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NEUTRONICS ACTIVITIES FOR NEXT GENERATION DEVICES*

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ABSTRACT

Neutronics activities for the next generation devices are the subject of this paper. The main activities include TFCX and FPD blanket/shield studies, neutronics aspects of ETR/INTOR critical issues, and neutronics computational modules for the tokamak system code and tandem mirror reactor system code. Trade-off analyses, optimization studies, design problem investigations and computational models development for reactor parametric studies carried out for these activities are summarized.

I. INTRODUCTION

Parametric studies for TFCX were performed to define shielding materials, compositions, and thicknesses based on design requirements, performance and cost. The nuclear responses in the toroidal field (TF) coils and the dose equivalent one day after shutdown in the reactor hall are the indicators for the shielding performance. Several shielding materials were considered in the analysis, e.g., W, steel, TiH_2, H_2O, concrete, Pb, and B_4C. Activation analysis and afterheat of four design options were performed for waste and site studies. Hot spots in the TF coils due to neutrons and photons streaming between the torus segments were analyzed for the TFCX inboard section. Design solutions were implemented to reduce the impact on the TF coils.

FPD neutronics activities concentrated on three areas: a) central cell blanket and shield trade-off studies to reduce the capital cost, b) detailed analysis for the whole reactor to define the neutron leakage to the end cell, net tritium breeing ratio, and energy deposition in the different reactor components, and c) end cell analysis to determine neutron wall loading, heat deposition in the Cee coils, the maximum nuclear responses in the Cee coils, and the performance of steel versus tungsten for Cee coil shielding as a function of the shield thickness.

Three neutronics studies were performed for ETR/INTOR critical issues. The shield requirements and the building wall thicknesses for different design criteria with respect to personnel access and maintenance scenario were determined to assess the impact on the reactor capital cost for the ETR/INTOR critical issue D. The nucleonics aspects of the plasma stabilization elements for the transient electromagnetics ETR/INTOR critical issue C were analyzed. The radiation damage in the insulator materials, the induced resistivity in the conductor materials, and nuclear responses in these elements were assessed as a function of the position relative to the first wall. Neutronics aspects of two radiofrequency launchers were analyzed in detail to determine the replacement frequency for the BeO windows and the shielding requirements for ETR/INTOR critical issue B.

Neutronics computational modules were developed for a tokamak reactor without tritium breeding blanket. The models estimate the performance parameter of two types of shields (steel and tungsten) and the nuclear responses in the TF coils. The modules have been integrated into the tokamak system code at FEDC and were used for TFCX study. For tandem mirror reactors, three computational modules were developed for the central cell, the choke area, and the end cell. The central cell module predicts the performance parameters for self-cooled liquid metal (Li or 17Li83Pb) blankets and the required shield to protect the central cell coils and the workers in the reactor hall one day after shutdown. The module calculates blanket and shield dimensions, composition and energy deposition in each zone of the central cell, tritium breeding ratio, radiation responses in the central cell coils, and cost of each component. This module can be used in two modes: a) perform the analysis for a specific blanket and shield design, or b) minimize, maximize, or use specific values for one or more of seventeen blanket and shield variables to define the blanket and shield configuration. The choke coil module calculates the radiation responses in the choke coil (normal copper and superconductor) and estimates the shield requirements for the superconductor coil based on the allowable nuclear responses. The end cell module has the same functions as the choke coil for each Cee coil and calculates the neutron wall loading distribution in the end cell.

This paper gives a summary for each activity including the key results and the main conclusions obtained during the last fiscal year.

II. TFCX SHIELDING ANALYSES

Design analyses and tradeoff studies were performed for the bulk shield of the Tokamak Fusion Core Experiment (TFCX). 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. 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.

A. Inboard Shield

For the hybrid design with the steel shield option, i.e., steel/H_2O/B_4C/Pb, a parametric study was performed for the inboard shield to reduce the nuclear responses in the TF coils.[1-3] 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. The main objective 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 nuclear responses, such as the maximum dose in the insulator material and the maximum fast neutron fluence in the winding material do not represent a concern for TFCX because of the short operating time of $2x10^5$ seconds. However these responses behave similar to the nuclear heating with respect to the composition and the thickness of the shield. As an example of the results of this study, Fig. 1 gives the maximum nuclear heating in the TF coil winding as a function of the water concentration in the steel shield with 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 boron carbide and the lead are located between the TF coil and the steel shield. The conclusions of this study[1] are the following: a) 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; b) this minimum reaches the lowest value with 1 to 2 cm of boron carbide and 2 cm of lead; c) at this composition, the lowest nuclear heating in the TF coil, winding and case, per unit length of the inboard section is obtained; and d) the elimination of the lead zone from this composition produces the lowest nuclear heating in the winding material of the TF coil per unit length of the inboard section. However, the total nuclear heating in the TF coil is slightly increased because more photons reach the TF coil case. The last result is important if two separate coolants with different operating temperatures are used to cool the winding and the case separately.

For the superconducting design, three shielding options were considered for the inboard section of TFCX. The first option is steel type shield which consists of 80% steel, 20% water with a 2 cm layer of boron carbide behind it based on the optimization studies performed for INTOR.[4,5] For this option the shield thickness was varied from 0 to 80 cm to analyze the radiation response in the TF coils. The steel concentration was also varied from 60 to 90%. The shield with a 60% steel concentration represents single diameter steels balls in a water tank which eliminates the fabrication cost associated with the steel type shield.[2] Fig. 2 shows the maximum nuclear heating in the components of the inboard TF coils based on 1.8 MW/m^2 DT neutron wall loading at the first wall and 60 cm total shield thickness. The increase in the water concentration from 20 to 40% by volume causes about a 35% increase in the total nuclear heating in the TF coil and the maximum nuclear heating in the TF coil winding materials. Also, the radiation dose in the insulator materials increases by ~40% for the same change in the water concentration. The atomic displacement values in the copper stabilizer show similar behavior to the dose in the insulator materials.

The second option is a tungsten type shield which consists of 80% W, 10% H_2O, and 10% steel. The motivation is to reduce the shield thickness required to achieve specific nuclear responses in the TF coils. As an example, two nuclear responses from the tungsten and the steel options are shown in Figs. 3 and 4 assuming theoretical densities for all materials. The results in these figures are based on separate calculations for each shield thickness (0, 10, 20, 40, and 60 cm) since the attenuation coefficients for the different nuclear responses are functions of the shield thickness. Compared to a 60 cm steel shield thickness, the tungsten option requires only 45 to 48 cm to achieve the same nuclear responses in the TF coils. This represents

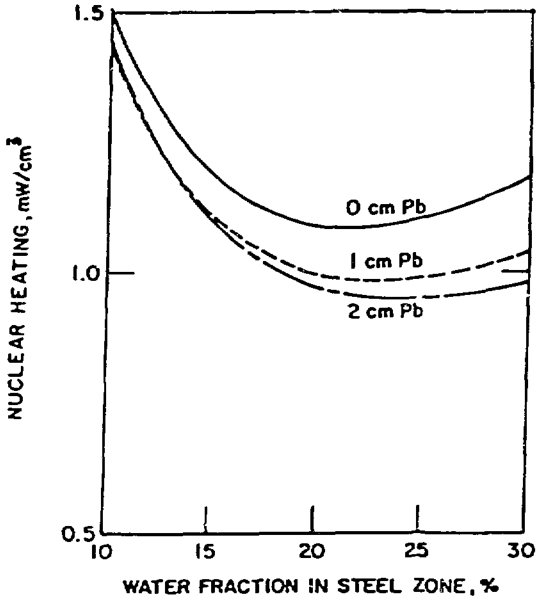


Fig. 1. Maximum nuclear heating in the TF coil winding as a function of the water concentration in the steel shield for different lead zone thicknesses, 70 cm total shield thickness and 3-cm boron carbide zone thickness normalized to 1.8 MW/m^2 DT neutron wall loading at the first wall.

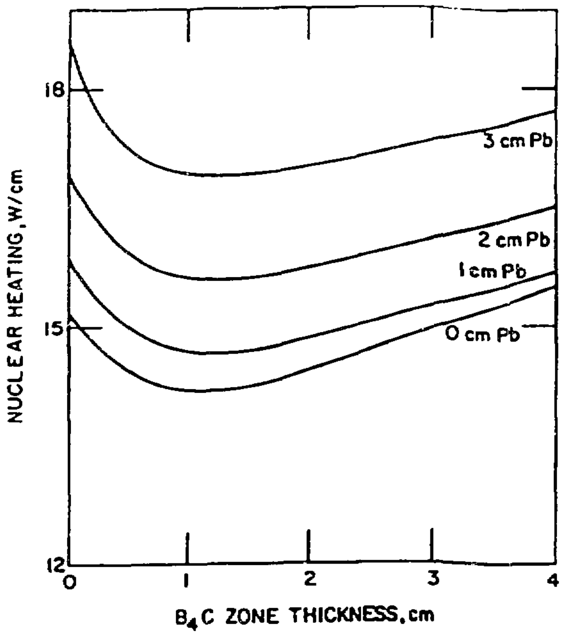


Fig. 2. Nuclear heating in the winding of the TF coils per unit length of the inboard section as a function of the B_4C zone thickness for different lead zone thicknesses with 20% H_2O in the steel shield and 70 cm total shield thickness normalized to 1.8 MW/m^2 DT neutron wall loading at the first wall.

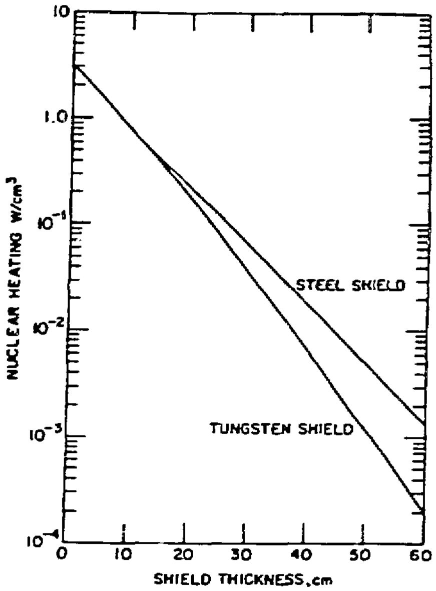


Fig. 3. Maximum nuclear heating in the TF coil winding normalized to 1 MW/m^2 neutron wall loading.

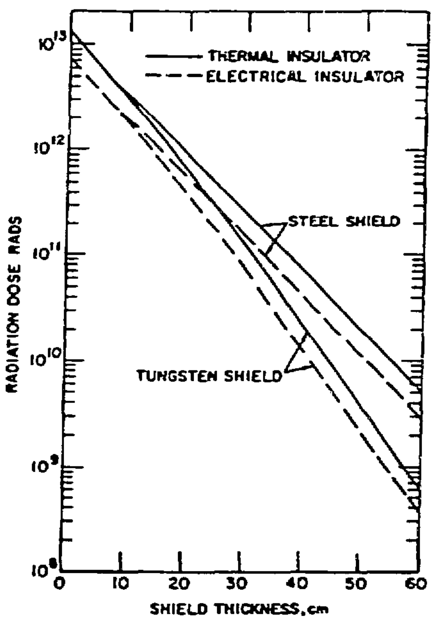


Fig. 4. Maximum radiation dose in the insulator materials normalized to 1 MW·y/m^2 DT neutron fluence at the first wall.

about 20 to 25% reduction in the inboard shield thickness for specific responses in TF coils. However, this reduction diminishes as the shield thickness becomes smaller. At a 20 cm inboard shield thickness, the difference between both options is not worth mentioning as shown in Figs. 3 and 4. Important factors in the comparison between the two options are the obtainable density and the high cost of the tungsten metal. For the same shield thickness, the cost of the tungsten option is ~ four times the steel. However, tungsten metal is the preferred option from the radioactivity point of view. Also, it should be noted that the use of 1 to 2 cm of boron carbide between the tungsten shield and the TF coil reduces the nuclear heating in the winding material by a factor of 2 to 2.7 as the total shield thickness changes from 25 to 75 cm.

The third option has the tungsten option with a layer of hydrogeneous material (H_2O or TiH_2) between the boron carbide and the tungsten shield to benefit from the high hydrogen content. The optimization analysis for this option gave the following conclusions: a) for the same nuclear responses in the TF coils, this shield option reduces the required thickness by 4 to 9 cm, relative to the second option; b) about 5 to 10 cm of a hydrogeneous layer (H_2O or TiH_2) gives the lowest nuclear responses in the TF coils for a specific shield thickness; and c) the use of 1 to 2 cm of boron carbide before the TF coil is beneficial for this improvement. The largest improvement occurs in the fast neutron fluence and the copper atomic displacement. Similar results were obtained in the blanket and shielding study of high power density tokamak reactor concepts.[6]

B. TFCX Outboard Shield

The TFCX 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 shielding system by using low cost materials.

The use of low cost materials (water, concrete, etc.) require a 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 magnet system. In order to avoid this situation, 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. The maximum nuclear heating in the outer section of the TF coils was calculated as a function of the shield thickness. About 70 cm of shield thickness is required to achieve about 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, b) concrete option, and c) steel balls option. The analyses for these options were carried out to determine the dose equivalent as a function of the shield thickness. Table 1 summarizes the results based on 1 MW·y/m^2 DT neutron wall at the first wall.

TABLE 1. TOTAL OUTBOARD SHIELD THICKNESS AS A FUNCTION OF THE DOSE CRITERION FOR THE DIFFERENT SHIELD OPTIONS CALCULATED FOR 1 MW·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

For 2×10^5 seconds of operation with 1.8 MW/m^2 DT neutron wall loading, the required shield thickness in Table 1 are reduced by 17, 24, and 27 cm for the steel balls, water, and concrete option, respectively.

C. Hot Spots in the Inboard Section of the TFCX Toroidal Field Coils

The TFCX conceptual designs call for the construction of the reactor torus through the use of "pie-shaped" segments for mechanical and maintenance considerations. The use of this concept results in hot spots in the inboard section of the toroidal field (TF) coils due to neutrons and photons streaming through the slots between the segments. The analysis was performed to study the effect on the nuclear responses in the TF coils and introduce design solutions to reduce the impact on the reactor design.[7]

Two types of slots, straight and single bend, with various widths (3, 5, and 7 cm) were considered. A sketch of the single bend configuration with a 3 cm width is shown in Fig. 5. A volumetric neutron source located between R=294 cm and 506 cm is used to model the DT neutrons from the plasma. The bulk shield consisted of a homogeneous mixture of 80% type 316 steel and 20% water based on previous optimization studies. A layer of boron carbide shield is located before the vacuum vessel to reduce the TF nuclear responses. The multigroup Monte Carlo code MORSE was employed for the calculations. An isotropic distribution was assumed for the DT

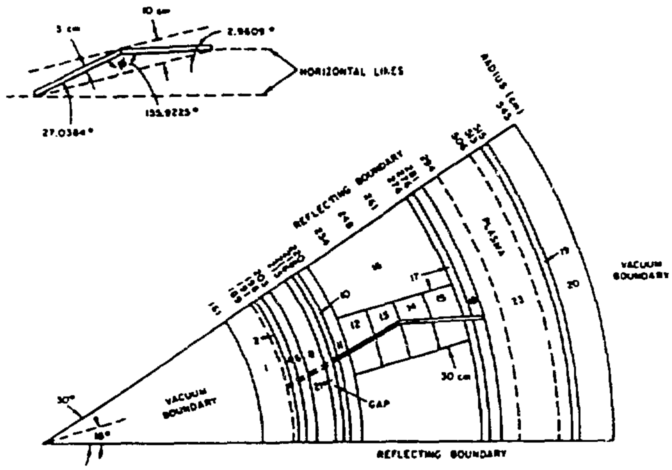


Fig. 5. Sketch of the single-bend slot model.

neutrons with spatial biasing to increase the number of the DT neutrons generated near the slot area. Geometric splitting and Russion Roulette were employed to increase the probability of neutrons passing through the inboard shield and the extremely narrow slots. A 67 group (46 neutron and 21 gamma groups) cross section set based on ENDF/B-IV was used in the calculations. Neutron and gamma kerma factors were included in the cross section set to generate the nuclear responses in the different reactor components. Nuclear heating results of these MORSE calculations relative to that of the no slot case usually used in the design process are given for the slot facing regions in the TF case and the front 2 cm of the TF coil winding materials in Table 2. As expected, the nuclear heating increases as the slot width increases or the slot bending angle decreases. These small slots increase the nuclear heating in the TF case and winding materials by one to two orders of magnitude which has a strong impact on TF coil design and performance. The 24° single bend slot reduces the streaming effect by a factor of 2 to 4 for the case and the winding, respectively, relative to the straight slot of the same width. The nuclear heating can further be reduced by increasing the bending angle, however, this will necessitate the use of a wider slot to enable the radial replacement of the individual segment. Also, the nuclear heating in the section of the TF coil which does not face the slot increases by factors of 5, 14, and 26 relative to the no slot case for the three straight slots of this study. This increase has a significant impact on the refrigeration power requirements.

This study demonstrates the strong effect of narrow slots on the TF coils and the effectiveness of bending the slot to reduce its impact on the design. The effect on other

TABLE 2. NUCLEAR HEATING FACTORS[a] IN THE TOROIDAL FIELD COIL

Slot		Nucl. Htg. Factor	
Type	Width (cm)	TF Coil Case	TF Coil Winding Material
Straight	3	16.5	10.0
Straight	5	37.8	30.0
Straight	7	78.0	59.9
24° Single bend	3	4.5	4.2
24° Single bend	5	17.9	19.9
24° Single bend	7	26.7	20.9

[a] Nuclear heating factor is defined as the ratio between the maximum nuclear heating due to the slot divided by the nominal nuclear heating without slot in the same reactor component.

nuclear responses such as radiation doses in the insulating material, neutron fluence in the winding materials, and atomic displacement in the copper stabilizer are similar to that of the nuclear heating.

III. FPD BLANKET AND SHIELDING ANALYSES

Neutronics analyses for FPD-I, -II, and -III (Fusion Power Demonstration reactor) of the mirror program were performed for the central cell, choke area, and the end cell. The central cell analysis includes a blanket optimization to reduce the capital cost, a shield design for the protection of the central cell coils, and a biological shield design for the reactor room access one day after shutdown. The choke area analysis was carried out to define the nuclear responses in the copper insert coil and the superconductor portion of the coil. Rigorous analysis was performed for the end cell since the capital cost of the end cell amounts to ~60% of the total capital cost. An elaborate three-dimensional (3-D) radiation transport analysis for the complete reactor system was carried out which includes the breeding blanket, the bulk shield, the central cell coils, the choke coils, the Cee coils, and the biological shield. The general purpose Monte-Carlo code MCNP was used for the calculations with a continuous energy representation for the nuclear cross sections based on ENDF/B-V nuclear data files. The energy spectrum and spatial distribution of the neutron source from the D-T plasma in the central cell and the end cell were modeled explicitly in the calculations. The Cee coil geometry was modeled rigorously without approximation by using the MIG code.[8] Results from the 3-D analysis include the net tritium breeding ratio, energy deposition in each component of the reactor, radiation damage parameters in each coil, neutron leakage from the central cell to the end cell and the detailed shield design for the end cell.

A. Central Cell Analysis

The central cell analysis covers the following three sections: a) the tritium breeding blanket, b) the shielding design to protect the central cell coils, and c) the biological shield. In the design process for the three sections, the emphasis was to reduce the capital cost and the total weight of the central cell. The blanket is designed to convert the kinetic energy of the DT neutrons to recoverable heat and produce adequate tritium breeding to supply the tritium fuel requirement during the whole reactor lifetime as well as generate surplus of tritium to compensate for operating with non-breeding blanket sections or to start another reactor within a reasonable period of time. Economic and shielding considerations require the blanket design to maximize the recoverable heat produced which is defined as the energy deposited in the first wall, breeding zone, and reflector per fusion neutron. Neutronics analysis was performed to study the performance of the self-cooled natural lithium-lead (17Li83Pb) blanket concept. The impact of the breeding zone thickness, reflector material selection, reflector composition, and reflector zone thickness were analyzed to determine the main blanket parameters. In the analyses, the five key parameters considered to define the blanket for FPD are the capital cost, the total weight, the tritium breeding ratio, the energy deposition in the blanket per fusion neutron, and the energy loss to the shielding system. The analysis was performed for a wide range for each key parameter. Table 3 gives the central cell parameters which resulted from this analysis.

The shielding thickness for the central cell was defined to achieve 0.1 mW/cm^3 maximum nuclear heating in the coil case. At this nuclear heating rate, the capital cost for the central cell is minimum based on the MFTF-α+T design analysis. The design criterion based on the nuclear heating and the five year operating time (100% availability) produce low nuclear responses in the superconductor coils as shown in Table 3. The design criteria for superconductor coil protection against radiation is about two orders of magnitude higher than the values in Table 3.

B. Choke Coil Analysis

The choke coil analysis was performed to define the nuclear responses in both sections of the choke coil; the normal copper insert and the superconductor. Based on the nuclear responses given in Table 3, the copper section can operate for 1.7 years (100% availability) to reach 10^{13} rads in the ceramic insulator which is corresponding to 3 vol% swelling in the $MgAl_2O_4$. The superconductor section has an adequate margin to operate for five years (100% availability) without reaching any radiation damage design limit.

C. End Cell Analysis

An accurate 3-D analysis was performed for the whole reactor to provide the following information: a) the neutron wall loading in the end cell, b) the energy deposition in each component, c) the net tritium breeding ratio, d) the neutron leakage from the central cell to the end cell, and e) the hot spot points in the Cee coils. The 3-D geometrical model for the calculations is shown in Fig. 6. The key results from this analysis are summarized in this section. The neutron wall loading in the end cell has a 0.15 MW/m^2 peak value at Z = ±46.1 m where Z is measured from the middle of the central cell. The neutron wall loading distribution shows that the Cee coils shield thickness can be tapered to less than the 35 cm starting from Z = ±54 m. Table 4 gives the energy deposition in each component per fusion neutron and the statistical error within one standard deviation. The energy deposition in the Cee coils and the end cell biological shield are 90.4 and 35.6 kw, respectively. The shield thickness for the Cee coils in the 3-D analysis is 35 cm. For the FPD design, it is possible to use 45 cm of shield instead of 35 except for a few spots, which reduces the energy deposition in the Cee coils to about 26.4 kw. The point with minimum space for shielding has an 8.6 mW/cm^3 maximum nuclear heating in the coil case (1.5 cm coil case thickness and 1.5 cm gap were assumed in the 3-D analysis) which can be accommodated in the Cee coil design. The corresponding dose in the electrical insulators is 1.1×10^{10} rads per full power year.

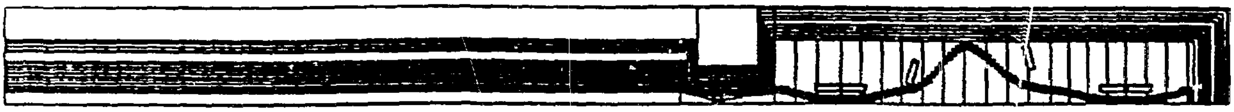
IV. NEUTRONICS ASPECTS FOR ETR/INTOR CRITICAL ISSUES

A. Personnel Access and Remote Maintenance

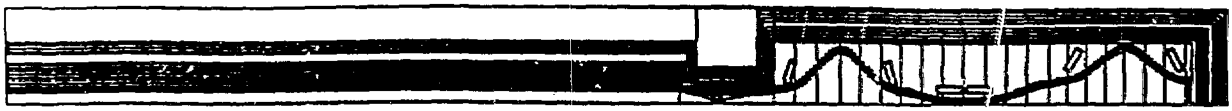
The INTOR shielding system was designed to achieve a 2.5 mrem/h dose equivalent in the reactor hall one day after shutdown with all shields in place.[4,5] This dose level permits a hands-on mode of operation in the reactor hall. A neutronics study was performed to define the shield and the reactor building wall thicknesses for three different configurations. Two configurations permit personnel access to the reactor hall one day after shutdown, and the third configuration assumes all-remote maintenance. Two dose equivalent values are considered for the personnel access configurations, 2.5 and 0.5 mrem/h. The 2.5 mrem/h represents the limit for the occupational exposure based on working

TABLE 3. FPD-I CENTRAL CELL AND CHOKE COIL PARAMETERS

Central Cell Blanket Parameter	
Materials	
Natural lithium-lead (17Li83Pb) breeder	
Carbon reflector	
Ferritic steel structure	
Dimensions and Composition in the Neutronics Model	
First wall thickness (50% ferritic, 50% 17Li83Pb), cm	1
Breeder zone thickness (7.5% ferritic, 92.5% 17Li83Pb), cm	29
Reflector zone thickness (10% ferritic, 10% 17Li83Pb, 80% C), cm	40
Breeding blanket length, m	70
Neutron wall loading, MW/m^2	1.95
Performance Parameters	
Tritium breeding ratio	1.11
Lithium-6 enrichment	Natural
Blanket energy multiplication factor	1.30
Total energy multiplication factor (blanket and shield)	1.37
Energy fraction deposited in the shield	4.83×10^{-2}
Central Cell Shield	
Materials	
Steel balls, water, B_4C (powder), lead	
Dimension and Composition Under the Coils	
Steel shield (60% Fe1422 steel alloy, 40% H_2O), cm	45
Boron shield (42% B_4C, 20% Fe1422 steel alloy, 20% H_2O), cm	5
Nuclear Responses in the Central Cell Coils	
Maximum nuclear heating in the coil case, mW/cm^3	0.10
Maximum nuclear heating in the winding, mW/cm^3	0.03
Maximum dose in the thermal insulator, $rads/MW \cdot y/m^2$	3.53×10^7
Maximum dose in the electrical insulator, rads, $MW \cdot y/m^2$	1.22×10^7
Maximum dpa in the copper stabilizer, $dpa/MW \cdot y/m^2$	3.44×10^{-6}
Fast neutron fluence in the winding material, $n/cm^2/MW \cdot y/m^2$	4.54×10^{15}
FPD-I Choke Coil	
Copper Insert	
Neutron wall loading, MW/m^2	0.6
Maximum nuclear heating in the coil case, W/cm^3	7.0
Maximum nuclear heating in the copper coil, W/cm^3	5.3
Maximum dose in the ceramic insulator, $rads/MW \cdot y/m^2$	6.1×10^{12}
Maximum atomic displacement in the copper, $dpa/MW \cdot y/m^2$	6.8
Superconductor	
Neutron wall loading, MW/m^2	0.8
Minimum shielding thickness, cm	70.0
Maximum nuclear heating in the coil case, mW/cm^3	0.46
Maximum nuclear heating in the winding materials, mW/cm^3	0.16
Maximum dose in the insulator, $rads/MW \cdot y/m^2$	6.05×10^8
Maximum dose in the electrical insulator, $rads/MW \cdot y/m^2$	2.53×10^8
Maximum fast neutron fluence in the winding material, $n/cm^2/MW \cdot y/m^2$	1.56×10^{17}
Maximum atomic displacement in the copper stabilizer, $dpa/MW \cdot y/m^2$	1.06×10^{-4}



X - Z Plane



Y - Z Plane

Fig. 6. Three dimensional geometrical model.

TABLE 4. NUCLEAR HEATING AND POWER DISTRIBUTION
Nuclear Heating Per Fusion Neutrin
(MeV/DTn)

	n	γ	Total
Blanket			
First wall	0.3288 ± 0.51%	0.5991 ± 1.09%	0.9279
Breeding zone	4.8433 ± 0.63%	5.8394 ± 0.60%	10.6827
Reflector	2.9348 ± 0.69%	2.4177 ± 0.94%	5.3525
Total	8.1069	8.8562	16.9631
Central Cell Coil Shield			
Steel shield	1.4112-2 ± 3.86%	1.0859 ± 11.52%	1.1000
B_4C shield	6.2042-6 ± 10.58%	8.6578-6 ± 11.52%	1.4862-5
Total	1.4118-2	1.0859	1.1000
Central Cell Coil			
Coil case	2.2878-7 ± 17.12%	7.0058-6 ± 14.94%	7.2346-6
Coil winding	7.1454-7 ± 14.67%	1.0479-5 ± 23.06%	1.1194-5
Total	9.4332-7	1.7485-5	1.8428-5
Central Cell Biological Shield			
B_4C shield	8.6626-8 ± 13.7%	5.4116-8 ± 15.0%	1.4074-7
Pb shield	62.417-9 ± 23.9%	4.5354-8 ± 18.1%	4.5978-8
Total	8.7250-8	9.9470-8	1.8672-7
Choke Coil Shield			
Steel shield	7.3229-2 ± 2.26%	4.5173-1 ± 2.29%	5.2496-1
B_4C shield	1.2498-7 ± 29.3%	1.3163-7 ± 34.6%	2.5661-7
Pb shield	2.4027-9 ± 80.1%	6.3297-8 ± 29.6%	6.5700-8
Total	7.3229-2	4.5173-8	5.2496-1
Copper Choke Coil			
Coil case	1.1434-3 ± 6.28%	4.5239-3 ± 6.13%	5.6673-3
Coil winding	2.0223-3 ± 5.05%	2.0665-3 ± 9.06%	4.0888-3
Total	3.1657-3	6.5904	9.7561-3
Superconductor Choke Coil			
Coil case	2.2060-6 ± 30.1%	8.7658-5 ± 27.8%	8.9864-5
Coil winding	1.5796-6 ± 26.1%	2.7122-5 ± 20.0%	2.8702-5
Total	3.7856-6	1.1478-4	1.1857-4
Cee Coils Shield			
Steel shield	3.9440-2 ± 2.26%	1.6293-1 ± 2.58%	2.0237-1
B_4C shield	1.7490-3 ± 4.36%	1.6874-3 ± 11.2%	3.4360-3
Total	4.1189-2	1.6462-1	2.0580-1
Cee Coils			
Coil cases	2.3006-5 ± 7.87%	4.9066-4 ± 3.92%	5.1366-4
Coil winding	1.7444-4 ± 5.78%	1.6501-3 ± 4.01%	1.8245-3
Total	1.9745-4	2.1408-3	2.3382-3
End Cell Biological Shield			
Water shield	9.1126-4 ± 7.33%	1.1430-5 ± 33.5%	9.2269-4
B_4C shield	2.4537-10 ± 38.6%	1.1064-6 ± 78.4%	1.0166-6
Pb shield	3.1575-12 ± 58.5%	4.0372-7 ± 55.6%	4.0372-7
Total	9.1126-4	1.2850-5	9.2411-4

Power Parameters

Total DT neutron power, MW	542.55
DT neutron power in the end cell, MW	5.54
Neutron wall loading at first wall, MW/m^2	1.95
Neutron leakage from central cell, n/DTn	6.50×10^{-4}

Power Distribution

	MeV/DTn	MW
Blanket	16.96	654.86
Central cell coil shield	1.10	42.47
Central cell coil	1.84-5	7.10-4
Central cell biological shield	1.87-7	7.22-6
Choke coil shield	5.25-1	20.27
Copper choke coil	9.76-3	3.76-1
Superconductor choke coil	1.18-4	4.56-3
Cee coils shield	2.06-1	7.95
Cee coils	2.34-3	9.04-2
End cell biological shield	9.24-4	3.56-2

8 h per day during the whole year. The 0.5 mrem/h satisfies the ALARA and DOE design guideline for occupational exposure. The remote configuration design is based on providing adequate protection for the toroidal field coils without any consideration for the dose equivalent in the reactor hall. For each configuration, the thickness of the reactor building wall was designed to attenuate the dose equivalent to 0.05 mrem/h during operation. This dose level satisfies the proposed radiation dose guideline in the "Clean Air Act, Radiological Emission Standards," which requires a 10 mrem/y on the site boundary.

Table 5 gives the shielding requirements for the three configurations. The key results from this study are the following: a) For all-remote operation, the radioactivity concentrations in the reactor components outside the bulk shield are increased by four to five orders of magnitude compared to the personnel access configurations, and b) Only a small difference in the overall cost of the reactor is observed among the different configurations.[9]

TABLE 5. INTOR SHIELDING REQUIREMENTS FOR THE DIFFERENT CONFIGURATIONS

Shield Design Criteria	Outboard Shield Thickness, cm	Wall Thickness cm
0.5 mrem/hr	116	172
2.5 mrem/h	105	198
All-remote operation	50	326

B. Nuclear Responses in the Plasma Stabilization Elements

A study was carried out to determine the nuclear responses in the plasma stabilization elements as a part of the transient electromagnetics critical issue C of ETR/INTOR activity. The main responses are radiation dose in the insulator material, induced resistivity and atomic displacement in the conductor material, neutron fluence, nuclear heating and life analysis. In this study, both copper and aluminum conductors with either $MgAl_2O_4$ or MgO insulating material were investigated.

The nuclear responses in the stabilization elements corresponding to the cases of Cu-$MgAl_2O_4$, Cu-MgO, Al-$MgAl_2O_4$, and Al-MgO lead to the following conclusions:

1. There are essentially no changes in the calculated nuclear responses when MgO is substituted for $MgAl_2O_4$ as an insulating material for the same conductor material.

2. The maximum insulator dose for the aluminum case is 1 to 10% smaller than for the copper case for both insulator.

3. The maximum neutron fluence above 0.1 MeV in the aluminum conductor is 6 to 7% less than in the copper conductor for both insulators.

4. The nuclear heating for the aluminum is 50-60% smaller than for the copper case.

5. The maximum atomic displacement in the aluminum material is 10 to 30% higher than in the copper material.

6. The copper resistivity at the end of life will increase significantly (~47% behind the first wall) due to atomic displacement damage and nuclear transmutations. The magnitude of the increase is a strong function of element position relative to the first wall.

C. Nuclear Responses and Shielding Analysis for RF Launchers

A study was carried out to determine the nuclear responses in two RF launchers and the shielding requirements to satisfy the design criterion of 2.5 mrem/h one day after shutdown. The two launchers for this study are: INTOR baseline design of a 4 x 4 array of loops each protected by a BeO radome which is cooled by a Faraday shield braced to the inside, and a stack of ridge-loaded waveguides. Neutron fluences, nuclear heating, gas production, and atomic displacements were calculated to determine the operating conditions and the expected life for the launchers. Neutron fluxes at the back of the launchers were calculated to estimate the bulk shield requirements.

Three-dimensional neutronics calculations have been performed for both launchers using the Monte-Carlo method. The MCNP code was used for the loop launcher design[10] with a continuous energy representation for the nuclear cross sections. The MORSE code was employed for the ridge waveguides with a 67 multigroup coupled cross section set. The results from this study show that the number of BeO replacements required during the INTOR 6.5 $MW\cdot y/m^2$ neutron fluence are 1 and 500 for the ridge waveguide and the loop launcher,[10] respectively. Also, the ridge waveguide needs a 39-cm additional shielding thickness to account for the gaps in the guide module while the loop launchers require only a 21 cm.

V. REACTOR SYSTEM CODES

A. Tokamak

Neutronics models for the tokamak reactor system code at FEDC were developed, implemented, and used for the TFCX design studies. The geometrical model used to generate the model is based on the FED design configuration.[11] Two shielding options, steel and tungsten, are included in the model for the inboard section of the reactor. Carbon or steel armor material for the inboard first wall can be used with both options. The steel type shield is used for the outboard section. The model can be used to calculate the shield dimensions to satisfy specific design criteria or calculate the nuclear responses for a specific reactor configuration. The nuclear responses include nuclear heating in the TF coils, fast neutron fluence, dose in the insulator material, atomic displacements in the copper stabilizer, and dose equivalent in the reactor hall one day after shutdown. The shield compositions used to generate the model are based on the homogeneous optimization study performed for INTOR.[4,5] A parametric study was performed to generate the nuclear responses as a function of the shield thickness. The results from this parametric study were used to generate numerical correlations as a function of the shield thickness, operating time, and neutron wall loading for the models.

B. Tandem Mirror

A blanket and shield computational tool for the tandem mirror reactors trade-off studies was developed. It consists of three modules for the central cell, the choke coil, and the end cell. The three modules were integrated in the Tandem Mirror Reactor System Code (TMRSC).[12]

The central cell module (first wall, blanket, and shield) calculates dimensions, composition, energy deposition in each region, tritium breeding ratio for the breeding blanket option, radiation responses in the central cell coil, and the cost of each component. The total material thickness inside the central cell coil is based on input design criteria or default values for the allowable nuclear responses in the coils. This module has several blanket options (Li or 17Li83Pb breeder; steel or carbon reflector; self-cooled for a liquid metal blanket; He or water coolant for nonbreeding blanket) and two shield designs (minimum thickness concept and low cost steel) which can be specified. The module can be used to minimize or maximize one or more of seventeen blanket parameters. Also, it can be used to perform the analysis for a specific design.

The choke coil module calculates the radiation responses in the choke coil (copper, superconductor, or hybrid design) and estimates the shield requirements for the superconductor coil based on the allowable responses and the shield cost.

The end cell module calculates the required shield thickness inside the Cee coils to satisfy the design criteria, the radiation responses in the Cee coils, and the biological shield thickness outside the Cee coils.

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