# Shielding Considerations for ITER: Current Status and Future Directions

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Presented at "New Horizons in Radiation Protection and Shielding", 26 April – 1 May 1992, Pasco WA.

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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. Shielding analysis for the reference conceptual design activity (CDA) design indicates that all magnet radiation effects are satisfied provided that a W back layer is utilized in the inboard region at midplane and the upper end of the outboard blanket is modified to provide additional shielding for the divertor region. The total magnet nuclear heating is 45 kW in the physics phase. The parts of the TF coils adjacent to the divertor vacuum pumping ducts are well protected against streaming radiation. For the US blanket design, all magnet radiation limits are met with the current 84 cm thick inboard region. For the other designs, the inboard blanket design should be modified, or 3-7 cm additional inboard shielding should be provided depending on the blanket type. The high aspect ratio (HARD) design results in $\sim$50% higher magnet damage compared to the CDA design.

INTRODUCTION

The ITER project has been moving steadily ahead since its inception in 1985. The first phase of the study, which is the Conceptual Design Activity (CDA),[1] has just ended and the ITER project is embarking on a new phase called the Engineering Design Activity (EDA). In this phase, the intent is to go deeper into the design developed in the CDA phase. The present ITER tokamak reactor design has a major radius of 6 m and an aspect ratio of 2.8. The reactor operates in two phases; a low fluence physics phase followed by a high fluence technology phase. Figure 1 shows a cross section view through the reactor.

The prime function of the bulk shield of ITER is to protect the toroidal field (TF) superconducting coils. The predominant magnet radiation limits are the total nuclear heating and the end-of-life insulator dose. Detailed shielding analysis has been performed and necessary reactor design modifications have been proposed during the CDA to satisfy the design limits.[2,3] The reference shield consists of 316 SS structure and water coolant and was designed to satisfy the neutronics, thermal hydraulics, and mechanical design requirements. Two regions with critical shielding space have been identified in ITER. These are the inboard and divertor regions. In this paper the results of the detailed neutronics analyses performed using one-, two-, and three-dimensional models are presented. 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 2-D calculations. Due to the geometrical complexity of the shield configuration, especially in the divertor region, 3-D models were employed to accurately determine the damage level at the magnet and the effect of radiation streaming through the divertor pumping ducts.

Different breeding blanket designs have been proposed during the CDA phase by the parties participating in ITER. The shielding performance of these blankets is compared based on the common reference ITER parameters and using the same computational tools. The issue of utilizing a breeding blanket in ITER has been debated since the end of the CDA phase. The impact of using a breeding blanket on magnet shielding is addressed. A High Aspect Ratio Design (HARD)[4] has been proposed by the US home team as an alternate to the CDA design. The shielding performance of the HARD design compared to the CDA design is presented.

DESIGN REQUIREMENTS AND OPERATING PARAMETERS

The ITER reactor is designed to achieve a fluence goal of $\sim$3 MW$\cdot$y/m$^2$. During the 15 year life of the machine, $\sim$3.8 full power years (FPY) of operation are expected; 0.05 FPY in the physics phase and 3.7 FPY in the technology phase. The overall dimensions of the reactor are fixed in both phases and 1100 MW and 860 MW of fusion power are anticipated in the physics and technology phases, respectively. The peak neutron wall loadings on the inboard and outboard are 0.88 and 1.2 MW/m$^2$, respectively, in the technology phase. Proper performance of the TF magnets is guaranteed if the radiation limits are met. These limits are $5\times10^9$ rads, 55 kW, 5 mW/cm$^3$, $10^{19}$ n/cm$^2$, and $6\times10^{-3}$ dpa for the peak end-of-life dose to the epoxy insulator, total nuclear heating in the 16 TF magnets, peak nuclear heating in the winding pack, end-of-life fast neutron fluence ($E_n > 0.1$ MeV) to the $Nb_3Sn$ conductor, and peak displacement damage in the Cu stabilizer, respectively. The heating limit is more important to be met in the physics phase due to the higher level of fusion power while the dose limit needs to be satisfied at the end of the technology phase. Another limit that needs to be satisfied is the He production in the vacuum vessel (V.V.). It should not exceed 0.1appm for reliable rewelding of the different V.V. components.

Before comparing the calculated results with these magnet design 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. They depend on the type of materials used in the blanket/shield, the characteristics of the assembly gaps, and the uncertainties in cross section data evaluation. They vary slightly with the response functions and differ for local and integral quantities. 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. The results presented here include the safety factors.

INBOARD SHIELD DESIGN OPTIMIZATION

Optimization studies were performed to design an efficient shield to protect the TF magnets.[2] Several options for the shield were examined and the shielding capability of many materials

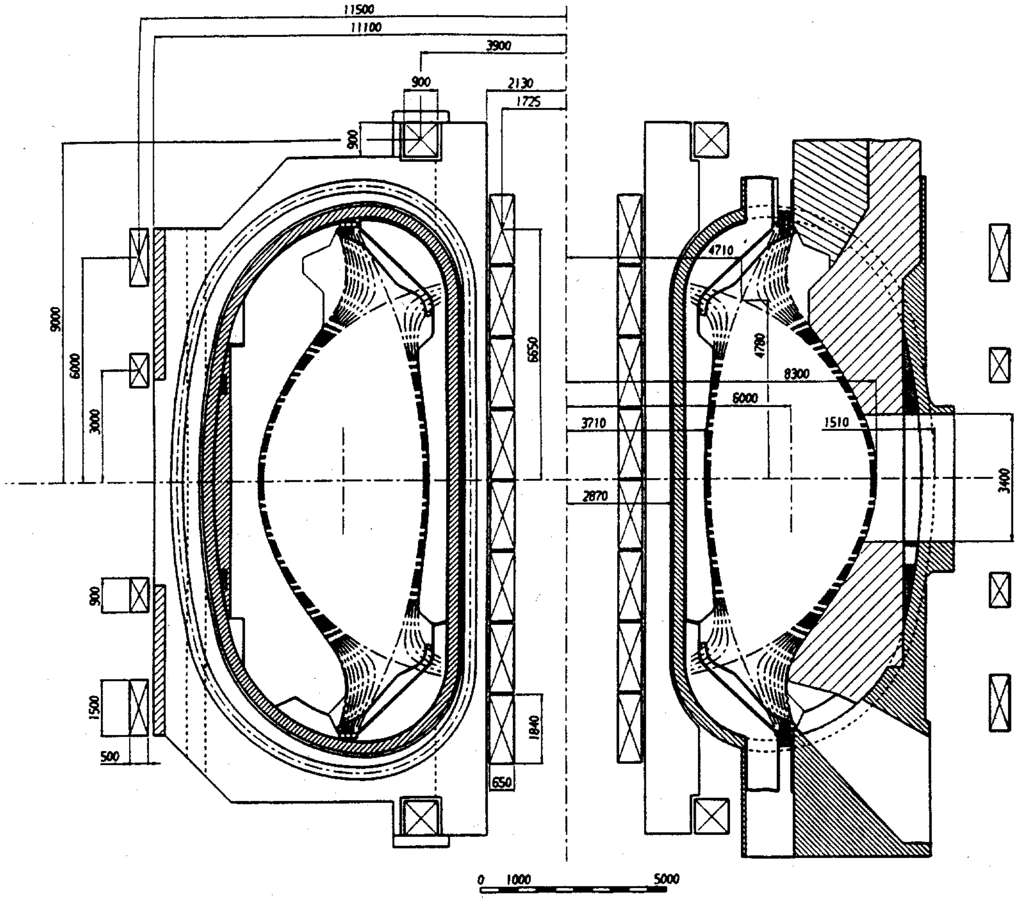


Fig. 1. Vertical cross section of ITER.

was assessed. Besides 316SS 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[5] and the cross section data based on the ENDF/B-V evaluation.[6] The 46 neutron and 21 gamma energy group structure and the $P_3$-$S_8$ approximation were used. Toroidal cylindrical geometry was used around the machine axis, permitting the representation of both inboard (IB) and outboard (OB) sides simultaneously.

The highest radiation damage in the inner legs occurs at the midplane where the space available for the IB 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 IB blanket/shield increases in thickness reaching 111 cm at the top/bottom. In the physics phase, 2 cm thick C tiles are used to protect the first wall (FW), while in the technology phase only 0.05 cm of W coating is used. The 1.5 cm thick FW is an integral part of the blanket/shield and consists of water cooled steel layers. The U.S. solid breeder IB blanket is 11.6 cm thick at the midplane and gradually increases in thickness toward the top/bottom.[7] The space between the blanket and the V.V. is occupied by the shield. The V.V. is 25 cm thick and has several coolant channels. Outside the V.V., there is a 5 cm thick back layer where 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 30.6 cm thick winding pack.

The shield is designed to satisfy the neutronics, thermal hydraulics, and mechanical design requirements. The reference shield is composed of 316 SS structure and $H_2O$ coolant. Alternate options such as B-SS shield, B-$H_2O$ coolant, and $^{10}B$ enriched borated materials were also analyzed.[2] Using B-SS shield with B-$H_2O$ coolant in the V.V. reduces magnet damage by about 20%. An additional 20% reduction is obtained by enriching the boron to 90% $^{10}B$. However, much 316 SS structure is needed to carry the electromagnetic load during plasma disruption diluting the effect of the B-SS shield. The B-$H_2O$ usually causes corrosion problems and 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.

The water cooled 316 SS 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 have been optimized to reduce the magnet damage and the SS layers have been checked with respect to maximum temperature and thermal stresses. The layered shield and V.V. at the midplane is shown in Fig. 2. Our analysis shows

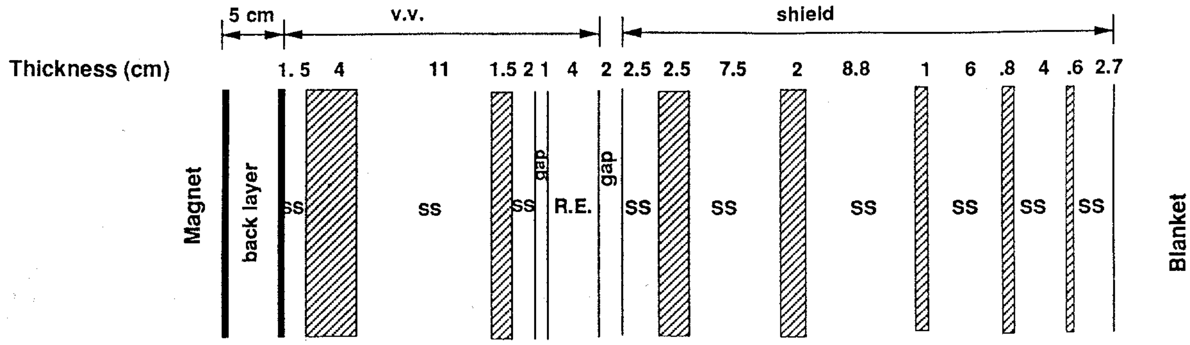


Fig. 2. Layered configuration of the inboard shield/V.V. at the midplane.

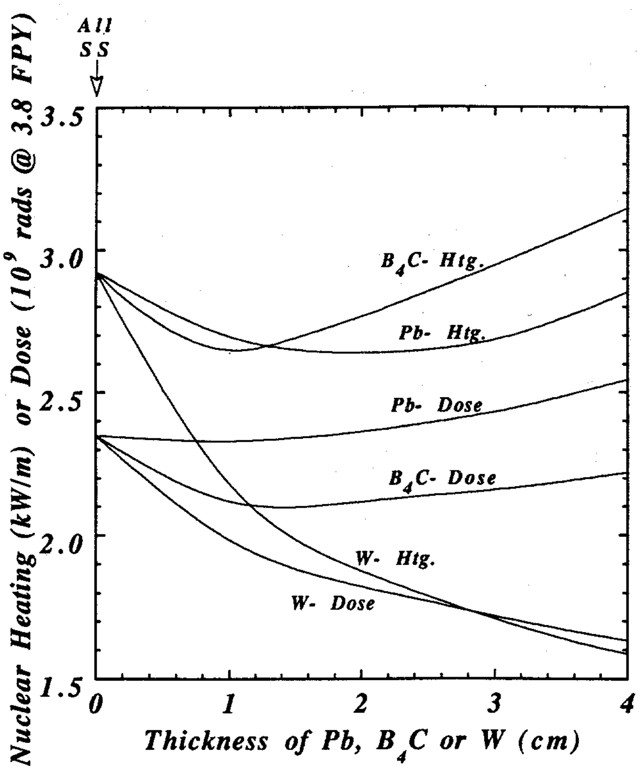


Fig. 3. Effect of the special materials used in the back layer (safety factors not included).

that for such a shield arrangement and for a pure SS back layer outside the V.V., the peak insulator dose and total nuclear heating in the inner legs amount to $7.2 \times 10^9$ rads at 3.8 FPY and $\sim$16 kW, respectively.

Other materials should be incorporated in the back layer to reduce the magnet damage to an acceptable level. 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 Fig. 3. In the analysis, 20% 316 SS structure is considered for canning the special materials. 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 magnet heating followed by Pb and then $B_4C$. It was found that two consecutive layers, 1 cm $B_4C$ followed by 3 cm Pb, is more effective than using each separately as shown in Fig. 4. 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 $\sim$20%. Because W is expensive, it can be used in limited places where the shielding space is critical. An acceptable peak end-of-life insulator dose of $4.7 \times 10^9$ rads is obtained 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 layer. All other magnet radiation effects are below the limits. The He production in the V.V. is excessive (1.5 appm at end-of-life). Different schemes other than welding are needed for the V.V. assembly especially in the high damage zone at midplane.

## MAGNET RADIATION EFFECTS IN THE RECESS AREA

Another critical area in the inner legs occurs behind the shield recess which is between the upper/lower end of the IB blanket and the inner end of the divertor plates (z = 3.8 to 4.7 m). The shield/V.V. therein is limited to $\sim$70 cm in thickness. 3-D Monte Carlo calculations[3] indicated that the neutron wall loading in the technology phase ranges between 0.086 and 0.17 MW/m$^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 $3.6 \times 10^9$ rads at 3.8 FPY. The nuclear heating deposited at both the upper and lower parts of the inner legs behind the shield recess totals 4.2 and 3.2 kW in the physics and technology phases, respectively. 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 end-of-life.

## MAGNET DAMAGE PEAKING

The mechanical design of the IB blanket/shield/V.V. calls for a wide variation in material arrangement within a single module,[1] as shown in Fig. 5. This toroidal variation in composition affects the damage level at the magnet. Furthermore, the thinning in the coil case of the inner legs 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 IB blanket/shield arrangement was modeled for the 2-D code TWODANT.[8] The variation in coil case thickness and the 2 cm wide assembly gaps were included

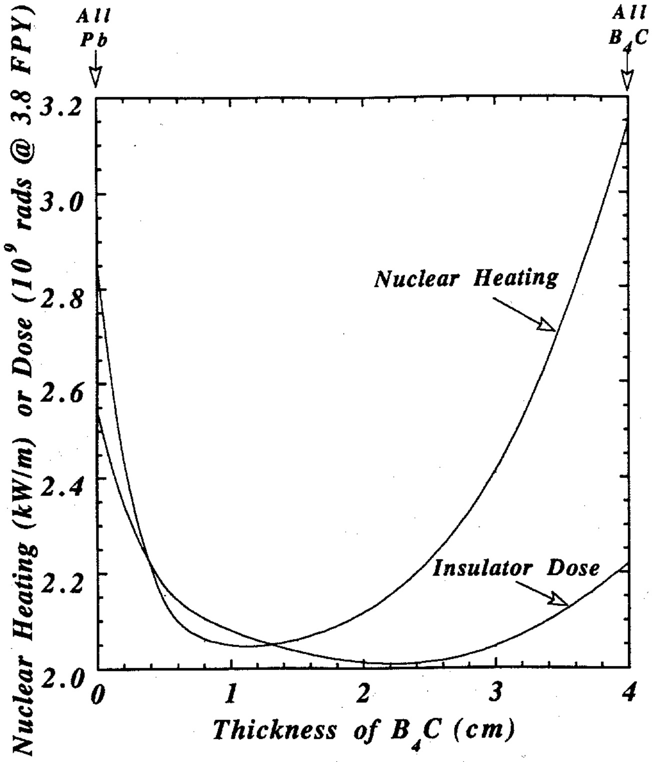


Fig. 4. Effect of using a combination of $B_4C$ and Pb in the back layer (safety factors not included).

in the model. The calculations were performed in x-y geometry using the $P_3$-$S_{16}$ approximation. The peaking factor is defined as the ratio of the 2-D to the 1-D values for the damage at the magnet. The peaking factors differ with the response function. The increase in damage due to the assembly gaps ranges between 1.63 and 1.85 for the local responses. The damage at the corners of the winding pack is enhanced further by a factor of $\sim$1.2 due to the relatively thin coil case (2.6 cm compared to 6 cm at the middle).

BULK SHIELDING IN THE DIVERTOR REGION

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 $\sim$70 cm at each of the top and bottom. The tilted divertor plates have limited the space available for the shield/V.V. behind the outer end of the DP as shown in Fig. 1. The minimum shield/V.V. thickness in this region is 48 cm. Additional shielding is provided by the DP and its support structure. The reference divertor design calls for a 5.5 cm thick DP in the physics phase and a relatively thicker DP (9.5 cm) for the technology phase. The electric insulator surrounding the winding pack is further protected by the fairly thick coil case ($\sim$30 cm). More shield is available when proceeding from the outer end of the DP towards the IB and OB sides. In the technology phase, the neutron wall loading, determined from the 3-D calculations, ranges between 0.27 and 0.46 MW/m$^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 OB blanket/shield. The divertor cooling tubes run toroidally after protruding vertically between coils.

The radiation effects at the top and bottom portions of the

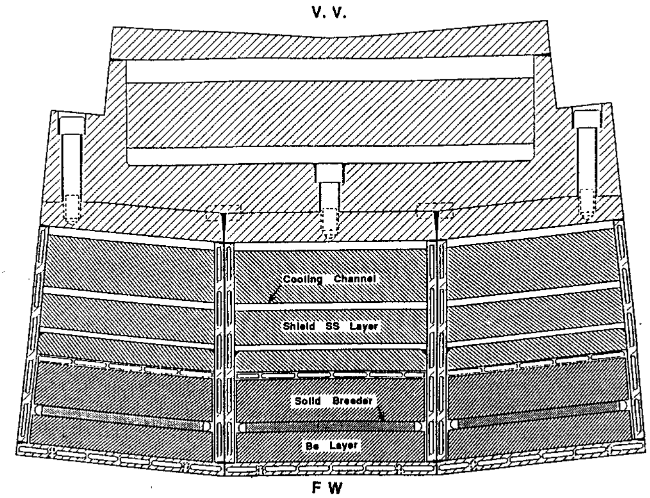


Fig. 5. Mechanical design of the inboard blanket/shield.

TF coils have been calculated using 1-D models. The peak end-of-life insulator dose values are $3.9 \times 10^9$ and $5.7 \times 10^9$ rads in the lower and upper divertor plates, respectively. Our results show that all magnet radiation limits will be satisfied if the upper end of the OB blanket is extended inward up to the plasma boundary to provide extra protection for the top divertor region. This modification seems feasible from the engineering standpoint and has the advantage of reducing the magnet heating by $\sim$5 kW. On this basis, the heating in the divertor region amounts to 26 and 15 kW in the physics and technology phases, respectively.

NEUTRON STREAMING THROUGH VACUUM PUMPING DUCTS

Neutron streaming through the vacuum pumping ducts in the lower divertor region can lead to additional damage in the adjacent TF coils. Detailed three-dimensional neutronics calculations have been performed to determine the radiation effects in the parts of the TF coils behind the lower divertor region and adjacent to the vacuum pumping ducts and divertor coolant tubes. The continuous energy, coupled neutron-gamma-ray Monte Carlo code MCNP,[9] has been used in the calculations. Because of symmetry, only 1/64 of the reactor was modeled with surrounding reflecting boundaries. Detailed configurations of the blanket, shield, V.V., coil case (C.C.), and winding pack (W.P.) in the divertor region were included in the model. In addition, the divertor plates, vacuum pumping ducts and divertor coolant tubes penetrating between TF coils were modeled in detail.

Figures 6 and 7 show vertical cross sections of the geometrical model used at toroidal locations through the vacuum pumping duct and the TF coil, respectively. The smallest shield/V.V. thickness is 48 cm at the edge of the divertor plate. A horizontal cross section is given in Fig. 8. At the side of the vacuum pumping duct the shield thickness is 30 cm and the V.V. thickness is 10 cm. The 2-cm-wide assembly gaps between adjacent blanket/shield modules are included in the model. 316 SS/$H_2O$ (at 20 vol.% $H_2O$) is used in the bulk and penetration shield and a 5 cm thick $B_4C$/Pb back layer is used outside the V.V. The results indicate that $2.82 \times 10^{-4}(\pm 3\%)$ neutrons stream into each vacuum pumping duct per DT fusion. The number in parentheses corresponds to the statistical uncertainty in the Monte Carlo calculation. 10.5% of these neutrons are uncollided source neutrons streaming directly into the duct. The leak-

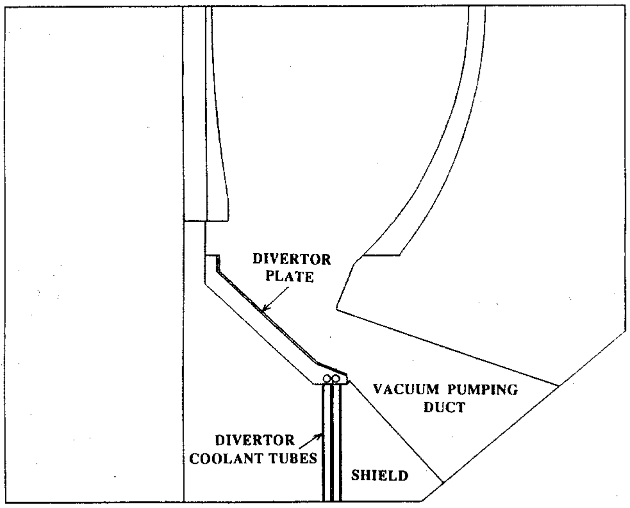


**Fig. 6. Vertical cross section through the vacuum pumping duct.**

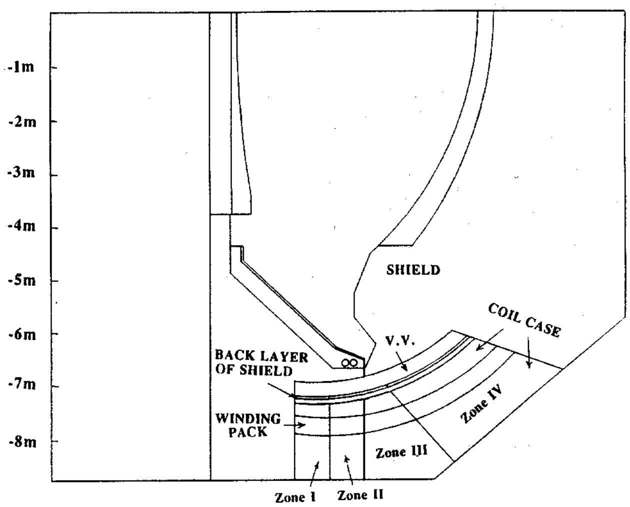


Fig. 7. Vertical cross section through the TF coil.

age out of the vacuum pumping duct at the back of the C.C. amounts to $7.75 \times 10^{-5}(\pm 4\%)$ neutrons per DT fusion with a very soft spectrum. The total number of neutrons leaking from the 16 vacuum pumping ducts is $1.24 \times 10^{-3}$ per DT fusion. This amounts to $3.78 \times 10^{17}$ neutrons per second in the technology phase.

## MAGNET RADIATION EFFECTS IN THE LOWER DIVERTOR REGION

The end of life insulator dose averaged over the front and side surfaces of the W.P. in the four zones of the TF coil is given in Table I. Values in parentheses correspond to the statistical uncertainty in the estimate of the response. It is clear that the sides of the coils are well protected from neutrons streaming into the vacuum pumping ducts and divertor coolant tubes. In all zones, the highest radiation effects occur at the front surface of the coil. The largest radiation effects are in Zones II and III because the shield is thinner at the edge of the divertor plate.

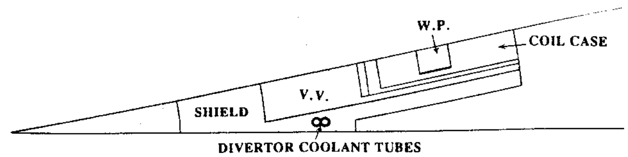


Fig. 8. Horizontal cross sections of the model at Z = -7 m.

Table I. Peak End-of-Life Insulator Dose (rads) in the Lower Divertor Region.

| | Front | Side |
|---|---|---|
| Zone I | $9.8 \times 10^8$ (.18) | $4.4 \times 10^8$ (.21) |
| Zone II | $2.7 \times 10^9$ (.15) | $1.2 \times 10^9$ (.19) |
| Zone III | $2.7 \times 10^9$ (.15) | $1.1 \times 10^9$ (.28) |
| Zone IV | $8.9 \times 10^7$ (.22) | $5.9 \times 10^7$ (.26) |

## OUTBOARD SHIELD

There is ample space available for the OB 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 vol.% $H_2O$) the peak dose to the insulator is $6 \times 10^4$ rads at 3.8 FPY and the total nuclear heating in the outer legs amounts to 0.06 kW. All other radiation effects are $\sim$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 which assures reliable rewelding for the V.V. assembly in the outboard region.

## 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. 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

Table II. Total Nuclear Heating (kW) in TF Magnets.

| | Physics Phase | Technology Phase |
|---|---|---|
| Inboard | 11 | 10 |
| Recess | 4 | 3 |
| Divertor | 26 | 15 |
| Penetrations | 4 | 3 |
| Total | 45 | 31 |

Table III. Pertinent Design Parameters for the Various Blanket Options.

| Blanket Design | US | J | EC | USSR |
|---|---|---|---|---|
| IB FW/Blanket Thickness (cm) | 11.6 | 12 | 18.8 | 18 |
| Breeder-Li Enrichment | $Li_2O$-95% | $Li_2O$-nat. | $LiAlO_2$-90% | LiPb-90% |
| Solid Breeder Form | Sintered Blocks | Sintered Pebbles | Sintered Cylindrical Pellets | |
| IB Blanket Composition | 10% $Li_2O$, 80% Be<br>5% SS, 5% $H_2O$ | 47% $Li_2O$, 27% Be<br>9% $H_2O$, 6% SS<br>11% void | 9% $LiAlO_2$, 37% Be<br>16% $H_2O$, 9% SS<br>6% He, 23% void | 43% LiPb, 30% SS<br>10% $H_2O$, 17% void |

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 radiation streaming through the various penetrations. Table II details the nuclear heating in the various regions in the physics and technology phases. The value of nuclear heating in the divertor region is based on the assumption that the upper end of the OB blanket is extended inward up to the plasma boundary. The total heating is well below the design limit in both phases.

## MAGNET SHIELDING EFFECTIVENESS OF THE ITER BLANKET DESIGN OPTIONS

During the CDA phase of ITER, each of the four international parties, the US, USSR, EC, and Japan, has proposed a blanket design that fulfills as much as practicable the ITER requirements.[1] No blanket has yet been selected as the reference design for ITER. However, the ceramic breeder blanket was chosen as the first option and the LiPb blanket as the backup. There are still some critical issues associated with each blanket. Work is underway to improve the various designs and to solve the identified problems hopefully before the start of the EDA. Some design parameters for the proposed blankets are listed in Table III. The US, J, and EC designs utilize solid breeders with Be multiplier while the USSR offers a LiPb blanket. All blankets are water cooled and employ 316 SS as a structural material. The blankets differ mainly in the mechanical configuration, forms of breeder and multiplier, and material density fraction.

During the CDA phase, the 4 parties have analyzed their blanket/shield designs with respect to the effect on magnet radiation damage. There are some differences between the various analyses regarding the codes and cross section data, V.V. configuration, magnet composition, FW tile/coating thickness, and machine operating time. Although close contact was maintained with the designers of the various reactor components, certain assumptions had to be made by each team to finalize the shielding calculations. We performed 1-D neutronics calculations to determine radiation effects in the inner legs of TF coils for the different blanket designs using the same calculational tools and reference parameters.

Table IV compares the results of all blanket/shield designs. For the purpose of comparison, no W back layer is used in any of these cases. The results for the latest layered solid breeder Japanese design are reported. Here, the comparison is done on the same bases and the difference in results is essentially due to the different blanket/shield concepts. From the shielding viewpoint, the US blanket/shield design has the highest performance. The J, EC, and USSR designs result in 1.3-1.6, 2, and 1.2 higher magnet damage, respectively, compared to US. The insulator dose exceeds the limit in all cases. As mentioned before, a W back layer reduces the dose below the limit for the US design. For the other designs, the W will not help. To meet the dose limit, the blanket designs need to be modified or, alternatively, extra shields should be added to the current 84 cm thick IB region. Approximately 5, 7, and 3 cm of extra shields are required for the J, EC, and USSR designs, respectively. Since most of the magnet nuclear heating results from the divertor region, the impact of the inboard blanket design on total nuclear heating is small. It varies between 45 and 57 kW in the physics phase with the US design yielding the lowest heating and the EC design resulting in the largest.

## IMPACT OF TRITIUM BREEDING IN ITER ON MAGNET SHIELDING

Since the end of the CDA phase of ITER, the issue of using a breeding blanket in ITER has been debated. Detailed cost/benefit/risk analysis has been carried out by the US ITER home team. One of the technical issues addressed is the impact of utilizing a breeding blanket on magnet shielding. The inboard blanket thickness is about 10 cm which is much smaller than the total inboard FW/blanket/shield/V.V. thickness (84 cm). If no breeding blanket is to be used during the whole ITER operation, the important impact on magnet shielding will be reducing the total required inboard region thickness. It is estimated that the same magnet damage will result by replacing the US blanket by about half the thickness of pure shield. Hence, a savings of about 5 cm in inboard region thickness can be achieved by not breeding in the inboard side. On the other hand, if an inboard breeding blanket is to be installed later in ITER, the inboard region thickness will not change and magnet heating will be lower in the early stage of ITER operation before installing the blanket. The blanket designs are different in their shielding performance with some blankets requiring up to additional 7 cm inboard shield to adequately protect the magnets as indicated in the previous section. Hence, careful design of the inboard blanekt, with magnet shielding in mind, can reduce the impact of inboard breeding on magnet shielding. Toroidal inhomogeneities, in addition to 2 cm thick assembly gaps between adjacent modules in the CDA design, yield magnet damage peaking factors in the range from 1.6 to 2. This peaking results mainly from streaming through the assembly gaps. Since not using an inboard breeding blanket results in a thinner inboard region, the peaking factors are expected to slightly increase. There is ample space available in the outboard side and using a breeding blanket is not expected to affect magnet shielding in the outboard region.

Table IV. Radiation Effects in Inner Legs of TF Magnets.

| Blanket/Shield Design | US | J‡ | EC | USSR |
|---|---|---|---|---|
| Physics Phase (2 cm C tiles) | | | | |
| Peak Nuclear Heating ($mW/cm^3$): | | | | |
| Winding Pack | 1.8 | 2.9 | 4.1 | 2.3 |
| Coil Case | 3.3 | 4.3 | 6.5 | 3.7 |
| Heating per Unit Length (kW/m): | | | | |
| Winding Pack | 2.5 | 4.3 | 5.6 | 3.1 |
| Coil Case | 2.1 | 3.2 | 4.6 | 2.6 |
| Total | 4.6 | 7.5 | 10.2 | 5.7 |
| Total Heating in Inner Legs (kW) | 11.1 | 18.2 | 21.2 | 15.7 |
| Technology Phase (0.05 cm W coating) | | | | |
| Peak Dose to Insulator ($10^9$ rads) | 5.8 | 10.3 | 12.4 | 6.8 |
| Peak Fast n Fluence to $Nb_3Sn$ ($10^{18}$ $n/cm^2$) | 6.6 | 12.0 | 14.3 | 7.8 |
| Peak dpa in Cu Stabilizer ($10^{-3}$ dpa) | 3.4 | 6.4 | 6.9 | 3.7 |
| Peak Nuclear Heating ($mW/cm^3$): | | | | |
| Winding Pack | 1.6 | 2.6 | 3.6 | 2.1 |
| Coil Case | 3.0 | 3.9 | 5.8 | 3.3 |
| Heating per Unit Length (kW/m): | | | | |
| Winding Pack | 2.2 | 3.8 | 5.0 | 2.8 |
| Coil Case | 2.0 | 2.9 | 4.1 | 2.3 |
| Total | 4.2 | 6.7 | 9.1 | 5.1 |
| Total Heating in Inner Legs (kW) | 10.2 | 16.3 | 18.9 | 14.0 |
| Extra Shield Required to Meet Dose Limit (cm) | 1† | 5 | 7 | 3 |

†Or W back layer over 1 m height with no need for extra shield
‡Layered pebble bed

## SHIELDING PERFORMANCE FOR THE HIGH ASPECT RATIO DESIGN (HARD)

A high aspect ratio design (HARD)[4] has been proposed by the US team to enhance the performance capabilities of ITER. Shielding analysis has been performed for the proposed HARD design and the shielding performance for this configuration is compared to that for the CDA design. Three different operating modes, namely, the inductive, steady state and hybrid modes are proposed for HARD. The fusion power varies depending on the mode of operation. The shielding analysis has been performed for the hybrid case which represents the worst case from the shielding standpoint since it yields the highest neutron wall loading. The peak wall loading in the hybrid case with 1080 MW fusion power is 1.39 $MW/m^2$. Magnet nuclear heating results can be determined for the other modes of operation by scaling with the fusion power (850 and 960 MW for the ignited and steady state modes, respectively). On the other hand, the end-of-life fluence and insulator dose should be independent of the fusion power as long as the ITER fluence goal of 3 $MW \cdot y/m^2$ is maintained.

Table V. Relevant Shielding Design Parameters.

| | CDA (Physics Phase) | HARD (Hybrid) |
|---|---|---|
| Fusion power (MW) | 1100 | 1080 |
| Average wall loading ($MW/m^2$) | 0.93 | 1.04 |
| Inboard region | | |
| Peak wall loading ($MW/m^2$) | 1.13 | 1.39 |
| Blanket/shield/gap/ V.V. thickness* (cm) | 84 | 83 |
| Gap thickness (cm) | 2 | 4 |
| Coil case and winding pack cross section area* ($m^2$) | 6.3 | 10.8 |
| Recess region | | |
| Peak wall loading ($MW/m^2$) | 0.22 | 0.25 |
| Blanket/shield/gap V.V. thickness (cm) | 70 | 70.6 |
| Divertor region | | |
| Peak wall loading† ($MW/m^2$) | 0.67 | 0.75 |
| Blanket/shield/gap/ V.V. thickness† (cm) | 58 | 56 |
| Coil case thickness (cm) | 26 | 39 |

*At midplane.
†At IB side of outer end of divertor plate.

Table VI. Total Nuclear Heating in TF Coils for the HARD Design.

| | HARD Hybrid (1080 $MW_f$) | Modified HARD Hybrid (1080 $MW_f$) |
|---|---|---|
| Inboard | 21 | 14 |
| Recess | 8 | 6 |
| Divertor | 35 | 23 |
| Penetrations | 4 | 4 |
| TOTAL (kW) | 68 | 47 |

The design parameters pertinent to shielding analysis are given in Table V for the HARD (hybrid) and CDA (physics phase) designs. The parameters are given for the three regions with critical shielding space. These are the inboard region, the divertor region and the shield recess area. For comparable fusion power, the peak neutron wall loading values in these regions are higher than those in the CDA design because of the higher aspect ratio in the HARD design. The assembly gap between the back of the shield and the V.V. is 4 cm in HARD compared to 2 cm in CDA. Furthermore, the inboard blanket/shield/gap/V.V. thickness at the midplane is 83 cm vs. 84 cm in CDA. This means that 3 cm of the shield is replaced by void which translates into ~50% higher damage at the magnet. Hence, the peak end-of-life insulator dose at the inner legs of the TF coils will exceed the design limit for epoxy. The insulator dose in the divertor region of HARD is acceptable as a result of using a thicker coil case.

The nuclear heating in the TF magnets for the 3 critical regions (IB, recess, and divertor) was calculated taking into account the poloidal variation of neutron wall loading and blanket/shield/V.V. thickness. The results are summarized in Table VI for the HARD hybrid design. These results are to be compared with those for the CDA design given in Table II. Radiation streaming through the different penetrations is assumed to contribute 4 kW to the total magnet nuclear heating. Most of the heating in the inboard region is generated in the 3 m high middle section. Heating in this region is doubled due to the larger magnet volume (70% more), higher wall loading, and thinner shield, compared to the CDA design. The results indicate that the total heating loads are 68, 53, and 60 kW in the hybrid, ignited, and steady state modes of operation, respectively.

In order to reduce the heating to a reasonable level and meet the insulator dose limit for 3 MW·y/m$^2$ fluence, several modifications to the present HARD design need to be adopted. The total IB blanket/shield/gap/V.V. thickness at midplane should be restored to 84 cm with the gap reduced to 2 cm as in the CDA design. In addition, it is necessary to extend the upper parts of the side modules of the ouboard blanket inward up to the plasma boundary (similar to the lower parts) in order to provide extra shielding for the upper divertor region. Notice that the side modules are located underneath magnets and there will be no interference with maintenance. The impact of this modification is to reduce the divertor heating by $\sim$5 kW. If these changes take place, the heating in the magnets will be 47 kW for the hybrid case, as detailed in the last column of Table VI.

## SUMMARY

Detailed shielding analyses have been performed and necessary reactor design modifications have been proposed in order to meet all radiation design limits for the magnets. 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 45 and 31 kW of heat load to the TF magnets are expected in the physics and technology phases, respectively. For the ITER fluence goal of 3 MW·y/m$^2$, the end-of-life dose to the epoxy insulator is below the $5 \times 10^9$ rads limit at all locations except in regions behind the outer end of the upper divertor plate. Proposed solutions include modifying the shape of the upper end of the outboard blanket or increasing the divertor shield thickness by a few centimeters. The 3-D neutronics analysis performed for the magnet shield design of ITER indicates that the parts of the TF coils adjacent to the lower divertor vacuum pumping ducts are well shielded against streaming radiation.

Based on the results of the shielding analyses conducted during the conceptual design phase for ITER, it is concluded that the current inboard configuration is suitable for the US blanket and shield design. On the other hand, it is mandatory to modify the J., EC, and USSR blanket designs or, alternatively, increase the inboard shield by 3-7 cm in order to satisfy the insulator dose limit. The latter option will surely require a revision of the overall ITER dimensions. Careful design of the inboard breeding blanket with shielding performance in mind will reduce the impact of breeding in ITER on the thickness of inboard zone and hence the reactor size. The high aspect ratio design (HARD) was found to have about 50% higher magnet damage due to the higher neutron wall loading, larger magnet volume and thinning of inboard shield.

## ACKNOWLEDGEMENTS

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