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Analysis of Heat Transport and Pipe-routing Considerations for Blanket to Steam Generator for A Fusion Reactor

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Extraction of heat from the breeding blanket in tokamaks requires a network of heat carrying pipes from the blanket modules (BM) to the steam-generator (SG). Regardless of the type of the blanket concept, the operational requirements will mandate the RH compatibility and remote maintenance. The pipe-network will need to connect all BM in a given sector by sector-manifold, which in turn will itself integrate with a common ring header after it exits from reactor vessel. While a number of concepts for DEMO and blanket modules are available, there is not sufficient information on how is the pipe routing planned for the case when the entire vessel is covered by BM. In the case of ITER, as the tokamak is progressing in construction, there are good reference layouts for the piping inside the vessel for the shield blankets. But, DEMO being an electricity producing reactor, it is necessary to examine various options for heat extraction while observing thermal constraints arising from interfaces with vessel, ports, cryostat, building, etc. Given the expertise and investment made for ITER, it would not be surprising to imagine that the next step tokamak will closely match the major parameters of ITER, hence it is of interest to examine the compatibility of present piping and manifolds of ITER (for water cooled shield BM) with that of a possible helium-cooled BM reactor.

In this work, we carry out an analysis of the hydraulic parameters assuming three different fusion reactor configurations. Configuration A is of 500MW with about 1.25 MW power to be extracted from each BM and in this case the layout within the vessel is assumed similar to ITER. Configuration B and C are similar to DEMO and a compact power plant like ARIES-ST. It is assumed that all three configuration used helium as a coolant for the BM. A sketch of the overall scheme for all the three configurations has been shown in Fig. 1. The supply ring header (SRH) connects the SG to the 18 sectors (shown as circles) from where the extracted heat is taken to the SG by the return-ring-header (RRH). Each sector will itself have a manifold to connect all the BM within that sector by a sub-header. With regards to a real electricity producing power plant like DEMO e.g. the design proposed in 1, the requirement of power extraction per module may be ~4 times that of ITER. The complexity of piping for DCLL blanket has been partially elaborated by Federici et al.2. A greater challenge will be encountered while analyzing the pipe-routing for future compact reactors proposed, e.g. ARIES-ST3 where a DCLL concept has been elaborated by Tillack et al.[4]. In general, the space limitations for the sector-manifold and pipe-entry and exit through the ports (of vessel, cryostat, bio-shield etc.) are already tight, so it becomes even more challenging for designing as the available space is less but the power-handling capacity needs to be almost an order of magnitude higher than configuration A.

Table 1 shows the parameters for the 500 MW scenario (Configuration A), with 440 BM, like it has been considered for ITER [5][6]. The inlet/outlet pressure and temperature of helium is taken to be 80/79.9 bar (to get adequate heat-capacity) and 300⊠C/500⊠C [7], and the diameters of the different headers have been evaluated in order to keep the velocity ~ 100 m/s (well below the 10% of the sonic speed). The pressure-drop seems to be nominal for the required flow parameters. The length of the pipes from BM to RRH and from RRH to SG are considered as ~40m and ~20m respectively. Following Ref. [8], we consider the diameter to be of ~ 48 mm for all the BM outlet pipes going up to RRH. The pressure drop in the above routing is found to be approximately ~ 2.25 bar. The power loss by radiation from all the hot pipes (500C) to the vacuum vessel (~200 C) seems negligible (1.23 MW). The detailed results of thermal and stress analysis of the piping network will be presented.

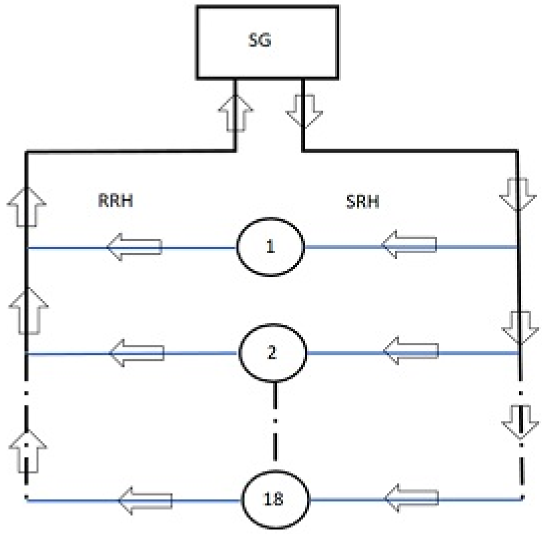

Figure 1: FIG:1

TABLE 1: Flow Parameters of Helium

Parameter	Ring Header	Sub Header	Blanket Module
Mass Flow rate (Kg/s)	482	24.4	1
Diameter of Pipe (m)	1.02	0.12	0.04
Pressure Drop (bar)	1.2	1.13	0.01

Figure 2: Table:1

The hydraulic parameters for the three configurations have been given in Table 2. Detailed results will be presented for all three cases.

TABLE 2: Hydraulic Parameters for different Tokamaks

Parameter	A	B	C
Fusion Power (MW)	500	2500	2980
He Mass flow rate (kg/s)	482	2170	1444
He Temperature (In/Out)°C	300/500	300/500	300/525
He Pressure (bar)	80	80	120
Velocity (m/s)	100	80	125

Figure 3: Table:2

In summary, the attempt to analyze piping from BM to SG is in itself an important step because the present designs (whether of ITER TBM[9] or DEMO TBM [10]) have considered only equatorial dedicated ports. The blanket piping design for considering entire vessel area covered by breeding blankets will present even bigger challenges of compact machines like DEMO-FNS [11].

Keywords: Fusion power plant design, Piping layout, Blanket module

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