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ABSTRACT

3-D neutronics and shielding analyses have been performed for the divertor region of the ITER interim design. The peak neutron wall loading in the divertor region is 0.6 MW/m^2 at the divertor cassette dome. The total nuclear heating in the 60 divertor cassettes is 102.4 MW. The peak helium production in the VV behind the pumping ducts is 0.5 He appm/FPY implying that rewelding might be feasible. The total nuclear heating in the parts of the TF coils in the divertor region is only 2.1 kW.

I. INTRODUCTION

The design of the International Thermonuclear Experimental Reactor (ITER) has been evolving during the Engineering Design Activity (EDA) phase. The latest interim design[1] has been driven by the need to simplify assembly, maintenance, and manufacturing, and to reduce costs. A main feature of the design is utilizing 20 toroidal field (TF) cased coils. The design has 60 divertor cassettes with vertical targets. Knowledge of nuclear heating and radiation damage levels in the different components of the divertor cassette is essential for proper design analysis. Twenty large divertor ports are utilized for assembly and disassembly of the divertor cassettes and for vacuum pumping. Radiation streaming into these ports can produce excessive heating and damage the TF coils. Reducing nuclear heating and radiation damage in the TF coils to acceptable levels behind the divertor cassettes and adjacent to the large divertor ports is an important shielding issue. Radiation damage to parts of the vacuum vessel (VV) in the divertor region need to be quantified to assess the feasibility of rewelding. Due to the geometrical complexity of the divertor region, three-dimensional (3-D) analyses are required. Two-dimensional discrete ordinate calculations can produce misleading results.

II. 3-D CALCULATIONAL MODEL

3-D neutron-gamma transport calculations have been performed for the divertor region using the continuous energy, coupled neutron-gamma-ray Monte Carlo code MCNP-4A[2] with cross sections based on the FENDL-1 evaluation.[3] The detailed geometrical configuration of the divertor cassette has been modeled for 3-D neutronics calculations based on drawings provided by the Joint Central Team (JCT) for the interim ITER design. The model represents a nine degree toroidal sector of ITER. Hence, it includes one and a half cassettes with the associated 1 cm gaps between adjacent cassettes. The model includes in detail the high heat flux plasma facing components (PFC), the vertical targets, the wings with associated plates, the gas boxes, as well as the central dome and cassette body segments. The 37.5 cm wide and 17.5 cm thick divertor pumping duct at the bottom of each cassette is included in the model. The rails upon which the cassettes move toroidal during maintenance are also included. The cassette model is divided into 32 cells to estimate the spatial variation of nuclear parameters.

The divertor cassette model has been integrated with the general ITER model that includes detailed modeling of the first wall, blanket with associated coolant manifolds and back plates, VV, TF coils,

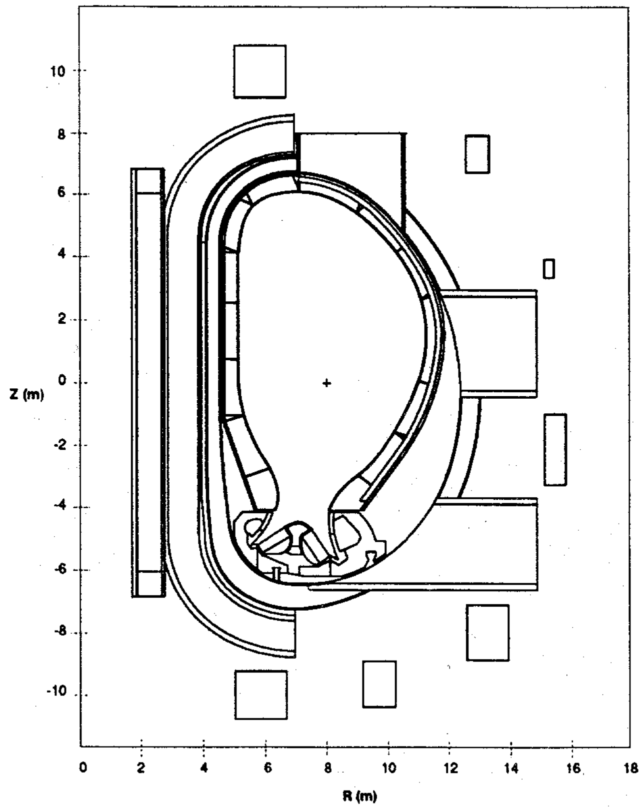

Fig. 1. Vertical cross section through the VV ports.

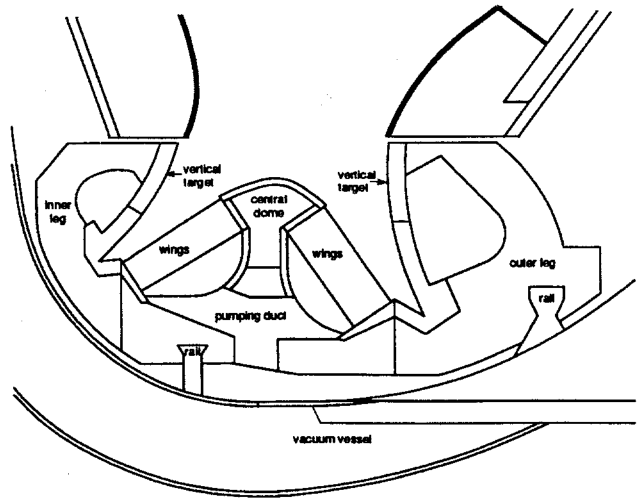

Fig. 2. Vertical cross section of the divertor.

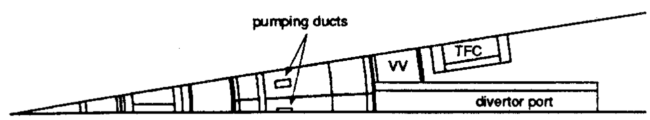

Fig. 3. Horizontal cross section at $z = -6$ m.

central solenoid, and PF coils. All toroidal and poloidal gaps between adjacent blanket modules are included. The major vacuum vessel penetrations are included in the model. The model includes half a TF coil and half a divertor port. The divertor port is 254.5 cm high with a width increasing from 97 cm at the bottom to 176 cm at the top. The port wall is 20 cm thick and is assumed to consist of 80% 316SS and 20% water. No additional shielding is included around the port. The TF coils are segmented to determine the nuclear heating and damage in the parts adjacent to the port resulting from radiation streaming.

Figure 1 shows a vertical cross section through the middle of the vacuum vessel ports. The detailed reactor geometrical modeling is illustrated. Figure 2 shows the detailed model for the divertor cassette. This cross section cuts through a pumping duct at the middle of the cassette. Figure 3 is a horizontal cross section in the middle of the divertor port. The divertor pumping ducts in the divertor cassettes are shown in this figure. Also shown is the part of the TF coil adjacent to the divertor port. A coil case which is about 20 cm thick surrounds the winding pack. Several additional surfaces have been added in the divertor region to allow for utilizing the geometry splitting with Russian Roulette variance reduction techniques employed in MCNP to improve the accuracy of the calculated nuclear responses.

A combination of cones, tori, cylinders, and planes was utilized for accurate modeling of the geometry. A total of 475 surfaces has been used in the model, of which 164 are fourth degree tori. The model employs 417 geometrical cells. The volumes of the different cells and the areas of surfaces of interest have been determined stochastically by ray tracing. In this calculation, all cells are voided and the geometrical model has been sprayed by 10 million particles at random directions. This calculation served also as a means for geometry checking.

A source subroutine has been written to modify MCNP to sample source neutrons from the source distribution in the plasma provided numerically by the San Diego JCT at 1600 mesh points. Surface flux tallies are used to determine the peak radiation effects at the front surfaces of the different components of the divertor cassette, VV and TF coil and cell flux and energy deposition tallies are used to determine the volume averaged parameters and total nuclear heating in the divertor and TF coil. The appropriate material compositions are used in the different cells. The VV consists of two 4 cm thick 316SS plates sandwiching a

TABLE I

Material Composition

Dome PFC	14% W, 13% Cu, 29% SS, 44% water
Cassette body	80% SS, 20% water
Wings	16% W, 79% Cu, 5% water packing fraction: 21% outer, 26% inner
Gas box liners	8% W, 74% Cu, 18% water
Vertical targets	top section: 7% W, 22% Cu, 52% SS, 19% water lower section: 17% C, 20% Cu, 46% SS, 17% water
Rails	100% SS

shielding region made of 60% 316SS and 40% water. The winding pack of the TF coil consists of 43.2% SS, 11.7% Cu, 2.9% Nb_3Sn, 7.4% Bronze, 16.8% liquid He, and 18% insulator (epoxy with 70% R-glass). The material composition used for the divertor cassette is given in Table I. The calculation has been performed using 100,000 source particles yielding statistical uncertainties less than 10% in the calculated nuclear responses at the locations of interest. The results are normalized to the nominal fusion power of 1500 MW. The end of life fluence related radiation effects have been determined for 1 full power year (FPY) of operation.

III. NEUTRON WALL LOADING DISTRIBUTION

The poloidal distribution of the neutron wall loading in the different regions of ITER has been determined using MCNP. In this calculation, only the uncollided neutron current crossing the first wall and the front surface of the divertor cassette is tallied. Two million source particles have been sampled yielding statistical uncertainty less than 0.5%. Surface current tallies have been determined by counting particles crossing the wall. The first wall surface has been segmented into 53 poloidal segments. The exact segment areas were calculated analytically. The plasma facing surface of the divertor cassette has been segmented into 21 poloidal segments to provide the neutron wall loading distribution in the divertor region.

The peak inboard and outboard wall loadings are 0.95 and 1.25 MW/m^2, respectively. The neutron wall loading peaks at vertical locations close to that of the plasma magnetic axis. The average values in the inboard and outboard regions are 0.69 and 1.06 MW/m^2, respectively. Figure 4 gives the poloidal

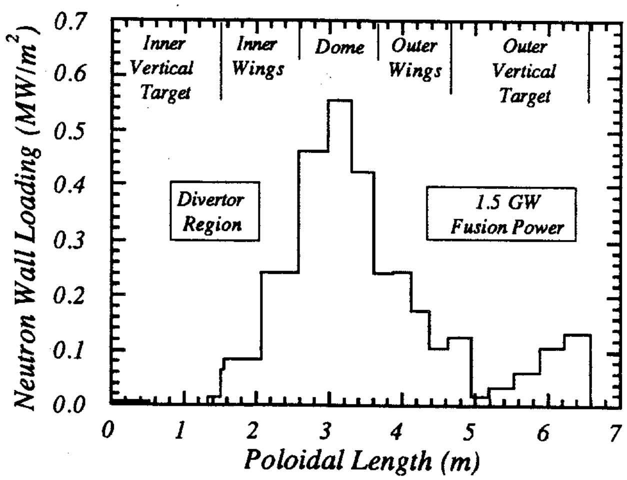

Fig. 4. Wall loading distribution in the divertor.

variation of neutron wall loading in the divertor as a function of toroidal length measured in the counter-clockwise direction from the upper corner of the inner vertical target. The neutron wall loading peaks at 0.56 MW/m^2 in the central dome which has the largest view of the plasma. The average neutron wall loading in the divertor cassette is 0.16 MW/m^2.

IV. NUCLEAR PARAMETERS IN THE DIVERTOR CASSETTE

The neutronics parameters have been calculated in the different components of the divertor cassette. These parameters include nuclear heating, atomic displacement and helium production. The radiation damage was calculated in both stainless steel and copper structures. The volume averaged parameters were determined for 32 segments of the cassette to provide detailed spatial distributions. The peak nuclear responses were also determined at the front surfaces of the cassette components. These are given in Table II. The largest heating and damage occurs in the dome PFC. The vertical targets and wings experience moderate levels of heating and damage with these values dropping rapidly as one moves deeper in the cassette body. For example, nuclear heating in the outer rail is only 2.9×10^{-4} W/cm^3 and the dpa and helium production values are 6.3×10^{-5} dpa/FPY and 1.3×10^{-3} He appm/FPY. The nuclear parameters in the inboard side of the cassette are lower than those in the outboard side that has a larger view of the plasma. Atomic displacements in Cu are slightly higher than in SS while helium production is much lower because of the higher threshold energy of nuclear reactions producing helium. The total nuclear heating in the 60 divertor cassettes is 102.4 MW. The

TABLE II

Peak Nuclear Responses in the Divertor Cassette

	Power Density (W/cm^3)	dpa (dpa/FPY)		He Production (appm/FPY)	
Dome PFC	10.75	4.69 SS	5.04 Cu	89.16 SS	55.39 Cu
Central Dome Body	5.39	2.28 SS		40.22 SS	
Outer Wings	7.34	2.98 Cu		31.26 Cu	
Inner Wings	5.79	2.17 Cu		20.40 Cu	
Outer Vertical Target	5.46	2.60 SS	2.77 Cu	46.54 SS	26.29 Cu
Inner Vertical Target	3.72	0.95 SS	0.98 Cu	18.31 SS	11.95 Cu

TABLE III

Peak Nuclear Responses in the VV Behind the Divertor Cassette

	Power Density (W/cm3)	dpa (dpa/FPY)	He Production (appm/FPY)
Behind pumping duct	0.034	0.025	0.48
Behind cassette body	0.018	0.005	0.09

major contributors are the outer vertical target with 23.1 MW and the dome PFC with 19.7 MW.

V. NUCLEAR PARAMETERS IN VV

Streaming through the pumping ducts in the bottom of the cassette can result in heating and damage hot spots in the VV. Previous designs were analyzed and recommendations regarding duct configuration and size were made. In this design, the pumping ducts are inclined towards the outer and inner divertor legs such that the VV behind them does not see any direct neutrons from the plasma. This helps to reduce the He production which is critical for rewelding. Only low energy secondary neutrons stream through the pumping ducts. The VV parameters were determined for toroidal locations away from and behind the ducts by segmenting the front surface of the VV below the cassette. The results are given in Table III. A peaking factor of about 5 results from streaming through the ducts. The peak helium production value indicates that rewelding of the VV behind the pumping ducts might be feasible. Since these areas of relatively high helium production are very small, the design of the VV might allow locating the welds away from streaming paths if these values are considered too high for rewelding. Another area of concern is along the divertor port where relatively high damage is expected due to neutron streaming. The largest damage occurs at the location where the port wall joins with the front VV wall and drops as one moves along the port away from the plasma chamber. The peak helium production is 0.036 appm/FPY and drops to 0.005 appm/FPY at locations adjacent to the back of the TF coil. It is clear from these results that rewelding of the divertor port is feasible.

VI. MAGNET RADIATION EFFECTS

The peak magnet radiation effects have been calculated in segments of the TF coils adjacent to the divertor port, between the divertor port and the horizontal port, and behind the divertor cassette below the divertor port. The radiation effects at the front surface of the TF coil adjacent to the divertor port are about an order of magnitude higher than those above and below the port. Table IV gives the magnet radiation effects in the part of the TF coil adjacent to the divertor port. The results are given at the front and side surfaces of the coil. The radiation effects are higher at the side surface due to streaming. The calculated radiation effects are much lower than the radiation limits proposed for ITER of 2 and 1 kW/m^3 for the coil case and winding pack power densities, respectively, and 10^9 rads for the end-of-life insulator dose.[4] Although no limits were specified for fast neutron fluence and Cu dpa in the EDA, the results are about three orders of magnitude lower than the limits of 10^{19} n/cm^2 and 6×10^{-3} dpa used in the CDA.[5] It is clear that the sides of the TF coils are well protected from radiation streaming into the divertor ports.

The total nuclear heating in the parts of the 20 TF coils in the divertor region is given in Table V. The results are given for the front, back, and side coil

TABLE IV

Magnet Radiation Effects at Front and Side Surfaces of the TF Coils Adjacent to the Divertor Port

	Front Surface	Side Surface
Coil case power density (kW/m^3)	0.080	0.126
Winding pack power density (kW/m^3)	5.12×10^{-3}	6.38×10^{-3}
Insulator dose (Rad/FPY)	4.46×10^{6}	6.78×10^{6}
Fast neutron fluence (n/cm^2 per FPY)	6.27×10^{15}	1.10×10^{16}
Copper dpa (dpa/FPY)	2.38×10^{-6}	4.46×10^{-6}

cases as well as the winding pack. The total nuclear heating is 2.081 kW with 1.575 kW contributed by the parts adjacent to the divertor port. The statistical uncertainty is less than 5%. The heating in the part below the port is about a factor of 5 more than that in the part above the port. Notice that contribution from streaming in the horizontal port is not included. The total nuclear heating in the TF coils should not exceed 17 kW. It is essential to determine the additional heating in the other parts of the coils and add them to the contribution from the divertor region to determine whether the total heating limit can be satisfied.

VII. SUMMARY AND CONCLUSIONS

3-D neutronics and shielding analyses have been performed for the divertor region using a detailed model for the ITER interim design. The model includes the first wall, blanket modules, back plate, coolant manifolds, VV with major ports, divertor cassettes, TF coils and PF coils. The peak inboard and outboard wall loadings are 0.95 and 1.25 MW/m^2, respectively, for the nominal 1500 MW fusion power. The peak neutron wall loading in the divertor region is 0.56 MW/m^2 at the divertor cassette dome.

The spatial distribution of the nuclear parameters has been determined in the divertor cassette. The total nuclear heating in the 60 divertor cassettes is 102.4 MW. The peak helium production in the VV behind the pumping ducts is 0.5 He appm/FPY implying that rewelding might be feasible. In addition, helium production in the divertor port wall is less than 0.04 He appm/FPY. Radiation streaming into the divertor ports results in peak end-of-life insulator dose, neutron fluence, and Cu damage well below the design limits. The total nuclear heating in the parts of the TF coils in the divertor region is only 2.1 kW. Heating in the remainder of the coils including contribution from streaming in other major ports needs to be calculated to determine the total magnet heating.

TABLE V

Total Nuclear Heating (kW) in the TF Coils in the Divertor Region

	Above Port	Adjacent to Port	Below Port
Inner case	0.031	0.409	0.036
Outer case	0.005	0.114	0.075
Side case	0.035	0.868	0.249
Winding pack	0.009	0.185	0.065
Total	0.080	1.576	0.425

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