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Analysis on damage to TF coils of a compact reversed shear tokamak CREST

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Abstract

CREST is a conceptual tokamak reactor design with high β plasma, high thermal efficiency, competitive cost and water-cooled ferritic steel components. Some of its parameters are similar to those of the ITER advanced mode plasma. In this manuscript, the specific issues and analysis on damage to TF coils of CREST were carried out based on the three-dimensional model of the CREST with the widely used code MCNP/4C and the IAEA latest released FENDL/2.1 data library. Damage to some specific regions of the TF coils near large openings and at the inboard mid-plane are calculated and analyzed. Parameters such as the distributions of nuclear heat density, fast neutron flux, dose rate to the epoxy insulator, and peak displacement dose to Cu conductor for the TF coil near these regions were calculated and analyzed. The shield thicknesses at these regions are optimized.
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1. Introduction

The compact reversed shear tokamak CREST [1,2] is a conceptual tokamak reactor with high β plasma, high thermal efficiency, competitive cost, solid pebble bed blanket and water/steam-cooled ferritic steel components. Some of its plasma parameters are similar to those of the ITER advanced mode plasma [3]. The design target for CREST is net electric power 1 GW with 2.97 GW fusion power and to operate for over 30 years with an availability of 75%. It has 14 large TF coils and 14 large horizontal ports. The oxide dispersion strengthened ferritic steel (ODS-FS) F82H [4] is chosen as the primary structure material. Water is used as coolant to cool the first wall (FW) and then it is changed to steam when reaching the blanket so as to get a high efficiency for the system. The solid pebble bed of Li_2ZrO_3 (50% ^6Li enrichment) and Be was chosen as the breeder and neutron multiplier, respectively, to achieve tritium self-sufficiency. Main parameters, materials and the radial arrangement for the inboard and outboard blankets of CREST can be found in Ref. [2].

A lot of work has been done on CREST design [1,2,5] such as the design and optimization of the core plasma parameters, estimation on cost of

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electricity (COE), neutronics, hydrodynamics and safety, etc. Most of the neutronics issues and analysis on CREST were included in Ref. [2]. In this manuscript, the specific issues and analysis of damage to TF coils of CREST due to large openings were carried out based on the three-dimensional (3D) model with the widely used transport code MCNP/4C [6] and the IAEA latest released fusion evaluated nuclear data library FENDL/2.1 [7]. Damage to some specific regions of the TF coils are calculated and analyzed. These regions are near the large openings, such as the neutral beam injecting (NBI) ports and the divertor cassette, or locate at the inboard mid-plane region. It is known that the void regions in the middle of the NBI port and around the divertor cassette let neutron stream through, and components near the inboard mid-plane region receive much more neutron fluence and irradiation, so TF coils near these large openings or at the inboard mid-plane will face higher neutron flux and then get much more damage. Neutron damages to TF coils around those regions have been evaluated and the shield thicknesses of these regions have been optimized.

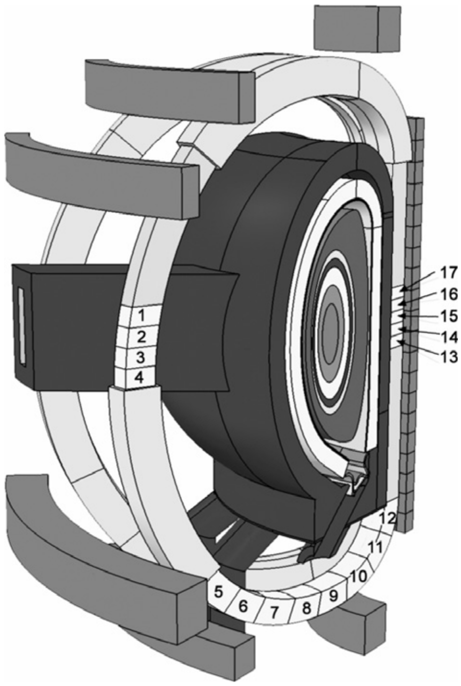

Fig. 1. 1/7 Sector of 3D CREST model.

2. Models

The first 3D neutronics calculations and analyses for CREST were introduced in Ref. [2], in which the 1/14 sector 3D model was established and used for neutronics calculations and analyses. The first model did not include the divertor cassette and NBI ports because no detail design was available at that time, while the new 3D model for CREST used in this manuscript includes the divertor cassette and NBI ports in order to perform the neutronics calculations and analyses for radiation damage to TF coils of CREST due to large openings at these regions and at the inboard mid-plane. It is a 1/7 sector of the 3D D-shape model for CREST (shown in Fig. 1). The NBI port and divertor cassette included in the 3D 1/7 sector model are shown in Fig. 2. Fig. 3 shows some details of the divertor cassette. The plasma core center is located at the position $PZ = 0$, the mid-plane of the NBI ports and TF coil inboard legs are at $PZ = 31$ cm and $PZ = -70$ cm, respectively. The dimensions of TF coils are 80 cm thick at the radial direction and 40–80 cm thick at the toroidal direction, respectively. Those for NBI ports are 50 cm in width and 200 cm in height with 50 cm thick concrete shield at each side of the ports. The 3D model was

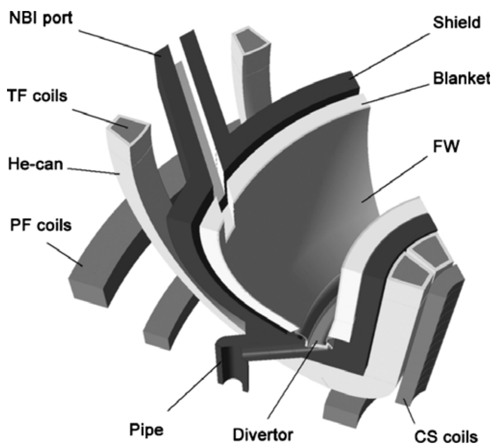

Fig. 2. NBI port and divertor cassette in CREST model.

produced by using the interface code MCAM [8]. Reflective boundary conditions were used in the model to make it equivalent to the full geometry. All essential components comprising the power core of CREST are included in the 3D models, i.e. the FW, blanket, shield, divertor cassette and magnet including TF coils, PF coils and central solenoid (CS). The divertor cassette included in the model is based on the ITER divertor design [9] and is within a factor of 0.6 of the size of the ITER's

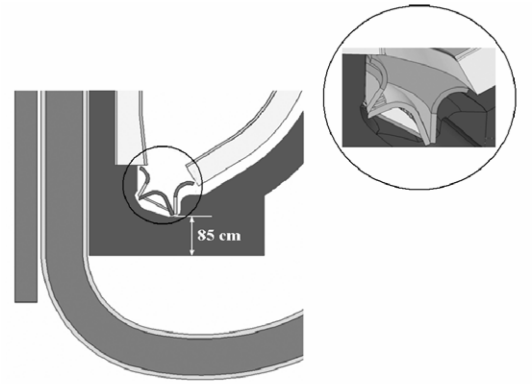

Fig. 3. Divertor cassette in CREST model.

divertor cassette. The materials used for divertor cassette are also the same as ITER divertor design, and this assumption for the calculations and analyses on damage to TF coils is reasonable. The minimum shield thickness at this region is 85 cm as shown in Fig. 3. The cryostat was not included in the 3D model analysis as mentioned in Ref. [2]. Material homogenization was used in each of the FW and blanket regions in the 3D model.

The neutron source used in the calculations was the sum of six nested plasma regions with different neutron production probabilities, as done in ITER analysis [9], according to one-dimensional system code calculation by FUSAC [10]. The probabilities of neutron production in the six regions are 0.216, 0.276, 0.296, 0.104, 0.070 and 0.038 from the inner region to the outer region, respectively.

3. Results and analyses

The 3D neutronics calculations for some key nuclear parameters, such as the distributions of nuclear heat density, fast neutron flux ($E > 0.1$ MeV), dose rate to the insulator epoxy, and peak displacement per atom (dpa) to Cu conductor for the TF coils of CREST near the large openings, i.e. NBI port and divertor cassette regions (N and D for short, respectively) and near the inboard mid-plane regions (M for short) were performed for CREST. These parameters could be used to estimate the damage to TF coils and can be used to optimize the shield thickness at these regions. The results are compared with the criterion of the ITER case to make sure that they are below the limitations and that the TF coils are well shielded and the device is safe during operation. The shield thicknesses were optimized according to the results for TF coils.

Fig. 1 shows some cells of the TF coils near N, D and M which will probably give the maximum damage to TF coil nearby and should be examined in details. Cells 1–4 are near N, each is 50 cm high and they are located from the position PZ = – 70 cm to PZ = 130 cm. Cells 5–12 are near D, and are ~100 cm long for each cell along the TF coil. Cells 13–17 locate at M and each is with a height of 35 cm from the position PZ = –105 cm to PZ = 70 cm. The dose rates to insulator epoxy are calculated with zones with 0.1 cm thickness at the plasma facing surface of TF coils near N, D and M. The radial and average toroidal thicknesses for TF coils are 100 cm and 80 cm, respectively, including the helium can.

The calculation results are shown in Tables 1–3 assuming an operation time of 30 years with an availability of 75%. All the results are with statistical errors less than 10%. Table 1 is for the average damage results in the individual cells near those regions. It is clear that the damages near M were the lowest compared with the high values of damages near N and D. Table 2 shows results for the front 7 cm thick cells facing plasma in the corresponding cells near N and D as in Table 1 in order to get the relatively high values for those damages. The results for damages listed in Table 2 are all higher than those listed in Table 1 for the corresponding cells and this is reasonable. Special analysis was then paid to Cell 9, according to Table 2, the cell with the highest values for the damages to TF coil, and a much smaller cell with 2 cm thick and 30 cm along the coil at the front (i.e. facing plasma) and middle (i.e. along the coil) region was used to calculate the peak values of damage to TF coil. The calculation results for three different thicknesses of Cell 9, i.e. 80 cm, 7 cm and 2 cm, respectively, are summarized in Table 3.

3.1. Nuclear heat density

From Table 1, the average nuclear heat densities near N are roughly the same, i.e. ~0.04 mW/cm^3, while those near D range from 0.01 mW/cm^3 to 0.07 mW/cm^3, and those are almost the same at M for ~0.02 mW/cm^3. The highest average heat density locates at Cell 9 near D from Tables 1 and 2 and the peak value for it is calculated as 0.325 mW/cm^3. It is lower than the ITER limit for nuclear heat density in TF coils i.e. 1 mW/cm^3

Table 1
Results of average damage to TF coils

Cell no.		Fast neutron flux (n/cm^2 s)	Fast neutron fluence (n/cm^2)	Heat density (mW/cm^3)	dpa to Cu (dpa)
Near NBI port	1	3.04×10^9	2.16×10^{18}	0.036	9.13×10^{-4}
	2	3.14×10^9	2.23×10^{18}	0.037	9.19×10^{-4}
	3	2.97×10^9	2.11×10^{18}	0.041	8.74×10^{-4}
	4	2.51×10^9	1.78×10^{18}	0.033	7.63×10^{-4}
Near divertor cassette	5	5.68×10^8	4.93×10^{17}	0.010	2.06×10^{-4}
	6	9.96×10^8	7.90×10^{17}	0.016	3.31×10^{-4}
	7	2.12×10^9	1.72×10^{18}	0.032	7.08×10^{-4}
	8	4.25×10^9	2.91×10^{18}	0.059	1.40×10^{-3}
	9	$\mathbf{4.99 \times 10^9}$	$\mathbf{3.46 \times 10^{18}}$	**0.070**	$\mathbf{1.68 \times 10^{-3}}$
	10	3.72×10^9	2.64×10^{18}	0.048	1.31×10^{-3}
	11	2.15×10^9	1.53×10^{18}	0.028	7.11×10^{-4}
	12	9.55×10^8	6.78×10^{17}	0.016	3.17×10^{-4}
Near inboard mid-plane	13	1.19×10^9	8.45×10^{17}	0.020	4.45×10^{-4}
	14	1.51×10^9	1.07×10^{18}	0.018	5.28×10^{-4}
	15	1.72×10^9	1.22×10^{18}	0.022	5.65×10^{-4}
	16	1.94×10^9	1.38×10^{18}	0.020	6.11×10^{-4}
	17	1.76×10^9	1.25×10^{18}	0.019	5.66×10^{-4}

Table 2
Results of damage for the front 7 cm thick cells in TF coils

Cell no.		Fast neutron flux (n/cm^2 s)	Fast neutron fluence (n/cm^2)	Heat density (mW/cm^3)	dpa to Cu (dpa)
Near NBI port	1	9.22×10^9	6.54×10^{18}	0.118	2.94×10^{-3}
	2	9.59×10^9	6.78×10^{18}	0.122	3.05×10^{-3}
	3	8.90×10^9	6.31×10^{18}	0.129	2.80×10^{-3}
	4	7.18×10^9	5.10×10^{18}	0.105	2.61×10^{-3}
Near divertor cassette	5	1.21×10^9	8.59×10^{17}	0.015	8.20×10^{-4}
	6	2.60×10^9	1.85×10^{18}	0.035	4.04×10^{-4}
	7	6.51×10^9	4.62×10^{18}	0.102	2.27×10^{-3}
	8	1.37×10^{10}	9.71×10^{18}	0.208	$4.58 x 10^{-3}$
	9	$\mathbf{1.65 \times 10^{10}}$	$\mathbf{1.17 \times 10^{19}}$	**0.259**	$\mathbf{5.90 \times 10^{-3}}$
	10	1.26×10^{10}	8.95×10^{18}	0.194	4.25×10^{-3}
	11	8.25×10^9	5.85×10^{18}	0.122	2.57×10^{-3}
	12	4.16×10^9	2.95×10^{18}	0.061	1.42×10^{-3}

Table 3
Damage results at the most severe region in TF coil (i.e. in Cell 9, near the divertor cassette region in Fig. 1)

Cell thickness[a] (cm)	Fast neutron flux (n/cm^2 s)	Fast neutron fluence (n/cm^2)	Heat density (mW/cm^3)	dpa to Cu (dpa)
80	4.99×10^9	3.46×10^{18}	0.070	1.68×10^{-3}
7	1.65×10^{10}	1.17×10^{19}	0.259	5.90×10^{-3}
2	1.94×10^{10}	1.38×10^{19}	0.325	6.66×10^{-3}

[a] In radial direction.

[11]. The total nuclear heat for TF coils is 4.05 kW and less than the ITER limit value of 14 kW as well [9]. So the shield provides good protection to TF coils of CREST from this point of view.

3.2. Fast neutron fluence

The most crucial radiation load to the TF coil is the fast neutron fluence to the superconductor. Suitable design radiation limits for the super-conducting TF coils were defined to be 10^{19} n/cm^2 by ITER [9] for fast neutron fluence.

It is clear from Table 1 that the maximum average fast neutron fluence near N appears above the middle region of the void region in the port. The trend of fast neutron flux and fluence is different from that of the nuclear heat density at the same region as shown in the table, because the nuclear heat is due to both neutrons and γ rays.

The maximum average fast neutron fluence in TF coil near M is 1.38×10^{18} n/cm^2 and below the limit for ITER.

The peak fast neutron fluence near D was calculated as 1.38×10^{19} n/cm^2 and higher than the radiation design limits for ITER. So an additional shield of 3 cm would needed near D according to this result.

3.3. Dose rate to the insulator

The radiation dose absorbed by the epoxy resin insulator is an important parameter to evaluate the shielding performance to TF coils. The radiation dose design limit of integral epoxy for ITER is 10^7 Gy [9]. Since the average fast neutron flux near D is higher than those near D and M, only the dose rate distribution of the insulator layer around D is evaluated and its distribution is similar to that of heat generation rate near D. The peak value is 9.9×10^6 Gy for an operation time of 30 years and an availability of 75%. It is roughly the same as the limit for ITER case.

3.4. Peak dpa to Cu conductor

As is well-known, the atomic displacement level is the best measure of expected neutron-induced effects in fusion reactor materials such as the structure materials and the TF coils. The ITER maximum dpa limit for Cu is 5×10^{-4} dpa [12]. From Table 1, the maximum average damage levels near N and M and D are 9.19×10^{-3} dpa and 6.11×10^{-4} dpa and 1.68×10^{-3} dpa, respectively, for an operation time of 30yrs and an availability of 75%. The peak damage level near D is calculated as 6.66×10^{-3} dpa. The peak value is about 4 times of the average damages for Cell 9 near D. So another ~33 cm thick shields near D should be added to reduce the radiation damage to Cu for TF coils near the large openings. When using the same factor 4 for estimation of shield thickness at N, another ~21 cm thick shield should be added, and this should be reasonable and safety consideration. And there is enough space for these as shown in Fig. 3. Another ~6 cm thick shield near M should be added as well when using the same factor 4 for the estimation of shield thickness at M. While there is not enough space for this, so the shield materials and their volume percentages in this region or the structural design, etc. of the machine should be optimized from the point of view of neutronics and hydrodynamics, etc.

4. Conclusions

Detailed analysis on radiation damage to TF coils near large openings (i.e. the divertor cassette and NBI port) and the inboard mid-plane of CREST was carried out. Some key nuclear parameters such as nuclear heat density, fast neutron flux, dose rate to insulator epoxy, dpa to Cu conductor, etc. were performed with the code MCNP/4C and the data library FENDL/2.1.

The damages near the inboard mid-plane were the lowest compared with the high values of damages near the NBI port and divertor cassette.

The peak values for nuclear heat density, fast neutron fluence, dose rate to insulator epoxy and dpa to Cu for TF coils locate at the divertor cassette region and are 0.325 mW/cm^3, 1.38×10^{19} n/cm^2, 9.9×10^6 Gy and 6.66×10^{-3} dpa, respectively. The total nuclear heat for TF coils is 4.05 kW.

Another ~21 cm and ~33 cm thick shields should be added at the NBI port and the divertor cassette regions, respectively, in order to reduce radiation damage to TF coils near the large openings below the ITER limits.

And another ~6 cm thick shield near the inboard mid-plane region should be added as well, but there is not enough space for this. So the shield materials and their volume percentages in this region or the structural design, etc. should be optimized from the point of view of neutronics and hydrodynamics, etc.

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