2.3 Blanket

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2.3.1 Blanket Function and Main Components

The basic function of the blanket system is to provide the main thermal and nuclear shielding to the vessel and external machine components. The blanket system is also designed to make possible the planned partial conversion (outboard area only) of the shielding blanket to the breeding blanket in a later stage of operation (if justifiable).

The basic concept of the blanket system is a modular configuration with a mechanical attachment system. The blanket modules (BMs) are attached directly to the vacuum vessel. Manifolds that supply cooling water to the modules are mounted on the vacuum vessel behind the modules.

The ITER-FEAT blanket module design is aimed of minimising, (a) the module cost, (b) the radioactive waste, and (c) electromagnetic (EM) loads due to disruptions/VDEs (vertical displacement events). The module configuration consists of a shield block to which separate first wall (FW) panels are mounted. The use of multiple flat panels for the FW provides a simple unit design and reduces the associated machining costs. A deeply slitted configuration minimises induced eddy currents and EM loads in the module. The main shield blanket parameters are given in Table 2.3.1-1.

Table 2.3.1-1 Shield Blanket Parameters for 500 MW Fusion Power Operation

Parameters	Unit	Value
Total blanket thermal power	MW	690
Heat flux on first wall (FW), average/max.	MW/m^2	0.25 / 0.5
Heat flux on limiter, average/max.	MW/m^2	~ 3 / ~ 8
Neutron wall loading, average/max.	MW/m^2	0.56 / 0.78
Number of modules, total/NB port modules		421 / 17
First wall surface area	m^2	680
Weight of modules	t	1,530
Weight limit for module	t/mod.	4.5
Typical blanket module dimension (Inboard equator)	mm	1415x 1095x450

2.3.2 Overall Configuration

2.3.2.1 Shield Blanket Module Arrangement

The 17-fold segmentation of the shield modules in the poloidal direction is established by taking into account the weight limit of 4.5 t per module imposed by the remote maintenance equipment, and the desire to minimise the number of modules for cost reduction. The module toroidal length varies from 1.25 to 1.96 m and the poloidal length from 0.85 to 1.24 m except for modules in the NB port region. The arrangement of shielding blanket modules is shown in Figures 2.3.2-1 and 2.

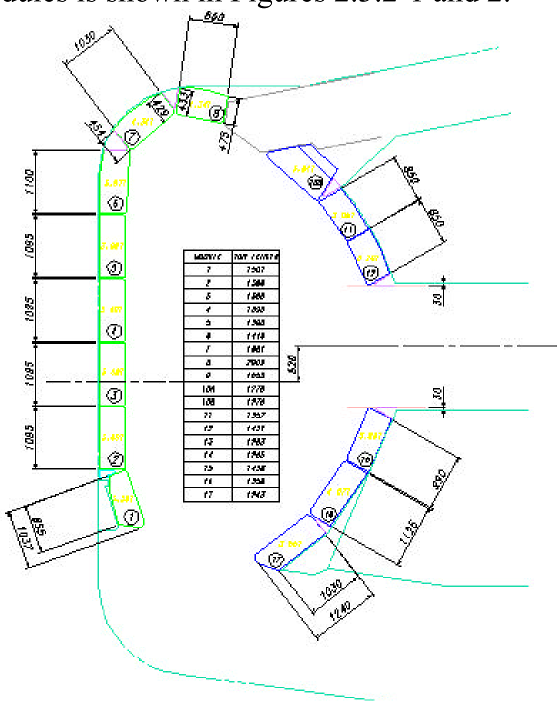

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Figure 2.3.2-1 Shield Blanket Module Segmentation
(poloidal cross-section through a port)

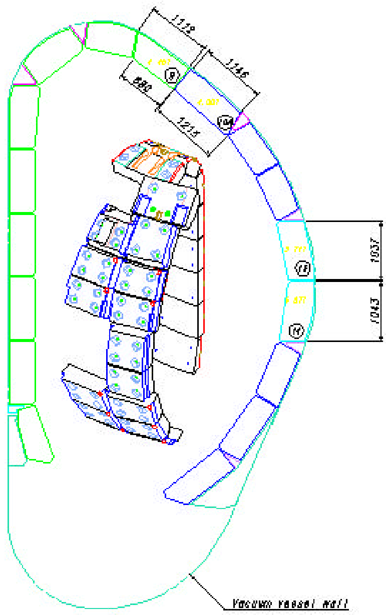

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Figure 2.3.2-2 Shield Blanket Module Segmentation
(poloidal cross-section in TF coil mid-plane)

The FW/shield modules have a toroidal segmentation of 20º (18 modules) on the inboard, and 10º (36 modules) on the outboard. In the region of the upper ports there are 18 modules between the upper port plugs. In the 13 positions between the 14 normal equatorial ports a polodal segmentation of 2 is used. A different segmentation is used in the region of the NB ports; 17 special modules are used to cover the area between port 3 and port 8.

2.3.2.2 Materials

Materials used for the primary modules and port limiter are shown in Tables 2.3.2-1 and 2, which also show the operating temperatures deduced from the thermal analysis and the estimated neutron damage for a local 14 MeV fluence at the first wall of 0.5 MWa/m^2.

Table 2.3.2-1 Primary Module Materials

Component	Material	Max. temp. (°C)	Neutron damage (dpa)
First wall			
- plasma-facing material	Be (S-65C or DShG-200)	700	1.6
- heat sink	DS Cu Al25-IG or CuCrZr-IG)	400	5.3
- tube and structure	SS 316L(N)-IG	160	2.7
Shield block	SS 316L(N)-IG	340	2.3
Flexible support			
- cartridge	Ti-6Al-4V with ceramic coating	200	0.03
- bolt/collar	Inconel 718/DS Cu Al60 with ceramic	210	0.15
Key structure / pad	SS 316L(N)-IG/bronze with ceramic insulation and MoS_2 coating	260	0.09
Electrical connection			
- bent sheets/support block	CuCrZr-IG/SS 316L(N)-IG	220	0.05
- bolt	Inconel 718	300	0.05
Hydraulic connection/manifold	SS 316L(N)-IG	150	0.05

Remarks: DS - dispersion strengthened, SS - stainless steel, IG - ITER Grade

Table 2.3.2-2 Port Limiter System Materials

Component	Material	Max. temp. (°C)	Neutron damage (dpa)
Limiter module			
- plasma facing material	Be (S-65C or DShG-200)	740	1.6
- heat sink	DS Cu Al25-IG or CuCrZr-IG)	450	5.3
- structure	SS 316L(N)-IG	160	3.4
Main structural and shield part and alignment system bellows	SS 316L(N)-IG	150	0.02
Flexible pivot	Ti-6Al-4V with ceramic coating	200	0.02
High strength bolt	Inconel 718	200	0.02

2.3.2.3 Cooling and Baking

The heat deposited in the blanket modules is cooled by the three independent loops of the primary first wall/blanket (PFW/BLK) primary heat transfer system (PHTS, see also 2.3). Each loop has the same coolant capacity, and feeds three 40º sectors. Adjacent 40º-sectors are cooled by different loops as a safety feature. The maximum heat load for the blanket modules is 690 MW, including the front part of the port plugs except for the port limiters. Table 2.3.2-3 shows the coolant flow condition for the blanket modules. The PFW/BLK PHTS loops cool the 421 blanket modules and the 18 in-port components in the upper ports. At the equatorial level, 12 in-port components and 3 NB liners are cooled by PFW/BLK PHTS loops, 2 port limiters are cooled by the divertor/limiter cooling loop (DIV/LIM PHTS) while the obscured port (port 7) has no port plug.

Table 2.3.2-3 Coolant Flow Condition for Blanket Modules

Maximum Heating Power (MW)	690
Number of PHTS loops	3
Cooling Operation	
- Coolant Inlet Temperature (°C)	100
- Temperature Rise (°C)	48
- Coolant Mass Flow Rate (kg/s)	3,378
- Coolant Inlet Pressure (MPa)	3.0
- Pressure Drop* (MPa)	≤ 1.0
Baking Operation	
- Coolant Inlet Temperature (°C)	240
- Coolant Inlet Pressure (MPa)	4.3

* in the in-vessel part of the blanket cooling system

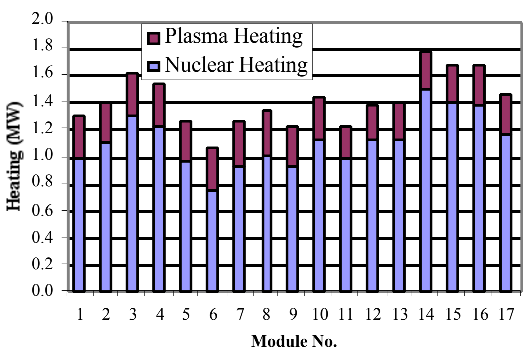

Figure 2.3.2-3 Heat Deposited in Each Module
(see Figure 2.3.2-1 for module no.)

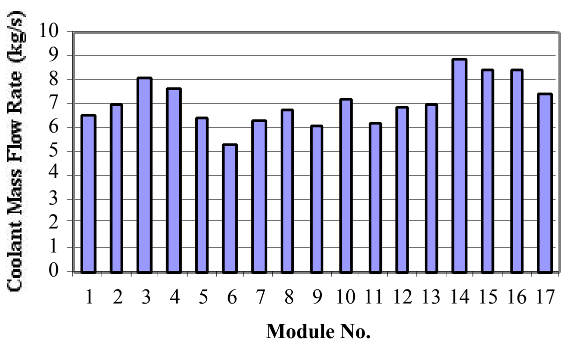

Figure 2.3.2-4 Required Coolant Mass Flow Rate for Each Module

The heat deposited in each module consists of nuclear heating and plasma heating as shown in Figure 2.3.2-3. The nuclear heating for each module is estimated based on the latest nuclear analysis and conforms with the nominal total nuclear heating of 554 MW. The plasma heating for each module is estimated assuming a uniform heat flux with nominal total value of 136 MW. The total surface area of the first wall is 680m^2, so the average heat flux due to plasma heating is 0.2 MW/m^2. Figure 2.3.2-4 shows the required coolant mass flow rate for each module. This required flow rate will be achieved using two kinds of orifices. One is installed inside the branch pipes of the blanket modules, providing the flow balance between blanket modules connected in parallel to the manifold. The other is installed inside the pipes around the interface point between the blanket system and the coolant loop to distribute the required mass flow to each manifold from the header of the PHTS.

2.3.3 Primary Modules

2.3.3.1 General

The module configuration consists of a shield body to which a separable first wall (FW) is mounted. The separable first wall has a facet geometry consisting of multiple flat panels, where 3-D machining will not be required. This produces a simple unit design with low associated machining costs. Several FW panels can be produced in each hot isostatic pressing (HIP) cycle. The use of small separate FW panels eases the Be tile HIP joint and will minimise the scrap rate. The separation of the FW from the shield body allows manufacturing process costs to be minimised, and solid HIP will be used only for the FW panel fabrication. The use of multiple panels also makes possible the replacement of individual damaged units, reducing nuclear waste volume, and it simplifies the repair and replacement methods in the hot cell. A configuration with deep slits minimises the induced eddy currents and EM loads.

The blanket module has four or six separate FW panels depending on the option chosen for FW attachment. Two attachment methods are being considered: one is based on a central beam attachment, which is connected to a shield block at its rear side (option B, see Figures 2.3.3-1), and the other is an attachment with bolts and small shear ribs to support EM loads and to prevent sliding due to thermal expansion (option A, see Figures 2.3.3-2). The shield block has radial cooling channels for both options. For hydraulic connection, two option are under consideration, i.e. separate connectors to use simple water pipes, and a co-axial connector to minimise the number of seal welds (see Fig. 2.3.3-3). In the following section, the option B primary module with separate connectors will be described in detail.

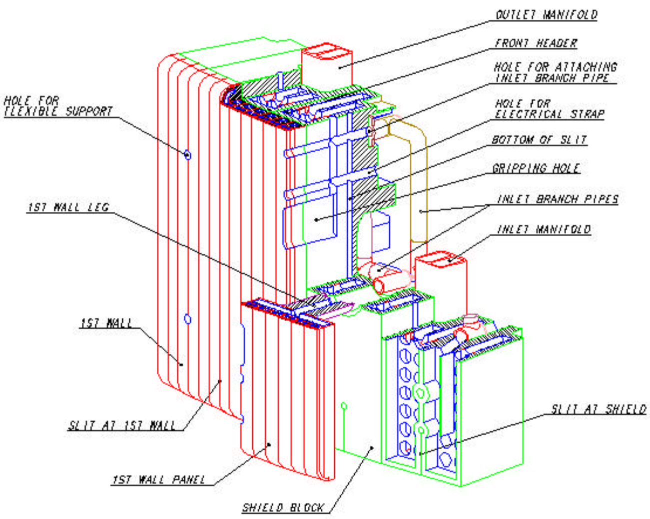

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Figure 2.3.3-1
FW Panel/Shield Block with Central Beam Attachment

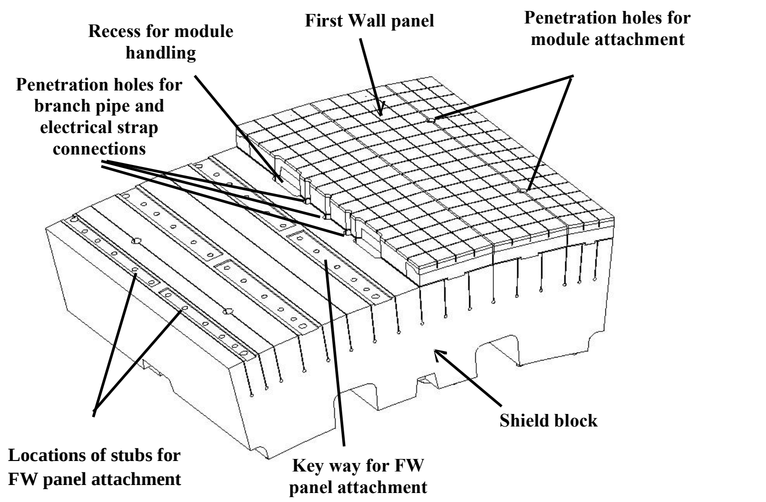

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Figure 2.3.3-2 FW Panel/ Shield Block Attached with Bolts and Shear Ribs

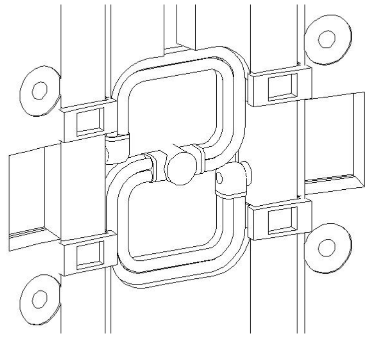

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Figure 2.3.3-3 Layout of Co-axial Hydraulic Connection for the Inboard Blanket Module

2.3.3.2 First Wall

The FW is composed of flat panels and attachments to the shield block. Each FW panel consists of beryllium armour in the form of tiles attached to a Cu-alloy heat sink plate internally cooled by SS cooling tubes. The Cu-alloy plate is attached to a ~ 50 mm thick steel backing plate that has a structural plus shielding function. A simple straight poloidal cooling channel layout is used. The FW panel will be manufactured using solid HIP as the reference method, with powder HIP also considered as an alternate method if fabrication is possible within acceptable dimensional accuracy.

The FW with central beam attachment (option B) consists of four individual, 81 mm thick, flat panels, as shown in Figures 2.3.3-4 to 6. The panel is provided with narrow (2 mm) poloidal slots, between every other FW cooling channel. The slots are continuous in the heat sink but are discontinued half way through the SS backing, where the toroidal inlet/outlet collectors are located. Front access penetrations (diameter 30 mm) are made in the poloidal slots. They are used to access the blanket module support, hydraulic, and electrical connections, for remote handling. On either side of the penetrations the cooling tubes are displaced inside the Cu-alloy heat sink to provide space for the holes.

The central beam is welded to the rear side of the FW panel at one end and is attached to the shield by welding (or by a bolted connection) on the other. The supporting beam is inserted inside a suitable penetration in the shield block. Its mounting and dismounting is made from the rear of the module. A weld thickness of about 12 mm is needed if the beam is supported at its end as a cantilever. The main advantages of the attachment system are the free thermal expansion of the FW panel, and the location of all the attachment fixtures on the back of module, where the volumetric heating and the irradiation damage is lowest.

The FW coolant inlet and outlet supply channels are routed inside the supporting beam. The inlet is connected to the main blanket supply lines, the outlet to the shield block headers.

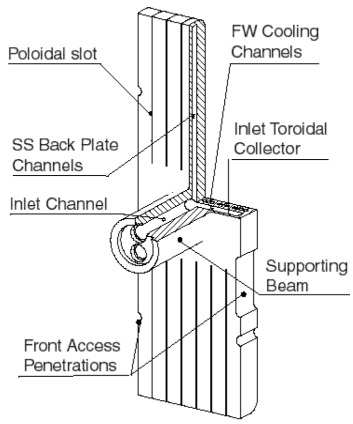

Fig. 2.3.3-4 FW Panel Rear Side View

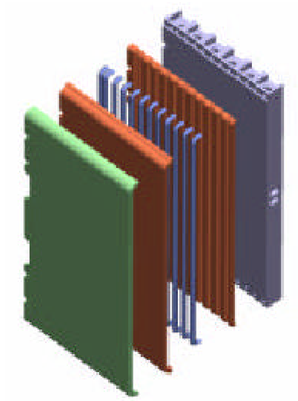

Fig. 2.3.3-5 FW Panel before Assembly

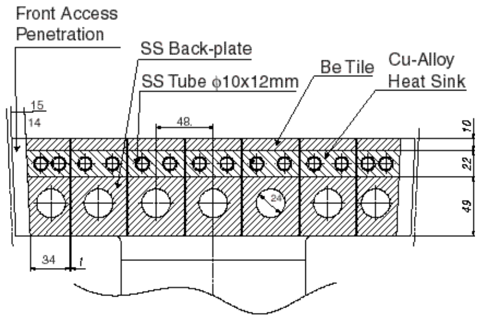

Figure 2.3.3-6 FW Panel Cross-section

2.3.3.3 Shield Block

The shield block cooling scheme consists of water headers that distribute the coolant in radial channels. The coolant flow is complicated by the presence of cut-outs in the rear part, radial access holes, and penetration holes for FW legs. Different schemes of cooling layout have been investigated: (a) one header in the front of the module and radial flow though co-axial channels, (b) front and rear headers, (c) a header in the middle of the module with co-axial flow. The layout with one front header has been preferred as in the rear part of the shield block there are many cut-outs and high stressed regions near the supports and keyways.

The shield block has a radial thickness of 0.37 m. It consists of four flat forged blocks electron beam (EB) partially welded together at the rear side (see Figure 2.3.3-7). Three poloidal slits of 0.17 and 0.19 m depths are retained on the front side between forged blocks and four slits 0.16 m deep are also cut from the top and from the bottom to reduce EM loads. Four penetration holes with 170 mm diameter allow the installation of the FW central beam. Eight 30 mm diameter access holes through the inboard FW/shield block provide front access to the bolts used for the flexible and electrical connectors and for laser-seal welding of the branch pipes. In the outboard module, 2 additional 30 mm diameter access holes in the mid-plane of the module (with 570 mm poloidal span) are provided for the temporary attachment of the module onto the VV during maintenance.

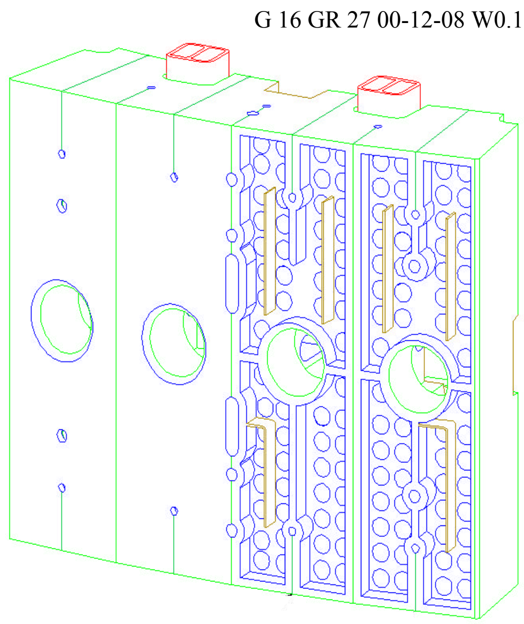

Figure 2.3.3-7 Front Headers and Radial Coolant Holes for Inboard Shield Blocks

The inboard and outboard modules have different arrangements of the support structure to resist the EM moments. Shear keys are used in the inboard (see Figure 2.3.4-2), whilst stub keys are used for the outboard modules (Figure 2.3.4-3). The rear surface of the inboard blocks is highly grooved to provide cut-outs for 3 keyways, 2 electrical straps and coolant branch pipes, coolant connections to FW panels, and cooling manifolds. In the outboard blocks, there are 4 cut-outs for the keyways, while there are no recesses for the cooling manifolds as they are in between modules.

The shield block can be fabricated by conventional drilling/plugging and machining/welding of flat forged blocks. Power HIP can also be applied to the shield block fabrication with different header configurations, such as the option b) described above.

2.3.4 Module Attachments

The mechanical attachment of the blanket module to the VV provides accurate positioning, movement allowances and load capability. It transfers radial loads while allowing toroidal/poloidal movements of 1 mm due to the different structural rigidities and thermal responses of the modules and the VV. The hydraulic and electrical connection have a similar compliance. The mechanical attachment includes four radial supports and three keys in the poloidal/toroidal directions. Electrical insulation in the mechanical attachment prevents the electrical current flowing through the supports and helps to reduce the EM forces. The hydraulic connection is metallic without any electric break, is compact, and is located in the centre of the module, in between two copper straps handling the large plasma halo current.

2.3.4.1 Flexible Support

The flexible supports react to the module loads in the radial direction while being compliant to the other directions. This feature is provided by a tubular cartridge with axial slits in the tube wall (Figure 2.3.4-1). One end of the cartridge is threaded outside M150 and connected to the VV. The other end is closed by an internal flange, which is used for bolting to the module. The supports are recessed in the VV wall in a shielded position where the low nuclear heating can be removed by thermal conduction. The flexible supports are installed, with the bolts in parking position, in the vessel recesses after the complete welded vessel assembly. The tolerances of the vacuum vessel are compensated by custom machining the large threaded end of the cartridge. A margin of ± 10 mm is available in the axial direction and ± 5 mm transversely. These adjustments allow accurate positioning of the first wall.

The electrical insulation consists of a ceramic coat applied to the flange of the cartridge and to the collar-washer of the bolt. The supports hold the modules by M45 bolts which are driven through the 30 mm access holes. The tool engages the bolt from the tip but does not need a large torque, because the elongation for the preload is achieved using differential thermal expansion provided by a heater in the 10 mm axial bore. The flexible cartridges are made from Ti-alloy (Ti-6Al-4V) due to its high strength, low Young's modulus and adequate toughness after irradiation. The high strength bolts are made of Inconel 718 (ASTM B637) precipitation hardened. The bolt preload (650 kN) is maintained under nuclear heating, and the number of cycles is within fatigue limits.

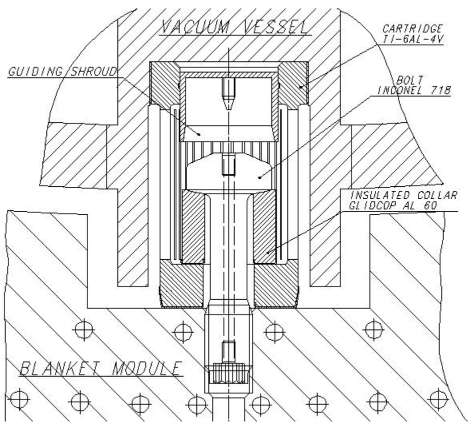

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Figure 2.3.4-1 Cross-section of a Flexible Support inside a Stub Key

2.3.4.2 Key

Each inboard module has three keys to react the radial torque, the poloidal forces and the smaller toroidal forces (Figure 2.3.4-2). The keys are fitted with bronze pads sliding against the key-ways of the modules during relative thermal expansion. Friction is reduced by coating the exposed surface of the pad with MoS_2 (dry deposition). The electrical insulation is made by ceramic coating of the hidden surface of the pad, which is fixed on the steel keys by insulated screws. The pads are sized for an average compressive stress of 100 MPa, to exclude the yielding of the bronze, the only condition which may damage the ceramic coat. In the inboard blanket the loads are about twice as large as in the outboard and there is less space because of the cooling manifolds. Two large poloidal keys for 1 MN load are built in between the flexible supports and a smaller toroidal key is located in the midplane.

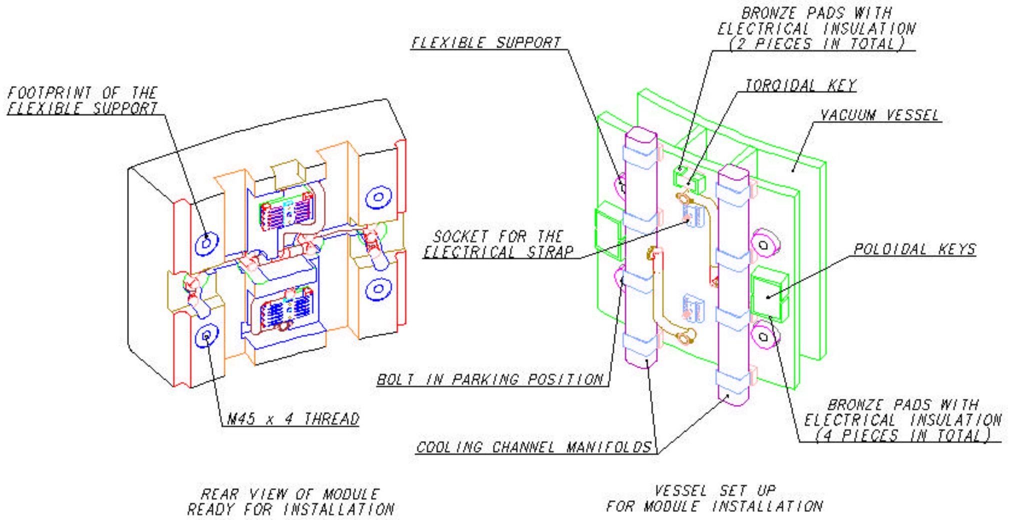

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Figure 2.3.4-2 Inboard Module Attachment with Poloidal and Toroidal Keys

In the outboard blanket the keys are built as stubs concentric with the flexible support where they extend from the VV (Figures 2.3.4-3). The flexible support is connected to the module at the bottom of the keyway, where the nuclear heating is high but still manageable. Bronze contact pads work as in the inboard blanket and are fixed by insulated screws. The pads are tapered to improve the assembly of the module, leaving a nominal clearance of 0.25 mm for each side.

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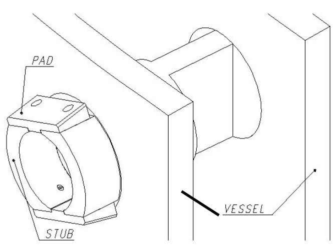

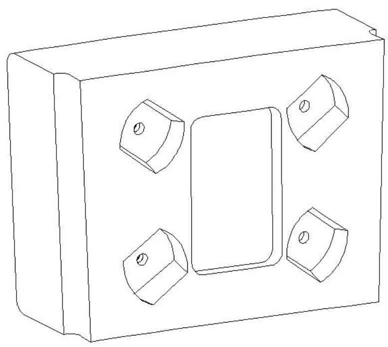

Figure 2.3.4-3 Stub Key and Back View of a Typical Outboard Module

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Only three stubs per modules are fitted with pads. The fourth remains loose in the module keyway to avoid over constraint during the assembly, and non-uniform thermal expansion. This configuration is used for modules 8 to 16. Module 17 and modules around the neutral beam penetrations have a combination of 2 stub keys with a shear key, owing to the particular space constraints.

2.3.4.3 Electrical Connection

During a slow VDE, the electrical connections discharge to the vessel up to 280 kA halo currents delivered to the modules by the plasma, and shield the metallic hydraulic connections. The strap withstands the EM forces generated by the large magnetic field, and accommodates displacements of 1 mm resulting from thermal expansion and the clearance of the keys.

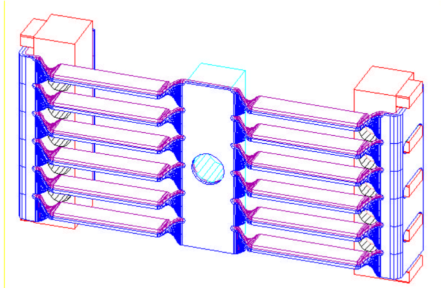

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Figure 2.3.4-4 One Strap of the Electrical Connection

The electrical connection includes two identical straps (see Figure 2.3.4-4) formed by two overlapped sheets of CuCrZr alloy, a material which provides low electrical resistance, good strength at high temperatures and after irradiation. The useful cross section is 12.8 cm^2. The sheets are bent 90° and louvered to achieve flexibility in all 3 directions. Two SS support blocks resist EM forces from the largest toroidal magnetic field component on the short sides. The blocks link the ends of the strap to the module through 5 M8 screws. The centre of the strap is connected to the vessel by a single M20 bolt. An intermediate SS socket is foreseen to spread the current on the wall and to accommodate the vessel tolerance. The socket is fixed to the wall by 8 M8 screws, or by welded studs.

2.3.4.4 Hydraulic Connection

The hydraulic connection is located in the centre of the module where the thermal displacements are the lowest. It includes flexible branches and a connector, which is kept as compact as possible to reduce the current loops between the vessel and the module. Two separate connectors to the modules are used to deliver the coolant to the first wall panels, whilst the return is from the shield block (see Figure 2.3.4-5). The branches are single pipes 48.6 mm OD routed above and below the electrical straps. The branches routed from the manifolds are remotely connected with the branches from the module at the points above and below the electrical straps.

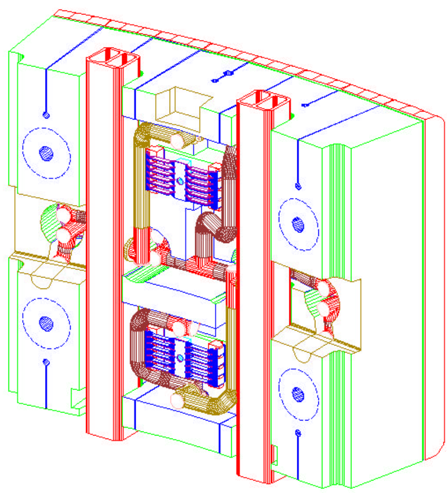

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Figure 2.3.4-5 Two Separate Hydraulic Connection for Inboard Module

2.3.5 Port Limiter System

There are two limiters in ITER and they are installed in the equatorial ports number 8 and 17. Each port limiter consists of a limiter module, a supporting and alignment system, and a port limiter shield. The location inside ports eases the maintenance, which is important for this high heat flux component. The limiter cooling system is in common with that of the divertor, because water is in contact with Cu-alloy tubes in both systems and they share the same water chemistry, as well as the same hydraulic parameters (pressure, subcooling, etc.).

2.3.5.1 Port Limiter Module

As shown in Figure 2.3.5-1 and 2, the limiter module is 2,100 high, 1,652 mm long and 458 mm thick and consists of an assembly of 36 mm thick vertical plates attached together at their rear section by welding or by bolting. The rear section of the plates forms a strong continuous backing and the ~ 1 mm gap between plates is insulated with alumina to prevent arcing. The vertical slots significantly reduce the electromagnetic loads due to plasma disruptions. The ability of the plates to independently expand also reduces the thermal stresses. Each plate is formed (Fig. 2.3.5-3) by a FW part attached to a shield part. The FW consists of beryllium armour in the form of small tiles (~ 6x6x4 mm) attached to a Cu-alloy heat sink plate internally cooled by Cu-alloy cooling tubes. The FW tubes are provided with swirl tapes to enhance the heat transfer and the margin to the critical heat flux. The shield part consists of a steel plate internally cooled by a SS serpentine tube. The FW tubes are connected to a common inlet manifold on one end and to the serpentine tube at the other end. A common outlet manifold is connected to the serpentine outlet. The FW can alternatively be fabricated separately and attached to the shield part of the plates e.g. by laser welding.

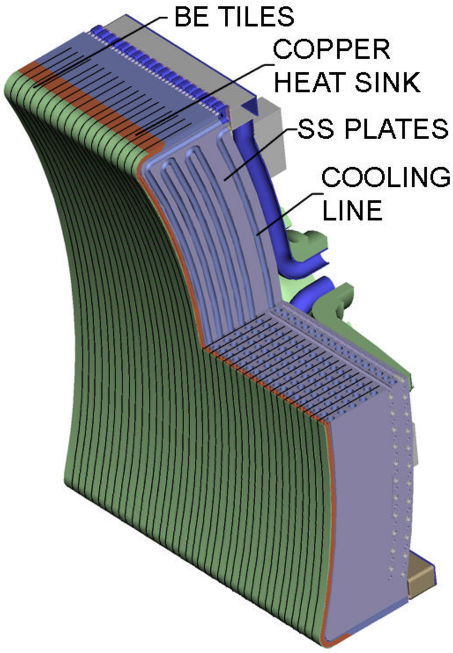

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Figure 2.3.5-1 Limiter Module Front View

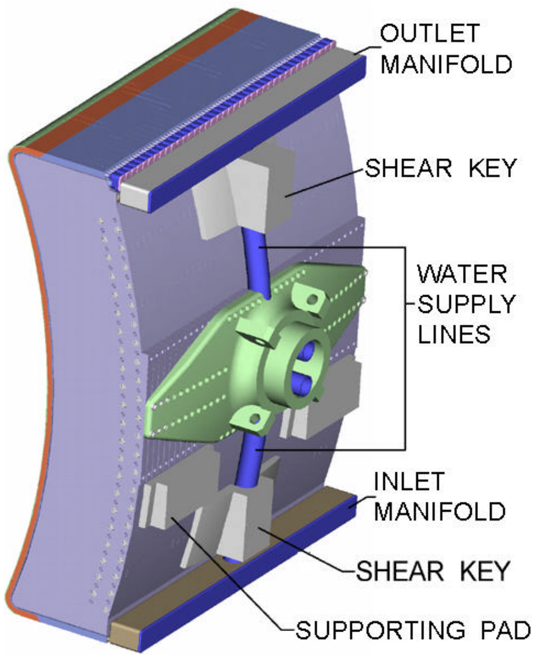

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Figure 2.3.5-2 Limiter Module Rear View

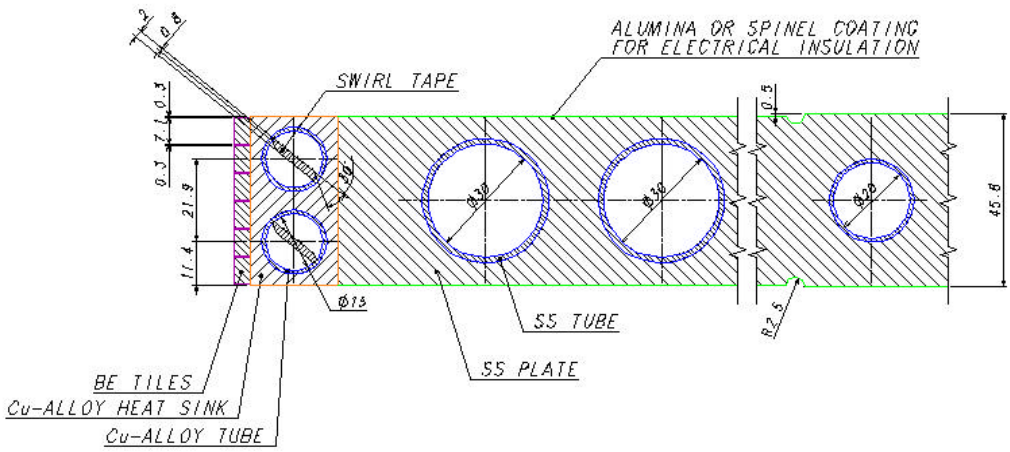

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Figure 2.3.5-3 Limiter Plate with Integrated FW

2.3.5.2 Plug Body with Supporting and Alignment System for the Limiter Module

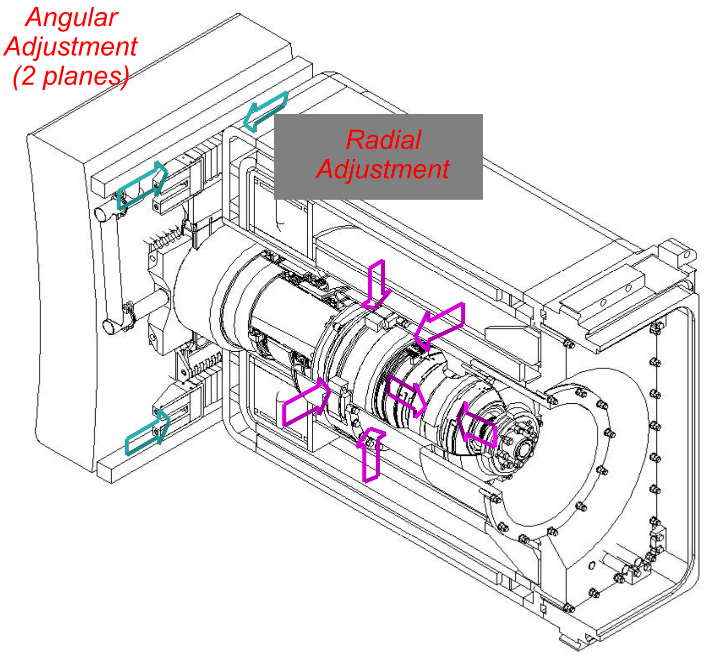

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Figure 2.3.5-4 Plug Body with Supporting and Alignment System for the Limiter Module

The supporting and alignment system, mounted on and inside the shield housing, supports a limiter module under the EM loads during disruptions, seismic loads and dead weight, providing it with all connections, including hydraulic (Fig. 2.3.5-4). This remotely-operated system adjusts the radial position of the limiter module. It also permits small rotations around poloidal and toroidal axes through a pivot point located near the centre of gravity in order to precisely align the front surface of the limiter module to the field surfaces. The adjustable driven supports are doubly sealed, thereby avoiding the need to operate under vacuum, and are powered by hydraulic stepping motors.

The shield housing is a massive welded structure, with internal water channels, providing structural support and nuclear shielding. The shield housing consists of two parts: inner frame and outer frame.

2.3.6 Cooling Manifold and Filler Shield

The coolant is delivered and returned to the blanket modules by a set of manifolds mounted inside the VV chamber aligned with the poloidal rows of modules from the upper port downwards. At the outboard the manifolds occupy the triangular voids between the modules and at the inboard they are located behind them (see Figure 2.3.6-1).

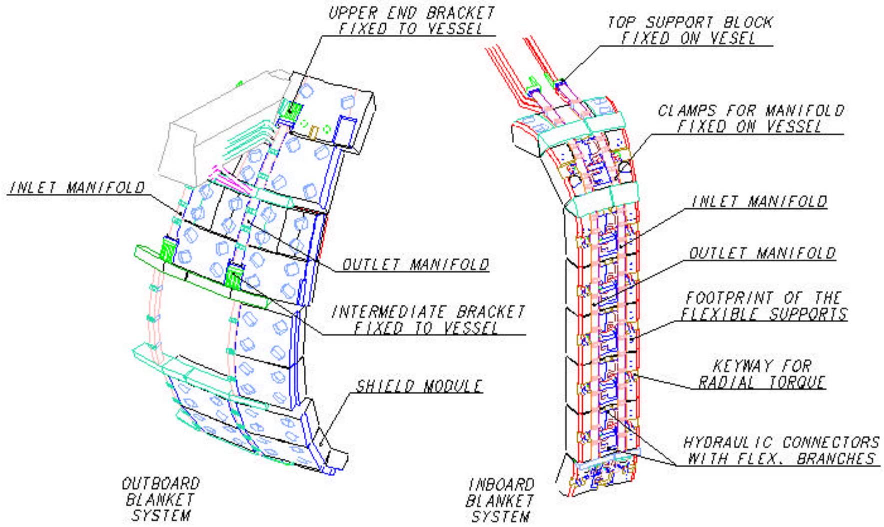

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Figure 2.3.6-1 Inboard and Outboard Blanket Manifold Overview with Modules

The manifolds include multiple channels feeding groups of 2-4 modules which are separated from the others as far as the tokamak cooling water system (TCWS) vault, where they can be isolated to improve the identification of a leak. The manifold size decreases downwards as the single channels reach the last module of their group.

All manifolds end near the upper port and are fed by eight pipes with ~ 60 mm inner diameter, arranged in two ranks on the side walls (Figure 2.3.6-2). The pipes do not pass through the flange of the VV port but are channelled upwards from the port duct through special twin chimneys built symmetrically either side of the split field joint. The chimneys end with a bulkhead occupied by the pipe feed-throughs. This pipe layout avoid interactions with the EC antennas and the diagnostic plugs.

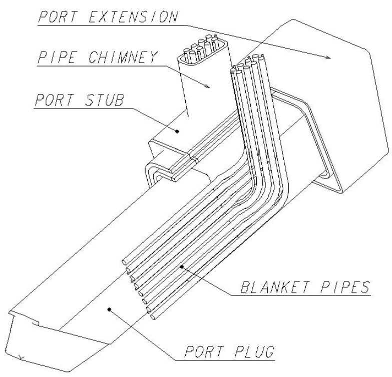

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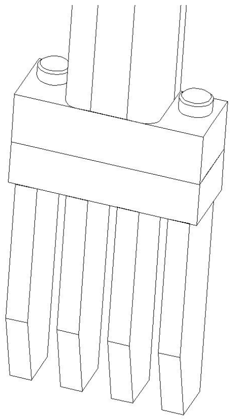
Figure 2.3.6-2 Blanket Pipes in Upper Port

The manifolds incorporate triangular filler shields in gaps between the blanket modules, between module 1 and 2 and between module 6, 7 and 8 in the inboard region, and also between modules 10 and 11, 13 and 14, 14 and 15, 16 and 17 in the outboard region. The filler shields are cooled with the water in the manifolds.

The flow cross section of the manifold is sized for a coolant speed of 6 m/s, giving a distributed pressure drop of 2.5 kPa/m. The highest manifold reliability has been pursued by adopting a 10 mm wall thickness, which is well above the size of any usual defect of the plates. The stainless steel plates and shaped profiles forming the cross-section of the manifolds are joined by triple pass filler metal welds, which can be fully non-destructively inspected. Any manifold failure will be repaired inside the VV, because the manifolds are too long for an easy replacement. Since the welds have higher failure probability than the base material, they have been concentrated on the accessible faces of the manifolds.

The longitudinal thermal expansion of the manifolds with respect to the VV is restrained to avoid the need for sliding supports, relying on low friction under vacuum. However, the transverse thermal expansion, in the toroidal and radial planes, is small and is unconstrained. The longitudinal end support of the manifolds is a strong flange bolted on a bracket welded on the VV wall (Figure 2.3.6-3). A similar restraint is used above the equatorial port, where the manifold reduces the cross section from 5 (see Figure 2.3.6-4) to 3 channels and changes the poloidal compressive force. The toroidal-radial supports handle lower forces and include an indented socket and a retention clamp fixed by thin welds. There are no lateral loads in the straight portion due to temperature differences (excluding some buckling stabilisation). The supports, spaced at 30-50 cm, are designed to withstand the radial forces directed towards the plasma and the smaller toroidal forces caused by a plasma disruption (thermal quench). They also restrain the radial movement of the manifold when it is colder than the VV (failure in the PHTS control system).



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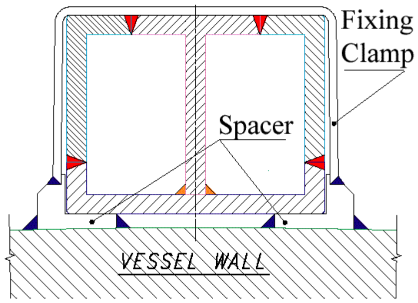

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Figure 2.3.6-3 Bolted End Support and Welded Intermediate Support for 2 Channel Inboard Manifold

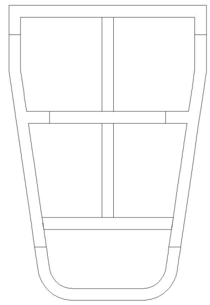

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Figure 2.3.6-4 Cross Section of Outboard Manifolds with 5 Channels

In the case of a vacuum leak, a cooling group of 2-4 modules is identified by tracer elements injected progressively in the three separate cooling circuits of the blanket. Inside each circuit the feeders of the 48 module loops are closed by ice plugs and progressively opened while watching the mass spectrometers of the VV pumps. The identification of the faulty module inside the leaking group requires a He leak test. Water is removed from the loop of the leaking group by a high speed nitrogen stream, which entrains the liquid upwards and dries the surfaces. The loop is evacuated and connected to the mass spectrometer. The VV is opened and the RH vehicle scans the leaking group of modules with a He puff. After the first wall, where the damage is most likely, the hydraulic connectors are checked because the field weld is the weakest part of the module.

The manifolds are fabricated and tested individually, then shipped to the VV production factory and installed on the 40° sectors. Here the supports are adjusted as required, tightened or welded. But the 9 outboard manifolds with filler shields over the field joint of the VV are only bolted, because they must be removed on site and then finally installed with the welds. The pipes going through the upper ports located in the middle of the 40° VV sectors are also installed and leak tested. The pipes in the 9 ports split by the field joint of the VV are installed in the laydown and assembly hall after the TF coils are in place. On one side of the VV field joints, the pipes are connected to the manifolds in the assembly hall. On the other side the pipes are welded to the manifold across the VV field joint in the pit, after the vessel field joint is done.

2.3.7 Design Features to Accommodate Breeder Blanket

The breeding blanket modules are designed to make possible the partial conversion of the outboard areas for breeding, under the same dimensional, installation, supports, coolant branch pipes and maintenance constraints as the outboard shield blanket modules. Therefore, the conversions should be done in the same way as the replacement of outboard shield blanket modules. Based on work carried out for the 1998 ITER design, the breeder candidate materials are lithium zirconate, lithium titanate and lithium silicate, used in a ceramic pebble bed. Beryllium pebbles are used for the neutron multiplication, and water for heat removal. SS 316L(N)-IG is used as structural material and Be armour is directly attached to the stainless steel without a copper heat sink. The first wall for the breeding blanket module will be integrated with the SS 316L(N)-IG box structure for breeding and cooling. Slits in the radial and poloidal directions to decrease electromagnetic loads will be useful also for the breeding blanket modules. The only additional feature required to allow the later option of breeding blanket installation is the helium purge gas lines for tritium removal. Those can be installed together with the shield blanket system during the initial construction phase. Each purge gas line will feed typically 3 outboard modules. A set of lines are mounted inside the VV chamber, and will have an entrance at the upper port. Layout of the purge gas lines is similar to the blanket cooling water manifolds, and consists of 12 mm OD purge gas manifolds (pipes) and 5 mm OD purge gas tubes. The purge gas tubes, which have one part attached to the breeding blanket module and the other attached to the purge gas manifolds, are able to connected by the remote handling system through the 30 mm front access holes.

2.3.8 Blanket Fabrication

2.3.8.1 Primary Module

The FW panel will be manufactured using solid HIP, the shield block from flat forged blocks, and the coolant channels produced simply by drilling and plugging. Powder HIP can also be used for the shield block or FW panel fabrication, and casting for the shield block.

FW

The main fabrication steps for option B with central mechanical attachment, schematically shown in Figures 2.3.8-1 and 2, are as follows:

- produce the Cu-alloy and SS plates by rolling;
- groove the Cu-alloy plates to fit the SS coolant tubes;
- machine the headers and drill the coolant channels;
- assemble and seal welds before HIPing;
- HIP the assembly at T = 1,050°C, p = ~ 150 MPa, t = ~ 2 h;
- attach the Be armour by HIP (e.g. with Ti interlayer at 800°C, 2 h, 120 MPa, or with Cu interlayer at 620°C, 140 MPa for 2 h);
- machine the poloidal slots;
- weld the support legs to the FW panels before drilling the coolant channels;
- weld headers and plug the channels before final surface machining.

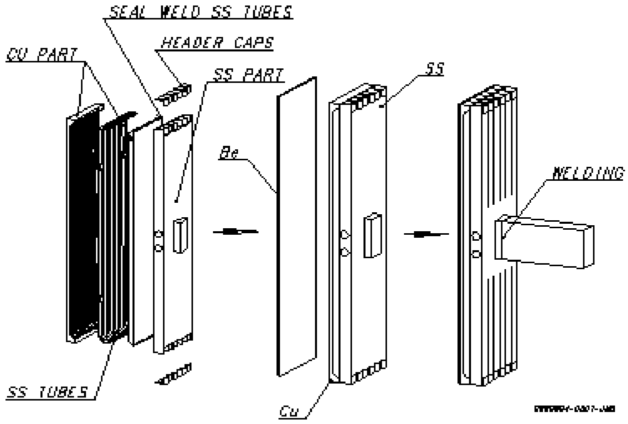

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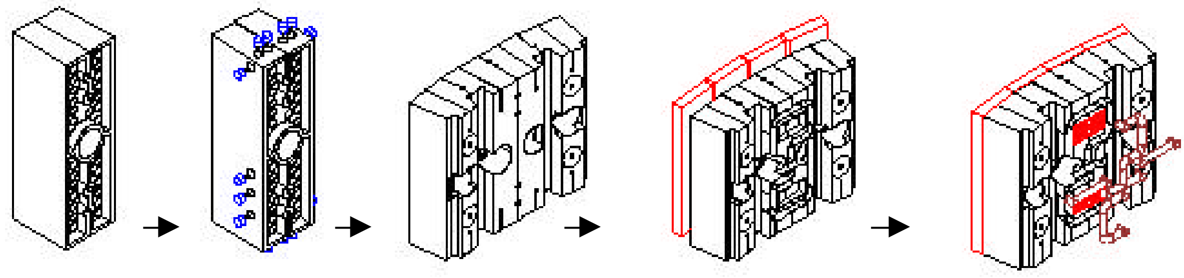
Figure 2.3.8-1 FW Panel Fabrication Method for the Option with Central Mechanical Attachment

Shield Block

The procedure of the shield block fabrication is schematically shown in Figure 2.3.8-3. The shield blocks are mainly fabricated by simple drilling and milling, and welding. The feasibility of obtaining drilled forged shield blocks within the required tolerances, and with welding closure plates, has been demonstrated in a module prototype. The manufacturing steps will be:

1) producing four separate forged blocks,
2) drilling radial coolant holes and the initial machining for headers,
3) drilling and machining of slits and FW holes,
4) plugging of coolant holes after drilling,
5) machining and the fabrication of coolant guides with screws,
6) attachment of coolant guides by welding after fixing by screws temporarily,
7) milling for the keys and drilling for flexible holes at the rear side as preliminary machining,
8) attaching lids to the headers by TIG welds,
9) EB welds to assemble separate forged blocks,
10) final machining of grooves at the rear side including all slits between the four shield blocks,
11) assembling shield blocks and FWs into unit,
12) attaching FWs to the shield block,
13) welding the hydraulic connections to the respective inlet/outlet headers and attaching electrical straps by bolts.



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1) Drilling & machining — 2)Drilling, plugging & machining — 3) Welding, milling & machining — 4) Final machining & assembly of first walls — 5) Welding & setting all attachments

Figure 2.3.8-3 Fabrication Procedures Based on Using Solid Forged Blocks

Powder HIP is also a possible fabrication method for the shield block whose feasibility has been demonstrated by manufacturing a prototype as part of ITER R&D. The fabrication error was still not quite as small as required, but it is expected to be reduced sufficiently in future manufacturing R&D.

2.3.8.2 <u>Limiter Module</u>

Listed below are the main manufacturing steps for the limiter module:

- machine to shape the Cu-alloy heat sink half-plates and machine half-channels;
- attach transition tubes to FW Cu-alloy tubes and join them by brazing or HIP;
- prepare the coolant SS serpentine tube and install swirl tapes;
- machine to shape the shield SS half-plates and the serpentine half-channel;
- bend the heat sink to shape;
- assemble the shield half-plates with the serpentine tube inside, together with the FW Cu-alloy half-plates and the FW cooling tubes; seal weld the tubes to the shield plates and provide canning; HIP the assembled parts at 1,050^{o}C,150 MPa, 2 h;
- machine the component to shape and provide electrical insulation coating;
- attach FW Be castellated plates to the heat sink with fast amorphous brazing (join the castellated side) after jigging;
- machine the continuous part of the Be plates to form tiles;
- make the FW collectors and provide connections to the shield part;
- EB weld the plates using a suitable jig, or bolt plates in sequence to form the module;
- weld coolant manifolds and supply pipes (TIG) and attach supporting fixtures.

If the FW were separately manufactured and attached by laser welding to the shield part of the plates (Fig. 2.3.5-4), the HIP parameters of the SS plates can be changed to 1,100^{o}C, 150 MPa, 2 h.

2.3.9 Blanket Assembly

The basic blanket assembly process is as follows (see also 2.10).

1. Pre-assembly stage

(i) 27 inboard and 27 outboard cooling manifolds (out of a total of 36) are mounted on the plasma-facing wall of VV sectors. The manifolds are checked for leakage.

(ii) The majority of diagnostics and helium purge lines are mounted on the plasma-facing wall of VV sectors. All the interior cabling and pipelines are routed to the vessel exterior and checked for continuity and leakage.

2. The vacuum vessel torus is aligned to the machine magnetic centreline, all global adjustments made and the vessel supports fixed. The measurement data will be utilised to provide a best-fit determination of the mounting profile information necessary to customise the module attachments.
3. Customisation of the module attachments.
4. Toroidal loops of diagnostics on the plasma-facing wall of the VV are installed and connected to the interior cabling.
5. The remaining 9 inboard and 9 outboard manifolds that cross the VV field joints are mounted on the plasma-facing wall of the VV. The manifolds are checked for leakage.
6. Mounting of module attachments.
7. Pre-mounting of blanket modules by temporary supporting bolts.
8. Fixing of blanket shielding modules. The module flexible support bolts are pretensioned using heat.
9. Welding of branch pipes, and fixing of electrical connectors.
10. Blanket commissioning test: global leak and pressure tests, global flow and pressure drop tests, global/local thermal transient response tests, and electrical resistance test.

2.3.10 Loads and Analysis

2.3.10.1 EM Loads

The plasma disruption database shows that high plasma current density leads to a fast current quench, resulting in a short quench time of 27 ms. By the use of deep slits, electromagnetic (EM) loads on the modules have been reduced to the values shown in Table 2.3.10-1. The FW-normal halo current density is assumed to be 0.18 MA/m^2 under the worst-case halo current event, as specified by I_{halo}/I_{plasma} multiplied by the toroidal peaking factor, TPF = 0.58. The design loads on modules are shown in Table 2.3.10-1.

Table 2.3.10-1 Maximum EM Loads on the Inboard Module (Option B)

		Shield block	FW
1) Centred disruption			
Torque M_r due to I_{rad}	MNm	-1.06	-0.008
Torque M_p due to I_{pol}	MNm	1.12	0.006
2) Fast VDE			
Torque M_r due to I_{rad}	MNm	-0.97	-0.008
Torque M_p due to I_{pol}	MNm	1.47*	0.006
3) Slow VDE**			
I_{halo} / module (FW)	MA	0.28	0.07
EM force F_r on module (FW)	MN	0.32	0.08
EM force F_p on module (FW)	MN	1.1	0.15
EM force F_{tor} on module (FW)	MN	0.21	0.03
Torque M_{tor} on module (FW)	MNm	0.36	0.027

* Maximum torque is produced on module 1 under the downward fast VDE. This torque can be reduced to ~1 MNm by increasing the deep slits as a special module.
** The eddy currents during a slow VDE are very small, thus no radial and poloidal torque are considered

2.3.10.2 Primary Module Structural Analysis

2.3.10.2.1 FW and Shield Block

FW

The FW panel and its attachment system have been sized to withstand the above electromagnetic loads caused by plasma disruption. The slotting of the FW results in a reduction of about 10 times in eddy currents and consequently in the stresses caused by the induced electromagnetic loads. The results of thermo-mechanical analyses performed to verify the FW during normal operation are summarised in Table 2.3.10-2, showing that thermal stresses in the FW panel are well below allowable values. These thermal stresses are somewhat lower in the FW part without penetrations.

Table 2.3.10-2 Main Thermo-mechanical Analysis Results for FW

Location	T range [°C]	Max. σ_{VM} [MPa]	Allowable [MPa]
Be tile	200 - 253	~ 171 (Cu interface)	-
Cu heat sink	165 - 218	~ 152 (Be interface)	294
SS tube	143 - 198	179 (wetted side)	444
SS back-plate	119 - 211	268 (at channels)	465

In the present design of the module, there is only room for a leg of 175 mm. With such a short leg, a scheme with cantilever support (Figure 2.3.10-1) is preferred, being the most reliable. A cantilever supported welded leg should have a big enough diameter to minimise the thickness of the weld. Space allows room for a leg of diameter 168 mm. The required thickness of the weld joint is 14 mm, and the bending stress in the weld joint is within the allowable (207 MPa).

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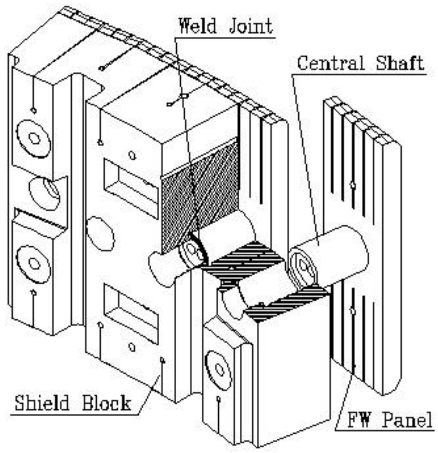

Figure 2.3.10-1
Cantilever-supported Welded Leg

<u>Shield Block</u>

Structural analyses of a 1/8 model of the inboard and outboard blanket modules have been conducted using ANSYS. Figure 2.3.10-2 and 3 show the temperature and stress distributions. Table 2.3.11-3 summarises results of the stress evaluation for the inboard module. The maximum temperatures, 287°C for the inboard module and 338°C for the outboard module, are below the allowable temperature of 427°C for non-active thermal creep. The calculated stress intensity for thermal and pressure loads on both modules are also within the allowable stress ranges of $3S_m$ and thermal fatigue allowable values.

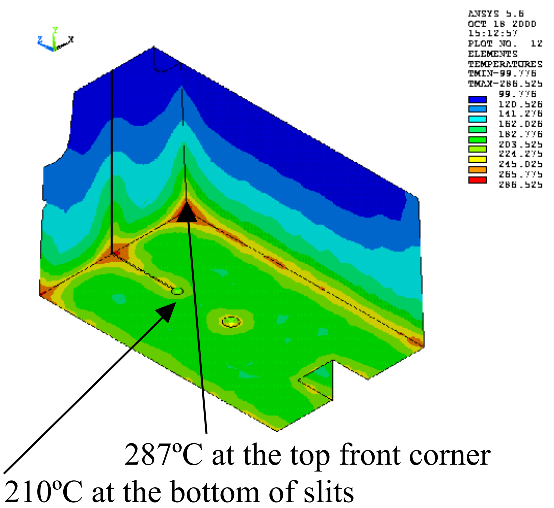

Figure 2.3.10-2 Temperature Distributions in the Inboard Shield Block

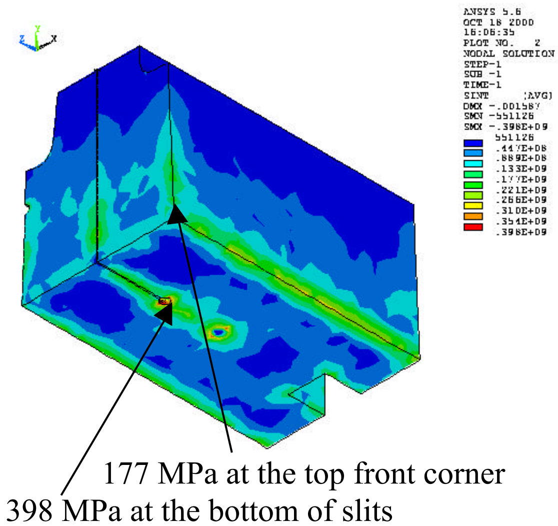

Figure 2.3.10-3 Stress Intensity in the Inboard Shield Block for Thermal and Pressure Loads

Table 2.3.10-3 Results of the Stress Evaluation for the Inboard Shield Block

		Calculated stresses (MPa)		Allowable stresses (MPa)		Accumulated fatigue damage factors	
Evaluation points	Temp (°C)	(1) Press.	(2) Press.+ Thermal	(1) $1.5S_m$	(2) $3S_m$	n/N_d	Allowable value
Bottom of slits	210	32	398 1)	202	405	0.06	1.0
Top front corner	287	<10	177 1)	184	369	<0.01	1.0

Note: 1) Values including thermal peak stresses

2.3.10.2.2 Module Attachment

Flexible Support

The loads on the flexible supports are shown in Table 2.3.10-4. The highest stress in the cartridge appears at the ends of the flexible spokes and is kept below the yield of the Ti-6Al-4V alloy at 200°C to avoid any permanent deformation affecting the alignment of the supports; thus the disassembly and the reassembly of the module is reasonable.

Table 2.3.10-4 Stress Components versus Allowables in the Flexible Cartridge

Load	**Load type**	**Stress[MPa]**	**Stress type**
Axial force 500 kN/-600 kN	Primary	+166/-200	membrane
Side displacement 1 mm	Secondary	±153	bending
Flange rotation ± 3 mrad	Secondary	±129	membrane
Combined Stress	**Stress[MPa]**	**Allowable [MPa]**	
Primary membrane	200	S_m	324
Primary (membrane + bending)	200	1.5 S_m	486
Primary + secondary (membrane + bending)	482	3 S_m	972

A high yield stress is important because it triggers buckling. With a combined stress of 500 MPa a margin of 3 on buckling is achieved. This was confirmed on prototypes (see Figure 2.3.10-4), with and without lateral displacements.

Fig. 2.3.10-4 Titanium Flexible Cartridge after Compression Buckling at 1.6 MN

The fatigue strength is satisfactory and was validated experimentally up to 10,000 cycles.

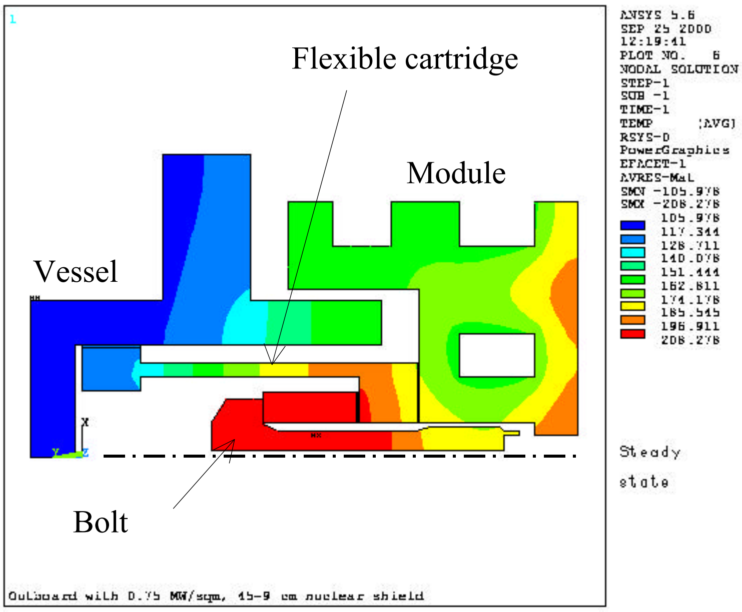

Fig. 2.3.10-5 Axisymmetric Temperature Map in the Outboard Support with Stub Key

The temperature of the bolt in the outboard blanket is higher, because the support is partially recessed in the back of the module together with the stub key in the outboard. The analyses show a maximum temperature of 208°C (see Figure 2.3.10-5) in the stem of the bolt which requires a bolt preload of 673 kN for the outboard module instead of 650 kN for the inboard.

<u>Key</u>

The key is cooled through conduction to the vessel. The temperature distribution has been verified with a 3D model to exclude hot spots and large thermal stress in the interface with the VV, and to limit the average shear stress to 35 MPa. In the front of the key at steady state the temperature reaches 256°C, while the inner VV wall reaches 125°C. The thermal stress inside the key is about 50 MPa, but it reaches a peak in the corner of 200 MPa.

Considering the loads from a centred disruption, the fast TF coil discharge and nuclear heating, the Tresca stress (see Figure 2.3.10-6) between the key and the VV reaches 300 MPa over a wide surface of the wall, with a peak of 390 MPa in the corner. These values are within the stress allowable $3S_m$ of the VV, but they require a smooth fillet between the key and the vessel and a defect-free joint. The compressive stress in the contact pad has also been calculated, to verify that local peaks do not reach yielding of the bronze which may break the ceramic electrical insulation. The maximum peak stress of 156 MPa (average 79 MPa) is acceptable, since the yield stress of aluminium bronze at 250°C is above 200 MPa.

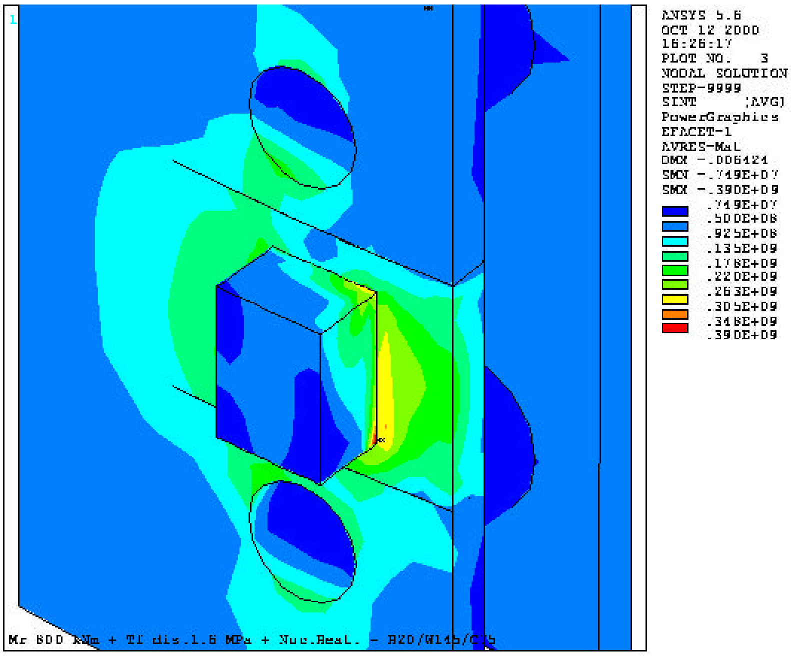

Figure 2.3.10-6 Total Tresca Stress Distribution during a Centred Disruption in the Inboard Key

<u>Electrical Connection</u>

The electrical strap has been analysed under EM forces and thermal displacement (Figure 2.3.10-7). The stress is mainly due to radial displacement and peaks at 315 MPa, below the allowable of 438 MPa for the high strength copper alloy. The experimental validation on a prototype mounted inside a solenoid with 7.5 T magnetic field has been successful completed.

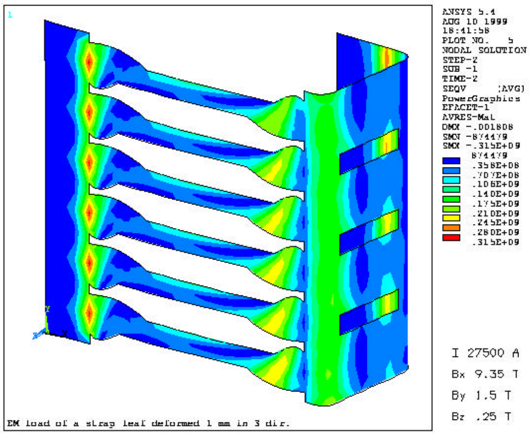

Fig. 2.3.10-7 Total von Mises Stress [Pa] in a Typical Leaf of the Electrical Strap

<u>Hydraulic Connection</u>

Structural analysis has been performed for inlet and outlet branch pipes by using a beam model. Four loading conditions have been considered: thermal load, EM load under fast VDE, coolant pressure, and relative displacement between cooling manifolds. The maximum stresses in the inlet pipe are 62 MPa at the elbow portion under the EM load, and 146 MPa at the edge portion of the FW connection pipe under the thermal loads. Both mechanical and thermal stresses are within allowables of $1.5S_m$ (220 MPa) and $3S_m$ (441 MPa), respectively. The maximum stresses in the outlet pipe at 150°C are 71 MPa at the elbow portion due to EM load, and 36 MPa at the other elbow portion due to thermal load. Both stresses are also within allowables. The branch pipe layout and the maximum stress position under the thermal load are shown in Figure 2.3.10-8.

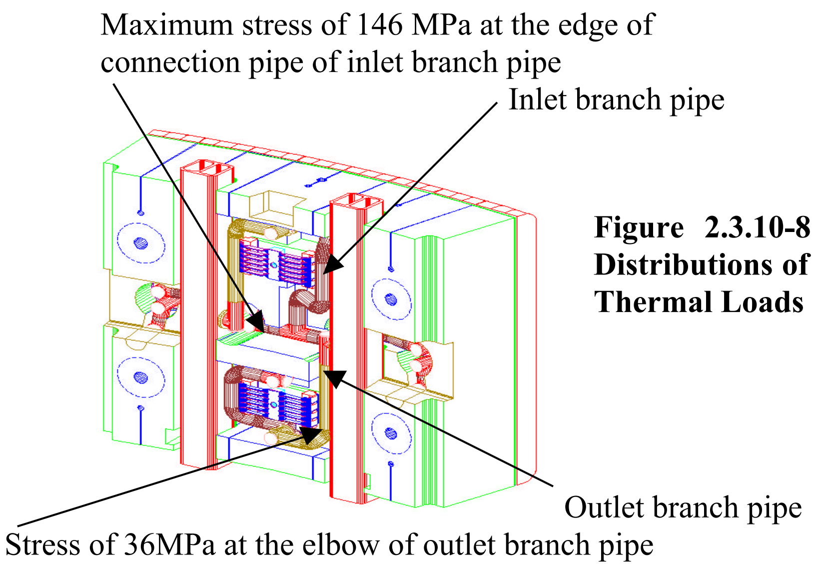

Figure 2.3.10-8 Stress Intensity Distributions of the Inlet Pipe under Thermal Loads

Figure 2.3.10-8 Stress Intensity Distributions of the Inlet Pipe under Thermal Loads

2.3.10.3 Port Limiter Structural Analysis

Port Limiter

In normal operation, only thermal stresses need to be analysed, since primary stresses are low. A transient elasto-plastic thermo-mechanical analysis was carried out for the 1998 ITER design to verify the ability of the limiter to withstand its thermal loads. This analysis is still valid because load conditions and geometry have not been changed significantly. A kinematic hardening model was used with elastic/full plastic material stress strain curves. During start-up the maximum temperature in the Be and Cu-alloy is ~740 and 450°C respectively, for a 4 mm Be armour. The maximum thermal stresses in the limiter structure are below 120 MPa, and the limiter works in low stress condition.

At the Be/Cu-alloy interface, however, plastic conditions are reached during plasma start-up and shutdown. Nevertheless, the maximum plastic strain values present in the DS Cu (Figure 2.3.10-9) are not excessive (max. total strain amplitude in Cu 0.56%), so that ratcheting is not expected. Both in Be and Cu the strain range decreases slightly every cycle. For the full thermal loads (8 MW/m^2), the allowable number of cycles of the FW exceeds 13,000 cycles, which satisfies the limiter design requirements.

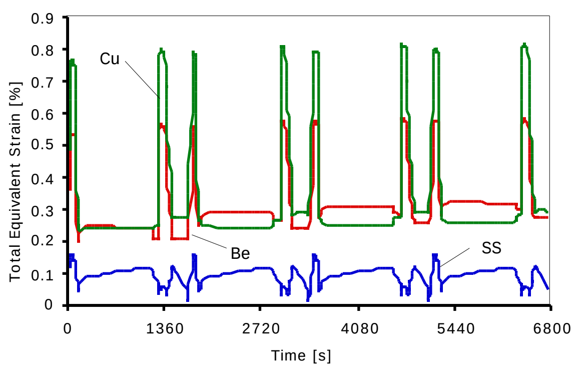

Figure 2.3.10-9 Time Evolution of the Total Equivalent Strain in Be, Cu and SS of Limiter

Port Limiter Supporting and Alignment System

Kinematics performance and the main structural components of the port limiter supporting and alignment system have been analysed. The chosen design of the support provides an accuracy of adjustment higher than required. Regarding EM loads, seismic loads and differential temperatures, maximum stresses of the sub-components in the system are within the allowables.

2.3.10.4 Cooling Manifold Structural Analysis

The highest load for the blanket cooling manifolds is the constrained thermal expansion with respect to the VV, then the coolant pressure, and finally the EM forces. The vacuum vessel and the blanket have the same inlet coolant temperature. Ideally, the inlet manifold has the same temperature as the vessel and no thermal stress, the outlet manifold is 50°C warmer with 157 MPa compressive stress. In reality the control systems of the cooling circuits have some tolerances, therefore the inlet manifolds can be 10°C cooler than the vessel and the outlet manifold can be 55°C warmer. If control fails, the blanket coolant enters at room temperature and quenches the inlet manifold to 70°C below the temperature of the vessel. The outlet manifold is shielded by the heat capacity of the blanket and becomes 30°C colder than the vessel, if the cooling circuit is stopped manually some minutes after the failure, otherwise it reaches also 70°C undercooling. The range of temperature differences is thus 75°C in the inlet and 85°C in the outlet manifolds.

For the assumptions above, the maximum value of the Tresca equivalent stress reaches 312 MPa at the inlet and 343 MPa at the outlet manifolds (Table 2.3.10-5), below the allowable $3S_m$ value of 441 MPa of the SS. If the coolant control is absent indefinitely the range of temperature difference in the outlet manifold becomes 125°C, with an associated Tresca equivalent stress range of 467 MPa (Table 2.3.10-5). The alternate yielding hardens the steel, which can nevertheless still withstand 20,000 cycles before failure.

Table 2.3.10-5 Inboard Manifold Stress [MPa]

Load source, stress component	Allowable stress	Inlet	Outlet
Coolant pressure 3 MPa, membrane (bending)		15 (72)	
EM, membrane shear (bending)		25 (19)	
Thermal range, longitudinal stress		235	267 [392]
All, Primary membrane Tresca	147	52	35
All, Primary membrane +bending Tresca	221	100	93
All, Total stress range Tresca	441	312	343 [467]

The constrained thermal expansion produces mainly longitudinal forces, but in the inboard top curved region the compressive force of the manifolds produces a radial component of 450 kN/m poloidally per 50°C difference. Supports are here spaced 30 cm apart instead of 50 cm as in the straight part.

The moderate effect of the coolant pressure on the manifold cross section is shown by the stress distribution in Figure 2.3.3-10. The finite elements of the mesh, corresponding to the welds between the C, L and T profiles forming the wall, are marked with a different colour.

The resulting stresses in these positions shows no bending and the welds experience only membrane stress as in circular pipes.

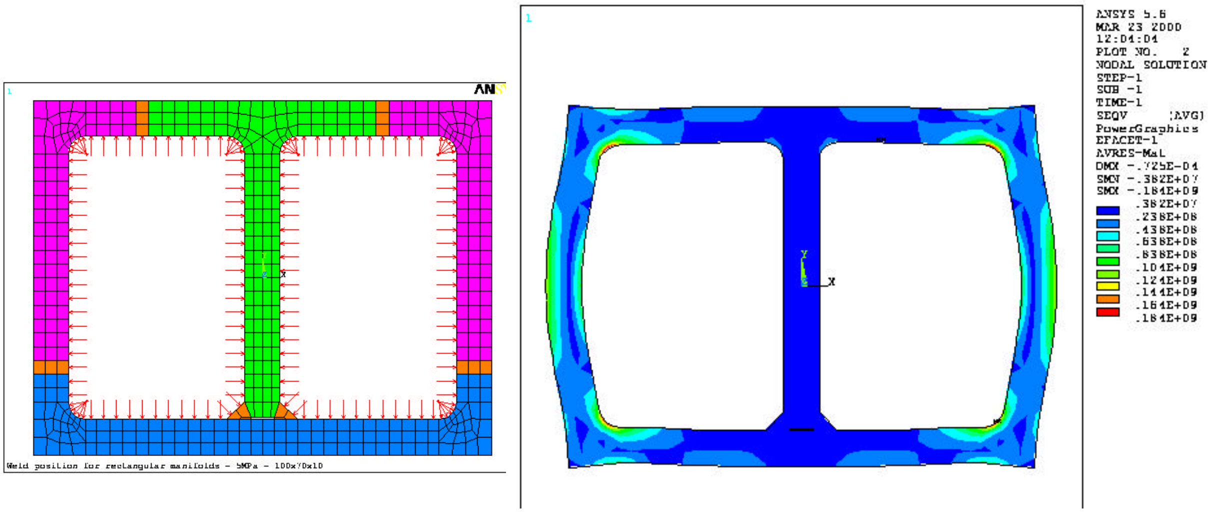

Figure 2.3.10-10 Stress in the Manifold with 5 MPa Pressure during the Baking.
The profiles are welded where there is no bending stress.

In the outboard the manifold is offset to bypass the equatorial port. Above it the channels reduce from 5 to 3 and the manifold has an intermediate restraint to balance the change of the longitudinal force. The support system of the outboard manifold has been assessed by 3D finite element analyses. Poloidal currents are induced in the manifolds by changes in the toroidal flux. They cross the toroidal field and generate radial pushing or pulling forces. At the inboard during a thermal quench of the plasma the pulling force reaches 120 kN/m length of manifold and is reacted by welded sheet clamps (Figure 2.3.6-3). During a TF coil fast discharge the manifolds are pushed against the support sockets on the vessel with a force of 40 kN/m length of manifold. The crossing of the currents with the small radial field generates minor toroidal forces (7.5 kN/m length of manifold in the inboard during the thermal quench) reacted by the indentation of the sockets. All EM loads give stress contributions far smaller than the thermal stress.

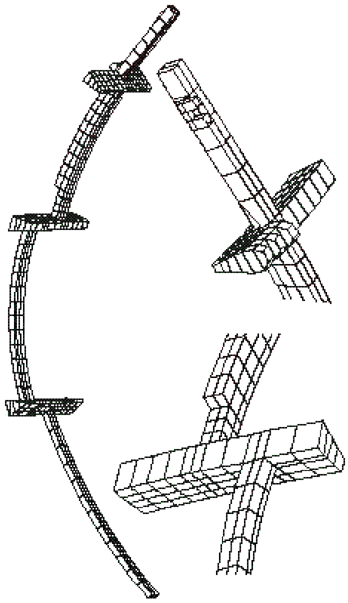

Figure 2.3.10-11 Finite Element Model of the Outboard Manifolds with the Filler Shields

2.3.10.5 <u>Thermal and Hydraulic Analysis</u>

Adequate coolant distribution to each blanket module is important to avoid any overheating in the modules resulting in excessive thermal stress. Coolant hydraulic analysis has been performed to assure that the required coolant can be distributed to each blanket module. The pressure drop has been calculated module 4 under option B. The obtained pressure drop is 0.1 MPa for the first wall and 0.3 MPa for the shield block in the case of a coolant mass flow rate of 7.48 kg/s.

The orifice diameter necessary for correct flow distribution of each blanket module has been calculated. Figure 2.3.10-12 shows the calculation model, with coolant distribution to three blanket modules (4, 5 and 6). The configuration of the blanket modules 5 and 6 are assumed to be the same as that of the blanket module 4. The required diameter for each module is shown in Table 2.3.10-6.

Table 2.3.10-6 Required Orifice Diameter for the Blanket Modules 4, 5 and 6

Module No.	Required Coolant (kg/s)	Orifice Diameter (mm)
4	7.71	42 (no orifice)
5	6.37	24.5
6	5.35	21

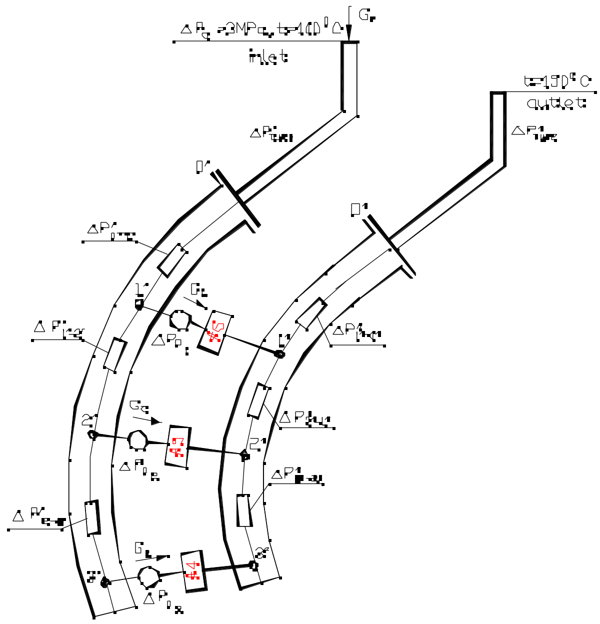

Figure 2.3.10-12 Example of Calculation Model for Coolant Distribution

2.3.11 Blanket Overall Assessment

A modular blanket design with a system of mechanical attachments to the vacuum vessel has been developed. A separate cooling manifold concept has been designed which is located on the vessel inner surface. The installation of a breeding blanket in the outboard area at a later date has not been precluded.

The blanket modules use a separable FW concept with a faceted geometry. Detailed design has been developed for a shield block with radial flow cooling and FW panel attachment structures for two options: (i) central beam and (ii) bolting with shear ribs. These concepts should result in significant cost and activated waste reductions compared to the 1998 ITER design. Fabrication methods for further cost reduction are covered by the on-going blanket R&D program which is investigating (a) the use of CuCrZr heat sink instead of DS Cu, (b) Be joining to Cu-alloy by brazing instead of solid HIPing, (c) powder-HIPed FW panels.

The EM, mechanical, and hydraulic analyses have been done for normal and off-normal events. Stress levels in the shield blanket modules and limiter are below allowable. Fundamental problems completing the further detailed design are not anticipated.