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AN OUTBOARD SHIELD DESIGN FOR TIBER-II WITH POTENTIAL FOR TRITIUM SELF-SUFFICIENCY

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Summary

The outboard breeding shield design for TIBER-II is described. The design allows for tritium self-sufficiency without compromising magnet protection, design simplicity, and the testing mission of the device. The shield consists of a beryllium pebble front zone backed by a steel pebble zone. The shield is cooled by an aqueous solution containing 16 g $LiNO_3$ per 100 cm^3. A double first wall is used to insure uniform cooling and minimize pressure. The design pressure for the outboard shield is 0.19 MPa and the coolant temperature is less than 75°C.

Introduction

Recently, there has been increased activity in the international fusion community aimed at designing an Engineering Test Reactor (ETR). TIBER-II [1] is the present U.S. candidate ETR design. The reactor is a compact steady state tokamak with a major radius of 3 m and 3.6 aspect ratio. The plasma shape and the neutron wall loading distribution obtained using the NEWLIT [2] code are given in Fig. 1. The space available for the outboard shield is 123 cm at the reactor midplane while the inboard shield thickness is only 48 cm. This implies that an efficient shield with large neutron attenuation is not required on the outboard side and a low cost simple design can be utilized. A low temperature, low pressure, water cooled shield that utilizes steel pebbles was considered for the outboard shield design. This low risk, low cost shield design results in magnet radiation effects which are several orders of magnitude below the radiation limits.

Although tritium self-sufficiency is not a requirement for an ETR, partial breeding might be essential in TIBER-II if tritium is available only

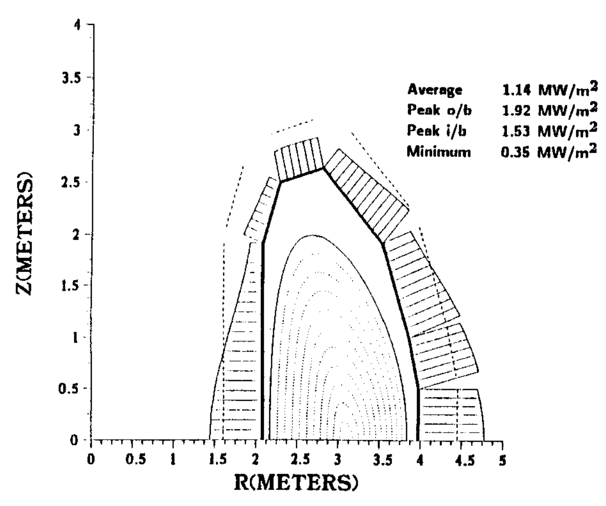

Fig. 1. Plasma shape and neutron wall loading distribution in TIBER-II.

from civilian sources. While the Canadian tritium production rate is ~ 2.5 kg/yr, the tritium burn rate in TIBER-II is ~ 5.3 kg/yr. In addition, tritium self-sufficiency is desirable due to the large cost of tritium ($10 M/kg). Modifying the shield design to breed as much tritium as possible without jeopardizing magnet protection and design simplicity was considered. An attractive concept that meets these requirements is the aqueous self-cooled blanket concept [3] where small contents of lithium compounds are dissolved in the water coolant. This design preserves the attractive features of low pressure and low temperature water cooled shields. It allows for inboard breeding without compromising magnet protection. Neutronics analysis for an early version of the TIBER-II design [4] indicated that while partial tritium breeding can be obtained by adding the Li salt to the water coolant, tritium self-sufficiency can be achieved using a front beryllium zone in the outboard region. The impact of these modifications on cost was also assessed. A cost saving of ~ $450 M over the life of the machine can be achieved by modifying the shield for tritium self-sufficiency. In addition, such a modification reduces the peak magnet damage occurring in the inboard legs of the TF coils. In this paper, we describe the design of the outboard breeding shield for TIBER-II and assess its tritium breeding potential.

Neutronics Analysis

Neutronics analysis was performed for the outboard breeding shield to determine the design that maximizes the local outboard tritium breeding ratio (TBR) while providing adequate magnet protection. Many Li compounds have high solubility limits in water allowing for significant tritium breeding. The nuclear performance of the different Li salts was assessed. The discrete ordinates code ONEDANT was used with ENDF/B-V data. Similar values of TBR were obtained for salts with high solubility when concentrations close to their solubility limits were used. Lithium salts with potentially large activation concerns such as LiI, LiBr, LiCl, $LiClO_4$ were not considered further in this study. In addition, salts with low solubility limits such as $LiHCO_3$ and Li_2CO_3 are not considered because of their limited tritium breeding potential. This lead to identifying lithium nitrate and lithium hydroxide as the primary candidates. $LiNO_3$ is used based on favorable preliminary corrosion experimental results [5].

Different outboard breeding shield design options were considered. These included using different neutron multiplying materials. Beryllium, lead, and zircaloy pebbles were considered. A high packing fraction required for enhancing tritium breeding can be achieved by utilizing different size pebbles. However, this can lead to a large pressure drop. A design with single diameter pebbles is preferred in order to minimize the pressure drop. Our results indicate that using single diameter pebbles will have only a small impact on the shield nuclear performance with the TBR decreasing by less than 10% and the magnet radiation effects increasing by less than 20%. Replacing Be by Pb reduces the TBR by 18%

with magnet damage decreasing by 25%. Since magnet radiation effects in the outboard side are ~ 3 orders of magnitude below the radiation limits, the 25% better shielding performance of lead is not considered as a major factor for multiplier choice. Because of its reduced tritium breeding potential in addition to the heavy weight, large creep, low modulus, low strength, high stress corrosion cracking in water, low melting point and safety concerns related to polonium production, lead was not considered further as a neutron multiplier. Calculations have also been performed for the option in which zircaloy is used as the structural material and neutron multiplier. This reduced the TBR by 36% with magnet radiation effects increasing by ~ 36%. Based on these results, Be is used as the neutron multiplier in the outboard breeding shield of TIBER-II.

Detailed thermal hydraulics considerations resulted in using a double first wall that insures uniform cooling and minimizes system pressure. Calculations have been performed for the baseline shield design with the first wall modelled separately. The aqueous solution cooled first wall is 1 cm thick with 1 mm thick PCA front and back zones. Single diameter balls were used in both the multiplier and steel zones. The multiplier zone consists of 58.8% Be (0.9 d.f.), 35.6% aqueous solution, and 2.6% PCA. The steel zone consists of 65.1% PCA and 34.9% aqueous solution.

Nearly the same TBR is obtained for $LiNO_3$ concentrations in the range 16-40 g/100 cm^3. Further concentration reduction leads to sharp decrease in TBR. A $LiNO_3$ concentration of 16 g/100 cm^3 is chosen for the baseline design to reduce corrosion and activation products without compromising the tritium breeding capability. Increasing the Be zone thickness beyond 40 cm results in only a very small enhancement in TBR (< 2%) with increased Be cost. A multiplier zone thickness of 40 cm is used in the baseline design. The nuclear performance parameters for the baseline outboard shield design are given in Table 1. The results are normalized to the peak outboard neutron wall loading. It is clear that magnet damage is at least three orders of magnitude below the radiation limits. The maximum accumulated damage in the PCA structure at end-of-life is 48 dpa.

Configuration Options

We have considered three shapes for the neutron multiplier/shielding material: rods, plates and spherical pebbles. From the standpoint of geometric accommodation, fabrication, support, penetration accommodation and cost, the spherical pebbles appear to have an edge over rods and plates. Rods and plates on the other hand can provide a higher material fraction which is a shielding advantage, and a lower coolant pressure drop with the potential of lower design pressure and temperature operation. Since there is much more room for shielding on the outboard side, the higher material fraction in not important; however, the irregular shape of the segments make geometric accommodation very critical. These and the other advantageous aspects of spherical pebbles make them the obvious choice for the outboard shield. This is especially true since we will show that pebbles can also be designed for low pressure and temperature operation.

Table 1. Nuclear Performance Parameters of the Baseline Outboard Shield Design
(16 g $LiNO_3$/100 cm^3, 40 cm Be Zone)

Local (full coverage) TBR	1.628
Local energy multiplication	1.404
Peak power density (W/cm^3)	13.70
Peak dpa rate in PCA (dpa/FPY)	19.2
Peak He production rate in PCA (He appm/FPY)	220.5
Peak fast neutron fluence in S/C (n/cm^2)	8.53×10^{15}
Peak winding pack power density (mW/cm^3)	4.16×10^{-3}

Thermal Hydraulics Analysis

One of the goals of the breeding shield is to make it low technology which means operating at a low pressure and temperature. For water cooling the operating pressure is determined by the hot spot temperature. Figure 2 is a midplane cross section of an outboard shield segment. We have considered two possible configurations of the first wall. In a single wall configuration, the pebbles rest against the first wall, and in the double wall configuration, the first wall is separately cooled, with the pebbles resting against the second wall. For both cases we use the average surface heating of 25 W/cm^2.

The heat transfer coefficient on the pebble surface in a pebble bed can be determined from empirical correlations. Heat transfer coefficients at the wall where the pebbles come in contact with it are somewhat lower. These coefficients are made up of a stagnant or conductive contribution and a turbulent or convective contribution [6]. Knowing the surface and nuclear heating in the shield and the heat transfer coefficients we can determine T_{max} within the shield if the coolant outlet temperature is known. The coolant outlet temperature depends on the inlet temperature and mass flow rate. The operating pressure depends on the pressure drop through the pebble bed and the hot spot T_{max}. The specific pressure drop

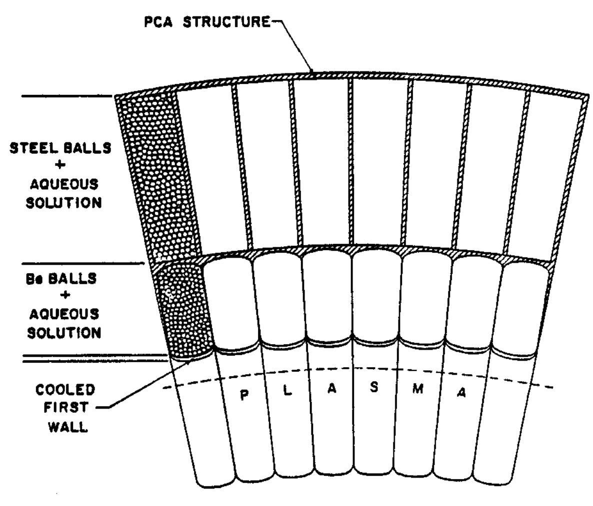

Fig. 2. Midplane cross section of an outboard shield segment.

through a pebble bed is obtained from well known empirical correlations. The optimum packing fraction for single size spherical pebbles is 62.5%.

To optimize for pebble size and temperature rise, we have generated a series of curves in which the operating coolant pressure is plotted as a function of temperature rise for a range of pebble diameters. Figure 3 shows these curves for the double wall design. The inlet coolant temperature was fixed at 40°C and the mass flow rate determined by fixing the temperature rise in the range 20-50°C. Once the velocity is determined, we calculate the heat transfer coefficients and the pressure drop for the longest coolant flow path through the pebbles of 4 m using the aqueous solution properties given in Table 2. From the surface and nuclear heating we determine the temperature drop across the film and obtain the hot spot temperature. To this we add a safety margin of 20°C and obtain the saturation pressure for the coolant at this temperature. The operating pressure is then determined by adding the maximum pressure drop to the coolant saturation pressure. The operating pressure for the single wall design is a factor of ~ 3 higher than for the double wall.

The 3 cm diameter spherical pebbles give the lowest design pressure. The hot spot occurs on the surface of the front row of pebbles instead of the surface of the wall with which they are in contact. Surface heat is dissipated in the coolant channel immediately behind the first wall where the flow is unimpeded, the heat transfer coefficient is very high,

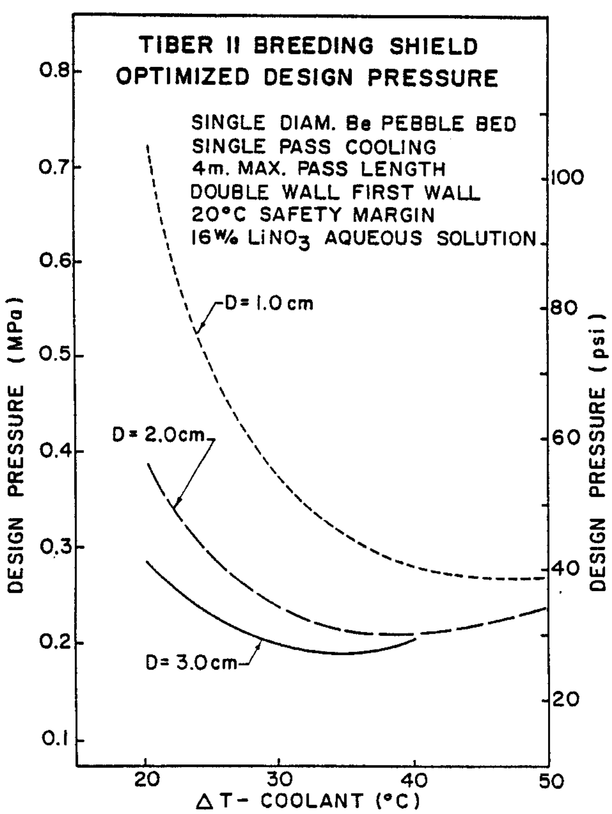

Fig. 3. Design pressure as a function of coolant temperature rise.

Table 2. Properties of 16 w/o $LiNO_3$ Aqueous Solution

Specific heat	C_p = 0.86 cal/g·°C
Density	ρ = 1.07 g/cm^3 (60°C)
Viscosity	μ = 5.5 x 10^{-3} g/cm·s
Thermal Conductivity	k = 1.5 x 10^{-3} cal/s·cm·°C

the coolant temperature rise is lower than in the pebble bed and the pressure drop is trivial. In this design, assuming an inlet coolant temperature of 40°C and a ΔT of 35°C (minimum for the 3 cm size pebbles), the hot spot on the Be pebble surface is 85.5°C giving a design pressure of only 0.19 MPa.

We have opted to use the double wall design for the base case. This is consistent with the original philosophy of operating the breeding shield at low temperature and pressure. Further, any uncertainty in the flow distribution through the pebbles will not affect the heat transfer on the first wall in the double wall design lending more confidence in the operation of the breeding shield. The thickness of the first wall facing the plasma is determined by material needed to withstand 1000 disruptions. Table 3 gives the thermal hydraulics parameters of the outboard breeding shield.

Table 3. Preliminary Parameters for Outboard Breeding Shield Using Single Pass Cooling

Be Pebble Diameter (cm)	3.0
Be Packing Fraction (%)	62.5
Total Thermal Power (MW)	176
Coolant ΔT (°C)	35
Inlet Coolant Temperature (°C)	40
Outlet Coolant Temperature (°C)	75
Mass Flow Rate (Kg/s)	1650
Superficial Velocity (m/s)	0.17
T_{max} in Breeding Shield (°C)	85.5
Safety Factor Used (°C)	20
Coolant Pressure Drop (psi/MPa)	10.5/0.072
Design Pressure (psi/MPa)	27.5/0.19

Tritium Breeding Potential

The shield configuration that provides adequate protection for the TF coils while maximizing tritium breeding is shown in Fig. 4. Although a significant contribution to the overall TBR is obtained from the outboard breeding shield, some additional breeding will occur in the aqueous solution used as coolant in different zones of the inboard and divertor region shield. In addition, since the neutron reflection capability of materials used in these zones varies, the impact on tritium breeding in the outboard region has to be considered. This was assessed by performing several toroidal cylindrical geometry calculations using the ONEDANT code. The detailed heterogeneous configuration of the inboard shield was included in the model. Calculations have been performed for the cases when Be or graphite tiles are used.

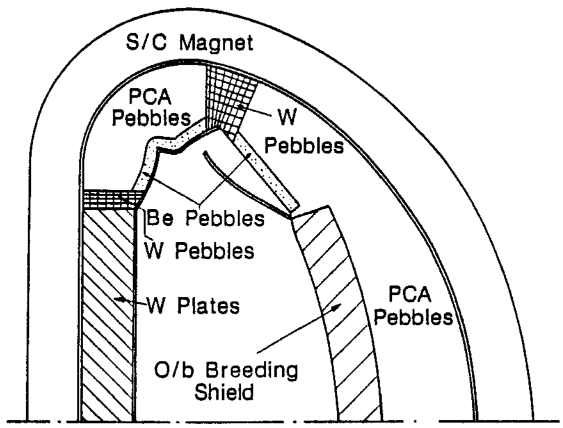

Fig. 4. TIBER-II shield configuration.

The results of the toroidal cylindrical calculations were used together with the coverage fractions of the different shield zones to estimate the overall TBR. Although several approximations are involved in this method, it is very useful in performing design iterations and parametric analysis. In the calculations three NBI ducts that are 0.8 m in diameter were assumed together with thirty-two vacuum pipes with a diameter of 0.2 m. The impact of test area on overall TBR was assessed. No credit was taken for tritium breeding in the test modules.

Different shield configurations that satisfy the magnet radiation protection requirement were considered to assess their tritium breeding potential. The first shield configuration option utilizes layered W shield in the inboard region and W pebbles in the divertor region. This option results in an overall TBR of less than ~ 0.8 for realistic test areas. The inboard shield was modified by replacing the front 2 cm of W by Be to enhance the TBR while still adequately protecting the magnet. This design option resulted in ~ 8% enhancement in overall TBR. Further shield design modifications were considered to enhance tritium breeding while still adequately protecting the magnets. This led to the baseline shield design configuration shown in Fig. 4. The achievable overall TBR for this configuration is given as a function of test area in Fig. 5.

In the baseline design a test area of 19.2 m^2 is utilized. The calculated overall TBR for this case is 1.015 when Be tiles and divertor plates are used and drops to 0.987 if graphite tiles and divertor plates are used. These values were determined by excluding tritium breeding in the test modules. Taking credit for tritium bred in the test modules with a conservative 0.5 local TBR, the overall TBR will be 1.086 with Be tiles and 1.058 with C tiles. This implies that tritium self-sufficiency is possible in TIBER-II. This conclusion was confirmed by the three-dimensional calculations that resulted in an overall TBR of 1.05 for the baseline design with graphite tiles. The results also indicate that the effect of using graphite instead of Be in the tiles and divertor plates will result in only a small drop in the overall TBR of ~ 2.7%. Hence the choice of armor material should be based on other non-neutronics considerations.

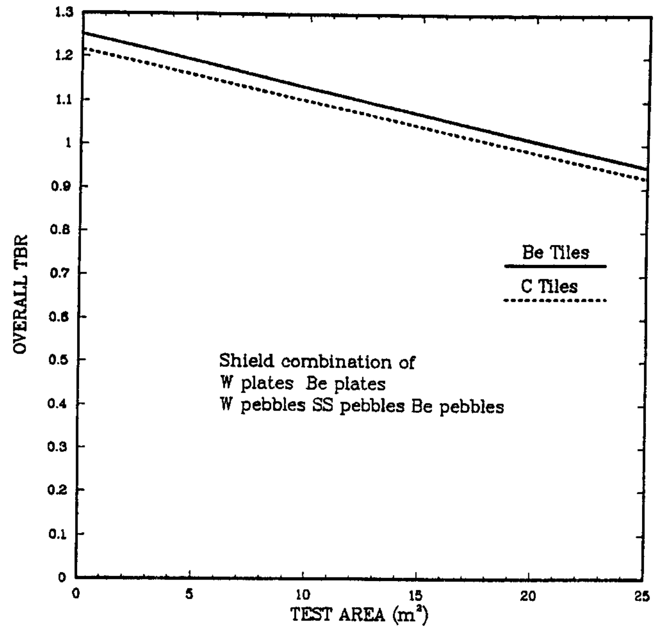

Fig. 5. Effect of test area on overall TBR.

Conclusions

A simple low risk outboard breeding shield was designed for TIBER-II. The shield utilizes aqueous solution coolant in which $LiNO_3$ is dissolved. Single size beryllium and steel pebbles are used in the shield resulting in a small pressure drop of 0.07 MPa. A double first wall is used to insure uniform cooling and minimize pressure. The design pressure is 0.19 MPa and the coolant outlet temperature is 75°C. Such a design allows for achieving tritium self-sufficiency in TIBER-II while adequately protecting the TF coils.

Acknowledgment

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