C]	532[b]	617[b]
Max. FW temperature [°C]	491[c]	554[c]
Helium mass flow in the 66mm unit cell [kg/s]	0.385[a]	0.404[a]
Helium velocity FW [m/s]	73.7[a]	79.1[a]
Helium velocity cooling plates [m/s]	20.7[a]	22.8[a]
Heat transfer coefficient FW [W/m^{2}K]	12400[a]	12900[a]
Heat transfer coefficient cooling plates [W/m^{2}K]	5050[a]	5300[a]
Pressure drop FW [MPa]	0.096[a]	0.108[a]
Pressure drop cooling plates [MPa]	0.011[a]	0.013[a]
Thermomechanical results		
Max. v.Mises stress in the FW [MPa]	392[c]	526[c]

[a] PBM of section 3.2
[b] BZM of section 3.2
[c] thermomechanical model of section 3.3

3.1 Neutronic calculations

Calculations have been carried out with the MCNP code [7] and nuclear data from the European Fusion File [8] to determine the breeding performance of the I-HCPB and provide nuclear heating data for the thermohydraulic analyses.

A generic 7.5 degree torus sector model was developed based on the reactor parameters and the neutron source distribution of the PPA reactor. 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 is an adaptation of the plasma boundary contour shape assuming a scrape-off layer of 150 mm at torus mid-plane. According to the MCNP reactor model, the blanket coverage is 82% with a resulting FW blanket surface of 1187 m^2.

Calculations for the PPA2 base case suggest that Tritium breeding ratios of 1.11 for EUROFER and 1.1 for ODS are reached at a radial breeding region depth of 490 mm and 30at% ^{6}Li-enrichment. Radial and/or poloidal gradation of the ^{6}Li-enrichment is not necessary.

Maximum values of the linearly scaled power densities in the EUROFER case are 49.8 MW/m^3 in the breeder, 14 MW/m^3 in Beryllium and 29.4 MW/m^3 the steel. For the ODS case, these values are 63.3 MW/m^3, 17.8 MW/m^3 and 37.4 MW/m^3.

At 1.34, the I-HCPB's energy multiplication factor is considerably larger than for other blanket concepts. The value is even larger for the overall plant, i.e. 1.41, which underlines the attractiveness of the I-HCPB for a power reactor.

3.2 Thermohydraulic calculations

Three different models have been applied in analysing the I-HCPB thermohydraulics. (i) A physical-basics model (PBM) was employed to compute overall energy balance, coolant mass flow, heat transfer coefficients and pressure drop in the blanket; (ii) a finite element FW model (FWM) was used to find local FW coolant temperatures; and (iii) a finite-element breeding zone model (BZM) was applied to analyse coolant, structural and pebble bed temperatures.

PBM The model covers a radial-toroidal blanket slice of 66mm height, which is the unit cell of the blanket. The radial extension agrees to the 490mm used in the neutronics model. Three important assumptions were made: (i) the connection of FW and breeding zone was not modelled; (ii) Power generated in the modelled section accounts for about 88% of the overall power. It was assumed that the Helium coolant removes all the power, and that the roughly 12% missing are equally divided between the in-flow and the out-flow leg. This corresponds to a loss of temperature increase in the blanket of 12% compared to the nominal values and is a conservative interpretation of power values; (iii) it was assumed that artificial roughening of the plasma-facing side of the FW channels doubles the HTC on that surface compared to smooth-surface channels, and that the increase in pressure drop for this scheme is 20% [9].

Preliminary analysis shows maximum temperatures in the breeder bed of 887°C for EUROFER, 878°C for ODS; in the Beryllium bed the maximum values are 564°C and 598°C. Pressure losses of about 0.12 MPa in FW and cooling plate channels together are about half the value of the DEMO HCPB blanket. Fluid mass flows, velocities, heat transfer coefficients and pressure drops computed with this model are given in Table 3.

FWM The three-dimensional FWM covers a 22mm high FW section including two half FW channels and 10mm Beryllium pebble bed on the side of the breeding zone. Figure 3 depicts the meshed cross-section of the model. Surface heat flux and neutron sources from the neutronic analysis are applied. Helium in the two channels flows in opposite toroidal directions; fluid velocity and heat transfer coefficients between fluid and structure were taken from the PBM. Figures 4 and 5 display the FW helium temperatures over the cannel length for the two reference cases.

BZM Figure 6 depicts a poloidal-toroidal cross-section of the three-dimensional BZM. While the width of three channel pitches is the toroidal unit cell width, the model covers two Be beds and two stiffening plates to account for poloidal boundary effects; it is thus higher than the previously noted 66mm for a unit cell. Calculations suggest that the heat generation in the additional bed and plate is just balanced by the additional stiffening plate cooling.

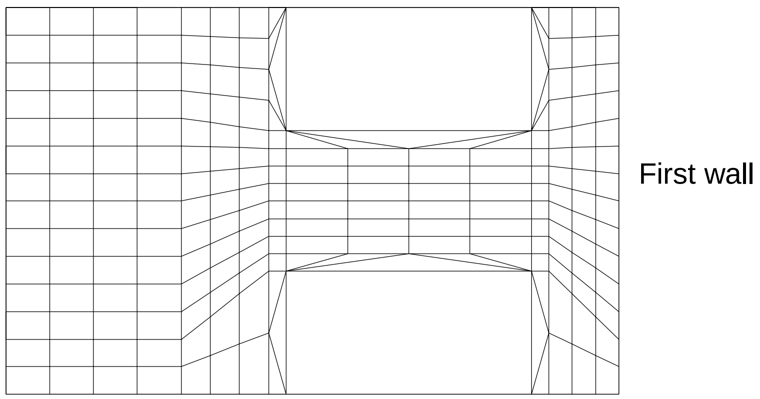


Figure 3 Radial-poloidal cross-section of the FWM mesh

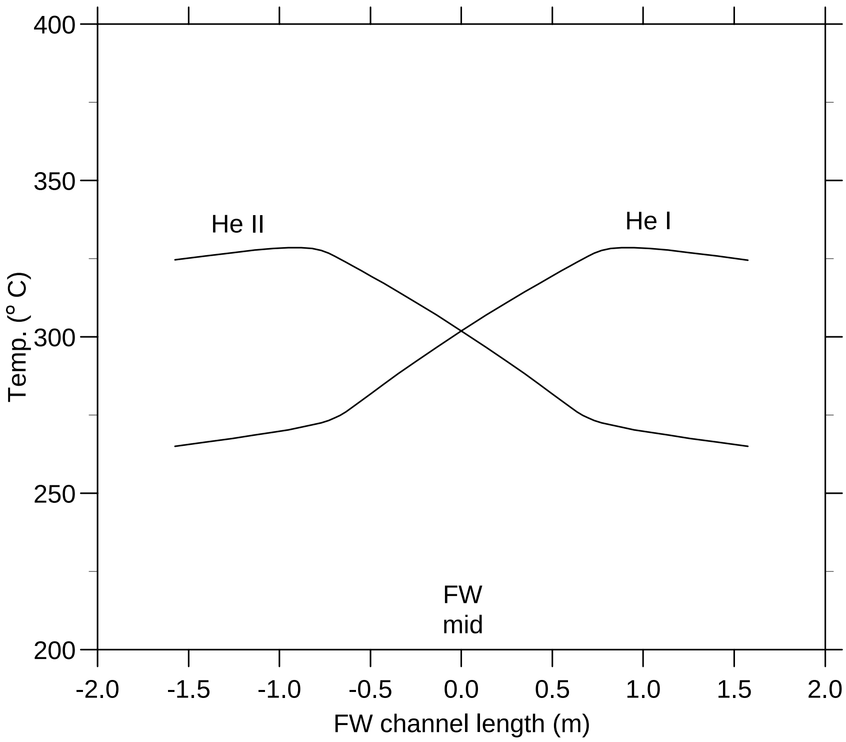


Figure 4 EUROFER reference case: FW coolant temperatures

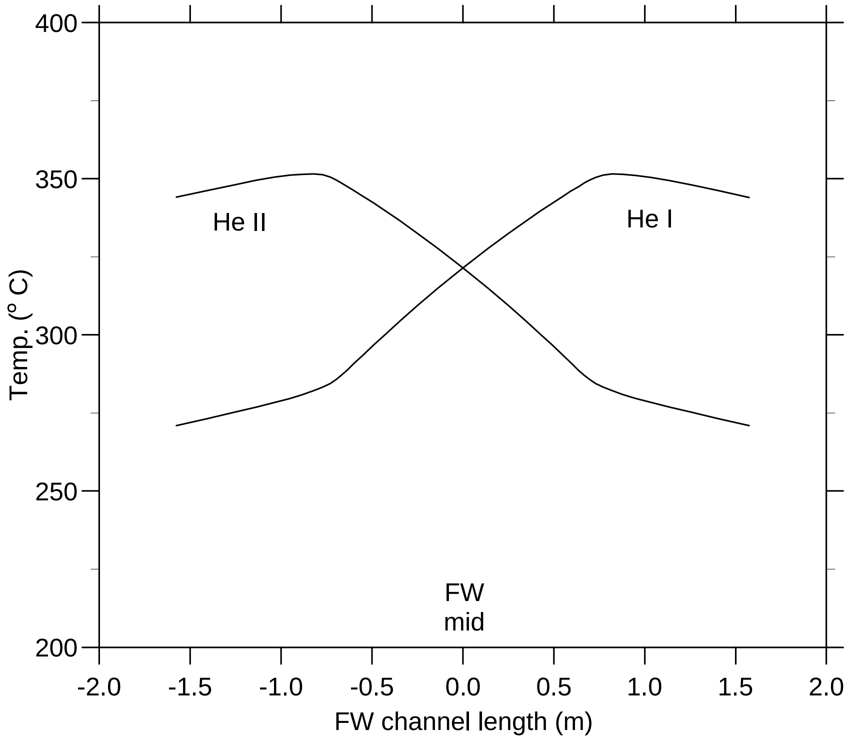


Figure 5 ODS reference case: FW coolant temperatures

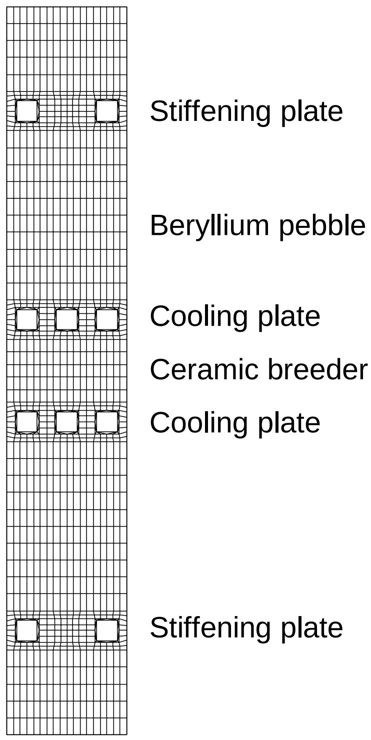


Figure 6 Toroidal-poloidal cross-section of the BZM mesh

The modelled mesh does not change in radial direction. In particular, the near-FW 180-degree bend in the ceramic-breeder container and the connection of breeding zone and FW has not been modelled. Flow in the container plates (the two in the middle) is directed to the FW in one plate and back in the other. In the stiffening plate (top and bottom plate) the flow is to the FW in one channel and back in the second channel of the same plate; between the two channels is a dummy channel that has been meshed in the figure but is modelled with void physical properties. All non-channel model faces have symmetry boundary conditions; HTCs between channels and structure have been applied, while HTCs between pebble beds and structure have been neglected. Inlet temperatures were drawn from the FWM.

The thermal conductivity of the Beryllium pebble bed is strongly dependent on the bed's deformation in the structure. Values of e.g. 8.8 $Wm^{-1}K^{-1}$ at 600°C were extrapolated from single-size Beryllium experiments [10] for an assumed interference of 0.5%. Recent experiments on interference of up to 0.45% support the view that thermal conductivity will be close to 10 $Wm^{-1}K^{-1}$ [11].

The resulting Helium temperatures are depicted in Figures 7 and 8. The most notable feature is the helium temperature maximum in the stiffening plates at about 0.2m from the FW, which is 40K larger than the outlet temperature for EUROFER and 45K for ODS. A heat flux from the hot leg to the cold leg using steel connections and the well-conducting Beryllium bed as a heat bridge is responsible for this effect. The stiffening plate temperature at the maximum location will be a limiting characteristic for the blanket concept.

In Figures 9 and 10, poloidal profiles through the three channels and for the radial location of the helium temperature maximum in the stiffening plate show steel temperatures of 525°C for EUROFER and 580°C for ODS. They indicate that there is space for an increase of the helium outlet temperature.

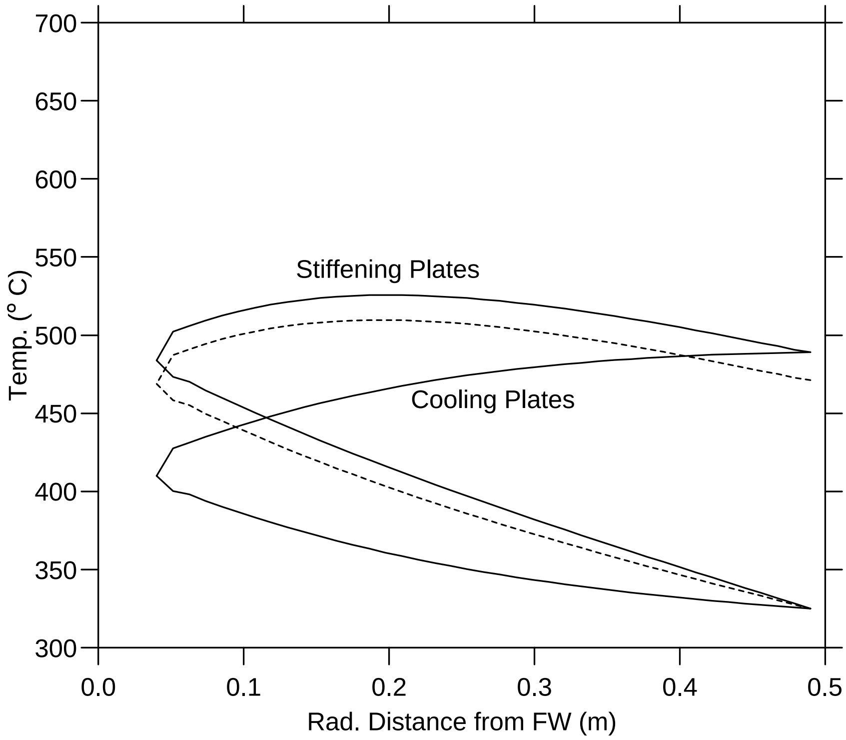


Figure 7 EUROFER reference case: cooling plate coolant temperatures

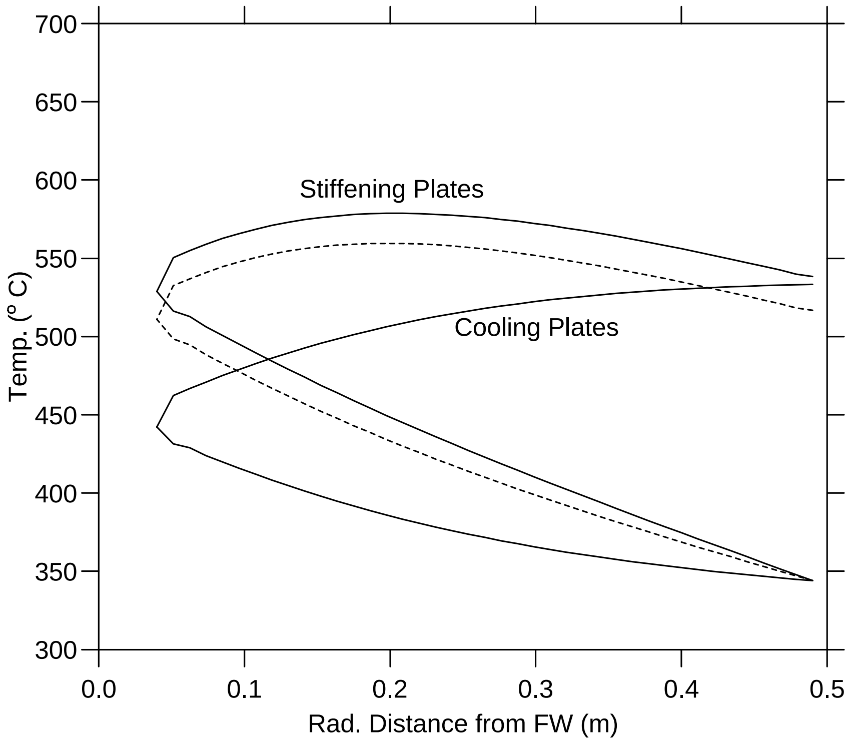


Figure 8 ODS reference case: cooling plate coolant temperatures

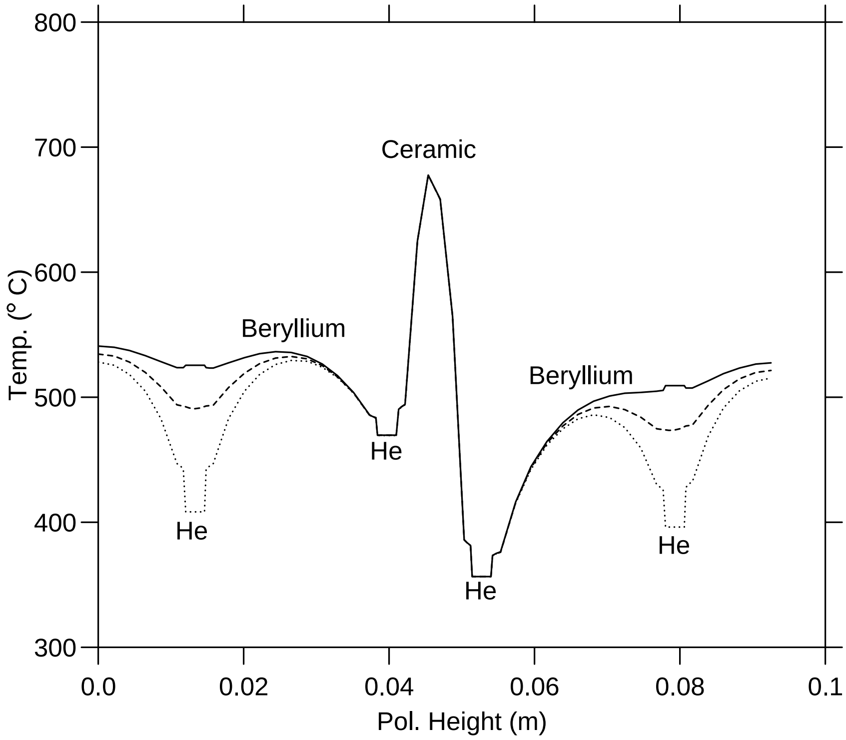


Figure 9 EUROFER reference case: poloidal temperature profile (radial position of the maximum stiffening-plate coolant temperature)

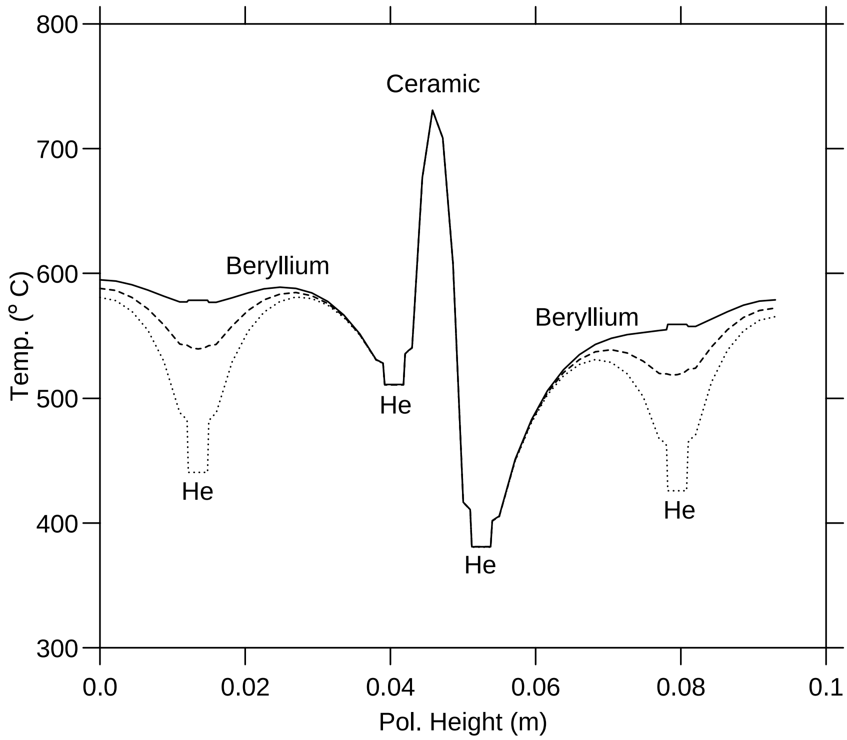


Figure 10 ODS reference case: poloidal temperature profile (radial position of the maximum stiffening-plate coolant temperature)

3.3 Thermomechanical calculations

A 66 mm high radial-toroidal slice of FW, blanket box and one stiffening plate was modelled using the CAD code CATIA as a preprocessor for the finite element solver PERMAS ver.7; the meshed section that is one toroidal half of the module is displayed in Figure 11. Pebble beds were not modelled apart from a 10mm Beryllium layer on the breeding zone side of the FW.

Structural temperatures under operating conditions were computed as an input to the thermal stress analysis. Boundary conditions for this temperature calculation were: (i) linearly scaled surface heat flux and heat source values were applied to the FW; (ii) convection of the fluid was modelled using local FW coolant temperatures from the FWM and a constant HTC from the PBM of the previous section; (iii) with the stiffening plate not being modelled in full detail, the plate temperature was set to 400°C, which is roughly in accordance with the FIDAP results of the previous section. Temperatures of the back wall (face D) were set according to neighbouring coolant channels foreseen for the I-HCPB (EUROFER/ODS):

0-40mm (from symmetry face C)	purge gas collector	450°C/450°C
40-390mm	He outlet	485°C/530°C
390-590mm	He inlet	265°C/270°C
590-765mm	FW He collector	325°C/345°C

The resulting temperature distributions are displayed in Figures 12 and 13. The maximum temperature of 491°C/556°C is in the corner of the FW.

The basic load case are then thermal stresses in the structure caused by the temperature distribution and the coolant pressure of 8 MPa. Boundary conditions for these FE calculations were: (i) the fixed origin is where faces A and C intersect with the FW face; (ii) face C is a symmetry plane; (iii) nodes on face A are fixed in z-direction; (iv) nodes on face B can uniformly move in z-direction, i.e. faces A and B stay parallel; (v) nodes on face D can uniformly move in x-direction, i.e. the back wall stays plane.

Figure 14 depicts the distribution of equivalent stress for the EUROFER reference case. FW stresses at the height of the FW webs between the coolant channels show local maxima, with the absolute stress maximum of 392 MPa occurring in the corner of the FW. The back wall sees stresses of the same order at the location of the He outlet. The maximum total stress limit $3 \cdot S_m$ at 500°C is 438 MPa. For ODS in Figure 15, the maximum stress is 529 MPa where 522 MPa are expected to be allowable.

In a coolant leak accident load case, preliminary calculations suggest that stresses are uncritical when the box is pressurised to 2 MPa; the stress peak shown in Figure 16 is localised and can be largely removed by design measures. Given the 8 MPa coolant pressure, this approach implies the provision of a pressure control system that limits the pressure in the box to the tolerable 2 MPa in case of a coolant leak.

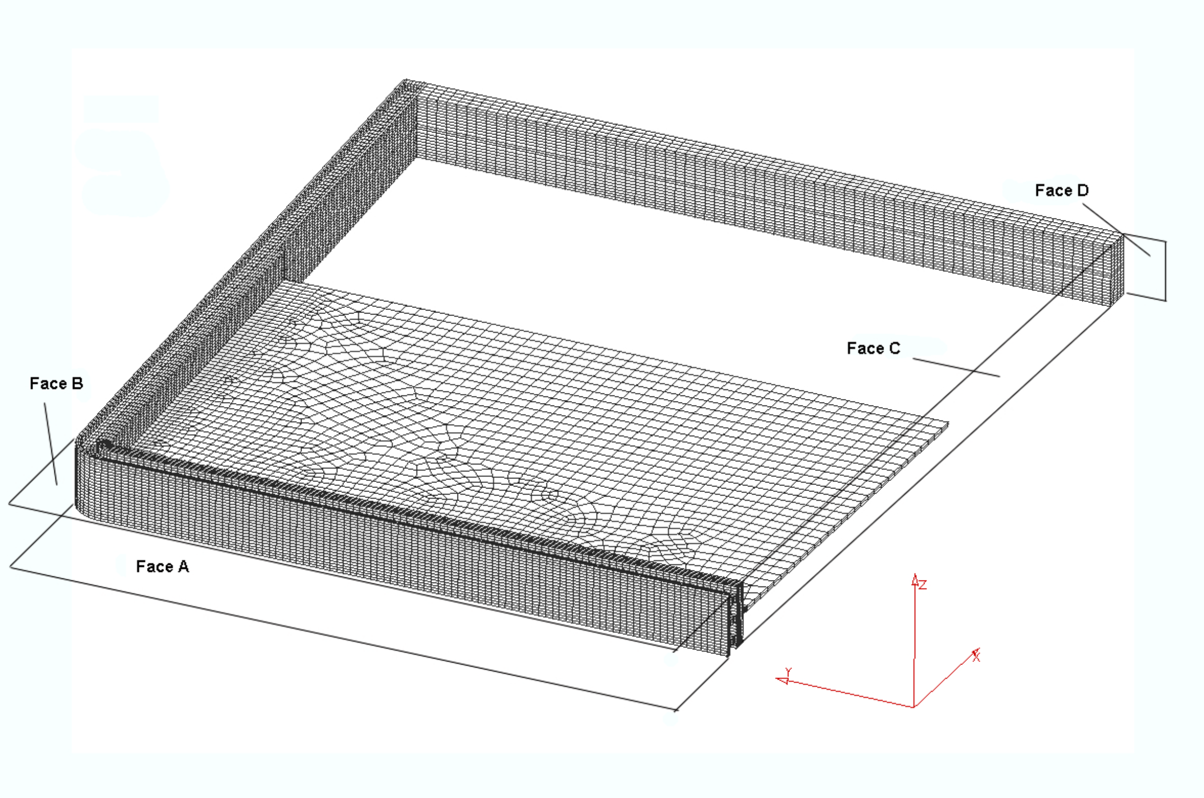


Figure 11 FE model of a slice from the blanket box

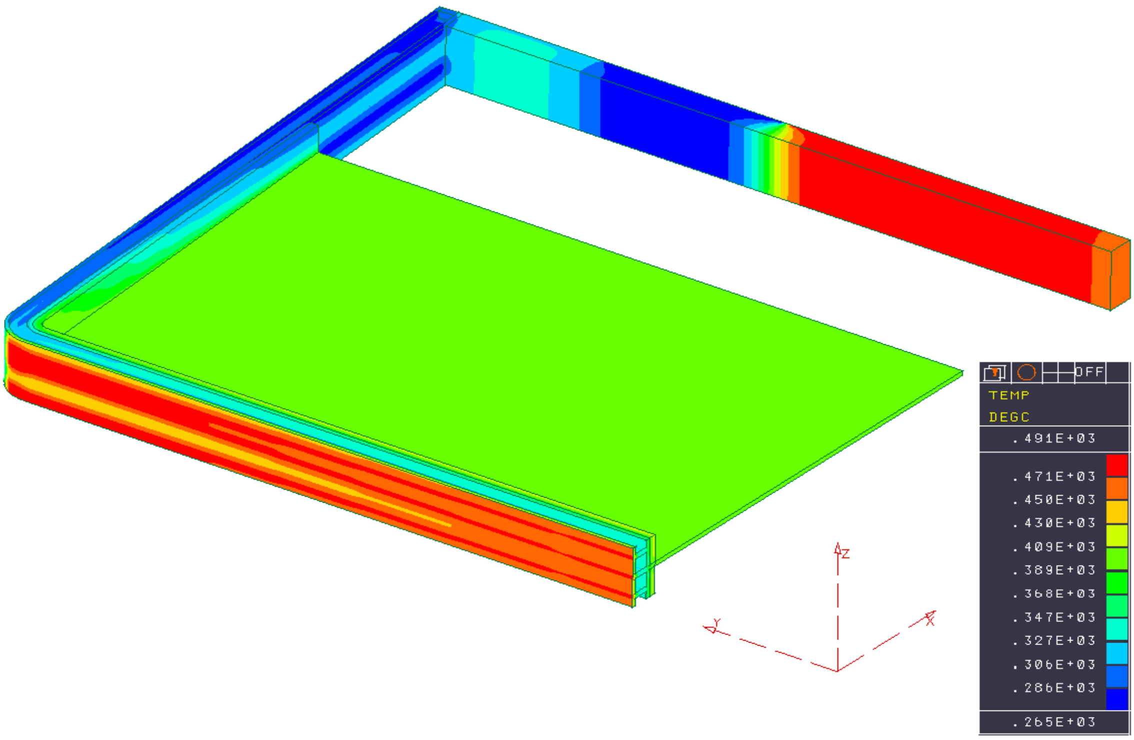


Figure 12 EUROFER case: temperature distribution in the blanket box

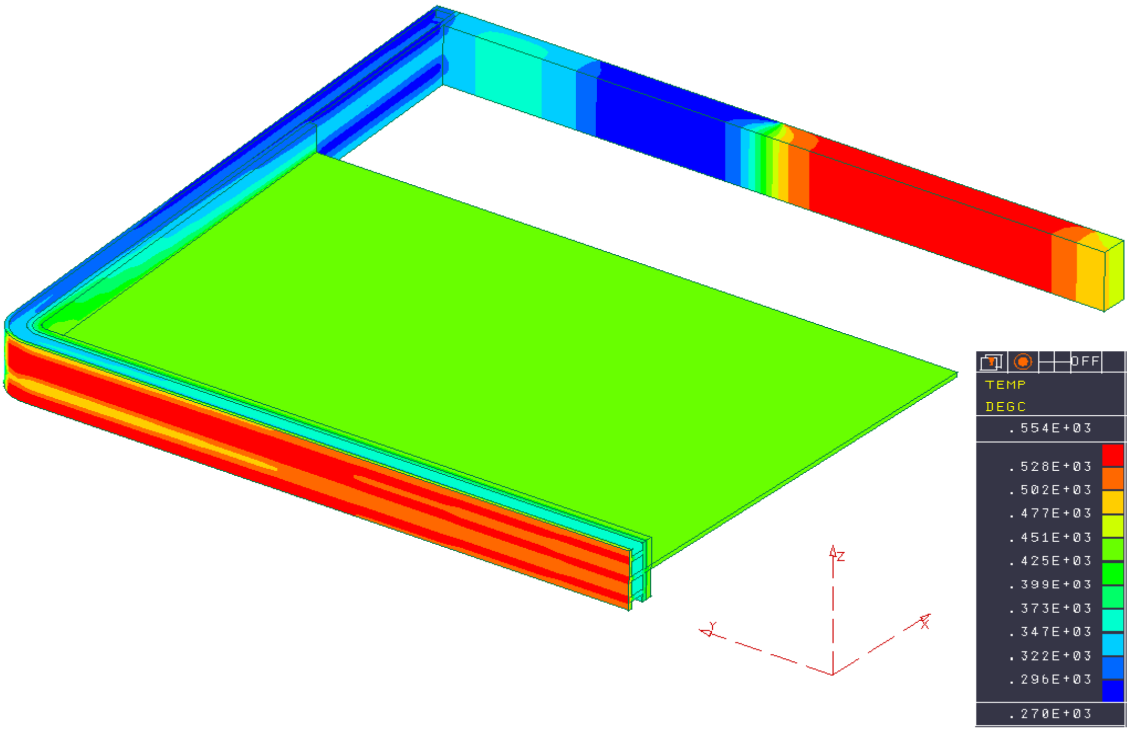


Figure 13 ODS case: temperature distribution in the blanket box

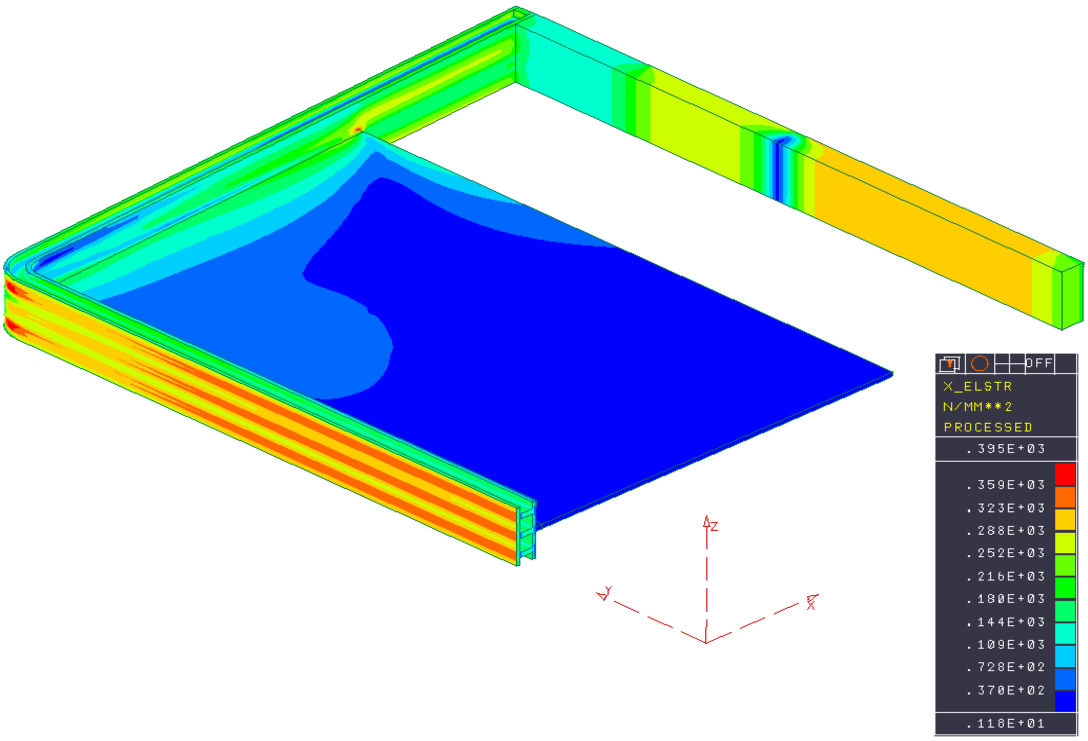


Figure 14 EUROFER case: v. Mises stress for stationary thermal load

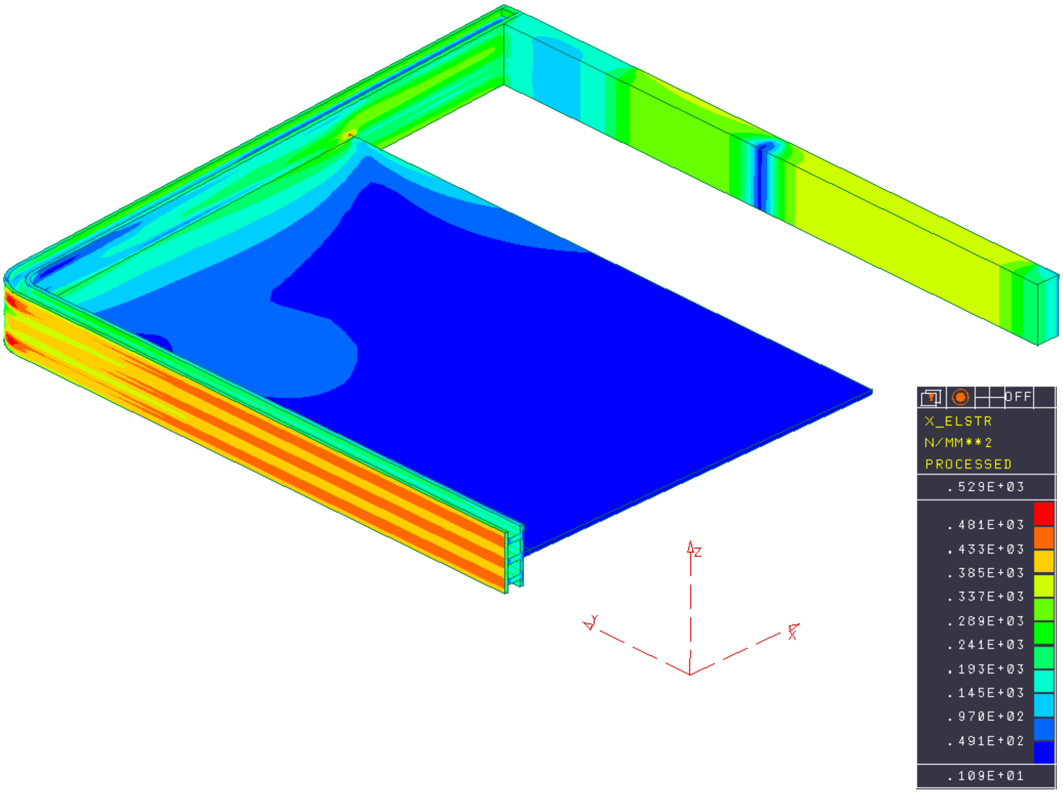


Figure 15 ODS case: v. Mises stress for stationary thermal load

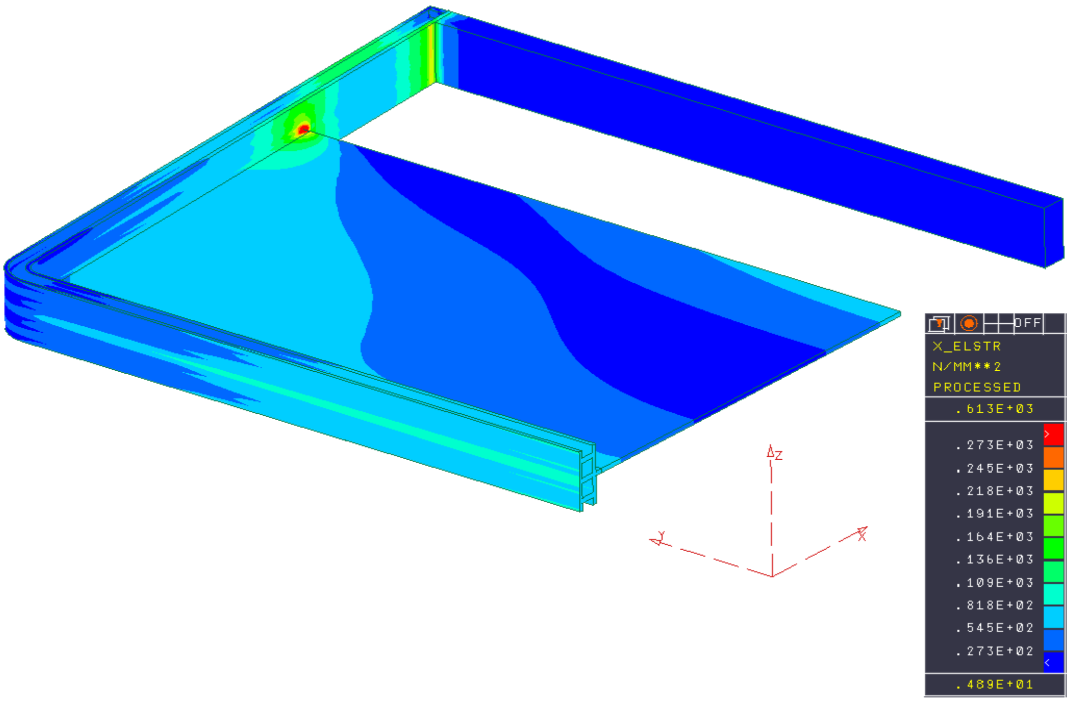


Figure 16 ODS case: v. Mises stress for 2 MPa internal box pressure

3.4 Steam circuits and power generation

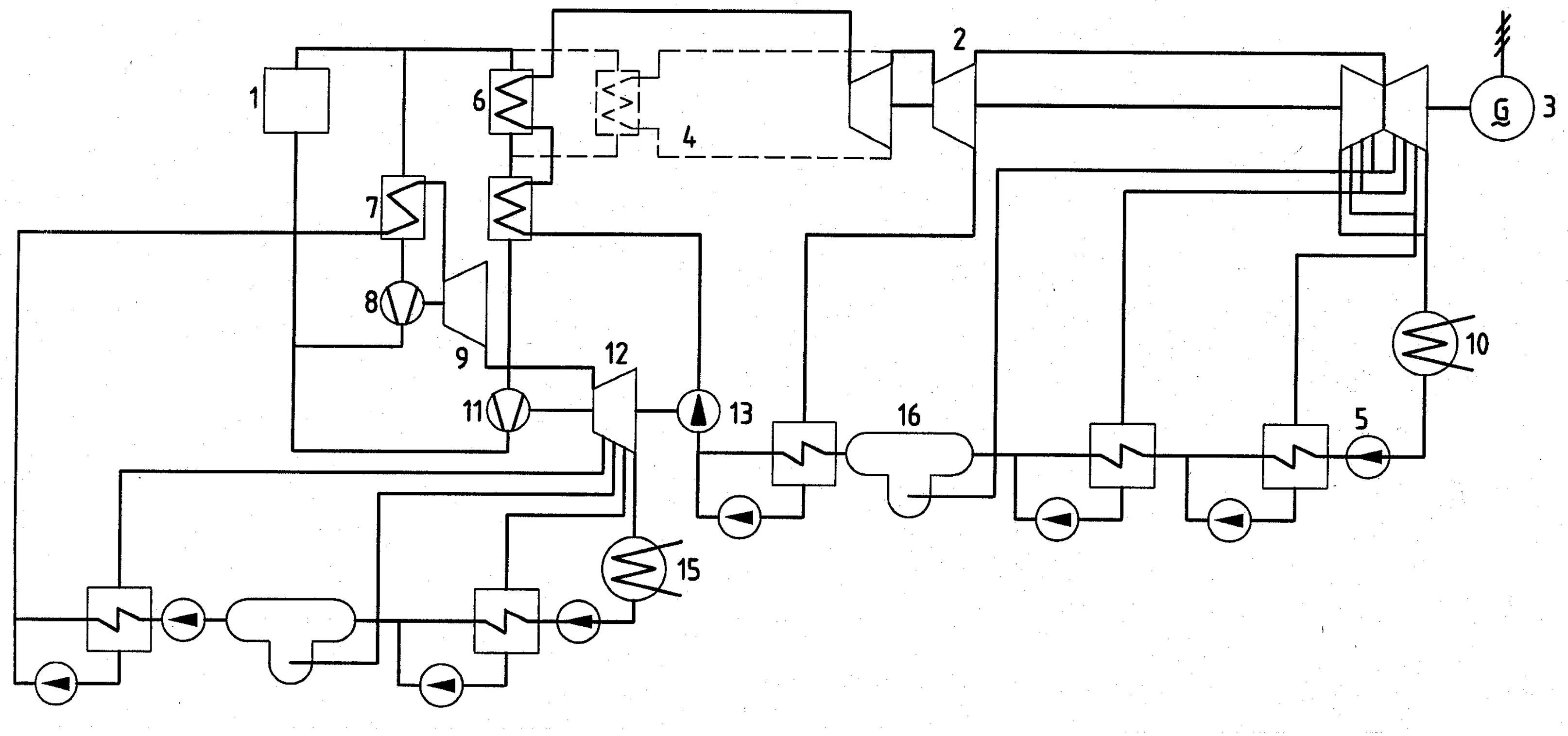
A Helium/water power generation circuit has been designed, Figure 17, to explore the achievable thermal and electrical efficiencies. The proposed circuit uses superheated steam; intermediate superheating (station 4) is an option to raise efficiency. In Figure 17, the main steam circuit has been employed for calculations, while the auxiliary circuit is meant to give an impression of how the driving of the auxiliary devices could be realised.

Eight Helium/water steam generators of equal size are foreseen for the power plant; as the key components of the power cycle they have been designed for the Helium conditions set by the blanket circuit. Figure 18 depicts the T,Q-diagram of two sizes of steam generator tailored to the needs of the EUROFER and ODS reference cases.

Employing life steam conditions of 11 MPa, 470°C for the EUROFER case and 11 MPa, 520°C for ODS, respectively, thermal efficiencies for the steam circuits including the power for the feed water pumps are 38.4% and 39.0%. Thus, overall electrical efficiencies in the blanket cycle of 35.8% and 36.7% are reachable. Intermediate superheating (station 4) is possible and would lift the electrical efficiency of the EUROFER concept to 36.5% and that of the ODS concept to 37.5%.

3.5 Tritium management

A new analysis was performed to improve the source-term and inventory evaluation for Tritium management in the HCPB blanket using Pick's model for the determination of Tritium permeation from ion impingement on the FW [12]. For the geometry and operating conditions of the HCPB DEMO blanket and MANET as structural material the Tritium inventory was re-evaluated. Surface conditions (sticking factor s) were confirmed to dominate the source term. They were varied between untreated (bare) steel ($s=3.6 \cdot 10^{-7}$) to sputter-cleaned steel (s=1) and to oxidised steel (permeation barrier). The calculated Tritium permeation rates vary between $2 \cdot 10^{-5}$ and 45 g/d. A value of 1 g/d may be reached if sputter-cleaning can achieve $s>5 \cdot 10^{-3}$ (even without accounting for coolant-side oxidation). This compares to permeation rates of 9 to 12 g/d calculated in earlier analyses for sticking factors between 10^{-5} and 10^{-4} [13]. According to this reference, the permeation rate from the purge gas system to the helium coolant amounts to 0.78 g/d. For these permeation rates, a permeation reduction factor (PRF) in the steam generators in the range of 20 is sufficient to keep the Tritium permeation to the water/steam below 20 Ci/day [13]. Such PRFs are easily achieved by Chromium oxidation of the Incoloy 800 steam generator tubes. Hence, there is evidence that the HCPB DEMO blanket does not necessitate a Tritium permeation barrier. Whether the Improved HCPB blanket possesses similarly favourable features must be clarified by future analyses of the Tritium permeation rates and inventories for the conditions of the I-HCPB including the steam generators, and the global Tritium balance.



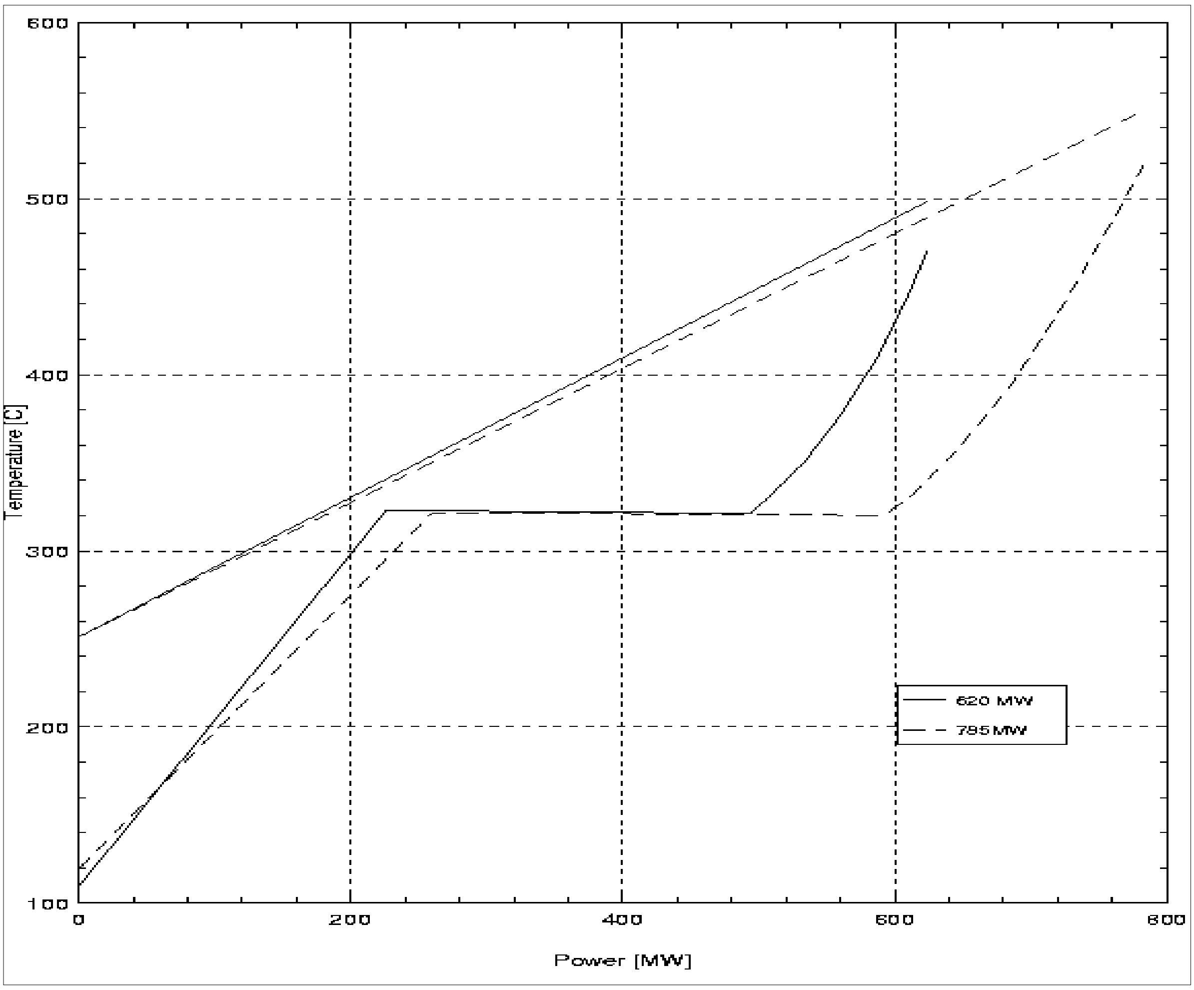
1 Blanket
2 Main steam turbine
3 Generator
4 Intermediate heat exchanger

5 Condensate pump
6 Steam generator (SG)
7 Auxiliary SG
8 Blower auxiliary circuit
9 Auxiliary turbine

10 Main condenser
11 Main blower
12 Main blower turbine
13 Main feed water pump
14 Preheater

15 Auxiliary condenser
16 Deaerator and feed water tank

Figure 17 Helium/water power generation circuit



	EUROFER	ODS
Power	620 MW_{th}	785 MW_{th}
He inlet temp.	500°C	550°C
He outlet temp.	250°C	250°C
Water inlet temp.	108°C	118°C
Water outlet temp.	470°C	520°C
Hydrostatic pressure	11 MPa	11 MPa
Pinch point distance	2.5 K	3 K

Fig. 18 Helium/water steam generator T, Q-diagram

3.6 Comments on lifetime and availability

The availability of the blanket system depends on the lifetime of its components, i.e. the time to its scheduled or – in case of a premature failure – to its unscheduled exchange, and on the time needed for the exchange operation (maintainability). Studies carried out for the DEMO blankets have shown that high reliability of the ex-vessel blanket sub-system can be achieved by providing sufficient redundancy [14]. Furthermore, easy access to the components assures good maintainability. In contrast, redundancy of in-vessel components is not possible, and all components must be available at the same time for operability of the system. In addition, maintenance of in-vessel components is a difficult and time-consuming task. Consequently, the availability of the blanket system is dominated by the in-vessel components, and the following discussion is restricted to this area.

The lifetime of the in-vessel blanket components is determined by the behaviour of the main blanket materials under the different operational loads. The dominating effect for Beryllium is the radiation-induced swelling. The envisaged use of a single-size pebble bed with a porosity of almost 40% and the rather low operational temperature provides a good potential for reaching high fluences. Ceramic breeder materials have been irradiated to high burn-up values successfully, but at low damage to burn-up ratios. The planned experiments with high-energy neutrons are expected to confirm the suitability of these materials for high fluence, too. For the structural materials under consideration (EUROFER and ODS) it is expected that a damage limit of 150 dpa (about 15 MWa/m^2) can be demonstrated in the future. This means a lifetime of the blanket segments of 3.4 to 4.3 FPY (= full power years).

To reach a blanket system availability of e.g. 0.8, the time needed to replace the complete blanket must be in the range of 9 to 13 months, or – if a cyclic 4-step replacement scheme is assumed – 2 to 3 months for one replacement step. In a well-designed plant with a developed technology the unscheduled down-times caused by premature blanket failures should be significantly less than the scheduled down-times. Assuming that a defective blanket segment can be replaced in half the time of one cyclic replacement step, i.e. 1 to 1.5 months, leads to the requirement that the failure rate for a single blanket segment must be significantly below 0.01 a^{-1}. Estimates for the HCPB DEMO blanket yielded an average blanket segment failure rate of this order of magnitude [14]. The similarity of key design features, particularly the use of hot iso-statically pressed (hipped) FW and cooling plates and the avoidance of fusion welds in regions of high neutron flux suggests that the good availability found in the DEMO HCPB concept study [1,14] applies to the I-HCPB concept. With the poor fusion weldability of ODS in mind it is now proposed to hip the stiffening plates into the blanket box, too. Experiments on this issue are carried out at the Forschungszentrum in the frame of the current blanket programme.

Whether the replacement times mentioned above can be realised depends on the blanket maintenance concept which can be elaborated only when a preliminary reactor design is available. The general design of the I-HCPB offers the possibility to divide the blanket into a small number of large units that can be handled as one piece. This is a precondition for optimising maintenance operations.

4 Limits of the proposed concept

Tab.1 specifies the limiting material temperatures and allowable stresses that impose operating boundaries on the I-HCPB concept. Key locations for checking are (i) both pebble beds for limiting temperature maxima inside; (ii) the cooling plates for limiting temperatures at their pebble bed interface; (iii) the plasma-facing FW for limiting stresses.

(i) The most notable benefit of the proposed breeding region design is the removal of design limits on minimum pebble bed heights. Preliminary one-dimensional calculations suggest that bed heights of 6mm for the ceramic breeder (probably the lower limit on the breeder bed at

10 times the pebble diameter) and 2x22.5mm for Beryllium can accommodate a maximum neutron wall load of 6MW/m^2. A coolant temperature increase of 320K was assumed for this calculation. It is important to note that the steel fraction was not increased for this case, which implies that Tritium breeding self sufficiency is still achieved.

(ii) Maximum cooling plate temperatures appear in the stiffening plates; essentially, they are determined by the helium outlet temperature. Calculations for the reference designs suggest that the maximum ODS cooling plate temperature is 70K below the limiting 650°C. This implies that the Helium outlet temperature could be lifted.

(iii) Stresses in the FW are due to a large radial temperature gradient caused by the surface heat flux. By reducing the FW between cooling channels and plasma interface to 3mm it is possible to accommodate a surface heat flux of about 1MW/m^2 and still keep surface temperatures of about 550°C, implying stresses close to the level of the ODS reference case.

Beyond these design measures, a high-temperature helium cooled divertor could make the divertor power available to the steam circuit and raise the electrical net efficiency significantly.

5 Open issues

Thermal and mechanical behaviour of Beryllium and Li_4SiO_4 pebble beds at operating conditions is the most important open issue for the I-HCPB concept. Bed models based on the so-called Drucker-Prager correlation and using the latest results from oedometer compression tests have recently been introduced into mechanical analyses with the finite element code ABAQUS. Even so, more material data at relevant operating conditions, e.g. orthosilicate creep data and Beryllium heat conductivity at relevant bed deformations, are needed before a reliable prediction of temperatures and stresses in the blanket can be made. Secondly, the life time of breeder and Beryllium pebbles is an issue that requires further experiments, e.g. high-energy neutron irradiation in the near future. Beryllium life is much influenced by irradiation swelling, but this issue is less concerning for the single-size pebble bed than it would have been for a much denser binary Be bed. Finally, different materials are currently used under the name of ODS steel; use in a fusion reactor environment would require a definition of a fusion ODS steel, its characterisation and the development of a blanket manufacturing methodology. Irradiation experiments up to high fluences have to be carried out for both materials.

6 Conclusions

An improved HCPB design has been proposed that lifts limitations of the DEMO HCPB while keeping design features that have been found useful. In particular, (i) an increased helium outlet temperature; (ii) flexibility of pebble bed heights; (iii) a nearly halved pressure drop in the blanket; and (iv) simple single-size Beryllium beds have been achieved while blanket box structure and manufacturing concepts have been kept.

The merit of the resulting blanket concept has been qualified by means of two reference cases for alternative structural materials EUROFER and ODS; preliminary calculations suggest strongly that these designs fulfil all requirements for a blanket at max. neutron wall loads of 3.5 MW/m^2 and 4.4 MW/m^2, respectively, and raise the electrical net efficiencies of the DEMO HCPB blanket by about 7% to 36.5% and 37.5%.

A minimum breeder-bed height limitation to ten times the pebble diameter suggests that the pebble bed concept has a limit of about 6 MW/m^2 max. neutron wall load; for the assumed FW design, stress limits imply that a max. surface heat flux in the region of 1 MW/m^2 could be accommodated.

It is a strength of the proposal that it produces an attractive blanket design on the basis of the readily available EUROFER; the concept can be enhanced by the use of an ODS steel for fusion applications, without depending on the successful development of such a material.

Although significant development work is necessary to qualify the ceramic breeder and Be, it seems that eventually blanket lifetime is dominated by the structural material. Even assuming that tolerable damage of 150 dpa can be demonstrated in the future, the lifetime limitation of exchangeable components requires an efficient maintenance concept which can only be developed in connection with a reactor design.

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