Engineering Solutions for Components Facing the Plasma in Experimental Power Reactors

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Abstract

A review of the engineering problems related to the structures in front of the plasma of experimental Tokamak-type reactors is made. Attention is focused on the so-named "first wall", i.e. the wall side of the blanket segments facing the plasma, and on the collector plates of the impurity control system, in particular for the case of the single-null poloidal divertor. Even if the uncertainties related to the plasma-wall interaction are still relevant, some engineering solutions which look manageable are identified and described.

1. Introduction

In experimental fusion power reactors such as INTOR [1] and NET [2], there will be a number of components facing directly the plasma which can be defined, in general, as first wall components. These components, subject to surface heat and neutron cyclic loads, are:

a) the side in front of the plasma (the so-named current "first wall") of the blanket segments which are arranged inside the vacuum vessel and which represent renewable parts of the reactor;
b) the collector plates and adjacent ducts of the systems (divertor or pumped limiter) intended for ash removal and impurity control. In the current designs these plates are distributed toroidally and they are segmented, as the blanket units, for an easy removal;
c) the beam tubes and the plugs needed for plasma heating either by neutral injection or radiofrequency systems.

In the following we will analyse some of the features related to the design of a) and b) because these two types of components are related among them and because they are characteristics of the type of problems which will happen in the heating system, once this will be chosen in a further phase of the design.

Concerning the impurity control systems, the consideration is limited here to the so-named "poloidal divertor" because this seems today the most attractive from the point of view of an INTOR/NET machine [3] and also because the input data from the plasma boundary conditions seem more defined.

2. Input Data from Plasma

Typical specifications of the plasma boundary conditions in the first wall components are shown in Table 1. These data, taken from recent INTOR studies [4] correspond to the so-named "high recycle regime" of the single-null poloidal divertor case. Under these hypotheses the plasma temperature at the edge is very low, then the erosion due to sputtering of the charge exchange neutrals is reduced as compared to previous INTOR evaluations [1]. The way to protect the first wall from plasma disruption effects (vaporization and melting) is still an open problem. This is due to the uncertainties on the possible instability of the melted layer, which is evaluated to be 0.05 - 0.1 mm per disruption. Protection by coatings such as TiC or SiC seems attractive. Indeed, recent experiments at JRC-Ispra have proved that the coating impedes the melted layer of the base metal (stainless steel) to be displaced.

The data given in Table 1 for the first wall are average values through the toroidal chamber. In reality, due to the toroidal plasma shape (of D-type), the power wall loading will vary along the poloidal direction; according to recent calculations [5], one has to expect a peak factor in the maximum thermal flux position as compared to the value of Table 1 of about 1.6, so leading to a maximum wall loading of about 0.2 MW/m^2.

3. First Wall Concepts

There is now a consensus in considering the first wall for experimental power reactors separate from the breeding blanket structures and with an independent cooling. A concept which has been developed at JRC-Ispra [6] in the framework of the INTOR studies and which is now adopted in the NET predesign evaluations is that of conceiving the first wall as one of the sides (this facing the plasma) of a box which envelopes the blanket segment, as shown in Fig. 1, which represents the cross-section of one segment on the equatorial plane of the reactor. Fig. 2 is another view of a possible configuration. Such a solution presents a number of attractive features for the design of the renewable components inside the vacuum vessel, namely:

- the surface facing the plasma is continuous for every blanket segment and is shaped according to the magnetic surfaces encircling the scrape-off layer, so improving the plasma vertical stability and limiting the vacuum pumping requirements in the plasma chamber;
- a secondary vacuum, or a controlled atmosphere, can be created inside the box, so eliminating the risks connected with possible leakages in the blanket units and related coolant circuits, as well in the coolant circuit of the first wall itself;
- a supplementary barrier is obtained in case of breeder-water interaction consequent to an accident in the blanket (i.e. coolant pipe rupture);
- a larger flexibility is left to the design of the breeding blanket units and on the choice of their cooling system, the only condition being that of fitting with the external envelope represented by the first wall box structure;
- the problem of the sacrificial layer needed on the first wall surface to deal with erosion (if the plasma edge condition will be changed in the future) and plasma disruption is separated from that of the breeding blanket thermomechanical design.

The main drawbacks of such a concept, as compared to that of a non-separate first wall (i.e. a first wall integrated on the breeding units) can be:

a) an increase of the structural material in front of the plasma, then a decrease of the breeding ratio;
b) the difficulty of designing the box-like structure, due to the expected high thermal stresses at the corners of such an envelope;
c) the need of a separate cooling circuit.

Point a) does not appear to be crucial in case of an experimental reactor; point b) is under study: the first results so far obtained will be briefly mentioned in the following. Concerning point c), the expected penalty seems to be acceptable, at least in case of a single type of coolant both for first wall and breeding blanket.

The first wall box proposed by JRC-Ispra comprises a shell of stainless steel, bonded to a row of parallel extruded tubes through which the coolant (water or helium) flows in toroidal direction across the shell width, from one inlet to one outlet collector. The cooling pipes can be attached to the shell by brazing (e.g. Nicrobrazing); the feasibility of using this technique in the fusion environment is under investigation. Other geometrical arrangements of the coolant flow and related fabrication techniques can be devised for the first wall. In Fig. 3 are summarized some of these options with their main technological features, in particular concerning the welding points which represent the most important aspect for the proper choice of the first wall design [7]. Solutions A, B and C concern different possible procedures of assembly extruded profiles with circular cross-section of the hole where the coolant flows, in such a way to realise a continuous first wall shell with the cooling channels integrated from the inner side of the box. Solutions A refer to assembly by brazing, solutions B by welding and solution C by metal deposition (e.g. plasma spraying). Solution D represents an alternative concept developed for NET by the NET-team at Garching, which uses cooling channels with rectangular cross-section.

The problem of the feasibility of the first wall-box from the structural mechanics point of view requires three-dimensional stress analyses with a sufficient degree of representativity of the real geometrical and operating conditions. An effort in this direction is in progress at JRC-Ispra. First results of stress analyses for the reference NET configuration (AISI-316 type as structural material, water as coolant) have been presented at the last SOFT-Conference in Varese [6]. Recent 3-D calculations enable to draw some first conclusions, namely:

a) the highest stresses arise at the edge of the box side in front of the plasma; they are about 1.6 times higher than the value obtained in the central part of the shell;
b) displacements of the order of few cm have to be expected in the structure; such type of deformations will have an impact in the design and accommodation of the segments but possibly will be manageable.

The absolute value of the maximum stress and strain will strongly depend on the thickness of the protective layer in front of the plasma. Under the hypotheses mentioned in section 2, for an AISI-316 first wall and a thickness of 5 mm for the protective layer, the maximum stresses should remain in the range 350-380 MPa, values which are not far from the allowable secondary stress for the case of annealed stainless steel ($\sim$ 330 MPa). The behaviour of the first wall in conditions of thermal fatigue, due to the cyclic nature of the heat loads, will be checked by tests in progress at JRC-Ispra [8].

4. Impurity Control Systems

The impurity control system (e.g. the single-null poloidal divertor) has to be designed in order to satisfy a number of functions, namely:

a) absorb the fraction of the total power transported in the plasma scrape-off layer to the collector plate;

b) reduce to minimum the level of impurities entering in the hot core of the plasma;

c) remove the helium ash produced in the DT reactions.

Consequently the components in front of the plasma connected with the impurity control system are:

- the particle collector, which covers axial-symmetrically the divertor region;
- the ducts to pump away plasma exhaust.

As for the first wall/blanket case, there is a consensus now in subdividing the impurity control system in sectors (or modules), which can be easily removed for substitution, including both the collector plate and the related first part of the exhaust pumping ducts. Various configurations have been devised for this purpose in INTOR and NET studies and they are discussed in another paper of the Conference [9]. The tendency is that of realising the removable modules in form of "cassettes" (Fig. 4) which can be separately replaced for maintenance or repair. Then one can assume that the lifetime both for the collector plate and the collector chamber walls, representing the inlet of exhaust plasma removal circuit, will be the same. In the case of single-null poloidal divertor which, as mentioned, represents today the most attractive solution for INTOR/NET (Fig. 5), typical engineering problems are:

a) the design of the collector plate and related cooling;

b) the design of the divertor chamber walls and related cooling;

c) the evaluation of the vacuum pumping requirements for the plasma exhaust removal.

The choice of the collector plate material in front of the plasma is related to the boundary conditions which one wishes to obtain. Recent studies [3] have shown that high-Z materials such as W or Mo are well suited to get low plasma edge temperature and then low erosion by sputtering. This approach has been followed at JRC-Ispra where the engineering feasibility of W-protected collectors has been investigated. The collector proposed consists of two layers toroidal plates, suitably shaped according to a proper poloidal profile; they are composed of a layer of W-Re 5% alloy in front of the plasma and an underlying matrix of copper joint together by melting of the copper on the tungsten (Fig. 6). Power fluxes incident on the divertor surface are removed by copper alloy cooling pipes brazed below the copper matrix and arranged in the radial direction with respect to the main axis. As mentioned, the impurity control system is subdivided into a number of independent modules (24 or more) placed side-by-side in the toroidal direction. Each module contains two plates, one on the inboard and one on the outboard side, as shown in Fig. 5, each one made by a high number of W-Re 5% tiles, typically of dimensions 40x30 mm, attached to the copper matrix. The two plates are arranged so to have an inclination of about 15° and 10°, respectively, as compared to the separatrix line. The power distribution on the plates has a peak value of $\sim 3.0 - 3.5$ MW/m^2. 3-D thermal and elastic stress analyses have been carried out to evaluate the cooling requirements and mechanical behaviour of the divertor plates [10]. Typical results are shown in Table 2. The thickness of the W-layer has been taken to be 5 mm, corresponding to a lifetime of about 5 years by assuming an average availability of 20%.

Apart from Cu, the maximum stresses seem not to exceed the allowable secondary stresses, as obtained by assuming the ASME-criteria also for the divertor materials. In any case these calculations have to be completed by an elasto-plastic analysis where also the problem of the residual stresses due to fabrication processes will have to be taken into account. On the other hand, the thermal fatigue effects should also be checked, once experimental results will be available. The feasibility of the W-Cu bonding and its behaviour under cyclic load conditions has been experimentally investigated at JRC-Ispra [11] and the results insofar obtained look promising.

5. Conclusions

During the past few years an important contribution has been given, mainly through the INTOR studies, to the identification of the engineering problems related to the "first wall" and impurity control systems of the future experimental reactors of Tokamak-type.

As far as the "first wall" is concerned, the concept of a box, separately cooled and acting as a containment for the breeding blanket segments, looks attractive. The real practicability of the concept, mainly from the point of view of the feasibility of the mechanical attachment to the underlying structures, remains, however, to be proved.

In case of the divertor type of impurity control, the solution involving W tiles on Cu heat sinks seems to offer chances for practical applications, provided that the plasma boundary conditions, as evaluated today, are proved. In any case a large experimental and analysis effort is still needed before getting a feasible approach for this component which certainly looks as one of the most delicate in proving the operability of a fusion reactor plant of Tokamak-type.

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TABLE 1 - Engineering specifications for the high recycling regime of the single-null poloidal divertor (INTOR, Phase 2A)

First wall	
- Radiation power	49 MW
- Charge exchange power	1 MW
- Average wall loading	∿ 0.12 MW/m^2
- Erosion rate due to sputtering	∿ 0.9 mm/a
- Erosion rate due to a plasma disruption	negligible
- Melt layer produced by a plasma disruption	0.05-0.1 mm
- Number of disruptions in the lifetime	∿ 10^3
- Neutronic heat deposition in first wall	∿ 18 W/cm^3
- Number of thermal cycles foreseen during the lifetime of the plant	∿ 10^5
Divertor	
- Total power to each collector plate (inner and outer)	34.5 MW
a) due to kinetic energy of DT plasma	16.2 MW (peaked)
b) due to recombination of DT ions at surface	9.8 MW (peaked)
c) radiation from diverted plasma	6.5 MW (uniform)
d) radiation from main plasma	2 MW (uniform)
- Peak power load to outer plate (target perpendicular to magnetic surfaces)	18 MW/m^2
- Peak power load to inner plate (target perpendicular to magnetic surfaces)	14 MW/m^2
- Erosion rate due to sputtering (outer collector plate, 15° inclined)	5 mm/a

TABLE 2 - Maximum temperatures and stresses on the collector plates (W-Re tiles on Cu heat sink) for a wall loading of ∿ 3 MW/m^2

Zone	Material	Maximum temperature T_{max} (°C)	Maximum Von Mises stress σ_{max} (MPa)	*Allowable secondary stress at T_{max} 3Sm (MPa)	Ratio $\sigma_{max}/3Sm$
Surface facing the plasma	W-Re	400	240	430	0.6
Connection between W-Re and copper	W-Re	180	410	606	0.7
	Cu	180	125	70	1.8
Coolant side (coolant channels)	Cu	160	60	100	0.6

* Sm = 1/3 Su for W-Re; Sm = 0.9 Sy for Cu.

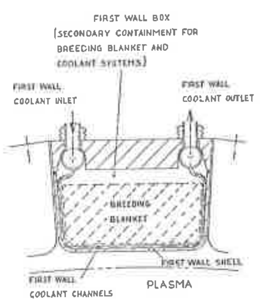

Fig. 1 First wall box concept (cross-section).

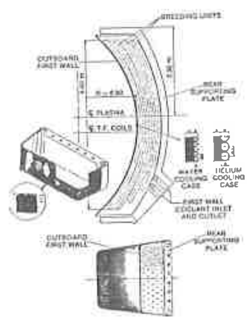

Fig. 2 Side view and perspective view of a possible first wall configuration.

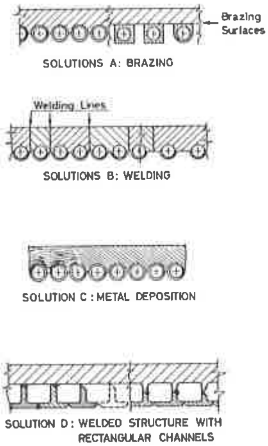

Fig. 3 Different technological options for the first wall panel.

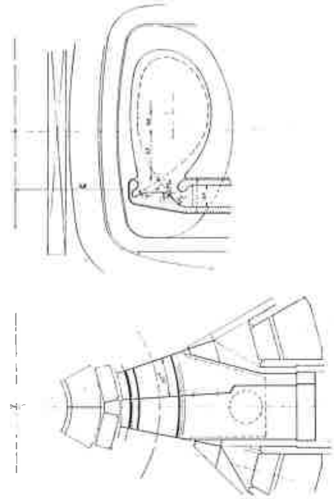

Fig. 4 The single-null divertor region in INTOR.

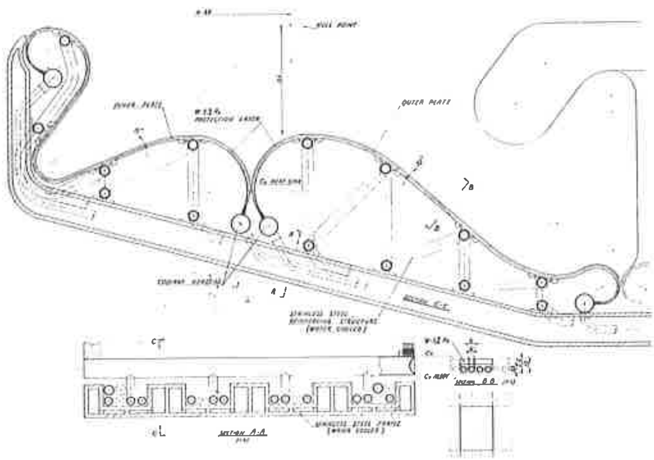

Fig. 5 Details of the inner and outer collector plates in INTOR.

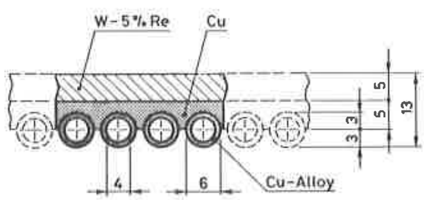

Fig. 6 Cross-section of the W-Cu collector plates.

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