ITER IN-VESSEL SYSTEM DESIGN AND PERFORMANCE

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Abstract

This paper reviews the design and performance of the in-vessel components of ITER as developed for the EDA Final Design Report (FDR). The double-wall vessel is the first confinement boundary and is designed to maintain its integrity under all normal and off-normal conditions, e.g., the most intense VDE's and seismic events. The shielding blanket consists of modules connected to a toroidal backplate by flexible connectors which allow differential displacements due to temperature differences. Breeding blanket modules replace the shield modules for the Enhanced Performance Phase. The divertor is based on a cassette structure which is convenient for remote installation and removal. High heat flux (HHF) components are mechanically attached and can be removed and replaced in the hot cell. Operation of the divertor is based on achieving partially detached plasma conditions along and near the separatrix. Nominal heat loads of 5-10 MW/m^2 are expected and these are accommodated by HHF technology developed during the EDA. Disruptions and VDE's can lead to melting of the first wall armour but no damage to the underlying structure. Stresses in the main structural components remain within allowables for all postulated disruption and seismic events.

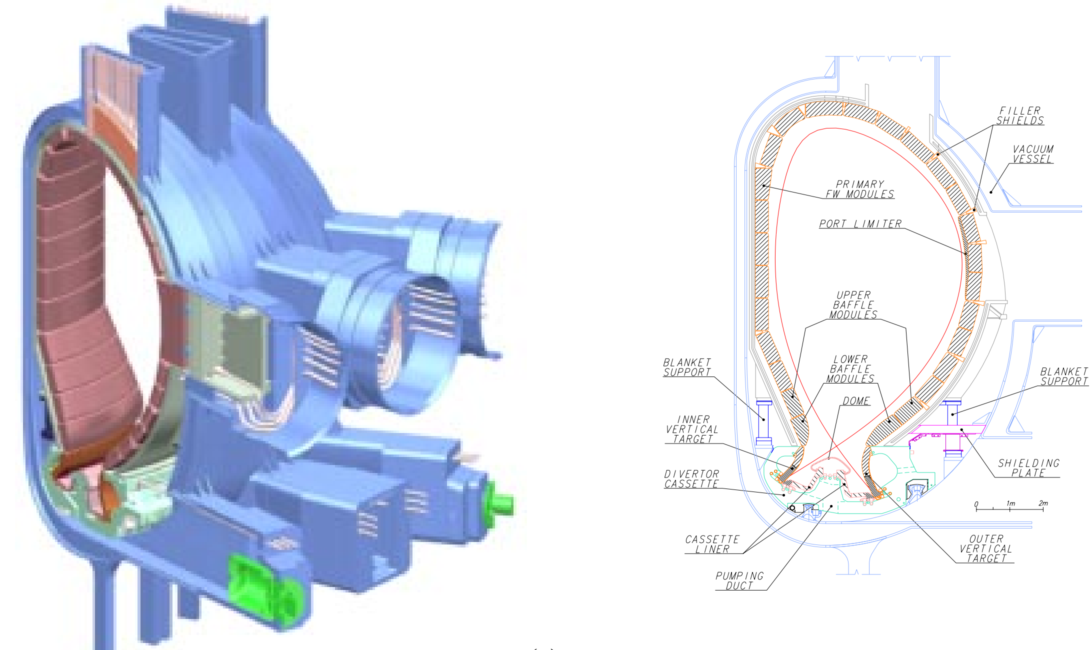
1. INTRODUCTION

The large currents, fields, stored energy and fluxes in ITER present many challenges to the design of the in-vessel systems. For example, during normal operation the divertor is required to dissipate a heat flux which upstream in the scrape off layer approaches 300 MW/m^2, while during off-normal events such as VDE's and disruptions, the vessel and its supports must accommodate a net vertical force F_v up to 150 MN and the first wall and divertor plates must withstand transient thermal loads of up up to 100 MJ/m^2 over durations of 3-300 msecs. The solution to such problems in the design of in-vessel components required a mix of innovative engineering approaches, the development of advanced technology, and detailed physics understanding and analysis of underlying phenomena. The intent of this paper is to review the main features of the design of the ITER in-vessel systems and to describe the nominal and limiting thermal and mechanical loads expected to be applied to these systems, and their consequences. A principle conclusion to be drawn is that the design developed by the JCT and Home Teams during the EDA for the main in-vessel components robustly meets the very demanding requirements imposed by ITER's normal and off-normal operating conditions.

2. DESIGN DESCRIPTION

The main features of the in-vessel components of the FDR design [1] are illustrated in Fig 1. The vacuum boundary is formed by a double-walled vacuum vessel which is supported by the TF coil structure and which contains the blanket and divertor systems. The shield blanket is composed of 739 modules, typically 1x2x0.35 m^3 and weighing 4 t, attached to a 160 mm thick backplate which, like the vessel, is a double-walled toroidal shell. The shielding modules are water cooled steel blocks, which are effective in removing energy from the 14 MeV neutrons, with a water cooled copper mat bonded to the surface of the modules on the plasma side and protected from interaction with the plasma by an armour material. The latter is beryllium for most of the modules, except those identified in the figure as the lower baffle for which the armour is tungsten. The blanket backplate is supported on the vessel by means of supports fabricated from laminated plates which allow differential radial displacements between vessel and blanket system. The divertor consists of 60 tapered steel cassettes whose plasma facing surfaces are protected from the plasma by demountable high-heat-flux (HHF) components. The latter are composite structures of steel, for structural support, lined on the plasma side with a water cooled copper heat sink and protected from the plasma by armour. CFC is used as armour for the lower vertical targets, the components receiving the highest heat flux, and tungsten elsewhere. The 20 equatorial ports are used for neutral beam and RF heating and current drive (8), test blanket modules (4), diagnostics (4) and remote handling (RH) equipment (4, shared with limiter(s) and additional diagnostics).

***For the ITER Joint Central Team and Home Teams**



(a) (b)

Fig 1. *a) Isometric view of vessel showing blanket and divertor. b) Cross-sectional view.*

The structural material for all in-vessel components is SS, generally 316 L(N) (ITER Grade - 0.06 to 0.08% N and ‹ 10 wppm B). The in-vessel components are cooled by water. The inlet temperature for the vessel is 100 ˚C while for the blanket and divertor it is 140˚ C. Design temperature rises are 9 ˚C for the vessel, 50 ˚C for the blanket and 40 ˚C for the divertor.

The **vessel** is a double walled structure fabricated from two toroidal shells joined by poloidal stiffening ribs [2]. The shells and ribs are 40 - 60 mm in thickness, and the total thickness of the vessel ranges from 0.37 to 0.79 m. Water cooled shielding plates are fixed between the walls and, in the region from 12 to 3 o'clock, ferromagnetic material is used in order to reduce the peak field ripple at the plasma edge to < 1%. A high boron content in the 304 SS shield plates improves shielding efficiency. The scale is impressive, 9 m x15 m in poloidal cross section, and the tolerances that can be obtained in a welded fabrication are critical to the assembly. A full scale 18˚ sector, shown in Fig 2, was fabricated by the Japanese Home Team as part of the L 3 Project [3]. The sector was fabricated from poloidal pieces into two half sectors by different companies and the obtained tolerances of ~ 5 mm in width and height of the final assembly are remarkably small. A combination of TIG, MAG and e-beam welding techniques, including a novel through-wall e-beam method, were used and the experience gained in constructing the model will be valuable in defining the final ITER vessel fabrication method.

The **blanket modules** are attached to the backplate by flexibles supports -- four titanium alloy cylinders which are slotted parallel to the cylindrical axis in order to permit relative displacements between module and backplate [4]. This approach was necessary to accommodate differential thermal and disruption-induced mechanical displacements between module and backplate of order 1 mm. A cutaway view of a module is shown in Fig 3. A single insulated centering pin locates the module on the backplate. Flexible pipes transport cooling water between each module and the 40 mm space between the walls of the backplate which serves as a manifold, and flexible copper connectors provide electrical contact between the modules and backplate. The modules are linked to each other by a system of insulated keys which react the radial torques produced by disruption-induced currents. Each of the four flexible supports is attached to the backplate by an M 45 Inconel bolt. All connections between backplate and module are accessible from the plasma side of the module. The mechanical attachment has been selected to facilitate remote assembly and replacement of the modules. RH access to the modules is by means of tools installed on vehicles which travel on a toroidal rail installed from equatorial ports. The thermal performance of the first wall and the feasibility of the fabrication processes have been validated in the L 4 large project [5]. Viability of the assembly and maintenance approach is being addressed in the L 4 and L 6 [6] Projects.

FIG. 2. *Full scale sector model (JAHT)*

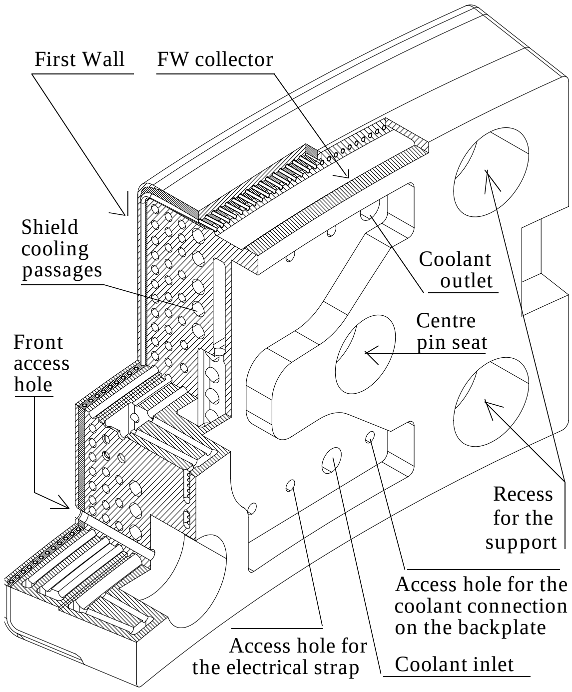


FIG. 3. *Cutaway view of blanket module.*

Tritium breeding blanket modules have been designed to replace the shielding for the Enhanced Performance Phase. The breeding modules are essentially interchangeable with the shielding modules, except for the addition of He purge gas tubes, and operate with the same hydraulic parameters. A tritium breeding ratio of 0.97 has been calculated for the reference design [7], thus exceeding the minimum required TBR of 0.8.

The **divertor** cassettes are 5 m long, 2 m high and taper from 0.5 to 1 m in width [8]. The HHF components are mechanically mounted on the cassettes (Fig. 4) and include the inner and outer divertor targets, liners which are perforated to provide pumping access through the cassette body to regenerating cryopumps located in 16 of the 20 divertor ports, and a dome which prevents neutrals from flowing back to the main plasma near the X-point from the private flux region which typically operates at neutral pressures $\geq$ 0.1 Pa. The cassette assembly weighs ~ 25 t and is mounted on two toroidal rails welded to the vessel. A pivoted attachment is used on the outer rail to allow differential thermal expansion. Repair or replacement of HHF components is carried out by removing the affected cassette through one of 4 divertor ports dedicated to RH access. Cassettes are removed by two vehicles, one which moves them toroidally along the rails, the other which moves them radially outward through the RH ports. Critical aspects of manufacturing a cassette and near-full-size HHF components have been demonstrated in the L 5 Project [9], while the RH approach has been successfully demonstrated with full size and weight components in the L 7 Project [10].

Installation of **port-mounted systems**, e.g., those providing RF H&CD, diagnostics and Test Blanket Modules is based on an integrated approach. In the case of the equatorial ports, these systems are cantilevered from the vacuum vessel closure plate. An example is illustrated in Fig 5 which depicts an ECH antenna array inserted into an equatorial port. In this way, that part of the system protruding from the closure plate to the plasma can be fully assembled and tested prior to insertion into the port thereby simplifying and shortening assembly and maintenance procedures. RF launchers and power trains have been designed to couple 50 MW of RF power for heating and current drive in each of three frequency ranges, namely electron cyclotron, ion cyclotron and lower hybrid [11]. Three equatorial ports are dedicated to neutral beam injectors thereby permitting 50 MW to be injected via NBI at an energy up to 1 MeV.

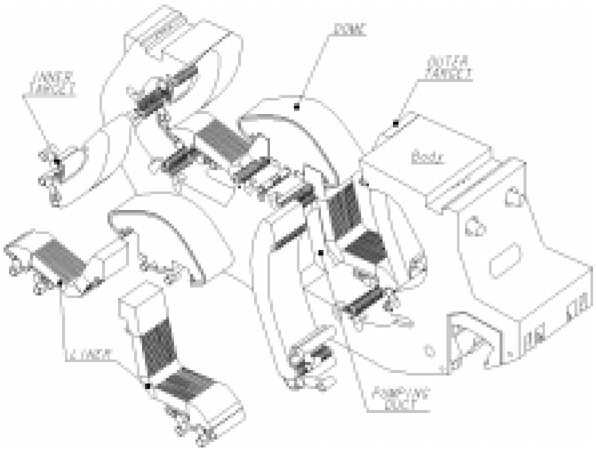


FIG. 4. *Exploded view of divertor cassette*

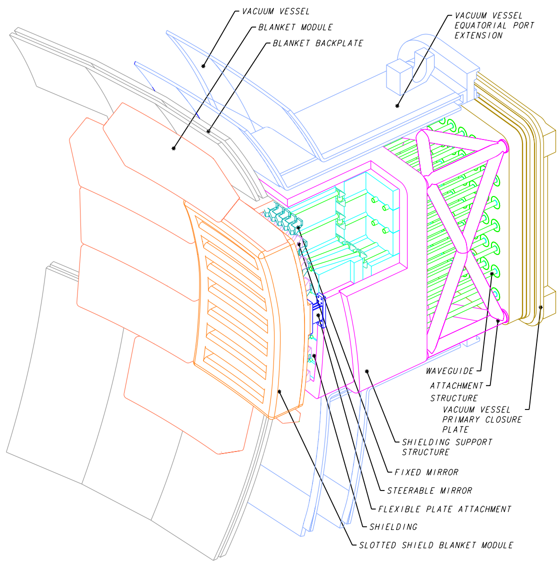


FIG. 5. *Cutaway view of ECH antenna.*

3.1. THERMAL LOADS AND PERFORMANCE

3.1. Normal operation

During a reference ITER discharge, 1500 MW of fusion power is produced of which 300 MW is associated with alpha particles, which must be dissipated by the first wall and divertor, while the remaining 1200 MW is in the form of 14 MeV neutrons whose energy is absorbed mainly by the blanket and divertor cassette. Because the decay length associated with the power deposited by the neutrons in the steel-water shield is rather long, typically 15 - 20 cm per order of magnitude, removal of heat in the shield under steady-state conditions does not present a difficult design challenge. However with the layout of cooling passages indicated in Fig 3, the thermal time constant varies with radial depth and under startup and shutdown conditions a radial temperature gradient exists ($\nabla T \sim 150$ ° K/m) which leads to a bowing of the module. Accommodation of differential thermal deformations was one of the principal requirements which led to the development of the flexible attachment concept.

The 300 MW of alpha power, plus up to 100 MW of auxiliary heating power, flows partly to the first wall of the blanket modules, mostly in the form of bremsstrahlung (~ 100 MW) and radiation from impurities, with the remainder flowing to the divertor. The exact split between first wall and divertor can not be predicted, and in fact it is desirable for the machine operator to be able to control it, and therefore the approach taken is to allow 75% of the power (300 MW) to be dissipated either on the first wall or in the divertor. Averaged over the first wall, 300 MW corresponds to a power density of 0.25 MW/m^2. The design requirement of 0.5 MW/m^2 allows for a peaking factor of 2 and is easily accommodated by the first wall design [5].

Practical solutions to handling up to 300 MW entering the divertor channels require that the divertor operate in a partially detached mode, where power and momentum entering the divertor channel near the separatrix are removed by radiation and charge exchange through interaction with neutrals and impurities before contacting the target. Partially detached divertor regimes are routinely observed in all divertor experiments and have been shown to be a reliable method of reducing the peak power flow on the divertor target plates, typically by a factor of 5 relative to what it would be in attached regimes. In order to promote the conditions for detachment, a vertical target design has been adopted in which recycling neutrals are directed toward the separatrix and private region. A generous fraction of in-vessel space has also been allocated to the divertor in order to permit a relatively long divertor leg, which is necessary to permit sufficient interaction between neutrals and plasma to remove most of the plasma power and momentum near the separatrix. (See Ref. [12] for a review of early detachment studies, and Refs. [13-15] to trace the main steps in the evolution of the ITER divertor design).

Detached plasmas allow neutrals to escape from the recycling region (or recombining region) near the target plates. On one hand this is beneficial since it fosters the buildup of high pressure in the private flux region, allowing efficient particle removal by the pumps, but on the other it could be deleterious in that it could allow neutrals to flow into the main chamber between the first wall and SOL. Fortunately the vertical target geometry naturally provides a solution: detachment occurs mainly near the separatrix, thus allowing the private zone to fill with neutrals, while further from the separatrix it remains attached, thus plugging neutrals from escaping to the main chamber.

Modeling of the complex physical processes taking place in the divertor has made rapid progress during the EDA and success in comparing results with experiments gives confidence in using codes to aid in ITER's divertor design and predicting its operating regime. The main code used to support ITER divertor design is B2-Eirene. This code predicts target heat loads in the range 5 - 10 MW/m^2 under nominal ITER conditions. A series of runs designed to investigate the divertor target heat load when a number of input parameters were varied about their reference values have also been carried out [16]. The variations include: 1) reducing edge transport coefficients, 2) lowering upstream density, 3) reducing chemical sputtering from the carbon target, 4) modifying the geometry by changing the length of the divertor dome, and 5) neglecting carbon radiation (Neon only). In cases 1) and 2) the peak heat flux rises to 13 MW/m^2 while in the other cases the target heat load remained within 5 - 10 MW/m^2. Throughout these runs the He neutral pressure was $\geq 1.0 \times 10^{-2}$ Pa, sufficient to remove the helium at the required rate of 2 Pa·m^3/s. Although the peak target heat load decreased with decreasing dome length the helium partial pressure also decreased and became marginal at the shortest dome length, thus indicating that the dome plays a useful role in the pumping of helium.

Excellent progress has been made in developing practical FW configurations for the three candidate PFC armours, Be, C and W [8,9]. Mockups have been fabricated and tested with each material, and the feasibility of using them as armours on plasma facing components capable of reaching steady-state conditions up to 10 MW/m^2 for ~1000 cycles has been established. Although any of the three materials could be used for the first wall, Be has been selected on the basis of its low Z, and absence of chemical sputtering and codeposition with tritium under ITER conditions. Thus Be is used as the PFM on the blanket modules, including the limiter, except on the lower baffle modules where lifetime considerations dictate the use of W because of its low sputtering coefficient at energies characteristic of charge-exchanged neutrals produced in this region. Solely on the basis of the nominal design heat load (5-10 MW/m^2), all three materials could also qualify as a PFM for the divertor target. However, in view of uncertainties in the control of the partially attached regime, a number of high power transients are foreseen (in 10 % of the discharges and lasting for ~ 10 sec) in which nearly attached conditions resulting in a peak heat flux of ~ 20 MW/m^2 could occur. Based on R&D results, C and possibly W could still meet this requirement with reasonable target thicknesses and projected lifetimes, but Be is ruled out.

3.2. Thermal response to disruptions and VDEs

During the thermal quench phase of a radial disruption, a large fraction of the plasma energy is dumped onto the divertor plates. Assuming the nominal ~ 1 cm scrape-off (at the outside mid-plane) expands to 3 cm during such an event, the energy deposition on the divertor is estimated to be ~ 100 MJ/m^2 occurring over a duration of 1-10 msecs. Substantial melting and evaporation of W targets, and evaporation of C targets, are predicted to occur, although the effects of such disruptions in simulations are mitigated by the formation of a vapour shield. Full power disruptions are expected to occur often in ITER, especially in the BPP phase where 10 % of the full power discharges are assumed to disrupt, and the associated loss of material may be dominant in determining divertor lifetime. In this regard C is superior to W, since for W neither the fractional loss of the melt layer nor the behavior of the resolidified material can be predicted. This is the principal reason that C is selected for the divertor targets in spite of concerns over the trapping of T through the co-deposition process. W remains an alternate, and could be used in a future divertor installation.

Following a thermal quench, a substantial fraction of the plasma current can be converted to runaways through the knock-on avalanche process. Calculations indicate that runaway currents of 15 MA with electron energies in the range 10 MeV could be expected, and that the total kinetic energy in the runaways could reach about 50 MJ [1]. However, a significant fraction of the magnetic energy may also be converted to runaways in which case the energy available for plasma-wall interaction would be significantly higher, e.g., 300 MJ. Considering the scrape-off width of the runaways, and the alignment accuracy of the first wall modules, the maximum energy density delivered on the wall as the plasma impinges on it is estimated to be 50 MJ/m^2.

For the case of a loss-of-control VDE, the plasma drifts vertically and quickly becomes limited. Plasma thermal energy can be transferred to the wall either during the slow VDE drift phase or during a final thermal quench phase. In either case it is estimated that 1 GJ may be lost to the wall over a toroidal band ≥ 0.5 m wide. With a peaking factor of 1.5, the maximum energy density deposited to the wall is then 60 MJ/m^2, similar to that deposited in the runaway case.

In calculating the expected damage to the wall, the duration of plasma-wall contact is an important parameter. Short interaction times are favourable since more of the incident energy is lost to evaporation and less energy is then available for transport to the armour joint and coolant. In modeling the plasma-wall interaction, the available energy is assumed to be released over a duration of 300 msec, a time which is considered to be a reasonable upper bound since during this time the plasma drifts vertically and the plasma-wall contact area moves out of the 0.5 m wide interaction region.

With these assumptions, the total melted and evaporated thickness for a 10 mm Be first wall is calculated to be 1.5 - 2 mm [17]. About 0.7 mm is evaporated in the case of a VDE but considerably less in the case of runaways. Although the incident energy density is similar for runaways and VDEs, the energy is deposited deeper in the Be by runaways and the incident energy is less effective in evaporating material. Aside from melting and evaporating a thin layer of armour, no other permanent damage to the first wall is expected as temperatures in the heat sink remain well below the melting temperature of copper.

4. MECHANICAL LOADS AND STRESSES

During normal operation the mechanical loads on the in-vessel components are insignificant. Instead, the main loads are due to off-normal events such as TF coil quenches, disruptions and seismic events.

Two classes of disruptions are foreseen in ITER. In the first, heat transport in the plasma increases dramatically and a large fraction (up to 100 %) of the plasma energy is lost within a few msecs mainly to the divertor target (thermal quench). Significant poloidal currents flow in the in-vessel components in response to the loss of diamagnetism. This is followed by a resistive current decay (current quench) and loss of vertical equilibrium. Poloidal currents again flow, this time in response to the loss of paramagnetism. If the current quench is rapid, e.g., 50 - 100 msec, the plasma stays relatively centered in the mid-plane and this is referred to as a radial disruption. For slower current quenches, a significant upward vertical motion occurs (disruption-induced VDE) followed by more rapid current quench which terminates the discharge. Toroidal and poloidal currents are induced in the in-vessel components, particularly those near the top of the vessel.

In the second basic type of disruption a VDE develops as a result of loss of control of the vertical stability. In this case, transport remains low and the plasma initially retains most of its thermal energy while it undergoes vertical motion driven by the unstable shaping field (loss of control VDE). Plasma-wall contact quickly ensues, the plasma becomes limited and shrinks in minor radius causing q to drop. When q drops to a sufficiently low level, taken to be q = 1.5 in the simulations, a thermal quench followed by a current quench occurs and large poloidal and toroidal currents are induced in the in-vessel components in the top or bottom of the vessel, depending on the direction of the initial perturbation.

During a VDE, currents flow through a force free region of cold plasma (halo) outside the main plasma and close through the vessel. These halo currents play a role in supporting the plasma's equilibrium, and their magnitude and toroidal distribution are important in determining the performance limits of the in-vessel components. Halo currents and associated forces have been evaluated for ITER using the axisymmetric MHD codes MAXFEA and TSC. While such codes are useful in the detail and insight they provide, they are limited quantitatively by the fact that halo currents are experimentally found to be non-axisymmetric; consequently an empirical approach has been adopted in which the predicted halo current I_h and toroidal peaking factor TPF for ITER are extrapolated from a VDE data base gathered from existing machines [18]. This leads to the bound

I_h/I_p*TPF ≤ 0.58, which may be conservative as there is a trend toward lower values of I_h/I_p*TPF for larger machines, in particular I_h/I_p*TPF ≤ 0.50 for JET and JT-60U. Also, TSC simulations indicate that the geometry and L/R time of the vessel are important in determining the amplitude of the halo current. Relative to existing machines, toroidal currents induced by vertical plasma motion in ITER should be more effective in the plasma force balance due to the long L/R time of the vessel. Such currents tend to be more axisymmetric than the halo current associated with non-axisymmetric disruptions and this could lead to lower halo current effects.

Indications from JET are that asymmetric VDEs can produce net horizontal forces acting on the in vessel components. In the absence of a more quantitative model, and motivated by the JET observations, the plasma in a maximum intensity disruption in ITER is assumed to undergo a helical perturbation with a peak amplitude of ~ 30 cm. This leads to a net horizontal force acting on the in-vessel components of about 50 MN. In the analysis, this force is further increased by the factor 1.4 to account for dynamic amplification.

For design purposes it is necessary not only to consider limiting cases in terms of severity of the loading conditions, but also the frequency with which various load cases are expected to occur. The severity of radial disruptions is dependent on the effective duration of the current quench; the shorter the quench time, the larger the induced currents and forces. From analysis of the disruption database, a model for the current quench has been proposed in which the plasma magnetic energy is dissipated by radiation. When extrapolated to ITER the fastest current quench time for a full performance plasma is projected to be 50 msec [18] while quench times of 100 msec are expected to be more typical. Nevertheless quench times of 50 msec are assumed to occur in a few percent of the BPP plasmas, and this is frequent enough to constitute a normal operating load condition for application of structural analysis codes.

From the results of TSC VDE simulations, slower disruption times in the final quench phase produce higher halo currents and forces since they allow larger vertical displacements and bring the disrupting plasma into regions of stronger destabilizing field. However, loss of control is expected to occur less frequently than radial disruptions and consequently the worst case VDE is presumed to occur only a few times during the machine lifetime. As a result a modest increase in allowable stresses is permitted by structural codes in carrying out stress analysis for such events.

The mechanical response of in-vessel components to seismic and disruption loads has been investigated using a variety of numerical and analytic methods. The process generally begins with a description of the currents and forces produced by a disruption, using one or more of the codes MAXFEA, TSC, EDDYCUFF and EMAS, and then uses these loads as input to stress analysis codes such as ANSYS and NASTRAN. Load combinations are then defined, e.g., thermal + disruption +seismic, etc., and the stresses resulting from the assumed loads are calculated and compared to limits appropriate to the category assigned to the load or load combination [19,20]. In summary, for all postulated loads and load combinations, including VDE's and disruptions, TF coil quenches, and seismic events, the in-vessel component design is determined to be structurally robust as the maximum stresses in the vessel, blanket and divertor systems stay within allowables.

5. NEUTRONIC PERFORMANCE

The neutronic performance of the ITER design has been extensively analyzed using a combination of 1-, 2- and 3-D neutronic codes [22]. The shielding performance of the in-vessel components, including effects of all penetrations, is found to limit the magnet heating to about half of the required value of 17 kW. Dose rates around most of the ports where hands-on maintenance is expected to be performed is calculated to be low enough (100-200 μSv/h) two weeks after shutdown to permit controlled human access for these repair work activities. He levels in the vessel remain low enough (≤ 1 appm) to permit rewelding during ITER's full experimental lifetime. However, due to streaming through penetrations in the modules, He levels in the backplate could somewhat exceed 1 appm at 1 MW·a/m^2. Near the 1 appm level, rewelding by conventional methods may be problematic and specialized techniques would then be necessary.

6. CONCLUSION

Owing to ITER's unprecedented size and performance, the design of its in-vessel systems is far more challenging than that of the largest fusion experiments now in operation. Solving the problems of removing the bulk heat deposited by neutrons and the surface heat deposited by the alpha particles, both in steady-state and transient conditions, required innovative design approaches, e.g., invention of a flexible attachment concept in the case of the blanket modules and exploitation of detached divertor regimes in the case of the divertor. Support for critical aspects of the design has been provided by a highly successful R&D effort. Worst case disruptions and VDE's have been modeled for ITER using a number of tools and are found to produce the most demanding mechanical loads. The design is found to be structurally robust as all normal and off-normal conditions load conditions lead to calculated stresses within allowable levels.

Acknowledgement. This report is an account of work undertaken within the framework of the ITER EDA Agreement. The views and opinions expressed herein do not necessarily reflect those of the Parties to the ITER EDA Agreement, the IAEA or any agency thereof. Dissemination of the information in these papers is governed by the applicable terms of the ITER EDA Agreement.

REFERENCES

[1] Technical Basis for the ITER Final Design Report, Cost Review and Safety Analysis, FDR0, IAEA/ITER EDA/Documentation Series (1998, to be published).
[2] IOKI, K., et al., "Design and R&D for the ITER vacuum vessel", 20th Symposium on Fusion Technology, September 1998, Marseilles.
[3] NAKAHIRA, M., et al., these Proceedings.
[4] ELIO, F. et al., "Engineering design of the ITER blanket and R&D results", 20th Symposium on Fusion Technology, September 1998, Marseilles.
[5] DAENNER, W., et al., these Proceedings.
[6] KAKUDATE, S., et al., these Proceedings.
[7] FERRARI, M., et al., "ITER reference breeding blanket design", 20th Symposium on Fusion Technology, September 1998, Marseilles.
[8] TIVEY, R., et al., "ITER divertor, design issues and R&D", 20th Symposium on Fusion Technology, September 1998, Marseilles.
[9] ULRICKSON, M., et al., these Proceedings.
[10] MAISONNIER, D., et al., these Proceedings.
[11] MAKOWSKI, M., et al., "Radio Frequency Auxiliary Heating Systems Design in ITER", 20th Symposium on Fusion Technology, September 1998, Marseilles.
[12] MATHEWS, G. F., J. Nucl. Mat., 220-222 (1995) 104
[13] JANESCHITZ, G., et al., J. Nucl. Mat., 220-222 (1995) 73
[14] PARKER, R., et al., J. Nucl. Mat., 241-247 (1995) 1
[15] JANESCHITZ, G., et al., in Fusion Energy 1996 (Proc. 16th Int. Conf. Montreal, 1996), Vol II, IAEA, Vienna (1997) 755
[16] PACHER, H., et al., "Operating window for high divertor radiation in ITER", 13th Int. Conf. on Plasma-Surface Interactions, May 1998, San Diego. See also KUKUSHKIN, A., et al., these proceedings.
[17] RAFFRAY, R., "Effect of runaway electrons and VDE's on ITER first wall", 20th Symposium on Fusion Technology, September 1998, Marseilles.
[18] WESLEY, J., et. al., in Fusion Energy 1996 (Proc. 16th Int. Conf. Montreal, 1996), Vol II, IAEA, Vienna (1997) 971
[19] SANNAZZARO, G., et al., "Seismic analysis of the ITER tokamak", 20th Symposium on Fusion Technology, September 1998, Marseilles.
[20] SANNAZZARO, G., et al., "Structural analysis of the ITER vacuum vessel", 20th Symposium on Fusion Technology, September 1998, Marseilles.
[21] MIKI, N., et al., "VDE/disruption EM analysis for ITER in-vessel components", 20th Symposium on Fusion Technology, September 1998, Marseilles.
[22] SANTORO, R., et al., these Proceedings.