ITER BLANKET AND SHIELD STUDIES FOR HIGH ASPECT RATIO DESIGN OPTION*

Y. Gohar, H. Attaya, M. Billone, C. Johnson, J. Kopasz, S. Majumdar, R. Mattas, and D.L. Smith
Argonne National Laboratory, 9700 S. Cass Ave., Argonne, IL 60439 USA

D. Lousteau, B. Nelson, and D. Williamson
Oak Ridge National Laboratory, P.O. Box 2009, Oak Ridge, TN 37831 USA

A. Raffray, A. Badawi, A. Ying, and M. Abdou
University of California, LA, 6288 Boelter Hall, Los Angeles, CA 90024-1597 USA

L. El-Guebaly, M. Sawan, I. Sviatoslavsky, E. Mogahed, and G. Kulcinski
University of Wisconsin, 1500 Johnson Drive, Madison, WI 53706-1687 USA

The attractiveness of the high aspect ratio design (HARD) option for ITER has motivated a study to assess the blanket and shield design performance for this configuration relative to the ITER reference design. The blanket and shield have been configured to take an advantage of the HARD option. The layered solid breeder blanket concept with water-coolant and steel-structure and the water-steel shield have been used. The changes in the neutron wall loading distribution, the mechanical design, the net tritium breeding ratio, the total tritium inventory, and the nuclear heating profiles are evaluated. The tradeoff between the net tritium breeding ratio, and the fuel operating cost is analyzed. The mechanical design and the structural interaction between the first wall and the blanket is studied.

Introduction

This paper examines the impact of HARD option on the blanket and shield performance as concluded from the ITER Conceptual Design Activity (CDA). The neutron wall load distributions have been calculated for the different operating modes of HARD. The average neutron fluence has been estimated over the test ports. The net tritium breeding capability of the layered solid breeder blanket with water-coolant and steel-structure has been calculated for the HARD configuration. The required tritium breeding capability is defined as a function of the external tritium supply. Several blanket modifications have been studied to improve the blanket performance. The shield performance has been analyzed and the results compared to that of CDA.

Neutron Wall Load

The neutron wall load distributions were calculated for the three operating modes of the High Aspect Ratio Design (HARD). The intent of the calculation is to compare the HARD capability for nuclear testing with the corresponding results from CDA. Also, the neutron wall loading distributions are used to define the net tritium breeding capability, the nuclear responses in the toroidal field coils, and the shielding performance.

The NEWLIT code [1] was used to perform the calculations. The code uses ray tracing techniques to determine the plasma contribution to the neutron current at a given point at the first wall. The neutron source distribution is considered as a function of the magnetic flux surfaces. The function is represented as $[1 - (a/a_o)^2]^3$. Where a is the minor radius of magnetic flux surface and a_o is the plasma minor radius. The magnetic flux surfaces were represented by the following parametric equations:

$$R = R_c + a_o \cos(t + C_o \sin(t)),$$

$$Z = K_o a_o \sin(t), \text{ and}$$

$$R_c = R_o + m(1 - (a/a_o)^2).$$

The parameter t varies from o to π. The plasma triangularity (C_o), the elongation factor (K_o), the magnetic shift (m), and the other parameters are given in Table 1. Figure 1 shows a schematic of the first wall geometrical model, which follows closely the field lines at the edge of the scrapeoff layer. At the midplane, the thickness of the scrapeoff layers are 15 and 14 cm for the outboard and inboard sections, respectively. The results of the calculations are shown in Fig. 2 for the different operating modes of the HARD where the peak reactor neutron wall loading occurs at the midplane of the outboard section. The peak values are 1.49, 1.73, and 1.95 for the inductive, steady state, and hybrid modes of operation. The variation in the peak neutron wall loading values reflect the difference in the fusion power, magnetic shift, and plasma triangularity. The average neutron wall loading values on the test section were also calculated for the three operational modes of HARD. The average values are higher than the corresponding values of CDA by 30 to 70% depending on the operating mode as shown in Table 1.

Breeding and External Requirements of Tritium

In this study, the design guidelines and the blanket design of CDA were used to study the impact of the HARD configuration on the net tritium breeding ratio. The solid-breeder with water-coolant and steel-structure of CDA was used for the inboard and outboard sections of HARD. Net tritium breeding estimates were made based on one-dimensional toroidal cylindrical geometry calculations. The one-dimensional results were coupled with the neutron coverage fractions of the different blanket regions. The neutron coverage fraction is corresponding to the fraction of source neutrons going directly to the region without any collisions. These coverage fractions were obtained from the neutron wall loading calculations.

The inboard blanket sectors extend vertically from z = -2.69 m to z = 2.69 m. The outboard blanket sectors extend from z = -3.85 m to z = 2.86 m with sixteen midplane ports occupied by other components. The port dimensions are 1.27 m and 2.3 m in the toroidal and poloidal directions, respectively. The neutron fraction lost to the sixteen ports is about 10%. It is about the same value as of CDA although the total port surface area is 24% less than CDA. Also, the neutron wall loading peaking at the ports is higher in HARD. These two factors are balancing each other, which leads to the same neutron coverage for the ports. The neutron coverage of the outboard blanket is reduced by 9% relative to CDA. Also, the divertor zone receives 21% of the neutrons which represents 34% increase from CDA. These two changes have a negative impact on the net tritium breeding ratio. The neutron coverage of the inboard blanket is increased from 16% in CDA to 20% in HARD. The net impact on the tritium breeding ratio is about 8% less tritium breeding as shown in Table 2.

The different blanket design options of CDA were reevaluated for HARD with respect to the net tritium breeding ratio. The blankets geometrical parameters of CDA [2,3] were used to study the impact of the HARD geometrical configuration. The radial build for the solid breeder blanket varies poloidally according to the neutron wall loading variation in order to maintain constant minimum breeder temperature in the poloidal direction. The effect of the 0.5 cm thick copper stabilizer loops in the outboard blanket has been evaluated. The blanket has a single Li_2O layer in the inboard blanket and two or three in the outboard blanket. The lithium in the Li_2O is enriched to 95%. The tritium breeding results have been modified to account for the actual neutron coverage, the assembly gaps, and the side walls. The net tritium breeding ratios are 0.78 and 0.87 for the design with two and three breeding

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Table 1
Comparison of Neutron Source Parameters and Neutron Wall Loading Results for HARD and CDA

Parameter	HARD			CDA	
	Inductive	Steady State	Hybrid	Physics	Technology
Major Radius (m)	6.33	6.33	6.33	6.00	6.00
Minor Radius (m)	1.58	1.58	1.58	2.15	2.15
Elongation	2.00	2.00	2.00	1.982	1.982
Triangularity	0.289	0.285	0.284	0.383	0.383
Magnetic Shift (m)	0.144	0.266	0.284	0.255	0.255
Fusion Power (MW)	850	960	1080	1100	860
First Wall Area (m^2)	833	833	833	944	944
Wall Loading (MW/m^2)					
First Wall Average	0.817	0.923	1.038	0.934	0.73
Maximum Value	1.490	1.728	1.952	1.540	1.204
Max. Inboard Value	1.149	1.245	1.392	1.131	0.844
Test Section Average	1.396	1.616	1.825	1.374	1.074

Table 2
Comparison of Neutron Coverage and Net Tritium Breeding Parameters for HARD and CDA

Parameters	CDA	HARD
Inboard Blanket Extent, m	-3.40 to 3.40	-2.69 to 2.69
Neutron Coverage of Inboard Blanket, %	16.4	19.6
Outboard Blanket Extent, m	-4.80 to 4.80	-3.85 to 2.86
Neutron Coverage of Outboard Blanket, %	69.1	59.6
Divertor Neutron Coverage, %	15.5	20.8
Port Neutron Coverage, %	10.3	9.9
Net Tritium Breeding Ratios of Different Options		
Two breeder zones and copper stabilizer	0.84	0.78
Two breeder zones without copper stabilizer	0.87	0.80
Three breeder zones and copper stabilizer	0.92	0.85
Three breeder zones without copper stabilizer	0.95	0.87

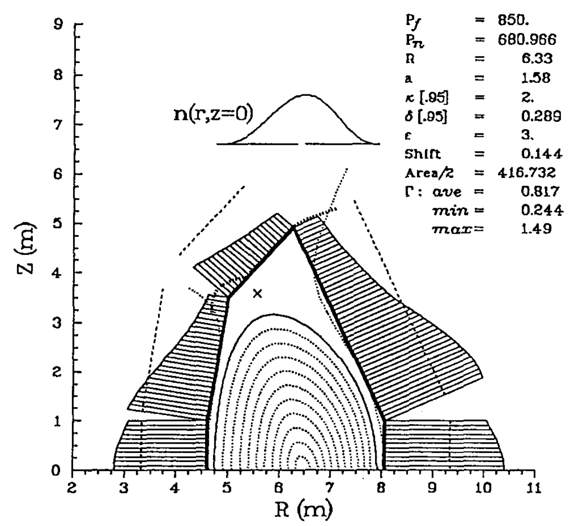

Figure 1
HARD first wall model and neutron wall loading for the ignited mode.

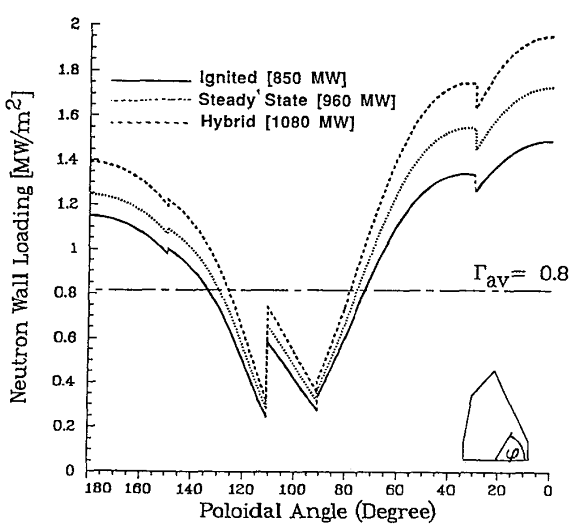

Figure 2
HARD neutral wall loading for the three modes of operation.

layers. The copper stabilizer loops in the outboard blanket reduce the net tritium breeding ratio by 3% as shown in Table 2. In inboard section, the effect of the loops is less than 1% on the net tritium breeding ratio because of low contribution of the inboard blanket.

The tritium requirements were evaluated for the HARD option as a function of the external tritium supply rates. It is assumed that the external tritium inventory available for ITER at start of operation is 20 Kg and the steady state inventory in the ITER components is 5 Kg. The calculations were normalized to the same average neutron fluence of 3 MW•a/m^2 at the midplane test sections. Table 3 gives the minimum tritium breeding ratios calculated for both options of ITER. The HARD option requires less net tritium breeding ratio for the same external tritium supply rate because of the higher average neutron wall loading at the test section compared to the CDA option. This can be translated to less external tritium supply, less operating time or relaxing the required tritium breeding ratio from the blanket to achieve the same fluence in the test sections.

Blanket Design Changes for HARD

Several blanket design changes have been considered to benefit from the HARD configuration and improve the performance. The first change is to divide the blanket sector to six modules in the poloidal direction. These blanket modules are attached to the continuous shield sector where the water manifolds are located. With this change, the replacement procedure for the first wall, blanket, and shield is the same as of CDA where the sector is replaced as a single unit. This design change does simplify the fabrication and assembly procedures of the blanket. Also, it increases the stiffness of the first wall structure to accommodate the plasma disruption loads.

The midplane module of the outboard blanket has been analyzed for thermal and helium pressure (0.1 MPa) loadings. The module extends about 1 m in the toroidal and poloidal directions. The poloidal and toroidal curvatures of the first wall were included in the model. Both the first wall and the blanket coolant panel were assumed to consist of three layers, a layer containing the water channels enclosed between two steel layers. The cross-section of the side walls between the first wall and the coolant panel were the same as that of the first wall. The first wall and the coolant panel were added to form the side wall between the coolant panel and the shield. The steel dimensions were taken as the CDA design. The heat flux on the front plate of the first wall as taken as 0.3 MW/m^2 while the heat flux on the back plate of the first wall from the blanket was treated as a variable. The radial variation of the temperature distributions was adjusted to the match the heat flux values. The results show the existence of an optimum heat load value from the blanket, which minimizes the thermal stresses as shown in Fig. 3. At the optimum heat load value, the first wall displacement is very small. The results show the maximum stress values occur at the corners of the module that can be reduced by adjusting the geometry at these points.

Another study was performed to improve the tritium breeding capability of the blanket and to study the use of high density beryllium. The CDA design guidelines with respect to the temperature limits for each blanket material were used for this study. The obtained local tritium breeding ratio is 9% higher than the corresponding value of CDA.

Shielding Performance

Shielding analysis has been performed for the proposed HARD design using one-dimensional model and the shielding performance for this configuration is compared to that for the CDA design [4]. The one-dimensional discrete ordinates code ONEDANT and a 67-coupled group nuclear data library based on ENDF/B-V were used to carry out the transport calculations. Three different operating modes, namely, the inductive, steady state, and hybrid modes are proposed for HARD. The fusion power and consequently, the neutron wall loadings vary depending on the mode of operation. The shielding analysis has been performed for the hybrid case which represents the worst case from the shielding standpoint since it yields the highest neutron wall loading. The peak inboard wall loading in the hybrid case with 1080 MW fusion power is 1.39 MW/m^2. Magnet nuclear heating results can be determined for the other modes of operation by scaling with the fusion power (850 and 960 MW for the ignited and steady state modes, respectively). On the other hand the end-of-life fluence and insulator dose should be independent of fusion power as long as the ITER fluence goal of 3 MW•y/m^2 is maintained.

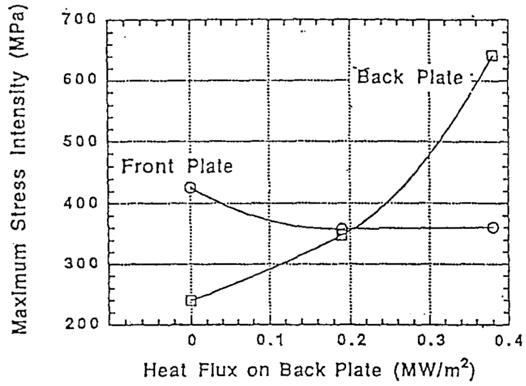

Figure 3
Variations of the maximum stress intensity in the front and back plates of the first wall as a function of the heat flux on the back plate for a 4x4x5 mm first wall design.

The design parameters pertinent to shielding analysis are given in Table 4 for the HARD (hybrid) and CDA (physics phase) designs. The parameters are given for the three regions with critical shielding space. These are the inboard region, the divertor region and the shield recess area located between the end of the inboard blanket and the inner end of the divertor plate. For comparable fusion power, the peak neutron wall loading values in these regions are higher than those in the CDA design because of the higher aspect ratio on the HARD design. The assembly gap between the back of the shield and the vacuum vessel (V.V.) is 4 cm in HARD compared to 2 cm in CDA. Furthermore, the inboard (i/b) blanket/shield/gap/V.V. thickness at the midplane is 83 cm vs. 84 cm in CDA. This means that 3 cm of the shield is replaced by void. 3 cm less shield translates into ~50% higher damage at the magnet. Hence, the peak end-of-life insulator dose at the inner legs of the TF coils will exceed the design limit for epoxy. The insulator dose in the divertor region of HARD is acceptable as as result of using a thicker coil case.

The nuclear heating in the TF magnets for the three critical regions (i/b, recess, and divertor) was calculated taking into account the poloidal variation of neutron wall loading and blanket/shield/V.V. thickness. The results are summarized in Table 5 for the HARD and CDA designs. Radiation streaming through the different penetrations is assumed to contribute 4 kW to the total magnet nuclear heating. Most of the heating in the inboard region is generated in the 3 m high middle section. Heating in this region is doubled due to the larger magnet volume (70% more), higher wall loading, and thinner shield, compared to the CDA design. The results indicate that the total heating loads are 68, 53, and 60 kW in the hybrid, ignited, and steady state modes of operation, respectively.

In order to reduce the heating to a reasonable level and meet the insulator does limit for 3 MW•y/m^2 fluence, several modifications to the present HARD design need to be adopted. The total i/b blanket/shield/gap/V.V. thickness at midplane should be restored to 84 cm with the gap reduced to 2 cm as in the CDA design. In addition, it is necessary to extend the upper parts of the side modules of the outboard blanket inward up to the plasma

Table 3. Minimum Tritium Breeding Ratio Required for Different Tritium Supply Rates Normalized to 3 MWa/m^2 Fluence at the Test Section

Tritium Supply Rate Kg/y	Total Tritium Purchased, Kg	Cost of Purchased Tritium B$	Required Tritium Breeding Ratio HARD	CDA
1	30	0.87	0.82	0.86
2	40	1.16	0.72	0.79
3	50	1.45	0.62	0.71
4	60	1.74	0.52	0.64

Table 4. Relevant Shielding Design Parameters

	CDA (Physics Phase)	HARD (Hybrid)
Fusion Power (MW)	1100	1080
Average Wall Loading (MW/m^2)	0.93	1.04
Inboard Region		
Peak Wall Loading (MW/m^2)	1.13	1.39
Blanket/Shield/Gap/V.V. thickness* (cm)	84	83
Gap Thickness (cm)	2	4
Coil Case and Winding Pack Cross Section Area* (m^2)	6.3	10.8
Recess Region		
Peak Wall loading (MW/m^2)	0.22	0.25
Blanket/Shield/Gap/V.V. Thickness (cm)	70	70.6
Divertor Region		
Peak Wall Loading† (MW/mw)	0.67	0.75
Blanket/Shield/Gap/V.V. Thickness† (cm)	58	56
Coil Case Thickness (cm)	26	39

* At midplane. † At i/b side of outer end of divertor plate.

Table 5. Total Nuclear Heating in TF Coils

	CDA Physics Phase (1100 MW$_f$)	HARD Hybrid (1080 MW$_f$)	Modified HARD* Hybrid (1080 MW$_f$)
Inboard	11	21	14
Recess	4	8	6
Divertor	31	35	23
Penetrations	4	4	4
TOTAL (kW)	50	68	47

* With 2 cm gap, 84 cm i/b blanket/shield/gap/V.V. and modified o/b blanket side modules.

boundary (similar to the lower parts) in order to provide extra shielding for the upper divertor region. Notice that the side modules are located underneath magnets and there will be no interference with maintenance. The impact of this modification is to reduce the divertor heating by ~5 kW. If these changes take place, the heating in the magnets will be 47 kW for the hybrid case, as detailed in the last column of Table 2.

Conclusions

This paper examines the impact of HARD option on the blanket and shield performance as concluded from CDA. The neutron wall load distributions have been calculated for the different operating modes of HARD. The average neutron wall load over the test ports is 30 to 70% higher than CDA which increases the testing capability of ITER. The net tritium breeding capability of HARD has been calculated based on the use of CDA blanket design. The net tritium breeding ratio is 8% less than CDA. However the increased average neutron wall load over the test sections overrides the impact on the external tritium supply because of shorter operating time for the same neutron fluence. The shielding analysis of HARD has suggested several modifications to meet the design limits of CDA. The total inboard thickness at midplane should be restored to 84 cm with the vacuum gap between the shield and the vacuum vessel reduced to 2 cm as in CDA, or 86 cm inboard thickness with the current gap thickness of 4 cm. In addition, it is necessary to extend the upper parts of the side modules to provide extra shielding for the upper divertor region.

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Magnet Reliability Workshop

[F. Puhn (GA)
and
D.B. Montgomery (MIT)]

The Magnet Reliability Workshop was a series of oral presentations by experienced magnet engineers and technical managers. The session was held in a workshop format to address the problems of attaining reliability in magnet systems. The experience of both fusion and SSC programs were well represented. The information presented ranged from detailed problem-solution descriptions to broad general advice on how to avoid problems in the future.

The recurrent general themes offered by the workshop participants could be summarized as follows:

- The things we worry about the most during design tend to fail the least.
- Little things tend to fail, particularly those things that get minimal attention during design.
- We need more engineering attention on details in order to improve reliability (and this will cost more money than we are used to spending).
- There are specific failure modes and effects that seem to repeat themselves, and these need more attention.

In the same vein, the recurrent specific failure modes offered in one form or another by the participants could be summarized as follows:

- Faults in electrical isulation caused by contamination or other unknown defects.
- Friction in clamped interfaces being either too high, too low, or intermittently disappearing, causing excessive forces, ratcheting deflection, or slippage.
- Structural stiffness being lower than anticipated in complex assemblies (perhaps by a factor of 2 or more).
- Leaks (coolant, vacuum), which propagate into other more severe failures.
- Loads not being completely defined, particularly secondary loads due to fringe fields, friction, excessive deflection, etc.
- Failures that occur or propagate due to inadequate information on the operational status of the magnet.

A general consensus of the attendees was that a design handbook for magnet engineers is needed. Advice, historical facts, and rules-of-thumb such as those presented in these talks, could all be included in such a handbook. The focus on improving magnet reliabilty stems from a need of large, future fusion projects such as ITER. A combination of cost, safety issues, and availability requirements all drive the need for more reliable magnets. The magnet reliability sessions were well attended, with considerable discussion from the audience. The subject is becoming of primary importance to major programs, and must receive more attention in the future.

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