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# TFCX SHIELDING OPTIMIZATION*

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ABSTRACT

Design analyses and tradeoff studies for the bulk shield of the Tokamak Fusion Core Experiment (TFCX) were performed. Several shielding options were considered to lower the capital cost of the shielding system. Optimization analyses were carried out to reduce the nuclear responses in the TF coils and the dose equivalent in the reactor hall one day after shutdown. Two TFCX designs with different toroidal field (TF) coil configurations were considered during this work. The materials for the shield were selected based upon tradeoff studies and the results from the previous design studies. The main shielding materials are water, concrete, and steel balls (Fe1422 or Nitronic 33). Small amounts of boron carbide and lead are employed to reduce activation, nuclear heating in the TF coils, and dose equivalent after shutdown.

I. INTRODUCTION

Design analyses and tradeoff studies for TFCX bulk shield are presented in this paper. Several shielding options were analyzed to lower the capital cost of the shielding system. Also, optimization analyses were performed for each option to reduce the nuclear heating in the TF coils and dose equivalent in the reactor hall one day after shutdown. This study was performed for two TFCX designs with different toroidal field coil configurations. The first design has superconductor coils to provide the required field-on-axis, designated superconducting design. The second design utilizes superconductor coils and normal copper insert coils located in the shield to produce the required field, designated hybrid design.

The bulk shield system for both configurations was designed to reduce the radiation leakage from the outer shield surface to an acceptable level. 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 outside the shield system are reduced to avoid high biological dose in the reactor hall after shutdown, and c) the workers and the public are protected from radiation exposure. Also, the bulk shield attenuates the decay gamma rays in the reactor hall to permit personnel access one day after shutdown with all shields in place. This access reduces the reactor downtime and the capital cost of the remote equipment required for maintenance. Also, it is a design criterion for TFCX.[1]

The shielding materials were selected based upon tradeoff studies and the results from the previous design studies.[2,3] Cost, resource, and performance were considered in the selection process. The main shielding materials are water, concrete, and steel balls with low nickel concentration (Fe1422 or Nitronic 33). A small amount of boron carbide with low density is used as a neutron absorber to reduce the activation and nuclear heating in the reactor components. Also, a thin lead layer is used as a gamma ray attenuator at the outermost shield surface to reduce the biological dose in the reactor building and the gamma heating in the reactor components. The low nickel concentration in Fe1422 and Nitronic-33 steel alloys reduces the production of long-lived isotopes, the biological dose after shutdown in the reactor hall, and the steel cost.[4] A description of the shielding system including the design methodology, as well as analyses and tradeoff studies for each shielding component are presented.

II. DESIGN METHODOLOGY

The TFCX coils were designed to last the lifetime of the reactor without change in their performance. Therefore, the shielding system must provides adequate protection for the different coil components from radiation damage and excessive nuclear heating.

For TFCX, the nuclear heating in the superconductor coils is the dominant design criterion for the inboard shield because of the low D-T neutron fluence. This nuclear heating impacts the refrigeration power required since about 500 watts of electrical power are consumed to remove one watt from the coils at 4°K. This removal efficiency calls for minimizing the nuclear energy deposition in the coils. Two values for the maximum nuclear heating in the TF coil winding, 0.3 and 1.0 mW/cm$^3$, were considered during the TFCX design process to assure cryogenic stability and design simplicity for these coils.[1]

The personnel access to the reactor hall within one day after shutdown requires the satisfaction of regulations pertaining to occupational exposure. The regulations[5,6] 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 in the nuclear industry, the exposure policy of the Department of Energy (DOE), and the national laboratories regulations are to reduce radiation exposures as low as reasonably achievable (ALARA). Specifically for facilities being designed, DOE-5480.1 limits onsite workers to less than 1 rem/y (0.5 mrem/h) as a design objective.

Each shielding material has certain physical constraints that must be taken into consideration during the design process. For example, water shielding material requires careful design for the shield system (pipelines, pumps, manifolds, etc.) 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 during operation. Concrete shield requires protective measures to avoid change in the performance over the lifetime of the reactor. The following design recommendations[5] adequately protect concrete against loss of water, temperature effects, radiation absorption effects, and stress conditions: a) maximum heat deposition, 1 mW/cm$^3$, b) maximum temperature gradient, 1°C/cm, c) maximum internal temperature, 80°C, and d) maximum ambient temperature, 71°C. The temperature limitations may be achieved, if it is required, by lining the concrete with a steel jacket and adequately ventilating the shield or cooling the jacket to be independent from the ventilation system for removing the nuclear heat load.

The transport calculations were performed using the discrete ordinate code ANISN[7] 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[8] was used for ANISN calculations. The MACKLIB[9] was employed to calculate the nuclear response functions (nuclear heating, radiation damage, gas production, etc.). For radioactivity and dose equivalent after shutdown, the calculations follow the ANISN-RACC[10]-ANISN path. The ANISN code was first used to obtain the steady state neutron fluxes in each interval of the geometry. These fluxes, after normalization for the proper wall loading, were used by the RACC code to generate the decay gamma source distributions for various operating and decay times. Decay gamma transport calculations were then performed with the ANISN code to obtain the dose equivalent for specific operating and decay time intervals.

## III. INBOARD SHIELD FOR THE HYBRID DESIGN

For the hybrid design, a parametric study was performed for the inboard shield of TFCX. The thicknesses, the compositions, and the geometry model used in the analysis are listed in Table 1. The total thickness of the inboard shield (t), thickness of the boron carbide zone (x), thickness of the lead zone (y), and water concentration in the steel zone (z) were varied in this study. The main objective of this study is to determine the best combination of values for x, y, and z such that the maximum and the total nuclear heating in the inboard portion of the TF coils are minimum for different t values. Other important design criteria, such as maximum doses in the insulator material and the maximum fast neutron fluences in the superconducting material will not be the contrasting requirements in this study because of the short operating time of 2 x 10$^5$ seconds. A small sample of the results will be given here due to the limitation on the paper length. For example, Table 2 gives the maximum nuclear heating in the TF coil winding as a function of the total shield thickness with three lead zone thicknesses of 0, 1, and 2 cm. The boron carbide zone thickness is 3 cm and the water concentration in the steel shield is 10% by volume. The results show that for fixed values of x, y, and z, the maximum nuclear heating in the TF coil winding reduces by about a factor of two every 5 cm of the shield thickness in the steel zone.

Figure 1 gives the maximum nuclear heating in the TF coil winding as a function of the water fraction in the steel shield with the lead zone thicknesses of 0, 1, and 2 cm. The boron carbide zone thickness is 3 cm and the total shield thickness is 70 cm. The maximum nuclear heating in the TF coil winding is at a minimum for a water concentration of around 20% in the steel zone by volume. This minimum also decreases as the lead zone thickness increases from 0 to 2 cm.

The maximum nuclear heating values in the TF coil winding as a function of the lead zone thickness with the boron carbide zone thickness of 0, 1, 2, 3, and 4 cm were calculated. The results show that the maximum nuclear heating in the TF winding is the lowest when

TABLE 1. GEOMETRY AND COMPOSITIONS FOR THE TFCX HYBRID INBOARD SHIELD STUDY

| Zone Description | Radius (cm) | | Thickness (cm) | Composition Volume Percentage |
|---|---|---|---|---|
| | From | To | | |
| TF coil case | 197.5 − t[a] | 202.5 − t | 5.0 | 100% type 304 steel |
| TF coil winding | 202.5 − t | 262.5 − t | 60.0 | 5% NbTi, 23% Cu, 45% type 304 steel, 8% insulator |
| TF coil case | 262.5 − t | 267.5 − t | 5.0 | 100% type 304 steel |
| Gap | 267.5 − t | 272.5 − t | 5.0 | Vacuum |
| Shield jacket | 272.5 − t | 274.5 − t | 2.0 | 100% Fe1422 |
| Lead shield | 274.5 − t | 274.5 − t + y | y[a] | 100% Pb |
| Boron carbide shield | 274.5 − t + y | 274.5 − t + x + y | x[a] | 100% $B_4C$[b] |
| Inner steel shield | 274.5 − t + x + y | 237.5 | t − (37 + x + y) | z[a]% $H_2O$, (100 − z)% Fe1422 |
| Copper coil | 237.5 | 257.5 | 20.0 | 73.7% Cu, 17.4% Fe1422 4% $H_2O$, 4.9% insulator |
| Outer steel shield | 257.5 | 271.5 | 14.0 | z% $H_2O$, (100 − z)% Fe1422 |
| First wall | 271.5 | 272.5 | 1.0 | 50% $H_2O$, 50% Fe1422 |
| Scrape-off | 272.5 | 280.0 | 7.5 | Vacuum |
| Plasma | 280.0 | 440.0 | 160.0 | Vacuum |
| Scrape-off | 440.0 | 447.5 | 7.5 | Vacuum |
| First wall | 447.5 | 448.5 | 1.0 | 50% $H_2O$, 50% Fe1422 |
| Outboard shield | 448.5 | 578.5 | 130.0 | 10% $H_2O$, 10% $B_4C$,[b] 80% Fe1422 |

[a] x, y, z, t are variables.
[b] A 0.7 density factor is assumed.

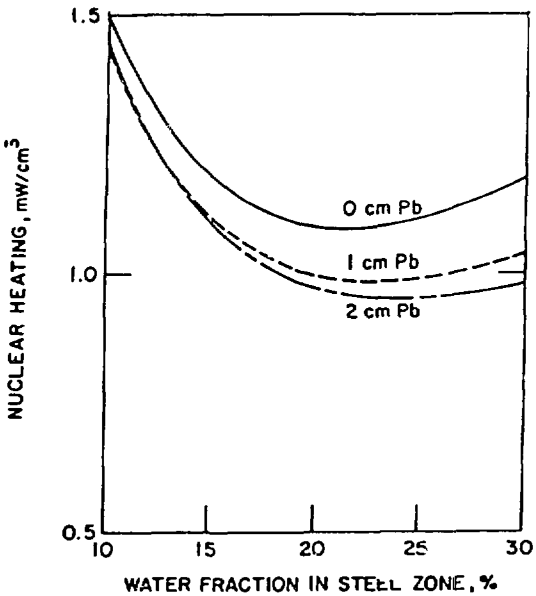


Fig. 1. Maximum nuclear heating in the TF coil winding as a function of the water concentration in the steel shield for different lead zone thicknesses, 70 cm total shield thickness and 3-cm boron carbide zone thickness normalized to 1.8 $MW/m^2$ DT neutron wall loading at the first wall.

TABLE 2. MAXIMUM NUCLEAR HEATING IN THE TF COIL WINDING AS A FUNCTION OF THE TOTAL SHIELD THICKNESS AND THE LEAD ZONE THICKNESS WITH 3 cm BORON CARBIDE ZONE THICKNESS AND 10% $H_2O$ IN THE STEEL SHIELD NORMALIZED TO 1.8 $MW/m^2$ OF NEUTRON WALL LOADING AT THE FIRST WALL

| Shield Thickness (cm) | Lead Zone Thickness (cm) | Neutron Heating ($W/cm^3$) | Gamma Heating ($W/cm^3$) | Total Heating ($W/cm^3$) |
|---|---|---|---|---|
| 60 | 0 | 5.28-4[a] | 5.23-3 | 5.76-3 |
| 60 | 1 | 5.77-4 | 4.96-3 | 5.54-3 |
| 60 | 2 | 6.31-4 | 4.99-3 | 5.62-3 |
| 65 | 0 | 2.62-4 | 2.68-3 | 2.95-3 |
| 65 | 1 | 2.87-4 | 2.54-3 | 2.82-3 |
| 65 | 2 | 3.14-4 | 2.54-3 | 2.86-3 |
| 70 | 0 | 1.30-4 | 1.37-3 | 1.50-3 |
| 70 | 1 | 1.42-4 | 1.29-3 | 1.43-3 |
| 70 | 2 | 1.56-4 | 1.29-3 | 1.44-3 |
| 75 | 0 | 6.42-5 | 6.94-4 | 7.58-4 |
| 75 | 1 | 7.03-5 | 6.52-4 | 7.22-4 |
| 75 | 2 | 7.72-5 | 6.50-4 | 7.27-4 |

[a] 5.28-4 reads $5.28 \times 10^{-4}$.

the boron carbide zone thicknesses is about 1 to 2 cm. Also, the lowest value for the maximum nuclear heating in the TF coil winding occurs with 2 cm lead zone thickness. Combining all the above results, it is evident that the lowest value for the maximum nculear heating in the TF coil winding for a specific shield thickness in the range of 50 to 80 cm occurs with about 20% water concentration in the steel zone by volume, 1 to 2 cm of $B_4C$, and 2 cm of lead.

The nuclear heating in the winding of the TF coils per unit length of the inboard section as a function of the total shield thickness, the water fraction in the steel zone, the lead zone thickness, and the boron carbide zone thickness, are also investigated. Same conclusions about the optimum water fraction in the steel zone and the boron carbide zone thickness can be made from the obtained results. However, the lowest nuclear heating in the winding of the TF coils per unit length of the inboard section is obtained without lead zone as shown in Fig. 2. For the total nuclear heating in the TF coils, winding and case, the use of 2 cm lead zone produces the lowest nuclear heating load. This result is important if two separate coolants with different operating temperature are used to cool the winding and the case separately.

Based on the above study for 0.3 $mW/cm^3$ maximum nuclear heating in the TF coil winding, the inboard shield for the TFCX hybrid design requires a total shield thickness of 80 cm with 20% water in the steel zone by volume, 1 cm layer of $B_4C$, and 2 cm layer of Pb at the back of the shield. The maximum nuclear heating in the TF coil winding with this shield is 0.22 $mW/cm^3$ as shown in Table 3. Other pertinent results for this shield are also given in Table 3. The use of 1.0 $mW/cm^3$ design criterion reduces the shield thickness by 11 cm and increases the nuclear responses in Table 3 by a factor of 5.

For the superconductor design, two shield concepts were considered for the inboard section of TFCX. The first concept is steel type shield which consists of 80% type 316 steel, 20% $H_2O$ with 2 cm layer of boron carbide behind it based on the optimization studies from previous designs.[3,11] For this concept, the total shield thickness is varied from 60 to 80 cm to analyze the radiation response parameters in the TF coils. Also, the impact of moving the boron carbide layer from the back of the shield to the position between the TF coil case and the vacuum vessel was considered. Table 4 gives the geometrical model and the composition of each zone used in this analysis.

The maximum nuclear heating in the TF coil winding, TF coil case, and the vacuum

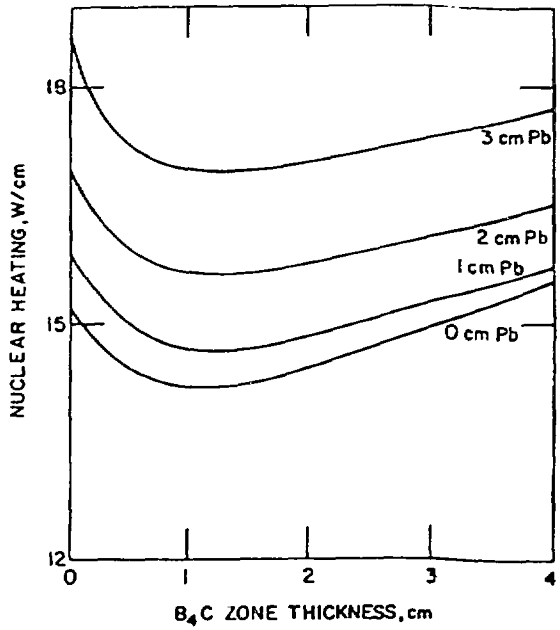


Fig. 2. Nuclear heating in the winding of the TF coils per unit length of the inboard section as a function of the $B_4C$ zone thickness for different lead zone thicknesses with 20% $H_2O$ in the steel shield and 70 cm total shield thickness normalized to 1.8 $MW/m^2$ DT neutron wall loading at the first wall.

TABLE 3. RADIATION RESPONSE PARAMETERS IN THE TF COIL BASED ON 1.8 $MW/m^2$ DT NEUTRON WALL LOADING AT THE FIRST WALL AND 2 x $10^5$ SECONDS OF OPERATION

| | |
|---|---|
| Shield parameters | |
| Total shield thickness, cm | 80 |
| Water concentration in the steel zone, % | 20 |
| Lead zone thickness, cm | 2 |
| Boron carbide zone thickness, cm | 1 |
| Maximum nuclear heating, $mW/cm^3$ | |
| the TF coil winding | 0.22 |
| the TF coil case | 0.43 |
| Total nuclear heating, W/cm | |
| the TF coil winding | 3.50 |
| the TF coil case | 2.03 |
| Maximum insulator dose, rads | |
| the TF coil | 2.34+5 |
| the normal copper coil | 1.68+9 |
| Maximum neutron fluences in the TF coil winding, $n/cm^2$ | |
| E > 0.1 MeV | 2.29+15 |
| E > 0.0 MeV | 3.84+15 |

TABLE 4. INBOARD SHIELD PARAMETERS USED IN THE ANALYSIS FOR THE SUPERCONDUCTOR DESIGN

| Zone Description | Radius (cm) From | Radius (cm) To | Width (cm) | Composition Volume Percentage |
|---|---|---|---|---|
| TF case | 121 | 126 | 5 | 100% type 316 steel |
| TF coil | 126 | 191 | 65 | 5% NbTi, 23% Cu, 8% insulator, 45% type 316 steel |
| TF case | 191 | 196 | 5 | 100% type 316 steel |
| Thermal insulator | 196 | 203 | 7 | 1% insulator |
| Vacuum vessel | 203 | 213 | 10 | 100% type 316 steel |
| Gap | 213 | 216 | 3 | Vacuum |
| Shield jacket | 216 | 218 | 2 | 100% type 316 steel |
| Boron carbide shield | 218 | 220 | 2 | 100% $B_4C$ (0.7 density factor) |
| Steel shield | 220 | 274 | 54 | 80% type 316 steel, 20% $H_2O$ |
| First wall | 274 | 276 | 2 | 50% type 316 steel, 50% $H_2O$ |
| Graphite armor | 276 | 281 | 5 | 100% C |
| Scrape-off | 281 | 294 | 13 | Vacuum |
| Plasma | 294 | 506 | 212 | Vacuum |
| Scrape-off | 506 | 513 | 7 | Vacuum |
| First wall | 513 | 515 | 2 | 50% $H_2O$, 50% type 316 steel |
| Outboard shield | 515 | 645 | 130 | 80% type 316 steel, 10% $H_2O$, 10% $B_4C$ (0.7 density factor) |

vessel are given in Table 5. Case 2 in this table is similar to Case 1 except the boron carbide layer is located between the TF coil case and the vacuum vessel as mentioned before. Comparisons between these two cases, one and two, show that moving the boron carbide layer near the TF coil reduces the nuclear heating in the TF coil winding by 15 to 17%, due to the neutron absorption increase in the boron carbide layer. On the contrary, the nuclear heating in the shield jacket and the vacuum vessel are increased by ~ 50%. The decrease in the nuclear heating of the TF coil winding is considered not sufficient to compensate for the difficulty and the complexity in the design with the boron carbide layer integrated with the vacuum vessel.

The second concept uses single diameter steel balls in a water tank to eliminate the fabrication cost associated with the steel type shield. The geometry and compositions used for this analysis is the same as that of the first concept listed in Table 4. The steel concentration in the shield was varied from 60 to 90% by volume. Figure 3 shows the maximum nuclear heating in the different components of the inboard section based on 1.8 $MW/m^2$ DT neutron wall loading at the first wall with 60 cm of shield. The optimal water fraction based on this analysis is around 20% which is consistent with the previous results for the hybrid design. The increase in the water concentration from 20 to 40% by volume causes about 35% increase in the total nuclear heating in the TF coils and the maximum nuclear heating in the TF coils winding materials. Also, the radiation dose in the insulator materials increases by ~ 40% for the same change in the water concentration. The atomic displacement in the copper stabilizer shows similar behavior to the dose in the insulator materials. These effects result from the increase in the slowing down of the neutrons with energy less than 7 MeV, the increase in the neutron absorption in the boron carbide zone, the neutron spectra hardening in the vacuum vessel and the TF coils.

## IV. OUTBOARD SHIELD DESIGN

The outboard shield design is driven by the biological dose requirements after shutdown and the radiation responses in the outboard section of the TF coils. Several shield options were analyzed with emphasis on reducing the cost of the shield system by using low cost materials.

First, the performance of the optimized steel type shield[11-14] was analyzed as a function of the shield thickness and the operating time. The thickness of the steel zone is varied and the dose equivalent one day after shutdown with type 304 steel for the TF coil case was calculated assuming 1.0 $MW/m^2$ DT neutron wall loading and 1 $MW \cdot y/m^2$ DT neutron fluence at the first wall. The results show that the dose equivalent one day after shutdown decreases by a factor of two for each 5 cm increase in the shield thickness. Figure 4 shows the dose equivalent as a function of the operating time and the time after shutdown with a 130 cm shield thickness. The dose equivalent is 0.3 mrem/h one day after shutdown from the 2 x $10^5$ seconds of operation

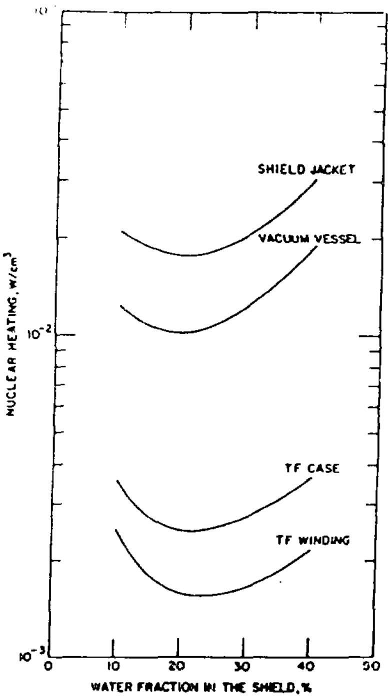


Fig. 3. Maximum nuclear heating in the different components of the inboard section normalized to 1.8 $MW/m^2$ DT neutron wall loading at the first wall with 60 cm total shield thickness.

TABLE 5. MAXIMUM NUCLEAR HEATING IN THE VARIOUS COMPONENTS BASED ON A 1.8 $MW/m^2$ DT NEUTRON WALL LOADING AT THE FIRST WALL

| Case | Shield Thickness (cm) | Neutron Heating ($W/cm^3$) | Gamma Heating ($W/cm^3$) | Total Heating ($W/cm^3$) |
|---|---|---|---|---|
| Maximum Nuclear Heating in the TF Coil Winding | | | | |
| 1 | 60 | 2.00-4 | 1.40-3 | 1.60-3 |
| 2 | 60 | 1.74-4 | 1.15-3 | 1.32-3 |
| 3 | 70 | 5.15-5 | 3.65-4 | 4.17-4 |
| 4 | 80 | 1.32-5 | 9.44-5 | 1.08-4 |
| Maximum Nuclear Heating in the TF Coil Case | | | | |
| 1 | 60 | 2.21-4 | 2.22-3 | 2.44-3 |
| 2 | 60 | 1.95-4 | 2.43-3 | 2.62-3 |
| 3 | 70 | 5.63-5 | 5.85-4 | 6.41-4 |
| 4 | 80 | 1.43-5 | 1.52-4 | 1.67-4 |
| Maximum Nuclear Heating in the Vacuum Vessel | | | | |
| 1 | 60 | 7.16-4 | 9.64-3 | 1.04-2 |
| 2 | 60 | 9.26-4 | 1.49-2 | 1.59-2 |
| 3 | 70 | 1.81-4 | 2.61-3 | 2.79-3 |
| 4 | 80 | 4.57-5 | 6.93-4 | 7.38-4 |

with 1.8 $MW/m^2$ DT neutron wall loading. However, the cost of this type of shield is relatively high.

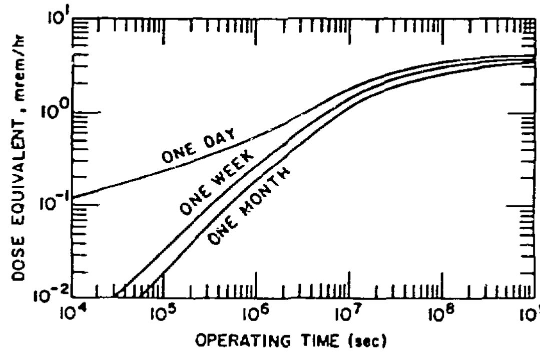


Fig. 4. Dose equivalent one day, one week, and one month after shutdown as a function of the operating time based on 1.8 $MW/m^2$ DT neutron wall loading and 1.3 m of shielding materials.

The use of low cost materials (water, concrete, etc.) require thicker shield to compensate for the difference in the attenuation characteristics, relative to the steel shield. This leads to larger TF coils which increase the cost of the magnetic system. In order to avoid such a case, the outboard shield is divided to two sections. The first section is designed to protect the TF coils from radiation damage and excessive nuclear heating. The second section is located between and outside the TF coils to satisfy the dose criterion.

The steel balls concept is used for the first section of the outboard shield to protect the TF coils based on the previous analysis of the inboard shield. Table 6 gives the geometry and composition for each zone used to perform the analysis for the first section of the outboard shield. The maximum nuclear heating in the outer section of the TF coils was calculated as a function of the shield thickness. About 70 cm shield thickness is required to achieve ~ 1 $mW/cm^3$ maximum nuclear heating in the TF coils.

For the second section of the outboard shield, three options are considered: a) water option (95% $H_2O$, 5% type 316 steel), b) concrete option, and c) steel balls option (60% type 316 steel, 40% $H_2O$). The analyses for these options were carried out to determine the dose equivalent as a function of the shield thickness. Table 7 gives the shield parameters used in the analyses. Figure 5 shows the dose equivalents at the outer surface of the shield one day after shutdown assuming type 316 steel liner for the reactor building and 1 $MW\cdot y/m^2$ D-T neutron fluence at

TABLE 6. GEOMETRY AND COMPOSITIONS FOR THE FIRST SECTION OF THE OUTBOARD SHIELD

| Zone Description | Radius (cm) From | Radius (cm) To | Width (cm) | Composition Percentage Volume |
|---|---|---|---|---|
| TF Case | 121 | 126 | 5 | 100% type 316 steel |
| TF Coil | 126 | 191 | 65 | 5% NbTi, 23% Cu, 45% type 316 steel, 8% insulator |
| TF Case | 191 | 196 | 5 | 100% type 316 steel |
| Thermal Insulator | 196 | 203 | 7 | 1% insulator |
| Vacuum Vessel | 203 | 213 | 10 | 100% type 316 steel |
| Gap | 213 | 216 | 3 | Vacuum |
| Shield Jacket | 216 | 216 | 2 | 100% type 316 steel |
| Boron Carbide Shield | 218 | 220 | 2 | 100% $B_4C$ (0.7 density factor) |
| Steel Balls Shield | 220 | 274 | 54 | 60% type 316 steel, 40% $H_2O$ |
| First Wall | 274 | 276 | 2 | 50% type 316 steel, 50% $H_2O$ |
| Graphite Armor | 276 | 281 | 5 | 100% graphite |
| Scrape-off | 281 | 294 | 13 | Vacuum |
| Plasma | 294 | 506 | 212 | Vacuum |
| Scrape-off | 506 | 513 | 7 | Vacuum |
| First Wall | 513 | 515 | 2 | 50% $H_2O$, 50% type 316 steel |
| Steel Balls Shield | 515 | 565 | 50 | 60% type 316 steel, 40% $H_2O$ |
| Boron Carbide Shield | 565 | 568 | 3 | 100% $B_4C$ (0.7 density factor) |
| Lead Shield | 568 | 573 | 5 | 100% Pb |
| Gap | 573 | 583 | 10 | Vacuum |
| Vacuum Vessel | 583 | 593 | 10 | 100% type 316 steel |
| Thermal Insulator | 593 | 600 | 7 | 1% insulator |
| TF Case | 600 | 605 | 5 | 100% type 316 steel |
| TF Coil | 605 | 665 | 60 | 5% NbTi, 23% Cu, 45% type 316 steel, 8% insulator |
| TF Case | 665 | 670 | 5 | 100% type 316 steel |

the first wall. For water shield, the second shield thickness required to limit the dose equivalent one day after shutdown to 2.5 mrem/h is 120 cm, which makes the total outboard shield 190 cm thick. For a limit of 0.5 mrem/h, the required thickness is 138 cm. For the concrete shield, the required thicknesses are 131 and 151 cm, for the limits of 2.5 and 0.5 mrem/h, respectively. The corresponding numbers for the steel balls option are 86 and 98 cm, respectively. Table 8 summarizes these results. For $2 \times 10^5$ seconds of operation with 1.8 $MW/m^2$ DT neutron wall loading, the required shield thicknesses in Table 8 are reduced by 17, 24, and 27 cm for the steel balls, water, and concrete option, respectively.

## V. CONCLUSIONS

Several low cost shielding options were developed for the two TFCX designs, the hybrid and the supercondutor.

For the inboard section, the use of 80 vol% steel - 20 vol% water shield composition with 1-cm boron carbide layer at the back of the shield results in the lowest value for the total nuclear heating in the winding of TF coils. Increasing the water concentration to 40 vol% for reducing the shield cost (steel balls in a water tank) results in about 35% increase in the total nuclear heating in the winding of the TF coils and the maximum nuclear heating in the winding materials of the TF coils.

For the outboard section, the shield is divided to two regions. The first region uses the 60 vol% steel-40 vol% water mixture to protect the outboard portion of the TF coils. The second region is located between and outside the TF coils to achieve the 0.5 mrem/hr dose equivalent in the reactor hall one day after shutdown. Water or ordinary concrete is the main material for the second region of the shield. The use of any combination of the low cost shield options studied in this paper for TFCX reduces the cost of the shield to less than 4% of the total direct cost[15] compared to 8 to 16% for other fusion reactor design studies.[3,4]

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TABLE 7. GEOMETRY AND COMPOSITION FOR THE OUTBOARD SHIELD

| Zone Description | Thickness, cm | Composition Percentage Volume |
|---|---|---|
| First Wall | 2 | 50% $H_2O$, 50% Type 316 Steel |
| Steel Balls Shield | 60 | 40% $H_2O$, 60% Type 316 Steel |
| Boron Carbide Shield | 3 | 100% $B_4C$ (0.7 DF) |
| Lead Shield | 5 | 100% Pb |
| Biological Shield | Variable | 95% $H_2O$, 5% Type 316 Steel, or 100% Concrete, or 40% $H_2O$, 60% Type 316 Steel |
| Boron Carbide Shield | 3 | 100% $B_4C$ (0.7 DF) |
| Lead Shield | 5 | 100% Pb |

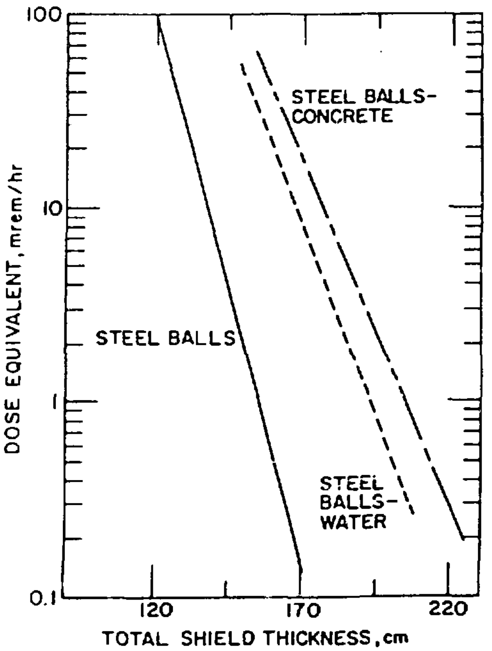


Fig. 6. Dose equivalent one day after shutdown based on 1 $MW \cdot y/m^2$ DT fluence at the first wall.

TABLE 8. TOTAL OUTBOARD SHIELD THICKNESS AS A FUNCTION OF THE DOSE CRITERION FOR THE DIFFERENT SHIELD OPTIONS CALCULATED FOR 1 $MW \cdot y/m^2$ D-T NEUTRON FLUENCE AT THE FIRST WALL

| Shield Option | Shield Thickness for 2.5 mrem/h, cm | Shield Thickness for 0.5 mrem/h, cm |
|---|---|---|
| Steel Balls | 156 | 168 |
| Steel Balls - Water | 190 | 208 |
| Steel Balls - Concrete | 201 | 221 |


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