

A HELIUM-COOLED SOLID BREEDER CONCEPT FOR THE TRITIUM-PRODUCING BLANKET OF THE INTERNATIONAL THERMONUCLEAR EXPERIMENTAL REACTOR

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The usefulness of the tritium-producing blanket in the International Thermonuclear Experimental Reactor (ITER) to the fusion research and development program can be maximized by selecting design parameters, features, and options that are reactor relevant without significantly increasing the risk in key areas such as device safety and operational reliability. For that reason, a helium-cooled solid breeder (SB) blanket is proposed since it combines the operation of the SB at high reactor-relevant temperatures with the operation of helium at moderate temperature and pressure to minimize risk. Results of the analysis done for this

blanket concept indicate that it is very attractive. It can achieve a high tritium breeding ratio without breeding in the space-limited inboard region. It offers important safety features, including the use of inert gas with no chemical reaction or corrosion, low activation SB, and multiple containment of tritium. The concept provides great operational flexibility to accommodate changes in ITER operating parameters, such as power level, and to optimize the operating temperature of the structure. A novel and practical concept is proposed for the thermal resistance gap between the coolant and SB to allow their operating temperatures to be optimized.

I. INTRODUCTION

Most major fusion program plans in the world call for a fusion engineering facility. Several designs for such a facility, e.g., the Next European Torus,¹ the Fusion Experimental Reactor,² the Test Fusion Reactor³ (OTR), the Tokamak Ignition/Burn Experimental Reactor⁴ (TIBER), and the International Tokamak Reactor⁵ were previously developed. Recently, it has been proposed that a facility of this type can be designed and possibly constructed through international cooperation. The design for the International Thermonuclear Experimental Reactor⁶ (ITER) is in a preliminary stage.

The ITER is expected to cost several billion dollars

and to be the primary fusion engineering facility during the first two decades of the next century. Given the mission, size of investment, and the time frame in which ITER will operate, it is expected that ITER will be a critical step in fusion development. The considerable experience gained from previous design studies will undoubtedly be utilized to make ITER more cost-effective and to maximize its usefulness to fusion development.

This usefulness can be maximized by selecting design parameters, features, and options that are reactor relevant without significantly increasing the risk in key areas such as device safety and operational reliability. A key component for which the questions of reactor relevance and risk must be carefully investigated is the

tritium-producing blanket. The ITER is expected to operate with a total fusion power of several hundred megawatts, an availability of up to ~25%, and a fluence of several megawatt-years per square metre. The tritium consumption over the lifetime of the ITER can be as large as 100 kg. Only a portion of this can be available from Canadian heavy water reactors and at a cost of ~\$10 million/kg (Ref. 7). Therefore, a blanket is required to supply ITER with its own tritium.

The tritium-producing blanket for ITER will occupy up to 90% of the space surrounding the first wall. This premium space in the only fusion engineering facility during the early part of the twenty-first century should be effectively utilized. If the blanket is reactor relevant, then the data obtained from its operation can be extrapolated to the demonstration device beyond ITER.

On the other hand, it must be recognized that the ITER blanket will have to be designed, constructed, and operated based only on information from testing in nonfusion facilities, without prior fusion testing. Therefore, high performance and availability cannot be guaranteed.

A good strategy for the ITER blanket that carefully balances the above considerations is

1. design the ITER blanket with reactor-relevant materials and configuration
2. operate parts of the blanket at reactor-relevant conditions where the information is most valuable and the penalty is minimum
3. operate the rest of the blanket at the most reliable conditions
4. provide flexibility in the blanket design to accommodate a variety of ITER operating conditions.

In this paper, we present a blanket concept that appears to be most compatible with the above strategy. The concept is based on a helium-cooled solid breeder (SB) blanket design. The SB is operated at temperatures suitable for *in situ* tritium recovery. The breeder material, configuration, and operating conditions are all reactor relevant. Helium is operated at modest pressure and relatively low temperature. A practical configuration for the thermal resistance required to decouple the coolant and breeder temperatures is proposed. It was found that this blanket concept meets all absolute ITER requirements, is most compatible with the ITER mission, and offers many attractive features.

II. KEY CONSIDERATIONS IN SELECTION

There are many important considerations in selecting an ITER blanket. Table I lists five key categories for the most important factors in such a selection:

TABLE I

Key Considerations in Selecting ITER Blanket

Engineering feasibility
Tritium breeding
Design margin
Flexibility, including $\pm 50\%$ power variation
Engineering complexity
Maintenance and repair
Safety and environment
Routine tritium release
Accidental release (tritium activation products)
Decay heat
Waste disposal
Chemical and thermal reaction potential
Fault tolerance
Economics
Change in reactor capital cost
Change in reactor operating cost
Change in cost of test program
R&D requirements
Cost of R&D required prior to ITER (This should be only the incremental increase required beyond existing blanket R&D programs.)
Cost of R&D required after ITER (for DEMO)
Benefits (value of information obtained from operation in ITER)
Impact on the blanket R&D after ITER
Reduced risk for DEMO

1. *Engineering feasibility.* This category includes absolute requirements, such as tritium breeding and maintainability. It also includes factors that provide a measure of the degree of confidence in the engineering feasibility, e.g., design margin and capabilities to accommodate power variation.

2. *Safety and environment.* This is a particularly important category for the first fusion engineering facility. It includes routine and accidental tritium and activation product release, chemical and thermal reaction potential, waste disposal, and a measure of fault tolerance.

3. *Economics.* The impact on ITER cost is a key consideration. The cost of ITER includes capital and operating costs and the cost of the test program. The selection of a basic blanket that is a primary candidate for commercial reactors will substantially reduce the cost of the test program for this particular class of concepts.

4. *Research and development (R&D) requirements.* Any blanket concept for ITER will require R&D because of the lack of prior experience and experimental data in a fusion-relevant environment. However, there are existing R&D programs now in many parts

of the world that address the most attractive blanket candidates for commercial reactors,⁸ e.g., liquid-metal and SB blankets. If one of these concepts is selected for ITER, the cost of R&D in this category should include only the incremental cost beyond existing programs. Since ITER is only a step in fusion development, the cost of R&D beyond ITER, e.g., to construct the demonstration reactor (DEMO), is also a consideration. Clearly, such cost will be impacted by prior experience and data from ITER.

5. Benefits. This category relates to the value of information obtained from the operation of the ITER blanket. The ITER blanket will occupy up to 90% of the premium volume in the unique fusion environment surrounding the plasma. The benefits of operating the ITER blanket will clearly depend on the type of blanket. The benefit category should include two sub-categories:

- a. a measure of the attractiveness of the ITER blanket for the DEMO and beyond, as measured by present worldwide interest⁸
- b. a measure of the completeness of the data obtained from operation in regard to reactor relevance, e.g., three levels can be identified:
 - i. relevant materials
 - ii. relevant design configuration
 - iii. relevant performance parameters.

The present blanket concepts that are of great interest to the world program have been summarized in Ref. 8. These concepts can be classified into liquid metals and SB blankets. Liquid metals are attractive, but their application to ITER appears to be limited by safety concerns and the need for relatively high temperature operation. On the other hand, SBs can be cooled with helium or water, with the well-contained breeder operating at high temperature based on tritium release and reactor-relevance considerations while the coolant is operated at low temperature for safety reasons. Helium offers advantages of safety and operation flexibility when compared to water and is the preferred coolant for this study.

The attractiveness of a helium-cooled SB blanket for ITER rests on several factors, the most important of which are the following:

1. Such blanket configurations have been studied in detail in the past for commercial reactor applications, e.g., Blanket Comparison and Selection Study,⁹ FINESSE (Ref. 7), ESPRESSO (Ref. 10), and TOKOPS (Ref. 11). This leaves a significant design margin since the wall loading (1 to 2 MW/m²) and fluence (3 MW·yr/m²) of ITER are substantially lower than the wall loading (up to 5 MW/m²) and fluence (up to 20 MW·yr/m²) of these commercial reactor designs and only part of the allowable SB temperature window

(maximum and minimum allowable temperatures limited by sintering and tritium release, respectively) needs to be used.

2. There is an expanding data base available from past and ongoing SB experiments carried out internationally. It is expected that data will be available from these experiments in fission reactors for ITER-relevant levels of fluence.

3. The SB can be designed to operate at high temperatures, providing valuable information applicable to reactor conditions, while the helium is kept at a moderate temperature and pressure for safety and reliability. In addition, since helium is a gas, the pressure increase for any temperature increase is minor compared to a saturated liquid. Thus, the helium can be run so as to optimize the structure temperature. Presently available data indicate that the optimum temperature for steel (both austenitic and martensitic) may be as high as 350°C. The use of water at such high temperatures requires high pressure, while helium can be used at low to moderate pressures. Furthermore, the helium flow rate and/or inlet temperature can be adjusted to change the helium and structure temperature without incurring a significant pressure drop penalty.

4. A key to the success of testing facilities such as ITER is maintaining flexibility in the machine operational capabilities. For example, the blanket should be able to accommodate variations in the steady-state neutron wall load and power level (e.g., 50%). The helium-cooled SB design can accommodate such changes because of

- a. large margin in the operating temperature range for SBs
- b. possibility of adjusting the flow rate and/or inlet temperature of helium.

5. Safety is expected to be a key factor for ITER, the first experimental fusion reactor. In this respect, the use of helium, an inert gas precluding any chemical reaction and corrosion, in combination with a low-activation SB, such as Li₄SiO₄ and Li₂O, is highly appealing. Furthermore, such a blanket can be designed to reduce tritium permeation to insignificant levels.

Section IV describes a helium-cooled SB design that makes maximum use of the attractive features described above and appears to offer an excellent potential for satisfying the selection criteria listed in Table I.

Some of the key technical areas addressed in Sec. IV for the helium-cooled SB design include the achievable tritium breeding ratio (TBR), the coolant pressure requirement, the provision for producing the required temperature drop between the high-temperature

breeder and the lower temperature coolant and the helium containment and manifolding. A key question for the ITER blanket is the magnitude of the required TBR. Since ITER is an experimental facility, the breeding requirements are different from those for commercial reactors. The ITER breeding requirements are discussed in Sec. III. Section V summarizes the technical issues and discusses the technical basis for a novel concept to provide a predictable thermal resistance between the SB and coolant.

III. REQUIRED TBR

The amount of tritium that will be available from nonfusion nonmilitary facilities around the year 2000 is ~ 25 kg, mainly from the Canadian fission reactors.⁷ However, a typical experimental fusion reactor with a 600-MW fusion power output, such as ITER, which is expected to operate around this time period, will need ~ 35 kg of tritium to operate for 1 full-power year (FPY). If the total time length of operation is equivalent to 2.5 FPY, the reactor requires 87.5 kg of tritium over the lifetime of the machine (~ 10 yr at 25% availability). This amount is well beyond the current capability of tritium supply. Therefore, it is necessary to have tritium breeding capability in the next experimental reactor to minimize the needed external supply of tritium to operate the machine. The question, then, is how much tritium must be generated in the reactor and the amount to be purchased from non-fusion sources. This question is addressed in detail below.

A model for the tritium fuel cycle in a reactor was generated¹² and applied based on principles developed in Ref. 13. The model accounts for the effect of several parameters on the required TBR. These include tritium residence times in various components of the cycle, tritium release rate, tritium burnup fraction in the plasma, the plasma burn time and downtime, and system availability. There are essentially two distinctive modes of operation. The first mode, the super-breeding mode, ensures that tritium produced over a specified period of operation exceeds the amount of tritium consumed by an amount sufficient to account for losses and for supplying tritium to start new reactors. Commercial reactors are most likely to operate in that mode. This scenario of operation was previously discussed.¹³ The other mode of operation, the sub-breeding mode, allows for tritium breeding; however, the amount of tritium that must be supplied externally for initial start-up of the device is consumed during the plant lifetime. In other words, at the end of operation, only a minimum amount of tritium, required in case of failure of the tritium processing system, is left in storage. The latter mode of operation is proposed for ITER. The TBR value, at which the transition between the two modes of operation occurs,

is called the critical TBR and is slightly larger than unity. Thus, the TBR required for ITER is at most unity. If the achievable TBR is less than this critical value, a certain amount of tritium must be supplied from external sources during the lifetime of the device (above and beyond the initial start-up inventory). For example, calculations have shown that the 600-MW fusion power for ITER needs ~ 20 kg of external tritium supply for a blanket that can achieve a TBR of 0.8 and operates for 2.5 FPY.

A parametric analysis was performed by varying various system parameters. Examples of results from the analysis are illustrated in Fig. 1, which represents the required TBR as a function of neutron fluence at the first wall and system availability for a 600-MW fusion reactor and 5 kg of initial tritium supply. During the engineering test phase of ITER, 25% availability is required to achieve $3 \text{ MW}\cdot\text{yr}/\text{m}^2$ of fluence within 10 yr at a neutron wall load of $1.15 \text{ MW}/\text{m}^2$. As the figure indicates, such an operation requires a TBR around unity. The amount of initial tritium inventory significantly affects the required TBR. Figure 2 shows the required TBR as a function of initial tritium inventory for different numbers of years of operation of a 600-MW reactor with 25% availability. Nine years of operation, which is equivalent to a fluence of $2.6 \text{ MW}\cdot\text{yr}/\text{m}^2$, require a TBR of 0.87 if 20 kg of initial tritium inventory is assumed.

There are uncertainties in predicting the required and achievable TBRs. These include, for example, uncertainties associated with obtainable tritium fractional burnup in the plasma and the mean residence time in the tritium processing system. Figure 3 shows

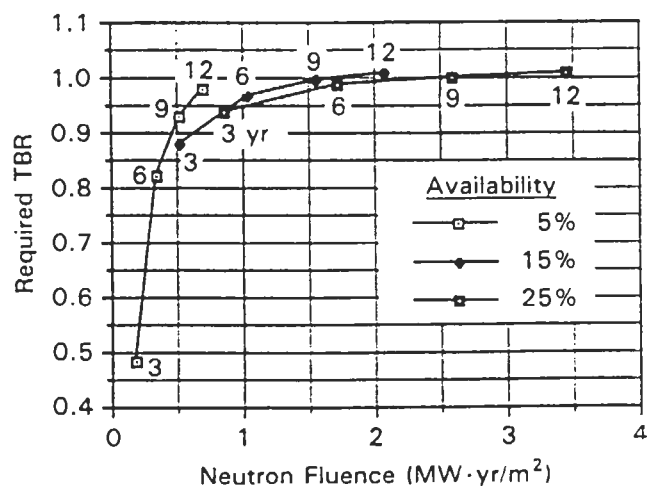


Fig. 1. Required TBR as a function of ITER fluence. Calculations assume representative fusion power of ~ 600 MW, a neutron wall load of $\sim 1.15 \text{ MW}/\text{m}^2$, and a 5-kg initial tritium supply. The numbers inside the figure indicate years of operations.

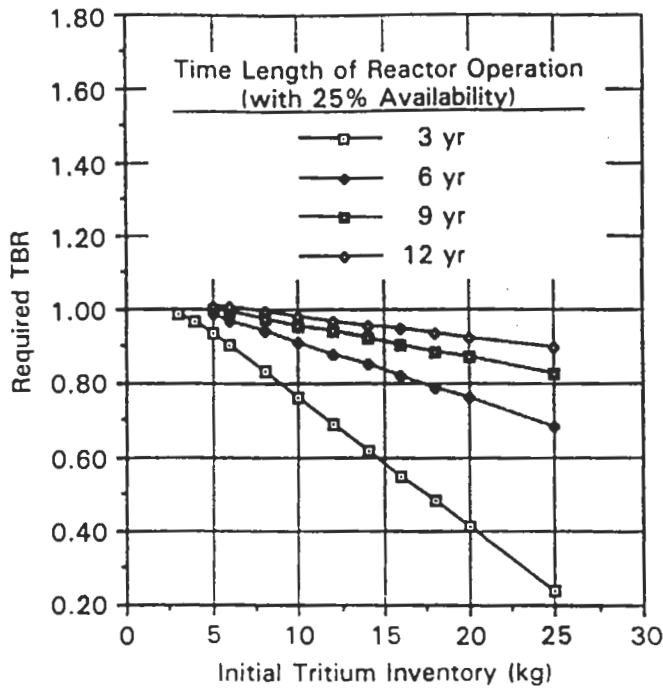


Fig. 2. Required TBR as a function of the initial tritium inventory for different numbers of years of operation assuming a 25% availability (fusion power = 600 MW, neutron wall load = 1.15 MW/m²).

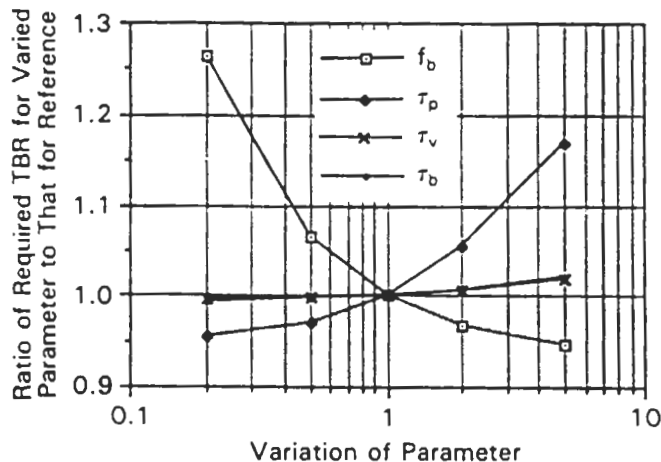


Fig. 3. Change in required TBR (assuming reference TBR = 1.0) with variations of the following parameters: tritium burnup fraction in the plasma (reference = 5%), tritium residence time in processing system (reference = 1 day), tritium residence time in vacuum system (reference = 2 h), and tritium residence time in blanket (reference = 10 days).

the relative variations of the required TBR with the following parameters: the tritium burnup fraction in the plasma f_b and the tritium residence times in the processing system τ_p , in the vacuum system τ_v , and in

the blanket τ_b . Both f_b and τ_p have a large impact on the required TBR because of the large tritium inventory in the tritium processing system. For example, if f_b decreases by a factor of 5 (i.e., from 5 to 1%), the required TBR increases by 26%; if τ_p increases by a factor of 5 (i.e., from 1 to 5 days), the required TBR increases by 17%.

Uncertainties in the achievable TBR arise from present uncertainties in basic data, calculational methods, and modeling of the reactor system. For ITER, the following suggestions are made:

1. Uncertainties in the required TBR should be substantially reduced by designing for high tritium fractional burnup in the plasma (>5%) and short residence time for tritium in the processing system (<1 day).
2. State-of-the-art calculations with sensitivity analysis should be performed to estimate the range of achievable TBR.

It is strongly recommended, however, that the ITER blanket not be burdened by an extra conservative breeding margin to compensate for uncertainties. A reasonable shortfall in TBR can be compensated for, if necessary, from external sources. For example, if a TBR of 0.9 is achievable, 17 kg of external supply of tritium would enable the reactor to operate for 9 yr with 25% availability. If the TBR is only 0.8, ~27 kg of initial tritium supply would be required for the same operating conditions.

IV. CONCEPT DESIGN AND ANALYSIS

This section presents the results of design and analysis for a helium-cooled SB blanket concept for ITER.

IV.A. Configuration

The design is based on previous studies^{10,14} and consists of a number of independent canisters, lying side by side in the poloidal direction, as shown in Fig. 4 for the outboard region of a representative ITER design. A rod bundle configuration inside each canister is chosen for design calculations. In this configuration, the SB and multiplier are in well-defined volumes to enhance the predictability of their properties. They can also be placed in the same rods or in different adjacent rods, creating a near-homogeneous mixture, which increases the neutronics performance. In addition, such a configuration provides the flexibility of varying individual rod sizes and compositions in different rows to allow for the exponential decrease in heat generation in the radial direction while maintaining the SB at an acceptable temperature. Finally, a rod bundle configuration provides reasonable heat transfer and pressure drop and has been well studied in heat exchanger designs. Note that the rods are staggered to reduce radiation streaming in the main blanket region.

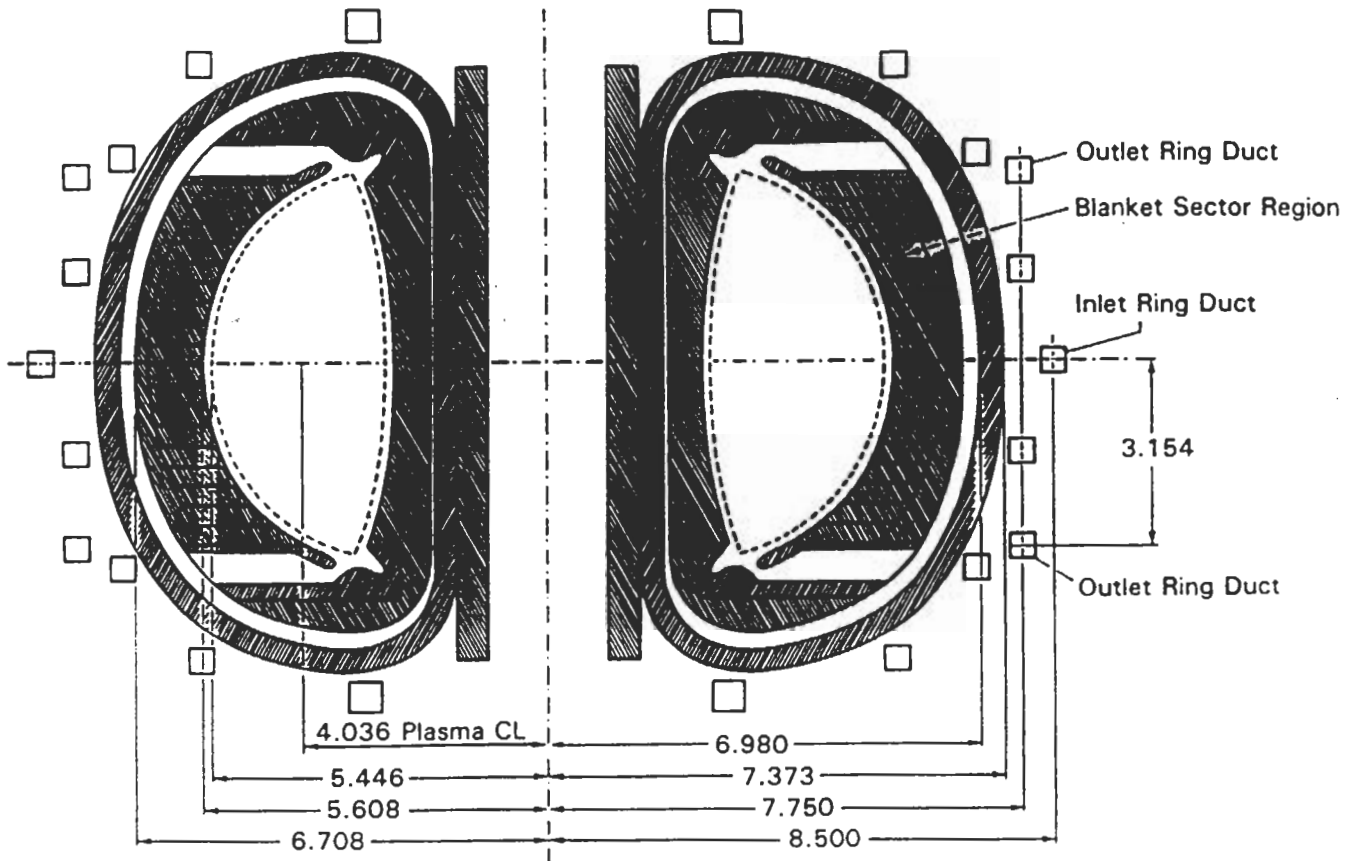


Fig. 4. Cross section of a representative ITER design showing canister layout. Dimensions are in metres.

However, the effects of radiation streaming in the plenum and manifolding must be accounted for in the design of the shield and manifolds.

Figure 5 shows a typical canister with the SB and/or multiplier rods. In the design, the canister is 33 cm wide and 1.1 m long. These small canisters were used to maximize the coverage of available space on the outboard of the ITER, where space allowances are required for test modules and penetrations. A single main helium flow is shown in this design to reduce the manifolding complexity. The flow comes in along the side wall to provide for the cooling of the first wall, and then enters the canister at the first wall and flows over the rod bundle before exiting the canister at the back. A porous mesh is provided along the side-wall helium channel so that a stagnant layer of helium inside the mesh will act as an insulator between the cooler side-wall flow and the hotter rod bundle flow. A 2-mm stagnant helium layer reduces the heat transfer between the canister flow and first-wall flow by a factor of ~ 7 , as compared to the case without the stagnant layer. The resulting heat transfer is then acceptably small and causes only an extra 10°C helium temperature rise in the first-wall channel.

Based on simple initial stress calculations, both the side and end walls are 5 mm thick with thin ribs

located ~ 10 cm apart, providing the support for each canister to be structurally independent. These calculations show that the stresses are up to an order of magnitude higher than the allowable stresses for a rectangular canister configuration that maximizes space utilization. A cylindrical geometry yields low stresses (~ 40 MPa) but poor space utilization. The indications are that between these configurations, there is a canister configuration maximizing the first-wall radius of curvature for optimum space utilization while keeping the stresses under the allowable limit.

For simplicity, a single rod configuration is shown inside the canister. However, the canister could easily be divided into two or three regions with different rod configurations in each, as required by neutronics or other considerations. The smaller rods are clad beryllium rods. The larger rods consist of a clad SB inner cylinder with a clad beryllium/helium packed bed annular region on the outside, as shown in Fig. 6.

The use of beryllium particles and helium to provide the thermal resistance between the SB and helium is proposed because it offers several advantages, including the following:

1. The thermal resistance between the SB material and coolant can be selected by properly tailoring the

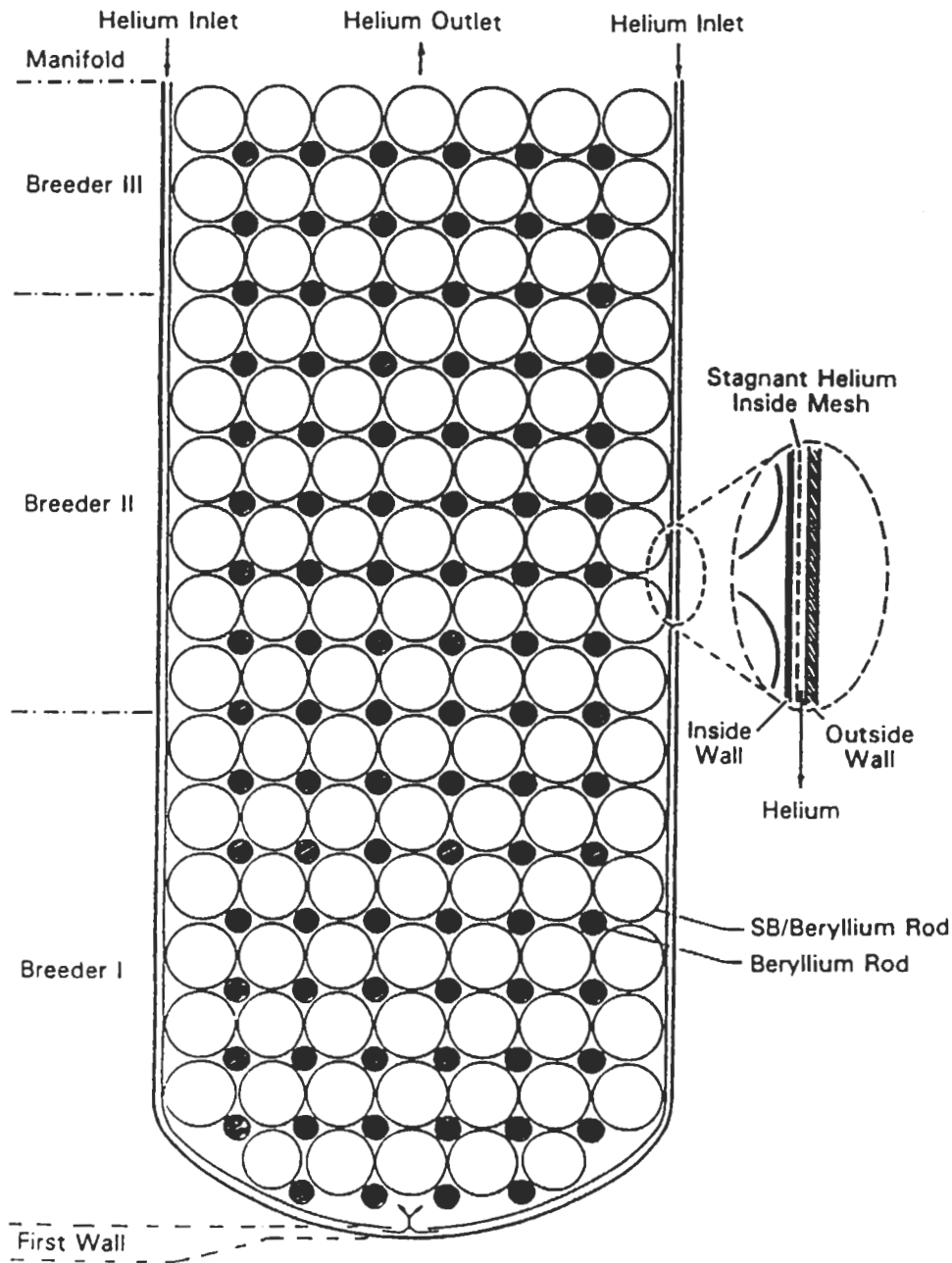


Fig. 5. Helium-cooled SB blanket canister for ITER.

bed characteristics (porosity, type, and size of particles, type, and pressure of gas).

2. The increased gap size, as compared to pure helium, allows for better uniformity and predictability of the thermal resistance.

3. The use of beryllium in a near-homogeneous mixture enhances the tritium breeding capabilities of the design.

4. The possibility exists of increased control over the SB temperature (through control of the effective

bed thermal conductivity by helium purge pressure and/or flow rate adjustments), which would permit operation over a wider range of power densities and transient conditions without violating temperature constraints.

5. Swelling can be partially accommodated by expansion into the bed void fraction.

6. Tritium generated in the beryllium multiplier is more easily removed, perhaps as a part of the basic SB purge system.

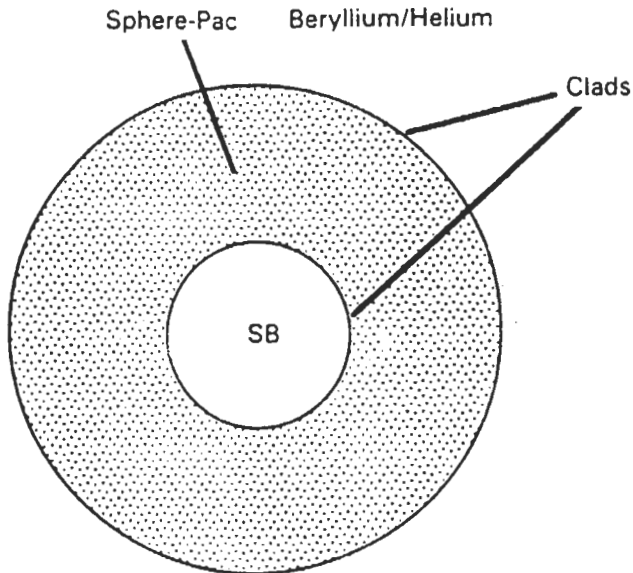


Fig. 6. SB/multiplier rod configuration.

7. An additional purged barrier is provided against tritium permeation to the coolant. The effects of the combination of SB, beryllium, and structural volume fractions in this configuration on TBR are addressed below.

A complete list of design parameters is given in Table II. Some of the more important ones are discussed in more detail in the following subsections.

IV.B. Neutronics

A series of one-dimensional calculations using the ONEDANT code¹⁵ and toroidal and poloidal geometry models was first performed to evaluate the impact of design options on tritium breeding. A three-dimensional Monte Carlo calculation using the MCNP code¹⁶ was performed for the final design to provide an accurate evaluation of the net TBR. The nuclear data used in the calculations were derived from ENDF/V-II with the beryllium data used in the MCNP analysis from the evaluation by Young and Stewart.¹⁷

The materials considered for the neutronics calculations are Li_4SiO_4 (based on the discussion in Ref. 18), beryllium, and primary candidate alloy (PCA) as breeder, multiplier, and structure, respectively. Figure 7 shows the outboard arrangement; the corresponding material volume fractions are shown in Table III. As an optimization procedure, the breeder region was initially divided into three neutronically homogeneous zones with different SB-to-beryllium ratios. The first zone (breeder I), which is the closest to the first wall, had the maximum beryllium-to-SB ratio of the three zones (86/14). The second breeding zone (breeder II) had a beryllium/SB mix ratio of 30/70, while the last

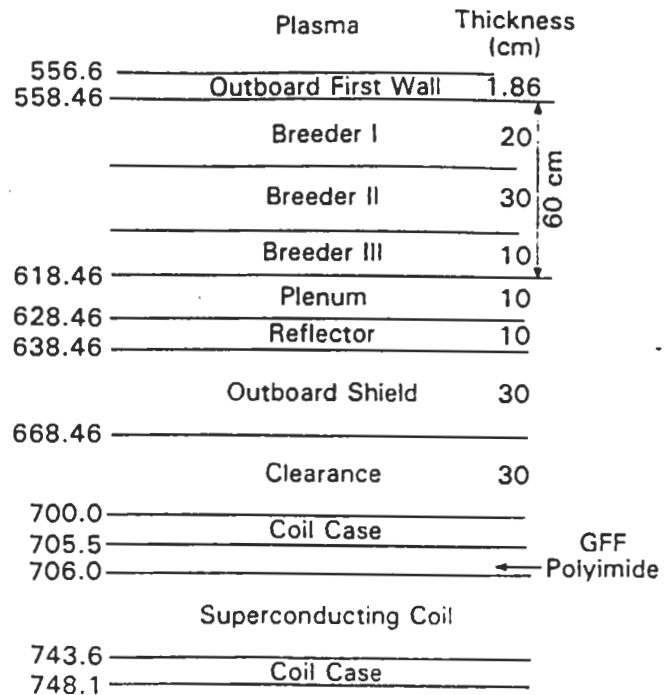


Fig. 7. Reference outboard blanket and shield configuration.

zone (breeder III) consisted of SB rods only. For these initial calculations, the TIBER water-cooled inboard shield configuration¹⁹ was adopted, with a total thickness of 55 cm. The idea was that the design and optimization of a helium-cooled inboard shield would be done later as a trade-off between the attractive safety feature of having an inert gas and the expected cost penalty of requiring more shielding material to account for the helium voids. In any case, since the inboard shield requires a separate cooling circuit for any blanket, it is reasonable to assume that the inboard shield, away from the lithium-carrying test modules in the outboard, can be water cooled.

Figure 8 shows the TBR as a function of the first breeder region thickness for a constant total breeder thickness of 60 cm and a 90% ^6Li enrichment. The TBR increases as the thickness of breeder I is increased, until saturation is reached at a thickness of ~ 50 cm. More importantly, the results illustrate the effect of the inboard shield configuration on the outboard TBR. The three cases shown correspond to (a) the water-cooled TIBER-II design,¹⁹ (b) a modified version of this design with the first 2.8 cm of water replaced by 1.4 cm of water and an additional 1.4 cm of beryllium, and (c) another modified version where the first 2.8 cm of water is replaced by 2.8 cm of helium. It is interesting to observe the effect of progressively removing the water. Using water as a coolant results in moderating the neutrons and decreasing the $(n,2n)$ reaction rate in the beryllium region in front of the

TABLE II
List of Major Parameters for Helium-Cooled SB ITER Blanket Design (Outboard Only)

Materials		Design Parameters (Continued)	
Coolant	Helium	Maximum nuclear heating rates (based on one-dimensional, homogenized region calculations)	
Breeder	Li ₄ SiO ₄	Breeder (W/cm ³)	16
⁶ Li enrichment (%)	90	Coolant	0
Neutron multiplier	Beryllium	Multiplier (W/cm ³)	16
Structure	PCA, 25% cold-worked	Structure (first wall) (W/cm ³)	9
Reflector	Carbon	Energy per fusion neutron (MeV)	19.7
Shield	Stainless steel and B ₄ C	First-wall/blanket description	
Configuration and Dimension		Total thickness (m)	1.1
Breeder/multiplier rods in canister		Manifold thickness (m)	0.1
Canister		Coolant total pressure drop (MPa)	0.1
Length (m)	1.1 to 1.3	Coolant pumping power (MW)	18
Width (m)	0.33	Coolant velocity (maximum/minimum)	
Number	480	First wall (m/s)	53/36
Rod		Canister (m/s)	24/20
Diameter (m)	0.042	Coolant flow rate (kg/h)	1.1 × 10 ⁶
Spacing between rods (m)	0.002	Tritium Removal	
Number per canister	98	Method	Purge flow
First wall		Steady-state blanket inventory	
Thickness (m)	0.005	Diffusion (g)	0.3
Channel width (m)	0.003	Solubility (g)	1.2
Second-wall thickness (m)	0.002	Surface adsorption (g)	12.2
Design Parameters		Purge gas parameters	
Coolant		Gas material	Helium
Inlet/outlet temperature (°C)	50/300	Inlet/outlet temperature (°C)	-400
Inlet/outlet pressure (MPa)	1.5/1.4	Pressure (MPa)	1.4
Breeder		Mass flow rate (kg/h)	590
Minimum/maximum temperature (°C)	325/500	Tritium production (g/FPY)	3.4 × 10 ⁴
Structure		HT partial pressure (Pa)	18
Maximum first-wall temperature (°C)	275	Heat Transport System	
Minimum/maximum temperature at Coolant interface (°C)	50/300	Helium side	
Breeder interface (°C)	325/400	Pressure drop (MPa)	0.1
Multiplier interface (°C)	150/400	Pumping power (MW)	2.2
Outboard neutron wall loading		Water side	
Average/maximum (MW/m ²)	1.5/1.9	Maximum pressure (MPa)	0.1
TBR		Inlet/outlet temperature (°C)	20/40
One-dimensional poloidal model with 100% coverage	1.6	Tritium barriers	Oxide layer probably
One-dimensional toroidal model with breeding in outboard only	1.3	LOCA	
Conservative net three-dimensional including all space losses	1.0	Method to accommodate	Initial calculations indicate passive method but needs verification

inboard shield. This in turn leads to decreasing the population of neutrons incident on the outboard region. A maximum TBR of 1.35 is obtained for the case where helium is substituted. This is another im-

portant factor to be considered for the comparison between a helium-cooled to a water-cooled inboard, particularly when considering that 3 to 4 cm of helium might be all that is needed to cool the inboard.

TABLE III
Compositions and Dimensions for Neutronics Calculations

Region	Thickness (m)	Volume Fraction			Shield Material
		Structure	Breeder [80% theoretical density (TD)]	Multiplier (90% TD)	
First wall	0.02	0.4			
Breeder I	0.2	0.09	0.07	0.46 ^a	0.38
Breeder II	0.3	0.16	0.37	0.14 ^b	0.32
Breeder III	0.1	0.20	0.51		0.28
Plenum	0.1	0.09			0.91
Reflector	0.1	0.05			0.1
Shield	0.3	0.4			0.1
					0.85 carbon 0.5 B ₄ C

^a90% TD.

^b80% TD.

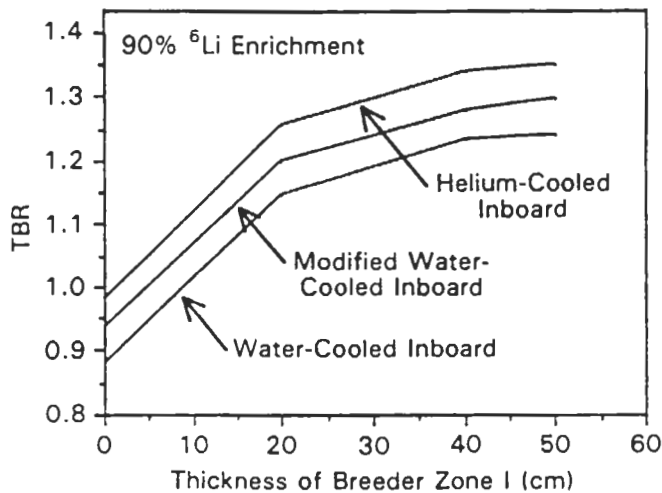


Fig. 8. TBR in outboard calculated using a toroidal arrangement as a function of breeder I thickness for a constant total breeder thickness of 60 cm.

The TBR can be enhanced by including blanket modules in regions other than the outboard blanket. Figure 9 shows the various geometrical regions in the blanket and shield. Assuming a single-null divertor at the bottom, the space available at the top is substantial and involves little penalty on the design. Adequate space is available for shielding and manifolding. Maintenance of that region should be comparable to other parts of the blanket, depending on the actual configuration and maintenance scheme for ITER. It is likely that maintenance operations in ITER will remove the entire sector, in which case the poloidal location of the breeding modules has little impact.

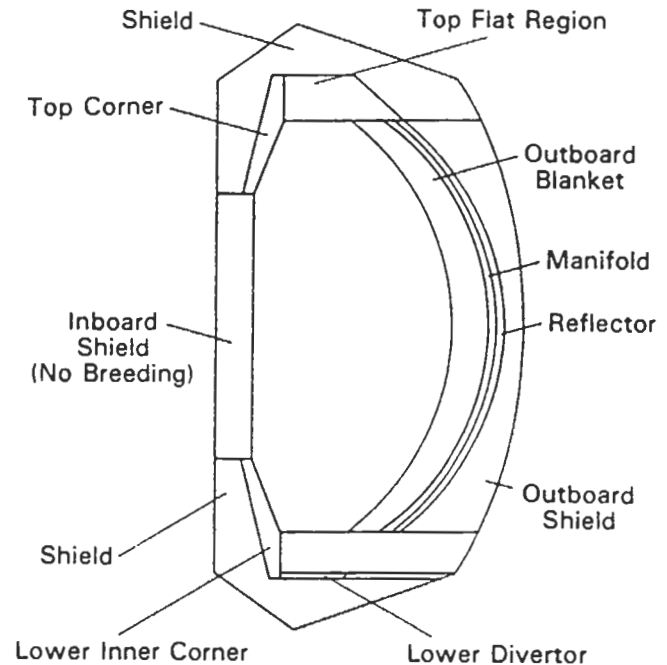


Fig. 9. Configuration for three-dimensional neutronics calculations.

The three-dimensional TBR was calculated using the MCNP code with a statistical error of 1%. The blanket and shield geometry used in the calculation is shown in Fig. 9. For simplicity, the model uses a uniform volumetric neutron source distribution in the plasma region. The breeding regions consist of lithium enriched to 90% ⁶Li and a beryllium-to-lithium ratio of 6.14.

Table IV summarizes the results. The net TBR without breeding in the space-limited inboard region is 1.27. The use of the actual neutron source distribution that is approximately parabolic squared with a magnetohydrodynamic plasma shift toward the outside would further increase the TBR in the outboard region. A 3% increase is estimated. Allowance needs to be made for the auxiliary plasma heating and diagnostic penetrations that were not included in the model. (Note that the divertor penetrations were included.) A 5% reduction in TBR is estimated. A portion of the blanket space will have to be used for test modules. About 20 m² of first-wall area will probably be occupied by test modules. However, since ~95% of this test space is probably used for blanket test modules, tritium will be produced and recovered from such space. Thus, loss of tritium breeding from test regions is expected to be small, no more than 5%.

It is apparent from the results in Table IV and the above discussion that a net TBR in excess of unity can be achieved in ITER with a helium-cooled SB blanket. This assumes no breeding in the space-limited inboard region and accounts for the divertor penetrations. There is ~30% excess breeding margin to compensate for other types of penetration and test modules. There is no incentive to have a TBR greater than unity as indicated in Sec. III. Therefore, this excess margin can also be used to further simplify the designs; for example, by eliminating breeding in the difficult access region of the lower inner corner (see Table IV and Fig. 9).

IV.C. Thermal Hydraulics

Figure 10 shows the canister helium pressure drop ΔP as a function of the inlet pressure $P(in)$ for different helium temperature rises ΔT_{He} for the case where the rod diameter is 4 cm and the spacing between rods is 2 mm. Even if the pressure drop is doubled or tripled to account for the helium flow outside the can-

ister (as our calculations indicate), a wide range of $P(in)$ and ΔT_{He} choices are available. As shown in Fig. 11, the corresponding pumping power requirement is <20 MW for all cases with $P(in) \geq 10$ atm and $\Delta T_{He} \geq 250$ K.

A $P(in)$ of 15 atm and a ΔT_{He} of 250 K are reasonable design choices; the required pumping power for the complete helium circuit, including the piping and heat exchanger, is ~22 MW. The low stress caused by the pressure will facilitate the design of the canister structural configuration.

IV.D. Power Variation Allowance

This blanket is very flexible in allowing for operation at 50% below or above the design power level. Both passive and active means can be employed to

TABLE IV

Contribution to TBR from Various Geometrical Regions as Calculated from Three-Dimensional Monte Carlo

Region Contribution to TBR	TBR
Outboard blanket	1.01
Top flat region	0.082
Top corner	0.064
Inboard shield (no breeding)	0.0
Lower inner corner	0.067
Lower divertor ^a	0.06 ^a
Total TBR	1.27

^aFrom extrapolation.

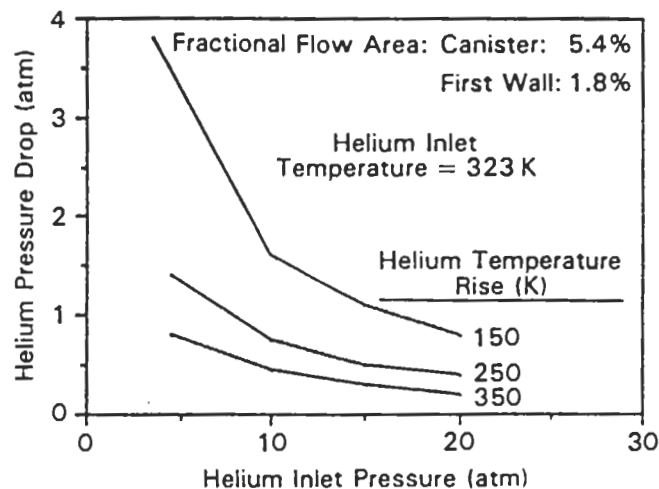


Fig. 10. Canister helium pressure drop as a function of the inlet pressure for different temperature rises.

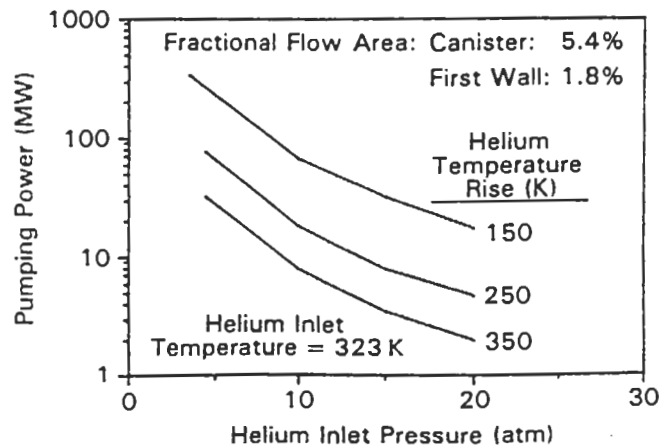


Fig. 11. Canister helium pumping power as a function of the inlet pressure for different temperature rises.

allow for power variation. The passive means make use of the large SB temperature window, whereas the active means make use of the helium temperature, which can be varied by changing the flow rate and/or inlet temperature without any significant pressure penalty.

Figure 12 illustrates the flexibility of the design in this regard for a canister located at the highest wall-loading location (1.9 MW/m²). The figure shows the maximum SB temperature in row (r) as a function of the thickness r of uniform breeder region I. The minimum temperature of the SB at $r = 0$ is 600 K. Any increase in the power level is allowed by letting the SB temperature in row (r) rise to the maximum allowable temperature, 1000 K. Thus, for 20-cm-thick breeder region I, as an example, the maximum SB temperature is 760 K, and the allowable increase in power is 50%. There is a trade-off here between increasing the allowable power variation and decreasing the number of uniform regions per canister for design simplicity. Since the wall loading varies poloidally, different canister configurations in the poloidal direction would also be required to account for the power variation. Note, however, that the 20-cm thickness required for breeder region I at the highest wall loading is the minimum uniform canister region thickness. There is much

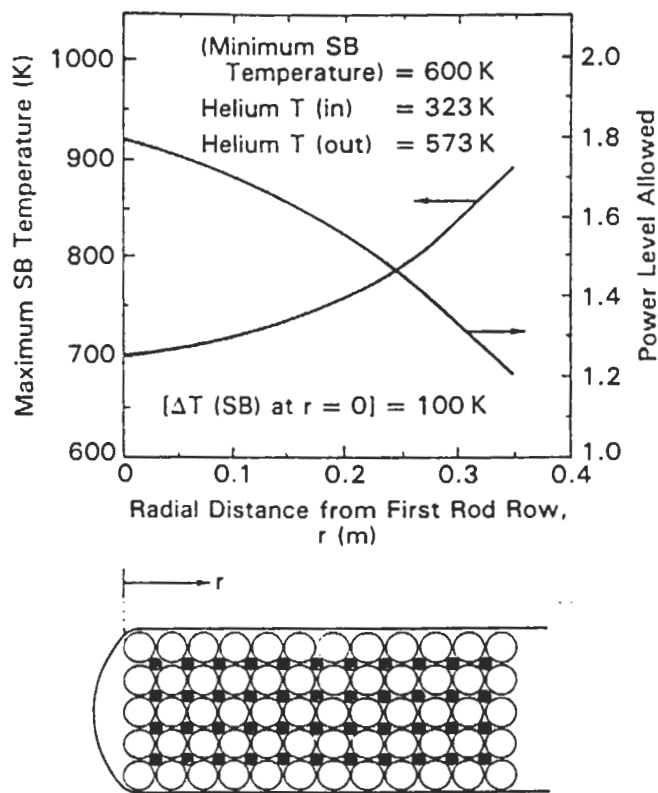


Fig. 12. Maximum SB temperature in row (r), and power level allowed as a function of the radial thickness of a uniform first breeder region.

more flexibility in setting the thickness of uniform regions in the canisters with lower power densities. Our calculations indicate that, to allow for a 50% increase in power, two or three different regions per canister and three or four different canister configurations along the poloidal direction are required. Note that any decrease in power can be easily accommodated with no apparent penalty by increasing the helium temperature through a reduction of its flow rate and/or an increase in its inlet temperature.

IV.E. Reliability

The reliability of the blanket is often thought to decrease as the number of rods increases. The overall consequence of failure, though, does not necessarily follow the same trend since the reduced consequences of failure per rod can outweigh the increased probability of rod failure in designs with a larger number of smaller rods. To address this issue, the probability of achieving a given level of consequences was investigated in different designs. The calculations performed assume that tritium contamination is one of the most serious consequences of rod failure.

Figure 13 shows the probability of tritium release for different SB volumes from which tritium escapes due to weld failure. Three cases are considered corresponding to rod diameters of 28, 40, and 56 mm. Note that the number of rods for a given canister varies inversely with the rod diameter. The total length of

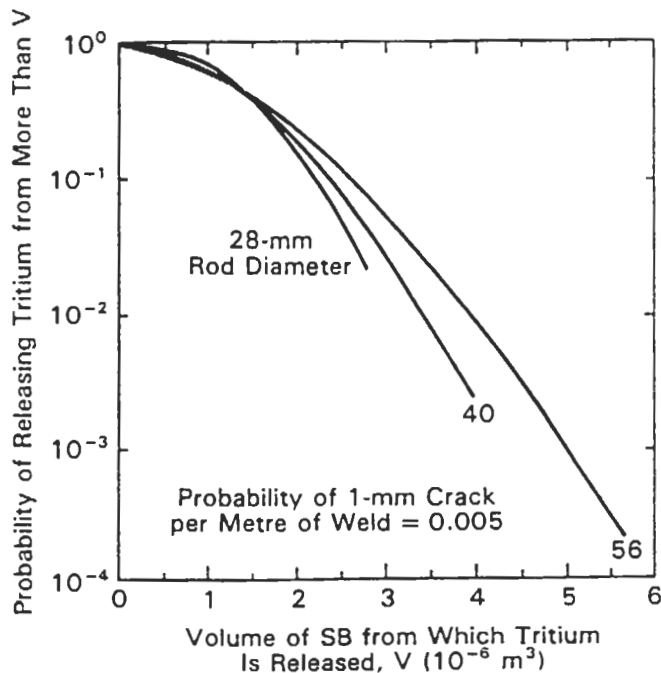


Fig. 13. Probability of releasing tritium from more than a given SB volume V for different rod diameters (i.e., different number of rods).

welds depends both on the size and the number of rods. A probability of 0.005 is arbitrarily assumed for a 1-mm crack per metre of weld. For this example case, depending on whether the critical amount of tritium release is less than or greater than the content of $1.5 \times 10^{-6} \text{ m}^3$ of SB, the most reliable design is the one with the smaller or greater number of rods, respectively. The 0.005 probability for a 1-mm crack per metre of weld used here does not affect the conclusion, as a different value will merely shift the point at which more rods are more reliable than fewer rods.

In summary, the rod design is very failure resistant. It provides multiple barriers against contamination. With the inert coolant, the consequences of failure are not catastrophic, at worst leading to contamination of the main coolant. The rod design also dilutes the source term, and increasing the number of rods can actually decrease the probability of exceeding contamination limits.

IV.F. Loss-of-Coolant Accident

Figures 14 and 15 summarize the results from the initial loss-of-coolant-accident (LOCA) calculations. They illustrate the change with time of the temperatures of the first wall and the Li_4SiO_4 SB in the first row of rods following a LOCA. Under adiabatic conditions, in time both the first-wall structure and SB would exceed their maximum allowable temperatures, resulting in loss of investment. If radiation to the cooled inboard is assumed, the first-wall temperature stays below half of its melting point of $\sim 1700 \text{ K}$. Including the effect of free convection (which results in heat transfer from the hot SB to the colder first wall) slightly increases the first-wall temperature, but only slows the time for the SB to reach its maximum allowable temperature of 1000 K (based on sintering). Including the effect of the purge flow to remove afterheat is found to be of substantial help. Even if radiation to the inboard is excluded, the SB and first-wall structure would not reach their maximum allowable temperatures for a purge flow average velocity of 0.005 m/s or more. The purge pressure drop for that case is reasonable, $\sim 50 \text{ kPa/m}$.

IV.G. Helium Leakage to Plasma

It is difficult to accurately predict the helium leakage to the plasma, but some observations can be made. Extrapolation of the data from Fort St. Vrain high-temperature gas-cooled reactor²⁰ indicates that the leakage rate to the plasma is only $\sim 1\%$ of the helium generation rate in the plasmas, based on an estimated 4700-m total weld length exposed to plasma. However, estimates assuming cracks through the whole thickness of the first wall could substantially increase this. The difficulty of accurate calculations is exacerbated because the number and size of cracks escaping inspection are not known and mechanisms

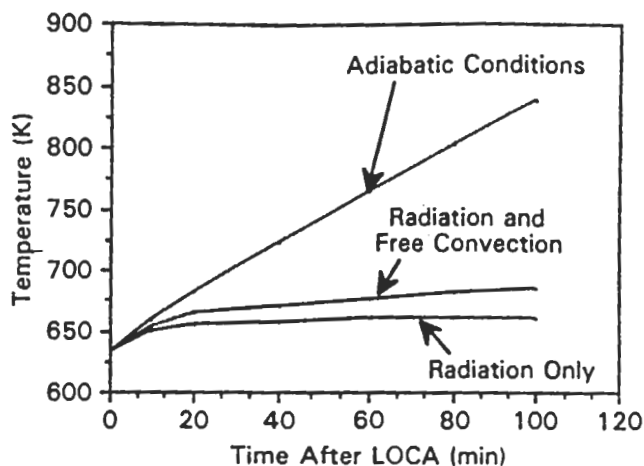


Fig. 14. First-wall temperature following a LOCA.

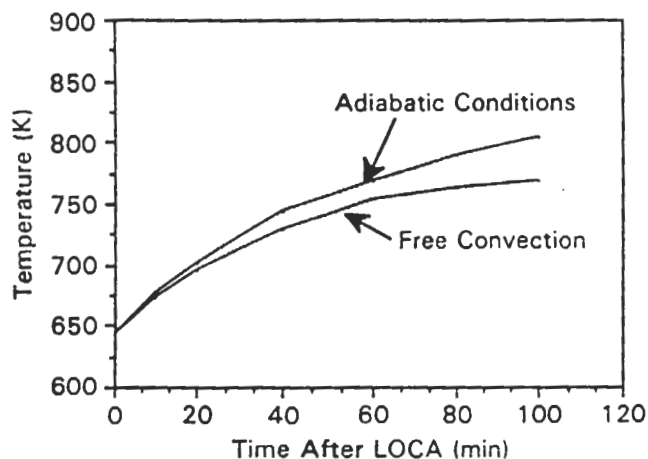


Fig. 15. SB temperature following a LOCA.

such as crack growth must be considered. Including this consideration in the decision process can help alleviate the problem. For example, crack propagation is more problematic in welds, and the number of welds in contact with the plasma can be minimized. Note also that any reasonable amount of helium leaking to the scrape-off region would be swept away in the divertor and would probably provide a beneficial effect by cooling the plasma edge.

IV.H. Tritium Inventory

The tritium inventory was estimated to be the sum of the diffusive, solubility, and absorption inventories. Expressions for estimating the diffusive and solubility inventories were based on Ref. 21. The diffusive inventory calculations assume a $2\text{-}\mu\text{m}$ grain radius, a diffusion preexponential constant of $2.1 \times 10^{-11} \text{ m}^2/\text{s}$, and an activation energy of 64 kJ/mol for Li_4SiO_4 . The solubility inventory was estimated to be one-third of

the corresponding solubility in Li_2O . The surface adsorption inventory was calculated by equating the expressions for the rates of adsorption and desorption²² for the resulting coverage, assuming a 1% hydrogen concentration in the purge. The heat of adsorption for Li_4SiO_4 was assumed to be 100 kJ/mol. As can be seen in Table II, the total steady-state tritium inventory in the Li_4SiO_4 breeder is estimated to be very low, 13.7 g. Radiation effects and other uncertainties could increase this inventory. However, the estimated inventory is low enough that even a factor of the order of 10 to 100 increase due to uncertainties would be acceptable. The concern regarding tritium accumulation in beryllium is lessened in this design since the beryllium can also be purged. However, the effect of a BeO layer and its thickness on the tritium inventory inside the beryllium spheres should be considered as it could freeze much of the tritium inside the beryllium.

V. TECHNICAL ISSUES

Important technical issues associated with this blanket design include the following:

1. the beryllium-packed bed conductance between clad and breeder. This is discussed in more detail in Sec. V.A.
2. irradiation effects on SB material properties, in particular those affecting the tritium transport. Note, however, that the estimated tritium inventory in the whole blanket is low and uncertainties of the order of a factor of 10 to 100 can be tolerated.
3. helium containment and manifolding. It is difficult to accurately predict the helium leakage to the plasma and further effort is required in this area to determine the extent of the problem and, if required, to develop acceptable solutions. The difficulty associated with helium manifolding is linked with the relatively large piping dimensions. This difficulty can be minimized by optimizing the configuration arrangement and maximizing the use of the outboard for breeding while excluding breeding in the space-limited inboard.

V.A. Packed Bed Conductance

A novel idea for producing the thermal resistance between coolant and breeder has been proposed in Sec. IV whereby a mix of beryllium particles and helium is used. The increased thermal conductance, when compared to earlier pure gas thermal resistance design, allows for larger dimensions and results in a more forgiving design since the effective conductance is not as affected by small changes in the gap geometry. Some of the other advantages of such a concept are listed in Sec. IV.A. Figure 16 illustrates the effect

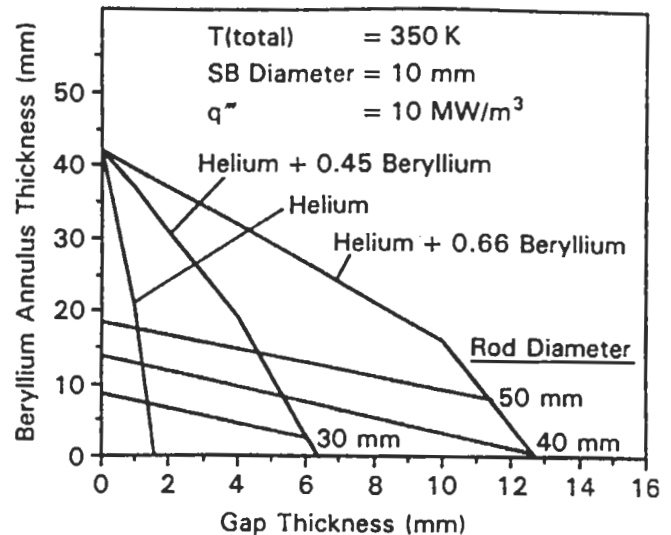


Fig. 16. Combinations of thicknesses of beryllium/helium annulus and helium gas gap to produce a temperature drop of 350 K for different beryllium/helium volume fractions.

of different helium/beryllium volume fractions on the required packed bed region thickness to achieve a temperature drop of 350 K for an SB inner rod diameter of 10 mm and a heat generation of 10 MW/m^3 . The calculations assumed that there is an extra solid beryllium annulus of thickness δ_{Be} surrounding the beryllium/helium packed bed region (of thickness δ_g) shown in Fig. 6. In that way, if δ_g is too small, extra beryllium can be available as required from neutronics consideration. The figure shows the combination of δ_{Be} and δ_g needed to achieve the 350 K temperature drop for different helium/beryllium volume fractions in the beryllium/helium-packed bed region. Lines of constant overall rod diameters are also shown. For no extra solid beryllium annulus (i.e., $\delta_{\text{Be}} = 0$), δ_g varies with the beryllium/helium region configuration from 1.5 mm for pure helium to 12.7 mm for a 34/66% helium/beryllium mix. Considerations of the neutronics requirement for adequate beryllium volume fraction and of the need for a large enough gap to allow for any small geometry changes indicate the need for a low-porosity (30% or less) helium/beryllium mix. A 30/70% helium/beryllium mix is chosen here as an example design value. The packed bed region thickness will then be ~ 14 mm. Note that these calculations were done based on an existing correlation for packed bed effective thermal conductivity²³ (see Table V) and include significant uncertainties.

In fact, a key issue is the predictability and reproducibility of the packed bed conductance. One of the advantages of a helium-cooled SB blanket design for ITER is its ability to accommodate power variation passively and/or actively. Although it is possible to

TABLE V

Comparison of Effective Thermal Conductivities of Packed Beds Based on Different Correlations

Formulas, ^a $\frac{k_{eff}}{k_{gas}}$	Gas		Divergence with 1 (%)		Comments	References
	Helium	Argon	Helium	Argon		
1. $32.8 \log \frac{0.43 + 0.31\epsilon}{\epsilon - 0.26}$	2.86	0.35			Experimental, graphite air beds; $d_g/d_s > 10$	Gorbis ²³
2. $(3.506 - 1.37 \times 10^{-3}T + 4.23 \times 10^{-7}T^2)k_{gas}$	3.2		+12		Experimental, three size sphere-pac U ₂ O fuel	Peddicord et al. ²⁴
3. $f\left(\frac{d_s}{\lambda} \frac{P}{2P_0} \frac{2-x}{x}, \frac{k_s}{k_{gas}}, \frac{k_{con}}{k_{gas}}\right)$	1.47	0.23	-48.6	-34.3	Theoretical, recommended for $P < 0.133$ Pa, $d_g/d_s > 10$	Wakao and Vortmeyer ²⁵
4. $e\left(1 + \frac{h_r d_s}{k_{gas}}\right) \times \frac{1 - \epsilon}{\frac{1}{1/t_g + h_r d_s/k_s} + \frac{2}{3} \frac{k_{gas}}{k_s}}$	3.54	0.55	+23.8	+57	Theoretical, recommended for nonvacuum beds, $d_g/d_s > 10$	Kunii and Smith ²⁶

^aHere, k_{con} is the contact thermal conductivity, h_r is the thermal radiation heat transfer coefficient, λ is the mean-free-path of gas molecules, x is the accommodation coefficient, P is the pressure, P_0 is atmospheric pressure, and t_g is the effective thickness of gas between spheres.

increase the helium flow rate to accommodate power increase, it is desirable to avoid as much as possible the pressure drop penalty associated with this and to keep this active method as an ultimate safeguard for large power increases. The passive method is then particularly attractive for allowing for increase in power. It makes use of the large SB temperature window and, as discussed in Sec. IV.D, could be designed to accommodate up to a 50% power increase. Under normal operation, the SB would operate over a small temperature range at the bottom of the temperature window. Long-term increase in power could be accommodated by allowing the SB operating temperature range to rise within the temperature window.

The passively allowable power increase δW_{Pas} can be estimated from

$$\delta W_{Pas} = \frac{[T_{max,a} - T_{He(in)}]}{[T_{max,0} - T_{He(in)}]} - 1, \quad (1)$$

where

$T_{max,a}$ = maximum allowable SB temperature

$T_{max,0}$ = maximum SB temperature under normal operating conditions

$T_{He(in)}$ = helium main flow inlet temperature.

Thus, as $T_{max,0}$ is increased, δW_{Pas} decreases, as illustrated by the two curves in Fig. 12. Figure 17 illustrates

the temperature distribution for a typical SB rod in the first row of rods, where the power generation is the highest. The gap component ΔT_{gap} is the major contributor to the overall temperature difference $[T_{max,0} - T_{He(in)}]$. Any relative uncertainty in ΔT_{gap} will significantly affect the prediction of $T_{max,0}$ and, in turn, the allowable power increase from Eq. (1). Due to the gap configuration, there are several sources of uncertainty that could affect the gap conductance during operation. These include the effects of irradiation swelling and thermal expansion on the gap geometry and on the beryllium thermal conductivity, bulk heating in the gap, and a slow-flowing helium purge.

The effective thermal conductivity k_{eff} of packed beds under normal conditions is a function of a number of parameters:

$$\frac{k_{eff}}{k_{gas}} = f\left(\frac{k_s}{k_{gas}}, \frac{k_{rad}}{k_{gas}}, \frac{k_p}{k_{gas}}, \epsilon, \frac{\delta_g}{d_s}, \phi, f_p \dots\right), \quad (2)$$

where

k_{gas}, k_s = thermal conductivity of the gas (helium) and solid (beryllium)

k_{rad}, k_p = effects of thermal radiation and low pressure

ϵ = porosity

δ_g = gap thickness

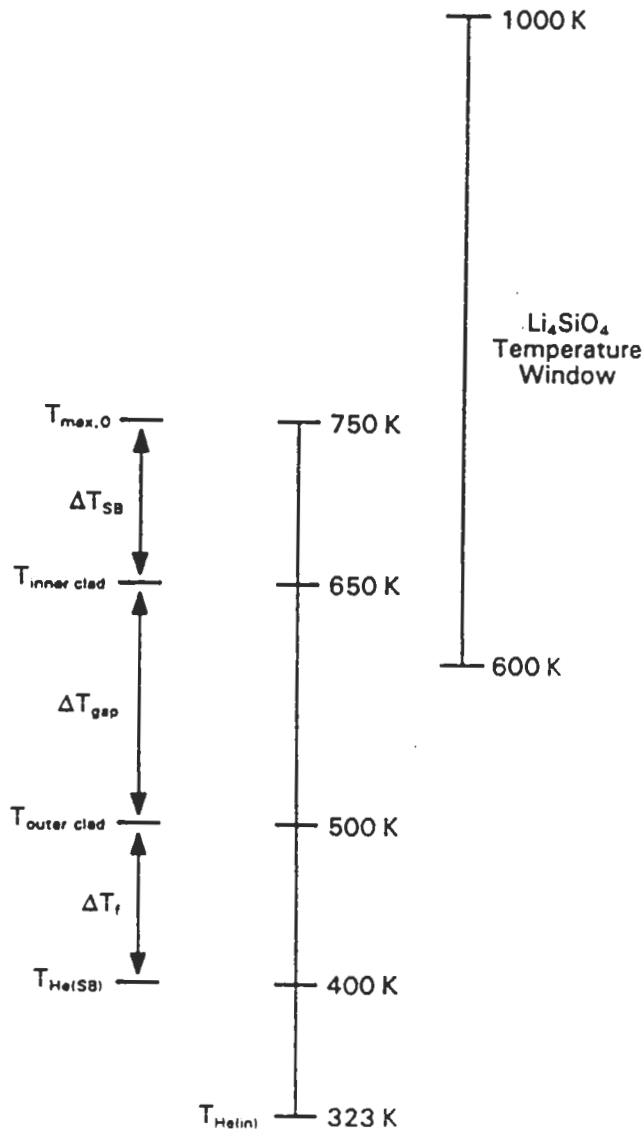


Fig. 17. Illustration of temperature distribution for a typical SB rod in the row closest to the plasma. (Note that ΔT_{clad} is negligible.)

d_s = mean diameter of the solid particles

ϕ = form factor

f_p = polydispersion factor.

The number of variables affecting k_{eff} can be reduced by assuming certain conditions. For example, the effects of thermal radiation and pressure can be minimized by assuming a temperature T lower than ~ 700 K and a high enough pressure P to effectively reduce the Knudsen number, respectively.

The effective thermal conductivity is still very dependent on a number of variables and on specific experimental or operational conditions. Table V shows a comparison of the calculated effective thermal conductivity, based on different available correlations, for packed particle beds with helium/beryllium and

argon/beryllium assuming the following conditions: porosity, $\epsilon = 0.4$; temperature, $T = 550$ K; pressure, $P = 10^5$ Pa; particle diameter, $d_s = 10^{-4}$ m; and packed bed thickness, $\delta g = 3 \times 10^{-3}$ m. The differences between the predicted values of k_{eff} are quite appreciable (up to 57%) when comparing the different correlations that are based on data obtained under different experimental conditions. For lower porosity (20 to 25%), the differences between the predictions from various models for k_{eff} are even larger.

The above discussion highlights the difficulty of estimating k_{eff} with acceptable accuracy from the available models and the importance of obtaining experimental data and deriving models for conditions as close as possible to those under which the packed bed will operate in a blanket. Based on this, it seems that an experimental program is needed with the aim of evaluating the packed bed effective conductivity under normal conditions for low-porosity and high-conductivity pebbles (to model beryllium) and then as required in the future to progressively include the effects associated with in-reactor operation (such as prototypic material, temperature, flowing helium, bulk heating, and eventually irradiation).

VI. SUMMARY

The helium-cooled SB blanket concept is an attractive candidate for the basic tritium-producing blanket in ITER from several points of view, such as the well-studied configuration, the expanding data base, the safety, and the information it provides on reactor-relevant SB operation at reactor temperatures. The key issues of neutronics, thermal hydraulics, and gap conductance have been addressed. A net TBR in excess of unity can be comfortably designed with a breeding margin for uncertainties. The helium pressure and inlet and outlet temperatures are moderate (15 atm and 323 and 573 K) and do not pose any significant penalty on the structural design. Helium cooling provides flexibility in selecting optimum temperature for structure and in allowing for power variation. A novel and practical concept has been proposed for the thermal resistance gap between the SB and the coolant. The concept is applicable to other blanket concepts and should be verified experimentally.

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