Overview of the US-ITER Magnet Shield: Concept and Problems

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L.A. El-Guebaly

Fusion Technology Institute
University of Wisconsin
1500 Engineering Drive
Madison, WI 53706

http://fti.neep.wisc.edu

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OVERVIEW OF THE US-ITER MAGNET SHIELD: CONCEPT AND PROBLEMS

Laila A. El-Guebaly
Fusion Technology Institute
University of Wisconsin-Madison
1500 Johnson Drive
Madison, WI 53706
(608) 263-1623

ABSTRACT

The International Thermonuclear Experimental Reactor (ITER) is designed to operate in two phases; physics and technology. The prime function of the shield is to protect the TF magnets. The predominant radiation limits are the nuclear heat load to the magnet and the end-of-life dose to the electrical insulator. These limits are specified by the magnet designers as 65 kW and 5×10^9 rads. Detailed shielding analysis has been performed and necessary machine modifications have been proposed during the conceptual design phase (1987-1990) in order to meet the magnet radiation limits. The shield is designed to satisfy the neutronics, thermal hydraulics, and mechanical design requirements. The reference shield consists of 316 SS structure and water coolant. A 5 cm thick back layer with special materials, such as W, Pb, and B_4C, is considered outside the vacuum vessel to reduce the magnet damage. Two regions with critical shielding space are identified in ITER, the inboard and divertor regions. This paper presents the various options for the shield design based on a variety of shielding materials and summarizes the different analyses carried out to guide the shield design.

INTRODUCTION

This paper presents the U.S. contribution to the shield design of ITER. The design calls for a 6 m major radius and 2.15 m minor radius. Figure 1 shows a cross section view through the upper half of the machine. The design of the shield has been optimized by a series of neutronics and thermal hydraulics calculations, subject to constraints such as the interface temperatures, thermal stresses, and radiation effects at the magnet. Detailed neutronics analyses were performed for regions with critical shielding space using one-, two-, and three-dimensional models. The 1-D method was heavily utilized to optimize the shield and to determine the peak radiation effects in the different components of the reactor. Peaking in magnet damage resulting from both assembly gaps and toroidal changes in configuration were computed using a 2-D code. Due to the complexity of the shield configuration, especially in the divertor region, 3-D models were employed to accurately determine the damage level at the magnet. The focus of this paper has been on the 1- and 2-D neutronics analyses. The 3-D analysis,

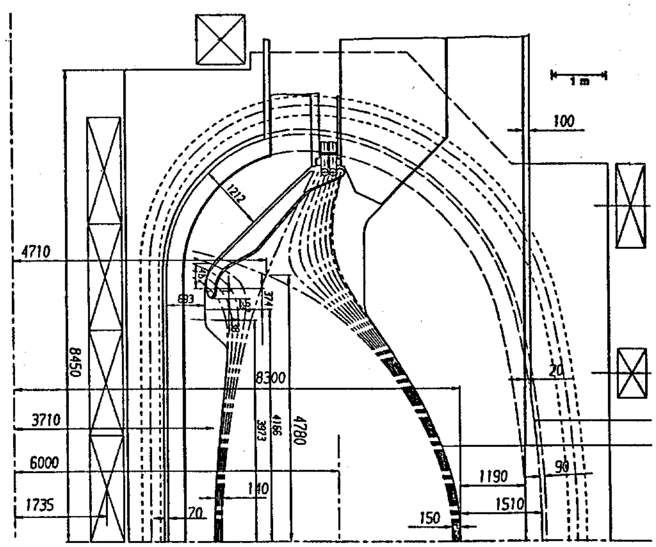

Figure 1. Vertical cross section of ITER.

mechanical design, and thermal hydraulics analysis are presented in companion papers[1-3] in these proceedings.

DESIGN REQUIREMENTS AND PARAMETERS

The ITER reactor is designed to accommodate two phases of operation and to achieve a fluence goal of ~ 3 MW·y/m^2. During the 15 year life of the machine, ~3.8 full power years (FPY) of operation are expected; 0.05 FPY in the physics phase and 3.7 FPY in the technology phase. The overall dimensions of the reactor are fixed in both phases and 1100 MW and 860 MW of fusion power are anticipated in the physics and technology phases, respectively. The shield is designed under a common set of design guidelines. For instance, the peak neutron wall loadings on the inboard and outboard are 0.88 and 1.2 MW/m^2, respectively, in the technology phase. Proper performance of the TF magnets is guaranteed if the radiation limits are met. These limits are 5×10^9 rads, 65 kW, 5 mW/cm^3, 10^{19} n/cm^2, and 6×10^{-3} dpa for the peak dose to the insulator, total nuclear heating in the magnet, peak nuclear heating in the winding pack, peak fast neutron fluence ($E_n > 0.1$ MeV) to the Nb_3Sn conductor, and peak

Table 1. Recommended Safety Factors for ITER Shielding Analysis

Responses	1-D Analysis Local	1-D Analysis Integral	3-D Analysis Local	3-D Analysis Integral
Correction factors for:				
Assembly gaps	1.7	1.2	-*	-*
Modeling	1.3	1.3	1.1	1.1
Uncertainties in Xn data	1.4	1.3	1.4	1.3
Safety factors	3	2	1.5	1.4

*Gaps included in 3-D models

dpa in the Cu stabilizer, respectively. Another limit that needs to be satisfied is the He production in the vacuum vessel (V.V.). It should not exceed 0.1 appm for reliable rewelding of the different V.V. components.

Before comparing the calculated results with these limits, safety factors should be used to correct the radiation damage obtained from the 1-D and 3-D analyses. In general, these factors account for the presence of the assembly gaps and for the uncertainties in the nuclear data and modeling. Because of the latter, different safety factors have to be applied to the 1-D and 3-D results. The safety factors consist of a set of correction factors which are design dependent. They depend on the type of materials used in the blanket/shield, the characteristics of the assembly gaps between the blanket/shield modules, and the uncertainties in cross section data evaluation. They vary slightly with the response functions and differ for local and integral quantities. Table 1 lists the correction factors for both 1-D and 3-D analyses. The recommended safety factors for the ITER shield design are 3 and 2 for the local and integrated 1-D results and 1.5 and 1.4 for the local and integrated 3-D results, respectively.

BULK SHIELD DESIGN

Inboard Shield Design

Optimization studies were performed to design an efficient shield to protect the TF magnets. Several options for the shield were examined and the shielding capability of many materials was assessed. Besides 316 SS and H_2O, these materials include B_4C, Pb, W, boron steel (B-SS), and borated water (B-H_2O). The effect of boron enrichment on both magnet damage and shield cost was also evaluated. In addition, the optimum coolant content and channel arrangement within the various layers of the shield were determined. The neutronics analysis was performed using the 1-D code ONEDANT[4] and the cross section data based on the ENDF/B-V evaluation.[5,6] The 46 neutron and 21 gamma energy group structure and the P_3-S_8 approximation were used. The different components were modeled as infinite cylinders around the machine axis, permitting the representation of both inboard (i/b) and outboard (o/b) sides.

The highest radiation damage in the inner legs occurs at the midplane where the space available for the i/b blanket/shield is constrained to 84 cm. The predominant magnet radiation limits are the end-of-life dose to the insulator and the total heat load to the magnet. The first wall follows the plasma contour and the i/b blanket/shield increases in thickness reaching 111 cm at the top/bottom. In the physics phase, the front 2 cm is composed of C tiles to protect the first wall (FW) while in the technology phase only 0.05 cm of W coating is needed. The 1.5 cm thick first wall is an integral part of the blanket/shield and consists of water cooled SS layers. The U.S. solid breeder i/b blanket is 11.6 cm thick at the midplane and gradually increases in thickness toward the top/bottom. The space between the blanket and the V.V. is occupied by the shield. The V.V. is 25 cm thick and has 10 cm steel ribs for blanket/shield support, 4 cm thick resistive elements (R.E.) and several coolant channels. Outside the V.V. (on the magnet side), there is a 5 cm thick back layer where one or a combination of special materials (such as B_4C, Pb, W) can be used to reduce the magnet damage. The inner coil case of the TF magnet varies poloidally in thickness. For the inner legs, the thickness of the inner coil case changes toroidally from 6 cm at the middle to 2.6 cm at the corners of the winding pack. The 30.6 cm thick winding pack contains 31.6 v/o SS, 26 v/o Cu, 2.5 v/o Nb_3Sn, 6.4 v/o bronze, 0.6 v/o V, 21.1 v/o He, and 11.8 v/o R-glass-fiber-filled epoxy insulator.

The reference shield is composed of 316 SS structure and H_2O coolant. Alternative options such as B-SS shield, B-H_2O coolant, and ^{10}B enriched borated materials were also analyzed.[7] Upon changing the coolant content in the shield/V.V., the magnet damage was found to minimize at 20, and 30% by volume for the SS shield and V.V., respectively. Our results show that about 20% reduction in magnet damage can be achieved by using B-SS shield (with 15 v/o 316 SS structure) in the bulk shield or B-H_2O coolant in the V.V. An additional 20% reduction is obtained by enriching the boron to 90% ^{10}B in borated materials (B_4C, B-H_2O, and B-SS). As expected, each option has some problems. The 15 v/o 316 SS structure of the B-SS shield is not sufficient to carry the electromagnetic load to the V.V. during plasma disruption and a higher 316 SS content dilutes the effect of the B-SS shield. The B-H_2O is effective when used to cool the V.V. whereas no significant reduction in damage is obtained when B-H_2O is used in the shield. This option raised some concerns because the B-H_2O usually causes corrosion problems, and besides, a separate cooling system with tritium removal scheme is needed. Enriching the boron to 90% ^{10}B significantly increases the cost of the borated materials. For example, a kilogram of B_4C, B-SS, and boric acid costs $270, 35, and 300 for natural B and $3420, 420, and 1000 for 90% ^{10}B in B, respectively. This translates into a difference in cost of 4-200 M$ between natural and enriched borated materials, depending on the amount of material used in the shield.

Clearly, the simplest and most reliable shield is the water cooled SS shield. Accordingly, the shield is configured in alternating layers of SS and H_2O coolant channels. More coolant channels are placed at the front of the shield to warrant proper cooling of this high heat load zone. The location and thickness of the coolant channels

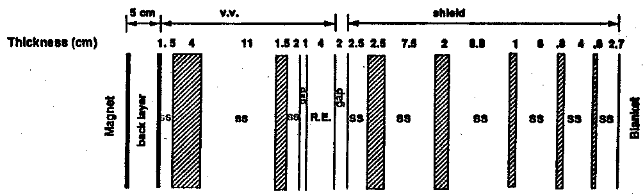

Figure 2. Layered configuration of the inboard shield/V.V. at the midplane.

have been optimized to reduce the magnet damage and the SS layers have been checked with respect to maximum temperature and thermal stresses.[3] The layered shield at the midplane consists of 2.7, 4, 6, 8.8, 7.5, and 2.5 cm thick SS layers spaced by 0.6, 0.8, 1, 2, and 2.5 cm wide coolant channels, as shown in Figure 2. Our 1-D analysis shows that for such a shield arrangement and for a pure SS back layer outside the V.V., the peak dose to the insulator and the total nuclear heating in the inner legs (z = -3.4 to 3.4 m) amount to 2.4×10^9 rads @ 3.8 FPY and ~8 kW, respectively. These results pertain to a cross section through the R.E. of the V.V. It should be mentioned that local values (such as dose, fluence, dpa, and appm) and integrated quantities (e.g. total nuclear heating) should be multiplied by safety factors of 3 and 2, respectively, before being compared to the design limits.

The pure SS back layer results in an excessive end-of-life dose to the insulator and heat load to the magnet (as will be shown later). Therefore, other materials should be incorporated in the back layer to reduce the magnet damage. The effect of using the special materials (B_4C with 80% density factor, Pb, and W with 90% density factor) in the back layer is illustrated in Figure 3. In the analysis, 20% 316 SS structure is considered for canning the special materials. The left axis of Figure 3 corresponds to the pure SS case. Upon replacing the SS by the special materials, the dose and heating minimize at thicknesses of 1 and 4 cm for B_4C and W, respectively. Replacing SS by Pb increases the dose and the heating minimizes at 2.5 cm Pb. To reduce the insulator dose, the W is the best followed by B_4C. The W is also the best for reducing the magnet heating followed by Pb and then B_4C. It was found that two consecutive layers of B_4C and Pb are more effective in reducing the damage than using either one separately. Figure 4 shows that 1 cm B_4C followed by 3 cm Pb is the optimum combination. No significant reduction in damage is obtained when W is combined with either B_4C or Pb. It is clear that W and Pb/B_4C are the most attractive options for the back layer. Notice that replacing Pb/B_4C by W in the back layer reduces magnet damage by ~20%. Because W is expensive, it can be used in limited places where the shielding space is critical and Pb/B_4C can be used elsewhere. It should be mentioned that the W and Pb are placed at the back of the shield where the neutron flux is low and no significant activation or decay heat hazards associated with them are anticipated. In addition, the loss-of-coolant accident analysis has indicated that there is no Pb melting under off-normal conditions.

The radiation effects in the inner legs for three back layer material options are listed in Table 2. The first option is for a pure SS back layer. Even with Pb/B_4C (option 2), the dose to the insulator is still high when the safety factor of 3 is included. An acceptable dose is obtained in option 3 when W is employed in the high damage zone which ranges from z = -0.5 to z = 0.5 m. Thereafter, the thicker blanket/shield and the lower neutron wall loading result in a lower damage and the Pb/B_4C can be utilized in the back

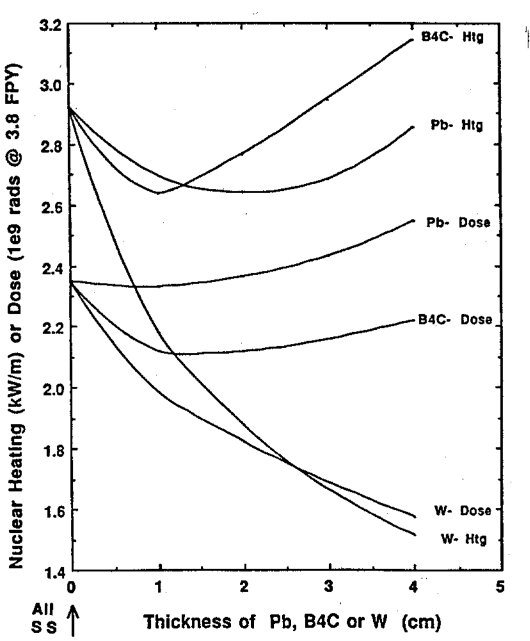

Figure 3. Effect of the special materials used in the back layer.

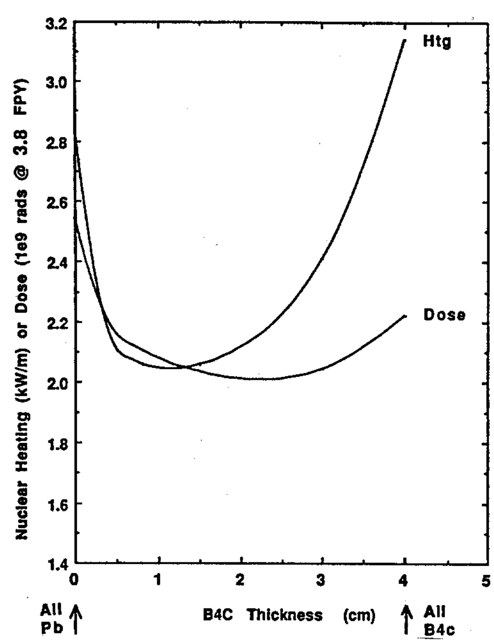

Figure 4. Effect of using a combination of B_4C and Pb in the back layer.

Table 2. Radiation Effects at Inner Legs of the TF Magnets

Option	1	2	3
Back Layer	SS	Pb/B_4C	Pb/B_4C & W†
Peak Dose to Insulator (10^9 rads @ 3.8 FPY)	2.35	2.10	1.58
Nuclear Heating in Inner Legs‡ (kW):			
Technology Phase	7.1	5.0	4.5
Physics Phase	7.8	5.5	5.0
Nuclear Heating per Unit Length* (kW/m):			
Winding Pack	1.35	1.21	0.95
Coil Case	1.57	0.85	0.56
Total	2.92	2.06	1.51
Peak Nuclear Heating* (mW/cm^3):			
Winding Pack	0.78	0.57	0.43
Coil Case	2.41	0.95	0.58
Peak Fast n Fluence to Nb_3Sn (10^{18} n/cm^2 @ 3.8 FPY)	2.2	2.4	1.9
Peak dpa in Cu Stabilizer (10^{-3} dpa @ 3.8 FPY)	1.1	1.2	0.9
He Production in V.V. (appm @ 3.8 FPY)	0.5	0.5	0.5

†W is used at the midplane from z = -0.5 to 0.5 m
‡z = -3.4 to 3.4 m
*For technology phase @ midplane

layer. All other magnet radiation effects are below the limits. The He production in the V.V. is excessive. Different schemes other than welding are needed for the V.V. assembly especially in the high damage zone from z = -2.5 to 2.5 m.

Another critical area in the inner legs occurs behind the shield recess which is between the upper/lower end of the i/b blanket and the inner end of the divertor plates (z = 3.8 to 4.7 m). The shield/V.V. therein is limited to ~70 cm in thickness. According to the 3-D Monte Carlo calculations,[1] the neutron wall loading in the technology phase ranges between 0.086 and 0.17 MW/m^2 in this region. Our results show that for the SS/H_2O shield and Pb/B_4C back layer, the peak dose to the insulator is 1.2×10^9 rads @ 3.8 FPY. The nuclear heating deposited at both the upper and lower parts of the inner legs behind the shield recess totals to 2.1 and 1.6 kW in the physics and technology phases, respectively. The peak nuclear heating in the winding pack, fast neutron fluence and Cu dpa are 0.3 mW/cm^3, 1.3×10^{18} n/cm^2 @ 3.8 FPY and 7×10^{-4} dpa @ 3.8 FPY, respectively. Hence, all magnet radiation effects are below the design limits in these regions. On the other hand, the He production in the V.V. exceeds the limit and amounts to 0.3 appm at the end-of-life.

The mechanical design of the i/b blanket/shield/V.V. calls for a wide variation in material arrangement within a single module, as shown in Fig. 5.[2] This toroidal variation in composition affects the damage level at the magnet. Furthermore, the thinning in the coil case of the inner legs

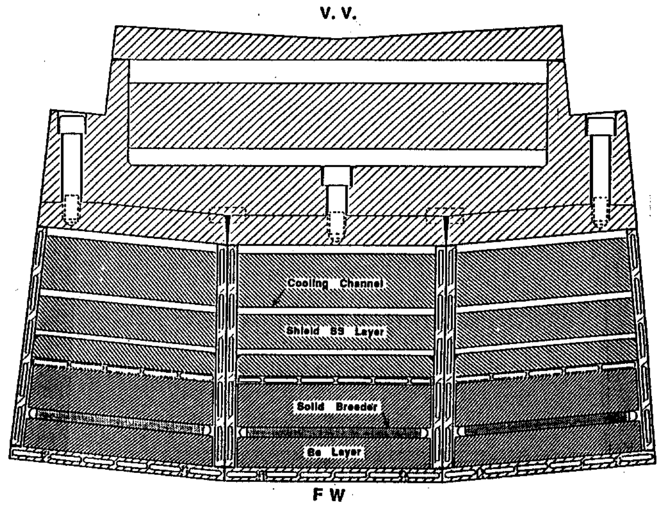

Figure 5. Mechanical design of the inboard blanket/shield.

creates hot spots at the corners of the winding pack. Other hot spots occur at the middle and corner of the winding pack due to the presence of the assembly gaps between blanket/shield modules. To quantify these effects, the V.V. and the i/b blanket/shield arrangement of Figure 5 was modeled for the 2-D code TWODANT. The variation in coil case thickness and the 2 cm wide assembly gaps were included in the model. The calculations were performed in x-y geometry using the P_3-S_{16} approximation. The peaking in damage is given in Table 3 at the middle and corner of the winding pack. The peaking factor is defined as the ratio of the 2-D to the 1-D values for the damage at the magnet. As noticed, the peaking factors differ with the response function. The enhancement in damage at the corners of the winding pack stems from both the presence of the assembly gap and the relatively thin coil case (2.6 cm compared to 6 cm at the middle). A possible solution to alleviate this problem is to design a uniformly thick coil case for the inner legs. The increase in damage due to the assembly gaps ranges between 1.63 and 1.85 for the local responses.

Table 3. Peaking Factors in Winding Packs of Inner Legs

	Middle of Winding Pack	Corner of Winding Pack
Peak dose to insulator	1.71	2.09
Peak nuclear heating	1.85	2.24
Peak fast neutron fluence	1.75	1.92
Peak dpa in Cu stabilizer	1.63	1.99

<u>Divertor Shield</u>

The most critical parts of the TF magnets are behind the outer end of the divertor plates (DP). The damage is high over a poloidal extent of ~70 cm at each of the top and bottom. The tilted divertor plates have limited the space available for the shield/V.V. behind the outer end of the DP to 47-56 cm. A vertical cross section through the lower divertor is shown in Figure 6. More shield is available when proceeding from the outer end of the DP towards the

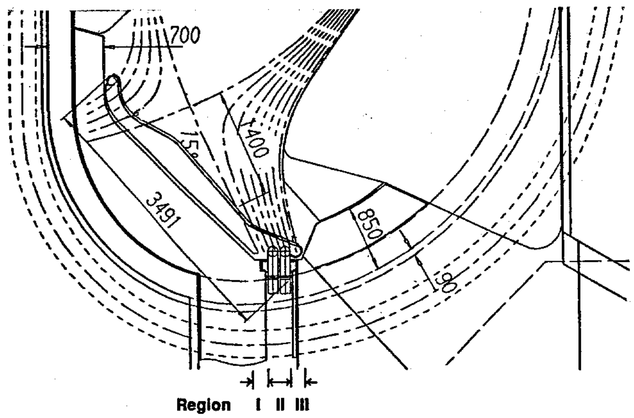

Figure 6. Detailed configuration of the divertor region.

i/b and o/b sides. In the technology phase, the neutron wall loading, determined from the 3-D calculations, ranges between 0.27 and 0.46 MW/m^2 over the outer end of the upper DP, with decreasing wall loading toward the outboard side. The outer end of the lower DP is subject to a lower wall loading (0.1 - 0.45 MW/m^2) due to the larger shadowing effect of the lower end of the o/b blanket/shield. The DP contain a total of 1.3 cm of materials (0.2 cm W, 0.6 cm Nb/Mo, and 0.5 cm H_2O). The divertor cooling tubes run toroidally after protruding vertically between coils. The magnets in the divertor region cover ~50% of the total toroidal space. The inner coil case is fairly thick (27-30 cm) and provides additional protection for the winding pack.

The radiation effects at the top and bottom portions of the TF coils have been calculated using 1-D models with poloidal cylindrical geometry around the plasma axis. For the neutronics analysis, the outer end of the DP is divided into three regions, as shown in Figure 6. An effective thickness of ~7 cm H_2O and 0.5 cm SS is taken into account for the toroidal divertor cooling tubes in region II. Tables 4 summarizes the radiation effects in the top divertor region for the three different back layer options. The He production in the V.V. is extremely high implying that V.V. rewelding should be avoided behind the outer end of the DP. Comparing the magnet damage in regions I, II, and III, one observes that region III has the highest damage due to the thinning in the shield. Region II has lower damage than region I due to the presence of the divertor cooling tubes. The lower divertor has relatively lower damage because of the lower wall loading, especially in region III. The calculated end-of-life dose to the insulator is excessive. Even with a W back layer, the dose, with a safety factor of 3, exceeds the limit by a factor of 2-3. The damage in region III of the upper divertor can be reduced to an acceptable level if the upper end of the side o/b blanket/shield module underneath the coil is extended inward up to the plasma boundary, similar to the lower end. This design modification reduces the neutron wall loading to 0.1 MW/m^2 in region III. It should be mentioned that the most recent divertor design calls for a fairly thick DP. In the physics and technology phases, the DP/structure are 5.5 and 11 cm thick, respectively. This design helps reduce the damage in the divertor region considerably and results in an acceptable radiation level at the magnet, even when Pb/B_4C are employed in the back layer.

Table 4. Radiation Effects in Parts of TF Magnets Behind the Upper Divertor

Option	1	2	3
Back Layer	SS	Pb/B_4C	W
Peak Dose to Insulator	3.5†	3.8	3.0
(10^9 rads @ 3.8 FPY)	2.5	2.7	2.1
	5.3	5.8	4.5
Nuclear Heating* (kW)	4.9	3.5	2.7
	6.3	4.6	3.5
	9.1	6.7	5.2
Total Nuclear Heating (kW):			
Technology Phase	20	15	11
Physics Phase	26	19	14
Peak Nuclear Heating* (mW/cm^3):			
Winding Pack	1.17	1.24	1.0
	0.85	0.9	0.7
	1.8	2.0	1.6
Coil Case	19.3	7.4	5.0
	15.2	5.9	4.0
	34.6	13.8	9.5
Peak Fast n Fluence to Nb_3Sn	4.1	4.5	3.3
(10^{18} n/cm^2 @ 3.8 FPY)	2.9	3.2	2.4
	6.1	6.8	5.0
Peak dpa in Cu Stabilizer	1.6	1.8	1.4
(10^{-3} dpa @ 3.8 FPY)	1.2	1.3	0.9
	2.4	2.7	2.0
He Production in V.V.	6.5	6.5	6.5
(appm @ 3.8 FPY)	5.4	5.4	5.4
	13.5	13.5	13.5

*In technology phase
†Region I, II, and III

Outboard Shield

There is ample space available for the o/b blanket/shield/V.V. This space is 151 cm thick at the midplane and increases toward the top/bottom. Therefore, there is no need for materials other than SS and H_2O to protect the outer legs of the TF magnets. Additional protection for the winding pack is provided by the 44 cm thick coil case. Our 1-D analysis shows that for the SS/H_2O shield (at 20 v/o H_2O) the peak dose to the insulator is 2×10^4 rads @ 3.8 FPY and the total nuclear heating in the outer legs amounts to 0.03 kW. All other radiation effects are ~5 orders of magnitude below the design limits. The end-of-life He production in the V.V. is as low as 10^{-4} appm and that assures reliable rewelding for the V.V. assembly. Although the o/b blanket/shield overprotects the magnets, it does not allow for hand-on-maintenance outside the V.V. Additional 40 cm of shield with a few centimeters of Pb are required at the midplane to permit manned access one day after shutdown.

ESTIMATE FOR THE TOTAL NUCLEAR HEATING IN THE TF MAGNETS

The total nuclear heating in the 16 TF magnets is calculated taking into account the following effects:

- the poloidal variation in the neutron wall loading
- the vertical variation in the blanket thickness and composition
- the poloidal variation in the shield and coil case thickness
- the toroidal variation in the V.V. thickness and composition
- the toroidal coverage of the magnets.

About 90% of the heating in the inner legs is generated in the 3 m high middle section. It is worth mentioning that the current design of a curved i/b FW that follows the plasma contour (with increased blanket/shield thickness toward the top/bottom) helps reduce the heating in the inner legs by a factor of ~2 compared to the previous same-thickness i/b blanket/shield design with a straight FW. Most of the nuclear heating is generated at the top/bottom parts of the magnet behind the outer end of the DP where the shield thickness is significantly reduced. Negligible heating is generated in the outer legs. A few kilowatts are anticipated to be deposited at the sides of the magnets as a result of the radiation streaming through the various penetrations.

Table 5 details the nuclear heating in the various regions in the physics and technology phases for the case of SS/H_2O shield. All values include the safety factor of 2 considered in the study. As expected, the heat load to the magnet in the technology phase is lower than that in the physics phase due to the smaller fusion power. Clearly, the SS back layer results in excessive heating in the magnet. The 65 kW heating limit is only met when W is partially employed in the back layer. It should be pointed out that the latest DP design results in about 40 and 23 kW of heating in the physics and technology phases, respectively, for Pb/B_4C back layer. The total heating in this case amounts to ~55 kW in the physics phase and ~35 kW in the technology phase. This heat load is well within the 65 kW heating limit for the TF magnets.

Table 5. Total Nuclear Heating (kW) in the 16 TF Magnets

Option	1	2	3
Back Layer	SS	Pb/B_4C	Pb/B_4C & W
Physics Phase			
Inboard	21	15	14
Divertor	88	64	48
Penetrations	4	4	4
Total	113	83	66
Technology Phase			
Inboard	19	13	12
Divertor	68	50	38
Penetrations	3	3	3
Total	90	66	53

CONCLUSIONS

Detailed shielding analyses have been performed and necessary reactor design modifications have been proposed in order to meet all radiation design limits. The detailed configuration of the various reactor components is taken into account in the analysis. The bulk shield consists of 316 SS structure and H_2O coolant and is designed to satisfy the neutronics, thermal hydraulics, and mechanical design requirements. To reduce the magnet damage, a combination of Pb and B_4C is used everywhere in the back layer outside the vacuum vessel except in regions with critical shielding space at the inboard midplane where W is employed. Based on one-dimensional analyses, all magnet radiation limits are satisfied in both phases of operation. About 55 and 35 kW of heat load to the TF magnets are expected in the physics and technology phases, respectively. For a 3 $MW \cdot y/m^2$ fluence, the end-of-life dose to the insulator is below the 5×10^9 rads limit everywhere in the TF magnets. The helium production in the vacuum vessel is excessive in the inboard and divertor regions and different schemes other than welding are needed for the vacuum vessel assembly.

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REFERENCES

1. M. SAWAN and L. EL-GUEBALY, "Three-Dimensional Neutronics Analysis for the U.S. Magnet Shield of ITER," these proceedings.

2. I. SVIATOSLAVSKY et al., "Mechanical Design and Fabrication of the U.S. Solid Breeder Blanket for ITER," these proceedings.

3. E. MOGAHED and I. SVIATOSLAVSKY, "Thermal Analysis of the Blanket and Shield Design for the U.S. ITER Design," these proceedings.

4. R. O'DELL et al., "User's Manual for ONEDANT: A Code Package for One-Dimensional, Diffusion Accelerated, Neutral Particle Transport," LA-9184-M, Los Alamos National Laboratory (1982).

5. R.W. ROUSSIN et al., "VITAMIN-E: A Coupled 174 Neutron, 38 Gamma Ray Multigroup Cross-Section Library for Deriving Application-Dependent Working Libraries for Radiation Transport Calculations," ORNL/DLC-113, Oak Ridge National Laboratory (1984).

6. Y. FARAWUKA, Y. GOHAR, and C. MAYNARD, "KAOS/LIB-V: A Library of Nuclear Response Functions Generated by KAOS-V Code from ENDF/B-V," ANL/FPP/TM-241, Argonne National Laboratory (1989).

7. L. EL-GUEBALY, "Shielding Design Options for the TF Magnets of ITER," Proceedings of IEEE 13th Symposium on Fusion Engineering, Knoxville, Tennessee (1989).