Thermal Analysis of the Blanket and Shield for the U.S. ITER Design

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October 1990

UWFDM-832

Presented at the 9th Topical Meeting on the Technology of Fusion Energy, 7–11 October 1990, Oak Brook IL (Fus. Tech. 19 (3) Pt. 2A 1546–1551).

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THERMAL ANALYSIS OF THE BLANKET AND SHIELD FOR THE U.S. ITER DESIGN

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ABSTRACT

Thermal analyses for the U.S. inboard and outboard shield and first wall of ITER have been performed utilizing the nuclear heating results obtained from the neutronics calculations. Several radial build configurations of the shield have been thermally analyzed. Different routes for the coolant are investigated to reduce the maximum temperature in shielding material which in turn reduces thermal expansion effects during ITER operation. The maximum thermal stresses are within the prescribed limits for the shield material at the maximum operating temperatures.

INTRODUCTION

In order for the blanket and shield of any thermonuclear reactor to perform optimally and safely, a reliable coolant system must be utilized. The parameters of the coolant system must be optimized simultaneously with neutronics and thermal stresses calculations. Iterations are needed to reach the best possible practical coolant/solid configuration. This approach is used here by utilizing the neutronics results to reach the optimal values for ITER coolant parameters.

HYDRAULICS ANALYSIS

The hydraulics analysis for the blanket is given. In addition, thermal-hydraulics analysis for the water-cooled shield and first wall/side walls are given in this work for inboard and outboard sides. The detailed geometry of the coolant systems is recommended based on thermal and mechanical considerations. Several analyses were performed to optimize temperature distribution, thermal stresses and pressure drop in the shield, first wall and side walls. In the shield thermal hydraulics analysis, the temperature distribution, overall coolant routing, thermal stresses and pressure drops have been calculated for both inboard and outboard sides. Several coolant channel thicknesses in the shield have been optimized to minimize the pressure drop in the coolant loop without affecting shield performance. Different routes for the coolant were investigated to reduce the maximum temperature in the shielding material. In all cases, the maximum thermal stresses are within the prescribed limits for the material at the maximum operating temperatures. A multilayer configuration is employed for the design of the blanket and shield. The optimum configuration of the shield has been determined from one-dimensional neutronics calculations. The blanket/shield radial build varies poloidally depending on the neutron wall loading.[1] The temperature distribution in the blanket is strongly connected to the optimum blanket design and is calculated during the optimization process of the blanket.[2] Several radial build configurations of the shield have been thermally analyzed. Thermal stresses in the shielding material determine the thickness of the different shielding material layers. The way the coolant is routed in the blanket and shield determines, to a large extent, the pressure drop results. Toroidal routes result in very low pressure losses while the reverse is true for poloidal routes. The reason for that is due to the fact that in the toroidal coolant routing the flow cross sectional area is larger which gives a lower flow velocity. Also, toroidal routing provides a shorter coolant path.

The coolant has been legislated to run poloidally in the inboard blanket/shield and outboard shield, while it runs toroidally in the outboard first wall and blanket. The coolant is supplied and collected at the top of the reactor except for the central lower outboard modules where the supply and the return have to be from the bottom. In this section, the thermal analysis is done for the technology phase. Scale-up may be used for physics phase calculations according to new heating loads. The following are the assumptions used in the hydraulics analysis:

- 1-D thermal analysis.
- The coolant is supplied to the first wall, blanket or shield at 60°C.
- The coolant returns upward and is collected at 100°C.
- The vertical extent of the blanket/shield is 8 meters for nuclear heating calculations, while it is 10 meters for pressure drop calculations.

- The coolant velocity in the coolant feed pipes to the FW, blanket or shield is 4.5 m/s.

INBOARD THERMAL HYDRAULICS ANALYSIS

First Wall and Side Walls

The inboard blanket consists of 32 modules. Each module has three sectors separated by an electrical insulator layer. Each sector has a first wall and two side walls. The first wall and the side walls are designed in the form of steel/water/steel layers. Figure 1 is a schematic which shows the general configuration of one sector of the inboard blanket/shield. Coolant channels are formed in the steel walls. For each sector the first wall has 6 coolant channels while each side wall has 5 coolant channels. Half of the coolant passing in the first wall, say through the left 3 coolant channels, returns back in the left side wall through the 5 channels[3] (see Figures 1 and 2). Note that we can consider the first wall/side wall coolant routes either as one single coolant system or two separate coolant systems; if one fails the other will do the job. The change in width of the first wall and side walls at the top and the bottom of the blanket has been considered in the pressure drop calculations. The results of the thermal-hydraulics analysis for a 0.5 cm FW channel thickness are given in Table 1.

Table 1. Inboard thermal-hydraulics parameters for FW/side wall.

Nuclear heating carried by the FW channel	41.7 kW/cm
Nuclear heating carried by the side wall channel	10.9 kW/cm
Temperature rise in FW coolant	31.7°C
Temperature rise in side wall coolant	8.3°C
Max. velocity in the FW channel (midplane)	7.08 m/s
Max. velocity in the side wall channel (midplane)	4.02 m/s
Average velocity in the FW channel	6.70 m/s
Average velocity in the side wall channel	3.02 m/s
# of coolant channels in one FW sector (midplane)	6
# of coolant channels in one side wall sector (midplane)	5
Pressure drop along the FW channel	4.20 atm
Pressure drop along side wall coolant channel	1.35 atm
Total pressure drop in FW/side wall coolant system	5.55 atm
Inlet Pressure	7.5 atm
Total volumetric flow rate to inboard side (FW)	687.84 liter/s
Coolant feed pipe diameter to one inboard module (FW)	7.8 cm

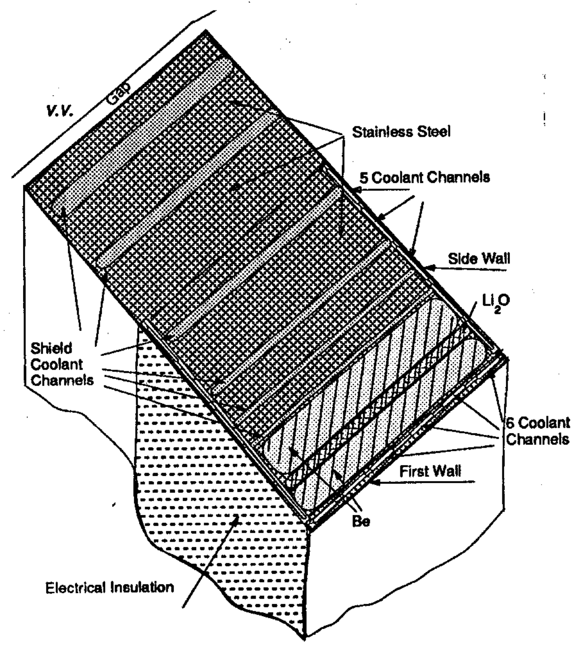

Figure 1. A schematic layout for 1/96 (one sector) of the inboard blanket and shield.

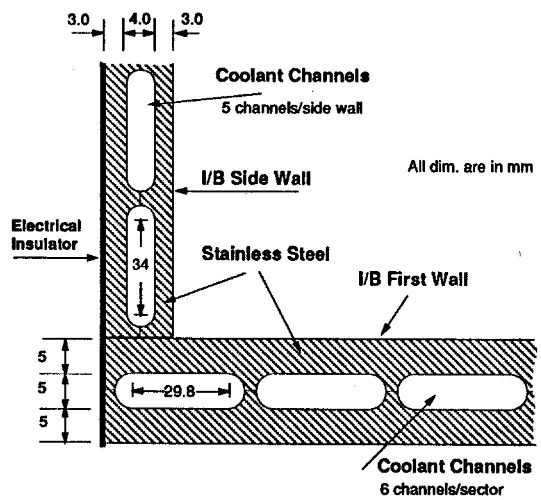

Figure 2. A schematic of the first wall and side wall coolant channels.

Inboard Shield

The shield consists of alternating layers of 316 stainless steel and water. Nuclear heating results obtained from one-dimensional neutronics calculations have been utilized to perform the thermal analyses for the inboard shield. Figure 3 is a nuclear heating distribution at the midplane and the top of the inboard blanket/shield. Figure 4 is a schematic of the optimized radial build of the inboard blanket/shield. It gives the coolant routes, coolant velocity, temperature rise in the coolant from top to bottom, pressure drop, steel maximum temperature and steel maximum thermal stresses at the reactor midplane. Figure 5 is a simple plane schematic of the radial build of the inboard blanket/shield at the top of the reactor. It gives the temperature of the coolant at the top of the blanket/shield, steel maximum temperature and steel maximum thermal stresses. Figures 6 and 7 are the temperature distribution at the midplane and at the top of the inboard blanket/shield respectively. Figure 8 shows a plot of the effect of the first wall coolant channel thickness on the pressure drop in the first wall alone. The coolant feeding pipe diameter to each inboard module is 8.3 cm and the total volumetric flow rate to each inboard module is 24.4 liter/s. Hence, the total volumetric flow rate to the 32 modules is 781 liter/s.

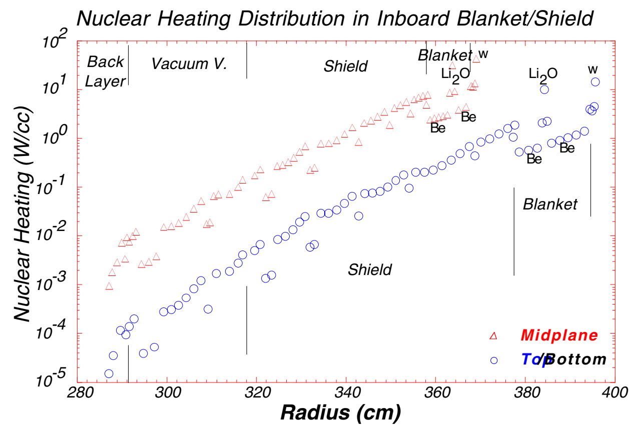

Figure 3. Nuclear heating distribution at the midplane and the top of the inboard blanket/shield.

OUTBOARD THERMAL HYDRAULICS ANALYSIS

The outboard blanket consists of 48 equal segments. Each segment has three modules, one central and two side modules. The central module is divided into an upper and lower module. All coolant connections (except for the lower central module, it is on the bottom) are at the top. The first wall is designed in the form of a steel/water/steel layer and the coolant channels are oriented toroidally. The first wall coolant is totally separate and independent from either the breeding zone or shield coolant. The first wall manifolds are located behind the breeding zone manifold and run poloidally. The first wall manifolds vary in toroidal width and have 30 mm radial depth. The first wall coolant traverses the first wall in the toroidal direction, then collects in the return manifold and exits the module at the top through the return pipe. The breeding zone consists

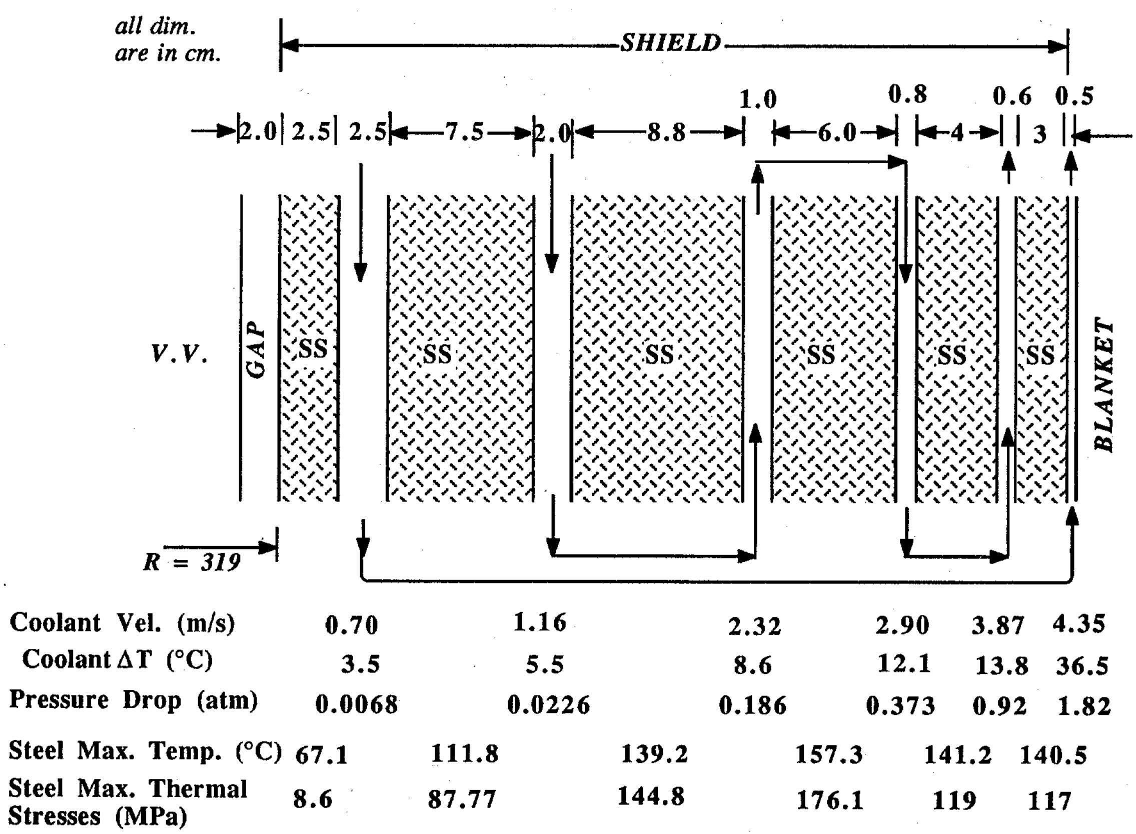

Figure 4. A schematic of the radial build of the inboard blanket

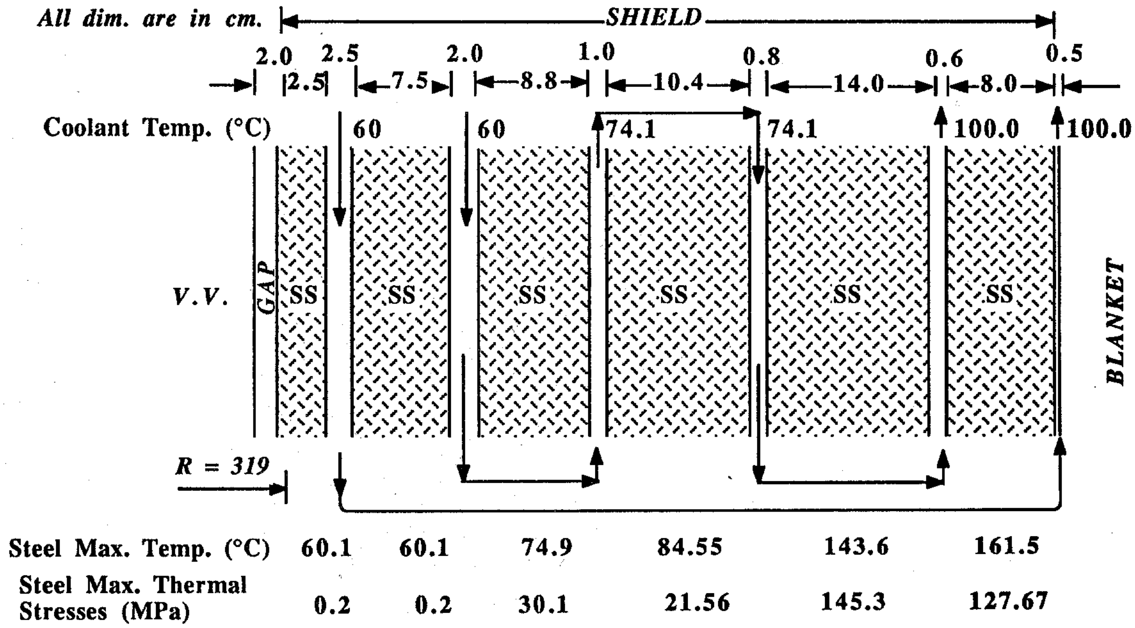

Figure 5. A schematic of the radial build of the inboard blanket

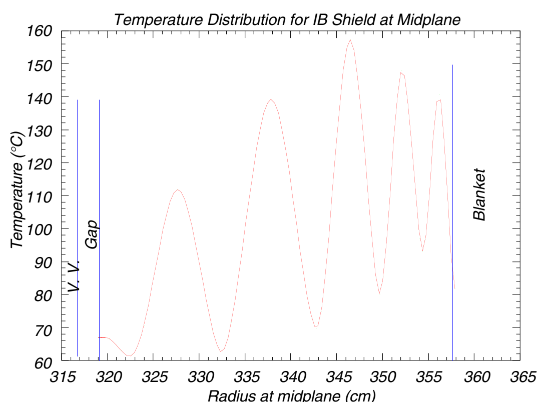

Figure 6. The temperature distribution for the I/B shield at the midplane of ITER.

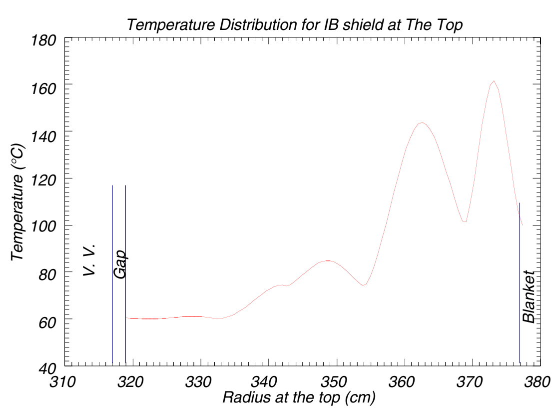

Figure 7. The temperature distribution for the I/B shield at the top of ITER.

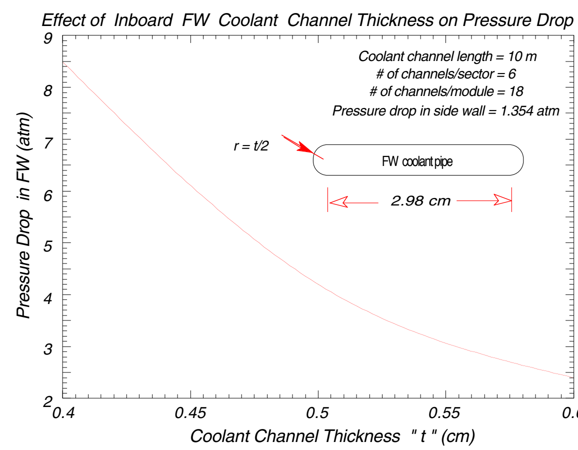

Figure 8. The effect of the first wall coolant channel thickness on the pressure drop in the first wall alone.

of beryllium plates interleaved with solid breeder plates and the cooling system. The details of the breeding zone can be found elsewhere.[2] Each breeding zone has 2 cooling panels. The cooling panels are 6 mm thick steel and have 2 mm × 22 mm cooling channels oriented toroidally. The coolant traverses the breeding zone in the toroidal direction, then collects in the return manifold and then is routed through the shield. The shield zone consists of steel plates with increasing thickness in the outward radial direction but with constant thickness in the poloidal direction. The plates are separated by coolant channels oriented poloidally. For the general configuration of the outboard blanket/shield see Reference 3. Figure 9 shows the radial build-up of one outboard module, general coolant

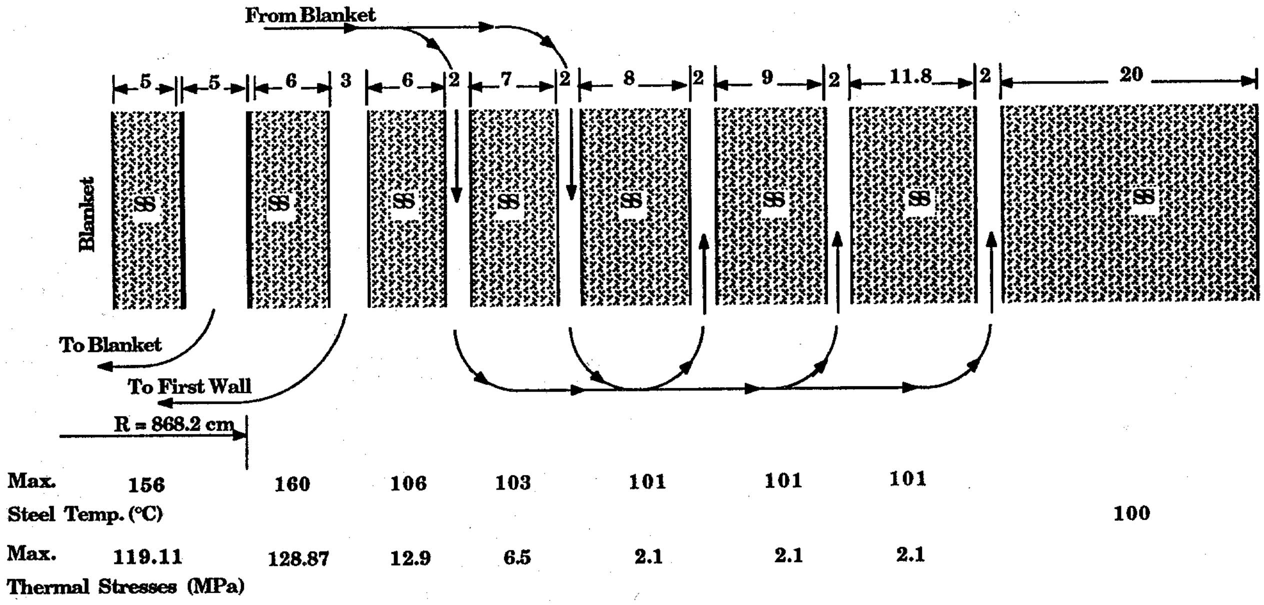

Figure 9. Outboard shield layout.

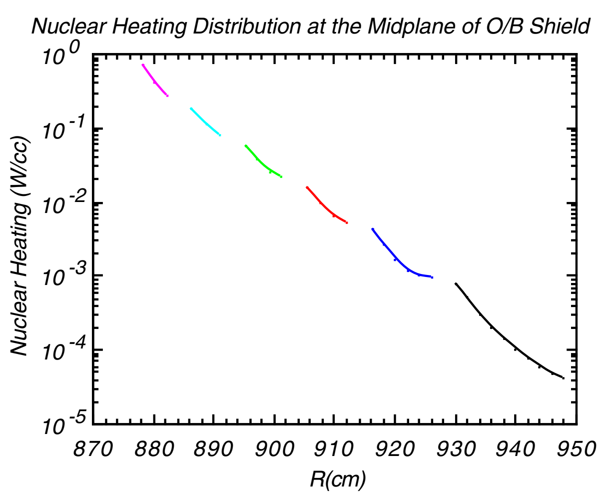

Figure 10. The nuclear heating for the outboard blanket/shield at midplane.

routing, maximum steel temperature and maximum thermal stresses in steel. Figure 10 shows a plot of the nuclear heating for the outboard blanket/shield at midplane. Figure 11 shows a plot of the temperature distribution for the outboard shield at midplane. The results of the thermal-hydraulics analysis for the outboard region are given in Table 2.

SUMMARY AND CONCLUSIONS

The US blanket for ITER consists of successive layers of solid breeder material and neutron multiplier cooled with water while the shield consists of alternating layers of 316 stainless steel and water. The optimum

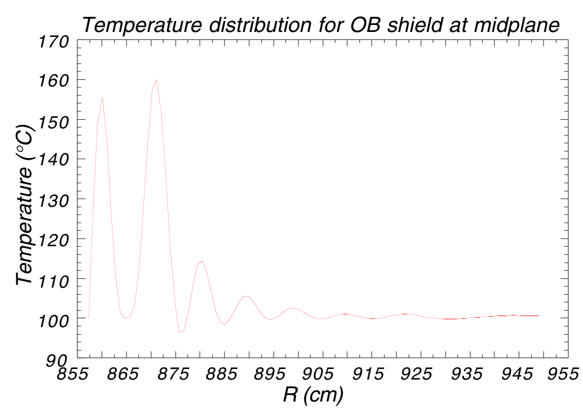

Figure 11. The temperature distribution for the outboard shield at midplane.

configuration of the blanket/shield has been determined from one-dimensional neutronics calculations. The blanket/shield radial build varies poloidally depending on the neutron wall loading. Nuclear heating results obtained from neutronics calculations have been utilized to perform the thermal analyses for the U.S. inboard and outboard blanket/shield for ITER. The temperature distribution in the blanket is strongly connected to the blanket optimum design and is calculated during the optimization process of the blanket. The optimum water content in the shield obtained by shielding optimization analysis, is about 20%. Several radial build configurations of the shield have been thermally analyzed. Thermal stresses in the shielding material determine the thickness of the different shielding material layers. In the blanket and shield thermal analysis, temperature distribution, overall coolant routing, thermal

Table 2. Outboard thermal-hydraulics parameters.

FIRST WALL	
Average Velocity	1.62 m/s
Pressure Drop	0.0686 atm
FIRST WALL HEADER	
Maximum Velocity	5.7 m/s
Pressure Drop (One Side)	0.107 atm
Total Pressure Drop in the First Wall Coolant System	0.283 atm
Total Volumetric Flow Rate/Module	90.6 liter/s
Total Volumetric Flow Rate	2900 liter/s
Coolant Feed Pipe Diameter to One Inboard Module (FW)	16.0 cm
BLANKET	
Average Velocity	
First Channel	2.1 m/s
Second Channel	1.2 m/s
Pressure Drop	
First Channel	0.194 atm
Second Channel	0.0739 atm
SHIELD	
Coolant Parameters	
All Channels Width	2.0 cm
Inlet Temperature	
First (Two) Channels	99.2 °C
Next (Three) Channels	99.8 °C
Exit Temperature	
First (Two) Channels	99.8 °C
Next (Three) Channels	100 °C
Velocity	
First (Two) Channels	2.17 m/s
Next (Three) Channels	1.45 m/s
Pressure Drop	
First (Two) Channels	0.122 atm
Next (Three) Channels	0.058 atm
Total Pressure Drop	0.18 atm
Total Volumetric Flow Rate/Module	92.0 liter/s
Total Volumetric Flow Rate	2944.3 liter/s
Coolant Feed Pipe Diameter to One Outboard Module	16.13 cm
COOLANT MANIFOLD	
Maximum Velocity	3.87 m/s
Pressure Drop (One Side)	0.027 atm
Total Pressure Drop in Blanket/Shield Coolant System	0.503 atm
Inlet Coolant Pressure	2.5 atm

stresses and pressure drops have been calculated for both inboard and outboard sides. Several coolant channel thicknesses in the shield have been optimizeed to minimize the pressure drop in the coolant loop without affecting shield performance. Different routes for the coolant are investigated to reduce the maximum temperature in the shielding material which in turn reduces thermal expansion effects during ITER operation. The coolant inlet temperature is 60°C, the outlet is 100°C and the maximum steel temperature in the shield is 150°C. In all cases, the maximum thermal stresses are within the prescribed limits for the material at the maximum operating temperatures. The thermal hydraulics system is tolerant to a fairly large range of uncertainty in nuclear heating and thermal hydraulics assumptions. However, the solid breeder performance is critical in that sense, and this issue has been addressed in Gohar's work.[2]

ACKNOWLEDGMENT

Funding for this work was provided by the U.S. Department of Energy.

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