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# Status Report
# KfK Contribution to the Development of

# DEMO-relevant Test Blankets for NET/ITER

## Part 2: BOT Helium Cooled Solid Breeder Blanket

## Volume 1: Summary

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Association KfK-Euratom
Projekt Kernfusion

Kernforschungszentrum Karlsruhe

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## Status Report
### KfK Contribution to the Development of Demo-relevant Test Blankets for NET / ITER
### Part 2: BOT Helium Cooled Solid Breeder Blanket
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**Abstract**

The BOT (Breeder Outside Tube) Helium Cooled Solid Breeder Blanket for a fusion Demo reactor and the status of the R & D program is presented. This is the KfK contribution to the European Program for the Demo relevant test blankets to be irradiated in NET / ITER. Volume 1 (KfK 4928) contains the summary, volume 2 (KfK 4929) a more detailed version of the report.

In both volumes are described the reasons for the selected design, the reference blanket design for the Demo reactor, the design of the test blanket including the ancillary systems together with the present status of the relative R & D program in the fields of neutronic and thermohydraulic calculations, of the electromagnetic forces caused by disruptions, of the development and irradiation of the ceramic breeder material, of the tritium release and recovery, and of the technological investigations. An outlook is given on the required R & D program for the BOT Helium Cooled Solid Breeder Blanket prior to tests in NET / ITER and the proposed test program in NET / ITER.

*"This work has been performed in the framework of the Nuclear Fusion Project of the Kernforschungszentrum Karlsruhe and is supported by the European Communities within the European Fusion Technology Program."*

## Status Bericht
### KfK Beitrag zur Entwicklung der Demo-relevanten Test Blankets für NET / ITER
### Teil 2: BOT Heliumgekühltes Feststoff-Brutblanket
### Band 1: Zusammenfassung

**Zusammenfassung**

Es wird ein heliumgekühltes Feststoffbrutblanket für einen Demo-Fusionsreaktor mit Brutstoff außerhalb von Kühlrohren (BOT) beschrieben und der Stand der F und E Arbeiten wird vorgestellt. Dies ist der Beitrag des KfK zum Europäischen Entwicklungsprogramm für die Demo-relevanten Testblankets, die in NET / ITER bestrahlt werden sollen. Band 1 (KfK 4928) enthält die Zusammenfassung und Band 2 (KfK 4929) den detaillierten Bericht.

In den beiden Berichten werden die Gründe, die zum gewählten Entwurf geführt haben, beschrieben. Es werden der Referenzentwurf für das Demo-Reaktorblanket und der Entwurf für ein Testblanket in NET / ITER mit den dazugehörigen externen Kreisläufen vorgestellt. Der Stand der Forschungs- und Entwicklungsarbeiten bezüglich Neutronen- und Thermohydraulikrechnungen, Rechnungen der elektromagnetischen Kräfte verursacht durch Plasmazusammenbrüche, Entwicklung und Bestrahlung der keramischen Brutstoffe, Tritiumfreisetzung und -gewinnung sowie technologische Untersuchungen werden aufgezeigt.

Es wird ein Ausblick gegeben auf die noch vor dem NET / ITER Test notwendigen F & E-Arbeiten für das BOT Heliumgekühlte Feststoff-Brutblanket und das zugehörige Testprogramm, das in NET/ ITER durchgeführt werden soll.

*"Die vorliegende Arbeit wurde im Rahmen des Projekts Kernfusion des Kernforschungszentrums Karlsruhe durchgeführt und ist ein von den Europäischen Gemeinschaften geförderter Beitrag im Rahmen des Fusionstechnologieprogramms."*

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## 1. PREVIOUS STUDIES AND REASONS FOR THE SELECTED DESIGN

The first helium cooled solid breeder blanket investigated at the Karlsruhe Nuclear Research Center (KfK) was with a lead multiplier integral with the first wall and toroidally arranged pressure tubes containing the breeder material. Due to the low melting point of lead the first wall region had to be cooled by a separate helium circuit at low temperature [1].

Subsequently, a design with poloidally running pressure tubes containing the ceramic breeder material and the beryllium multiplier in the form of a pebble bed was investigated [2]. The resulting one-dimensional tritium breeding ratio (TBR) was 1.08 (based on 100 % torus surface coverage) which would imply a real tridimensional breeding ratio lower than one. Therefore, it was decided to investigate other blanket forms with higher breeder and multiplier amounts per unit volume. Indeed, the investigated poloidal solution was rather leaky for the neutrons, which was the main reason for the rather modest breeding ratio.

In 1988 the conceptual design of a blanket with a tridimensional TBR higher than one was published [3]. This design was based on the use of radial canisters containing the breeder material and the multiplier. However, this blanket was designed for NET, i.e. for a considerably lower neutron flux at the first wall (1 $MW/m^2$ against 2.2 $MW/m^2$ for the Demo blanket) and a lower neutron fluence (0.8 $MWy/m^2$ against 5 $MWy/m^2$ for the Demo blanket)

The here proposed design for the Demo blanket is similar to the canister design for NET. There are however differences, the main one is the use of a martensitic steel as a structural material (Manet) in place of the austenitic steel 316 L. Manet is able to withstand the high neutron fluence in the Demo blanket (70 d.p.a.) without swelling.

The present design is based on the following choices:

- the use of small lithium orthosilicate pebbles as breeder material
- the use of high pressure helium as coolant
- the use of beryllium as neutron multiplier
- the use of the martensitic steel Manet as structural material.

The combination ceramic breeder material 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 military programmes. Tritium extraction from the ceramics by means of a helium purge gas flow has been extensively and successfully used. A further earlier decision within the EFTP was to concentrate on the ternary ceramics, because $Li_2O$ is more chemically active, with possibly more compatibility problems with structural and breeder materials, and also because $Li_2O$ tends to swell more when subjected to neutron irradiation [4]. Among the ternary lithiated ceramics, aluminates and silicates are the most actively studied so far, thus it was decided that CEA-ENEA would concentrate on aluminate investigations and KfK on the silicates. Recently both groups have started some activity on lithium metazirconate $Li_2ZrO_3$ as well. Namely on metazirconate pellets (CEA) or pebbles (KfK).

Early during the KfK ceramic breeder R. & D. program it was established that tritium release from orthosilicate was much faster than from metasilicate [5], lithium orthosilicate $Li_4SiO_4$ was therefore chosen as reference breeder material. This material however has a relatively low thermal conductivity and high thermal expansion coefficient and might break up due to excessive thermal stresses when irradiated at high power densities. This is confirmed by irradiation experiments at high burn-ups, which show that $Li_4SiO_4$ pellets break up in smaller pieces due to excessive thermal stresses [6]. Obviously the use of sufficiently small $Li_4SiO_4$ pebbles would avoid this problem and this is the solution which was chosen for the present design.

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 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 consequences than water leakages.

The only two elements which in practice could be used as neutron multiplier 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. For this reason 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 behaviour under neutron irradiation.

Due to the high neutron fluence in the structural material expected in the Demo blanket ($\approx$ 70 dpa) 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. 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 Manet is part of a separate program within the EFTP, parallel to the blanket program and will not be discussed in the present report.

In summary the present design is based on the following principles:

a) The use of lithium orthosilicate ($Li_4SiO_4$) as breeder. This offers the advantage to have a fast tritium release and thus a low tritium inventory in the breeder material. Lithium orthosilicate is quite stable at high temperatures: in vacuum or dry pure helium the total lithium pressure above $Li_4SiO_4$ at 1120 °C is only $10^{-2}$ Pa. Of the considered lithiated ceramics, lithium orthosilicate, together with $Li_2O$, is the one which has the lower induced radioactivity after short (compared to aluminate) and very long (compared to aluminate and zirconate) times. This offers safety advantages (low after-heat) and makes the waste disposal easier. Finally, lithium orthosilicate has considerably more lithium per unit volume than lithium aluminate $LiAlO_2$ (a factor 2) and than lithium metazirconate $Li_2ZrO_3$ (a factor 1.5). This offers the advantage of having a proportionally smaller lithium burn-up for the same fluence.

b) The use of small $Li_4SiO_4$ pebbles. This avoids the thermal stress problem in the breeder material. With full or annular pellets, cracks, caused by thermal stresses and high burn-ups, are more likely, as pellets are necessarily of greater size. These cracks, if large enough, may cause uncontrolled and very high temperature increases in the ceramic, especially for the pieces which are displaced away from the tube walls. In case of a pebble bed, these large displacements are not possible and the deterioration of the thermal conductivity of the ceramic, likely at high burn-ups, has little effect on the thermal conductivity of the bed, as this is mainly controlled by the thermal conductivity of the helium purge gas. Furthermore, a bed of pebbles allows for a large volume for the flow of the purge helium gas. This reduces the tritium partial pressure in the purge gas and, consequently, the tritium losses from the purge flow system.

c) The use of a helium purge flow at a pressure below atmospheric: This of course reduces the amount and probability of tritium losses by leakage. Furthermore, as the tritium partial pressure in the purge gas in the blanket depends on the purge helium volume flow and not on its density, the low pressure reduces the mass flow rate of the purge helium and thus the dimensions of the purge gas system.

d) The use of the Breeder Out of Tube (BOT) solution: The coolant helium gas at high pressure is contained inside tubes, the breeder material and the low pressure purge gas is placed outside the pressure tubes. This avoids the instability problems (tube collapsing) which could occur, in the case of higher pressure being outside the tubes (Breeder Inside Tube solution) especially in presence of thermal and / or irradiation creep. Furthermore the use of a BOT solution allows to use flat outer surfaces and thus to pack more breeder and multiplier in the limited space available in the blanket.

e) Good temperature conditioning of the blanket components: This requires that the first wall, where the highest heat flux and power density occur is cooled by the inlet helium. The blanket is cooled by the helium preheated in the first wall region, so that the minimum temperature of the breeder is 380 °C, which allows to have a very low tritium inventory in the breeder material. The beryllium temperature is kept as low as possible to reduce the beryllium swelling under irradiation. Beryllium and breeder are well mixed to increase the tritium breeding ratio.

f) The use of relatively small radial canisters: The coolant moves essentially in radial direction thus producing a more uniform temperature distribution in the blanket due to the large power density gradients in radial direction. The use of radial canisters allows a good filling with breeder and multiplier of the space available in the blanket region. The subdivision of the blanket in small modules reduces the thermal stresses and the stresses caused by the plasma disruptions, makes a precise construction easier and gives the possibility of making significant tests starting from the smallest submodules.

g) The use of a redundant convective cooling system: The use of two completely separate coolant systems with the helium flowing alternatively in two opposite directions renders more uniform the temperature distribution in the first wall and in the blanket. The use of a redundant convective cooling system offers safety advantages and is particularly important with helium, which has not as good natural convection cooling capabilities as other, more dense, fluids.

h) The use of a double containment against tritium losses: This is realized in the blanket region by the presence of the canister and of the segment box walls. Obviously, the control of the tritium losses is an important safety issue.

i) The use of a double containment against high pressure coolant helium losses: This is realized in the blanket region by the presence of the high pressure tube and of the canister walls. Also this point has important safety implications.,

In the following Sections the design of Demo blanket and of the relative test objects to be irradiated in NET/ITER will be presented as well as the status of the R&D programme, the required R&D programme prior to the NET/ITER tests and the test programme in NET/ITER. More detailed information on this matter can be obtained from Volume 2 of this Status Report [7].

## 2. THE BLANKET DESIGN FOR THE DEMO REACTOR

### 2.1 The Blanket Configuration

The outer blanket segment is illustrated in Figs. 1 through 5 and exhibits the following basic design features:

1. The ceramic breeder material ($Li_4SiO_4$ pebbles) and beryllium multiplier are contained in 28 separate, nearly rectangular canisters, which are mounted on a back plate (Figs. 1 and 3).
2. The whole arrangement of canisters is contained in a tightly closed box called a blanket box (Fig. 4).
3. The plasma facing surface of the blanket box consists of the first wall. The back side of the blanket box is the plate on which the canisters are mounted. This back plate is formed by poloidal coolant helium feeding manifolds (Fig. 4).
4. At the back of the blanket box there are the main coolant helium feeding tubes. These are contained in a closed box which is fixed to the back of the blanket box. Blanket box plus feeding tubes box form the segment box (Fig. 2).
5. A helium cooled vertical radial shield is provided at the back of the segment box (Fig. 1)
6. A horizontal shield is installed inside the segment box upper part above the blanket (Fig. 1).
7. The blanket box and blanket structure are cooled by helium at 8 MPa. The coolant flows in series through the blanket box and the blanket structure.
8. The blanket structure consists of vertically arranged 21-mm-thick beryllium plates with 6-mm slits in between. The slits are filled with a bed of $Li_4SiO_4$ pebbles of diameter in the range 0.35 to 0.6 mm. A coolant coil of the shape seen on Fig. 5 is embedded in each beryllium plate.
9. A separate purge gas system at 0.08 MPa carries away the tritium generated in the breeding material.
10. For safety reasons, the coolant flow is divided into two completely independent coolant systems, which alternately feed consecutive coolant tubes.
11. Passive plasma stabilization requires a saddle loop of high electrical conductivity. This is achieved with 13-mm-thick copper plates attached on the side walls of the segment box and an increased thickness of the first-wall plate in

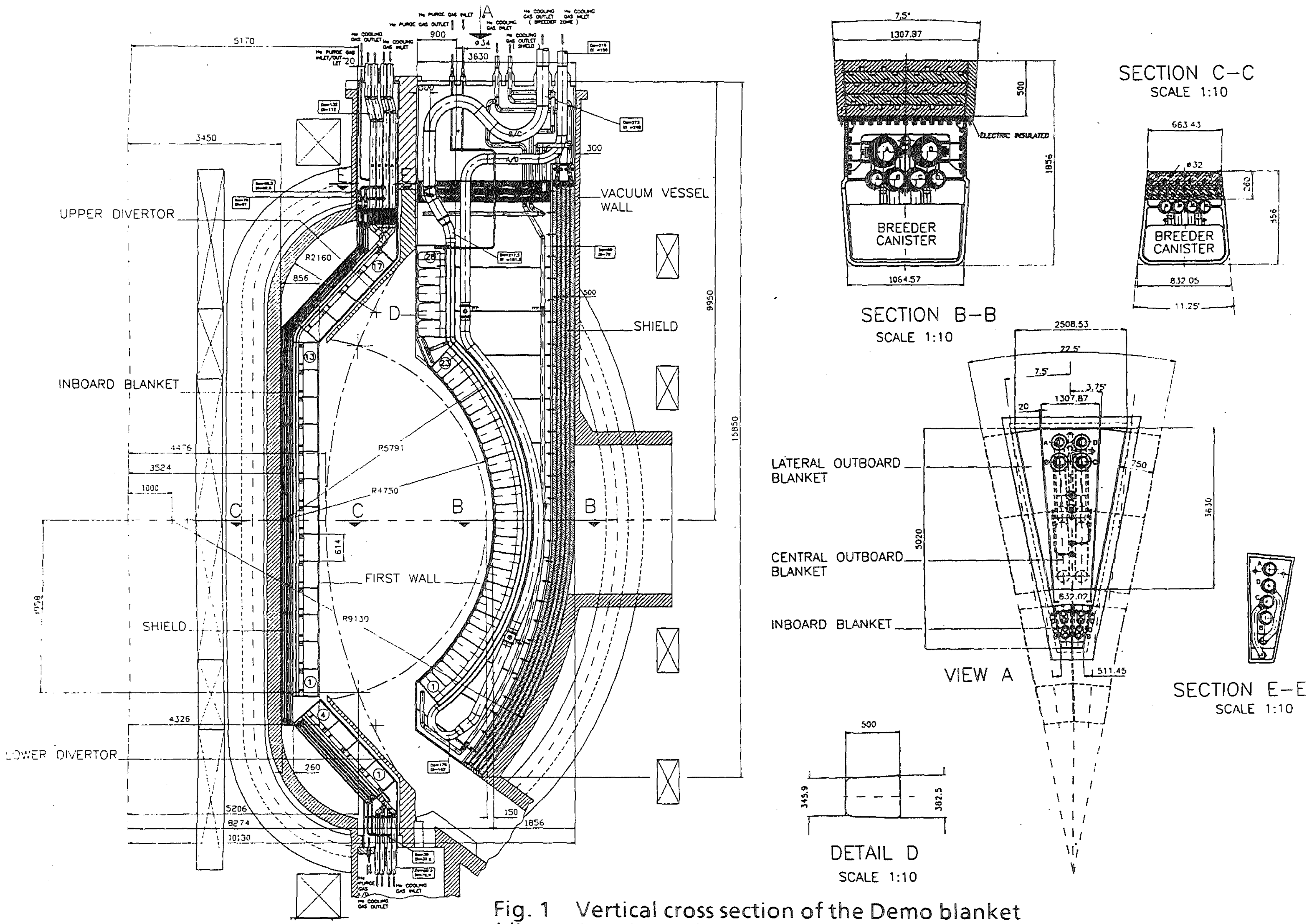



Fig. 1 Vertical cross section of the Demo blanket (dimensions in mm).

two 500 mm-wide regions at the poloidal angle of 60 deg from the plasma magnetic axis (Fig. 3).

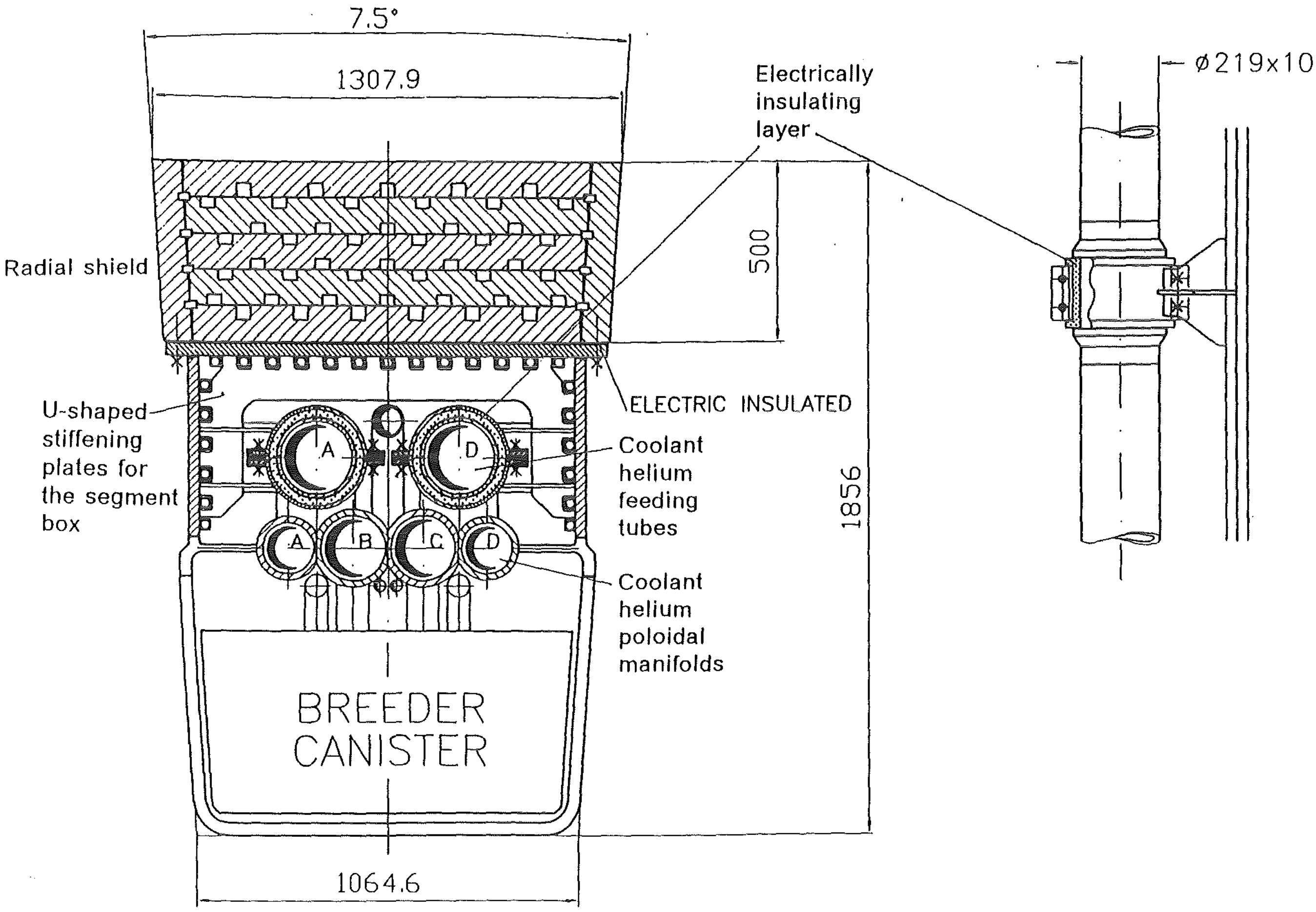



Fig. 2 Left: equatorial cross section of the out-board segment box (Section B.B. of Fig. 1). Right: electrically insulated clamp supporting the main feeding gas pipe (dimensions in mm).

The inboard blanket segment mainly consists of blanket box, piping and radial shield, all contained in the segment box. A horizontal shield is installed in the upper segment region to protect the TF-coils, the flange region and the piping above from neutron radiation (Fig. 1).

The radial shield consists of steel blocks with cooling channels which are welded together. The steel blocks should contain 10 ÷ 20 % zirconium hydride pellets to reduce the neutron fluence in the vacuum vessel and in the magnets below allowable limits.

Fig. 6 shows a toroidal-radial cross section of the inboard segment at the equatorial plane. The general arrangement in the blanket region is similar to that of the outboard segment. However, the coolant coils, the beryllium plates and the pebble beds are placed in toroidal-radial planes, rather than poloidal-radial ones like in the outboard blanket. This is due to the fact that the segment cross section in this plane is unchanged in poloidal direction, thus allowing the use of the same unit cell of breeder, multiplier and coolant channel for the whole inboard blanket. For the outboard blanket segment this is only possible by placing the breeder-multiplier-coolant channel unit cells in poloidal-radial planes, due to the variation of the segment width in toroidal direction.

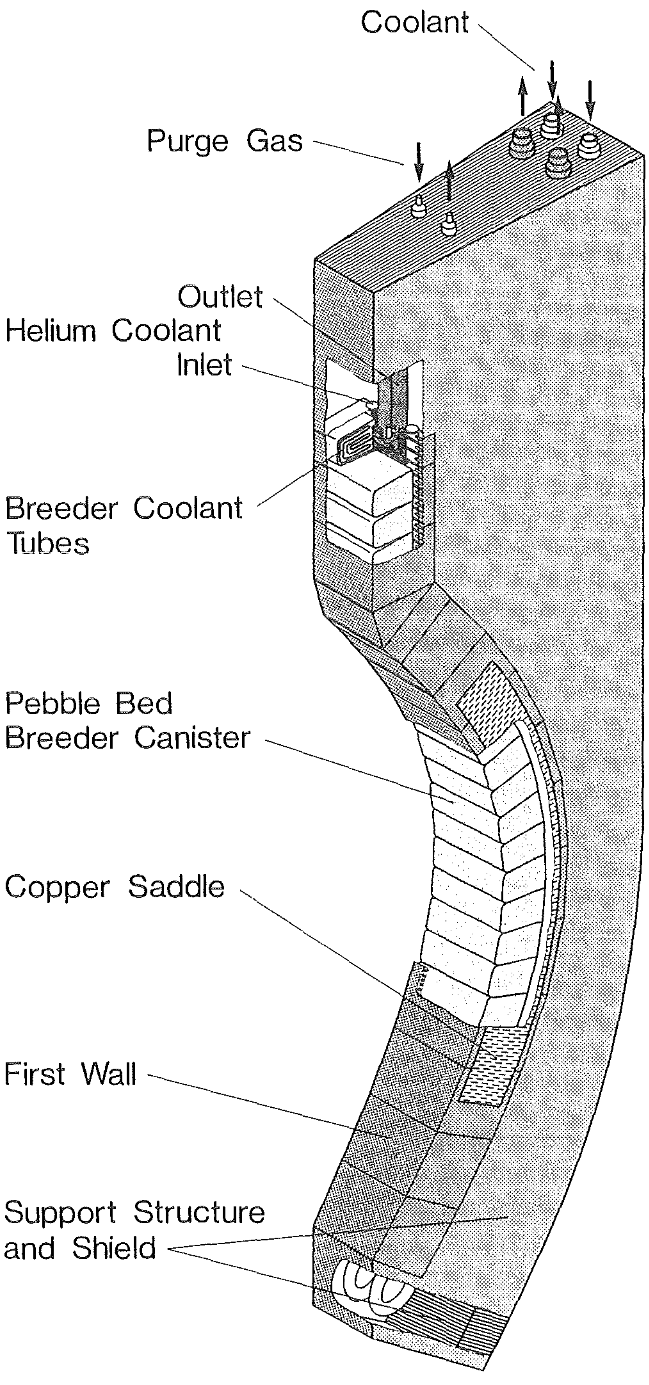



Fig. 3 Isometric view of the out-board blanket segment.

## 2.2 <u>Neutronic, Thermohydraulic, Stress Calculations. Forces Caused by Disruptions</u>

The power distribution, the local tritium production and the tritium breeding ratio have been evaluated by means of three dimensional Monte Carlo calculations using the code MCNP. These calculations account also for the presence of 10 ports placed at the equatorial plane of ten outboard blanket segments [7, 8]. The neutronic calculations have allowed to design properly the shields behind the blankets to maintain the radiation loads on the TF-coils within

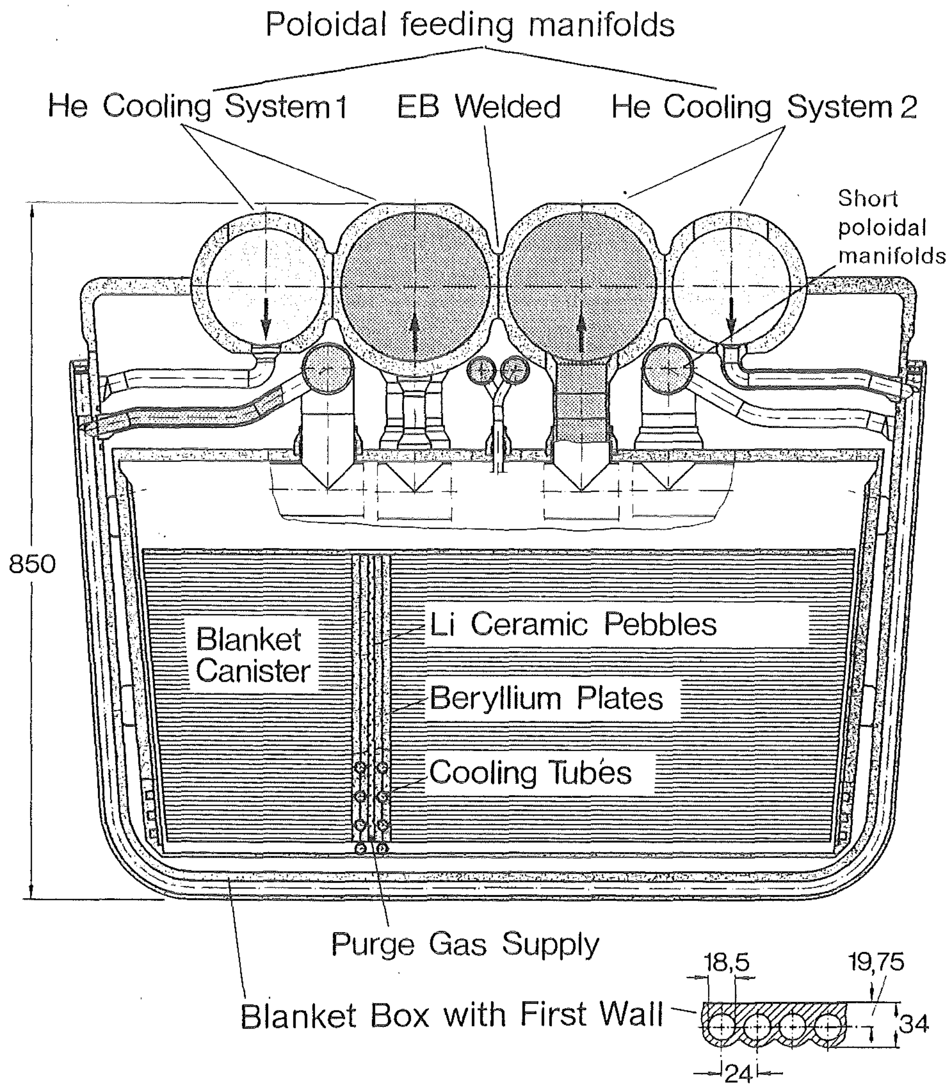



Fig. 4 Radial-toroidal cross section at the equatorial plane of the outboard blanket box and (below) radial-poloidal section of the first wall (dimensions in mm)

acceptable limits. The blanket temperatures and stresses in the structural material (first wall and boxes) have been evaluated by tridimensional calculations using the computer code ABAQUS. Fig. 7 shows as an example of the calculations the temperature distribution in the beryllium and in the $Li_4SiO_4$ pebble bed at the equatorial plane of the outboard blanket. Temperatures and stresses are within acceptable limits [7].

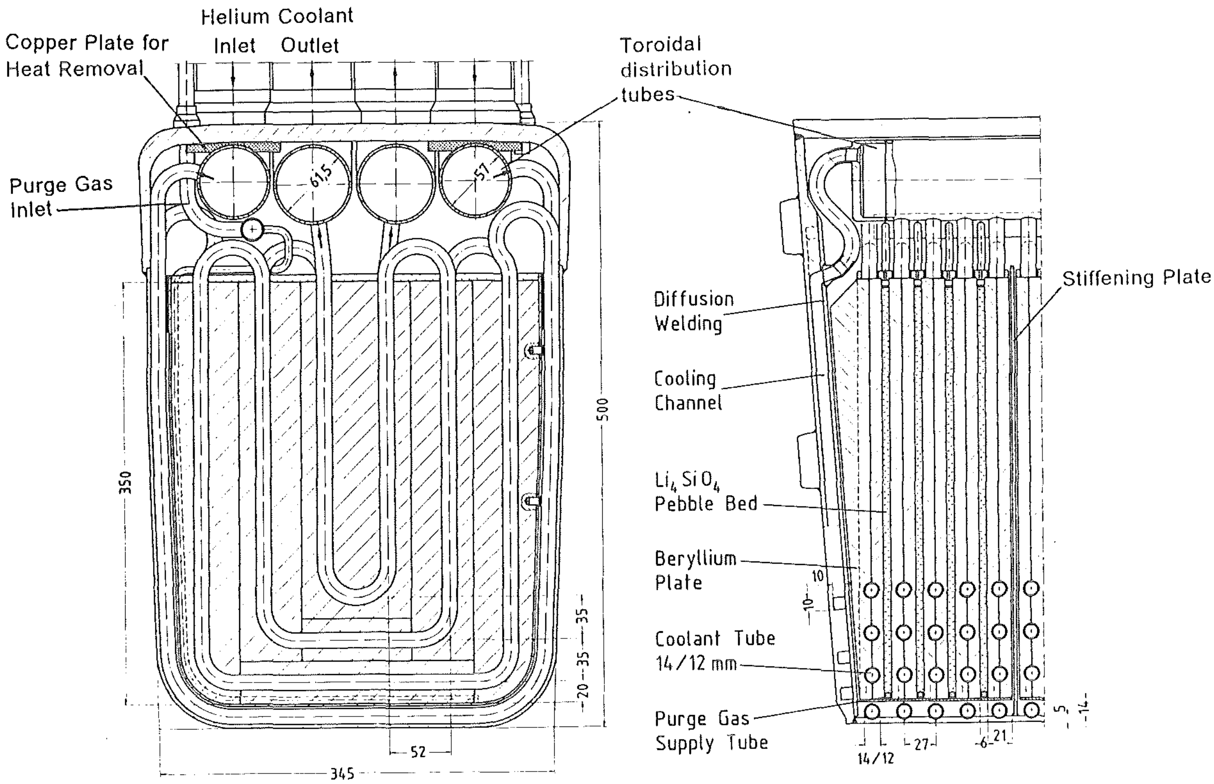



Fig. 5 Radial-poloidal (left) and radial-toroidal (right) cross section of a outboard canister (dimensions in millimeters).

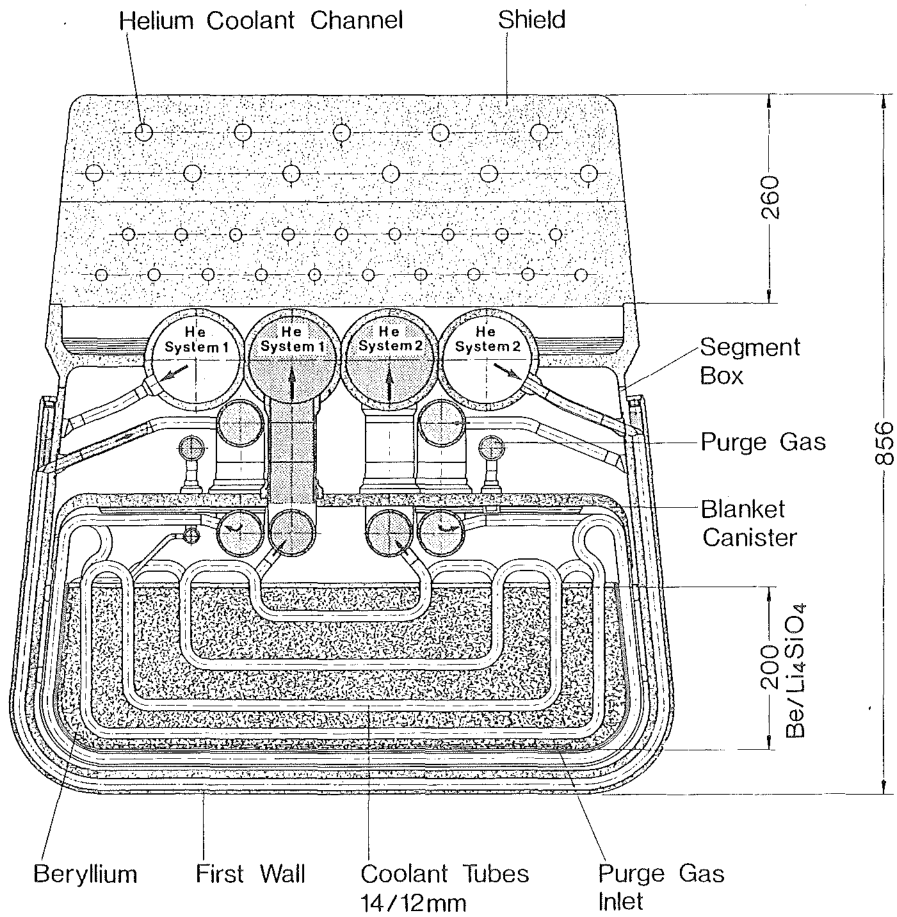
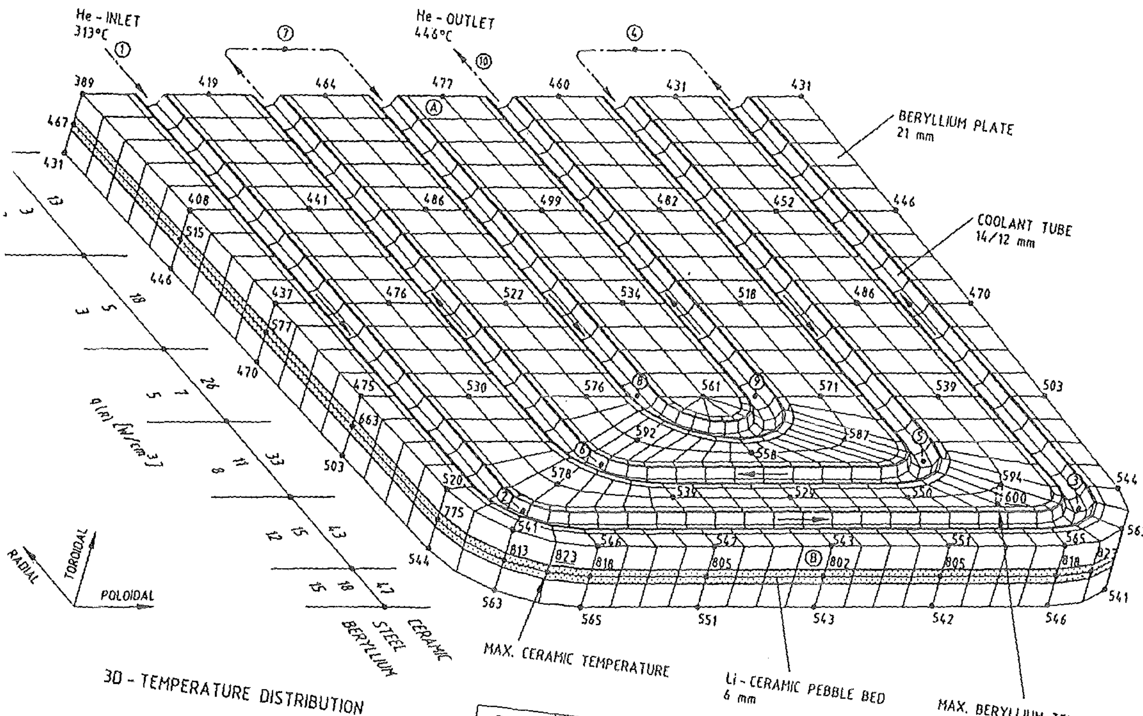



Fig. 6 Radial-toroidal cross section of the inboard segment box (dimensions in millimeters).




| POINT | 1 | 2 | 3 | 4 | 5 | 6 | 7 | 8 | 9 | 10 |
|---|---|---|---|---|---|---|---|---|---|---|
| He - TEMP. (°C) | 313 | 332 | 363 | 380 | 394 | 410 | 421 | 431 | 437 | 446 |

He - MASS FLOW PER TUBE: $43.5 \cdot 10^{-3}$ kg/s (COUNTER FLOW IN ADJACENT COILS OF TUBE)

Fig. 7 Tridimensional temperature distribution in the beryllium and in the $Li_4SiO_4$ pebble bed at the equatorial plane of the outboard blanket.

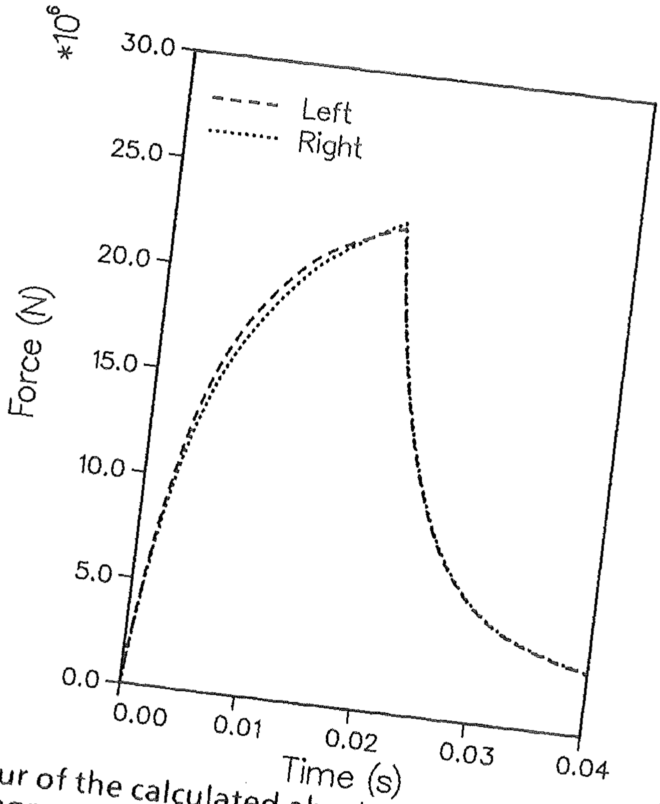



Fig. 8 Time behaviour of the calculated absolute value of the resultant Lorentz forces on the Demo segment box. Each curve refers to a half of the blanket structure in respect of the vertical middle plane: "left" in the positive vertical direction and "right" in the negative vertical direction.

Table 1 shows the main results of these calculations. The helium temperatures of the cooling system (inlet 250°C, outlet 450°C) are modest compared to those obtained in high temperature gas-cooled fission reactors. Although detailed calculations have not yet been performed, preliminary estimates show that these temperatures would be sufficient to allow a plant efficiency comparable to that of a pressurized water reactor ($\approx$ 33 %). The helium temperature limitation is given by the conservative choice of the structural material (martensitic steel) and the relative limitation of 550°C maximum operation temperature. Higher temperatures could be achieved by the use of a more advanced structural material, such as for instance a molybdenum alloy as proposed for a Japanese blanket design [9].

The obtained tritium breeding ratio (TBR) value of 1.11 (which accounts for the presence of ten ports on the outboard blanket) is high enough to ensure a sufficient margin against the uncertainties in the TBR calculations (estimated at 5 %) and the reserve to compensate for the tritium decay and tritium inventories in the system (estimated at 1 - 2 %).

The total amount of beryllium is rather high (228 tonnes). Neutronic calculations have shown that this value could be reduced to 100 tonnes with a loss in TBR of only 4 %. However, design work for this alternative has not yet been performed.

Calculations have been performed of the electromagnetic forces (Lorentz forces) caused in the blanket by a large plasma disruption. By the reference plasma disruption it is assumed that the plasma current decreases linearly from the initial value of 19.8 MA to zero in 20 ms. The calculations have been performed with the computational system CARIDDI / KfK. Fig. 8 shows a typical result of these calculations. CARIDDI / KfK cannot take account of the ferromagnetic behaviour of the structural material MANET. An appropriate computer code for this purpose is under development. For the time being the ferromagnetic effects will be accounted for by a separately estimated safety factor. A stress analysis at the time of occurence of the maximum forces will be performed in the future to compare different blanket support options.

## 2.3 Tritium Inventories and Control, Tritium Extraction

The evaluation of the tritium inventories and the tritium control in the blanket are particularly important as the Demo blanket has to show the capability of producing sufficient tritium for a continuous plasma operation (TBR $>$ 1). Similarly to the blanket design for NET [3], the tritium extraction and control are based on a tritium purge flow system using helium plus 0.1 % hydrogen at subatmospheric

**Table 1** **Main characteristics of BOT Helium Cooled Solid Breeder Blanket for the Demo reactor.**

| | |
|---|---|
| Breeding material: | 0.35 - 0.6 mm $Li_4SiO_4$ pebbles ( 90 % $^6Li$ enrichment), total amount = 60 tonnes |
| Multiplier: | beryllium, total amount = 228 tonnes |
| Total blanket power: | 2500 MW ( + 300 MW in the divertors) |
| Coolant helium temperature: | inlet = 250 °C<br>outlet = 450 °C |
| Coolant helium pressure: | 8 MPa outboard, 10 MPa inboard |
| Coolant helium pressure drop (first wall, blanket, feeding tubes): | 0.26 MPa outboard, 0.4 MPa inboard |
| First wall maximum steel temperature: | 550 °C |
| Max. temp. in beryllium: | 600 °C |
| Max. temp. in pebble bed: | 820 °C |
| Min. temp. in pebble bed: | 380 °C |
| Peak thermal and pressure load: | 437 MPa (Von Mises, primary plus secondary stress, at first wall, outboard equatorial plane, T = 481 °C) |
| Real tridimensional tritium breeding ratio (assuming ten 3 x 1 m ports on outboard blanket for heating systems and others ): | 1.11 |
| Tritium production rate: | 390 g/d |
| Peak burnup in $Li_4SiO_4$: | 10 at% referred to total lithium, 22 dpa |
| Peak fluence in Manet: | 70 dpa |
| Peak fluence in beryllium: | 2.2 x $10^{22}$ n/$cm^2$ (E > 1 MeV), 15000 appm He |

pressure to extract the major fraction of the tritium produced in the blanket. Furthermore, 0.1 % of the helium mass flow is continuously extracted from the main helium coolant circuit and sent to a helium purification plant for the extraction of the impurities and of the tritium coming by permeation from the purge flow or directly injected from the plasma. The assessment of these tritium quantities has been performed with the methods illustrated in Ref. [3] and [10]. The permeation data for Manet are based on the experimental results of Ref. [11].

Table 2 shows the main results of these calculations. The greatest tritium inventory is in beryllium. This has been calculated on the assumption that all the tritium produced in the beryllium is trapped in it. Recent experimental information shows that beryllium irradiated to a total fluence of $5 \times 10^{22}$ $cm^{-2}$ ($E \geq 1$ MeV) releases 99 % of tritium only at temperatures above 600 °C [12]. These data might not be completely relevant to the present blanket as the beryllium of Ref. [12] was irradiated at low temperatures (< 75 °C), was probably considerably anisotropic and contained a relatively large amount of BeO.

The tritium purge system data appear quite feasible. The tritium losses by leakage are neglibible. The tritium losses by permeation from the purge flow system to the main helium system are considerably higher than in the case of NET. This is due to the fact that the permeability of tritium through Manet is considerably higher than through austenitic steels [11].

The tritium inventory in the first wall and the tritium direct losses from the plasma to the main helium cooling system are known only very roughly. They have been calculated with the code DIFFUSE for a similar blanket [13]. The amounts of tritium coming directly from the plasma by permeation may be very large (estimated at 1-100 g/d). Only with an effective oxidizing atmosphere in the helium main cooling system and / or permeation reducing coatings on the permeating surfaces could it be possible to reduce the tritium losses to the environment to the desired value of less than 10 Ci/d (0.37 TBq/d).

Tritium can be separated more easily from helium than from other coolants such as water or liquid metals. Various techniques have been sucessfully applied to helium cooled fission reactors. However, in the case of a fusion blanket the amounts of tritium are much higher than in fission reactors, so that special techniques have to be developed. Several options have been considered, namely:

a) Cryoadsorbtion / permeation / electrolysis concept: $H_2O$ and HTO is adsorbed in a cold trap at -100 °C and then the water is electrolysed. $H_2$ / HT with all other impurities are separated from the helium by sorbtion in a molecular sieve at -196 °C. Then the hydrogen species can be separated from the

**Table 2** **Tritium inventories and control**

---

Tritium inventories:

- in $Li_4SiO_4$ pebbles = 10 g
- in first wall: 3 to 300 g
- in beryllium at the end of blanket life: 2080 g
- in solution in blanket structural material: 0.15 g

---

Tritium purge system:

- Total purge helium mass flow: 0.67 kg
- Average helium pressure: 0.08 MPa
- Purge helium velocity in the bed: 0.3 m/s
- Pressure drops in the bed: 0.011 MPa outboard, 0.006 MPa inboard
- HT partial pressure in purge helium: 0.85 Pa outboard, 0.44 Pa inboard
- H/HT ratio: 94 outboard, 182 inboard

---

Tritium losses (neglecting the direct losses from plasma to main helium cooling system)

by permeation from purge system to main helium coolant system = 2.9 g/d

by permeation from the main helium coolant system to water /steam circuit:

reducing atmosphere in He-system : 175 Ci/d (6.5 TBq/d)

oxidizing atmosphere in He-system $<$ 10 Ci /d (0.37 TBq/d).

---

remaining helium and the other impurities by means of a palladium / silver permeator.

b) Freezer / adsorbtion process concept: this process is similar to the previous, however here, rather than by the use of a permeator, HT / $H_2$ is separated from the impurities and from the helium by the successive use of an adsorber at 75 K and a freezer at 10 K.

c) If most of the tritium is in form of molecular hydrogen the use of a permeator to extract the hydrogen isotopes from the purge gas may be an attractive alternative.

d) If, on the other hand, a significant fraction of the tritium in the purge gas exists in the oxide form, it appears preferable to convert all the hydrogen into tritiated water and process the latter by liquid phase / vapor phase catalytic exchange.

So far, no decision has been taken on a particular option. This will also depend on the results of planned in-situ tritium release experiments to determine the HT / HTO ratio in the purge flow at the blanket outlet.

## 2.4 Safety and Reliability Considerations

The safety problems of a helium cooled solid breeder blanket are similar to those of a helium cooled fission reactor, so that a lot of experience is available. In respect of fission reactors the fusion reactor has, however, considerable safety advantages:

a. Generally an accident causes an increase of the impurities in the plasma, which cause a plasma disruption. The power production in the blanket is rapidly switched off.

b. The after-heat in the blanket is much smaller than in a fission reactor, (it is essentially produced in the steel, the after-heat produced in the orthosilicate and in the beryllium is negligible), after a very short time it is only about 1 % of the full power production.

Helium, an inert, one-phase and transparent coolant, offers considerable safety advantages. However, it loses its good heat transfer properties by the decrease of pressure and velocity. For this reason the safety investigations have the objective of maintaining the high pressure and mass flow of the helium coolant. Detailed safety investigations for the Demo blanket and its helium and water / steam circuits have not yet been performed. They will be carried out during 1992 - 1994.

However preliminary estimates and considerations can be already mentioned at this stage:

a. The blanket is cooled by two completely separate helium cooling systems. Each of these systems is able to carry away the after-heat by temperatures lower than those by normal operation. By the full power production , the operation of a single coolant circuit would result in blanket temperatures higher than the normal operation temperatures. However the resulting deformations in the blanket are such that a blanket segment can still be replaced.

b. The helium / water-steam heat exchangers can be placed sufficiently above the blanket, so that the after-heat in the blanket can be carried away by the natural convection of helium, provided the helium pressure is maintained.

c. For the blanket module to be tested in NET / ITER it has been shown that even in the case of failure of both main helium cooling systems, the tritium purge gas system is able to carry away the blanket after-heat [7].

d. Preliminary investigations show that, in case of failure of both main helium cooling systems, it is possible to remove the blanket after-heat by introducing a heavy gas (argon or $CO_2$) in the gaps between the segment box and the inner structures (canisters, tubes, shields).

e. Recent experiments show that the outboard canisters fail at a pressure of 9 MPa [14], which is greater than the maximum pressure that would occur in the canister due to the failure of one or more cooling tubes. Of course, excessive pressurization of the helium purge flow system and plastic deformation of the canisters during an accident should be avoided; thus, a pressure limitation system,. for instance a burst membrane acting at ~ 0.3 MPa, should be provided. Furthermore, the canister walls under the effect of the fast neutrons could become brittle and allow only a small plastic deformation. However, the calculations and experiments show that even a complete pressurization of the canisters at relatively high pressures would not produce a catastrophic failure of the blanket.

f. The purge gas sytem, which contains the major part of the tritium, is relatively small, is operating at subatmospheric pressure and relatively low temperatures and has a double containment.

Detailed reliability investigations have not yet been performed. However also here it is possible to make some preliminary considerations. It has been frequently stated that the number of welds has an important effect on the reliabily of a system. In the present blanket design there are about 54000 tubes contained in 2016 canisters, as against 60000 tubes and 8000 modules for the CEA-ENEA Breeder Inside Tube Demo blanket. 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 [15, 16]. The two welds of each coolant tube in the present design are placed in the back region of the blanket where the smallest neutron fluences are expected, as in the case of the fission reactors. A failed weld causes a leakage of coolant helium in the purge flow system, which can be detected by the local increase of helium pressure. If the crack is small enough, this increase in pressure is small and operation with the damaged element can be continued until the next planned blanket discharge operation.

## 3. TEST OBJECT DESIGN FOR NET / ITER

### 3.1 Introduction

One of the main objectives of NET / ITER is to test Demo-relevant blankets. Presently the NET / ITER testing program foresees, first, module or submodule tests in ports placed at the equatorial plane of outboard blanket segments and later tests of whole segments or sectors.

The testing philosophy for the Demo-relevant BOT-HCSBB (Breeder Out of Tube, Helium Cooled Solid Breeder Blanket) foresees that the module testing should be made in three stages in order to reduce the risk of malfunction. First, the module should be made of austenitic steel 316 L, which is a fairly well known material, rather than Manet, and placed behind the standard water cooled NET / ITER first wall. Then, a 316 L module should be placed with its helium cooled wall directly facing the plasma. Finally, a Manet module facing the plasma should be tested.

### 3.2 Test Module Design

To assess the feasibility problems and the requirements posed by the testing programme on the NET / ITER machine, a conceptual design of a test module was made. For this purpose the first kind of arrangement, namely the module placed behind the NET / ITER first wall was chosen.

Fig. 9. shows a vertical and horizontal cross section of the test module placed behind the first wall. The module composed of a box containing six canisters, the neutron shields and the various pipes, is contained in an adapter box. The adapter box is attached to the vacuum vessel by electrically insulated bolts. The test module and the shield are mounted on rails (also electrically insulated from the box) and can be withdrawn and brought to a manipulator box by opening two horizontally slidings doors. The module inside the manipulator box can be transported to the hot cells.

Calculations were performed of the electromagnetic forces caused by the maximum disruption (linear decrease of the plasma current of 19.8 MA to zero in 20 ms). Due to the use of the electrical insulations, the stresses in the structure proved to be within the ASME norms.

Table 3 shows the geometrical dimensions and the weights of the test module.

A considerable amount of work was performed for the design of the ancillary loops. As in the case of the Demo reactor, there are two completely independent helium cooling systems, one purification system for the coolant helium and one

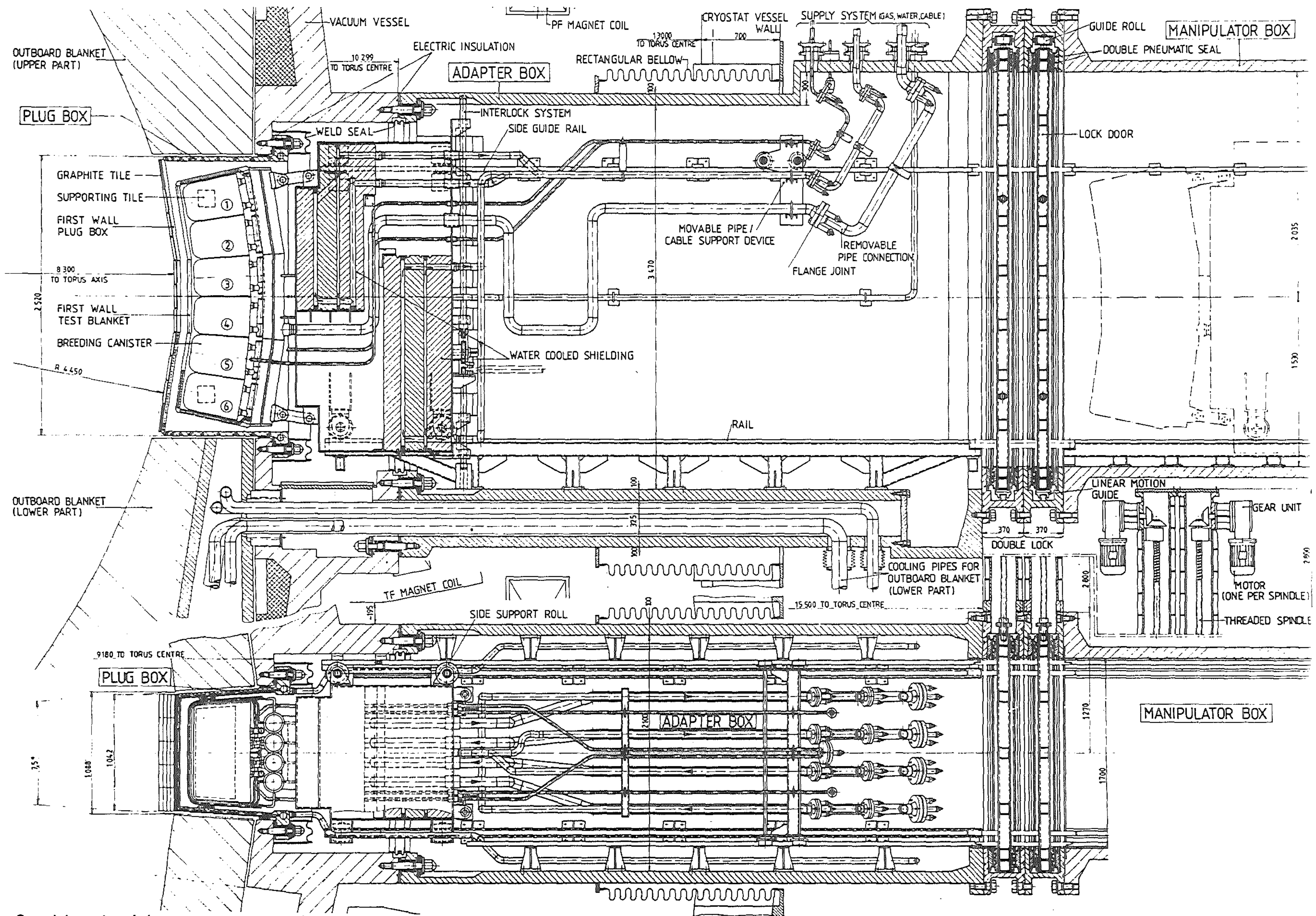



Fig. 9 Vertical (upper picture) and horizontal (lower picture) cross section of the test module placed behind the first wall in a horizontal port of the outboard region. The figure shows also the shield. Test module and shield are shown also inside the manipulator box (dotted lines) in the position used for transportation to the hot cells (dimensions in millimeters).

**Table 3: Dimensions and weights of the test module**

| | |
|---|---|
| Number of canisters | 6 |
| Main dimensions | approx. 2.5 m x 1.2 x 2.8 m |
| Total weight | appros. 12.5 t |
| Weight of blanket | appros. 2.3 t |
| Total weight at vacuum vessel flange | appros. 120 t |
| Dimensions of vacuum vessel opening | 1.2 m x 2.56 m |
| Piping dimensions | |
| - Poloidal cooling tubes | 193.7 x 18 mm |
| - First wall cooling channels | 18.5 mm |
| - Cooling gas intermediate collectors | 60.3 x 2.6 mm |
| - Cooling gas supply lines | 88.9 x 3.2 mm |
| - Purge gas supply lines | 25 x 2 mm |

**Table 4: Main design data for one cooling loop**

| | | |
|---|---|---|
| Cooling power | 1.2 | MW |
| Temperatures | | |
| Test plug inlet | 200 | °C |
| Test plug outlet | 450 | °C |
| Coolant pressure | 60 | bar |
| Pressure drop of test plug | 1.14 | bar |
| Helium mass flow rate | 0.924 | kg/s |

tritium purge system. Table 4 shows the main design data of one of two cooling loops. These data correspond to the case where the module is placed directly facing the plasma. In the case of the test module placed behind the NET / ITER first wall the power decreases by about 30 %. The power densities in the Demo blanket are about 2.5 times higher. The structural material temperatures of the Demo blanket can be obtained only by a proper reduction of the coolant helium mass flow.

Detailed calculations allowed to assess the tritium losses from the system, which are within acceptable limits (Table 5 and 6). A further interesting result was that it is possible to use the purge gas system as emergency cooling to remove the afterheat produced in the test module (Table 7).

Figures 10, 11 and 12 show a schematic representation of the ancillary loops. The conceptual design of the loops had a twofold objective, namely to assess the space requirements of the loops at the periphery of the NET / ITER reactor and to estimate the time constants of the various components and thus allow the time requirements given by the testing of the modules on the NET / ITER machine (burn time, duty cycle and duration of continous operation). These requirements are given in Chapter 6. Fig. 13 shows the chosen arrangement for the four ancillary loops. Table 8 gives the space requirements of the loops.

**Table 5: Data basis for the tritium balance of one cooling loop**

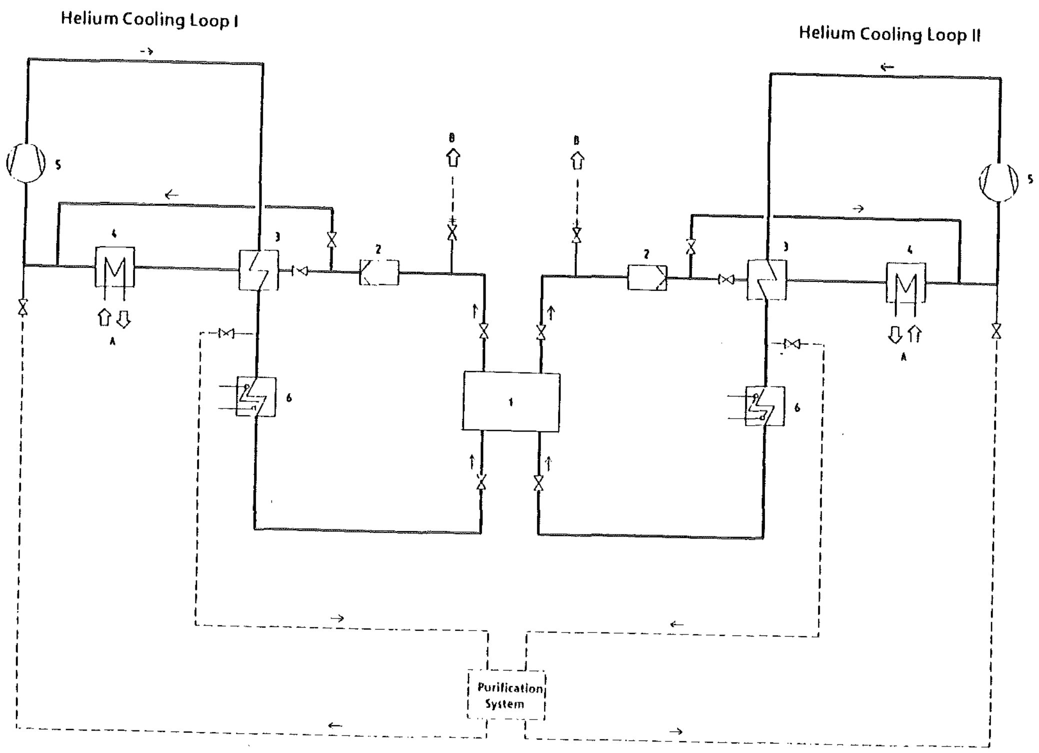
| | | |
|---|---|---|
| Temperature of pipings between: | | |
| Test plug - recuperative HX | 450 | °C |
| Recuperative HX - main cooler | 300 | °C |
| Main Cooler - recuperative HX | 50 | °C |
| Recuperative HX - test plug | 200 | °C |
| Mean He-Temperature | 280 | °C |
| He-Volume | 3 | $m^3$ |
| Partial pressures: $p_{HT}$ | 0.264 | Pa |
| $p_{H_2}$ | 100 | Pa |
| He mass flow rate (cooling loop) | 0.924 | kg/s |
| He mass flow rate (purification loop) | $0.924 \times 10^{-3}$ | kg/s |
| Tritium permeation rate into the cooling loop | 6 | Ci/d |

**Table 6: Tritium permeation rates for one cooling loop**

| | Tritium Mass Flow Rate [kg/s] | Activity Flow Rate [Ci/d] |
|---|---|---|
| In via first wall | $6.03 \cdot 10^{-12}$ | 5.0 |
| In from purge gas system | $1.21 \cdot 10^{-12}$ | 1.0 |
| Losses into components room | $2.54 \cdot 10^{-13}$ | 0.21 |
| Losses through piping walls outside components room | $7.04 \cdot 10^{-13}$ | 0.58 |
| Losses into water loop at main cooler | $7.35 \cdot 10^{-17}$ | $6.10 \cdot 10^{-5}$ |
| Separation in the cooling gas purification system | $6.27 \cdot 10^{-12}$ | 5.2 |

Table 7: Main design data for the purge gas system

| | | |
|---|---|---|
| Operating Mode "Purging" | | |
| Helium mass flow rate | $3 \cdot 10^{-3}$ | kg/s |
| Helium pressure level | 1.0 | bar |
| Blanket pressure loss | 0.13 | bar |
| He-temperatures | | |
| Test plug inlet | $\geqq$ 20 | °C |
| Test plug outlet | 350 | °C |
| Breeding zone | 450 | °C |
| $H_2$-Injection | 0.1 | Vol-% |
| Partial pressure | | |
| $p_{H_2}$ | 100 | Pa |
| $p_{HT}$ | 0.17 | Pa |
| $p_{HTO}$ | 0.03 | Pa |
| $p_{H_2O}$ | negligible | |
| Operating Mode "Emergency Cooling" | | |
| Cooling power | 10 | kW |
| He-temperatures | | |
| Test plug inlet | 50 | °C |
| Test plug outlet | 450 | °C |
| Helium pressure level | 1 | bar |




1 Test Plug
2 Dust Filter
3 Recuperative Heat Exchanger
4 He/$H_2O$ Heat Exchanger
5 Blower
6 Electrical Heater

A Cooling Water Supply
B Evacuation System

Fig. 10 Schema of the two helium cooling systems for the test module.

1 Test Plug
2 He/$H_2O$ Heat Exchanger (Main Cooler)
3 Fine Filter
4 Room-Temperature Adsorber
5 Electrical Heater
6 Pd/Ag-Permeator
7 He/$H_2O$ Heat Exchanger
8 Recuperative Heat Exchanger
9 Low Temperature Adsorber
10 Blower
11 He/$H_2O$ Heat Exchanger
12 Vacuum Pump
13 Hydrogen Intermediate Storage (U-Beds)

A Cold Water System
B Helium Supply and Storage
C Adsorber Regeneration System (Purge Gas System)
D Cooling Water Supply
E $LN_2$ Supply and Storage
F Evacuation Nozzle
G $H_2$ Supply
H Monitoring System of Double-Wall Piping
I Isotope Separation System
K Pressure Relief System

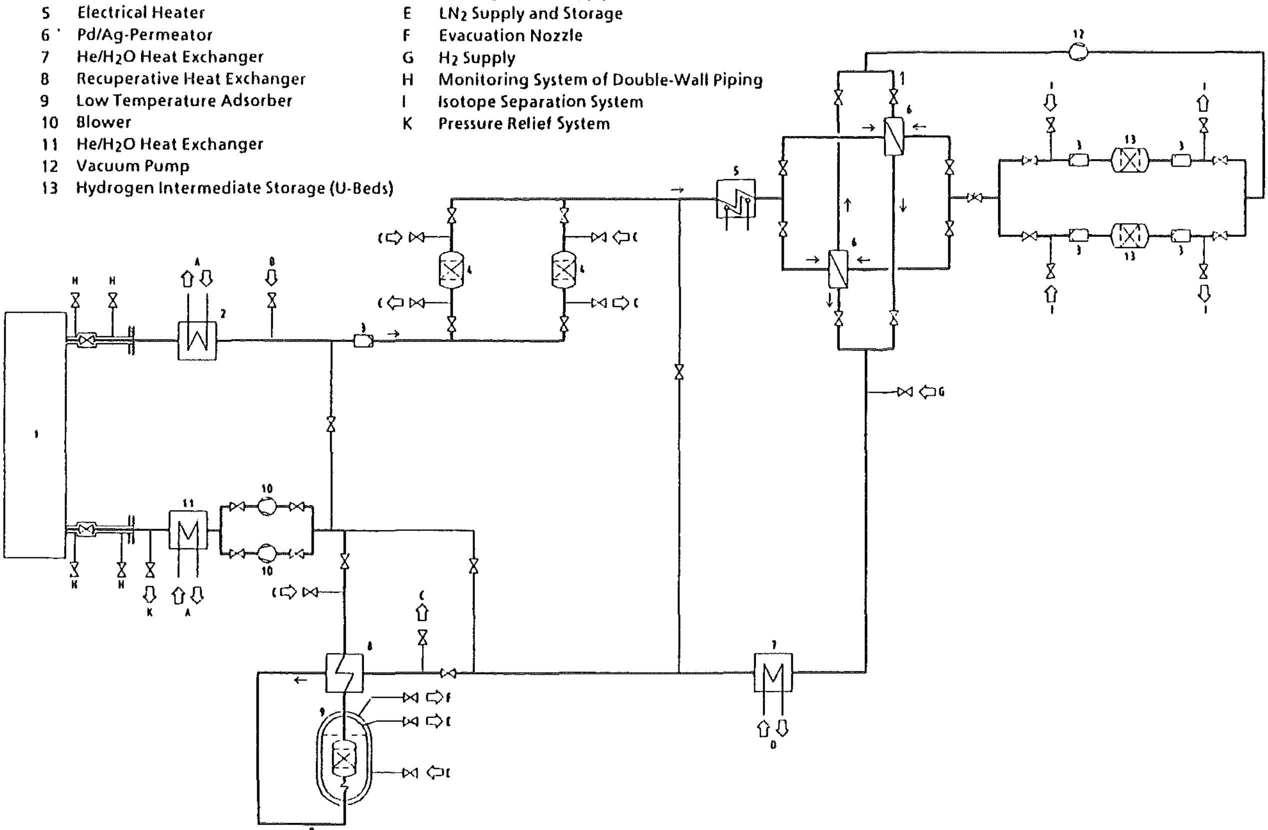


Fig. 11 Schema of the helium purification system for the helium cooling systems.

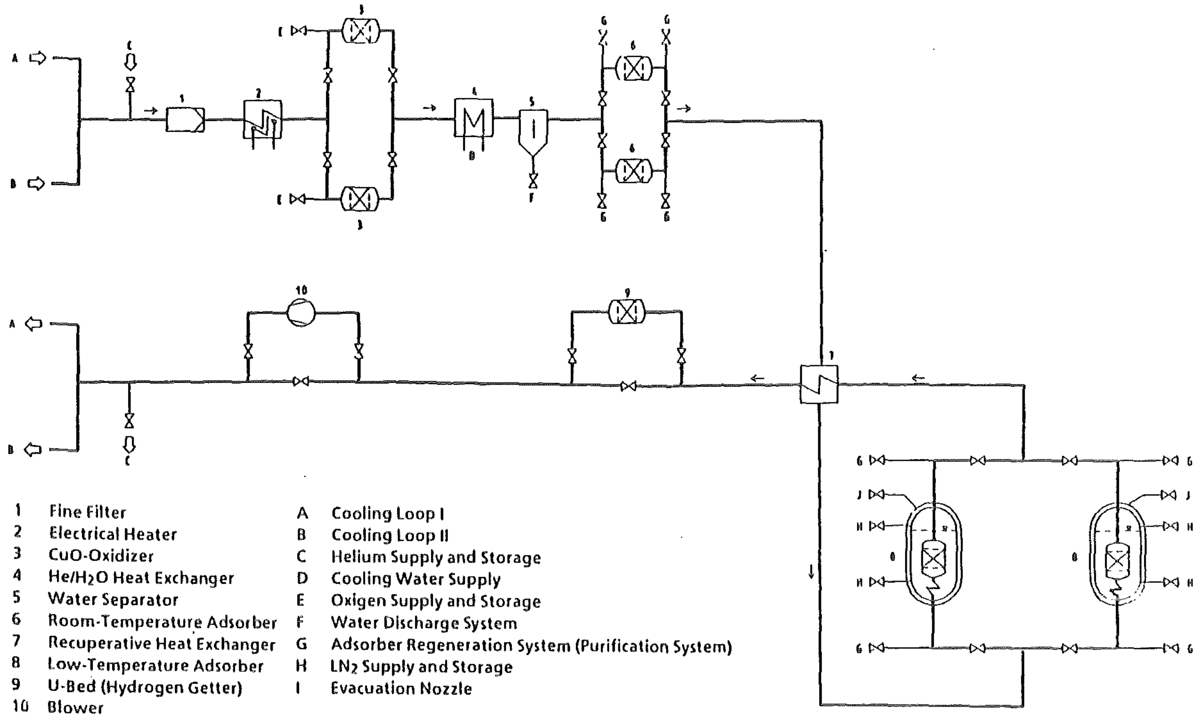



Fig. 12 Schema of the purge gas system for the test module.

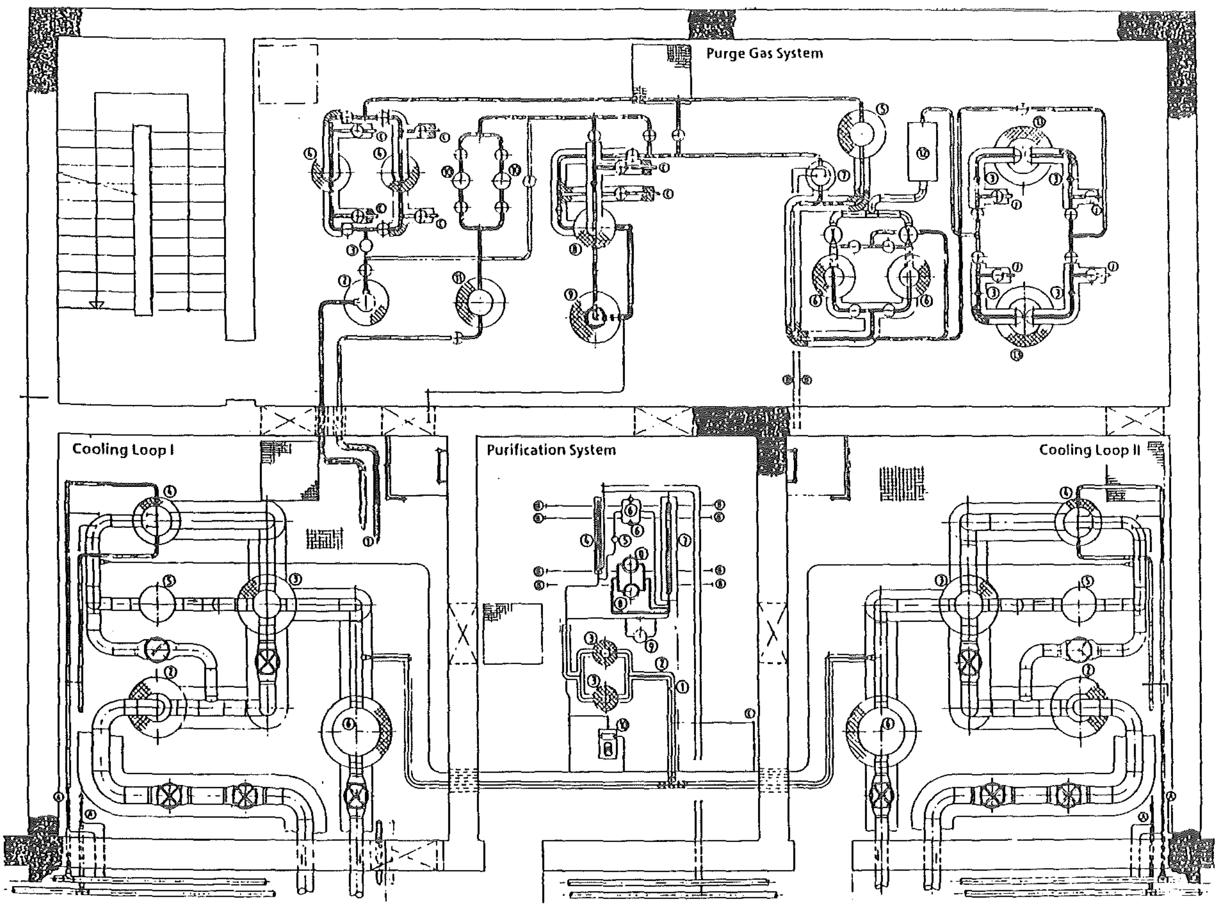



Fig. 13 Room arrangement for the test module ancillary loops

Table 8: Space requirements for ancillary systems of the NET test plug

| System | Space requirement [$m^3$] |
|---|---|
| Cooling loop I | 315 |
| Cooling loop II | 315 |
| Helium purification system | 110 |
| Purge gas system | 440 |
| Regeneration system for adsorbers of the purge gas system | 105 |
| Helium supply and storage | 315 |
| Pressure relief system | 115 |
| Evacuation system | 40 |
| Room for electrical cabinets | 70 |
| Control station | 60 |
| Water extraction system | 35 |
| Gas analysis room | 55 |
| Store for gas samples | 30 |

## 4. STATUS OF THE R & D PROGRAM

The Helium Cooled Solid Breeder Blanket project at KfK started in 1983. In these 8 years a considerable amount of work both theoretical and experimental has been performed. In the frame of the present Summary Report it is not possible to describe all this in detail. Much more information is given in Volume 2 of this Status Report [7]. The design activities for the Demo blanket and for the test objects to be irradiated in NET / ITER are summarized in the Chapters 2 and 3 of the present report. In the following we will illustrate the work which has been performed at KfK to provide the information required by the design.

### 4.1 Neutronics: Methods and Data

The neutronic calculations for the blanket layout are performed with the Monte Carlo transport code MCNP and nuclear data from the European Fusion File EFF-1. In the Monte Carlo transport procedure the accuracy of a specific calculated quantity depends on the number of events contributing to this quantity and on the involved cross-section data. Therefore, the accuracy of the calculated quantities (reaction rates, flux densities, power densities etc.) is limited mainly by the accuracy of the applied nuclear cross-section data.

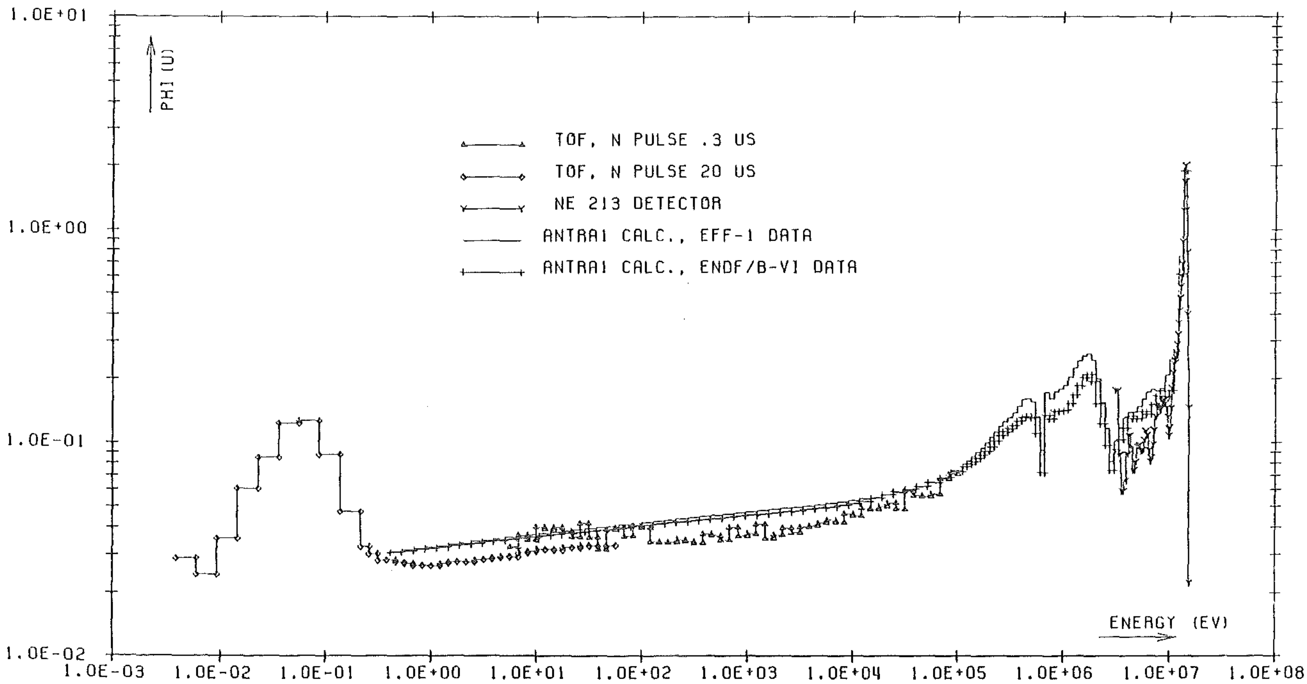



Fig. 14 Comparison of the preliminary data from KfK experiments and the calculations based on EFF-1 and ENDF/B-VI data [17]. The ordinate is leakage neutrons per source neutron per unit lethargy, the abscissa the neutron energy.

With regard to tritium breeding the most important nuclear interaction processes are the $^6Li$ (n,α)t- and the Be (n, 2n)-reaction. The nuclear cross-section of the $^6Li$ (n,α)t-reaction is rather well known and the cross-section data of the Be (n, 2n) reaction are thus the main sources of uncertainty. This holds for the magnitude of the (n, 2n) cross-section itself and even more, for the angle and energy distributions of the secondary neutrons, which are referred to as double-differential neutron emission cross-sections (DDX-data).

A beryllium transmission experiment with a 14 MeV neutron source is being performed at KfK for checking the beryllium nuclear data. In this experiment the neutron leakage spectrum of beryllium spherical shells is measured over the whole relevant energy range, i.e. from thermal energies up to 14 MeV, using various experimental techniques. Preliminary results for a 17 cm thick spherical shell have been published and are shown in Fig. 14 together with results of calculations [17].

A $S_N$-transport procedure with rigorous treatment of the neutron scattering has been developed to analyze this 14 MeV neutron experiment [18]. It avoids the usually applied Legendre approximation of the scattering kernel through the use of angle-dependent scattering matrices. Thus, it allows to use the tabular beryllium DDX-data on the very recent ENDF/B-VI and the forthcoming EFF-2 data file in an appropriate way.

Benchmark calculations and comparisons with experiments showed the validity of this computational procedure. With respect to the beryllium data, however, it is so far not possible to draw a final conclusion on their quality. Further work is required as the neutron multiplication of beryllium affects directly the tritium breeding ratio.

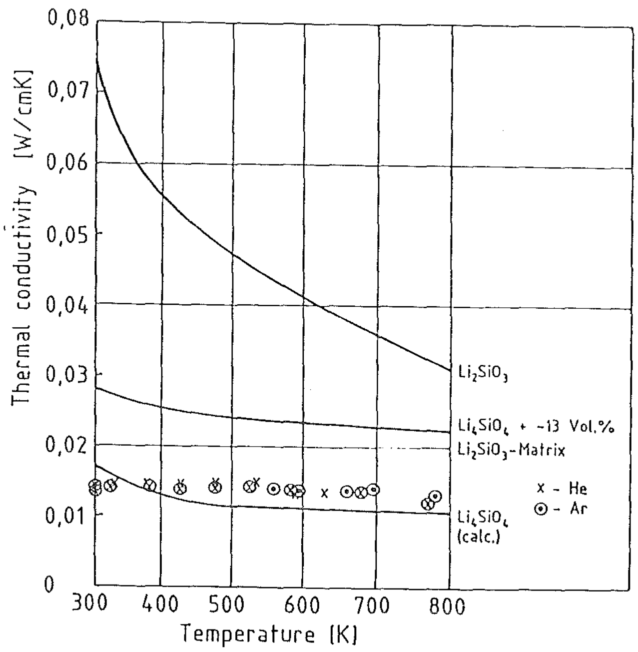



Fig. 15 Thermal conductivity of $Li_2SiO_3$, sintered $Li_4SiO_4$ ( + $Li_2SiO_3$) and pure $Li_4SiO_4$.

## 4.2 Lithium Orthosilicate Properties

The knowledge of thermophysical properties of $Li_4SiO_4$ is essential for the calculations of temperatures and stresses in the blanket. Specific heat, thermal expansion coefficient and thermal conductivity of unirradiated material were measured at KfK. Analytical correlations which account for the effects of temperature, porosity and the presence of a second phase ($Li_2SiO_3$) have been established. Fig. 12 shows the results of the thermal conductivity measurements at KfK

Similarly correlations have been produced which describe the most important mechanical properties of unirradiated $Li_4SiO_4$. These are the Young's modulus, the Poisson number, the ultimate compressive strength, the ultimate bending strength and the creep behaviour. The data are based on measurements performed at KfK and on ANL published data. The main parameters which affects those properties are temperature, porosity and grain size. Fig. 16 shows the results of the KfK measurements of the ultimate compressive strength of $Li_4SiO_4$.

The out-of-pile-compatibility capsule tests of $Li_4SiO_4$ with 316L austenitic steel and with the martensitic steel 1.4914, very similar to Manet, in presence of $H_2O$ and NiO have indicated that the compatibility limit is higher than 800 °C for both materials. Subse-

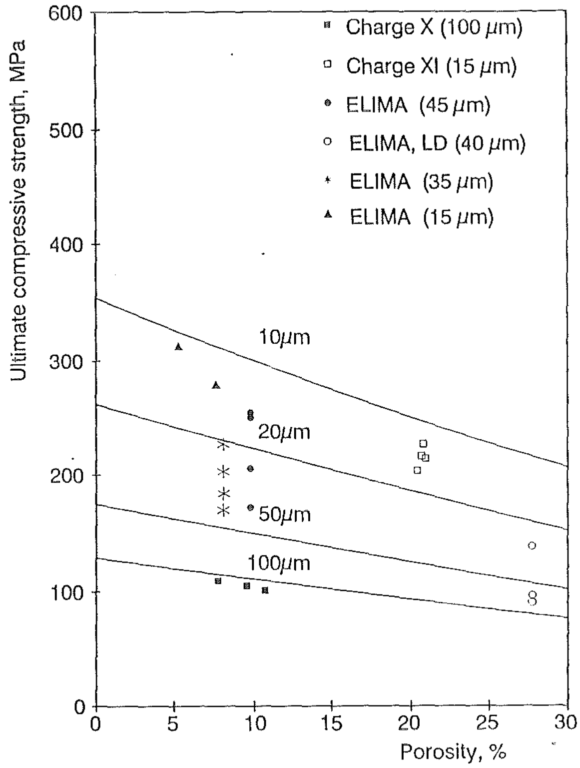



Fig. 16 Ultimate compressive strength of $Li_4SiO_4$ with different mean grain size versus porosity.

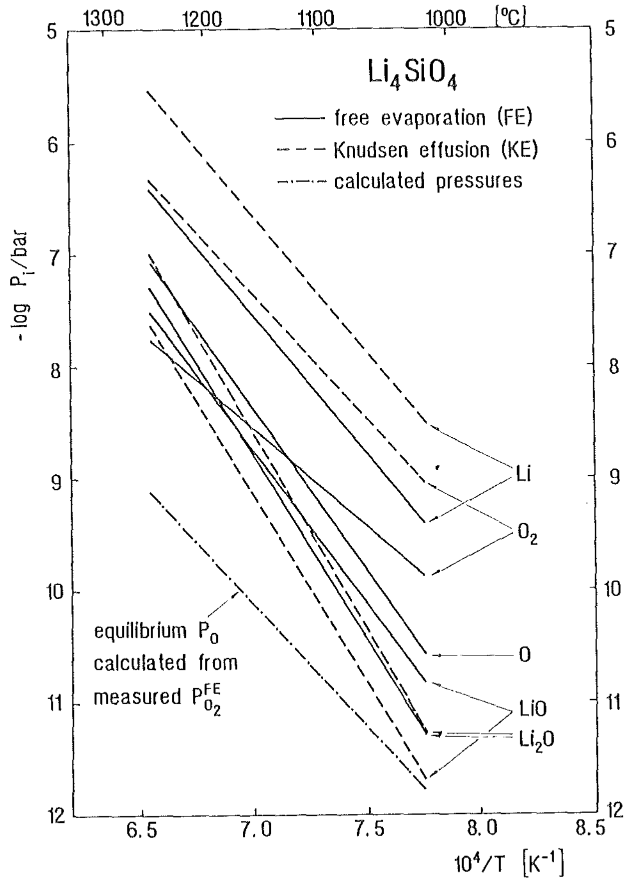



Fig. 17 Results of free evaporation and Knudsen effusion measurements at KfK.

quent tests with flowing argon and $H_2O$ at 10 and 100 Pa partial pressure respectively indicate that the temperature limit should be 800 °C. However these partial pressures are much higher that those expected in the helium purge flow system during blanket operation ($\approx$ 0.1 Pa).

The equilibrium partial pressures of Li, $O_2$, $Li_2O$, LiO and $Li_3O$ over lithium orthosilicate were measured in vacuum by the analysis of the vapors effusing from platinum Knudsen cells. Also the free evaporation of these vapors over $Li_4SiO_4$ was experimentally determined (Fig. 17). Free evaporation partial pressures are almost an order of magnitude lower then equilibrium ones. These data indicate that the temperature at which the total lithium pressure exceeds $10^{-2}$ Pa is higher that 1120 °C. This would lead to conclude that the lithium transport in the blanket would be acceptably low for temperatures lower than this limit. However, this does not account for the presence of $H_2O$ and $H_2$ in the helium purge flow. Experiments to investigate these effects have yet to be performed.

Measurements of the hydrogen solubility in $Li_4SiO_4$ show that this is very small and its contribution to the tritium inventory in the blanket is negligible.

### 4.3 $Li_4SiO_4$ Pebble Development

The reference lithium orthosilicate pebbles are fabricated by melting and spraying the liquid material by the firm Schott Glaswerke, Mainz. This production method requires rapid quenching from the melting temperature, thus some of the pebbles may contain cracks and stresses that lower the mechanical stability. The pebbles have been optimized at KfK in view to improve their mechanical properties and still maintain the excellent tritium release behaviour of $Li_4SiO_4$. The best results have been obtained so far by the use of $Li_4SiO_4$ with an excess of 2.2 wt% of $SiO_2$ over the stoichiometric composition. After melting and quenching the pebbles are annealed in a special rotating furnace. First they are heated slowly up to 1030 °C and kept at this temperature for 5 minutes. At this temperature there is a liquification of a $SiO_2$ rich phase. Afterwards the pebbles are cooled slowly down to room temperature. Scanning electron microscope (SEM) analyses of the pebble microstructure indicate quite clearly that the $SiO_2$ rich phase has precipitated in the grain boundaries healing the microcracks and preventing the scatter of mechanical properties (Fig. 18b).

With a new wind-sifting machine it has been possible to separate not only the unround pebble but the hollow ones as well. The reference pebbles have a density of about 98 % of theoretical. The reference pebble bed with pebbles in the dia-

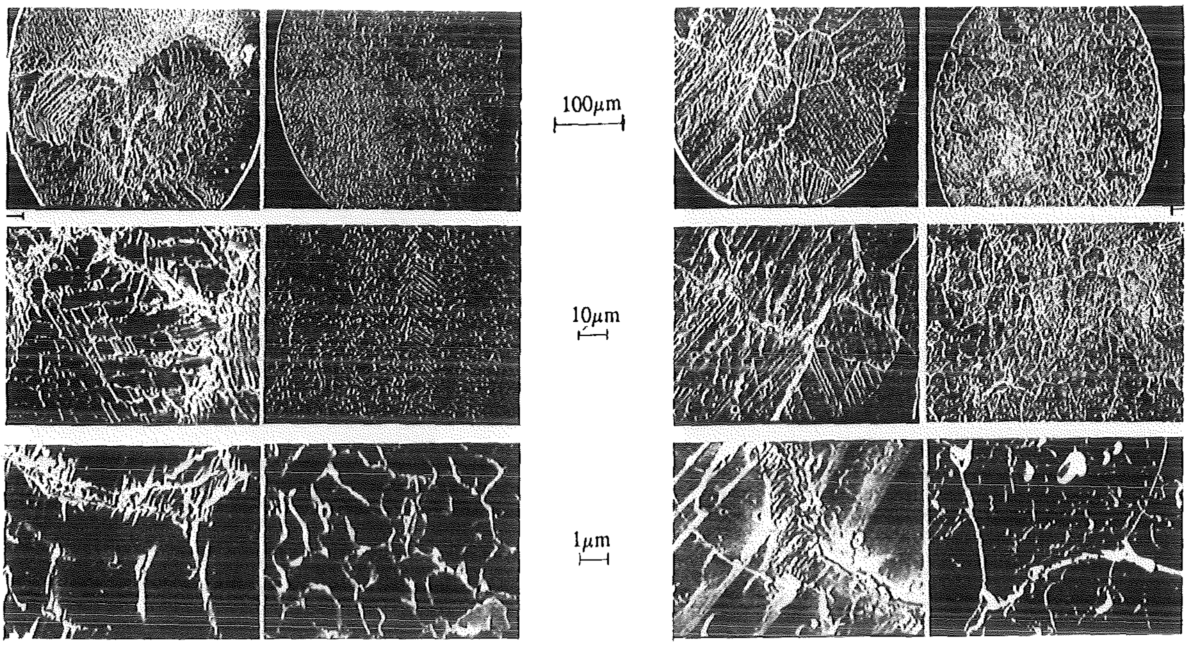



Fig. 18a SEM of a polished and etched cross section of lithium orthosilicate pebbles as produced by Schott, Glaswerke Mainz. Left: $Li_4SiO_4$, right: $Li_4SiO_4$ + 2.2 w% $SiO_2$

Fig. 18b SEM of speciments of Fig. 18a after annealing 5 min at 1030 °C. Left: $Li_4SiO_4$, right: $Li_4SiO_4$ + 2.2 w% $SiO_2$.

meter range between 0.35 and 0.6 mm has a density of 1.49 g/cm³ which corresponds to a packing factor of 63 - 64 %.

The sintered pebbles are fabricated at KfK. The $Li_4SiO_4$ sinterable powder is produced by an especially developed "methanol process". To obtain the pebbles the powder is wetted with small amounts of water, passed through a sieve, the obtained sticks are broken by shaking and the primary graines are then rounded-up on a rotating disk. Then the pebbles are dried, dewaxed, presintered at temperatures up to 900 °C, and finally sintered in a fluidized bed at 1030 °C for two hours and cleaned from dust. The pebbles have a density of 90 % of theoretical (Fig. 19). The bed of pebbles in the diameter range 0.35 - 0.63 mm has a density of 1.37 g/cm³ corresponding to a packing factor of 64 %.

An important factor for the choice of the pebbles is the optimization of their mechanical properties. At first, a screening is made by a simple text. An increasing weight load is imposed to a single pebble by a piston. The load is made by a water container that gradually fills with water. The pebble is placed on a balance that measures the load. The load at which the pebbles break, averaged through 10 tests, is called the fracture load. The pebbles which have perfomed best in this first screening are then subjected to a thermocycling test, which simulates the mechanical and thermal stresses that are imposed on the pebble bed in the blanket

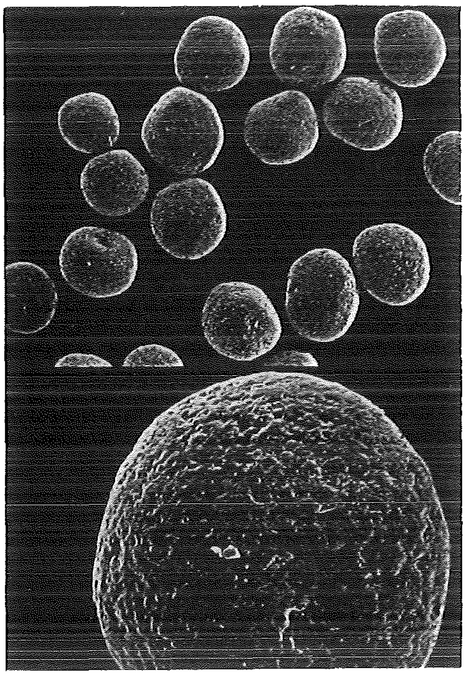


Fig. 19 Sintered pebbles of $Li_4SiO_4$.

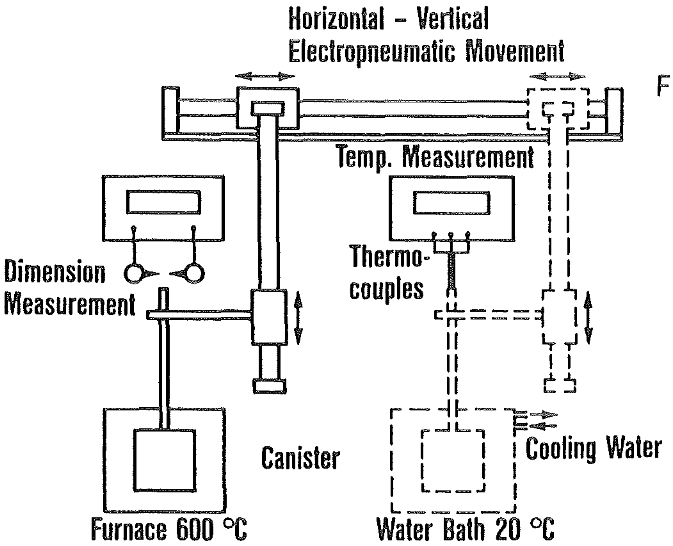



Fig. 20 Apparatus for the thermal cycle test.

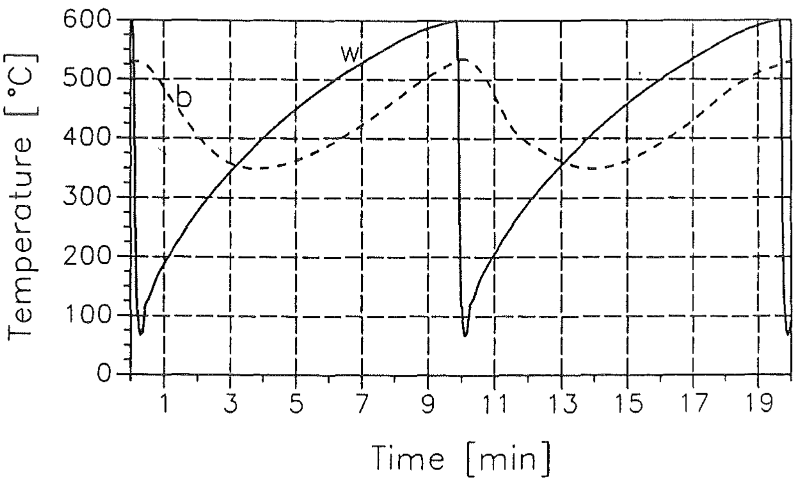



Fig. 21 Temperature in container wall (w) and pebble bed (b) during thermocycling.

during reactor operation. The pebbles are filled in a steel container with a packing factor of 63 - 64 %. The container is then subjected to rapid periodic temperature variations (Fig. 20). When the container is cooled in the water bath, its wall is at a temperature 450 - 500 °C lower than the temperature at the pebble bed center (Fig. 21), thus exerting strong compressive stresses on the pebbles. Only the pebbles which have a failure rate of 2 % or less and a negligible amount of fine fraction after this test are deemed to be acceptable. Both the pebbles fabricated by melting with 2.2 wt % $SiO_2$ excess and the last sintered pebbles have performed well by the thermal cycle tests.

### 4.4 Lithium Metazirconate ($Li_2ZrO_3$) Pebbles

Some work has been performed on $Li_2ZrO_3$ pebbles as a back-up solution in case the reference $Li_4SiO_4$ pebbles would encounter unexpected difficulties. Data on the mechanical properties have been obtained from a literature research. Compatibility tests performed at KfK indicate that the maximum allowable contact temperature with stainless steel is above 700 °C. Thermal conductivity measurements have been performed at KfK. The agreement with data available from the literature is poor. Data on the specific heat and thermal expansion coefficient have been obtained from the literature.

Sintered pebbles of 0.5 mm ± 10 % have been fabricated at KfK by the same route used for the sintered $Li_4SiO_4$ pebbles. An alternative fabrication process has been developed by the firm HITEC in collaboration with KfK. By this process the $Li_2ZrO_3$ powder is wet milled in polyvenyl alcohol and then spray-dried to obtain a powder with 4 µm grain size. Then the powder is dispersed in water containing a special wax. The granulation process performed by the firm Alpine produces the pebbles in the desired diameter range (0.5 mm ± 10 %) with a yield of 90 %. Sintering at 1200 °C for 2 hours results in pebbles of 86 % theoretical density. The pebbles have been used essentially for irradiations tests.

## 4.5 Technological Investigations

The design work for the Demo-blanket and for the test modules has pointed out which are the main technological issues on which the R. and D. work has to be concentrated. Besides the development of the breeder material other technological work is being performed at KfK. Four items appear to be very important as they may effect the blanket feasibility:

- effects of the periodic temperature variations on various blanket components,
- first wall fabrication methods,
- development and testing of the brazing of the beryllium plates to the coolant tubes,
- measurements of the thermal conductivity of the bed of $Li_4SiO_4$ pebbles.

### 4.5.1 Thermal cycling tests: the HEBLO loop

At the various stages of blanket development individual structural elements and

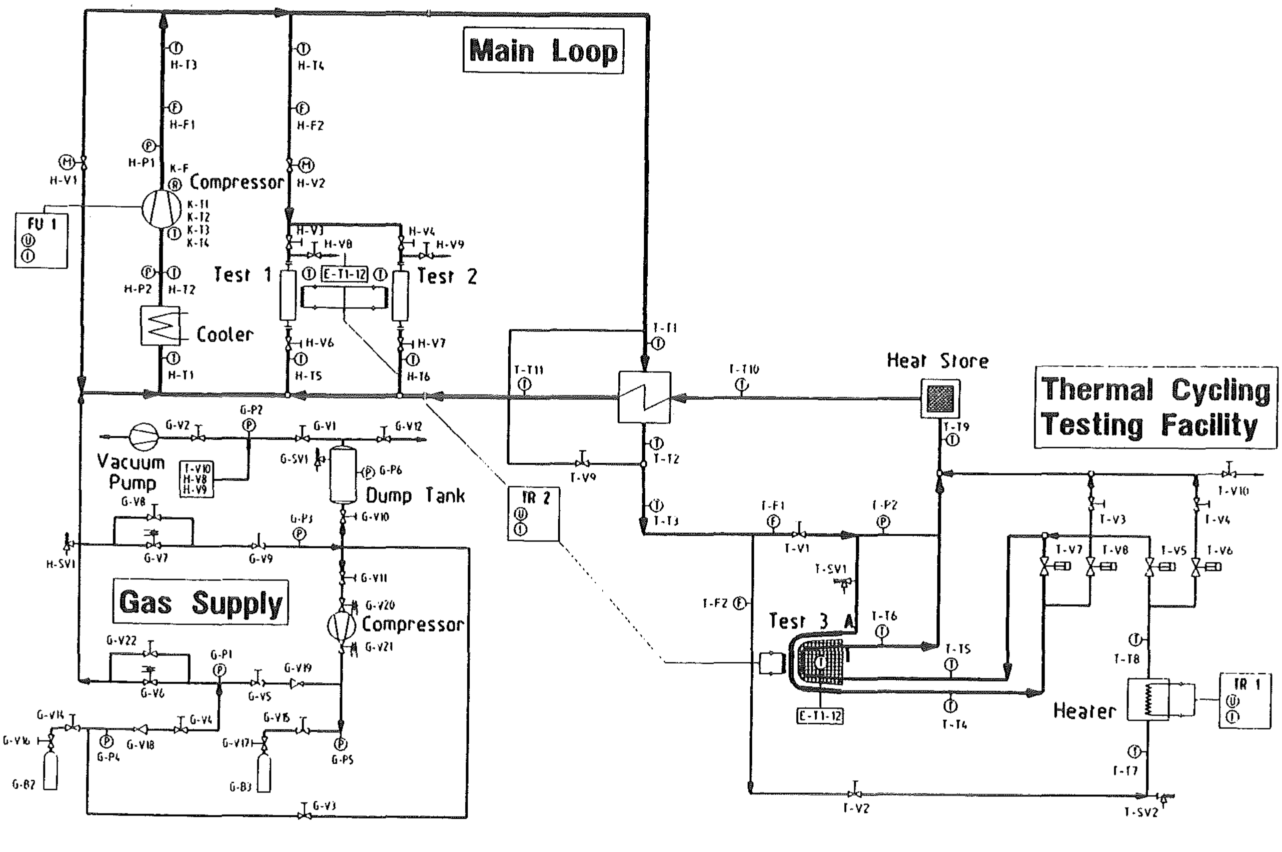



Fig. 22 Helium blanket test loop HEBLO.

original blanket assemblies have to undergo long term testing. Simulation of the loading due to the great number of thermal cycles during the service life of a fusion blanket is in the foreground of the investigations. To be able to perform such tests on small structural units a helium blanket test loop (HEBLO) is being erected at KfK. This loop will allow to perform two kind of experiments:

1) Thermal cycling tests involving parts which can be directly heated electrically. Typical examples include wall elements with brazed or welded cooling coils or channels. The thermal cycles can be generated in the test by variation of the heating power and/or the coolant flow through the coolant channels.

2) Tests on components or - preferably - on original blanket assemblies which cannot be directly heated electrically. The cyclic variation of the temperature in the test object and hence the thermal stresses must be generated here by quick variations of the cooling gas temperature.

First tests will be performed with blanket components of the KfK design, however tests for designs of other associations could be envisaged.

The layout of the HEBLO loop is represented schematically in Fig. 22. The loop consits of two parts:

The main loop accommodates as essential components the circulating compressor, the cooler, a filter, and a bypass line with control valve (H-V1) for adjusting the compressor output to a constant set value. A turbo compressor with gas bearing is used whose speed can be continuously adapted to the conditions prevailing in the experiment by use of a transistor frequency transformer (FU-1).

The characteristic data of the helium system are:

- operating pressure — 80 bar
- helium output of the compressor — 100 $m^3/h$, 0.33 kg/s
- pressure rise in the compressor — 1.5 bar

In the main loop connections have been provided for two test sections where objects for testing can be operated which can be directly heated electrically (type 1 tests). The coolant flow can be controlled by means of a control valve (H-V2) installed in the common feed line. Heating of the objects to be tested is via a continuously controllable transformer (TR-2) of 30 kW maximum power.

Connected with the main circuit will be constructed a testing facility to be able to carry out thermal cycling tests on original blanket structures (type 2 tests). This facility will be designed such that an object can be tested which consists of a maximum of eight true size beryllium plates with cooling tubes and ceramic layers in-between. The object will be subjected to a maximum of $2x10^4$ thermal cycles. The temperature variations are obtained by circulating alternatively through the test section helium at 280 °C and 460 °C respectively.

### 4.5.2 First wall fabrication methods

The reference concept of the first wall foresees coolant channels fabricated by deep-hole-boring followed by U-bending of the wall plate segment. The maximum length of the bore is 2.4 m and the diameter 18.5 mm, and the toroidal pitch of the bores is 24 mm. A total of 12 bores of 1.2 m length with the proper diameter were drilled in a 39.5 mm thick plate made of a steel similar to the austenitic steel 316 L. The tests showed that the precision requirements can be met at a relatively low cost.

An alternative fabrication method consists of milling grooves into a plate and to surface weld the plate with a metal sheet covering it. A series of experiments were performed on various sample 316 L stainless steel plates up to 240 mm in diameter, which allowed to determine the optimum temperature, time and pressure of the diffusion welding operation. Post-test investigations indicated that the bonding is good also around the narrow webs.

### 4.5.3 Beryllium-structural material brazing tests

A first series of tests on the development of brazed joints between beryllium plates and 316 L and Manet cooling pipes has already been completed at the W.C. Heraeus company, Hanau. There tests allowed to choose the optimum braze, namely foil type AgCuSn with 5 % and 10 % tin content.

Subsequent brazing tests were performed between 316 L steel tubes and two beryllium plates 100 x 40 x 10 mm in size each. These tests have demonstrated that reliable Be/Be and Be/steel brazes can be achieved with Ag60Cu30Sn10 braze at 700 °C brazing temperature. Values of about 200 N/mm$^2$ were measured for the shear strength of the steel / beryllium compound at room temperature. At 500 °C the shear strength was still at least 70 N/mm$^2$.

### 4.5.4 Measurements of the effective thermal conductivity of the pebble bed

A series of measurements of the effective thermal conductivity of pebble beds and the heat transfer coefficient at the walls of the bed container were performed by KfK in an especially KfK developed experimental apparatus.

Fig. 23 shows schematically the experimental apparatus. The pebble bed is contained between two concentrical cylinders. The inner cylinder contains an electrically heated rod. The gas can flow in axial direction through the bed.

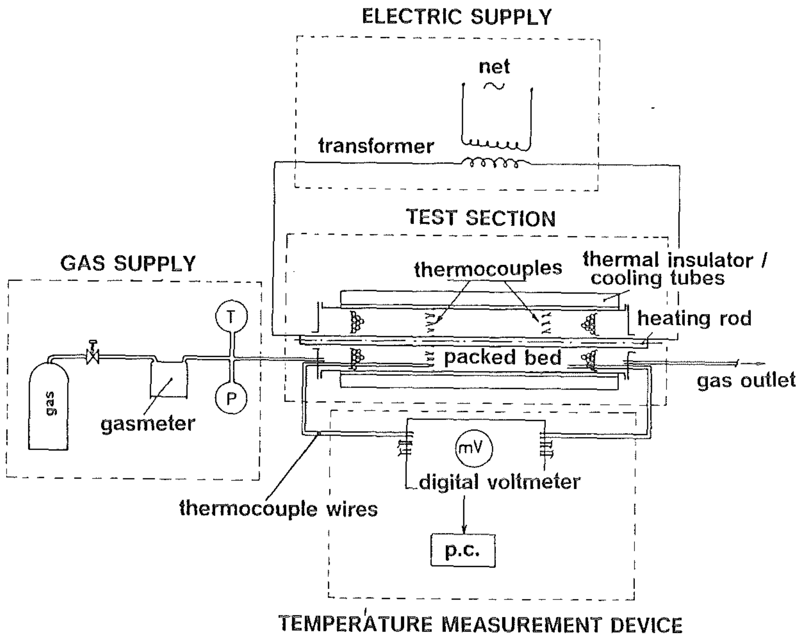



Fig. 23 Schematic representation of the experimental apparatus for the measurements of the pebble bed effective thermal conductivity and of the wall heat transfer coefficient.

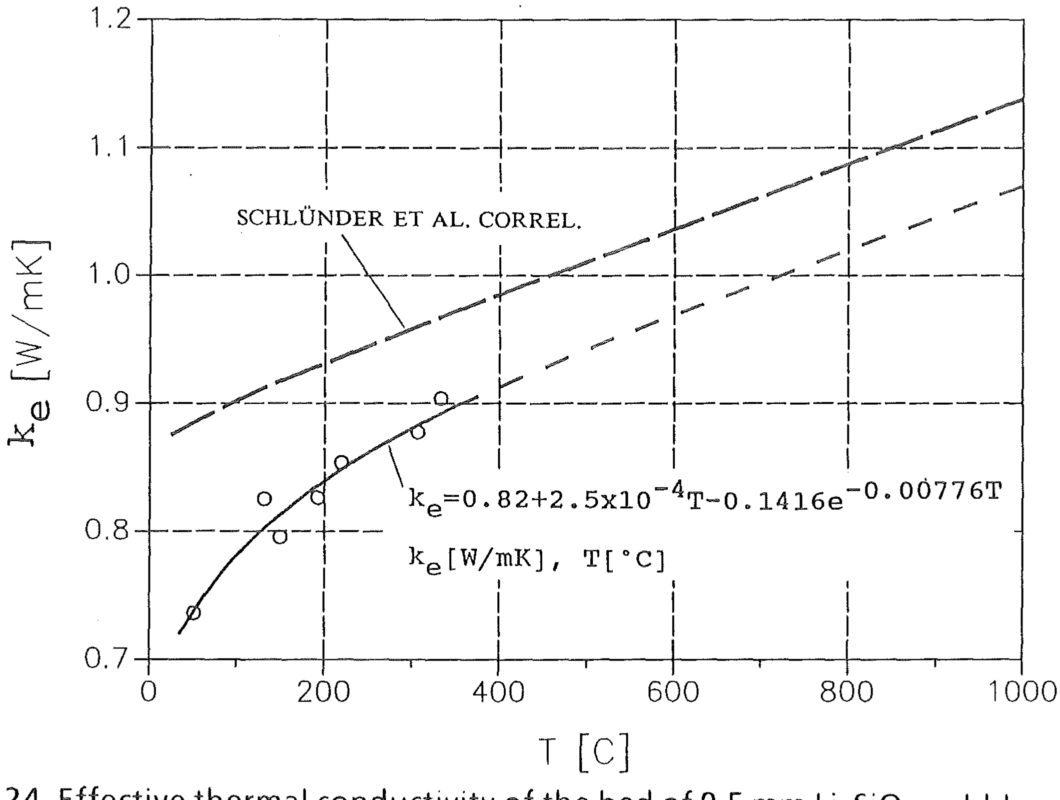



Fig. 24 Effective thermal conductivity of the bed of 0.5 mm $Li_4SiO_4$ pebbles and stagnant helium.

The radial distribution of the temperature in the bed is measured in two axial positions in the central part of the test section, by means of two banks of 32 thermocouples each, placed at various radii at four different azimuthal angles. The temperatures on the inner and outer cylinder surfaces are measured by thermocouples placed into the walls. The measurements were performed with beds of pebbles of various materials and diameters with stagnant or flowing helium or argon. Also binary mixtures of pebbles of different thermal conductivity and diameter have been investigated.

Fig. 24 shows the measured effective thermal conductivity of the bed formed of 0.5 mm $Li_4SiO_4$ pebbles in stagnant helium. Recently, effective thermal conductivity measurements have been carried out for a bed of $Li_4SiO_4$ pebbles with diameters in the range 0.35 - 0.6 mm, which is the reference pebble bed for the Demo blanket design. The measured effective thermal conductivity data differ little from those of the bed with 0.5 mm $Li_4SiO_4$ pebbles. Thus the correlation of Fig. 21 has been used for the temperature calculations in the Demo blanket.

## 4.6 Irradiations and Tritium Release of Breeder Ceramics

### 4.6.1 Irradiations

Within the framework of breeder ceramic development KfK, in cooperation with

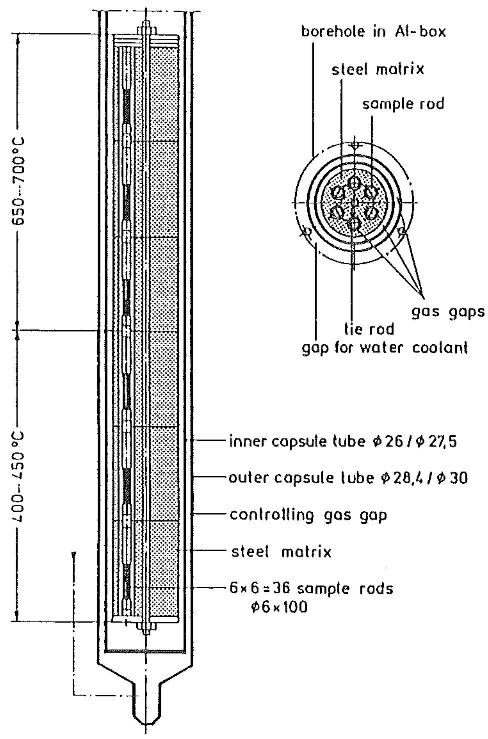



Fig. 25a Scheme of DELICE capsule.

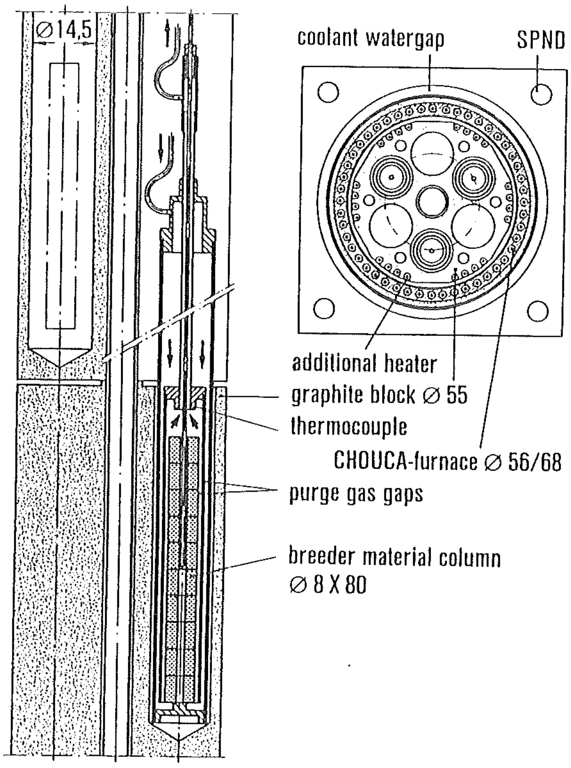



Fig. 25b Scheme of LISA capsule.

EC partners, has initiated and performed since 1983 quite a number of irradiation experiments. Table 3 is a survey of the general programme and the goals pursued with the individual experiments. Furthermore KfK is participating in the joint European irradiation programme (EXOTIC, ALICE, SIBELIUS). Fig. 25 shows the schemes for the test rigs of the DELICE and LISA irradiations respectively.

**Table 9 Irradiation experiments and goals**

| Name of Experiment | Reactor | $LiAlO_2$ | $Li_2SiO_3$ | $Li_4SiO_4$ | $Li_6SiO_6$ | $Li_2ZrO3$ | $Li_2O$ | Experimental Goal *) | Status **) |
|---|---|---|---|---|---|---|---|---|---|
| MERLIN | FRJ 1/Jülich | | X | X | X | | | A | compl. |
| DELICE 01/02 | OSIRIS/Saclay | | X | X | | | | A, B | compl. |
| LISA 1/2 | SILOE/Grenoble | X | X | X | | | | A, C | compl. |
| LILA/LISA 3 | SILOE/Grenoble | X | | X | | X | | A, C | compl. |
| TRIDEX 1-6 | FRJ 2/Jülich | | X | X | X | | | A, C | compl. |
| ELIMA 1 | KNKII/Karlsruhe | | X | X | | | | A, B | in pile |
| ELIMA 3 | KNKII/Karlsruhe | | | X | | | | A, B | rfi |
| COMPLIMENT | | | | | | | | | |
| DELICE 03 | OSIRIS/Saclay | X | X | X | | X | X | A,B,D | PIE |
| ELIMA 2 | HFR/Petten | X | X | X | | X | X | A,B,D | PIE |

*) Experimental goals

A) Evaluate the mechanical and chemical integrity of the ceramic material

B) Evaluate the tritium retention and release properties after irradiation

C) Evaluate the tritium retention and release properties during irradiation

D) Investigate if there is a different impact of neutron-dpas or t + α-dpas to the relevant properties

**) Status

compl.: irradiation and PIE completed

rfi: ready for irradiation

in pile: under irradiation

PIE: post irradiation examination started

### 4.6.2 Tritium release

Tritium inventory is an important aspect in blanket design. For economical and safety reasons it should be small, i.e. tritium release should be fast. This Section deals with tritium inventory in the breeder ceramics. For the total inventory see Section 2.3.

The main goal of the tritium release investigations is the determination of the tritium inventory in the blanket for representative conditions. In addition it is important to identify and model tritium release controlling processes.

Three different kinds of tritium release measurements are performed:

1. Tritium annealing of low-activity samples irradiated in closed capsules at room temperature
2. Tritium annealing of high-activity samples irradiated in closed capsules at elevated, blanket-relevant temperatures
3. Purged inpile tests under blanket-relevant conditions.

#### 4.6.2.1 Annealing tests

The "low"-activity (tritium activity $\leqq$ 1 MBq) samples are irradiated for about 20 h at room temperature in a field of moderated neutrons at the KfK cyclotron. The "high"-activity (tritium activity up to $10^4$ MBq) samples are normally irradiated in closed stainless steel capsules under He atmosphere and at elevated temperatures in the OSIRIS reactor at Saclay or the HFR reactor in Petten.

Several metasilicate and metazirconate samples and a large number of orthosilicate samples were annealed. Generally, tritium release from orthosilicate and metazirconate is observed to be much faster than from metasilicate.

#### 4.6.2.2 Purged inpile tests

Two series of tests are performed: the LISA experiments together with CEA at the SILOE reactor in Grenoble and the TRIDEX experiments together with KFA at the DIDO reactor in Jülich. The samples are contained in stainless steel capsules and surrouned by electrical heaters. Zn-reductors are used to convert any tritium water to gas. The facilities allow systematic variations of sample temperature (between about 300 and 700 °C), purge gas composition and flow rate. Moreover, in the LISA tests the tritium production rate can be varied by moving the inpile test section. Selected KfK samples are also being tested in the common European inpile tests EXOTIC-5 and -6 , performed in Petten. Before irradiation the samples are generally dried under vacuum or by He purging at temperatures between 500 and 800 °C for several tens of hours.

The most important parameter determined in the inpile tests is the tritium inventory as a function of sample temperature. Usually the inventory I is normalized by the production rate p to yield the residence time $\tau = I/p = I/r$ (at equilibrium p = r, r release rate). The concept of a residence time is only useful if $\tau$ is independent of p. This requires $r \sim I$, i.e. first order release processes. In this case blanket inventories can be predicted using (calculated) production rates.

Because of its importance, the question of the order of release reactions was studied in LISA-2 and in the recently performed test LILA/LISA-3 for ortho- and metasilicate, metazirconate and aluminate. The kinetics was observed to be independent of, and the inventories to be proportional to the production rate. This indicates, in accordance with the annealing results, that the controlling processes obey first order kinetics.

The inpile results concerning purge gas effects too are in agreement with annealing observations. Inpile tritium release was found to be independent of purge gas flow rate and pressure. It is enhanced by $H_2$ and $H_2O$ additions to the purge gas and strongly retarded by $O_2$. For orthosilicate the residence time was found to vary according to $\tau \sim 1/p_{H2}$ with the $H_2$ partial pressure. Because reasonable residence times are observed for the most promising ceramics with He + 0.1 vol. %

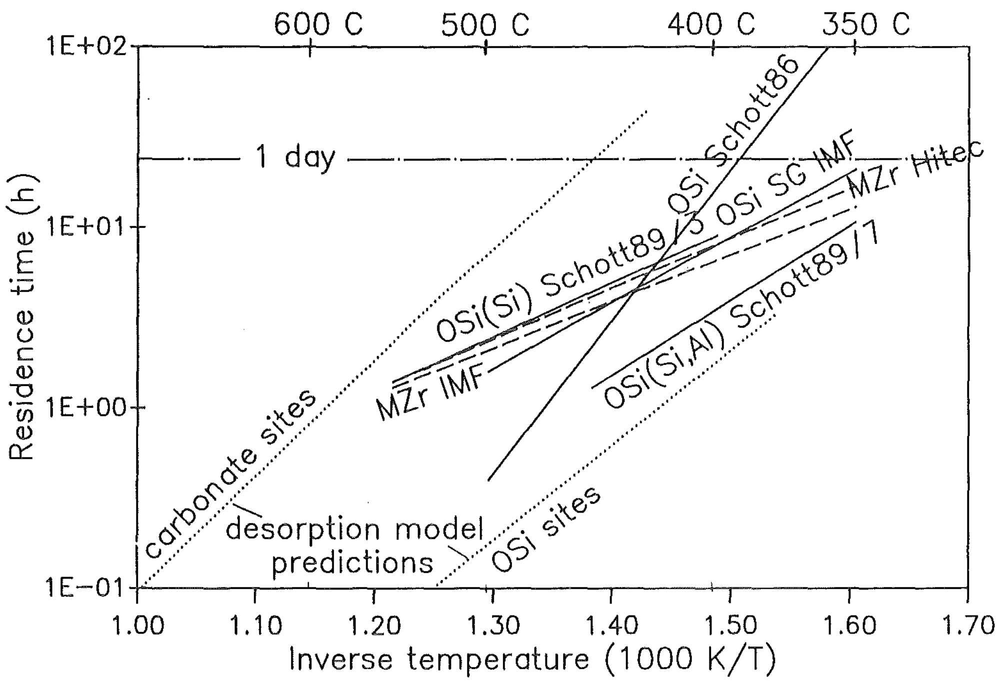



Fig. 26 Measured inpile residence times for recently produced metazirconate (MZr) and orthosilicate (OSi) and desorption model predictions for He + 0.1 % $H_2$ purge gas. SG: sinter granulate; Schott: spheres prepared from melt.

$H_2$, this mixture has become the reference purge gas in most studies.

Inpile residence times with $H_2 + 0.1$ vol% $H_2$ purge gas for recently produced orthosilicate and metazirconate sinter granulate and orthosilicate spheres prepared from the melt (Schott) are shwon in Fig. 26. At 400 °C ($\approx$ minimum blanket temperature) the residence times of all new charges are rather consistant with those of the older charge Schott 86 (if anything, they are smaller), hitherto considered as reference, and sufficiently small (< 1d) for blanket applications. The data for the $Li_4SiO_4$ pebble Schott 86 have been used to calculate the tritium inventory in the breeder ceramic of the Demo blanket.

### 4.6.3 Post irradiation examinations

Measurement and comparison of the properties of various breeding ceramics irradiated at different temperatures and neutron fluences are carried out in the Fusion Ceramics Laboratory (FKL). The FKL is located in the Hot Cell Facility of KfK. It consists of four lead shielded boxes and four glove boxes. An inert and dry atmosphere ($N_2$) is needed within the boxes to protect the ceramics from reactions with water vapor.

Heating of the ceramic samples according to a special program is used to evaluate the tritium release characteristics. Irradiation induced swelling and open porosity changes of the samples are measured by mercury intrusion porosimetry. Changes of the thermal diffusivity of $Li_4SiO_4$ pellets due to irradiation is measured by the laser flash apparatus. The thermal conductivity of beds of sintered and melt produced lithium orthosilicate pebbles will be measured by the radial heat flow method in a glovebox. Thermal cycling is applied to breeding ceramics to study thermomechanical properties.

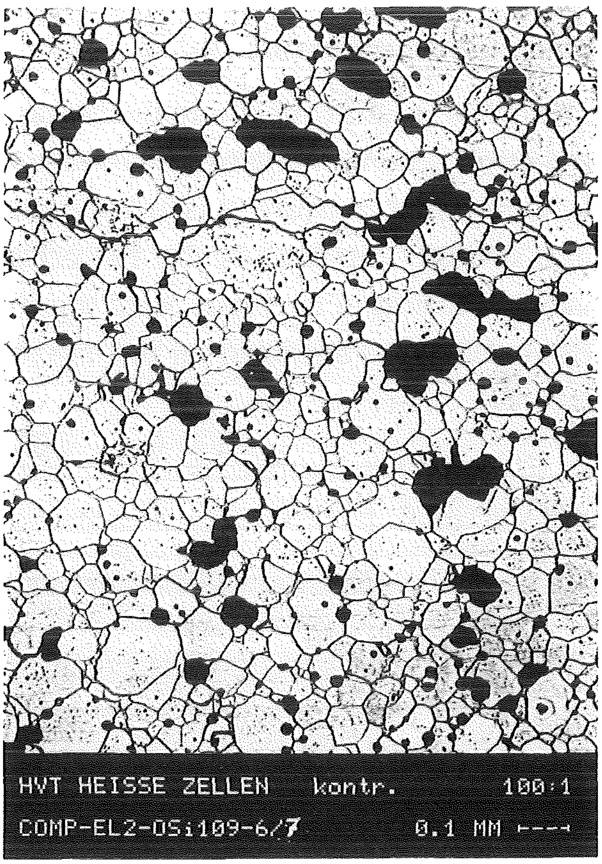


Fig. 27 Ceramographic section of $Li_4SiO_4$ sample P109 COMPLIMENT.

Selected samples are transferred to the shielded metallographic unit located in the same building for further characterization. Photographs of fractured surfaces are taken by a scanning electron microscope. The elemental composition in the range from carbon to heavier elements is determined. The abundance and size distribution of grains and closed pores are explored by image analysis (Fig. 27).

## 4.7 Beryllium

### 4.7.1 Beryllium properties

A recommendation of beryllium properties is difficult because the measured properties are unique to a particular specimen, batch to batch variations were observed and the properties depend on many parameters, including:

1. fabrication process,
2. grain size
3. impurities
4. direction of forces (longitudinal, transversal)
5. temperature
6. strain rate, load time
7. porosity
8. irradiation
9. test procedure

The dependence on these parameters often is not separable and much more data would be needed for a real assessment than are presently available. Moreover, important changes in manufacturing beryllium powder product have improved the mechanical properties significantly since 1970 and a number of products with different material properties are on the market. Materials favourable with respect to strength may not be so with respect to brittleness or irradiation resistance.

No material property measurements were made at KfK but the existing data were evaluated and best values were recommended for use in the EC-Demo reactor blanket studies. The recommended correlations for

- specific heat
- vapour pressure

- thermal conductivity
- electrical resistivity
- thermal expansion
- Young modulus
- Poisson ratio
- yield strength
- ultimate tensile strength
- compressive strength
- elongation
- fracture toughness
- impact strength

are listed in Ref. [7].

### 4.7.2 Compatibility with $Li_4SiO_4$ and structural materials

Out-of-pile tests performed at KfK have shwon that the maximum allowable temperature at the interface beryllium - $Li_4SiO_4$ is greater than 700 °C, while for the coupling Be-316L it is 600 °C. These results have been confirmed by tests performed at CEA, which show also that the compatibility with beryllium of the ferritic steel 1.4914, similar to Manet, is better than that of 316 L [7].

The first results from visual inspection of the Sibelius compatibility irradiation samples indicate that at 550 °C there was no reaction between $Li_4SiO_4$ and Be, and only a slight reaction at the interfaces Be-Manet and Be-316 L [7].

### 4.7.3 Irradiation effects

Most of the available data on the behaviour of beryllium under irradiation is for beryllium fabricated 25 to 35 years ago. This material is generally made of relatively coarse lamellar grains and contains high amounts of beryllium oxide. Because slip can occur only along basal planes, this kind of beryllium is very anisotropic and has low ductility. Recent developments in the production of beryllium powder (impact grinding, atomisation) and powder consolidation methods (cold isostatic pressing / vacuum sintering, hot isostatic pressing) produce better materials, more isotropic and with higher ductility. Irradiation results are already available which indicate that these materials behave better under irradiation [19].

Furthermore, all the irradiation data at high neutron fluences have been obtained by irradiations at low temperatures (< 100 °C) and subsequent out-of-pile annealing. These data may be quite different from the those obtained by irradiations at high temperatures and fluences. All this points out to the necessity of performing irradiations at high fluences and temperatures of beryllium samples produced with the new improved fabrication methods. Section 5.2.2 shows the proposed European beryllium irradiation program based on these requirements.

## 5. REQUIRED R. & D. PROGRAM PRIOR TO TESTS IN NET / ITER

The design and R. & D. work performed so far during the first three year period of the European Blanket Development Programme have shown that there is still a number of major technical issues to be solved. These may be divided in four parts:

a. Design problems

b. Material development and testing

c. Non-nuclear tests of blanket components and blanket sections

d. Nuclear tests of blanket-subsections.

The annual operation cost of ITER is presently estimated at about 300 M $/a. With the low availability expected, testing in NET / ITER will be very expensive and limited in scope. Moreover safety and reliability of NET should not, or only marginally be reduced by the incorporation of test blankets. Therefore an extensive blanket development and testing program has to be carried out prior to construction of the NET / ITER test modules.

Only the second three years period of this program, running until 1994, has been specified up to now, for the rest not more than general ideas exist. The later part of the program will strongly depend on the blanket concept which is selected for final development and testing.

### 5.1 Design Problems

Table 10.1 summarizes the work performed in 1989 - 91 and the proposed until 1994. This refers essentially to design modifications in view to improve the blanket reliability and, if necessary, to make the blanket more compliant with a large beryllium swelling. Detailed safety investigations have still to be performed for the Demo blanket and for the test modules. The development work for a computational system to calculate the Lorentz forces and stresses caused by disruptions in ferromagnetic structures (Manet) will continue until the end of 1994.

### 5.2 Material Development

The R & D relevant materials for the present solid breeder blanket are $Li_4SiO_4$, beryllium and Manet. With Manet development forming a separate part of the fusion R & D program, only the breeder material and beryllium will be discussed here.

In the area of material development the main issue will be the impact of long term irradiation effects on the material properties. Most of the material data measurements described earlier in the paper (Sections 4, see also Ref. [7]) have to be repeated with materials irradiated up to Demo relevant fluences.

### 5.2.1 Breeder material

The tritium release in-pile experiments at low burn-ups and the out-of-pile tests (mechanical, thermal cycle and heat transfer tests) performed so far show that the chosen reference solution of breeder material - $Li_4SiO_4$ pebbles in the diameter range 0.35 to 0.6 mm and fabricated by melting - does not present so far any feasibility problem. There is still work to be done on the effects of the presence of beryllium and Manet on the tritium release and some out-of-pile tests which will be discussed in the Section 5.4 (Non-Nuclear Tests), however the future work will be mainly concentrated on the effects of high burn-ups (Table 10.2). Of interest is the behaviour of the reference $Li_4SiO_4$ pebbles at high neutron fluences (10 at % total lithium burn-up, 22 dpa) in the temperature range 380 - 800 (900) °C and namely:

- mechanical stability
- tritium release and transport
- lithium transport
- thermal conductivity of the pebble bed
- compatibility with beryllium and with Manet.

As shown in Table 10.2 two end-of-life irradiations are proposed by ECN-Petten and by CEA in the HFR and in the Phenix reactor respectively. Within the European collaboration both will contain samples of the KfK reference $Li_4SiO_4$ pebbles. In both reactors it is possible to achieve the objective burn up of 10 at %, however in the case of HFR the achieved fluence will be lower than the aimed one (5.8 against 22 dpa), while in Phenix it will be higher (39 dpa). Considering the preparatory work requirements, the irradiation in Phenix cannot be started before January 1994, so that the results of the post-irradiation-examinations will be available only at the end of 1995.

Table 10.1 R & D program prior to tests in NET / ITER. Design

| Activity / Milestones | | 1989 | 1990 | 1991 | 1992 | 1993 | 1994 |
|---|---|---|---|---|---|---|---|
| 1. | **Design** | | | | | | |
| 1.1 | Demo blanket conceptual design | | | | | | |
| | - Basic design | | ▽ | ▼ | | | |
| | - Modification due to results of investigations | | | | | | ▽ |
| 1.2 | Test module and relative ancillary loop conceptual design | | | | | | |
| | - Basic design | | ▼ | | | | |
| | - Modifications due to results of investigations | | | | | | ▽ |
| 1.3 | DEMO blanket safety investigations | | | | ▽ | | |
| 1.4 | Design of He temp. and flow control for test module coolant loop | | | | | ▽ | |
| 1.5 | Safety investigations for test module and its ancillary loops | | | | | ▽ | |
| 1.6 | Definition of large scale tests and design of large helium loop for Non-Nuclear Tests | | | | | | ▽ |
| 1.7 | Development of a computational system to calculate Lorentz forces and stresses caused by disruptions: | | | | | | |
| | - For non-ferromagnetic structures (316LSS)-ELSA | | ▼ | | | | |
| | - For ferromagnetic structures (MANET)-CARMA | | | | | | ▽ |

Milestones: ▽ planned ▼ achieved

Table 10.2 R & D program prior to tests in NET / ITER. Breeder material

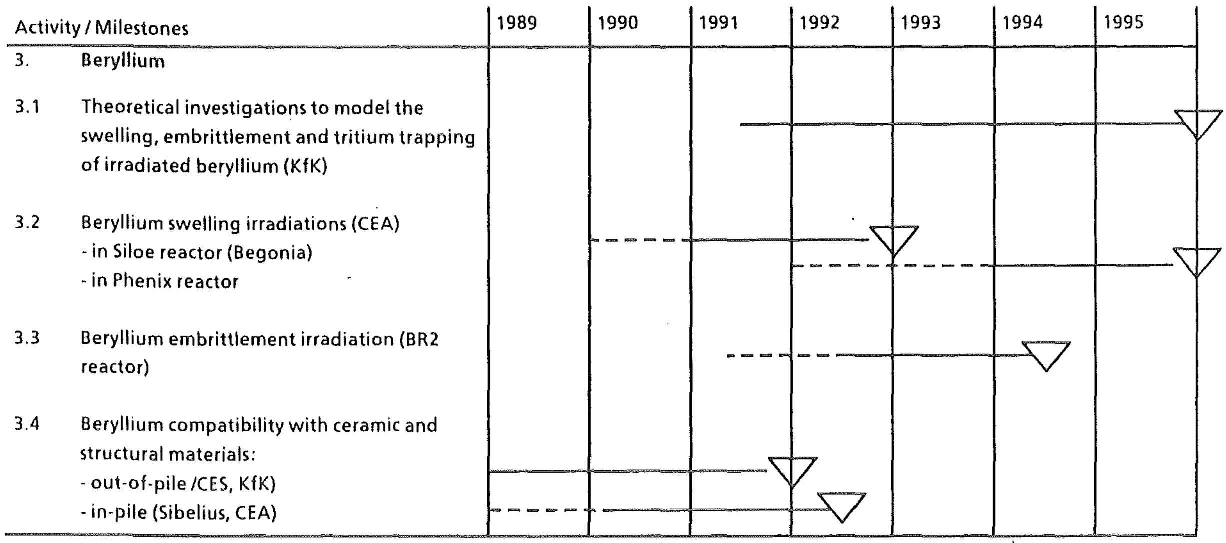
| Activity / Milestones | | 1989 | 1990 | 1991 | 1992 | 1993 | 1994 | 1995 |
|---|---|---|---|---|---|---|---|---|
| 2. | **Breeder Material** | | | | | | | |
| 2.1 | Choice of reference $Li_4SiO_4$ pebbles | | ▼ | | | | | |
| 2.2 | Reproducibility of tritium release results for reference $Li_4SiO_4$ pebbles | | ▼ | | | | | |
| 2.3 | Further improvements of the $Li_4SiO_4$ pebbles | | | | | | | ▽ |
| 2.4 | Effects of beryllium and Manet presence on tritium release | | | | ▽ | | | |
| 2.5 | Determination of released HTO / HT ratio | | | | ▽ | | | |
| 2.6 | Effects of high burn-up: | | | | | | | |
| | - EXOTIC 7 irradiation (ECN, Petten) | | | | | | ▽ | |
| | - Phenix irradiation (CEA) | | | | | | | ▽ |

Milestones:

▽ planned

▼ achieved

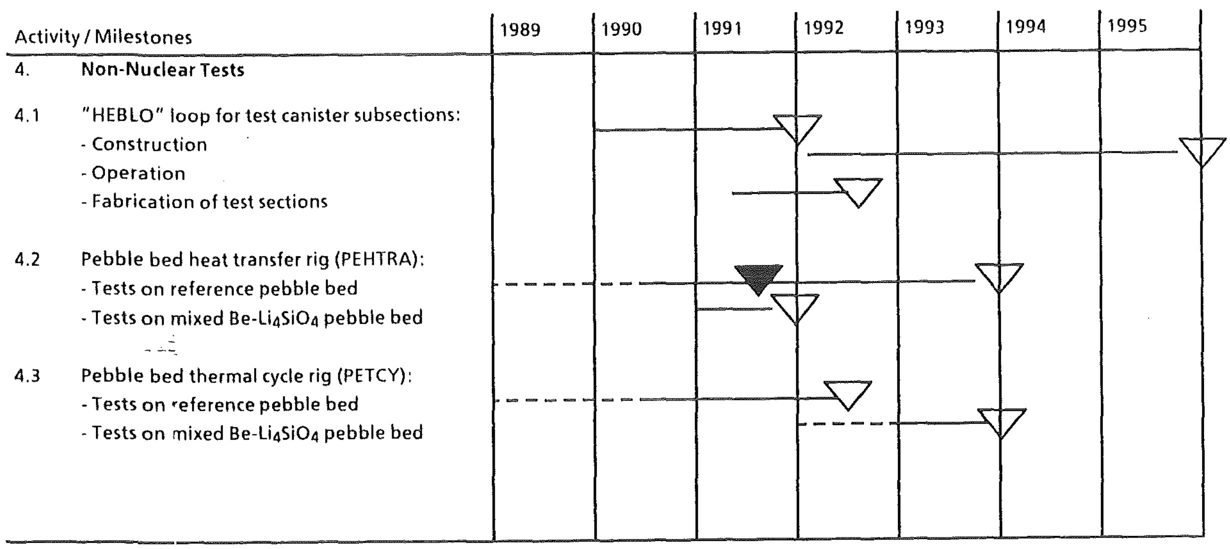
Table 10.3 R & D program prior to tests in NET / ITER. Beryllium




Milestones:

planned achieved

Table 10.4 R & D program prior to tests in NET / ITER. Non nuclear tests.




Milestones:

planned

achieved

### 5.2.2 Beryllium

The design work for the Demo blanket has shown which are the problems related to the use of beryllium in the blanket. These are:

- swelling under irradiation
- embrittlement caused by irradiation
- tritium retention
- compatibility with Manet and $Li_4SiO_4$ pebbles
- behaviour of the interface between beryllium and cooling tubes under thermal cycling

As discussed in Section 4.7.3 the irradiation effects, especially on swelling, embrittlement and tritium retention, should be investigated for beryllium fabricated with the modern, improved methods. These irradiations should be performed up to high fluences for the relevant high temperatures.

Table 10.3 shows the European beryllium program. The irradiation in the Phenix reactor covers the neutron effects expected in Demo, but not the complete range of temperatures (Table 11). This should be covered by the irradiation in the Siloe reactor, but at much lower neutron fluences. The proposed beryllium embrittlement irradiation covers the temperature range, but at much lower fluences than in the Demo blanket (Tables 10.3 and 11). It is known that embrittlement occurs already at much lower fluences than swelling. These irradiations will also provide relevant information on the tritium retention in beryllium.

**Table 11: Beryllium irradiations: temperatures and fluences**

| | Temp range °C | Peak fluence (E > 1 MeV) | Peak fluence (dpa) | He content (appm) |
|---|---|---|---|---|
| Demo blanket | 250 - 600 | $2.2 \times 10^{22}$ | 30 - 60 | 15000 |
| Be-swelling irr.: | | | | |
| - Siloe | 250 - 700 | $\approx 2.2 \times 10^{21}$ | 3 | $\approx$ 800 |
| - Phenix | 400 - 550 | $\approx 4 \times 10^{22}$ | 60 | $\approx$ 15700 |
| Be-embrittlement irr.: (BR2) | 200 - 600 | $1.5 \times 10^{21}$ | - | $\approx$ 600 |
| Be-compatibility | 450 - 550 | $0.55 \times 10^{21}$ | 0.7 | $\approx$ 200 |

### 5.3 Non-nuclear Tests

In a fusion reactor the heat sources result from nuclear reactions and occur within the materials. This cannot be realistically simulated in non-nuclear tests. Moreover no tritium production and radiation damage effects occur in such tests. Nevertheless quite a number of important issues can be addressed in non-nuclear tests.

An attractive possibility for helium cooled solid breeder blankets consists in constructing a realistic mock-up of a blanket or blanket subunit, including the breeder material, and feeding the helium coolant system periodically with cold (≈ 260 °C) or hot (≈ 460 °C) helium. In this way the total structure will periodically be heated up and cooled down thereby simulating the thermal cycling effects of a Tokamak reactor. A possibility to technically realize the temperature changes is described in Section 4.5.1 and more extensively in Ref. [7] for the HEBLO-loop. The KfK design of a helium cooled ceramic breeder blanket has a large degree of modularization: Each segment is composed of 28 toroidally arranged canisters, each canister is subdivided by stiffening plates into subsections, each subsection contains a periodic structure of beryllium plates and slits filled with breeder material pebbles. Therefore testing of a subunit often is already characteristic for the whole system.

Issues addressed by the subsection tests include:

- investigation of the wall coolant connection
- effective heat conductivity of the pebble bed
- local heat transfer coefficients and hot spot factors
- purge flow distribution and pressue drop
- local mass transfer (local substitution of pebbles by sublimating simulating material)
- hydrogen permeation into coolant
- effect of purge flow on lithium transport.

The full canister tests partly address the same issues but in addition they provide information on:

- blanket pressure drop
- flow distribution in the canister
- breeder / structure / multiplier thermal expansion interaction

- determination of fundamental frequency
- response to LOCA or other coolant transients
- response to structure failure, such as for instance, the break of an inner pressure tube.

With the HEBLO-loop canister subsection tests will be performed in the period 1992 - 1995. For the full size canister tests an existing helium loop will be used but details are not yet specified and the tests fall into the period after final solid breeder blanket concept selection.

Table 10.4 summarizes the non-nuclear tests performed and proposed in the period 1989 - 1995. Beside the HEBLO tests, also tests to measure the thermal conductivity and the wall heat transfer coefficient of a pebble bed in the test apparatus PEHTRA and thermal cycle tests of pebble beds in PETCY are being performed [7].

Parallel to the thermal cycling tests fabricability tests of various components are being conducted [7].

The helium coolant system is similar to what has been used for fission reactors and does not need special R & D work except for the thermal cycling operation which has to be tested.

## 5.4 Nuclear Tests

In principle it would be desirable to perform the tests mentioned under point 5.3 with nuclear heating also, so that a prototypic heat source distribution is obtained and the behaviour of tritium could be studied. However, inspection of the possibilities at existing research reactors revealed that this is not feasible.

In order to get prototypic neutron flux values one has to go right into the center of a reactor. Here the space is very limited, however. Thus, only a small subunit of a blanket structure can be used to study local geometry effects. In view of the high degree of modularization and periodicity of the structure such small size experiments are still quite valuable.

The minimum dimensions to represent the KfK solid breeder blanket configuration are: one 6 mm thick slab filled with breeder material pebbles, sandwiched between two beryllium plates which contain helium coolant tubes. The dimensions of the breeder material slab should be large compared to its thickness what means about 60 mm. A purge system with supply tubes at the ends of the slab should be provided.

The neutron flux, flux gradient, and neutron spectrum should be matched as closely as possible to the fusion reactor conditions, also the heat source should be simulated.

Practically only the thermal research reactors of the European Community are available for this type of experiments. But then the thermal neutron flux impinging on the test assembly has to be suppressed, otherwise the tritium production rate and the heat generation from the $^{6}Li(n, \alpha)T$-process would be by far too high when compared with other quantities. Therefore the test assembly has to be surrounded by cadmium.

A research reactor in which a large central loop and a test position covered by cadmium already exists, is the BR2. Therefore this reactor was taken for a first feasibility assessment. Only some neutronics calculations have been performed, a detailed engineering design does not yet exist.

In spite of the uncertainties and differences to ITER, it can be concluded that the conditions are quite acceptable to get important informations on the nuclear behaviour of the blanket. Issues which could be addressed by such tests include:

- tritium release and transport by the purge flow system,
- tritium permeation to the coolant,
- lithium transport,
- beryllium / ceramic / steel compatibility under irradiation.

Similar or slightly degraded test conditions can be obtained in the core center of OSIRIS or HFR, provided that a cadmium covered loop can be tolerated there. With the expectation that no fundamental non-feasibility problem will be revealed by the nuclear tests they are postponed - mainly for economic reasons - to well behind the year 1994.

## 5.5 Conclusions

The essential tests to judge the feasibility of the Demo blanket, namely high burn-up irradiations of breeder material and beryllium and non-nuclear tests on blanket subsections, will be terminated in 1995. Only at this stage it will be possible to make a founded choice between the two European solid breeder blankets. This date was already foreseen in the program proposition to the Fusion Technology Steering Committee (FTSC) in 1989. After 1995 more expensive tests will be necessary, namely out-of-pile canister or segment tests in already available large helium loops, and nuclear tests of blanket subunits in fission reactors. This would allow sufficient time for the design and construction of the test modules to be irradiated in NET/ITER in the foreseen time (2005 - 2010).

# 6. TEST PROGRAM IN NET / ITER

## 6.1 Introduction

NET/ITER offers the unique possibility to test simultaneously all aspects of a DEMO relevant blanket concept in the real geometrical configuration, with the real magnetic field, and with an incident neutron flux having the real neutron spectrum and spatial distribution. The main differences to DEMO are the lower wall load (1.0 $MW/m^2$ instead of 2.2 $MW/m^2$), and the shorter burn time, or lower fluence.

In spite of the lower power density the average temperature in the blanket can be adjusted to DEMO values by flow reduction. However, the local temperature distribution and temperature gradients in the materials will be different. Anyway, the most important use of NET/ITER is the comparison of the actual test object performance in NET/ITER with calculated predictions and to make code validations. Here testing in NET/ITER is only the last step of a design validation process which includes the R. and D. program described in Chapters 4 and 5.

Testing in NET/ITER will be extremly expensive and, because of the limited amount of tritium available for NET/ITER operation, also will be strongly limited in scope. Therefore it was tried to specify a test program which is sufficiently detailed to confirm the viability of the blanket concept for Demo and allows some blanket variations and modifications to be tested, but on the other hand keeps the NET/ITER burn time requirements sufficiently low.

In order to arrive at such a compromise the following procedure was adopted: First the critical issues of the blanket concept were identified. Then for the issues relevant to testing in NET/ITER the time constants to reach equilibrium conditions in NET/ITER-tests were estimated. From these the duration of a single test and the required mode of NET/ITER operation were deduced. Finally a test program was formulated and the total burn time to conduct the program was estimated.

## 6.2 Critical Issues of the B.O.T. Helium Cooled Solid Breeder Blanket

The subject of critical issues was discussed in the Finesse Study and in several ITER workshops. This will not be repeated here, only a list is given which may contain some arbitrary distinctions. Under critical issues we understand both, those which have an impact on the concept feasibility and those which refer to details of a particular design. For the KfK solid breeder blanket concept (see Chapter 5)these include:

1. Beryllium swelling and embrittlement
2. Response of the structure to plasma disruptions
3. Breeder material behaviour at high burn-up (mechanical integrity, thermal conductivity, tritium release)
4. Tritium inventory, tritium permeation and losses to the steam circuit,
5. Behaviour of the mechanical structure with welds and brazed connections under cyclic loads and with a volumetric heat source

5a. The same as 5. but with irradiated material

6. Temperature and flow distribution
7. Tritium self-sufficiency
8. Failure modes and reliability
9. Behaviour under abnormal coolant conditions

Points 1 and 3 do not only refer to material property data but also to the behaviour of the material in the blanket configuration. Thus, all the points are subject to testing in NET/ITER. Points 1, 3, 5a and 8 cannot be adequately covered in NET because of too low fluence and need extrapolation. Only a limited amount of abnormal operation conditions testing will be possible in NET/ITER. Post irradiation testing up to destruction may be needed in some cases.

### 6.3 Requirements Posed on NET / ITER Operation by the Testing of the Blanket Module

Important parameters which determine the minimum duration time of a test are the time constants to reach equilibrium conditions in the test module for temperature and tritium distribution after a step in power production.

An assessment of these time constants was made in the framework of two ITER workshops, and a general agreement was achieved on the values. They are based on a neutron wall load at the test blanket position of 1.2 $MW/m^2$ what corresponds to an average neutron wall load of 0.8 $MW/m^2$. In particular, the design of the test module for the KfK solid breeder blanket (see Chapter 3) has allowed to estimate the time constants of the various components and thus assess the requirements given by the testing of the test modules in the NET / ITER machine. Table 12 shows these requirements.

**Table 12** **Minimum requirements posed on NET / ITER operation by the testing of the BOT-HCSBB module**

---

<u>Time to reach:</u>

| | |
|---|---|
| - steady state temp. in the breeder * | 100 s in blanket front<br>300 s in blanket back |
| - fraction of steady state tritium inventory (tritum release) in the breeder* | 50 % with 1000 s burn time shot length<br>67 % with 3000 s burn time shot length |
| - equilibrium tritium permeation to the coolant | ≈ 1 day |
| - steady state in purge flow and tritium extraction system | ≈ 6 hours + test. |

<u>Duration of continuous operation</u>

- during early technology phase*: 10 to 20 hours (duty cycle ≧ 50 %)
- at the end of the technology phase: 5 days (duty cycle ≧ 70 %)

<u>Recommended cycle parameters during continuous operation*:</u>

| | |
|---|---|
| - burn time shot length | ≥ 1000 s |
| - duty cycle ** | ≥ 50 % |
| - off burn time | ≤ 1000 s |

---

* Coolant flow and inlet coolant temperature control during off-burn-time

** Duty cycle = $(\text{burn time} / (\text{burn time} + \text{off-burn time}))_{\text{average}}$

## 6.4 A NET Testing Program Scenario

A test programme for solid breeder blanket tests in NET was elaborated and with the test duration estimates mentioned above the burn time requirement for NET was calculated. The data are summarized in Table 13 and the following text explains essentially this table.

Blanket testing in NET does not mean to put a fully developed blanket module in a port and to study its behaviour. On the contrary, the test programme is the most important part of the blanket development process which serves to decide between design options, although only one solid breeder blanket concept will be tested.

The programme is subdivided into three steps of increasing risk and DEMO relevance which are implemented in sequence and in the following are called generations:

A Test module with austenitic steel behind the NET first wall

B Test module with austenitic steel with its own first wall

C Test module with the martensitic steel Manet with its own first wall

In each generation functional tests of one day duration are foreseen which are followed by real measurements. Depending on the test objective these tests may last 3 days or 5 days. It is assumed that a few modifications will be tested to select the best one. Depending on the test objective, modules with different instrumentation and possibly also different design details will be required. Moreover the outcome of the test will lead to design changes so that some iterations are needed. By multiplication of the number of modifications, the number of test objectives and the number of design iterations the number of tests per type are deduced.

The realization of the tests is done in so-called test campaignes where - because of the two ports available for the solid breeder blanket concept - two tests are being conducted in parallel. The required burn time is deduced from the number and duration of the test campaignes.

From all the module tests the best design solution will be selected and a complete NET segment of this type will be constructed. With this segment longer duration tests will be made to accumulate experience, to have some failure statistics, and to study irradiation effects. The segment will be made of Manet with its own first wall as a box design similar to what is foreseen for DEMO. The segment should re-

Table 13 Solid Breeder Blanket Testing Scenario

| generation* | A | | | B | | | C | | | port tests total | segment |
|---|---|---|---|---|---|---|---|---|---|---|---|
| test duration | 1d | 3d | 5d | 1d | 3d | 5d | 1d | 3d | 5d | | 3d |
| burn time | 20h | 50h | 80h | 20h | 50h | 80h | 20h | 50h | 80h | | 50h |
| modifications | 2 | 2 | 2 | 2 | 2 | 2 | 4 | 2 | 2 | | |
| objectives | 1-2 | 1-2 | 1 | 1-2 | 2 | 1 | 1-2 | 2 | 1 | | |
| iterations | 2 | 2 | 1 | 2 | 1 | 1 | 2 | 2 | 1 | | |
| tests | 4-8 | 4-8 | 2 | 4-8 | 4 | 2 | 8-16 | 8 | 2 | 48 | 20 |
| campaignes | 2-4 | 2-4 | 1 | 2-4 | 2 | 1 | 4-8 | 4 | 1 | 24 | 20 |
| integral burn time | 60 | 150 | 80<br>290 | 60 | 100 | 80<br>240 | 120 | 200<br>400 | 80 | 930 | 1000 |

*A = test module with austenitic steel behind NET first wall

B = test module with austenitic steel with own first wall

C = test module with martensitic steel with own first wall main in place as long as possible and is subject to repeated careful measurements. Here 20 measurement campaignes of three days duration each are foreseen what leads to a test burn time of about 1000 h.

If only one port is available for the solid breeder blanket concept (ITER case) the number of tests is equal to the number of test campaignes and the integral burn time for the port test would be doubled. The number of positions in which test segments can be inserted is not so much limited. Therefore more positions can be taken when more concepts have to be investigated and the segment test time will remain unchanged.

With this overall frame given, a more detailed test program could be formulated. However, in view of the lead time of more than 15 years and the many open questions - such as NET driver blanket and total burn time available - this was considered as premature. The general tendency of the test program should be to select as fast as is reasonably achievable the final blanket solution and to leave a segment of the selected blanket in NET / ITER as long as possible in order to gain operational experience and to accumulate radiation dose.

An irradiation of 7000 h instead of 1000 h would make the test much more Demo relevant. Demo mid-life conditions could be achieved with respect to embrittlement and stress relaxation, however the regime of onset of swelling will not be reached.

## REFERENCES


1. M. DALLE DONNE, S. DORNER, and S. TACZANOWSKY, " Conceptual Design of Two Helium Cooled Fusion Blankets (Ceramic and Liquid Breeder) for INTOR", KfK - 3584, EUR-7978e, Kernforschungszentrum Karlsruhe (1983).

2. M. DALLE DONNE, U. FISCHER, and M. KÜCHLE, "A Helium-Cooled, Poloidal Blanket with Ceramic Breeder and Beryllium Multiplier for the Next European Torus", Nucl. Technol., 71, 15 (1985).

3. M. DALLE DONNE, U. FISCHER, M. KÜCHLE, G. SCHUMACHER, G. SORDON, E. BOJARSKY, P. NORAJITRA, H. REISER, H.D. BASCHEK, E .BOGUSCH, "Pebble-bed canister: The Karlsruhe ceramic breeder blanket design for the Next European Torus", Fusion Technol., 14, 1357 (1988).

4. G.W. HOLLENBERG and D.L. BALDWIN, "The Effect of Irradiation on Four Solid Breeder Materials", J. Nucl. Mater., 133 & 134, 242 (1985).

5. H. WERLE, J.J. ABASSIN, M. BRIEC, R.G. CLEMMER, H. ELBEL, H.E. HÄFNER, M. MASSON, P. SCIERS and H. WEDEMEYER, "The LISA-1 Experiment: In-Situ Tritium Release Investigations", J. Nucl. Mater. 141-143 (1986) 321.

6. G.W. HOLLENBERG, "Swelling of Lithium Ceramics During Irradiation", presented at American Ceramic Society Mtg., Pittsburgh, Pennsylvania, May 1987.

7. M. DALLE DONNE (ed) et al., "Status Report. KfK Contribution to the Development of DEMO-relevant Test Blankets for NET/ITER. Part 2: BOT Helium Cooled Solid Breeder Blanket. Volume 2: Detailed Version", KfK 4929 (1991).

8. U. FISCHER, "Impact of Ports on the Breeding Performance of Liquid Metal and Solid Breeder Blankets for the Demo-NET Configuration", presented at Second Int. Symp. on Fusion Nuclear Technology, Karlsruhe, Germany, June 2 - 7, 1991.

9. T. KURODA, K. SATO, T. SUZUKI, T. TONE and M. SEKI, "Technical Considerations on the Tritium Breeding Blankets for a Fusion Power Reactor", Fusion Engineering and Design 8, (1989) 219 - 226.

10. M. DALLE DONNE, S. DORNER, "Tritium Control in a Helium Cooled Ceramic Blanket for a Fusion Reactor", Fusion Technol. 9, (1986) 484 - 491.

11. K. FORCEY, D. ROSS, I. SIMPSON, D. EVANS, "Hydrogen Transport and Solubility in 316 L and 1.4914 Steels for Fusion Reactor Applications", J. Nucl. Mat. 160, (1988) 117 - 124.

12. D. BALDWIN, D. GELLES, O. SLAGLE, "Tritium Release from Irradiated Beryllium at Elevated Temperatures", 4th. Int. Conf. Fusion Reactor Materials, Kyoto, Dec. 5 - 8, 1989, J. Nucl. Mat. 179 - 181 (1991).

13. E. PROUST, CEA, April 1990, unpublished..

14. E. WEHNER, R. KRIEG and B. DOLENSKY, "Canister Type Blanket Elements for NET Under Accident Overpressure. Theoretical and Experimental Investigations", Fusion Eng. Design, 6, 69 (1988).

15. P. RAU, Siemens Kraftwerk Union, Private communication, 1990.

16. E. FRANCILLON, Framatome, Private communication, 1991.

17. H. EBI, W. EYRICH, H. FRIES, H. GIESE, K. HAYASHI, F. KAPPLER, U. von MÖLLENDORFF and T. TSUKIYAMA, "Measurement of Neutron Transport and Multiplication in Beryllium", Proc. 2nd Int. Symp. Fusion Nuclear Technology, Karlsruhe, 3 - 7 June 1991, to appear in Fusion Eng. and Design.

18. A. SCHWENK-FERRERO: "Verfahren zur numerischen Lösung der Neutronentransportgleichung mit strenger Behandlung der anisotropen Streuung", Kernforschungszentrum Karlsruhe, KfK-4788 (September 1990).

19. J.M. BEESTON et al., "Comparison of Compression Properties and Swelling of Beryllium Irradiated at Various Temperatures", J. Nucl. Mat. 122 & 123 , 802 (1984).