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STATUS OF EC SOLID BREEDER BLANKET DESIGNS AND R & D FOR DEMO FUSION REACTORS

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Abstract

Within the European Community Fusion Technology Program two solid breeder blankets for a DEMO reactor are being developed. The two blankets have various features in common: helium as coolant and as tritium purge gas, the martensitic steel MANET as structural material and beryllium as neutron multiplier. The configurations of the two blankets are however different: in the B.I.T. (Breeder Inside Tube) concept the breeder materials are $LiAlO_2$ or Li_2ZrO_3 in the form of annular pellets contained in tubes surrounded by beryllium blocks, the coolant helium being outside the tubes, whereas in the B.O.T. (Breeder out of Tube) the breeder and multiplier material are Li_4SiO_4 and beryllium pebbles forming a mixed bed placed outside the tubes containing the coolant helium.

The main critical issues for both blankets are the behavior of the breeder ceramics and of beryllium under irradiation and the tritium control. Other issues are the low temperature irradiation induced embrittlement of MANET, the mechanical effects caused by major plasma disruptions, and safety and reliability.

The R & D work concentrate on these issues. The development of martensitic steels including MANET is part of a separate program. Breeder ceramics and beryllium irradiations have been so far performed for conditions which do not cover the peak values injected in the DEMO blankets. Further irradiations in thermal reactors and in fast reactors, especially for beryllium, are required. An effective tritium control requires the development of permeation barriers and/or of methods of oxidation of the tritium in the main helium cooling system. First promising results have been obtained also in field of mechanical effects from plasma disruptions and safety and reliability, however further work is required in the reliability field and to validate the codes for the calculations of the plasma disruption effects.

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1 *Introduction*

In the framework of the European Community Fusion Technology Program four blanket concepts for a DEMO reactor are being investigated. DEMO is the next step after ITER. It should ensure tritium selfsufficiency and operate at coolant temperatures high enough to have a reasonable plant efficiency [1].

These investigations, which include R & D work, aim to a well founded choice among various concepts in view of developing blanket modules to be tested in ITER only for the most promising blankets. It is foreseen to reduce the number of blanket options to two in 1995. To facilitate a comparison of the four blanket designs common specifications have been suggested by the European Test Blanket Advisory Group. Table 1 shows the most important ones. The resulting fusion power is about 2200 MW [1].

Two of the four blankets are based on the use of a liquid metal breeder and will be dealt with in a companion paper at this Conference [2]. The present paper deals with the two solid breeder concepts. They have many features in common: both use high pressure helium as coolant and helium to purge the tritium from the breeder material, martensitic steel as structural material and beryllium as neutron multiplier. The configurations of the two blankets are, however, different: in the B.I.T. (Breeder Inside Tube) concept the breeder material is $LiAlO_2$ or Li_2ZrO_3 in the form of annular pellets contained in tubes surrounded by beryllium blocks, the coolant helium being outside the tubes, whereas in the B.O.T. (Breeder Out of Tube) the breeder and multiplier material are Li_4SiO_4 and beryllium pebbles forming a mixed bed placed outside the tubes containing the coolant helium. The paper gives a short description of the two designs, discusses the principal critical points and illustrates the related R & D program.

The associations primarily involved in the design and R & D work are CEA, ENEA, and KFK while ECN and SCK/CEN contribute to the irradiation program on breeder materials and beryllium respectively.

2. *The European Solid Breeder Blankets*

The combination ceramic breeder materials and helium coolant was an early choice (1983 - 1984) of the European Fusion Technology Program (EFTP).

Ceramic materials have high melting points. They are not very chemically active and of course do not present MHD problems. The best known lithiated ceramics are lithium oxide Li_2O or the ternary ceramics $Li_xM_yO_z$ (M = metal). These materials have been extensively used in some countries for tritium production in the frame of their defense programmes. Tritium extraction from the ceramics by means of a helium purge gas flow has been extensively and successfully used in in-pile and out-of-pile experiments. A further earlier decision within the EFTP was to concentrate on the ternary ceramics, because Li_2O is more chemically active, with more compatibility problems with structural and breeder materials, and also because Li_2O tends to swell more when subjected to neutron irradiation [3]. Among the ternary lithiated ceramics, aluminates and silicates were the most actively studied so far, thus it was decided that CEA-ENEA would concentrate on aluminate investigations and KfK on the silicates. Later both groups started some activity on lithium metazirconate Li_2ZrO_3 as well. Namely on metarzirconate pellets (CEA) or pebbles (KfK). Since 1992 CEA is focusing its activities on Li_2ZrO_3.

Helium is better suited than water as the coolant of a lithium ceramic blanket. Water reacts with lithiated ceramics producing lithium hydroxide, which has a relatively high vapor pressure. This, besides affecting the integrity of the ceramic, could cause unduly high lithium transport due to the temperature gradients present in the blanket. Helium, on the contrary, is an inert gas and, as the experience with helium cooled fission reactors shows, can be kept extremely pure (total amount of impurities < 1 ppm) even in large and complex circuits. Unlike water, helium can be kept at high temperatures without need to increase the pressure, thus the problem of keeping the minimum temperature in the breeder material above a certain level, to ensure low tritium inventories in the breeder, becomes much easier, as thermal insulating gaps between breeder and coolant are not required. A further advantage of helium is that leakages to the plasma chamber have much less severe safety consequences than water leakages.

The only two element which in practice could be used as neutron multipliers are lead and beryllium. Due to the low melting point (327 °C), it is very difficult to keep lead in the solid state in a ceramic blanket: changes of phase during operation are practically unavoidable. Furthermore beryllium is a better neutron multiplicator than lead and, as a good neutron moderator, decreases the neutron leakage from the blanket. For these reasons beryllium is generally the preferred choice. This material has been extensively studied 25 - 35 years ago as a possible cladding material for fission reactors and it has been used as reflector in various material testing reactors. It is an excellent neutron multiplier. Its high melting point, high thermal conductivity and low specific weight are considerable advantages. There are, however, problems related to its behavior under neutron irradiation. Also the choice of beryllium as neutron multiplier favors the use of helium rather than water as coolant. Indeed water steam reacts with beryllium at temperatures ≥ 600 °C forming hydrogen. In case of porous beryllium the reaction is selfsustaining [4].

The common specifications for the European DEMO blankets (Table 1) imply an average neutron wall fluence of 5 MWy/m^2, which corresponds to a peak irradiation damage of 70 dpa. For this reason an austenitic steel cannot be used as it would swell too much under irradiation. Within the European Fusion Technology Program the martensitic steel MANET has been chosen as the structural material for all four blanket designs. This steel is similar to the martensitic steels which are being investigated within the various fast reactor programs and promises to be able to withstand very large neutron fluences with very small amounts of swelling. The R. & D. work for the European martensitic steels, including MANET, is part of a separate program within the EFTP, parallel to the blanket program, and will not be discussed in the present paper.

In the following two sections the two European Solid Breeder Blankets will be shortly described. Table 2 shows the main data of the two blankets.

2.1 *The European BIT blanket with $LiAlO_2$ (or Li_2ZrO_3) pellets [1, 5, 6]*

A blanket segment of this design is made of banana-shaped poloidal breeder modules arranged in rows inside the segment box (Fig. 1).

The design of the generic breeder module depends on its location in the segment box:

The generic breeder module of the back rows consists of a banana-shaped martensitic steel pressure vessel on the outer wall of which sintered blocks of beryllium are either brazed or mechanically attached (see Fig. 2). Inside this vessel, martensitic steel breeder tubes equipped with helical wire spacers and containing a stack of annular $LiAlO_2$ (or Li_2ZrO_3) pellets are arranged in a shrouded bundle surrounded by a baffle.

Coolant and purge gas are both fed and collected through the upper end of the module. The coolant, helium at 6 MPa, enters the module at 250 or 310 °C (depending on the cooling scheme). It first extracts heat produced in the vessel wall, the beryllium and the baffle while flowing downwards in the annular space between the vessel wall and the coaxial baffle. Once having attained the bottom end of the module, it reverses direction and flows upwards through the breeder tube bundle before exiting the module at 520 °C. The stagnant helium gap between the bundle hexagonal shroud and the circular baffle provides a thermal insulation which limits the heat losses from the hotter upflowing to the colder downflowing coolant.

The purge gas, helium at a pressure close to the coolant one, flows inside the breeder tubes through the central hole of the annular pellet stack. After being fed to an inlet chamber located at the top end of the module, it flows first downwards in half of the bundle tubes, reaches an intermediate chamber welded to the bottom end of the tubes, and then flows upwards up to an outlet chamber connected to the outlet header.

The generic breeder module of the front rows differs from the above design mainly by the arrangement of the beryllium blocks (see Fig. 3). Indeed, Be/Steel brazing or mechanical attachment being not well suited to the accommodation of the large swelling rates expected from this material in regions of high fluence, the beryllium, in form of quarters of an annulus sintered blocks, has been placed inside the pressure vessel between two concentric steel tubes surrounding the breeder tube bundle. A narrow gap is provided between the Be blocks and the outer tube so as to accommodate Be swelling, while the outer side of the inner tube is equipped with helical wire spacers which create a stagnant helium gap for thermal insulation purposes. If needed, this beryllium zone can be actively purged (by helium drawn directly from the coolant at the bottom of the module) in order to minimize the permeation to the coolant of the tritium produced within the beryllium. Beryllium and breeder purge circuits are connected in series.

Another difference between front and back row breeder modules relates to the breeder tubes which are U-shaped (in order to avoid the welds associated with an intermediate purge chamber) and spaced with grids.

The module assembly is contained in the segment box whose first and side walls are equipped with toroidal cooling tubes (see Fig. 4) connected to poloidal headers located at the back of the blanket. In order to simplify the plant, the same coolant is used first to cool the first wall and box side walls, and then the blanket proper. However, the in-series connection is made outside the blanket.

2.2 *The European BOT blanket with Li_4SiO_4 (or Li_2ZrO_3) pebbles [7, 8]*

Fig. 5 shows a vertical cross section across the right side of the torus. Fig 6 shows an isometric view of a segment of the outboard blanket. The blanket is contained in the segment box, which is divided in 10 elements. In each element there are 9 stiffening plates placed in radial-toroidal planes. The breeding /multiplier material is in form of a mixed bed of beryllium and Li_4SiO_4 pebbles (Fig. 7). The main structure of the bed is made by 2 mm beryllium pebbles, which fill up 63 % of the bed volume. Between the large Be-pebbles there are 0.1 - 0.2 mm pebbles which take up 17.5 % of the bed volume. In the front part of the blanket (radial thickness = 33 cm) 50 % of the little pebbles are of beryllium and 50 % of Li_4SiO_4, in the back part (radial thickness = 17 cm) they are of Li_4SiO_4. Due to the presence of the stiffening plates, the segment box is strong enough to withstand the full helium coolant pressure.

The high pressure helium coolant tubes, divided in two completely separated independent systems, enter the segment from the top then are routed upwards in poloidally running distribution ducts (Figs. 5, 6), from which the tubes cooling the box walls, including the first wall, depart. After having cooled the box walls, the helium enters poloidal manifolds from [illegible] distributed to the coolant coils of the beryllium/Li_4SiO_4 pebble beds (Fig. 7). Thereafter, the coolant is collected in poloidal tubes and exits the segment at the top (Figs. 5, 6). The box walls are cooled alternatively by the two independent cooling systems running in opposite directions to avoid large poloidal deformations of the first wall due to the temperature increase of the helium coolant in toroidal direction. Also in the pebble bed region two adjacent layers are alternatively cooled by the two independent systems. This time, however, running in the same direction. The tritium purge flow at 0.1 MPa runs in radial direction through the pebble beds from the front towards the back of the blanket (Fig. 7). A separate high pressure helium system parallel to the main one, is used to cool the shield at the back of the segment (Fig. 5).

Fig. 8 shows a toroidal-radial cross section of the inboard blanket. The arrangement of the cooling coils and of the beryllium-Li_4SiO_4 pebble beds is similar to that of the outboard blanket.

3. *Main critical issues of two European Solid Breeder Blankets*

3.1 *Behavior of the structural material under irradiation*

A fully martensitic steel containing 10 - 11% Cr and additions of approximately 0.6 % Mo, 0.65 % Ni, 0.25 % V and 0.15 % Nb, referred to as MANET, has been evaluated among other martensitic steels for first wall and breeder structural component applications in the European DEMO blankets.

This high chromium martensitic steel has many advantages over the austenitic steels, including lower decay heat, increased resistances to thermal stress development and irradiation-induced high temperature (helium) embrittlement and void swelling.

No major difficulties have been encountered in the primary fabrication of MANET; however, secondary fabrication poses some problems in that pre-weld heating is sometimes required and post-weld heat treatment at ≥ 700 °C is essential to temper the brittle martensite formed in the weld and heat-affected zones.

The main disadvantage of the ferritic and martensitic steels is that they suffer radiation hardening (as evidenced by increases in tensile yield stress) and embrittlement (increases in ductile-brittle transition temperature and reductions in upper shelf energies in Charpy V-notch impact tests); the irradiation-induced reductions in the fracture toughness are also manifested in compact tension tests. It is necessary to demonstrate and ensure that the fracture toughnesses of the steel plates and welds are not impaired to the extent that there is a high probability of failure by fast brittle or ductile fracture during the fusion reactor start-ups and shutdowns and at the coolant temperature during operation.

The magnitudes of the radiation hardening and embrittlement are dependent on the neutron fluence and spectrum, irradiation temperature, steel composition and structure. The extents of the hardening and embrittlement are greater following irradiation at the lower temperatures in the range 250 to 550 °C, with recovery of the radiation damage occurring above about 400 °C. Thus, the irradiation-induced increase in the ductile-brittle transition temperatures of the MANET steel decreases from about 280°C, after exposure to fluences of ≤ 14 dpa at 300 °C, to 30 °C following irradiation at about 500 °C in HFR [9]. Recent data have also shown that fully martensitic steels containing 7 - 9 % Cr exhibit significantly reduced hardening and embrittlement than the higher Cr steels following irradiation to 7 and 10 dpa at 365 °C with a low He: dpa ratio in a fast reactor [10]. In particular, a 9 % Cr - 2WVTa steel showed virtually zero embrittlement after irradiation to 7 dpa and only a 15 °C increase in DBTT after 13 dpa at 365 °C [11]. However, it is necessary to demonstrate that this increased irradiation resistance, which is associated with greater micro-structural stability, is maintained following irradiation at temperatures of ≤ 300 °C and at the higher He: dpa ratios typical of fusion.

3.2 *Behavior of breeder ceramics under irradiation*

Extensive investigations of the out-of-pile and in-reactor behavior of the three breeder ceramics have been performed [12]. Fabrication routes have been established and the out-of-pile thermal properties measured. Thermomechanical tests have been carried out on the ceramic specimens with geometrical forms and structures relevant to the application; these are important considerations as, for instance, small pebbles of high density Li_4SiO_4 have proved to be superior in this respect than sintered Li_4SiO_4 of lower density [13] or of larger Li_4SiO_4 pellets [14]. Irradiations have been performed in thermal as well as fast reactors and thereby it has been possible to obtain data on the tritium release, mechanical stability (swelling, fragmentation) and in-pile compatibility with the structural materials and beryllium [12, 15].

The two parameters characterizing the irradiation damage are the lithium burn-up (referred to total lithium content) and the displacement per atom (dpa). For the European DEMO reactors, with an average neutron wall fluence of 5 MWy/m^2, the peak burn-up would be about 10 % for Li_4SiO_4, about 19 % for Li_2ZrO_3 and 25 % for $LiAlO_2$ (see Table 2). The peak burn-ups achieved to date for these ceramics are in the range 3 - 4 %, in irradiations carried out mainly in thermal reactors, where the dpa rates are less than in a fusion blanket and the lithium burn-up may be inhomogenous, especially for larger samples.

Further in-pile tests are necessary with ceramic samples of the correct dimensions, geometrical shape and structure up to the peak values of lithium burn-ups and dpa's expected in the Demo blankets, over the complete temperature range of application in the blanket with the appropriate composition of purge gas and in the presence of the relevant structural material and, if necessary, beryllium.

3.3 *Behavior of beryllium under irradiation*

The behavior of beryllium under irradiation, particularly with respect to swelling, embrittlement and tritium release, is probably the key issue for the solid breeder blankets. Swelling of the beryllium results from the helium produced by the (n, 2n) reaction while tritium is produced by secondary reactions. The available data on these effects are not adequate for the design of a fusion blanket, for the following reasons:

a) Beryllium was extensively evaluated as a potential fuel element cladding material for thermal reactors about 25 - 40 years ago and has subsequently been used as a reflector in material testing reactors. In the meantime, however, the quality of the beryllium produced by powder metallurgical routes has improved considerably due to the developments in powder production and consolidation techniques. Thus it is now possible to obtain commercially beryllium which is much more isotropic and containing reduced amounts of oxide. There are already indications that these new grades of beryllium are less brittle when irradiated at high temperatures [16].

b) Most of the irradiations at high temperatures have been performed to relatively low neutron fluences ($\leq 10^{22}$ cm^{-2}, E > 1 MeV) whilst the irradiations to higher fluences (up to 5×10^{22} cm^{-2}, E > 1 MeV) have been carried out at low temperatures and the swelling data obtained by post - irradiation annealing at high temperatures [17, 18]. The out-of-pile annealing data may not be relevant for the assessment of the swelling during high temperature in-pile irradiations to high fluences.

Two different approaches have been proposed for the two European Solid Breeder Blankets to solve the problem of beryllium swelling. In the BIT concept beryllium is used, in the regions of higher fluences and temperatures where the highest swelling is expected, in form of small slabs, whereby a gap is left to accommodate the beryllium swelling. As the swelling closes the gap the beryllium temperature decreases and this reduces the further swelling rate of beryllium. In the BOT concept beryllium is used in form of small pebbles. The pebble bed is constrained by well cooled walls, thus the swelling results in an increase of the contact surface of the pebbles and thus of the thermal conductivity of the bed. The temperature decreases, with, also here, a consequent reduction of the swelling rate. The use of beryllium in form of relatively small pebbles offers the advantage of avoiding the problem of differential swelling and excessive thermal stresses, particularly as it becomes progressively more brittle during irradiation. Excessive thermal stresses and differential swelling due to thermal gradients can also be avoided with beryllium in form of small slabs.

The irradiation-induced low and high temperature embrittlement is very difficult to model and irradiation experiments are essential. In chapter 4 the European program to investigate beryllium embrittlement will be illustrated.

Although the tritium production in beryllium is less than 1 % of that produced in the ceramic breeder, 2 - 3 kilograms of tritium could accumulate in the beryllium at the end of the Demo blanket life if the tritium release is slow. This is a safety problem.

There are two possible approaches for reducing the safety concerns related to the tritium inventory in beryllium. Recent experimental data show that porous beryllium releases tritium at lower temperatures than the fully dense material [19]. The alternatives are to use beryllium with very low oxygen content (BeO $<$ 300 ppm) so as to minimise the chemical trapping or a relatively high amount of oxygen (BeO $>$ 1.5 - 2 %) so that it would require a long time (hours) at high temperature for the tritium to be released in the event of an accident [20].

3.4 *Tritium Control*

The tritium control in solid breeder blankets has been examined in some detail for the two European ceramic Demo blankets [7, 21, 22]. The main objective of tritium control is to limit the tritium leakage from the helium coolant system to the water/steam cycle by permeation through the steam generator. The potential sources of contamination of the helium are the tritium permeation through the first wall and the walls separating the ceramic breeder and the beryllium from the helium coolant. The tritium permeation through the first wall has been estimated using the DIFFUSE code to be between 1 and 100 g/d [21]. Clearly the upper value of permeation would not be acceptable for any blanket, as it would imply that 1/3 of the tritium produced in the blanket would permeate through the first wall to the coolant. The tritium permeation through the first wall, a problem common to all the blanket concepts, depends mainly on the surface state of the first wall [21]. If, for instance, plasma sprayed beryllium, which at present appears to be the preferred solution of first wall protection, is used, the recycle rate to the plasma of deuterium and tritium is very high and the resulting permeation to the coolant should be small [23].

The tritium permeation from the breeder has been calculated to be 2.9 and 25 g/d for the BOT [7] and BIT blanket [21] respectively. There are two possibilities for reducing the tritium losses through the steam generators to the water/steam circuit to an acceptable value of 10 - 20 Ci/d, namely using a permeation barrier and /or oxidizing the tritium in the main coolant system.

Al_2O_3 permeation barriers, which can be formed by aluminising, are used industrially and have been shown to be stable to mechanical and thermal shocks. Laboratory tests have shown that they can reduce the tritium permeation through the austenitic and martensitic steel by up to four orders of magnitude [24]. This method has been employed successfully in the BEATRIX II irradiation [25], but it may not be applicable for the long and narrow inner walls of the first wall coolant channels and other techniques have to be envisaged such as steam oxidation [21].

Oxidation of the tritium in the main helium cooling system will ensure that the tritium is in the form of HTO, which does not permeate through the steel. This could be done by adding relatively small amounts of O_2 or H_2O to the coolant and have catalyzers in the main coolant system which would promote the oxidation of HT to HTO.

3.5 *Mechanical effects caused by major plasma disruptions*

As for the other blankets, major plasma disruptions could cause unduly high stresses and deformations in the blanket structures. These are caused by the interaction of the currents, induced by the rapid variation of the magnetic field due to the plasma current decrease during the disruption, with the toroidal and poloidal magnetic fields used for the plasma confinement. Generally the highest stresses occur in the segment box surrounding the blanket.

The evaluation of the forces and stresses caused by the plasma disruptions is very important as they could seriously damage the blanket and make the removal of the blanket segments impossible.

3.6 *Safety and reliability*

3.6.1 *Safety*

As for the helium cooled fission reactors, the most dangerous accidents are the loss of flow (LOFA) and the loss of coolant accidents (LOCA). Although at first sight these accidents appear not to be so problematic for fusion blankets as for fission reactors, due to the lower power density and lower afterheat, they could still present some problems as there are parts of the blankets, for instance the first wall, where the heat fluxes and power densities are comparable with those in fission reactors.

3.6 *Reliability*

Studies, based on the quality standard presently achieved by the nuclear fission industry in the manufacturing of tubes, welds and so on, indicate clearly that the availability of the blanket system is a crucial problem. This is especially true for concepts using a large number of small diameter tubes inside the blanket for heat extraction, resulting in a relatively high frequency of coolant leaks.

The two ceramic blankets have about 50000 to 60000 tubes. These numbers are comparable with the number of fuel rods in a fission reactor of equivalent power (the German standard PWR of 1300 MWe has about 59000 fuel rods, while the fast reactor Superphenix with 1180 MWe has about 100000 rods in the core and 20000 in the blanket). Both for the standard German PWR of 1300 MWe and the Superphenix reactor a failure rate of 10^{-5} rods/year ($\approx$ 10^{-9} rods/hour) has been achieved [26, 27]. However, the comparison between the fuel rods of fission reactors and the blanket coolant tubes may be not a straightforward one: the first are straight and free to move at one end to accommodate for thermal expansion, while the latter are bended and welded at both ends. On the other hand the blanket tubes are not in contact with a fuel which may reach very high temperatures and contains a considerable amount of fission products, some of which very chemically aggressive.

The time required to replace a blanket segment is probably considerably larger than that required by the replacement of a fuel element of a fission reactor, so that there is great incentive to provide full redundancy in the heat extraction system.

The ceramic BIT concept could claim some reliability advantage if normal operation of the blanket could be continued despite the failure of a breeder tube, which is not yet clear. The operating pressure of the sweep gas flowing inside the breeder tubes was chosen close to the one of the coolant in this view. This offers also the advantage that the primary stresses on the tube walls are very low, thus reducing the failure rate.

Also in the case of the BOT concept operation with a failed coolant tube is possible, as the segment box can sustain the full pressure of the coolant gas. In this case however, the sweep gas flow of the blanket segment in question should be routed to a small tritium recovery system working at high pressure, kept usually in service.

4. *The European R & D Program*

4.1 *Ceramic breeder materials*

As mentioned in Section 3.2 the main issue is the behavior of the ceramic materials under irradiation, namely tritium release, mechanical integrity and compatibility with the martensitic structural material and with beryllium. These problems are being investigated within the European Program both with out-of-pile and in-pile experiments.

The in-pile experiments have been performed jointly among the European partners at thermal reactors in France (SILOE and OSIRIS) and in Holland (HFR). Some experiments were made in American fast reactors (EBR2 and FFTF) and in the FRJ-2 thermal reactor in Germany. The materials investigated were $LiAlO_2$, Li_4SiO_4 and Li_2ZrO_3, the first in form of pellets,and the other two either in form of pellets or pebbles. The peak lithium burn-ups were about 3 - 4 % and the dpa's about 2 - 3 in the thermal reactors, while in the fast reactors about 6 % burn-up and about 20 dpa were achieved.The irradiation temperatures were in the range 300 - 750 °C [12, 15, 28 - 30].

The irradiations performed so far, although generally successful, do not cover the complete range of operational conditions expected in the European DEMO blankets: temperatures between 300 °C and 900 °C, lithium burn-ups up to 10 - 25 %, dpa rates up to 20 - 40. The temperatures and lithium burn-ups can be achieved in the available thermal reactors, however the required dpa rates can be obtained in a reasonable time only in fast reactors. Furthermore, fast reactors allow to perform experiments with more uniform power distribution. This is important especially with large irradiation samples. It is planned to perform irradiations in the fast reactor Phenix and/or in the EBRII. If both do not become available a Russian fast reactor could be a possibility.

4.1.1 *Tritium release*

Tritium release from ceramics is a very complex phenomenon, depending mainly on the diffusion of tritium within the grain and on various desorption mechanisms occurring at the grain and at the particle surface. Models to describe these phenomena have been so far not very successful and may be not very useful to assess the most important parameter, namely the tritium inventory in the blanket breeder ceramic. Therefore many in-pile experiments have been performed to have data directly applicable to the blanket situation. These data are generally represented is terms of the tritium residence time τ, namely the ratio of tritium inventory to tritium production, as a function of temperature. Because the relevant tritium release phenomena are all of first order, the τ-temperature relationship can be applied directly to the blanket case even for different power densities, provided the experiments have been performed with exactly the same breeder material and the same chemical composition of the purge gas to be used in the blanket.

To compare tritium release of various ceramic breeder materials the temperature at which τ is equal to 1 day is often used. With the standard purge gas atmosphere of helium + 0.1 % hydrogen this temperature has been found by the in-pile experiments [15, 28, 31] to be about 450 °C, 350 °C and 320 °C for $LiAlO_2$, Li_4SiO_4, and Li_2ZrO_3 respectively. For $LiAlO_2$ also the purge gas helium + 0.01 % H_2O has been used with comparable tritium release behavior. The addition of H_2O could decrease the permeation of tritium out of the purge flow system [28]. All this holds for the material fabricated with the methods and in the shapes to be used in the two European Solid Breeder DEMO Blankets and, of course, for the conditions in which the irradiations were performed.

So far, the results indicate that the tritium release characteristics are not affected by the irradiations. The irradiation EXOTIC-7 in the HFR reactor will allow to obtain information on the effects of high lithium burn-ups (up to 10 %) on the tritium release behavior. The results of this irradiation will be available in the fall of 1995. Higher temperatures and dpa rates may be not so important, as it is expected that the tritium release would be faster (higher temperatures) or not much affected by higher dpa rates.

4.1.2 *Mechanical integrity*

The mechanical integrity of the pellets or pebbles under irradiation is important, as excessive fragmentation could decrease or even locally stop the flow of the purge gas. Other consequences could be displacements of the breeder material and local hot spots due to an excessive local power production and bad thermal conductivity. By the irradiations performed so far in thermal and fast reactors the breeder ceramics remained essentially intact. The only exception is the DELICE3 experiment in OSIRIS/Saclay where all the samples irradiated at 400 °C, with the exception of the $LiAlO_2$ pellets fabricated by ENEA, were considerably damaged, even if the power density and lithium burn-up were smaller than in other irradiations. A tentative explanation of this behavior could be the steep temperature transients caused by trips in the OSIRIS reactor [30].

The sensitivity of ceramic materials to rapid rates of change in temperature is well known. Thus, a series of thermal cycle tests between 230 ° and 600 °C have been performed for beds of Li_4SiO_4 and mixed beds of beryllium and Li_4SiO_4 pebbles. The tests have shown that the Li_4SiO_4 pebbles break in the temperature range

430 ° to 500 °C for negative temperature rates of change of about 50 to 54 °C/sec [31]. These values are considerably higher than the values expected in the blanket. Similar thermal cycle experiments are planned for irradiated material.

Also here, one sees the necessity of obtaining material irradiated at the peak DEMO Blanket values of lithium burn-up and dpa's.

4.1.3 *Compatibility with beryllium and structural material*

Various experiments performed within the European Community Fusion Technology Program have shown that the out-of-pile compatibility limit of the breeder ceramics $LiAlO_2$, Li_4SiO_4 and Li_2ZrO_3 with the austenitic steel 316 L, with MANET and with beryllium is 700 °C [33 - 36]. The irradiation experiment SIBELIUS has confirmed the out-of-pile results for the beryllium compatibility with the three breeder ceramics [37, 38]. The SIBELIUS irradiation was performed in a thermal reactor at 550 °C for 2000 hours with a He + 0.1 % H_2 atmosphere. Data for DEMO blanket peak conditions are required.

4.2 *Beryllium*

As mentioned in Section 3.3 the main issue is the behavior of beryllium under irradiation, namely swelling, embrittlement and tritium release. The European R & D work has concentrated on these three problems.

4.2.1 *Beryllium swelling*

As the results of in-pile irradiations at high temperatures are rather scarce, irradiations were performed in the thermal reactor SILOE /Grenoble, in the temperature range 240 - 700 °C and for neutron fluences of 2 - 2.5 x 10^{21} cm^{-2} ($E_n \geq 1$ MeV) corresponding to helium contents of 1100 - 1400 appm (BEGONIA experiment) [39]. These fluences are of course considerably smaller than the peak values expected in the DEMO (15000 - 20000 appm helium), therefore irradiations are planned in the fast reactor Phenix, which will be performed as soon as the reactor will become available.

As the swelling data for the DEMO peak neutron fluences are not yet available, modeling work on helium behavior in irradiated beryllium has been initiated in Europe. A rather sophisticated computer code (ANFIBE) has been developed. This code describes the precipitation of helium atoms into intragranular bubbles, the migration of these bubbles to the grain boundaries to form intergranular bubbles, the growth and coalescence of these bubbles and their interlinkage with pores. The mathematical model is expressed by a system of seven reaction rate differential equations. The description of the helium behavior in beryllium accounting for the beryllium properties (mainly surface tension, temperature and irradiation induced creep) allows to calculate the volume swelling. The comparison of the ANFIBE calculations with the experiments available from the literature, including the BEGONIA experiments, both at low and at high temperatures shows a good agreement [40]. The experimental data covers independently the ranges of temperatures, fast neutron fluences and helium contents for the DEMO blankets. However, more data are required to qualify ANFIBE especially from experiments in which irradiations to high fluences have been carried out at high temperatures.

4.2. *Beryllium embrittlement*

The data on beryllium embrittlement under irradiation is either for very old beryllium or contradictory [41].

The irradiation-induced embrittlement is very difficult to model and irradiation experiments are essential. However, contrary to swelling, beryllium embrittlement is produced at relatively low fluences which can be easily achieved in thermal test reactors. An European collaborative program to investigate the irradiation embrittlement of beryllium has been initiated; it involves the exposure of tensile, disc compact tension and transmission electron microscope specimens to a fluence of 1.5×10^{21} cm^{-2} ($E_n > 1$ MeV) at temperatures of 200°, 400° and 600 °C in the BR2 reactor at Mol. The following grades of beryllium supplied by Brush Wellman are included: S-65 and S-200F containing 0.5 - 0.6 % and 0.9 - 1 % BeO respectively and produced using impact ground powders and consolidated by (a) axial vacuum hot pressing (VHP) and (b) direct hot isostatic pressing (HIP). Pre- and post-irradiation as well as thermal control tensile and fracture toughness tests will be carried out at the respective temperatures of irradiation and also at a "stand-by" (coolant) temperature of 200° or 250 °C. Immersion density measurements have been performed on the tensile samples before testing to evaluate the swelling; the helium and tritium contents of the irradiated specimens will also be determined. The irradiation has been started in August 93 and has been terminated in March 1994.

4.2.3 *Tritium release from beryllium*

Some work on out-of-pile tritium release from beryllium irradiated in the SIBELIUS experiment has been started in Europe [38]. Further out-of-pile tritium release is being performed at KfK Karlsruhe with beryllium irradiated at low temperatures and high fluences (1 - 4 x 10^{22} cm^{-2}, $E_n \geq 1$ MeV) in the BR2 reactor. Similar experiments will be performed with beryllium from the BR2 embrittlement experiment and from a recently started irradiation experiment in HFR. This latter experiment, performed with beds of beryllium pebbles and mixed beds of beryllium and Li_4SiO_4 pebbles will allow to separately measure the release of the tritium directly produced in the beryllium and the tritium injected in the beryllium by the recoil from the adjacent breeder ceramic pebbles. Supplementary information may be obtained on the embrittlement of the beryllium pebbles and on beryllium swelling.

The results of these experiments cannot be directly applied to calculate the tritium inventory in the DEMO blanket. Therefore it was decided to extend the code ANFIBE for the calculation of the tritium release from beryllium. The diffusivity of tritium in beryllium is much higher than that of helium, but the tritium release is hindered by the helium bubbles (physical trapping) or by the beryllium oxide impurity (chemical trapping). The model describing the behavior of tritium is formally identical to that of helium, however, a rate-equation has been added, which accounts for the chemical trapping of the tritium by the oxygen impurities.

The comparison of the ANFIBE calculations with the experiments of Ref [38] and other experiments from the literature shows a good agreement at higher temperatures. However, at temperatures lower than 500 °C the code underpredicts the tritium release by a factor 2 - 3 [42]. This may be attributed to the formation of microcracks in the beryllium due to the rapid temperature change. Quite clearly this part of the code needs further development. Furthermore, more extensive experimental data, particularly with respect to in-pile tritium release, is required to qualify the code.

4.3 *Tritium control and extraction*

4.3.1 *Tritium control*

As discussed in section 3.4 R & D work is required in the following areas:

a) Development of an aluminizing system for the tubes of the helium/water-steam heat exchangers.

b) Development of an aluminizing system for the tubes separating the breeder and beryllium from the main coolant system. In this case in-pile tests are required to investigate a possible degradation of the Al_2O_3 layer under irradiation, especially in presence of thermal stresses, and the compatibility with lithiated ceramics.

c) Development of a permeation barrier for the inner walls of the MANET first wall coolant channels.

d) Investigation of the necessity and/or possibility of maintaining an oxidizing atmosphere in the main helium coolant system to allow the selfhealing of the Al_2O_3 layer on the surface of the permeation barrier.

e) Development of catalyzers to promote the oxidation of HT to HTO in the main helium coolant system.

This work should addressed to in the near future.

4.3.2 *Tritium extraction*

Tritium extraction from helium is less problematic than the extraction from water or liquid metals. however some work has been initiated to identify the method most suited to the particular conditions of tritium contained in the purge helium of a solid breeder. In particular a feasibility study has been performed in cooperation with the firm LINDE for the BOT DEMO blanket [43].

4.4 *Non nuclear tests*

Two thermal cycle tests have been performed in the HE-FUS2 rig at ENEA-Brasimone for the thermomechanical characterization of breeder rods containing $LiAlO_2$ pellets typical of the BIT blanket option. In the first of the two tests, 5000 cycles were performed at a peak tensile stress in the ceramic pellets of 17 MPa. In the second test the number of cycles was 20000 and the peak stress of 22 MPa. The tests were quite successful, only at the end of the second test one pellet was found cracked in two half pieces without any smaller fragments or powder [28]. Subsequently, thermal cycle tests have been performed with single rods containing Li_2ZrO_3 pellets. The results of these tests indicated that the behaviour of the Li_2ZrO_3 pellets is similar to that of the $LiAlO_2$ ones of comparable density [44].

Thermal hydraulic tests for the BIT blanket option were performed with a curved 7 rod bundle to investigate the curvature effects on the thermal mixing within the bundle and on subchannel flow distribution and to measure the wire-wrap/guide spacers pressure drops. The tests were performed at the AIR-FUS1 facility at ENEA-Brasimone [45]. A high pressure helium loop (HE-FUS3) will be shortly constructed at the same center with the capability of testing mock-ups of both the European Solid Breeder blanket options.

A test with 700 thermal cycles between 440 °C and 280 °C has been performed in the high pressure helium loop HEBLO at KfK-Karlsurhe on twelve beryllium plates of 200 mm length brazed pairwise to each other and with a leg of the cooling coil containing high pressure helium. The test was successful: the brazing has remained intact [46]. This test relates to the first BOT blanket option with beryllium plates rather than pebbles. A new test section with beryllium and Li_4SiO_4 pebbles has been manufactured and will be tested in the fall of 1994.

4.5 *Mechanical effects caused by plasma disruptions*

The three dimensional computational system DEMETRA for the calculation of currents, magnetic forces and stresses has been developed at KfK-Karlsruhe. With this system it has been possible to show that the BOT blanket can withstand the reference major disruption of Table 1 (plasma rate of change = 1 MA/ms), even accounting for the ferromagnetic properties of MANET [47, 48]. However, these calculations, performed for blanket segments electrically insulated from each other, indicated that the blanket could not withstand plasma disruptions much faster than 1 MA/ms.

At a recent meeting it has been suggested to connect electrically in toroidal direction the first wall of the various blanket segments [49]. This improves the situation quite considerably, as the currents induced in the blanket are mainly running in toroidal direction, i.e. parallel to the major component of the magnetic field. The forces on the blanket are therefore caused only by the coupling of the induced currents with the much smaller poloidal magnetic field. In this case, the DEMETRA calculations show that the blanket can withstand centered disruptions as fast as 10 MA/ms as well as vertical displacements of the plasma. In these recent calculations first attempts have been made to account also for the irradiation induced embrittlement of MANET [50, 51]. This work is important because the dynamic forces caused by disruptions are particularly dangerous for brittle materials.

This work will continue on two lines:

a) Validation of the DEMETRA code by comparison of the computed results with induced currents in tokamaks such as ASDEX upgrade and JET.

b) More accurate evaluation of the effects of thermal cycles (fatigue) and of the irradiation induced embrittlement of MANET.

4.6 *Safety and reliability*

Calculations have been performed by KfK-Karlsruhe and SIEMENS-KWU of the two most dangerous accidents, namely the loss of flow accident (LOFA) and the loss of coolant accident (LOCA) for the BOT blanket option [52]. The results of the calculations can be summarized as follows:

For both accidents it was assumed that the neutron power is switched off 1 second after accident begin and that the plasma power radiated to the first wall decreases linearly to zero in 10 seconds.

a) LOFA: the assumption is that there is a sudden and complete loss of power to the helium blowers of both independent cooling systems. Due to the run-time of the blowers the first wall and blanket temperatures decrease during the transient. Also for long times after the accident no problem arises due to the helium natural convection cooling, provided that the center of the heat exchangers is at least 5 meters higher than the center of the blanket segments.

b) LOCA: as for the Light Water Fusion Reactors, a sudden guillotine cut of the largest tube downstream the helium blower in one of the two independent coolant systems is assumed. The helium pressure is reduced to 0.1 MPA in 0.5 seconds. Also here, due to the help of the other independent helium cooling system, the blanket temperatures decrease, while the maximum first wall temperatures increase from 550 to 600 °C for a period of about 10 seconds and than decrease. This probably does not cause an appreciable damage to the blanket. Similar results have been obtained for the LOCA at CEA for the BIT design.

The calculation of the reliability of the main helium coolant system outside the BOT blanket (heat exchangers, blowers, main coolant tubes) has shown that the availability of this system is higher than 98 % [53]. Similar work for the blanket itself is in progress.

5. *Conclusions*

The two European solid breeder blankets differ in their configuration, however their similarities have as a consequence that the critical issues are in principle the same and that the major part of the R & D work can be performed jointly.

The main critical issues for both solid breeder blankets are:

a) Behavior of the breeder ceramics under irradiation. The irradiations performed so far, mainly in thermal reactors, were encouraging. The tritium release, mechanical stability and compatibility results were satisfactory. However, these data have been obtained for lithium burn-ups of up to 3 - 4 %, dpa rates of up to 2 - 3, i.e. for conditions considerably lower than the peak values of the DEMO blanket. By the recently started EXOTIC-7 experiment lithium burn-ups of up to 10 % (dpa $\approx$ 6) are foreseen, however, the peak dpa rates of the DEMO blankets are achievable only in fast reactors.

b) Behavior of beryllium under irradiations. The data on swelling, embrittlement and tritium release from modern beryllium are so far not sufficient to cover the peak conditions in the DEMO blankets. The results of the BR2 irradiation experiment will probably provide sufficient data on embrittlement. The computer code ANFIBE may be able to extrapolate the irradiation data on swelling and, to a lesser extent, on tritium release to DEMO blanket conditions. However, more data is required from high temperature irradiations to high fluences in fast reactors.

c) Tritium control. The condition that only 10 curie/d is released from the DEMO blanket to the steam-water circuit requires permeation barriers (on the cladding of the breeder material and/or on the helium/water heat exchangers) and/or an oxidation system of the tritium in the main helium system. The resulting effect of these measures must able to reduce the tritium losses of at least two orders of magnitude.

Besides the martensitic steel, which is part of a so-far separate European R & D program other issues require R & D work:

d) Mechanical effects from plasma disruptions. The computational system DEMETRA to calculate the forces and stresses caused by plasma disruptions in the blanket has been developed. The code accounts for the ferromagnetic properties of MANET and it is being modified to account for the irradiation induced embrittlement of this material. The calculations performed so far show that the BOT blanket can withstand a centered disruption with a plasma decrease of 10 MA/ms. DEMETRA requires further validation by comparison with experimental data.

e) Safety and reliability. First calculations show that the BOT and BIT blanket can withstand severe loss of flow and loss of coolant accidents. The consequences of other incidents, loss of coolant to the plasma region and others, have to be evaluated as well.

First reliability analyses of the BOT blanket coolant system outside the blanket proper show an acceptable availability ($\geq$ 98 %). These analyses have to be extended to the blanket proper.

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Table 1 Specifications of DEMO

1. <u>Demo Specifications</u>

1.1 **Geometry:**

- Double null		
- Major radius	=	6.3 m
- Minor radius	=	1.82 m
- Elongation	=	2.2
- Number of sectors		16
- Number of segments		48 outboard, 32 inboard
- Max. blanket + shield thickness:		1.856 m outboard 1.176 m inboard

1.2 **Operation:**

- Neutron wall load		2.2 MW/m^2 average
- Surface heat flux		0.4 MW/m^2 average 0.5 MW/m^2 peak
- Continous operation	=	reference mode
- Pulsed operation	=	optional mode with burn time = 1 hour, number of cycles = 2.10^4
- Disruptions:		linear decay of plasma current from 20 MA to zero in 20 ms.

2. <u>Blanket requirements</u>

2.1 **Tritium breeding:**

TBR > 1.0 in 3D neutronic calculations with 10 ports at outboard equatorial plane, 3.4 m high. full segment width

2.2 **Operation time of segment**

20000 hours

2.3 **Coolant conditions:**

As required for electricity production with a thermal efficiency $\eta \geq 30$ % where

$$\eta = \frac{\text{electrical power output from the blanket}}{\text{(neutron power + surface heat) to the blanket}}$$

2.4 **Resistance to disruptions**

After one disruption the blanket may be deformed and non operational but it must be removable from the port.

Table 2 *Main Data of the European Solid Breeder Blankets*

	B.I.T.	B.O.T.
Breeding and multiplier material	Annular $LiAlO_2$ pellets Beryllium slabs	Mixed bed of Li_4SiO_4 and Beryllium pebbles
Li^6 enrichment (at %)	90	25
Total blanket power	2500 MW (+ 300 MW in divertors)	
Cooland helium pressure:	6 MPa 6 MPa	8.8 MPa 10 MPa
Coolant helium pressure drop (F.W., blanket, feeding tubes) outboard inboard	 0.34 MPa (F.W. not included) 0.34 MPa (F.W. not included)	 0.31 MPa 0.37 MPa
Coolant helium temp:	250 °C 520 °C	250 °C 450 °C
Max. steel temperature	550 °C	540 °C
Max. temp. in beryllium	600 °C (BOL 500 °C (EOL)	660 °C (BOL) 500 - 550 °C (EOL)
Max. temp. in breeder material Min. temp. in breeder material	590 °C 410 °C	660 °C 300 °C
Real tridimensional tritium breeding ratio (without ports) Accounting for 10 outboard ports	1.12 1.07	1.14 1.08
Peak lithium burn up	25 at %	10 at %
Peak fluence in beryllium	21000 appm He	17000 appm He
Tritium purge system pressure Tritium inventory in breeder mat.	5.7 MPa 200 g	0.1 MPa 60 g

List of figure captions

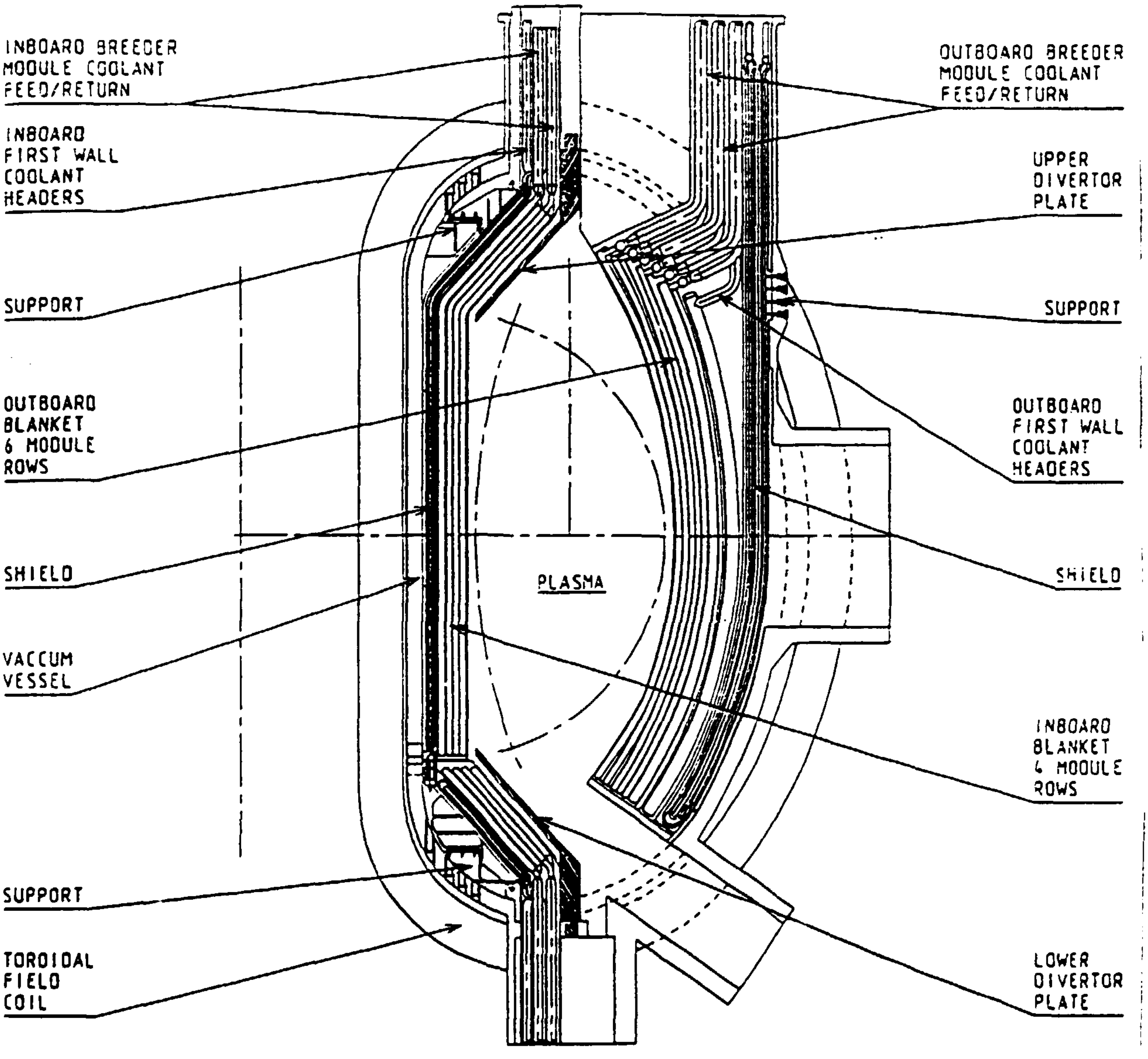
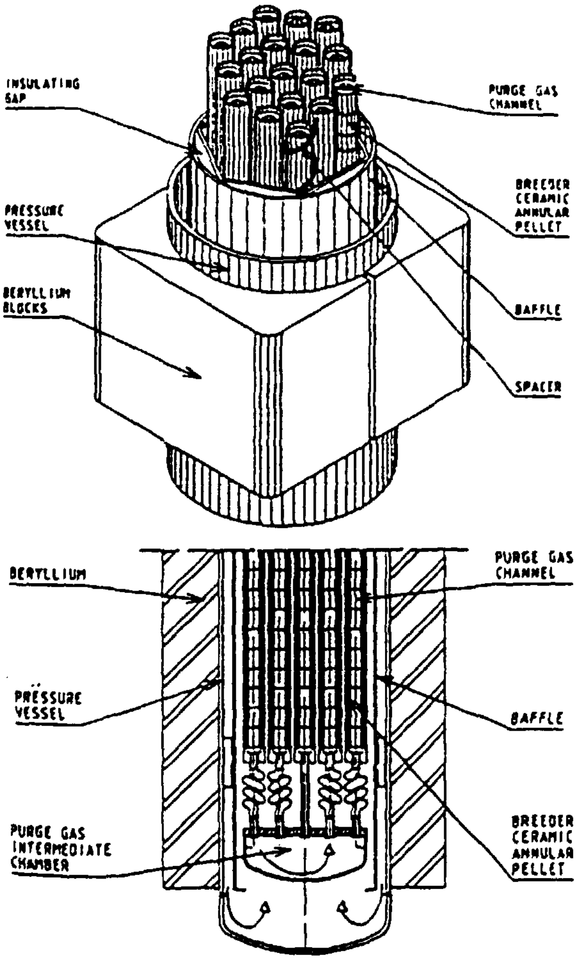

INBOARD BREEDER MODULE COOLANT FEED/RETURN
INBOARD FIRST WALL COOLANT HEADERS
SUPPORT
OUTBOARD BLANKET 6 MODULE ROWS
SHIELD
VACCUM VESSEL
SUPPORT
TOROIDAL FIELD COIL
OUTBOARD BREEDER MODULE COOLANT FEED/RETURN
UPPER DIVERTOR PLATE
SUPPORT
OUTBOARD FIRST WALL COOLANT HEADERS
SHIELD
INBOARD BLANKET 4 MODULE ROWS
LOWER DIVERTOR PLATE
PLASMA

Fig 1


INSULATING GAP
PURGE GAS CHANNEL
BREEDER CERAMIC ANNULAR PELLET
PRESSURE VESSEL
BERYLLIUM BLOCKS
BAFFLE
SPACER
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PURGE GAS CHANNEL
PRESSURE VESSEL
BAFFLE
PURGE GAS INTERMEDIATE CHAMBER
BREEDER CERAMIC ANNULAR PELLET

Fig. 2

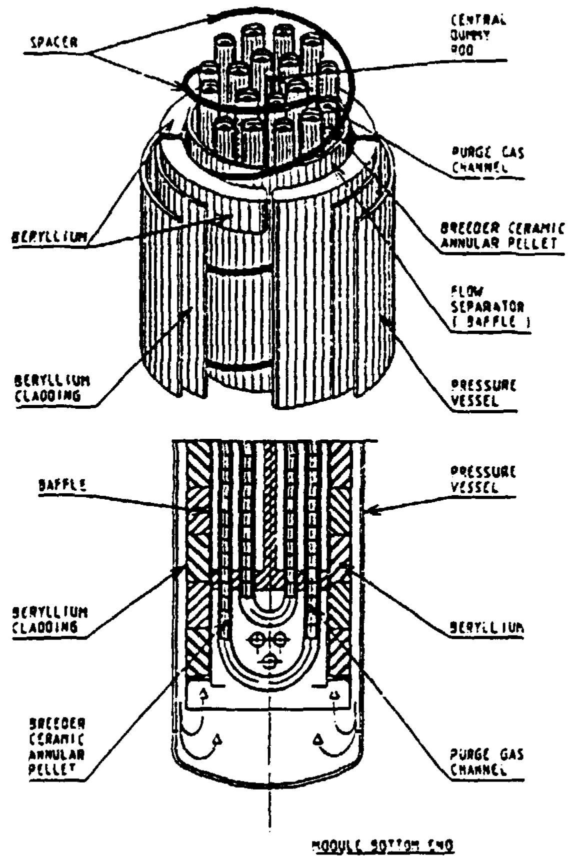
SPACER
CENTRAL DUMMY ROD
PURGE GAS CHANNEL
BERYLLIUM
BREEDER CERAMIC ANNULAR PELLET
FLOW SEPARATOR (BAFFLE)
BERYLLIUM CLADDING
PRESSURE VESSEL
BAFFLE
PRESSURE VESSEL
BERYLLIUM CLADDING
BERYLLIUM
BREEDER CERAMIC ANNULAR PELLET
PURGE GAS CHANNEL
MODULE BOTTOM END

Fig. 3

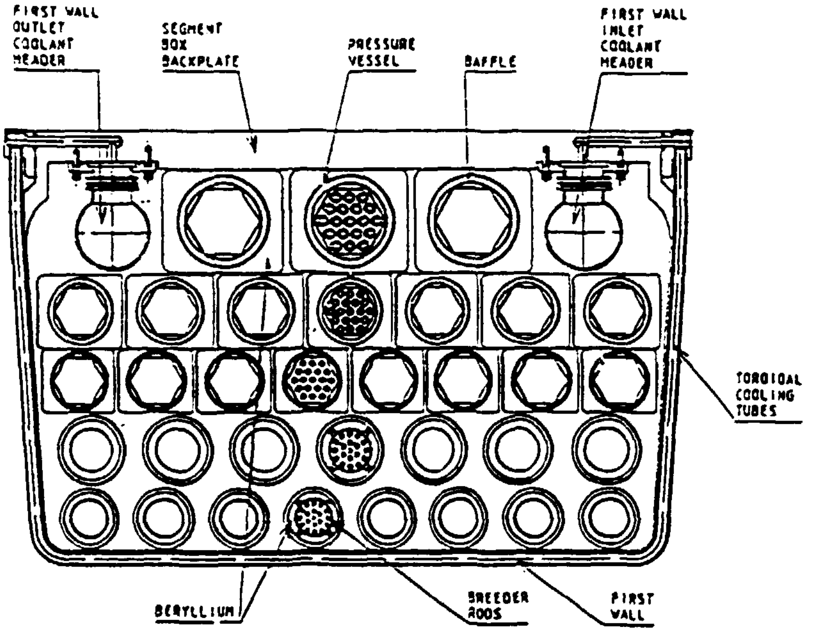
FIRST WALL OUTLET COOLANT HEADER
SEGMENT BOX BACKPLATE
PRESSURE VESSEL
BAFFLE
FIRST WALL INLET COOLANT HEADER
TOROIDAL COOLING TUBES
BERYLLIUM
BREEDER RODS
FIRST WALL

Fig. 4

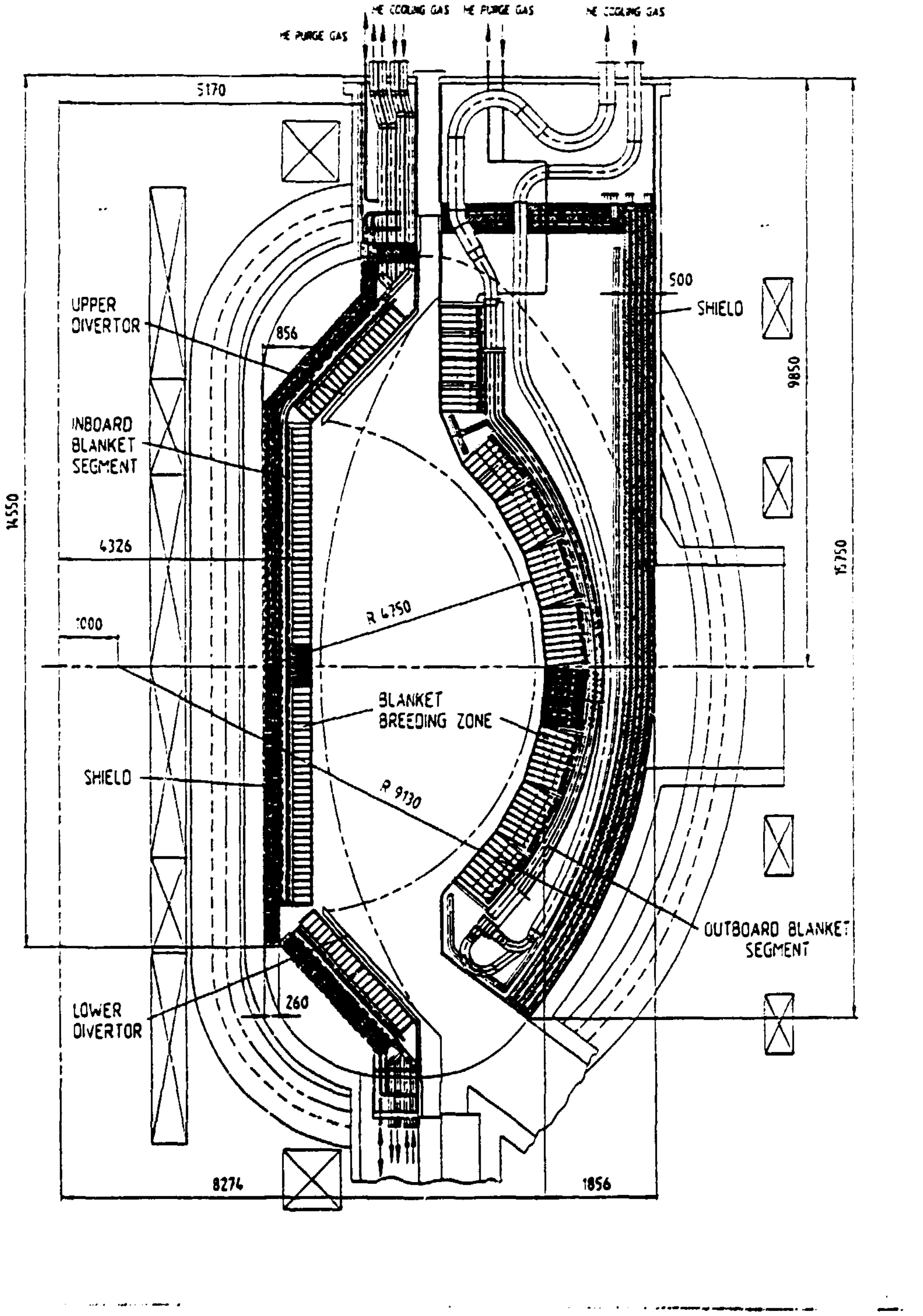
HE PURGE GAS
HE COOLING GAS
HE PURGE GAS
HE COOLING GAS
5170
UPPER
DIVERTOR
856
INBOARD
BLANKET
SEGMENT
14550
4326
1000
SHIELD
LOWER
DIVERTOR
260
8274
1856
500
SHIELD
9850
15750
R 4750
BLANKET
BREEDING ZONE
R 9130
OUTBOARD BLANKET
SEGMENT

Fig. 5

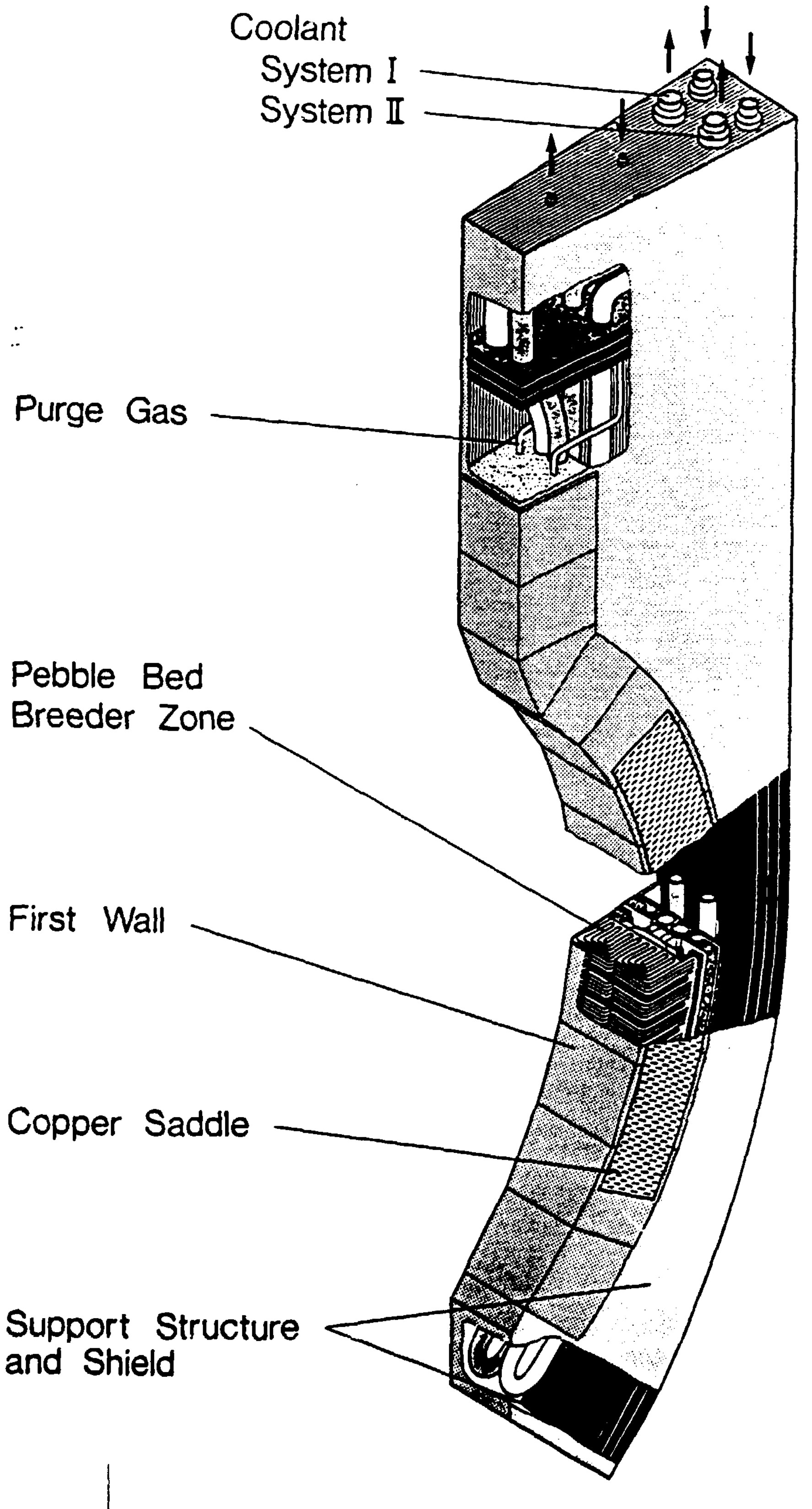
Coolant
System I
System II
Purge Gas
Pebble Bed
Breeder Zone
First Wall
Copper Saddle
Support Structure
and Shield

Fig. 6

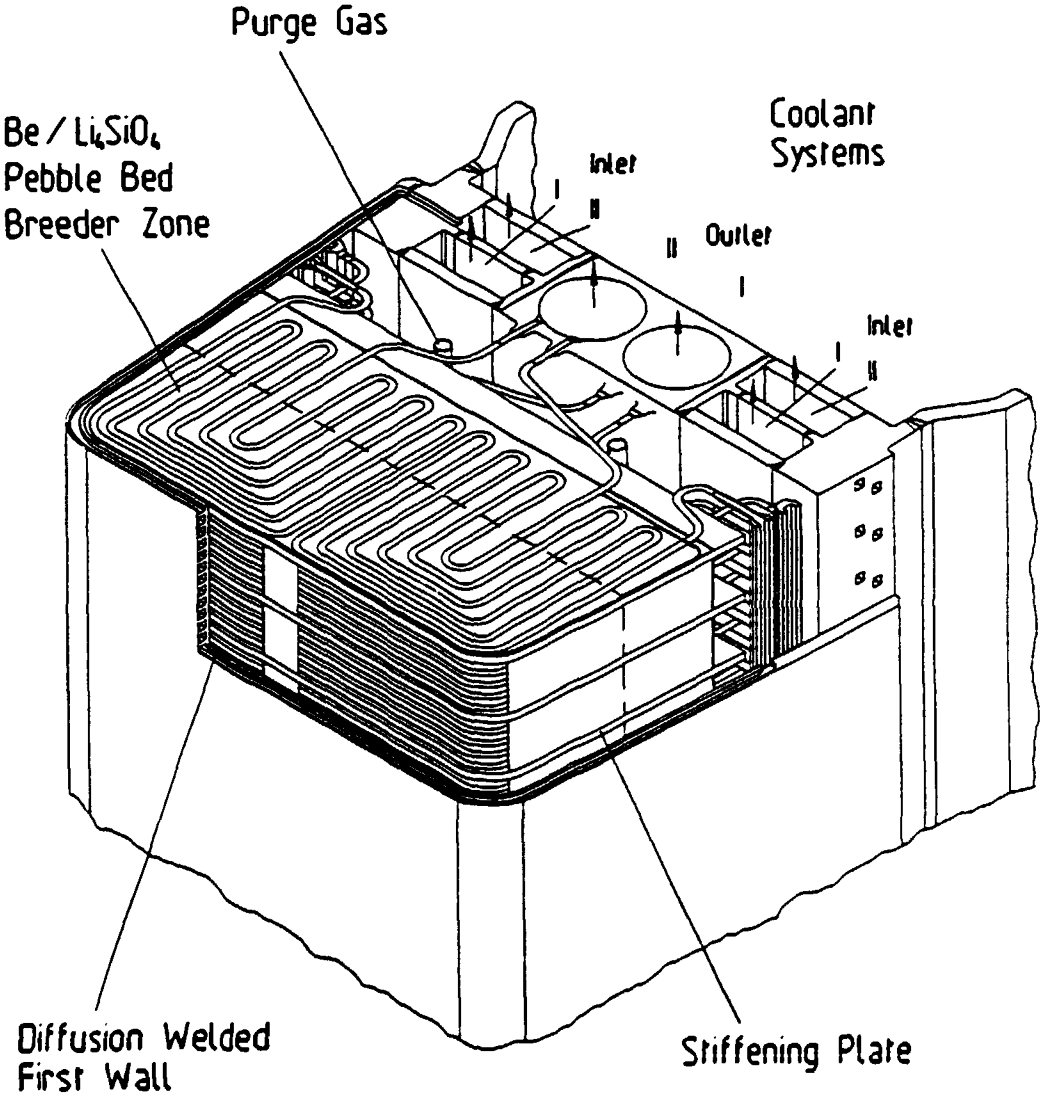
Purge Gas
Be / Li_4SiO_4 Pebble Bed Breeder Zone
Coolant Systems
Inlet
I
II
II
Outlet
I
Inlet
I
II
Diffusion Welded First Wall
Stiffening Plate

Fig. 7

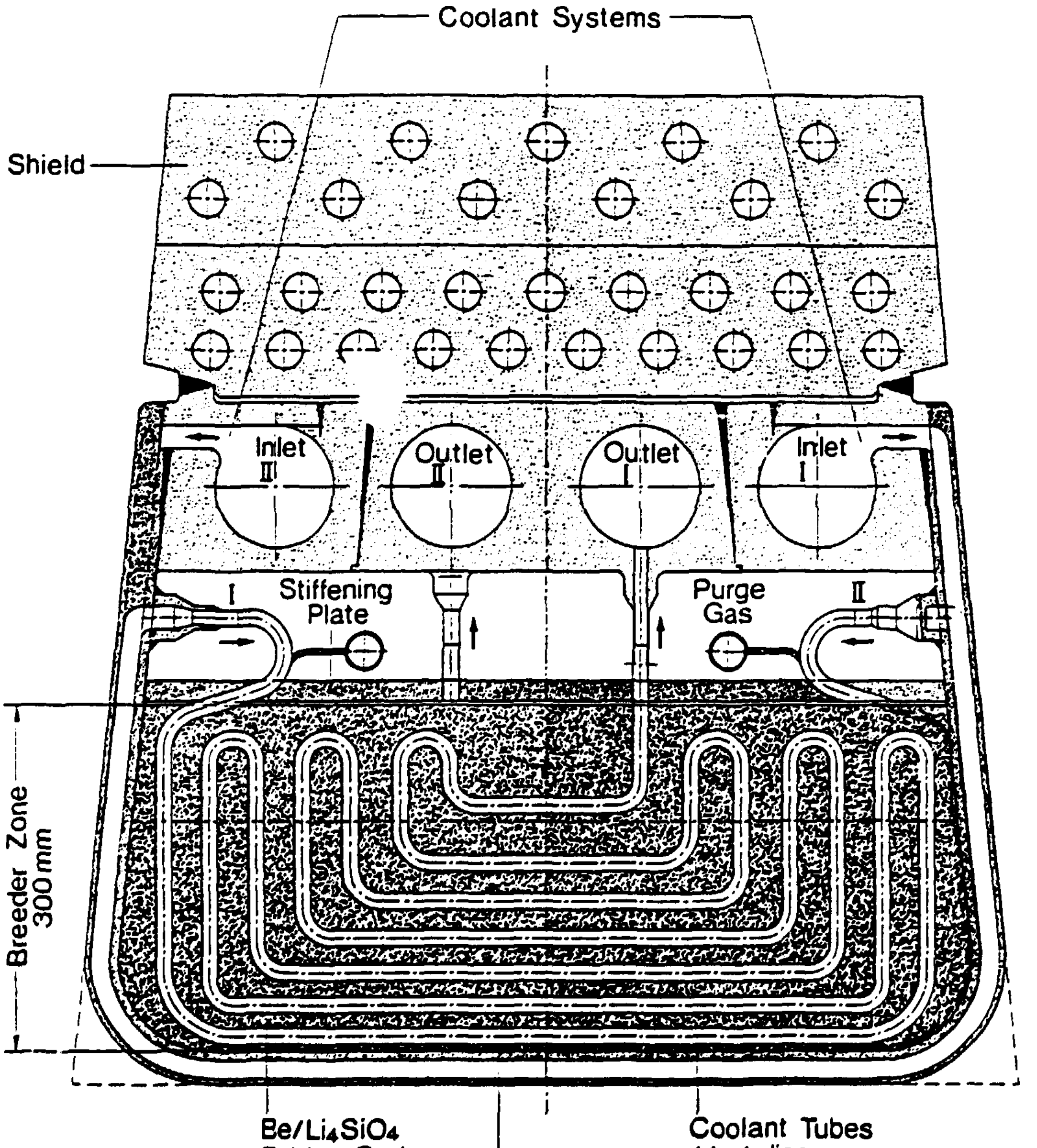
Coolant Systems
Shield
Inlet
II
Outlet
II
Outlet
I
Inlet
I
I
Stiffening
Plate
Purge
Gas
II
Breeder Zone
300 mm
Be/Li4SiO4
Pebble Bed
Coolant Tubes
14 x 1 diam
Diffusion Welded First Wall

Fig. 8