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**DESIGN ANALYSIS AND OPTIMIZATION OF SELF-COOLED
LITHIUM BLANKETS AND SHIELDS***

by

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DESIGN ANALYSIS AND OPTIMIZATION OF SELF-COOLED LITHIUM BLANKETS AND SHIELDS*

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ABSTRACT

A study of self-cooled lithium blankets was carried out to define the performance of these blankets and to determine the potential to operate at the maximum possible values of the performance parameters. The main design parameters considered during the course of the study were the tritium breeding ratio, the blanket energy multiplication factor, the energy fraction lost to the shield, the total blanket thickness, the reflector material selection, and the compositions of the different blanket zones.

Another study was carried out to determine materials, compositions, arrangements, and thickness of the shield zone for the reference blanket. Helium and water-cooled shields were optimized for the inboard and outboard sections of the reactor. Based on the above two studies, the reference blanket and shield configurations were developed for the ANL Tokamak Power Systems Study. The helium-cooled shield was selected for use with liquid metal blankets to reduce safety concerns related to lithium-water reactivity. This helium-cooled shield provides shielding characteristics similar to a conventional water-cooled shield. The analyses and results from these studies are the subject of this paper.

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1. INTRODUCTION

The ANL Tokamak Power System Study [1] focussed on the self-cooled lithium blankets with vanadium structural material because of the favorable characteristics of this concept [2]. In the design process of these blankets, a special consideration was given to maximize the performance parameters, reduce the radioactive waste, and reduce safety concerns about lithium interaction with shielding materials. The coolant channel walls are electrically insulated to eliminate MHD-related pressure constraints. This leads to a simple blanket configuration with a single coolant pass in the poloidal direction. Three performance parameters were used to compare the different blanket configurations; the tritium breeding ratio (TBR), the blanket energy multiplication factor, and the energy fraction lost to the shield. Parametric studies were systematically done to study the changes in these parameters due to the following variables: a) the breeder zone thickness, b) the reflector material selection (W or CaO), c) the reflector zone thickness, and d) the reflector zone composition.

A shielding study was carried out to determine shield materials, compositions, arrangements and thickness of the shield zones for the reference blanket. Helium and water-cooled shields were optimized for the inboard and outboard sections of the reactor. A special attention was placed on helium-cooled shields to reduce concerns related to lithium-water reactivity. Two shielding criteria are adopted for this study: a) toroidal field coils are required to function for a $150 \text{ MW}\cdot\text{yr}/\text{m}^2$ deuterium-tritium neutron exposure at the first wall without any change in their performance, and b) workers are

permitted access to the reactor hall one day after shutdown with all shields in place. Based on the above studies, the reference blanket and shield configurations were developed.

1.1 Blanket Analysis

The blanket is designed to convert the kinetic energy of the deuterium-tritium neutrons to recoverable heat and produce adequate tritium to supply the tritium fuel requirement as well as generate enough surplus tritium to start another reactor within a reasonable period of time. In the reactor design process, it is desirable to maximize the recoverable heat produced in the blanket which is defined as the energy generated in the first wall, breeder, reflector, and plenum zones per fusion neutron. Another important function of the blanket is to perform as a part of the reactor bulk shield.

The first wall and the tritium breeding zone composition are dictated by thermal-hydraulics, stress, MHD, and materials considerations. Previous studies [3,4] showed that good neutron moderators combined with high Z materials in the reflector zone improve the blanket performance. The selection of tungsten or calcium oxide in the reflector zone was made for different reasons. Tungsten reduces the blanket and shield thickness and increases the blanket energy multiplication factor. The combination of the tungsten reflector, the liquid lithium, and the vanadium structure produces a low activation blanket. The main concern with this blanket is the cost of the tungsten material. For this reason, calcium oxide was considered because of significant cost reduction compared to tungsten. However, the tungsten reflector is superior to calcium oxide with respect to the blanket performance as will be shown in this section.

In order to study the performance of a self-cooled liquid lithium blanket

with a tungsten reflector, a wide range of design parameters was considered. The breeding zone thickness was varied from 20 to 50 cm with a 7.5 Vol.% V-15Cr-5Ti structural material and 92.5 Vol.% Li. The reflector zone consists of 10 Vol.% V-15Cr-5Ti structural material with the balance used for the liquid lithium and either tungsten or calcium oxide. The lithium fraction in the reflector zone was varied from 0.1 to 0.4. The tungsten and calcium oxide materials are used in a powder form with a 0.9 density factor to reduce the materials unit cost. The reflector zone thickness was varied from 20 to 80 cm. Different combinations of the blanket parameters were used in the transport calculations.

1.2 Blankets with Tungsten Reflector

The change in the tritium breeding ratio as a function of the blanket parameters was analyzed. Figure 1 shows a strong correlation between the tritium breeding ratio and the breeding zone thickness. For a 30 cm reflector zone thickness and a 10 Vol.% lithium in the reflector, the tritium breeding ratio changes from 0.85 to 1.28 as the blanket thickness increases from 20 to 50 cm. Increasing the lithium fraction in the reflector zone from 10 to 40 Vol.% results in ~ 0.1 increase in the tritium breeding ratio. This increase decreases to 0.04 as the breeding zone thickness increases to 50 cm. This clearly shows a weak correlation between the tritium breeding ratio and the lithium fraction in the reflector zone for these types of blankets. Also, the results show a very small increase in the tritium breeding ratio as the reflector zone thickness increases.

Figure 2 shows the energy generated in the blanket as a function of the breeding zone thickness with different lithium fractions. The energy generated in the blanket increases as the lithium fraction in the reflector

zone decreases. Modest increases are also observed as the breeding zone thickness increases for lithium fractions ≥ 0.2 in the reflector zone. However, for a lithium fraction of about 0.1 in the reflector zone, the energy generated in the blanket changes very little as the breeding zone thickness increases. These results suggest a 0.1 lithium fraction in the reflector zone for the reference design to achieve the maximum possible energy generated per fusion neutron. Also, the energy generated in the blanket increases as the reflector zone thickness increases for any breeding zone thickness. Above a 30 cm reflector zone thickness, the energy generated in the blanket increases as the breeding zone thickness decreases. These results indicate that it is desirable to increase the reflector zone thickness and decrease the lithium fraction in the reflector zone.

The energy generated in the shield is lost to the environment because it is not suitable for power generation. The plant efficiency drops by about 1% for every 3% of the total energy generated in the shield. The results show that the minimum breeding zone thickness is 35 cm for a 30 cm reflector zone thickness with a 0.1 lithium fraction to achieve the 3% value. The energy generated in the shield significantly increases as the lithium fraction in the reflector increases. Also, an increase in the blanket thickness (breeding or reflector zone) reduces the energy generated in the shield.

1.3 Blankets with Calcium Oxide Reflector

The analysis performed for the blanket with tungsten reflector was repeated for the calcium oxide. The tritium breeding results show the strong dependence on the reflector zone thickness. Also, the tritium breeding ratios of these blankets are higher than the corresponding values from the similar blankets with the tungsten reflector. These results show that any of the

blanket parameters (the breeding zone thickness, the lithium fraction in the reflector zone or the reflector zone thickness) can be increased to improve the tritium breeding ratio.

In contrast to the blankets with a tungsten reflector, an increase in the breeding zone thickness with any reflector zone thickness improves the energy generated in the blanket. Also, the increase in the thickness of the reflector increases the energy generated in the blanket. However, the increase in the lithium fraction of the reflector zone results in a small increase in the energy generated in the blanket. In general, the use of calcium oxide in the reflector zone improves the tritium breeding ratio by about 0.1 and reduces the energy generated in the blanket by about 13% relative to the same blankets with tungsten reflector.

The results for the calcium oxide blankets show that a reflector zone thickness greater than 65 cm is required to reduce the energy fraction generated in the shield to 0.03. Such thick blankets will cause design difficulties if it is considered for the inboard section of the reactor.

1.4 Reference Blanket

Based on the analysis given in the previous sections, the reference blanket parameters for the Tokamak Power System Study [1] are defined in Table 2. The blanket with the tungsten reflector produces a 15% more energy and requires a 20-cm less thickness relative to the corresponding blanket with the calcium oxide reflector. The blanket with the tungsten reflector was the choice for the inboard section of the reactor because of the desire to reduce the inboard blanket/shield thickness and to get the benefits from the higher blanket energy multiplication.

In the outboard section of the reactor, the main criterion is to reduce

the capital cost of the blanket by using low cost materials. The calcium oxide reflector is used for this function. The breeding zone thickness of a 30 cm is based on the calculated performance of this blanket concept. The energy loss to the shield dictates the thickness of the reflector zone. In this design, a 55 cm thick reflector is used to reduce the energy loss to about 5% of the total energy generated per fusion neutron. Further increases in the reflector zone thickness has a small effect on improving the blanket performance. The main performance parameters for the reference blankets are given in Table 2. The natural lithium is the choice since the use of enriched lithium does not improve the performance of this blanket concept [4]. Also, the use of natural lithium reduces the breeder material cost.

2. SHIELD ANALYSIS

A study was performed using the reference blanket to determine the shielding requirements and the nuclear responses in the superconductor coils. The shield zone was optimized to protect the superconductor coils from radiation damage during the reactor operation and the workers in the reactor hall from the decay gamma rays after shutdown. The study employed the same data libraries and the same radiation transport code used for the blanket analysis. The design criteria, the analysis procedure, and the results are discussed in this section.

2.1 Shield Design Criteria

The effects of neutron and photon irradiations on superconductor coils tend to lower their performance. For NbTi superconductor material, results from neutron irradiation experiments [5,6] have shown that the critical current density is decreased by about 10% at a fast neutron fluence of about

4×10^{18} n/cm² ($E > 0.1$ MeV) while the critical temperature is unchanged. The degradation of the critical current is expected to reach a saturation value at high neutron fluences. At ambient temperature, neutron irradiation experiments show a saturation value of 20%. The extrapolation of the existing data at 4°K to 10^{19} n/cm², results in a 17% reduction in the critical current. Such fluence degradation does not represent a problem for the current 7T magnet design since the NbTi superconductor must operate below the initial current density due to coil protection considerations.

Experimental studies show that the copper resistivity increases rapidly with the fast neutron fluence and reaches a saturation value of $\sim 4 \times 10^{-7}$ Ω -cm. At this level of neutron fluence, the total copper resistivity in the 7T magnet is ~ 1.6 times the saturation level. This 7T magnet uses a large copper fraction, $\sim 50\%$ of conductor cross section, and forced liquid helium flow for heat removal. Such a design permits the copper stabilizer to operate at the saturation level of the irradiation induced resistivity.

The insulator material is the most sensitive component in the superconductor coils because the radiation damage is irreversible and limits the operating life of the coils. The properties of interest for the coil designs are the electrical resistivity, dielectric strength, the mechanical strength, and the thermal insulation. Experimental results [7] from neutron irradiation at 5K suggest that polyimides can withstand a radiation dose of 10^{10} rad and retain high resistivity and mechanical strength. The maximum tolerable dose in the insulator material at the end of life was assumed to be 5×10^{10} rad for the current analysis. The low value of the magnetic field and improvements in insulator materials are expected to allow for this dose limit.

The nuclear heating in the coils impacts the required refrigeration power since about 500W of electrical power is consumed to remove 1W at 4K. This low

removal efficiency calls for minimizing the nuclear energy deposited in the superconductor coils. The maximum nuclear heating was limited to $<1.0 \text{ mW/cm}^3$ to maintain reasonable refrigeration capacity and to use a simple design for the heat removal system.

For personnel exposure, the U.S. Federal Regulations limit the occupational exposure to 5 rem/y with a maximum of 3 rem/quarter. Based on working 8 h/day and 40 h/week, the occupational exposure is 2.5 mrem/h.

2.2 Parametric Analysis

First, one-dimensional nucleonics calculations were performed as a function of the shield thickness. The analyses for the inboard and the outboard section were carried out separately to account for the differences in the blanket materials and the design criteria. The total thickness of the inboard blanket and shield is an important parameter for the reactor design. In the current study, the shield section was designed to protect the inner leg of the toroidal field coils. The outboard shield was designed to permit personnel access to the reactor hall with all shields in place one day after shutdown.

The materials for the shielding system were selected based on trade-off studies and the results of the previous design studies [8-12]. In the selection process, safety, cost, and performance were considered. The shielding materials selected for this study are Fe-14Mn-2Ni-2Cr steel alloy (Fe1422), titanium hydride (TiH_2), boron carbide (B_4C), water, and lead. Two coolants, water and helium, were considered for the shield. Special attention was given to the helium-cooled shield to improve its performance because of the safety advantage with liquid metal systems. Fe1422 is the structural material for the shield. TiH_2 is used as a substitute for water in the

helium-cooled shield. A small amount of B_4C is used at the back of the shield as a neutron absorber to reduce the activation in the reactor components and the structural materials. A 0.7 density factor was used for the B_4C 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 outboard shield so that the dose equivalent in the reactor hall is reduced after shutdown.

For the water shield (x cm of 80% Fe1422, 20% H_2O ; 5 cm of 60% B_4C -0.7 DF, 20% Fe1422, 20% H_2O by volume) with the reference inboard blanket, the maximum insulator dose results indicate that a 35 cm thick shield (101 cm thick blanket and shield) is required to achieve the 5×10^{10} rad. The 10^{19} n/cm² fast neutron fluence in the superconductor material require that a 37 cm thick shield. At this shield thickness, the maximum nuclear heating rate is less than the design limit. On the other hand, the helium-cooled shield (x cm of 90% Fe1422, 10% He; 5 cm of 60% B_4C -0.7 DF, 30% Fe1422, 10% He by volume) requires 63 cm of shielding material (129 cm thick blanket and shield) to achieve the same neutron fluence in the superconductor coils. This represents an extra 28 cm of shield material in the inboard section.

The desire to use the helium-cooled shield motivated an optimization study to reduce its thickness. The approach was to introduce a hydrogenous material to improve the slowing down characteristics of the shield. Titanium hydride (TiH_2) was selected and used in the steel zone of the shield. The volume fraction of TiH_2 was varied from 0.0 to 0.5 with a helium fraction of 0.1 and the balance as Fe1422 in the steel zone. A similar analysis was carried out for the water-cooled shield to define the water volume fraction in the steel zone. Figures 3 and 4 give the maximum thermal insulator dose and the maximum nuclear heating in the toroidal field coil as a function of the hydrogenous material in the steel zone (TiH_2 or H_2O). A 37 cm total shield

thickness is used which is based on the water shield results for the water shield.

The results of the water-cooled shield calculations confirm that the 0.2 water/0.8 Fe¹⁴²² composition is an optimum choice to achieve the minimum insulator dose and nuclear heating in the superconductor coil. For the helium-cooled shield, the 0.3 TiH₂/0.6 Fe¹⁴²² composition is the optimum choice. The calculated nuclear responses for the insulator dose and the nuclear heating with the optimum helium-cooled shield are 13 and 8% lower than the corresponding values from the optimum water-cooled shield.

The outboard shield thickness was designed to provide 2.5 mrem/h dose equivalent in the reactor hall one day after shutdown. Previous design analyses [11,12] show that a 2×10^7 n/cm²·s total neutron flux at the outer shield surface results in the required dose equivalent. For the reference blanket with the calcium oxide reflector, a 90 cm of water-cooled shield is required. For the TiH₂ shield, it is expected to require about the same thickness based on the total neutron fluence results for both shields.

2.3 Reference Shield

Safety considerations associated with the liquid lithium system and the optimized performance of the helium-cooled shield indicate a preference for using this shield over the traditional water-cooled shield. Table 3 gives the reference shield parameters. The inboard shield has a total thickness of 37 cm with 10% of the volume used by the helium coolant. The maximum dose in the electrical insulator is 9.6×10^9 rad which is quite satisfactory based on the current experimental data for polyimide insulators. The thermal insulator dose is 2.1×10^{10} rad. The fast neutron fluence is 8×10^{18} n/cm² in the NbTi superconductor. The nuclear heating is only 0.24 mW/cm³. These nuclear

responses do not degrade the performance of the toroidal field coils. The saturation value of the copper induced resistivity is used in the toroidal field coils design.

The outboard shield requires 90 cm of shield material to achieve the 2.5 mrem/h in the reactor hall one day after shutdown. The composition is the same as the inboard shield except for a 5 cm layer of lead at the outer surface as shown in Table 3.

3. SUMMARY

A first wall/blanket/shield system using the self-cooled lithium blanket concept with vanadium structural material has been developed and optimized for the Tokamak Power System Study. The entire system is designed with reduced activation materials. The tungsten and calcium oxide for the reflector zone were selected based on their performance. The helium-cooled shield developed for this system eliminate concerns related to lithium interaction with shielding materials.

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TABLE 1 -- Reference Blanket Parameters

Blanket Parameter	Reflector Material	
	Tungsten	Calcium Oxide
First wall ^a thickness, cm	1.0	1.0
Breeder ^b zone thickness, cm	35.0	30.0
Reflector ^c zone thickness, cm	30.0	55.0
Tritium production in the first wall per DTn	0.027	0.026
Tritium production in the breeding zone per DTn	1.059	1.009
Tritium production in the reflector zone per DTn	0.027	0.214
Tritium breeding ratio	1.113	1.249
Energy generated in the first wall MeV/DTn	0.68	0.66
Energy generated in the breeding zone MeV/DTn	11.93	11.13
Energy generated in the reflector zone MeV/DTn	5.19	4.64
Energy generated in the shield zone MeV/DTn	0.59	0.81
Energy generated in the blanket MeV/DTn	17.80	16.43
Total energy generated in the blanket/shield MeV/DTn	18.39	17.24
Blanket energy multiplication factor	1.27	1.17
Energy fraction lost in the shield	0.03	0.05

^a First wall composition is 50 Vol.% V15Cr5Ti alloy, 50 Vol.% natural lithium.

^b Breeder zone composition is 7.5 Vol.% V15Cr5Ti alloy, 92.5 Vol.% natural lithium.

^c Reflector zone composition is 10 Vol.% V15Cr5Ti alloy, 10.0 Vol.% natural lithium, 80.0 Vol.% reflector material with 0.9 density factor.

TABLE 2 -- Reference Shield Parameters^a

A. Inboard Shield

Total shield thickness, cm	37
Steel zone ^b thickness, cm	32
Boron zone ^c thickness, cm	5
Thermal insulator dose, rad	2.1×10^{10}
Electrical insulator dose, rad	9.4×10^9
Fast neutron fluence in the winding material, n/cm^2	8×10^{18}
Total neutron fluence in the winding material, n/cm^2	1.3×10^{19}
Nuclear heating in the coil case, mW/cm^3	1.2
Nuclear heating in the winding material, mW/cm^3	0.24
Nuclear heating in the coil case, W/cm	11.3
Nuclear heating in the winding material, W/cm	6.8
Atomic displacement in the copper stabilites, dpa	2.8×10^{-3}

B. Outboard Shield

Total shield thickness, cm	90
Steel zone ^b thickness, cm	80
Boron zone ^c thickness, cm	5
Lead zone ^d thickness, cm	5

^a All responses normalized to 5 MW/m^2 D-T neutron wall loading or $150 \text{ MW}\cdot\text{yr/m}^2$ D-T neutron fluence at the first wall.

^b Steel zone composition is 60% Fe1422, 30% TiH₂, 10% He by volume.

^c Boron zone composition is 60% B₄C (0.7 DF), 30% Fe1422, 10% He by volume.

^d Lead zone composition is 100% Pb.

FIGURE CAPTIONS

Figure 1. Tritium breeding ratio as a function of the breeding zone thickness for a 30 cm tungsten reflection zone thickness with different lithium fractions in the reflector zone.

Figure 2. Energy generated in the blanket as a function of the breeding zone thickness for a 30 cm tungsten reflector zone thickness with different lithium fractions in the reflector zone.

Figure 3. Maximum thermal insulator dose as a function of the hydrogenous material in the steel zone of the shield normalized to 150 $\text{MW}\cdot\text{yr}/\text{m}^2$ D-T neutron fluence at the first wall.

Figure 4. Maximum nuclear heating in the toroidal field coil as a function of the hydrogenous material in the steel zone of the shield normalized to 5 MW/m^2 D-T neutron wall loading at the first wall.

FIGURE 3. Y. GOHAR

FIGURE 4. Y. Gohar