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THE TECHNOLOGY TEST REACTOR TIBER-II

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ABSTRACT

The breeding shield design for the compact technology test reactor TIBER-II is described. The shield allows for tritium self-sufficiency by utilizing a beryllium multiplier zone in the outboard shield and an aqueous solution coolant in which $LiNO_3$ is dissolved. This low temperature and low pressure design yields an overall tritium breeding ratio of 1.05 and provides adequate protection for the TF coils. The design features of this breeding shield render it an attractive low risk candidate for ITER.

1. INTRODUCTION

TIBER-II [1] is a compact tokamak technology test reactor developed as a candidate for the International Thermonuclear Experimental Reactor (ITER). The reactor has a major radius of 3 m and an aspect ratio of 3.6 and produces 314 MW of fusion power yielding an average neutron wall loading of 1.14 MW/m^2. The conceptual design and analysis of the shielding system was led by the University of Wisconsin with contributing effort by personnel from Lawrence Livermore National Laboratory (LLNL), Rensselaer Polytechnic Institute (RPI), Grumman Corporation, the Canadian Fusion Fuels Technology Project (CFFTP), Idaho National Engineering Lab (INEL) and University of California-Los Angeles (UCLA).

The main objective of the shield design is to provide adequate protection for the TF superconducting coils. This goal can be achieved by using a 48 cm thick high performance tungsten based inboard shield [2]. The space available for the outboard shield is 123 cm at the reactor midplane. Thus a lower cost steel based shield is used. While tritium self-sufficiency is not a requirement, partial tritium breeding is essential in TIBER-II if tritium is available only from civilian sources. In addition, tritium self-sufficiency is desirable due to the large cost of tritium. An important secondary objective is to modify the shield design to breed as much tritium as possible without compromising magnet protection and design simplicity, increasing machine size or encroaching upon the test modules. An attractive concept that meets these requirements is the aqueous self-cooled blanket concept [3], where a soluble lithium compound is added to the water coolant. This design has the flexibility of adding the lithium salt only when tritium breeding is required. It preserves the attractive features of low pressure, low temperature water cooled shields and allows for inboard breeding. Using this concept in

TIBER-II was also inspired by Ontario Hydro's demonstrated capabilities of removing tritium from water at acceptable concentrations and cost. The tritium processing equipment direct cost for TIBER-II is estimated to be \$25M. Tritium self-sufficiency is possible using a front beryllium zone in the outboard shield resulting in a lifetime cost savings of ~ \$450M [4].

In this paper, detailed nuclear analysis for the outboard breeding shield is given as well as the three-dimensional neutronics analysis for TIBER-II. The inboard and divertor zone shield neutronics analysis [2] and detailed shield mechanical design [5] are presented in separate papers in these proceedings. The overall shield design parameters will also be given here.

2. NUCLEAR ANALYSIS FOR OUTBOARD SHIELD

Many Li compounds have high solubility limits in water allowing for significant tritium breeding. The major considerations in the choice of Li salt are compatibility with shield materials, tritium breeding and shielding performance, induced radioactivity, salt radiolysis, and impact on tritium recovery and containment. Our results show no significant difference in breeding potential when high solubility salts are used with concentrations close to their solubility limits. Lower magnet damage is obtained with higher solubility salts due to the enhanced shielding effect of Li. Lithium salts with potentially large activation concerns such as LiI, LiBr, LiCl, and $LiClO_4$ and salts with low solubility limits such as $LiHCO_3$ and Li_2CO_3 were not considered. This led to identifying lithium nitrate and lithium hydroxide as the primary candidates. $LiNO_3$ is used in the baseline design based on favorable preliminary corrosion experimental results [6].

Different neutron multiplying materials including beryllium, lead and Zircalloy pebbles were considered. While a high packing fraction of neutron multiplier required for enhancing tritium breeding can be achieved by utilizing different size pebbles, a design with single diameter pebbles is preferred in order to minimize the pressure drop. One-dimensional neutronics calculations were performed for four outboard breeding shield designs. These include using two size Be pebbles, single size Be pebbles, single size Pb pebbles and Zircalloy structure with single size Zircalloy pebbles. The results for a 40 cm thick multiplier zone given in Table 1 show that using single size pebbles instead of two size pebbles reduces the tritium breeding ratio (TBR). While the smaller packing fraction of solids results in less breeding in the multiplier zone, more breeding occurs in the steel zone due to increased Li content and smaller amount of steel. Therefore, the largest drop in TBR (~ 10%) occurs for the 60 cm thick multiplier zone with the drop decreasing for thinner multiplier zones. Using single size pebbles, the shielding capability of the multiplier zone improves due to the increased amount of Li and decreased amount of multiplier, while the shielding capability of the steel zone decreases. Therefore, the largest magnet radiation damage enhancement (~ 50%) occurs when no multiplier zone is used with only a slight increase for thick multiplier zones. These results indicate that using single diameter pebbles, which has attractive thermal-hydraulics aspects, will have only a small impact on the shield nuclear performance.

Replacing Be by lead reduces TBR by 18% with magnet radiation effects 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

Table 1. Impact of Outboard Shield Design on Nuclear Parameters for a 40 cm Thick Multiplier Zone

Design Option	Two size Be pebbles	Single size Be pebbles	Single size Pb pebbles	Zircalloy structure with single size Zr pebbles
Local TBR	1.97	1.79	1.46	1.14
Local M	1.52	1.46	1.19	1.23
Peak end-of-life neutron fluence in coil (E > 0.1 MeV) (n/cm^2)	4.5×10^{15}	5.2×10^{15}	3.9×10^{15}	7.1×10^{15}
Peak end-of-life insulator dose (rad)	4.8×10^6	6.1×10^6	4.6×10^6	8.4×10^6
Peak winding pack power density (mW/cm^3)	1.7×10^{-3}	2.0×10^{-3}	1.5×10^{-3}	2.6×10^{-3}

Table 2. Nuclear Performance 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
Power density (W/cm^3)	
Front of first wall	13.70
Front of Be zone	15.76
Front of SS zone	0.81
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 @ 2.5 FPY (n/cm^2)	8.53×10^{15}
Peak winding pack power density (mW/cm^3)	4.16×10^{-3}

production, lead was not considered further. Replacing Be by Zr reduces TBR by 36% with magnet radiation effects increasing by ~ 36%. Based on these results, single size Be pebbles are 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 [7]. Calculations have been performed for the baseline shield design with the 1 cm thick first wall modeled separately. The peak outboard neutron wall loading of 1.92 MW/m^2 was used with total DT reactor operation time of 2.5 full power years (FPY). 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 a 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. The effect of Be zone thickness on TBR is shown in Fig. 1. 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 2.

3. IMPACT OF INBOARD AND DIVERTOR ZONE SHIELD ON OVERALL TBR

While most of the overall TBR is obtained from the outboard breeding shield, additional breeding will occur in the aqueous solution used in other zones. In addition, the impact of neutron reflection from these zones on outboard tritium breeding has to be considered. This can be properly

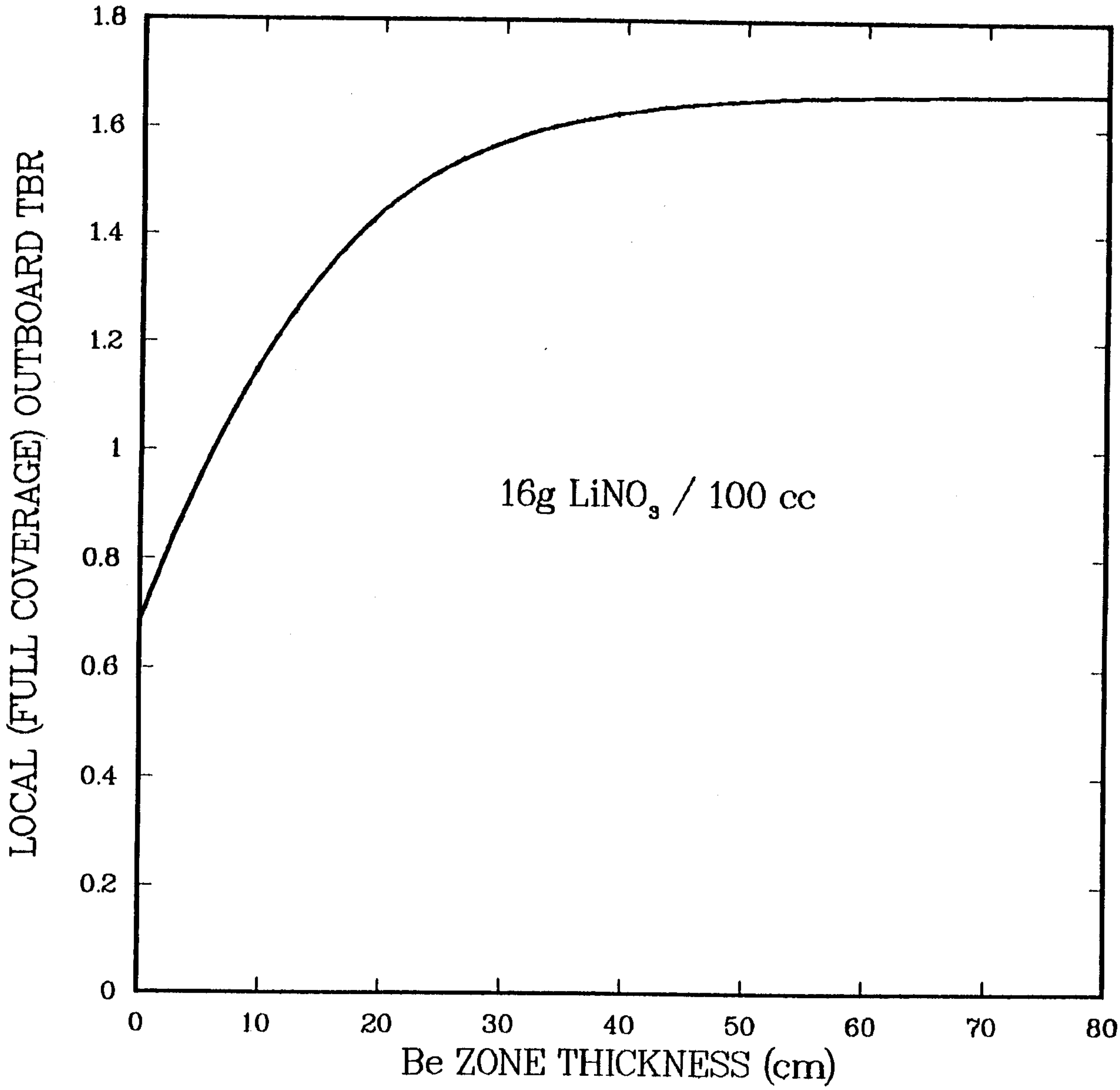

Fig. 1. Effect of Be zone thickness on local TBR in outboard shield.

assessed by performing neutron transport calculations using a toroidal cylindrical model. Several toroidal cylindrical geometry calculations have been performed using the one-dimensional discrete ordinates code ONEDANT [8] and ENDF/B-V cross-section data. Different materials were used in the inboard region of the toroidal cylindrical model to simulate the effect of the different inboard and divertor shield zones. In all cases a front 0.8 cm thick zone of Be or graphite was used to represent the armor or divertor plate. This zone is followed by a 0.4 cm thick PCA first wall and a 0.8 cm thick aqueous solution coolant zone.

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. The penetrations, including the NBI ducts and the thirty-two vacuum pipes, were considered. The impact of test area on overall TBR was assessed with no credit taken for tritium breeding in the test modules. Different shield configurations that satisfy the magnet radiation protection requirement were considered. With the option utilizing layered W shield in the inboard region and W pebbles in the divertor region, an overall TBR of only less than ~ 0.8 is achievable for realistic test areas ($\gtrsim$ 15 m^2). Replacing the front 2 cm of W by Be 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. 2. The calculated overall TBR for this case is 1.015 when Be tiles are used with 19.2 m^2 test area. The TBR drops to 0.987 if graphite tiles are used. Taking credit for tritium bred in the test modules with a conservative 0.5 local TBR, the overall TBR will be 1.086 with

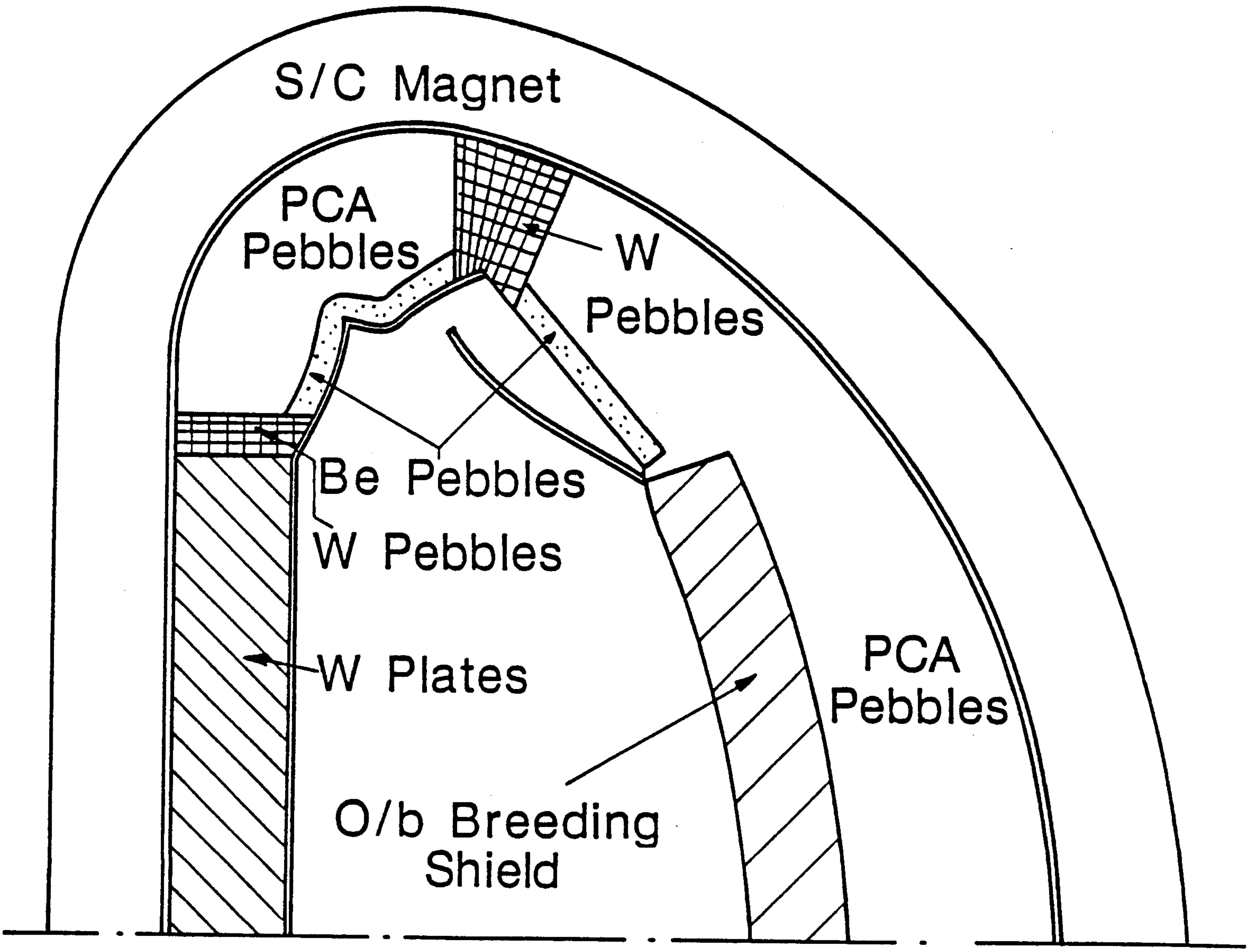

Fig. 2. Baseline Shield Configuration.

Be tiles and 1.058 with C tiles. It is clear also 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.

4. THREE-DIMENSIONAL NEUTRONICS ANALYSIS

The three-dimensional (3-D) calculations have been performed using the continuous energy coupled neutron-gamma Monte Carlo code MCNP [9] and cross section data based on the ENDF/B-V evaluation. Because of symmetry only 1/64 of the reactor was modeled with reflecting boundaries. The model includes one-fourth of one of the sixteen TF coils and one half of one of the 32 vacuum pipes used in the divertor region. A vertical cross section of the geometrical model used is given in Fig. 3 and a horizontal cross section at Z = 2.5 m that shows the divertor region vacuum pipe is given in Fig. 4. The first wall, tiles and divertor plates were modeled separately. The neutron source was sampled from a D-shaped plasma zone with the baseline spatial source distribution. The calculations were performed for the two cases of Be and graphite tiles and divertor plates. Several variance reduction techniques were utilized to improve the accuracy of the calculations. Fifty thousand histories were used in each calculation yielding statistical uncertainties of less than 0.5% in the calculated overall TBR and energy multiplication.

The overall TBR values determined from the 3-D results are compared to the values estimated from 1-D results and coverage fractions in Table 3. It is interesting to note that the estimates for overall TBR based on the one-dimensional analysis are only slightly lower than those obtained from the three-dimensional calculations. This demonstrates that the method used to estimate the overall TBR from one-dimensional results is a powerful tool that

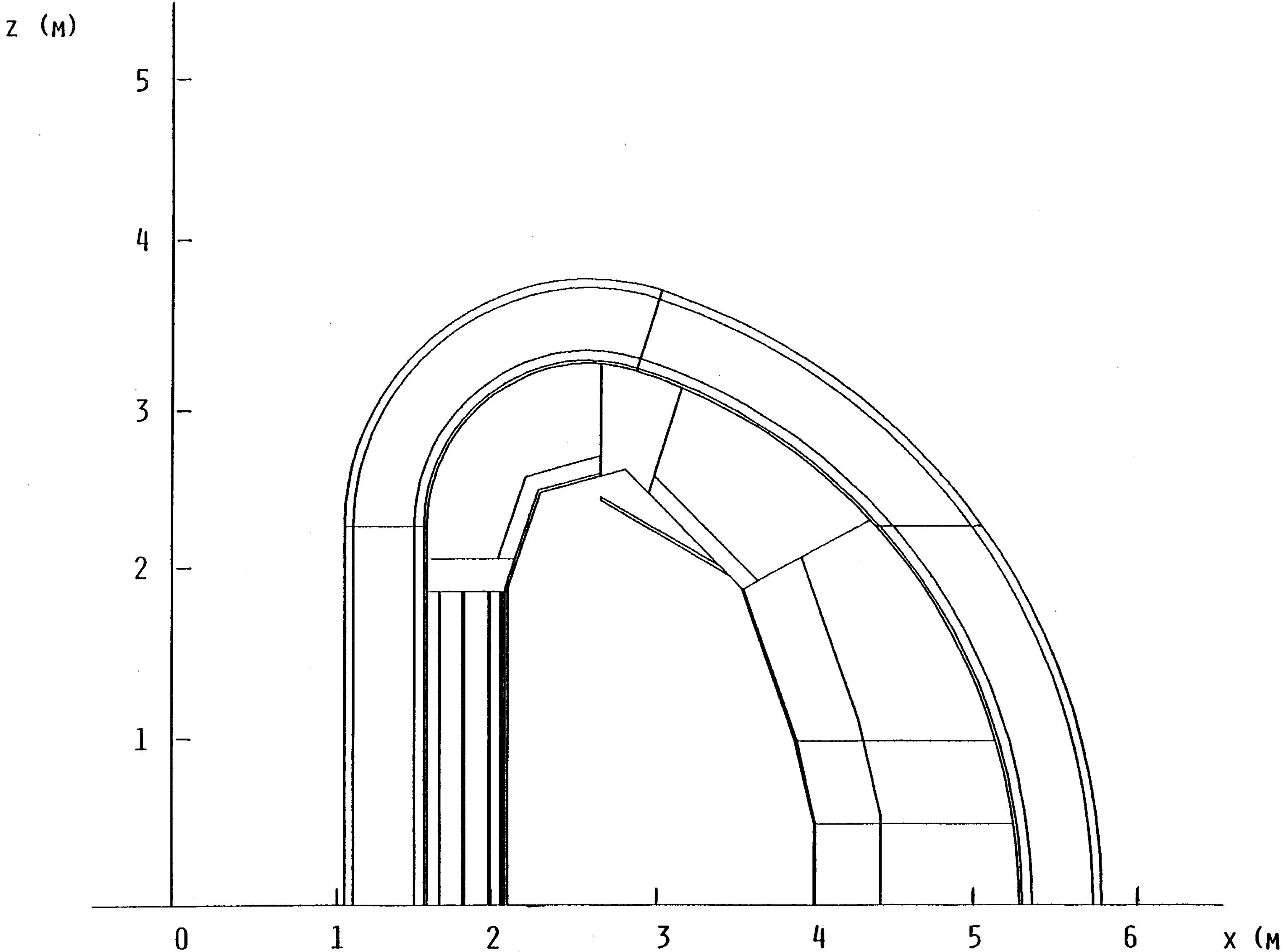

Fig. 3. Vertical cross section of the geometrical model used in the three-dimensional calculations.

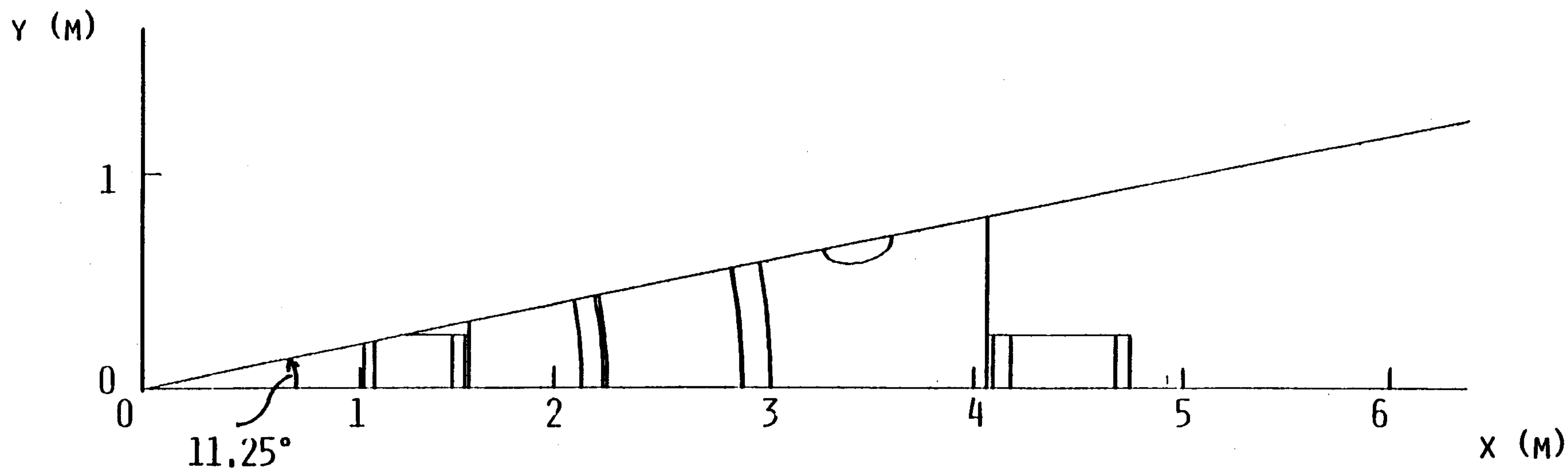

Fig. 4. Horizontal cross section at $Z = 2.5$ m.

Table 3. Comparison Between Values of Overall TBR Obtained from 3-D Calculations and Those Estimated From 1-D Results

Test Area	From 3-D Results		Estimate from 1-D Results		% Difference	
(m^2)	Be Tiles	C Tiles	Be Tiles	C Tiles	Be Tiles	C Tiles
0	1.254	1.230	1.251	1.216	-0.2	-1.1
5	1.199	1.178	1.189	1.156	-0.8	-1.9
19.2	1.053	1.036	1.015	0.987	-3.6	-4.7
25	0.991	0.976	0.943	0.917	-4.8	-6.0

Table 4. TIBER-II Nuclear Parameters Determined from 3-D Calculations

			Nuclear Heating in TF Coils (kW)	
Region	TBR	Nuclear Thermal Power (MW)	Winding Pack	Case
Outboard	0.70	152	0.1	0.6
Divertor	0.27	90	10.8	4.7
Inboard	0.08	62	20.8	18.1
Total	1.05 (±0.4%)	304 (±0.5%)	31.7 (±11%)	23.4 (±14%)

can be utilized in early stages of the design where several iterations and parametric analyses are needed. The TBR and shield nuclear thermal power results for the baseline design that utilizes graphite tiles and two rectangular NBI ducts with 19.2 m^2 devoted for test modules are given in Table 4. It is clear that tritium self-sufficiency is achievable in the TIBER-II design. A larger tritium breeding margin can be obtained if credit is taken for tritium bred in the test modules. The results for the total nuclear heating in the winding pack and case of the TF coils are given in Table 4. The total nuclear heating in the 16 TF coils is 55 kW which is below the design limit of 72 kW required to maintain a stability margin of at least 300 mJ/cm^3.

5. SUMMARY AND CONCLUSIONS

Table 5 summarizes the major features and performance parameters for the breeding shield design of TIBER-II. Details of the activation analysis are given elsewhere [10]. The inboard and outboard breeding shields were analyzed with respect to LOCA and LOFA [11]. In the case of LOFA, natural coolant convection will exist everywhere to dissipate the decay heat. In the case of LOCA, the outboard shield possesses adequate thermal inertia as to be able to absorb the decay heat adiabatically. Establishing natural circulation of air through the inboard coolant channels is capable of maintaining the maximum inboard shield temperature below 510°C following a LOCA.

It is concluded that all requirements for nuclear performance of TIBER-II can be met using the breeding shield design presented here. Tritium self-sufficiency can be achieved with a total reactor lifetime savings of ~ \$450M. This aqueous breeding shield is a simple, flexible, low risk, low temperature, and low pressure design which is considered as an attractive candidate for ITER.

Table 5. Major Parameters for the Breeding Shield of TIBER-II

Coolant/breeder	Aqueous solution with 16 g $LiNO_3$/100 cm^3
Structural material	PCA
Neutron multiplier	Be
Inboard shielding material	W
Overall TBR	1.05
Shield thermal power	304 MW
Total nuclear heating in TF coils	55 kW
Peak end-of-life neutron fluence ($E > 0.1$ MeV) in winding pack	9.3×10^{18} n/cm^2
Peak end-of-life insulator dose	1.25×10^{10} rad
Design pressure	0.2 MPa
Inlet coolant temperature	40°C
Outlet coolant temperature	75°C
Maximum Be ball surface temperature	86°C
Maximum structure surface temperature	80°C
Tritium concentration in coolant	10 Ci/ℓ
Tritium recovery	continuous
Total activity at shutdown after 2.5 FPY operation	800 MCi
Total decay heat at shutdown after 2.5 FPY operation	2.7 MW
Waste disposal rating	
inboard	0.152 (Class C)
outboard	0.042 (Class C)

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