2.13 Materials Assessment

2.13.1 Introduction

The material choice for ITER has been orientated toward industrially available materials and well-established manufacturing technologies, while taking account of physical and mechanical properties, maintainability, reliability and safety requirements. The materials chosen not only need to have good basic material properties, but also to maintain these properties sufficiently through component manufacture. Materials proposed for ITER can be divided into two groups:

- standard materials with well-established manufacturing technologies, which do not require any modifications (e.g. AISI 304L, 316L, 316LN, Inconel 718, etc.).
- standard materials, which require some modifications, such as more stringent limits on the alloying elements, some changes of the processing technology, etc.

To support the materials selection, ITER R&D has been carried out with the following objectives:

- characterisation of selected materials, study of the critical issues, and provision of the required database for the design;
- justification of the materials selection.

The following sections briefly describe the selection and assessment of materials for vessel and in-vessel components, diagnostic components, and magnets, indicating the reasons for the choice and the advantages of these particular materials over the alternatives.

2.13.2 Materials for Vessel and In-Vessel Components

2.13.2.1 Plasma-Facing Materials

2.13.2.1.1 Rationale for the Selection of Armour Materials

The choice of plasma-facing materials for different components is determined mainly by plasma compatibility and erosion lifetime (ion erosion and thermal erosion during disruption and transient events).

Beryllium has been chosen as the armour material for ~ 80% of the total surface exposed to the plasma (primary first wall and port limiter). The main reasons for the selection of Be are the low effect on plasma contamination, low radiative power losses, good oxygen gettering ability, absence of chemical sputtering (in comparison with carbon), low bulk tritium inventory, and the possibility of in-situ (or in hot-cell) repair of damaged surfaces using plasma spray.

W has been selected as the material for the divertor baffle areas where there exists a high concentration of neutral particles. In this area the key issue is erosion lifetime, and W has a lower erosion rate, due to its low sputtering yield and its higher sputtering threshold energy, as compared to those of beryllium and carbon. Another advantage of W is its low tritium retention. The plasma compatibility of W is an issue, because a small amount of W in the confined plasma region could lead to a very large radiation power loss from the plasma.

In areas exposed to high thermal fluxes during normal operation, and large energy excursions during plasma disruptions (lower part of the divertor vertical target), carbon fibre composite is selected. Use of CFC in this area can provide the needed erosion lifetime, whereas the use of W and Be in this area is limited due to the creation and loss of a melt layer during disruption events. However, the use of CFC as armour has to be restricted, because of chemical erosion and tritium retention, especially in co-deposited layers.

All plasma-facing components will create dust which can interfere with plasma operation, and which will generally end up in the divertor region (see 2.4).

2.13.2.1.2 Beryllium

Commercially available beryllium grades from the USA and from the RF have been evaluated as candidate materials. The selection of the optimum grade is driven by those properties which are very sensitive to the impurity levels - grain size, method of production, thermomechanical treatment - which usually differ for the different Be grades.

Among various industrial beryllium grades, S-65C VHP (vacuum hot pressed) grade (manufacturer Brush Wellman Inc., USA) has been selected as a reference grade. The main reasons were:

- lowest BeO and metallic impurity content among the other structural grades;
- high elevated temperature ductility;
- excellent low cycling thermal fatigue performance and thermal shock resistance;

The similar DShG-200 (manufactured by the RF) was selected as a back-up.

The base properties of these grades are well documented. The key issue for application of Be as armour is its behaviour after neutron irradiation and thermal transient events. Neutron irradiation typically leads to degradation of the Be properties. For first wall irradiation conditions (temperature range 240-480°C, maximum damage level ~ 1.7 dpa, He concentration 1,700 appm) the main concern is the embrittlement of Be at low irradiation temperatures (less than 300°C). It is expected that the ductility of Be at this temperature and conditions will be at the level ≤ ~ 1%. The embrittlement of Be at low temperature could lead to brittle destruction of the tiles and affect the thermal erosion of Be during transient events. The study of thermal erosion and damage of the neutron irradiated Be S-65C VHP confirms that this grade has better thermal shock resistance. The use of Be tiles, without critical defects that may result in crack initiation, could partly solve this problem.

Another possible material is plasma-sprayed beryllium. Be plasma spray (PS) has the potential to be used for the manufacturing of the first wall modules and for in-situ or hot cell repair of the damaged armour, which would save maintenance time and reduce radioactive waste. By comparison with S-65C, plasma-sprayed Be has lower thermal shock resistance, lower mechanical properties, and higher porosity.

2.13.2.1.3 Tungsten

A preliminary selection of W grades has been made taking into account the near term availability, cost, technological features, and behaviour under thermal fatigue and disruption conditions. Several W grades from different suppliers (pure sintered W, cast W alloys, W-1%La_2O_3, chemical vapour deposited (CVD) W etc.) were selected for investigation. The recent results from the different tests have demonstrated that among the studied materials, pure W has lowest thermal erosion (in comparison with W-1%La_2O_3) and adequate thermal fatigue resistance (also at heat fluxes much higher than required). The key issue is the proper design of the W armour (size of tiles, grain orientation): the recommended grain orientation is parallel to the direction of the heat flow, in which case possible delamination is not so critical.

Pure sintered W is recommended as the reference material for divertor components. This material is available in a wide range from different suppliers. The W properties database is also well established.

Typically for bcc metals, irradiation leads to an increase in the ductile-to-brittle transition temperature (DBTT). Based on the available data, all W grades have the problem of brittleness at the expected fluence of 0.1-0.5 dpa in the region near the heat sink, for an irradiation temperature less than ~ 500°C. To avoid damage and possible delamination of this brittle material it is recommended to avoid the use of W in geometries with crack initiators. One way to solve this problem is to orient the W armoured design toward concepts that reduce thermal stresses in the armour and in the joints i.e. brush, rod or lamella structures.

2.13.2.1.4 Carbon Fibre Composites

Carbon fibre composites have been selected as the reference material in certain parts of the divertor due to the absence of melting, their high thermal shock and thermal fatigue resistance (low crack propagation) and their high thermal conductivity in comparison with conventional graphites. The preferable materials are CFCs with:

- high thermal conductivity: the higher the thermal conductivity, the larger the sacrificial thickness and hence the higher the erosion lifetime;
- 3D fibre structure is preferable because it results in more isotropic properties and higher thermal shock resistance in comparison with 1D and 2D materials;
- high density and porosity: higher density and lower porosity are preferable to minimise gas absorption and outgassing.

Among the available CFC grades, the 3D CFCs Sepcarb®NB 31 (produced by Société Européene de Propulsion (SEP), France) and NIC-01 (produced by Nisseki-Corporation, Japan) have been selected as the references. The back-up option is CFC CX 2002U (2D felt-type material, produced by Toyo Tanso Co., Japan). CFC with 10%Si doping (Sepcarb-inox NS31) has also been developed by SEP. The goal of the Si doping is to reduce the chemical erosion. This material is therefore still under study as a possible candidate material.

The properties of CFCs strongly depend on the fibre type, the architecture (size of cells, volume of fibres), the method of production, and the thermal treatment. Being newly developed materials, the existing database on their properties is usually more scattered than that of metals. For the reference materials, data are available only up to ~ 800-1,500°C.

For a damage fluence ~ 0.1-0.3 dpa and irradiation temperature 150-1,200°C the changes in properties are not expected to be crucial except for the thermal conductivity. CFCs with high initial thermal conductivity retain high conductivity after irradiation, but at low irradiation temperatures (less than 300°C) the thermal conductivity could be ~ 3-5 times lower than the unirradiated CFC. The performances of mock-ups using the Sepcarb®NB 31 and NIC-01 armour tiles have been studied including irradiation effects. Generally, the performance was as expected (higher temperature due to the loss of thermal conductivity). The decrease in the thermal conductivity due to neutron irradiation leads to an increase in thermal erosion under disruption conditions and this may have to be taken into account during the erosion lifetime assessment.

2.13.2.2 Structural Materials

2.13.2.2.1 Stainless Steels

SS 316L(N)-IG Type

Austenitic stainless steel (SS) of 316 type is the main structural materials for the ITER vacuum vessel and for in-vessel components (shielding blanket, divertor cassette body) because this steel is qualified in many national design codes, has adequate mechanical properties, good resistance to corrosion, weldability, forging, and casting potential, is industrially available in different forms, and can be manufactured by well-established techniques. Among the 316 steel family, 316L(N) type SS (based on 316L(N)-SPH SS developed for the European Fast Breeder Reactor program) has been selected for ITER application.

The main reasons for the selection of this grade are the following.

- The proposed grade has an optimal combination of the main alloying elements (C, N, Ni, Cr, Mn and Mo) with a tight specification of their allowable composition range. The narrow specification provides an optimal microstructure (austenitic phase with a δ-ferrite content ≤ 1%) and good control of the heat-to-heat variation of mechanical properties.
- The tight control of the carbon and nitrogen provides a satisfactory resistance to stress corrosion cracking of base metal and welds, and an adequate level of material strength.
- The mechanical properties of the proposed grade are better than those of 316L and 316LN steels. The higher strength is combined with a good ductility. The design allowable stress is higher than in the other SS grades.
- The proposed grade is less prone to delayed reheat cracking than Ti or Nb stabilised steels.
- 316L(N) is less sensitive to irradiation embrittlement than 304 steel.
- There is a comprehensive database for this grade, including heat-to-heat variations and product size.

For ITER, only minor modifications in chemical composition are required, in order to cope with the radiological safety limits and with the re-welding requirement. The safety implications of the Co and Nb content show the following.

- For a fusion reactor, activated cobalt in corrosion products plays an important role in determining the occupational dose level during maintenance after a coolant leak.
- Reducing the Co content from 0.25% to 0.05% decreases the total decay heat in the vacuum vessel by ~20% and helps to reduce the activation of components.
- Further decrease of Co to 0.01% does not further reduce the decay heat because the Ni^{60} (n,p)->Co^{60} reaction becomes dominant.
- Cobalt is one of the main components of activated corrosion products in the water cooling system.
- Nb produces long-lived radioisotopes that could become important for the decommissioning and waste disposal of in-vessel components. In 316 type SS, niobium is present as a trace element picked up during the melting process from the ferroalloy addition: existing data indicate that it is possible to reduce Nb to ≤ 0.01%.

To provide a common designation and to avoid any confusion with similar steel grades, the designation 316L(N)-IGX is used, where 316 indicates the type of steel, L low carbon content, (N) controlled nitrogen content, and IG - ITER Grade. X is a progressive number indicating the procurement specification, which defines the additional requirements in terms of product form, impurity content, QA procedure, and delivery conditions needed for the different components as follows:

- 316L(N)-IG1 for the modules of the primary wall should have a cobalt content < 0.05%;
- 316L(N)-IG2 for the vacuum vessel should have a cobalt and niobium content as low as possible; cobalt and niobium < 0.05% and < 0.01%, respectively, have been assumed as industrially feasible limits; taking into account the requirement of re-welding the irradiated material, the boron content should be limited to 10 wppm;
- 316L(N)-IG3 cast plus HIPed steel for one of the options for the divertor cassette body; cobalt is limited to 0.05%;

- 316L(N)-IG4 thin walled tubes for the primary first wall; the cobalt content is limited to < 0.05%;
- 316L(N)-IG5 in in-vessel cooling pipes: Co < 0.05% and B < 0.0010%;
- 316L(N)-IG6 produced by the powder HIP process; the composition is the same as IG3 with the exception of higher oxygen and silicon concentrations.

The physical and mechanical properties of the 316L(N)-IG steel which is available in the different forms (rolled stocks, bars, plates, tubes, etc.) are well known and sufficient for design evaluation. For some components (e.g. shield blanket modules) high temperature hot isostatic pressing (HIP) at ~ 1,000-1,100°C will be used for the manufacturing. In some cases a few HIP cycles may be applied. Also, a manufacturing option is to use stainless steel powder and HIP as a consolidation technique. As a result, the properties of the HIPed steel may differ from those of wrought material. A study of the effects of HIP cycles on the mechanical properties of solid and powder HIPed steels has been performed in the frame of ITER R&D. It was shown that the tensile properties of HIPed steel are within the design allowables. In some cases the yield strength of solid-HIPed base metal is close to the minimum specified values. Powder-HIP gives tensile properties above the average of the wrought material, but fracture toughness of powder-HIPed steel degrades faster under irradiation than wrought material.

Neutron irradiation is another factor which affects the properties of stainless steel. At the maximum expected neutron fluence ~ 0.3 MWa/m^2, the peak damage in the steel for the in-vessel components will be ~ 2 dpa with a peak He generation of about 55 appm. The irradiation temperature for these components is in the range 100-300°C. Hardening of the steel will occur, but predicted uniform and total elongation of the wrought steel will remain higher than 10% and 20%, respectively, at the end of the operation phase. For the solid-HIP steel, the estimated uniform and total elongation will be above 18% and 31%, respectively. Therefore, materials will be relatively ductile and retain work hardening capability. Initial neutron data on 316L(N)-IG, prepared by powder HIP and irradiated at ~ 75°C to 2 dpa, indicates that the tensile properties are similar to those of wrought material. However, for powder HIPed material, the fracture toughness decreases with increasing dose from ~ 1,000 kJ/m^2 in the unirradiated steel to ~ 250 kJ/m^2 after 2-2.5 dpa at 75-290°C.

For the blanket manifolds and vacuum vessel the irradiation dose will be low, less than 0.06 dpa for an average first wall 14 MeV neutron fluence of 0.3 MWa/m^2. This dose will not result in significant property changes, and design allowables for the unirradiated steel have been used for these component designs.

Vacuum vessel, manifold and branch pipe connections must remain weldable throughout the machine lifetime in spite of the He generated by neutron irradiation. The helium content generated by gaseous transmutation should be kept below a threshold value, which depends on the welding method. Based on existing data, re-welding of 316L type SS can be successfully carried out on thick welds when the He content is 1 appm. For thin single-pass welds, low energy welding can raise this limit to 3-10 appm. Helium generation in the vacuum vessel will be below the re-weldability limit of 1 appm. For the blanket manifolds, the helium generation exceeds this value.

He generation can be minimised by reducing the boron content of the steel. Neutronics calculations show that decreasing the boron to 10 wppm lowers the helium generation by ~ 44% in the vacuum vessel to a level less than 1 appm (for 1 MWa/m^2). Reducing the boron even further results in more than a factor of two suppression in the helium production. Therefore, the boron concentration in the steel used for the manifold should be as low as feasible industrially. The effect of boron on He generation is most significant for the steel close to the water cooling channels due the thermalisation of neutrons by water.

Borated Steel

A standard borated stainless steel is proposed for the shielding inserts of the VV to increase the shielding efficiency. 304B7 (UNS designation S30467) steel is used in the form of plates fixed between two VV walls. The ASTM standard A 887-89 is referred to for the base properties and the procurement specifications. Cobalt concentration should be limited to 0.2% max, unless a lower concentration can be agreed upon between the purchaser and the supplier.

Precipitation-Hardened Steel

High strength precipitation-hardened stainless steel type 660 (also known as A-286) is proposed for the port stiffening extensions and bolting the port plugs. The UNS designation of SS 660 (A-286) grade is K66286.

The heat treatment indicated by ASME SA 638 can be used for SS 660. It should be performed in two stages: solution annealing (from 900 ± 15°C, 2 h, oil or water quench) and ageing for 16 h at 705-760°C, air or furnace cooling. This is a well-known standard material, and a large database is available.

For the port plugs, doses of irradiation are relatively low (< 0.1 dpa). These doses will not result in significant property changes. Thus, properties of unirradiated material can be taken for the design analysis.

Ferritic Steel

Inserts of ferromagnetic material are used in the outboard area inside the vacuum vessel in the shadow of the TF coils to reduce the toroidal field ripple. The working temperature is ~ 20-200°C, the irradiation damage less than 0.03 dpa, and the environment is water, in galvanic contact with the VV material. SS 430 is recommended for the ferromagnetic plates, because it has relatively good corrosion resistance in spite of its slightly lower magnetisation than the best materials. The procurement specification ASTM A 240-86b and/or ASTM 883 (alloy 2) can be used for the required product form. The UNS designation is S43000.

No specific data on the corrosion of SS 430 is available for the water chemistry specified in ITER. However, SS 430, which has a chromium content of ~ 17%, should have a uniform corrosion rate similar or better than that of 304 type SS.

2.13.2.2.2 Copper Alloys

Heat-sink Copper Alloys

Cu alloys are considered as heat sink materials for the ITER first wall, divertor and limiter. Two materials have been retained as candidate heat sink materials for the high heat flux components: CuCrZr and GlidCop®Al25. The main differences between these materials are:

- the properties of CuCrZr alloy strongly depend on thermomechanical treatment which could be part of the manufacturing cycle, whereas properties of GlidCop®Al25 alloy are relatively independent of the heat treatment;
- fracture toughness of CuCrZr alloy is significantly higher at high temperatures (>~ 200°C) than that of GlidCop®Al25;
- CuCrZr is a weldable material, whereas fusion welding is not recommended for GlidCop®Al25;
- CuCrZr alloy is available from different suppliers, whereas GlidCop®Al25 is produced by one firm, OMG Americas, USA.

Starting from the original specification, a tighter limitation for composition and a specific heat treatment were proposed for the CuCrZr alloy, with the definition of an ITER grade specification, CuCrZr-IG. The proposed composition of CuCrZr alloy differs from the standard one, mainly by its narrower range of the Cr (0.6-0.9%) and Zr (0.07-0.15%) content. In different national standards the chromium content varies from 0.4% to 1.5% and the zirconium content varies from 0.03% to 0.25%. The reason for limiting the Cr content to the narrower range in the alloy is that otherwise it may result in the formation of coarse Cr precipitates which affect the radiation resistance. Zr promotes hardening of the alloy by providing a good homogeneity of precipitates. Limitation of oxygen (< 0.002%) and of the total amount of impurities (< 0.03%) is required for the same reason and for a better resistance against embrittlement. Based on industrial experience, the reference ITER heat treatment for CuCrZr-IG is the following: solution anneal at 980-1000°C for 1 h, water quench then age at 450-480°C for 2-4 hours. Ageing can be performed at any time during component manufacturing.

For ITER application, the fabrication process of GlidCopAl25 was optimised, with an improvement of ductility and reduction of anisotropy. The material is provided by the manufacturer under the trademark Glidcop®Al25-LOX-CR (low oxygen, cross-rolled). A high temperature annealing (at 950°C) is performed after cross rolling. The optimised grade is indicated as CuAl25-IG.

During component manufacturing, the base material is generally subjected to additional thermal cycles e.g. welding, brazing, HIPing, etc. Brazing, even at relatively low temperatures, results in a significant decrease of strength in the case of long term exposure and/or slow cooling rates. Brazing or HIPing at high temperature can be combined with solution annealing. In this case the brazing/HIPing temperature should be about 950-980°C, followed by fast cooling and ageing at 475-480°C. The most critical step is the cooling rate from the brazing/HIPing temperature. If a fast cooling rate can be realised after high temperature brazing/HIPing, the loss of strength is minimised.

CuAl25-IG shows much less sensitivity to heat treatment than CuCrZr-IG. Thermal stability is one of the main advantages of CuAl25-IG. This allows a wider flexibility in the use of different joining technologies for high heat flux components.

Irradiation results in an increase of strength of Cu alloys at low temperatures, below 300°C. Properties of unirradiated materials should be used in this temperature range to be conservative for analysis. At temperatures > 300°C, the strength of irradiated materials is lower, and irradiated material strength properties should be used.

Ductility of CuCrZr alloy increases with increase of irradiation temperature. At irradiation temperatures < 200-250°C, criteria for immediate plastic strain localisation and fracture due to the exhaustion of ductility are paramount in structural assessments. Above these temperatures the material will be ductile, and creep and creep fatigue properties are more critical. For CuAl25-IG in all temperature ranges, ductility decreases significantly with irradiation, and plastic strain localisation and local fracture criteria should be used for structural analysis.

Nickel-Aluminium Bronze

Nickel-aluminium bronze C63200 (82Cu-9Al-5Ni-4Fe) is proposed for nuts, bearings and rods connecting plasma-facing components to the divertor cassette. Several standard specifications can be used for the material procurement (ASTM B150, ISO 428 CuAl10Fe5Ni5, DIN CuAl10Ni and BS CA 104). High strength, low friction and spark resistance are the key properties for these applications.

The typical tensile strength of C 63200 bronze is 540-725 MPa, and yield strength is ~ 330-380 MPa, depending on the heat treatment. The ultimate compressive strength is ~ 760 MPa. There is experience of using this material in JET for similar applications, and testing for ITER is now under way. The properties of irradiated C 63200 bronze are unknown.

DS Copper Alloy, Glidcop Al60

DS Cu alloy, Glidcop®Al60, is proposed for the compression collar of the bolt fastening of flexible supports. The material is manufactured by OMG Americas Company under the trade mark Glidcop®Al60-LOX. The UNS alloy number is C15760. High material strength is provided by aluminium oxide (~ 1.1 %) particles uniformly distributed in the copper matrix. The material is used to compensate for a pre-load decrease due to thermal expansion of bolt and flexible cartridge materials, and to equalise temperatures in the module support unit due to high thermal expansion and thermal conductivity. Strength and thermal resistance of Glidcop Al60 are superior compared with other Cu alloys (ultimate strength ~ 500-530 MPa and yield strength ~ 400-440 MPa).

Irradiation results in a decrease of ductility and increase of strength for compression collar working conditions. However, the material of the collar is under compression stresses, and fracture due to immediate plastic flow localisation and fast fracture is avoided in this configuration. Irradiation may accelerate creep that should be taken into account in the design analysis.

2.13.2.2.3 Ti Alloys

For the blanket flexible supports, a flexible cartridge in the form of a cylinder with axial slots is screwed into the vessel from one side and bolted through access holes in the blanket. The design requirement for the material of the flexible cartridge is complex and somewhat contradictory. The material should have a high strength, because of high axial loading force, and at the same time allow for a wide range of elastic deformation during bending. The estimated temperature range of the flexible cartridge is between 100 and 260°C, and damage is ~ 0.1 dpa.

A comparative analysis of the properties of several candidate materials was performed. The ability of materials to withstand impact loading is proportional to $U = 0.5\ (YS)^2/E$, where YS is yield strength, and E is Young's modulus. Comparison of the candidate materials shows that titanium alloy has a yield strength slightly less than that of Inconel 718, the highest flexibility and the best U parameter, because of the relatively high strength, and low Young's modulus (approximately a factor of two less than Inconel 718 and 316 steel). Also, Ti alloys have lower resistance to buckling compared with high strength nickel alloys, since the buckling stress limit is proportional to Young's modulus.

Titanium alloys are widely used in the chemical and aerospace industries. Among the commercial Ti alloys, Ti-6Al-4V ($\alpha+\beta$ alloy) was proposed as reference grade. Since it is widely used in different countries, there is an industrial experience in many fields and the database of unirradiated material is relatively complete. Ti-6Al-4V can also be used according to ASME specification SB-381, "Titanium and titanium alloy forgings".

Ti alloy property change due to irradiation was studied for doses in the range 0.01-0.4 dpa and temperatures 40-350°C. The following general conclusions can be drawn from the available R&D results.

- Strength of all materials is increased due to irradiation.
- Ductility is decreased. However, in all experiments (even at ~ 0.4 dpa) the uniform elongation was above 2%. Irradiation dose expected for flexible cartridges in the ITER design is below 0.1 dpa. Structural criteria of unirradiated materials therefore should be used for the design analysis.
- There was no significant alteration of fatigue lifetime due to irradiation.
- Fracture toughness decreases (2-5 times) due to irradiation. Nevertheless, J_c (or K_c) values remain at a relatively high level (~ 20-60 kJ/m^2) at room temperature.

α-based Ti alloy (Ti-5Al-2.4Sn) exhibited less embrittlement due to irradiation. However, the strength of the alpha-based alloy is lower than that of the $\alpha+\beta$ alloy. Ti-5Al-2.4Sn alloy is thus considered as a back-up.

The cartridge operates in the environment of the vacuum chamber, but it is shielded from the direct bombardment of energetic particles by the module itself. The possibility of hydrogen or helium implantation by energetic particles is, therefore, eliminated. Available data indicates that a relatively high hydrogen content of 3-5 wt. % is necessary to produce significant embrittlement, in terms of impact fracture energy, in Ti alloys. ITER R&D shows that hydrogen saturation of Ti-6Al-4V alloy up to 200 wppm of hydrogen did not result in changes of tensile properties.

2.13.2.2.4 Ni Alloys

Inconel 718

High strength, fatigue and fracture toughness are required for the bolt attaching the shielding blanket modules through the flexible support, to minimise the size of attached units and to provide reliability of the attachment. The operational conditions for bolts are the following: temperature ~ 150 – 300°C, damage dose ~ 0.5 dpa, and a design fatigue lifetime that is ~ 30,000 full power cycles.

Inconel 718 is selected as the reference material due to its high strength and good ductility. As a back up the high strength grade 660, also known as A-286, precipitation-hardened stainless steel could be considered. Inconel 718 has satisfactory fracture toughness and fatigue lifetime. The limited data available suggest that SS 660 has lower stress relaxation than Inconel 718 under low dose irradiation.
Inconel 718 is used in the nuclear industry. The material is produced commercially in forms of bars, rods, plates, strips, etc. The specifications, chemical compositions, required heat treatments and properties of these materials are well established. Selecting the correct heat treatment provides materials with a high level of strength and ductility. Inconel 718 has higher strength allowing for a higher pre-loading and a reduction of the bolt diameter. Careful control of the grain size of the mill product of Inconel 718 is needed to provide good fatigue lifetime. The material has relatively good fracture toughness after appropriate heat treatment.

Data on the irradiation behaviour of Inconel 718 is sparse. Preliminary results shows that irradiation up to 0.5 dpa will result in an increase of strength and in a slight decrease of ductility that will not affect the component structural integrity and lifetime.

Significant stress relaxation is expected under irradiation. Results show that for the bolts of flexible cartridge fastenings where the irradiation dose is ~ 0.02-0.1 dpa, the stresses reduce by 10-20% from initial pre-load to the end of life, so, the required pre-stress should be ~ 800 MPa.

Nimonic 80A

Superalloy Nimonic 80A is proposed for the divertor keys (elements for fixing PFCs in the cassette body) where high strength is required. This material is produced by Inco Alloy International under the specifications ASTM B637, DIN 17742, DIN 17754 and BS 3076. The UNS designation of alloy is N07080. High strength can be achieved in the aged state. The ultimate tensile strength is about 1,250 MPa, and yield strength is about 780 MPa at RT. The material is used for components where relatively low doses of irradiation will be achieved at the end of life, i.e. < 0.1 dpa. These low doses will not result in significant properties changes, and properties of unirradiated material should be used for the design analysis.

2.13.2.3 Joining Technologies

The design of the ITER PFCs involves various combinations of joints between armour materials and Cu alloy heat sinks, and between Cu alloys and austenitic steels. The joints must withstand the thermal, mechanical and neutron loads, the cyclic mode of operation, and operate under vacuum, while providing an acceptable design lifetime and high reliability. This section summarises the results of the R&D program on the development of joining technologies. Further developments are still needed with the goal of improving reliability and improving manufacturing efficiency, and the final selection must take into account the estimation of the cost of these technologies.

2.13.2.3.1 Be/Cu Joints

The main problem of bonding Be to Cu alloys is that Be reacts with almost all metals at moderate and high temperatures and forms brittle intermetallic phases that are detrimental for the joint reliability and the fatigue lifetime. To solve this problem and to provide good quality Be/Cu joints, different approaches have been studied.

- Use of materials as fillers or interlayers between Be and Cu alloy which do not form intermetallic phases with Be (e.g. Al, Ge, Si, AlSi or AlBeMet). The joining temperature for these materials is typically equal to or less than the melting temperature of these filler materials.
- Use of diffusion barrier materials with less affinity for the formation of beryllide intermetallics. Different types of these barriers have been studied: Ti, Cr, Ti/Ni, Al/Ni Ti/Cu, Al/Ti/Cu, Cr/Cu. Typically for this type of joints, HIP is used as a joining procedure with the temperature range 500 - 850°C.
- Brazing with Cu-based brazing alloys. At a brazing temperature more than 650°C (typical temperature for these types of brazing alloys) intermetallic formation occurs, but using a "fast" brazing technique, this formation can be limited.
- Direct bonding of Be to Cu alloy (with or without a soft intermediate layer of Cu) at moderate temperature (~ 500-700°C). Direct Be plasma spray can also be carried out under this condition.

Based on the result of tests on small scale mock-ups, the following joining technologies have been determined as references.

- For ITER first wall, HIP at 850°C with a Ti interlayer. Be plasma spray could also be used. Both technologies have demonstrated satisfactory performance at a heat flux ~ 1- 2.5 MW/m^2.
- For high heat flux components (port limiter): "fast" brazing with CuInSiNi alloy and HIP at 625°C with AlBeMet interlayer. These technologies demonstrated the best thermal durability, e.g. fast brazing resisted 4500 cycles at 12 MW/m^2.

Based on preliminary data, both methods (HIP and brazing) seem to be resistant to neutron irradiation.

2.13.2.3.2 W/Cu Joints

The main problem in the development of W/Cu joints is the large difference in the coefficient of thermal expansion and of elastic modulus. With a conventional flat tile geometry, this difference creates very large stresses at the W/Cu interface. From an engineering point of view the solution is to use brush-like (rectangular or rod) or lamella type W armour design. The advantages of the brush structure are that the stresses at the W/Cu interface may be reduced as the single elements are free to expand under the heat flux, reducing the thermal stress in the tile.

Several joining methods have been developed.

- Casting of pure Cu onto W. This process consists of casting a soft compliant layer of pure copper onto the activated/or not activated surface of W. The good joint during casting is based on the high wettability and high creep relaxation ability of pure Cu. The W/cast Cu elements are then joined by different methods such as e-beam welding, brazing or HIP to the copper alloy heat sink.
- High temperature brazing technology using CuMn base braze alloy has provided reliable and good quality joints of W tiles and copper heat sink.
- Several methods for joining W rods (1.6-3.2 mm dia) to CuCrZr heat sinks have been developed. One of them is the diffusion bonding of W rod tips directly into the OFHC/CuCrZr substrate at 450°C.

- The use of CVD (chemical vapour deposition) of W onto Cu also produces good joints, but has limitations in thickness. The use of W plasma spray also gives good joints with a Cu alloy heat sink. Both these methods have been successfully applied for manufacturing of components with curved surfaces. However, as was demonstrated by testing of representative mock-ups, these technologies can be applied only for components with low and moderate heat flux.

Several technologies (casting, brazing, direct diffusion bonding of W into Cu) provide excellent high heat flux durability performance of W/Cu joints and can be recommended for the manufacturing of the divertor components. Further development has to be focused on the cost optimisation of these technologies.

2.13.2.3.3 CFC/Cu Joints

The problem in the development of CFC/Cu joints is worse than for the W/Cu joints, due to the even larger difference in the coefficient of thermal expansion of the materials. To provide a high quality CFC/Cu joint, the surface of the CFC has to be activated to increase the wetting and a compliant layers between the CFC and Cu alloy heat sink is needed to relieve the residual stresses. Several joining technologies have been developed and studied.

- Active metal casting (AMC®) technology, which was originally developed for the Tore Supra limiters. This technology includes special laser treatment of the CFC surface, followed by casting of pure Cu onto CFC, machining and final joining with the Cu alloy heat sink. In the monoblock configuration, CFC tiles with AMC® Cu can be joined to Cu alloys by brazing or HIP. For flat tiles, the same joint could be obtained by e-beam welding. AMC® technology was applied to different CFC grades (SEP N11, SEP N31, etc.).
- Brazing with silver-free alloys (CuMn, CuSiAlTi) and HIP-assisted brazing with CuMn and CuTi alloys. The best high heat flux results have been observed for CuMn brazes.

Few technologies (AMC® and brazing with CuMn) provide the required high heat flux durability performances of the CFC/Cu joints. Based on preliminary data, both methods are resistant to neutron irradiation.

2.13.2.3.4 SS/Cu Joints

Plasma-facing components involve the use of three different types of SS/Cu joints: tube-to-tube, plate-to-tube, and plate-to-plate. For the first wall, solid HIP is considered as the most promising method of achieving the required quality of joints. Two ranges of HIP temperatures have been studied: high, ~ 1,050°C, which permits the combination in one treatment of the joining of Cu to steel and steel to steel, and moderate temperatures ~ 920°C, in which case steel-to-steel joining has to be performed by a separate procedure. For HIP joining the use of CuAl25-IG alloy is preferable due to its excellent resistance to high temperature heating. With fast cooling (~ 2°C/s) and applying further ageing, the properties of CuCrZr-IG after HIP are acceptable. Both methods give high quality bonding and tensile properties of joints very close to the strength of Cu alloys. Optimised SS/CuAl25-IG joints have good radiation resistance. Strength of the HIP-bonded joints after 0.4 dpa irradiation dose at 150 and 300°C is higher than or equal to that of the base metal, with the failure location within the Cu part close to the interface.

Another joining method, i.e. explosion bonding, seems promising and provides a good quality of joint, but it is applicable only for simple geometric component shapes. For Cu tube to stainless steel tube joining, friction welding, diffusion bonding (for DS Cu) and TIG welding with Ni interlayer (for CuCrZr-IG) can be used. The casting of CuCrZr onto stainless steel also gives a good quality of joint, that is also resistant to neutron irradiation.

2.13.2.3.5 SS/SS Joints

All austenitic stainless steels are readily weldable with a wide range of well-established methods, such as narrow gap TIG welding, electron beam welding, and laser welding. The weld joints should be performed in accordance with internationally accepted procedures such as described in ASME section IX. 316L(N)-IG can be considered to be a member of the P-8 Group 1 weld joints. Narrow gap TIG welding is selected as reference for the vacuum vessel manufacture. The properties of the weld are well-defined and compatible with the load requirement of these components. For TIG joints, the unirradiated welded material shows, in general, a higher yield stress, lower ultimate tensile strength and lower ductility than plate material. The effect of neutron irradiation on the tensile properties is similar to that for plate material, i.e. significant hardening and loss of ductility occurs. It is expected that the worst ductility reduction occurs in the irradiation temperature range of 275-375°C, well above that appropriate for ITER.

For some components, the welding of irradiated steel is needed. From the material point of view, helium content generated by gaseous transmutation should be kept below a threshold value. The threshold He content is rather low and although it will not cause any change of the mechanical property of the base material, it can still promote the formation of cracks in the heat-affected zone during welding. Certain precautions have to be taken, such as low weld heat input, surface preparation, avoiding high tensile stresses, etc. The welding method must therefore be carefully chosen.

The joining of stainless steel to steel by HIP is considered as the main method for the manufacture of the shield blanket modules. The quality of joint strongly depends on the quality of the base metal and on the surface preparation during the HIPing process. The base metal should be vacuum re-melted for a low content of impurities and of non-metallic inclusions. The surfaces should be polished, etched and HIPed right after surface preparation. Any delay in HIPing may result in the formation of an oxide layer with unsatisfactory properties of the joint. The manufacturing process should be carried out under strict quality control.

Most of the tensile data for SS/SS HIP joints are within the design allowable but close to the lower bound of the specified strength. Neutron irradiation has a slight effect on the mechanical properties of the joints, demonstrating a similar behaviour as for the base metal.

2.13.2.4 Ceramics for Electrical Insulation

Aluminium oxide and aluminium magnesium oxide (spinel) are proposed for the insulation of the modules and for the limiter plates. The same materials are used as for electrical insulator for the diagnostic components. The main difference between the diagnostic application and that in the blanket module and in the limiter and baffle plates is the additional requirement to sustain significant static and dynamic loads. The ceramic provides not only passive electrical insulation but also fulfils a structural function.

Plasma spray is one of the most promising options for components manufacture where insulation is required. Coatings have good mechanical properties in the unirradiated state. Results of R&D shows that the coating is not damaged if the compression stress does not exceed the yield point of the substrate material. In ITER, the maximum design load during operation, 200 MPa, is below the yield point of 316L(N)-IG.

Impact tests of unirradiated and irradiated ceramic coatings have been performed. ITER R&D has demonstrated that a 12 kg weight falling from 20 mm did not cause the coating to disintegrate (> 10,000 cycles).

The radiation-induced conductivity is estimated at about 10^{-6} - 10^{-4} S/m for the ITER working conditions (first wall and limiter). Therefore the estimated electrical resistivity of coatings will be no worse than ~ $3x10^{2}$-10^{5} Ω. Dielectric breakdown strength either does not change or decreases under irradiation depending on the material. But in the worst case the estimated dielectric breakdown strength of alumina and spinel is almost one order of magnitude better than required for the insulators of the first wall module. It is not expected that significant strength degradation of the ceramic materials, alumina and spinel will occur for the dose of irradiation anticipated within the flexible attachment, ~ 0.3 dpa.

2.13.2.5 Material Selection for Vessel/In-Vessel Components

A summary of the materials grades selected as reference for ITER vessel/in-vessel components is given in Table 2.13-1. Behind the use of each material is a substantial body of R&D to substantiate the properties and the materials use, in the main carried out in the frame of ITER EDA R&D. For the current ITER design a proven material solution exists for each component. However, work is continuing so as to confirm that material properties can be preserved under the most cost-effective component manufacturing processes.

Table 2.13-1 Materials for Vacuum Vessel (VV) and In-vessel Components

Material	Material Grade	Components
Beryllium	S-65C VHP (backup DShG-200)	· Armour tiles for first wall and limiter
Tungsten	Pure sintered W	· Armour tiles for divertor components
Carbon fibre composite (CFC)	SEP NB 31, NIC 01 (back-up CX 2002U, SEP NS31)	· Armour tiles for divertor vertical target
Cu and Cu alloys	CuCrZr-IG	· Heat sink for plasma-facing components (PFCs) and for heating systems
	CuAl25-IG	· Heat sink for PFCs
	Nickel-aluminium bronze	· Nuts, bearings and other friction parts
	Glidcop Al60	· Compression collar of the flexible support bolts
Austenitic and precipitation hardened steels	316L(N)-IG1 plates and forgings	· Shield modules
	316L(N)-IG2 plates and forgings	· Vacuum vessel, blanket cooling manifolds
	316L(N)-IG3 cast	· Some vacuum vessel components and back-up material for divertor body
	316L(N)-IG4 tubes	· Thin walled tubes for first wall
	316L(N)-IG5 tubes	· In-vessel cooling pipes
	316L(N)-IG6 powder HIP	· Back-up material for shield modules
	AISI 660 (A-286)	· Fastening components for the port plugs (e.g., fixing wedges and bolts)
	SS 30467	· Boronised steel for in-wall shielding structures (plates)
Ni alloys	Inconel 718	· Bolts for the flexible supports and electrical straps, blanket cooling manifold support
	Nimonic 80A	· Keys in divertor PFCs
Ti alloy	Ti-6Al-4V	· Flexible cartridges for the module support
Ferritic steel	SS 430	· Ferromagnetic insert
Ceramic	Al_2O_3 or $MgAl_2O_4$	· Electrical insulators of module attachment and limiter plates

Note: Materials used for commercial components are not included in the table;

2.13.3 Materials for Diagnostic Components

Diagnostic components use a variety of different materials. These materials are distributed at different locations within ITER and, as a result, the operational conditions for them are very different. However, the key issue for these materials is radiation resistance under ITER conditions.

2.13.3.1 Electrical Insulators/Ceramics

Insulating ceramics will be used in feedthroughs, connectors, mechanical supports and general stand-offs, mineral insulated (MI) cables and their seals, substrates of bolometers, and other sensor devices such as pressure gauges. Ceramic windows are also used in diagnostic systems and heating/current drive systems. Almost all the candidate ceramics retain the geometrical stability demanded by the design, namely within ± 5% (± 1% for magnetic coils) for the full temperature and neutron fluence. In nearly all cases they must be high-vacuum compatible, and the electrical conductivity of ceramics should remain below a specified value depending on the diagnostic systems. This in general is ~10^{-6}S/m.

For ceramics, radiation induced conductivity (RIC), radiation induced electrical degradation (RIED), dielectric, thermal and mechanical properties, and tritium diffusion, are the key properties.

Present data indicates that, with careful choice of material and operating temperature range, the long-term volume degradation of the electrical insulation (bulk RIED) should not impose serious technological problems for ITER conditions (up to 2 dpa). RIC plays a major role in determining the electrical insulating ability at the onset of operation, and is quantitatively well evaluated. In general, RIC will not impose a serious technological problem except possibly for applications such as the bolometer substrate, where cross-leakage may be important.

Additional problems which have been identified during RIED investigation, such as surface conductivity, insulator cracking and electric charging effects, require further investigation to determine the physical mechanisms responsible for associated RIED-like electrical degradation, and to assess their possible influence on the insulator performance and lifetime. The possibilities of surface contamination and degradation must be taken into account and countermeasures to mitigate such effects should be considered.

2.13.3.2 Wires/Cables

Different cable types (mineral-insulated, steel and glass-braided) with different configurations (twisted pairs, coaxial, multi-axial etc.) are under consideration. For cables/wires, RIC, RIED, radiation induced electromotive force (RIEMF) between sheath and centre conductor of mineral insulating (MI) cables, conductor resistance, and dielectric breakdown strength, are important properties. Reliable cable termination techniques with low leakage currents for MI cables are also an important issue.

RIC and RIEMF are quantitatively well documented. RIC is evaluated to be ~10^{-10} S/Gy s. RIEMF current will be ~10^{-10} A/Gy s m for typical MI-cable below 600˚C.

Magnetic coils made from MI-cable require an insulating conductivity of less than 10^{-6} S/m. This requirement for magnetic coils set behind shield blankets will be satisfied.

In ITER geometry, the voltage generated between two ends of a centre lead by the RIEMF is evaluated to be of the order of 1 μV. However, the full resolution of the RIEMF topic is still to be completed due to uncertainities in the interpretation of the results. Indications are encouraging but further tests and analytical work are required to clarify the matter.

2.13.3.3 Windows

Windows, acting as vacuum and tritium confinement barriers, transmit optical and microwave signals. Window assemblies must be vacuum tight to UHV standards and also be able to withstand the potential 0.2 MPa pressure rise during an in-vessel coolant leak.

Some issues for windows are common with ceramic issues. For potential window materials, the optical properties including radioluminescence and optical transmission have been extensively studied. In addition to enhanced tritium diffusion, there is a possibility of enhanced tritium leaks due to microcracking in the ceramics-to-metal joints due to effects such as radiation-enhanced segregation and sub-critical crack growth (SCCG). For LIDAR applications, synergistic effects of high power laser-beams and irradiation and/or surface degradation are very important from the mechanical damage point of view.

Based on ITER R&D, fused silica glass or quartz windows are recommended for spectroscopic systems for the visible region, and sapphire windows should be used for the IR region. From the viewpoint of radioluminescence, fused silica has an advantage over sapphire as a window material. On the other hand, optical absorption in silicas (covalent-bonding) is sensitive to gammas (ionising radiation), namely radiolysis, as well as to atomic displacements. Optical absorption in sapphire is relatively insensitive to gamma radiation and only displacement damage associated with neutron irradiation induces notable absorption. Thus, under high-flux ionising-radiation, sapphire will have some advantages over fused silica. Crystalline quartz is more resistant to radiolysis than fused silica.

The availability in quantity and in sizes of window materials made of fused silica (synthesised silica especially for large windows and optical fibres) should be considered. Two possible fused silicas which have been found to have radiation-resistant optical properties are KU-1 (high OH) and KS-4V (low OH). In the case of diamond, excellent material grades have been developed in the EC H&CD programme (at 145 GHz), which may possibly be used for diagnostic windows from the GHz region to the IR /visible range.

2.13.3.4 Optical Fibres

For a large variety of optical diagnostics it is extremely desirable to be able to use fibre optic transmission close to the plasma region to take advantage of the limited spatial access and easy alignment of optical components.

Important issues for ITER diagnostics are radiation resistance of optical fibres especially for the visible region, and their application in the vicinity of the first wall. (Optical fibres will be employed in such a way that only a few metres will be exposed to the high radiation flux region.) Irradiation effects on radioluminescence and degradation of optical transmission due to permanent and transient absorption are important. Key parameters of optical fibres, which affect radiation resistance, are fibre composition (dopants, OH-content, impurities, cladding type), fibre fabrication (manufactured drawing speed and temperature) and preform fabrication.

The behaviour of optical fibres is excellent in the IR region (low absorption and low radioluminescence). Serious technical problems are not anticipated and suitable fibres already exist. The main problem will be in the associated electronics in an optical communication system. R&D indicates that the radiation-induced loss could be less than 10 dB/m in the wavelength region of 800-1,200nm for some optical fibres, with a fast neutron fluence of 10^{25} n/m^2 and ionising radiation dose of 10^{10} Gy.

Recently developed fluorine-doped optical fibres revealed better radiation resistance, 20dB/m or less in the visible region for a fast neutron fluence of 10^{23} n/m^2 and 10^8 Gy. Taking into account these results, there is a possibility to use the optical fibres for the visible region inside the cryostat during operation. More detailed, specific and extensive studies, including manufacturing processes, are needed before final conclusions can be made for the application of optical fibres in ITER. Further radiation-resistant optical fibre development, and round robin irradiation tests, are underway.

2.13.3.5 Mirrors/Reflectors

The first mirrors will experience the most severe environmental conditions in the ITER device. In addition to irradiation effects on mirrors, other adverse effects of sputtering, evaporation or coating on the mirror surfaces can change the reflectivity of the mirrors.

R&D has been focused on the irradiation on metal bulk mirrors, single coated mirrors, dielectric coatings on ceramics for high power laser beams, layered synthetic microstructures for UV reflectors, crystalline materials for X-ray spectroscopy, and graphite elements for submillimeter plasma diagnostics.

It is believed that solutions now exist for situations where the dominant potentially damaging mechanism is erosion due to the CX flux of energetic neutrals. This is likely to be the case for the plasma facing mirrors mounted in the equatorial and the upper ports.

For mirrors mounted in ducts in the shielding labyrinths however, deposition could be the dominant potentially damaging mechanism. This will most likely be the situation in the divertor where there may be substantial erosion. There are only limited experimental results on deposition and models of the process are still at an early stage. There is therefore a need for controlled experiments on present-day devices and the development of models and possible mitigating methods.

Dielectric mirrors and layered synthetic microstructures should be used in well-shielded locations and under temperature control, to avoid differential nuclear and thermal swelling.

Some inorganic X-ray crystals, though they are vulnerable to radiation damage, do not have a change in their reflectivity, interplane distance, and width and form of diffraction lines, up to 10^{23} n/m^2 for Ge, Si, SiO_2 and Graphite, and to 10^{22} n/m^2 for Mica monocrystals.

2.13.3.6 Materials Selection for Diagnostic Components

Reference materials for application in the diagnostic components in ITER have been selected (Table 2.13-2). This choice depends on a comprehensive irradiation database of diagnostic components which has been accumulated during ITER EDA R&D. While not all material choices are yet proven, promising candidates exist for nearly all applications. The most serious potential problem identified so far concerns noise, for example from radiation-induced emfs, and its effect on plasma control sensing magnet coils, which could cause the baseline to drift to the point at which control becomes difficult for very long pulses (> 1000 s), and alternative solutions would then be needed.

Table 2.13-2 Materials for Diagnostic Components and Results of Irradiation Tests

Diagnostic components/ sensors	Candidate materials	Main R&D results: Accumulated effects (Dose)	R&D results: Dynamic effects (Dose rate)
Ceramics (electrical insulators)	Single crystal and polycrystal alumina (Al_2O_3)	2 dpa in helium atmosphere (RIED: < 10^{-6} S/m)*1	10^4 Gy/s (RIC: < 10^{-6} S/m)
Wires /Cables	MI-cables: SUS, Inconel (sheath)/MgO, Al_2O_3 (insulator)/ Cu, Ni (centre conductor)	1.8 dpa (RIED: No catastrophic degradation)*2	10^4 Gy/s (RIC: < 10^{-6} S/m) 10^3 Gy/s (RIEMF: < 10V)
Windows	Fused Silica/Quartz KU-1 (400-1500 nm; high OH)	10^{-3} dpa (Transmission: 5% degradation; 8 mm thickness) 6x10^{19} n/cm^2 , 2.5x10^9 Gy (High trans. for $\lambda > 350$nm)	Radioluminescence: 10^7 photons/Gy.Å.sr.cm^3 at 410 nm
	Sapphire (800-5000nm)	0.4 dpa (Transmission: No degradation; 1mm thickness)	Radioluminescence: 10^{10} photons/Gy.Å.sr.cm at 410 nm
	Diamond (GHz-IR)	10^{-2} dpa	
Optical fibres (Visible region)	Pure silica (core)/F doped (clad)/Al jacket (RF:KS-4V)	10^7 Gy (pure gamma) (Transmission loss: 2-2.5 dB/m)	Radioluminescence: (Cerenkov + 450-650 nm)
(Visible region)	F doped silica (core)/F doped (clad)/Al jacket (JA F-doped)	10^{-2} dpa (10^8 Gy) (Transmission loss: 20 dB/m)	Radioluminescence: (Cerenkov + 450-650 nm)
(IR region)	Pure silica, F doped (core)/F doped (clad)/Al jacket	0.5 dpa (10^{10} Gy) (Transmission loss: 5 dB/m)	Radioluminescence: (Cerenkov + 1270 nm)
Mirrors /Reflectors	First mirrors: Metal (Cu, W, Mo, SS, Al)	40 dpa (Cu)*3 (Reflectivity: No degradation)*4 0.1 dpa (Mo, Al) (Reflectivity: No degradation)	
	First mirrors for LIDAR: Single coated (Rh/V)*5		
	Dielectric mirrors: (HfO_2/SiO_2, TiO_2/SiO_2)	< 10^{-2} dpa *6 (Flaking, Blistering)	
	LSMs *7: (Mo/Si, W/B_4C and W/C)	< 10^{-2} dpa (the shift of the peak reflectivity to shorter wavelength)	
	X- ray crystals: (Ge, Si, SiO_2, Graphite)	10^{-2} dpa	
Magnetic coils	MI cables	10 dpa (Dimensional stability)	Asymmetry in output voltage
Bolometers	Mica substrate*8	10^{-2} dpa (Expansion: 0.15% at 150 ˚C)	
	Au meander	10^{-2} dpa (Resistivity: 20% increase)	
Pressure gauges	Alumina feedthrough, W filament, SS frames		

*1 Substantial surface degradation in Wesgo 995 and Ruby is observed.
*2 Some long-term degradation even without voltage application is observed.
*3 Simulation experiment using Cu+ ions of 1 or 3 MeV.
*4 Sputtering by energetic particles near plasma will affect reflectivity seriously.
*5 Sputtering and redepostion will be important.
*6 Partially damaged. Dielectric mirrors are used as second mirrors.
*7 LSM (Layered Synthetic Microstructures): in well-shielded location and temperature control.
*8 Al_2O_3, AlN being considered for thin substrate.
Note: Abbreviations are defined in the text.

2.13.4 Materials for Magnets

2.13.4.1 Introduction

The magnets have two classes of structural materials: metallic and non-metallic, and one class of electrically conducting materials. Within the metallic materials, there are two subclasses: conductor jackets and structural elements. The conductor jackets require special consideration because of their close association with the superconductor fabrication and (for Nb_3Sn) their potential influence on its superconducting properties. Within the non-metallic components, there are four subclasses: high voltage insulation, low voltage insulation, non-metallic structural components and low friction surfaces. The only fully non-metallic component (apart from small local fillers that form part of the electrical insulation although they have no role in forming the electrical barrier) is the precompression ring. The electrically conducting materials have four subclasses: Nb_3Sn cables, NbTi cables, joints and current leads.

2.13.4.2 Metallic Materials

2.13.4.2.1 Requirements

The operating temperature of the magnet structures is predominantly in the range 4.5-20 K. Under fault conditions loads can occur up to 300 K but these do not influence the material choice. The gravity supports and the vacuum vessel supports have a temperature gradient, from 4.5 K at one end to 300 K or above at the other. Some keys and bolts (especially those associated with precompression structures) require preloading at room temperature.

The nuclear radiation levels on the magnets are low (in terms of the potential impact on material properties) since the most sensitive components are the organic fillers used in the electrical insulation. There is a requirement to reduce impurities that result in long life residual radioactivity (cobalt and niobium).

There is a generic need for good material properties for the magnets. The critical properties are those relating to plastic yield, fast fracture and fatigue crack growth rate. For keys and bolts, typically a high yield strength at room temperature is required.

2.13.4.2.2 Material Characteristics for Structural Elements

The most obvious material to meet most of the structural requirements is austenitic steel. There are two well known classes frequently used in cryogenic applications, AISI 304L(N) and 316L(N). Unstabilised austenitic steels such as 304 can, under some conditions, undergo a martensitic transition in association with fatigue crack growth, which at 4K can lead to fast fracture. This is not generally acceptable for stressed components, and the fully stabilised 316 class is required. The cost penalty of this choice is small.

The 316L and LN class has a fairly wide range of chemical compositions and the properties at 4K have a substantial range. To achieve properties towards the upper end of the range for yield stress and fracture toughness K_{ic} (typically over 1000 MPa and 200 $MPam^{1/2}$ respectively), a narrower composition range is required, typically with nitrogen, nickel and chromium levels at the maximum allowable for 316LN. This has led to the definition of a class of 'strengthened austenitic steels' for the magnet structural elements. These are steels with a specific individual composition that are close to the 316LN range. In some recent developments, the composition is extended outside the 316LN range, usually by the addition of extra manganese to increase nitrogen solubility. These new steels have been qualified after trial component fabrication to confirm that the manufacturing route does not lead to martensite formation. For less critical components, 316LN and 316L remain suitable candidates (yield stress and fracture toughness K_{ic} in the range 700-900 MPa and over 200 $MPam^{1/2}$ respectively).

The fatigue properties of these steels are generally associated with impurity content, grain size, and the formation of brittle phases as the result of heat treatments (including welding). For high performance components, removal of impurities using vacuum oxygen degassing and electroslag refining are required for the material preparation. Welding procedures and weld fillers have to be qualified on an individual basis and avoidance of brittle delta ferrite formation is a strict requirement.

The main welding techniques qualified are:
- electron beam (EB) of strengthened austenitics (forgings and castings) for thicknesses up to 50mm;
- submerged arc welding of strengthened austenitics (forgings and castings) for thickness up to 250mm (also used combined with EB);
- gas tungsten arc welding of strengthened austenitics (forgings and castings) for thicknesses up to about 30mm (to be used combined with submerged arc welding where low heating is required for the initial weld passes, as in the TF coil case closure).

In addition to the austenitic steels, exceptionally high strength materials may be required for local components such as keys and bolts. Two standard varieties of Inconel are selected for this purpose.

2.13.4.2.3 Material Characteristics for Conductor Jackets

The material for the conductor jacket has an important structural role in the CS and PF coils.

For the PF coils, the cable material is NbTi. The jacket is fabricated as extruded or drawn sections which are assembled onto the cable by butt and/or longitudinal welding. The stress issues are entirely tensile fatigue (the jacket does not operate at high Tresca stress levels) and a refined 316LN or L austenitic steel is suitable.

For the CS, the cable material is Nb_3Sn and the material choice is much more complicated. The jacket material may have to be co-reacted through the Nb_3Sn heat treatment (approximately 200 h at 600-650°C) and both high yield strength and good fatigue resistance are required. Assembly onto the cable after heat treatment is possible but requires extreme care as the Nb_3Sn cable is brittle (a thin protecting jacket is used). It is preferable (but not an absolute requirement) for a co-reacted jacket to have a thermal contraction coefficient close to that of the Nb_3Sn strands, since the strand superconducting properties are decreased by strain.

Three materials are available for the conductor jacket which can be co-reacted with the cable. In addition, any of the strengthened austenitic steels or standard 316LN can be used as a jacket material if assembled onto the cable after the heat treatment.

The following materials are all specially developed for co-reaction:
- Incoloy 908;
- pure titanium (with controlled oxygen content);
- modified 316LN.

The first two of these have thermal contraction coefficients that match the Nb_3Sn. Incoloy 908 is a nickel-based superalloy that undergoes precipitation hardening in the reaction heat treatment, leading to a good fatigue performance. It is however extremely sensitive to stress accelerated grain boundary oxidation and results in stringent requirements on oxygen in the heat treatment atmosphere. Pure titanium is not sensitive to oxygen but has a low modulus. The final option, a modified 316LN steel, has a significantly higher thermal contraction. It has a low carbon level to avoid embrittlement by carbide precipitation during the heat treatment, and enhanced nitrogen to improve the yield strength.

2.13.4.2.4 Material Selection

Table 2.13-3 gives a summary of materials allocated to the various components. EK1, EC1, JJ1 and JK2 refer to specific varieties of strengthened austenitic steels that have been qualified by particular suppliers.

Table 2.13-3 List of Materials for Magnet Structures

Class	Specification	Components
Strengthened austenitic steel	EK1 or JJ1 for cryogenic applications as forged sections	· TF coil case (inner leg basic elements) including stub elements · Pre-compression ring flanges
	EK1 or JJ1 for cryogenic applications as forged plates	· Gravity support stacked plates, flanges and joint keys · Friction joints · Webs of intermediate outer intercoil structure stub elements (fabrication option)
	JK2 as extruded/rolled sections	· Reinforcing for Ti CS conductor jacket
	JK2 as forged plates	· Buffer elements of CS pre-load structure (due to low thermal contraction of JK2)
	EC1 for cryogenic applications as cast sections	· TF coil case (outer leg basic elements) including (fabrication option) outer intercoil structure stub elements
316 austenitic steels	316LN as forged plates	· PF coils supports (frames, clamp plates, tie rods, flexible plates and bolts) · Parts of radial plates and covers · CS supports · Tie plates, flanges, adjustable wedges of CS pre-load structure
	316L as extruded square tubes, circular tubes and circular sections	· PF coil jackets · Parts of radial plates and covers · VV support stacked plates and flanges · CCs supports · Cooling pipes
Nb_3Sn conductor jacket materials	Incoloy 908 extruded square tubes	· CS conductor jacket
	Modified 316LN	· TF conductor jacket
	Titanium	· CS conductor jacket
Inconel	718 as fasteners	· Pre-compression ring bolts, inner intercoil structure poloidal keys and outer intercoil structure bolts · Joints bolts for gravity support and VV support

2.13.4.3 Non-Metallic Materials

2.13.4.3.1 Requirements

Electrical Considerations

The high voltage insulation in the coils is designed for a maximum level of 20 kV which allows for anticipated levels of faulted operation as well as effects such as switching jitter that increase the nominal maximum operating level. All high voltage insulation must incorporate a true electrical barrier which is capable of withstanding the electrical fields without a filler material (because filler materials may contain gas-filled voids which can break down). Low voltage insulation must withstand < 10 V and the insulation has only to maintain separation of the conducting surfaces. The insulating breaks are electrical breaks that have to be placed in each helium cooling pipe that is connected to the coil and which have to withstand a maximum voltage level of 20 kV.

Mechanical Considerations

The insulation material contributes to the support of the magnets by acting both as a bonding agent within a coil (only for high voltage insulation) and a filler material that transmits pressure loads.

The non-metallic insulation material used in the coil has an important secondary role as a filler that is used to absorb conductor manufacturing and winding tolerances.

The precompression ring is required to have high tensile strength in the circumferential direction both at room temperature and 4K.

Some structural supports (for the PF coils) must provide a reaction against vertical load, while allowing relatively free component movement parallel to the surface. This requires generally a non-metallic interface that can provide low sliding friction and is wear resistant under high compression loads.

The insulating breaks may be loaded by extension loads due to relative movement of the coil and the cryostat.

Manufacturing Considerations

The filler material used in vacuum impregnation processes (usually a form of epoxy resin) has several constraints because of the small gaps that have to be filled. The resin must have a low viscosity and a pot life significantly longer than the filling time at the filling temperature (a minimum of several hours, depending on the precise resin composition) to allow proper filling of the insulation. Flexibiliser is generally used to increase the resin fracture toughness and therefore to reduce the susceptibility to cracking at low temperature.

The reinforcement material and electrical barrier used together with the filler for high voltage insulation is required to be sufficiently robust (i.e. resistance to tearing if in the form of tapes) during manufacturing to allow easy wrapping of the conductor and a high filler density to be obtained by compression.

Nuclear Radiation

Extensive investigations of candidate insulation materials have been performed, based around mechanical and electrical tests of insulation systems after irradiation in a fission reactor.

Tetraglycidyl diaminodiphenyl methane epoxy (TGDM) and polyimide resin systems, suitable for pre-impregnation and high-pressure laminates, lose little strength and stiffness and do not change dimensionally after irradiation at 4 K as high as 1.8×10^{22} fast (>0.1MeV) neutrons/m^2.

Diglycidyl ether of bisphenol A (DGEBA) epoxy resins are more sensitive to radiation than TGDM systems. The principal deleterious characteristics of this resin system following irradiation are significant loss of shear strength and the expansion caused by increased porosity from gas evolution during warm-up to room temperature. Despite this, it is the first choice for impregnated systems because of its low viscosity and long pot life.

Mica barriers do not offer improved radiation resistance compared to other electrical barrier systems and have a low shear capacity. The use of polyimide film or inorganic coatings for electrical barriers will provide higher shear strengths and better radiation resistance.

2.13.4.3.2 Material Selection

Table 2.13-4 Potential Non-metallic Materials

Material		Process
Resins		
DGEBA	diglycidyl ether of bisphenol A epoxy, anhydride curing agent	VPI*
flex. DGEBA	flexibilized diglycidyl ether of bisphenol A epoxy, anhydride curing agent	VPI*
TGDM	tetraglycidyl diaminodiphenyl methane epoxy, amine curing agent	PP*
Reinforcements		
S-2 glass (also known as R-glass or T-glass)	S-2 glass (boron free) with silane finish	wrapped as tapes or injected with resin as chopped fibres
Electrical Barriers		
PI (HA) film	polyimide film, 0.025 mm thick, relatively amorphous structure	
PI (H) film	polyimide film, 0.025 mm thick	
PI (HPL)	polyimide	HPL*
Low Friction Surface		
Fibreslip	woven mat of PTFE and glass fibre with epoxy binder	bonded to support surface

*VPI: Vacuum pressure impregnation; PP: Pre-impregnation; HPL: High pressure laminate

The list of potential non-metallic materials is given in Table 2.13-4 and the materials applicable to the various magnet components in the current design are given in Table 2.13-5. The high voltage insulation system is chosen on the basis of its manufacturing convenience which outweighs its sensitivity to radiation damage. High pressure laminate is chosen for the low voltage insulation because of its high compressive strength and tolerance to small local gaps and sliding of the contact surfaces.

The pre-compression ring is formed by winding glass filaments coated with epoxy binder around the circumference. The winding-case filler is applied by mixing the resin and filler before filling the case with a VPI process. The insulating breaks are preformed (i.e. before welding into the helium cooling tubes) by winding glass filaments coated with epoxy binder around a central former which includes the steel end connections and the central electrical break. Steel reinforcement and voltage screens may be included within the winding.

Table 2.13-5 Non-Metallic Materials for Magnets

Components	Material and Form
TF, CS, PF Turn Insulation	Pre-impregnated glass epoxy TGDM type with polyimid film barrier and S-2 glass reinforcement or VPI DGEBA flexibilised epoxy with polyimid film barrier and S-2 glass reinforcement
Filler for Radial Plate Grooves, CS, PF Winding Pack Filler	VPI DGEBA flexibilised epoxy with S-2 glass reinforcement (only if pre-impregnation used for turn insulation)
Radial Plate Insulation	VPI DGEBA flexibilised epoxy with S-2 glass-kapton barrier (with bonded or unbonded sheets)
TF, CS, PF Ground Insulation	VPI DGEBA flexibilised epoxy with S-2 glass-kapton barrier (with bonded or unbonded sheets)
TF winding - Case Filler	VPI DGEBA flexibilised epoxy glass with S-2 chopped glass filler
Low Voltage Insulation	High pressure laminate
TF Coil Precompression Ring Insulating breaks	High density unidirectional glass fibre reinforced epoxy
PF1, 6 sliding surfaces at the supports	Fibreslip bonded to coil ground insulation

2.13.4.4 Conductor Materials

2.13.4.4.1 Requirements

The main coils are superconducting and the type of superconductor used depends on the field, temperature and current levels associated with the different coils. Two types of superconductor are commercially available and can be considered: Nb_3Sn and NbTi. The choice is related to the overall machine design (see 2.1). Nb_3Sn is of the A15 type and has the the highest critical temperature at high field. However it is a brittle compound and all of the mechanical forming operations associated with the coil have to be completed before the compound is formed by a reaction heat treatment at about 600°C for about 200hrs. This reaction forms the Nb_3Sn from niobium filaments distributed in a tin-bearing matrix (either bronze or copper with a central tin core). NbTi is a metal alloy and is ductile.

Nuclear Radiation

Nb_3Sn and NbTi filaments show negligible effects of radiation at levels up to those corresponding to the coil insulation limits. At higher levels, there is evidence of changes in critical properties of the strands (initially an improvement, followed by degradation). The copper stabiliser in the conductor undergoes a resistance increase with neutron irradiation at 4K that can be largely recovered by annealing on warm-up of the coils. For the expected dosage over the lifetime of the machine, about 5 warm-ups of the TF coils will be required to keep within the electrical conductivity specification of the stabiliser.

Joints

Each coil is made up of a stack of pancakes (or multiple pancakes) of wound superconductor, with a joint at each end. The joints of adjacent pancakes are connected together and those at the ends are connected to the coil busbars. The joints generally represent a structurally weak point in the conductor as the current must be transfered from the superconducting strands to a copper interface before entering the next conductor. In some joint techniques it is possible to join the cables directly and extend the jacket containment around the whole electrical contact part. However, in some locations (the coil terminals to the busbars), this is not practical and the copper must form part of the jacket containment. In this case, methods of joining the copper contact interface to the conductor jacket are required.

2.13.4.4.2 Material Selection

Nb_3Sn strands are formed in a multistage drawing process with intermediate restacking of the billets. The strand layout consists of a core of a bundle of filaments in a matrix, surrounded by a diffusion barrier (that prevents the tin contaminating the outer copper during the heat treatment). Outside the diffusion barrier there is a layer of copper stabiliser and then a coating which controls the contact resistance to adjacent strands (Table 2.13-6). NbTi strands consist of NbTi filaments in a copper matrix with again a surface coating to control the contact resistance (Table 2.13-6).

Table 2.13-6 Materials Selected for Nb_3Sn and NbTi Strands.

Type	Matrix	Diffusion Barrier	Filaments	Stabiliser	Coating
Nb_3Sn	Bronze or tin in copper	Tantalum or tantalum/niobium, thickness about 10µm	Ti doped with Ta, diameter about 2-5µm	High conductivity copper, RRR*>120	Electroplated Cr
NbTi	copper	none	NbTi, diameter about 5µm	High conductivity copper, RRR>120	Electroplated Ni

*residual resistivity ratio

Jointing techniques that are available are defined in the Table 2.13-7, and those selected are given in Table 2.13-8.

Table 2.13-7 Jointing Techniques

Type	Jacket Seal	Electrical Seal
Lap	Explosion bonded steel-copper, steel-jacket weld	Solder
Butt	Jacket-jacket weld	High pressure sintering

Table 2.13-8 Superconductor and Joint Types Used in the Various Coils

Coil	Superconductor Type	Internal Joint Type	Terminal Joint Type
Toroidal Field	Nb_3Sn	Lap/Butt	Lap
Central Solenoid	Nb_3Sn	Butt	Lap
Poloidal Field	NbTi	Lap	Lap
Correction	NbTi	Lap	Lap