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Abstract

Several shield options were analyzed for the ITER conceptual design to minimize the nuclear responses in the toroidal field (TF) coils. The total nuclear heating in the physics phase and the insulator dose in the technology phase are the most critical parameters in the design process. The first shield option has type 316 stainless steel and water shielding material. Steel and water also serve as structural material and coolant, respectively. The second option is similar to the first except that borated water is used instead of ordinary water. The other two options include a small layer of lead or boron carbide (B_4C) at the back of the shield. The last three shield options were considered to reduce the nuclear heating in the toroidal field coils relative to the steel/water shield. An optimization process was performed taking into consideration the thermal-hydraulics and the engineering requirements to define the shield configuration. A careful integration was performed to calculate the total nuclear heating in the toroidal field coils which account for the neutron wall loading distribution, the change in the shield thickness in the poloidal direction, and the space between the toroidal field coils in the divertor zone. The results show that the steel/water/Pb and the steel/borated-water shield options are very close in terms of the total nuclear heating in the toroidal field coils and the dose in the insulator material. The other two options, steel/water and steel/water/ B_4C deposit more nuclear heating in the toroidal field coils.

Introduction

The shielding analyses including nuclear responses in the toroidal field coils were performed for each phase of ITER operation. This division is dictated by the different design configurations and reference parameters for the physics and the technology phases. In the technology phase, the analyses are given for the reactor configuration with solid breeder blanket. Several shielding options and shielding optimization analyses are presented in this paper. The total nuclear heating in the toroidal field coils is the key parameter for the shield design in the physics phase. The corresponding parameters in the technology phase is the dose at the end of life for the insulator material in the toroidal field coils.

Physics Phase Analyses

Several shield options were analyzed for the physics phase of ITER to minimize the total nuclear heating in the toroidal field coils. The first shield option has type 316 stainless steel and water shielding material. Also, steel and water serve as structural material and coolant, respectively. The second option is similar to the first except that borated water is used instead of ordinary water. The other two options include a small layer of lead or boron carbide (B_4C) at the back

of the shield. The last three shield options were considered to reduce the nuclear heating in the toroidal field coils relative to the steel/water shield option. The use of the first shield option produces high gamma heating in the toroidal field coils because of the gamma rays generated from the (n,γ) reactions. Therefore, lead was introduced at the back of the shield to reduce the gamma heating in the toroidal field coils as much as possible. The other approach is to use borated water to absorb the neutrons in the boron to avoid the generation of gamma rays from neutrons capture in the steel. The use of boron carbide at the end of the shield is employed for the same function. An optimization process was performed taking into consideration the thermal-hydraulics and the engineering requirements to define the shield configuration.

The shielding analysis is based on a one-dimensional calculation. The one-dimensional discrete ordinates code ONEDANT [1] was used to perform the transport calculation with a P_3 approximation for the scattering cross sections and an S_8 angular quadrature set. A 67 coupled group nuclear data library [2] (46 neutron and 21 gamma) based on ENDF/B-V was employed for these calculations. VITAMEN-E [3] and KAOS/LIB-V [4] libraries were used to obtain this working library. The calculational model is based on a midplane cross section for the reactor configuration.

Several design analyses steps were performed to optimize the shield composition and material arrangement to reduce the nuclear responses in the TF coils to the minimum possible values under several design constraints. The constraints are as follows: a) the materials in the shield are limited to water, type 316 stainless steel, lead, and boron (borated water or boron carbide), b) the shield thickness is constrained by the current ITER configuration, c) the need to have a few steel layers (3-4) with a total thickness of about 30 to 40 cm in the semipermanent section of the shield for structural purposes, d) the desire to avoid temperature changes across the vacuum gap between the removable and semipermanent sections of the shield, and e) the inclination to operate the shield at low temperature as much as possible.

As a first step in the shielding optimization process, a calculational model was developed based on the ITER configuration at the midplane. Three shielding options were considered using the same calculation model as shown in Table 1. The first option has steel and water materials while the second option uses borated water (1 gm natural boron per 100 cm³ of water) instead of pure water. The third option is similar to the first option with the exception of 3 cm lead layer at the back of the shield. The total shield thickness is 75 cm including 2 cm gap for all the options.

The poloidal neutron wall loading distribution was calculated with the NEWLIT code. The shield calculational model uses a uniform neutron source distribution over the plasma volume which results in different neutron wall loading values relative to the actual neutron

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Table 1
Shielding Calculational Model for the ITER Physics
Phase Used in the First Step of the Shielding
Optimization Process

Reactor Component	Zone Thickness, cm		Zone Composition (Vol. %)
	Option 1	Option 3	
A. Toroidal Field (TF) Coils			
TF coil case	20	20	steel ^a
Insulator	1	1	polyimide insulator
Winding pack	48	48	winding pack ^b
TF coil case	5	5	steel
Gap	5	5	Vacuum
B. Inboard Semipermanent Shield Section			
Shield section	0	3	lead
	2	2	steel
	7	7	water
	10	10	steel
	2	2	water
	12	10	steel
	1	1	water
	10	10	steel
	1	1	water
	8	8	steel
	1	1	water
	6	5	steel
Gap	2	2	Vacuum
C. Inboard Removable Shield Section			
	5	5	steel
	1	1	water
	3	3	steel
	1	1	water
First wall	1	1	steel
First wall tile	2	2	carbon
D. Plasma Chamber			
Scrapeoff layer	15	15	Vacuum
Plasma	440	440	Vacuum
Scrapeoff layer	15	15	Vacuum
E. Outboard Removable Shield Section			
First wall tile	2	2	carbon
First wall	1	1	steel
	1	1	water
	3	3	steel
	1	1	water
	5	5	steel
Gap	2	2	Vacuum
F. Outboard Semipermanent Shield Section			
	6	6	steel
	1	1	water
	8	8	steel
	1	1	water
	12	12	steel
	1	1	water
	16	15	steel
	1	1	water
	16	15	steel
	1	1	water
	16	15	steel
	7	7	water
	2	2	steel
	8	8	water
	2	2	steel
	0	3	lead

^a Type 316 stainless steel.

^b 0.34 Type 316 stainless steel, 0.06 polyimide insulator, 0.10 liquid helium, 0.39 copper stabilizer, 0.11 Nb₃Sn superconductor.

wall loading values calculated by NEWLIT. Therefore, the results are normalized to the actual values of the neutron wall loading at the midplane. The nuclear responses calculated at the midplane are given in Table 2. The results show that the use of the small lead layer or boroated water reduces the nuclear responses by about a factor of two except for the fast neutron fluence. Also, the maximum nuclear heating in the TF coil winding pack is less than the 5 mW/cm³ specified by the TF coil design for both shield options. All the other nuclear responses have small values because of the low fluence (0.05 MW·y/m²) of the physics phase. The total nuclear heating in the TF coils was calculated by integrating the total nuclear heating in the TF coils per unit length calculated from the previous model over the length of the first wall in the poloidal direction. The neutron wall loading distribution and the first wall/shield thickness as a function of the first wall length in poloidal direction including the divertor area are accounted for in the integration process.

Table 2
Nuclear Responses in the Toroidal Field Coils for the
Physics Phase Resulted from the First Step of the
Shielding Optimization Process

Nuclear Response	Shield Option		
	Steel/H ₂ O	Steel/H ₂ O-B	Steel/H ₂ O/Pb
Max. nuclear heating, mW/cm³			
case	9.57	3.57	2.98
winding packs	2.27	1.17	1.30
Max. fast neutron fluence^a, n/cm²			
	2.98x10 ¹⁶	2.92x10 ¹⁶	4.15x10 ¹⁶
Max. insulator dose^a, rads			
	6.05x10 ⁷	3.70x10 ⁷	4.47x10 ⁷
Max. atomic displacement^a, dpa			
	1.66x10 ⁻⁵	1.62x10 ⁻⁵	2.15x10 ⁻⁵
Total nuclear heating, W/cm			
cases	47.0	19.5	19.3
winding packs	22.4	14.6	18.2
Total	70.0	34.1	37.5
Total nuclear heating, kW			
cases	14.16	5.80	5.74
winding packs	6.66	4.34	5.41
Total	20.82	10.14	11.15

^a A total fluence of 0.05 MW·y/m² was used for the physics phase.

The second step in the shield optimization process was to calculate the maximum temperature in the shield materials and vary the shield zones thicknesses given in Table 1 to get the minimum possible nuclear heating in the TF coils. The thicknesses of the last two steel layers in the inboard removable section were adjusted as shown in Table 3 to keep the temperature across the vacuum gap about the same and the maximum steel temperature less than 300°C. The thicknesses of the steel layers in the inboard semipermanent section of the shield were allowed to vary while the total first wall/shield thickness is kept constant by adjusting the thickness of the water zones as shown in Table 3. A sample of the results from this parametric analysis are shown in Fig. 1. The configuration with minimum total nuclear heating in the TF coils was chosen for further optimization where the thickness of the lead zone was allowed to vary without changing the total first wall/shield

thickness. The last layer of water in the shield was adjusted to balance the change in the lead zone thickness. Based on the obtained results, the shield configuration with 3 cm lead layer was chosen. Table 4 gives all the nuclear responses for this configuration.

Table 3
Inboard Shielding Calculational Model^a for ITER Physics Phase Used in the Second Step of the Shielding Optimization Process

Zone ^b Number	Zone Thickness, cm	Zone Material
8	3	Lead
9	2	Type 316 stainless steel
10	8	Water
11	Variable	Type 316 stainless steel
12	Variable	Water
13	Variable	Type 316 stainless steel
14	Variable	Water
15	Variable	Type 316 stainless steel
16	Variable	Water
17	8	Type 316 stainless steel
18	1	Water
19	5	Type 316 stainless steel
20	2	Gap
21	4	Type 316 stainless steel
22	1	Water
23	4	Type 316 stainless steel
24	1	Water
25	1	Type 316 stainless steel
26	2	Carbon Tile

^a The magnet and outboard are similar to the model of Table 1.

^b The zone numbers are used in Fig. 1.

Figure 1

Total nuclear heating in the toroidal field coils as a function of the thickness of Zone 12 for different thicknesses of Zone 14 with a 1-cm thickness for Zone 16 and a total shield thickness constrained to 75-cm during the ITER Physics Phase.

Table 4
Nuclear Responses in the Toroidal Field Coils for the Physics Phase Resulted from the Third Step of the Shielding Optimization Process

Nuclear Response	Value
Maximum nuclear heating, mW/cm ³	
TF coil case	2.31
TF coil winding pack	1.17
Maximum fast neutron fluence, n/cm ²	4.35x10 ¹⁶
Maximum insulator dose, rads	4.36x10 ⁷
Maximum atomic displacement, dpa	2.46x10 ⁻⁵
Total nuclear heating, W/cm	
TF coils cases	15.48
TF coils winding packs	17.41
Total	32.89
Total nuclear heating, kW	
TF coils cases	4.60
TF coils winding packs	5.18
Total	9.78

Another calculation was performed with natural boron carbide instead of the lead to investigate the impact on the nuclear responses in the TF coils. The comparison with the lead configuration shows that the total nuclear heating in the TF coils is increased by a factor of two. In order for the B₄C configuration to compete with the lead case, the boron has to be distributed in the shield to reduce the gamma rays production from the neutron capture in the steel shield. Such configurations will be similar to the borated shield option considered before.

Technology Phase Analyses

The shielding optimization for the technology phase of ITER (the fourth step in the optimization process) started with the optimum configuration of the physics phase. The removable sections were replaced by the solid breeder blankets [5]. The main design criterion in this phase is to insure adequate protection for the toroidal field coils up to an average fluence of 3 MW·y/m² of DT operation. The thickness of the inboard removable section was increased by 10 cm from the physics phase which is expected to reduce the total nuclear heating in the toroidal field coils by about a factor of two. This leads one to consider the fluence parameters as the objective functions in the optimization process. The fluence parameters are the insulator dose, fast neutron fluence in the winding packs, and the atomic displacement in the copper stabilizer of the toroidal field coils. The most sensitive component of these parameters is the insulator dose. Therefore, six of the shielding zones (8, 10, 11, 12, 13, and 14) are allowed to vary during the optimization process to achieve the minimum insulator dose. The insulator dose and the total nuclear heating in the TF coils show minimum values at the shielding configuration of the physics phase as shown in Figs. 2 and 3. The atomic displacement and the fast neutron fluence have a minimum value at a different configuration with more steel. However, their minimum values are very close to the values obtained with shielding configurations of the physics phase.

The performance parameters of the shield for the technology phase are shown in Table 5. The total nuclear heating is about half the corresponding value in the physics phase. All the other parameters are less than the design requirements of the toroidal field coils.

Table 5

Nuclear Responses in the Toroidal Field Coils for the Technology Phase Resulted from the Fourth Step of the Shielding Optimization Process Based on the Geometry Model Given in Table 1 with 3,8 and 8-cm Thick for Zone Numbers 8, 11, and 13, Respectively

Nuclear Response	Value
Maximum nuclear heating, mW/cm^3	
TF coil case	1.38
TF coil winding pack	0.65
Maximum fast neutron fluence, n/cm^2	1.34×10^{18}
Maximum insulator dose, rads	1.40×10^9
Maximum atomic displacement, dpa	7.43×10^{-4}
Maximum copper resistivity, $\text{n}\Omega \cdot \text{cm}$	1.19×10^2
Total nuclear heating, W/cm	
TF coils cases	9.04
TF coils winding packs	9.29
Total	18.33
Total nuclear heating, kW	
TF coils cases	2.86
TF coils winding packs	2.94
Total	5.81

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Figure 2

Total nuclear heating in the toroidal field coils as a function of the thickness of Zone 12 for different thicknesses of Zone 14 with a 3-cm thickness for Zone 8 and total shield thickness constraints to 85 cm during the ITER Technology Phase.

Figure 3

Maximum insulator dose in the winding pack of the toroidal field coils as a function of the thickness of Zone 12 for different thicknesses of Zone 14 with a 3-cm thickness for Zone 8 and total shield thickness constrained to 85-cm during the ITER Technology Phase.

Also, the shield performance with natural boron carbide (0.8 DF) instead of the lead in the last layer of the inboard shield were calculated. The insulator dose and the total nuclear heating in the toroidal field coils values are about twice the corresponding value obtained with the steel/water/Pb shield option.