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LOW-COST SHIELD FOR TOKAMAK FUSION REACTORS*

by

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LOW-COST SHIELD FOR TOKAMAK FUSION REACTORS

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ABSTRACT

A low-cost shielding system was developed for the Demonstration Tokamak Power Plant. The shielding system was designed with low-cost materials to fulfill several functions and satisfy the design criteria. The reference shielding materials are water, ordinary concrete, Fe14Mn2Ni2Cr steel alloy, boron carbide, and lead. The cost of this shielding system is only 3% of the total direct cost compared to 11% in the STARFIRE design.

1. INTRODUCTION

The development of a low cost shield design for Tokamak fusion reactors has evolved from a strong incentive to reduce the capital cost of the shielding system which represents 8 to 16% of the total direct cost,¹⁻³ and from a desire to improve the reactor maintainability and performance. Based on this motivation, a shielding system for the Demonstration Tokamak Power Plant⁴ (DEMO) was designed to perform several functions and satisfy the design criteria. The main function during reactor operation is to reduce the neutron and photon leakage intensities from the shield. This reduction ensures that a) the different reactor components are protected from radiation damage and excessive nuclear heating, b) the neutron reaction rates in the reactor components which produce undesirable radioactive isotopes are reduced, and c) the workers and the public are protected from radiation exposure. Another function is to attenuate the decay gamma rays so that personnel are permitted access to the reactor building, with all shields in place within one day after shutdown. The personnel access to the reactor building significantly reduces the reactor downtime and the capital cost of the remote equipment required for maintenance.

The shielding materials were selected based upon tradeoff studies and the results from the previous designs;¹⁻³ in the selection process, cost, resource availability, activa-

tion, and performance were considered. The reference DEMO shielding materials are 68% ordinary concrete, 9% lead, 6% boron carbide, 7% Fe14Mn2Ni2Cr steel alloy (Fe1422), and 10% water by volume. Ordinary concrete and water are the main materials for the outboard and penetration shield. Water and Fe1422 are the coolant and structural materials for the whole shielding system. Also, Fe1422 is employed for the inboard shield where a material with high shielding performance is required. The low concentration of nickel and chromium motivated the use of Fe1422.^{1,5} A small amount of boron carbide is employed as a neutron absorber to reduce the activation in the reactor components and structural materials. A density factor of 0.7 is assumed for the boron carbide to avoid the fabrication cost required for a high density material. Lead is used as a gamma ray attenuator at the outermost surfaces of the shields so that the dose equivalent in the reactor building is reduced after shutdown.

The composition and arrangement of the inboard shield were optimized to achieve maximum protection for the toroidal field (TF) coils. The outboard shield consists of two sections, water and ordinary concrete. The first section was designed to perform several functions: a) protect the ordinary concrete shield against radiation damage and excessive nuclear heating, b) act as a neutron reflector to enhance the tritium breeding ratio and energy multiplication factor, and c) reduce the average neutron energy entering the concrete shield to minimize the production rate of long-lived isotopes which are generated mainly by high energy neutrons.

A description of the reference shielding system, including design criteria and constraints adopted for the DEMO design, as well as analyses and tradeoff studies are presented in this paper. Induced activity, biological hazard potential (BHP), and decay afterheat are analyzed for several shielding options. The radiation exposure dose in the reactor building after shutdown are presented as a

function of the shield thickness for the promising shielding options. Also, the dose equivalent outside the reactor building during operation is analyzed for the reference design. The economic analysis shows that the cost of this shielding system⁴ is only 3% of the total direct cost compared to 8-16% in the other designs.¹⁻³

II. SHIELD DESIGN CRITERIA

In the design of a shielding system, a careful definition of the design criteria is an essential step.⁶⁻⁸ The TF coils are designed to last the lifetime of the reactor without change in their performance. Therefore, the shielding system must protect the different components of the TF coils from radiation damage. In general, neutron irradiation of the superconductor materials tends to lower the critical current density (J_c) and the critical temperature (T_c). For NbTi superconductor, it has been shown that J_c is decreased by $\sim 10\%$ at 4×10^{18} n/cm² ($E > 0.1$ MeV) dose while T_c is unchanged.⁹ The copper resistivity is more sensitive to the neutron fluence, a neutron fluence of 10^{18} n/cm² generates an induced resistivity of 7×10^{-8} Ω -cm. This increase in the resistivity can be accommodated by using more copper stabilizer, or can be annealed out by warming the TF coils. The TF coil design allows 5×10^{-8} Ω -cm induced resistivity by radiation before an anneal process is required. The most sensitive component in the TF coil is the insulator materials because the radiation damage in them is irreversible and limits the operating life of the TF coils. It was demonstrated⁹ that epoxy insulator (Kapton) can withstand a radiation dose of 10^{10} rads at cryogenic temperature and retains high resistivity and mechanical strength. The nuclear energy deposition in the magnet impacts the refrigeration power required since about 500 W of electrical power are consumed to remove one watt from the TF coils at 4 K. This removal efficiency calls for minimizing the maximum nuclear energy deposition in the TF coils to few mW/cm³.

The personnel access to the reactor building within one day after shutdown requires the satisfaction of regulations pertaining to occupational exposure. The federal regulations limit the occupational dose to 5 rem/y, with a maximum of 3 rem/quarter. Occupational exposure based on working 8 h per day and 40 h per week is 2.5 mrem/h. However, the current practice limits worker exposure to 0.5 mrem/h. The DEMO design goal is 0.5 mrem/h in

the reactor building within one day from shutdown.

Each shielding material has certain constraints that must be taken into consideration in the design process. For example, water shielding material requires careful design for the shield system so that the formation of gas pockets from radiolysis is not possible. In addition, special attention must be given to the pH-value and corrosion products. Concrete requires protective measures to avoid change in the performance over the lifetime of the reactor. The following design recommendations¹⁰ adequately protect concrete against loss of water, temperature effects, radiation absorption effects, and stress conditions: a) maximum internal heat generation, 1 mW/cm³, b) maximum temperature gradient, 1 °C/cm, c) maximum internal temperatures, 80°C; and d) maximum ambient temperature, 71°C.

III. ANALYSIS AND TRADEOFF STUDIES

Several analyses and tradeoff studies were performed to define the reference shield system for DEMO. The impact of the inboard shield thickness and material arrangement on the radiation response in the TF coils was analyzed for two shielding options, tungsten and steel-boron carbide. The effect of the boron carbide density factor on the shield response was studied. Several shielding options were analyzed for the outboard shield. The dose equivalent results as a function of the radiation shield thickness are presented. A three-dimensional geometrical model was developed and used for the relativistic electron beam (REB) duct analysis. The radiation responses in different ceramic materials for the REB duct window were calculated as a function of the distance from the first wall.

One-dimensional calculations were performed using discrete ordinate code ANISN with S_8 symmetric angular quadrature set and P_3 legendre expansion for the scattering cross sections. A 67-multigroup cross section set (46 neutrons and 21 photons) collapsed from the CTR Library was used for ANISN calculations. The MACKLIB-IV was employed to calculate the nuclear response functions. DKR radioactivity code was used for activation analysis. A general Monte Carlo code for neutron and photon transport, MCNP was used for the three-dimensional analysis of the REB duct. A continuous energy representation for the nuclear cross sections from ENDF/B-IV was

employed for MCNP calculations. The energy spectrum of the DT source neutrons was sampled for the REB duct analysis. The dose equivalent analysis is based on the neutron and gamma fluxes using the flux-to-dose conversion factor of the American National Standard ANSI/ANS-6.1.1-1977.

IV. INBOARD SHIELD

The inboard blanket and shield thickness (the radial thickness from the edge of the scrape-off region to the position of maximum toroidal field) is 120 cm. This thickness includes 2-cm vacuum gap between the blanket and shield, 5-cm vacuum gap between the shield and the TF coils, and 27 cm for the inboard blanket. In the inboard shield analysis, the inboard blanket employs Li_2O with 0.7 density factor for tritium breeding which results in another 7 cm vacuum in the inboard section of the reactor. Two shielding options were studied for DEMO design, the tungsten option (90% tungsten, 10% water) and the steel option (80% Fe1422, 10% B_4C , and 10% water). The compositions of both options were obtained from previous optimization studies given in References 1, 2, and 11. The thickness of the inboard shield, the section between the inboard blanket and the magnet case, was varied from 64 to 94 cm for both shield options, and a homogeneous composition was assumed. Figure 1 gives a sample of the nuclear responses calculated for the steel and tungsten shield options. All the results are normalized to 2.12 MW/m^2 neutron wall loading or $21.2 \text{ MW}\cdot\text{y/m}^2$ integrated neutron wall loading for the operating life of the DEMO reactor (both loadings at the first wall).

Figure 1 gives the maximum dose in the electrical insulator at the end of the operating life. The tungsten shield option reduces the electrical insulator dose by a factor of ~ 25 compared to the steel option at 79 cm of shield thickness (the reference shield thickness for DEMO). The corresponding maximum nuclear heating in the magnet material is 0.4 MW/cm^3 for the steel option. The tungsten shield option reduces the nuclear heating in the TF coils by a factor of ~ 20 relative to the steel option.

In the above analysis for the inboard shield, a homogeneous composition and theoretical density were assumed for each material. However, practical considerations (cost, fabrication, and engineering) dictate a heterogeneous composition and density factor less than one for boron carbide. The hetero-

Fig. 1. Maximum dose in the electrical insulator as a function of the inboard shield thickness.

geneous arrangement considered for the inboard shield is based on the study in References 1 and 11 for similar shield compositions. The boron carbide density factor was varied from 0.7 to 1.0 in the heterogeneous arrangement, and the impact of this density variation on the shield performance was examined. Figure 2 displays the insulator dose in the magnet material from a heterogeneous shield as a function of the boron carbide density factor. The heterogeneous arrangement for the shield reduces the dose in the insulators, neutron fluence in the TF coils, atomic displacement, and copper-induced resistivity by 26 to 52% relative to the equivalent homogeneous shield assuming a theoretical density for each material. However, the nuclear heating in the TF coils increases by a factor of two. Gamma heating causes this increase due to an increase in the (n,γ) reaction rate. A reduction in the boron carbide density factor increases the radiation response in the TF coils. At 0.7 density factor, all the design criteria are satisfied. The damage in the TF coils in terms of insulator dose, copper-induced resistivity, neutron fluence, or atomic displacement is almost doubled when the density factor of boron carbide is changed from 1.0 to 0.7. The corresponding increase in the nuclear heating is only 40%.

Table 1 gives the key radiation response parameters in the inboard section of the TF

difficulty for the TF coils. The total nuclear heating in the inboard section of the TF coils is < 1.7 kW based on the maximum heating per unit length of the TF coils at the midplane. The maximum dose in the electrical insulator is 1.05×10^9 rad, which is well below the acceptable limit.

V. OUTBOARD SHIELD

In addition to radiation protection of reactor components, the outboard shield was designed with the objective of meeting the dose equivalent requirement to allow for personnel access to the reactor building and reducing the cost of the shield system. Six shield options were considered with different combinations of shielding materials behind a 50 cm Li_2O blanket; borated water, ordinary concrete, heavy concrete, water, Fe1422-ordinary concrete, and H_2O -ordinary concrete. Each shield option consists of one or two basic shielding material (water, ordinary

Fig. 2. Maximum dose in the electrical insulator as a function of the density factor of the B_4C zone in a heterogeneous analysis.

Table 1. RADIATION RESPONSE PARAMETERS IN THE INBOARD PORTION OF THE TF COILS

Maximum fluence in the superconductor material, ($E > 0.1$ MeV)	7.05×10^{17} n/cm ²
Maximum induced resistivity in the copper stabilizer	5.30×10^{-8} $\Omega \cdot \text{cm}$
Maximum atomic displacement in the copper stabilizer	3.59×10^{-4} dPa
Maximum nuclear heating in the superconductor material	8.67×10^{-5} W/cm ³
Maximum nuclear heating in the magnet case	4.63×10^{-4} W/cm ³
Maximum nuclear heating in TF coils including the magnet case	4.18 W/cm
Maximum dose in the electrical insulator	1.05×10^9 rad
Maximum dose in the thermal insulator	3.41×10^9 rad
Total nuclear heating in the inboard shield	17.1 MW

coils for the reference design (2 cm 100% Fe1422, 10 cm 75% B_4C , 25% H_2O , 65 cm 92% Fe1422, 8% H_2O , 2 cm 100% Fe1422, and 27 cm inboard blanket; percentage by volume). The maximum neutron fluence with energy above 0.1 MeV in the superconductor material is 7×10^{17} n/cm², which does not produce significant change in the critical current density or the critical temperature of NbTi alloy.⁹ The induced resistivity in the copper stabilizer which has a maximum of 5.3×10^{-8} $\Omega \cdot \text{cm}$, drops very fast with respect to the depth in the TF coils. This result assumes no magnet annealing during the operating life of the reactor, with a $21.2 \text{ MW} \cdot \text{y/m}^2$ integrated neutron wall loading at the first wall. The TF coil design accommodate this change in the resistivity without difficulty. However, magnet annealing can be used to lower the maximum induced resistivity. The maximum nuclear heating in the superconductor material is 8.7×10^{-5} W/cm³, which does not cause any design

concrete, heavy concrete, or Fe1422), boron as a neutron absorber (B_4C or H_3BO_3), and lead at the outermost surface of the shield to attenuate the decay gamma rays.

The shield thickness required for heavy concrete shield is about two thirds of that for ordinary concrete shield to achieve the same flux level. The heavy concrete installed cost per cubic yard is about three times the ordinary concrete which makes the ordinary concrete shield option more attractive. For 200 cm shield thickness, the borated water reduces the total neutron flux, an order of magnitude less than the ordinary concrete. However, the attenuation performance is reversed with neutron energy above 7 MeV. For example, the 14 MeV neutron flux from the borated water shield is about a factor of two higher than the corresponding value from ordinary concrete. This difference in the attenuation of the high energy neutrons is a

key factor for determining the required shield thickness to satisfy the dose equivalent requirements. Nuclear heating at the surface of the ordinary concrete behind the Li_2O blanket is 0.23 W/cm^3 which exceeds the 1 mW/cm^3 design limit. As a result of this high heat load at the concrete surface, another shielding section before the concrete was considered. Water and Fe1422 were considered for the first section of the shield to protect the concrete from excessive heating. The results show that a front shield thickness of 45 cm reduces the maximum nuclear heating in the ordinary concrete to $\sim 1 \text{ mW/cm}^3$. The maximum nuclear heating is nearly the same whether Fe1422 or H_2O is used in the front section of the shield, but water is preferred from the cost and activation point of view.

Figure 3 shows the maximum dose in the reactor building one day after shutdown as a function of the total shield thickness for the different shielding options. The reference design with water-ordinary concrete option requires a total material shield thickness of 200 cm (2 cm 100% Fe1422 , 25 cm 100% H_2O , 12 cm 42% Fe1422 , 58% B_4C , 10 cm 100% Pb , 140 cm 100% ordinary concrete, 6 cm 42% Fe1422 , 58% B_4C , 5 cm 100% Pb ; percentage by volume) to achieve 0.4 mrem/h one day after shutdown. For the same total shield thickness, a 47 cm of borated water followed by 150 cm of ordinary concrete increases the dose equivalent from 0.4 to 12.5 mrem/h one day after shutdown. This increase in the dose equivalent is due to the drastic change in the neutron spectrum at the outer surface of the shield. The neutron flux is roughly doubled for neutron energy above 2 MeV, while that below 0.1 MeV increases significantly. In addition, the water has a large mean free path for the gamma rays; for example, the required thicknesses of water and ordinary concrete to reduce the 1.25 MeV gamma-ray flux by an order of magnitude are 34 and 16 cm, respectively. Borated water and water shield options require 28 to 30 cm additional shield thickness to get the same dose level as water-ordinary concrete shield option.

The average activation parameters in the outboard shield as a function of the time after shutdown are given in Table 2. The results show that most of the outboard shield material (95% by volume) can be classified as a low level waste based on the radioactivity concentration.

The reference design uses a 150-cm thick wall for building structural purposes which reduces the dose equivalent to 0.1 mrem/h

Fig. 3. Maximum dose equivalent one day after shutdown in the reactor building as a function of shield thickness.

outside the building during operation. This dose represents an upper estimate since the dose calculations ignore the attenuation effect of the different components existing in the reactor building. However, the 0.1 mrem/h satisfied the design criterion discussed before.

VI. PENETRATION SHIELD

Since the penetration shield significantly affects reactor cost, personnel access, and reactor availability, special attention has been given to the main penetrations: (a) the relativistic electron beam (REB) system, (b) the vacuum duct system, and (c) the limiter system. The penetration shield was designed to perform the same functions of the outboard shield.

In order to determine the radiation damage in the ceramic material of the REB system, a three dimensional Monte-Carlo transport calculation was performed for the whole reactor, including the REB annular duct. The general-purpose Monte-Carlo code MCNP was used for the calculation, with a continuous energy representation for the nuclear cross sections. The energy spectrum of 14 MeV neutrons from the DT plasma was considered in the calculations. The REB annular duct has 13-cm inner radius and 26-cm outer radius. The radiation dose

Table 2. AVERAGE ACTIVATION PARAMETERS

Time After Shutdown	Activity		After-heat (% of Thermal Power)	BHP (km ³ air per kW _{th})
	Ci/cm ³	Ci/W _{th}		
A. All the Outboard Shield				
1 min	6.12-2*	1.02-1	1.32-1	7.49
1 d	9.64-3	1.61-2	2.22-3	3.17
1 wk	9.39-3	1.57-2	1.97-3	3.09
1 mo	8.86-3	1.48-2	1.82-3	2.91
1 y	6.12-3	1.02-2	9.25-4	1.46
10 y	5.49-4	9.17-4	4.66-5	9.52-2
100 y	2.65-5	4.43-5	5.44-7	2.19-2
1000 y	2.70-7	4.51-7	2.35-8	4.27-5
B. Ordinary Concrete Shield				
1 min	4.00-5	4.95-5	7.92-5	4.33-3
1 d	3.37-6	4.18-6	1.05-6	2.18-3
1 wk	3.17-6	3.93-6	4.28-7	2.09-3
1 mo	3.09-6	3.82-6	3.65-7	2.03-3
1 y	2.72-6	3.37-6	1.24-7	1.78-3
10 y	2.43-6	3.01-6	5.75-8	1.54-3
100 y	1.23-6	1.52-6	1.84-8	7.57-4
1000 y	3.35-9	4.15-9	4.86-10	9.79-7

* 6.12-2 reads 6.12×10^{-2}

and neutron fluence were calculated for three ceramic materials: magnesium oxide, aluminum oxide, and silicon oxide. The calculations show that locating the ceramic material 270 cm from the first wall results in total dose $< 10^{12}$ rads at the end of the operating life, ($21.2 \text{ MW}\cdot\text{y}/\text{m}^2$ integrated neutron wall loading at the first wall). This dose corresponds to a neutron fluence of $\sim 10^{20} \text{ n/cm}^2$ ($E > 0.1 \text{ MeV}$). At such a level of dose MgO or Al_2O_3 show insignificant change in the structural properties. However, the DEMO design has a 90° bend for the REB duct outside the bulk shield. The bend reduces the dose in the insulator by an order of magnitude. The DEMO reference design locates the ceramic materials after the bend, where the dose is $\sim 10^{11}$ rads at the end of the operating life.

The penetration shield thickness for the REB duct and the vacuum lines were determined based on a similar analysis to that for the INTOR shield system.^{2,11} The shield thicknesses were designed to get a total neutron flux at the outermost surface of $2 \times 10^6 \text{ n/cm}^2\cdot\text{s}$ during operation. This flux value is

corresponding to a biological dose of 0.4 mrem/h after one day from shutdown for water-ordinary concrete shield. The penetration shield uses water-ordinary concrete option with the thickness varies from 65 to 95 cm.

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