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Equatorial electron cyclotron port plug neutronic analyses for the EU DEMO

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Within the Power Plant Physics and Technology (PPPT) programme in the EUROfusion Consortium design activities are currently in progress for the development of a DEMOnstration Fusion Power Plant (DEMO). The design of the machine and the integration of in-vessel components require neutronics analyses to verify the tritium self-sufficiency, the shielding requirements, and the structural integrity of its components. In particular, the effect of penetrations in the blanket and in equatorial port plug introduced by the electron cyclotron (EC) heating system, namely due to openings for the antenna waveguides, were analyzed. In this study three-dimensional MCNP calculations were conducted for the pre-conceptual designs of the EC port plugs and shielding optimization were performed in order to ensure that the DEMO design limits are not exceeded.

Two configurations of the EC heating system were performed using a DEMO Water Cooled Lithium Lead (WCLL) with integrated EC configurations model of a half of the sector (i.e. 10° model) and relevant ones repeated with a full sector model (i.e. 20° model) to test the reliability of results. Additionally, the effect of the radiation on the WGs closest to the plasma was analyzed as well as the impact on Tritium Breeding Ratio.

Keywords: DEMO, equatorial port plug, electron cyclotron, nuclear heating, material damage

1. Introduction

In support of the integration of EC heating system [1] neutronic simulations were performed to make sure that all design requirements are met and to optimize the design when needed. In this process the reduction in the T breeding performance of the breeding blanket as well as shielding performance of the design was assessed.

Two different remote steering EC system designs were analyzed both consisting of 8 waveguides (WGs). Due to penetrations in the tritium breeding blanket (TBB) and port plug it was expected that further shielding optimization would be required to keep the nuclear heating of the toroidal field (TF) coils and the neutron induced damage of the exposed part of the vacuum vessel (VV) below their design limits. Additionally, the nuclear heating and the neutron damage to the exposed parts of the EC WGs were assessed.

To reduce the CPU time requirements, first a half-sector model (10°) was used while later the important results were confirmed using a more suitable full-sector model (20°).

2. Configurations and models

2.1 EC configurations

The first analyzed EC configuration consisted of two rows of four WGs (EC#1, Figure 1) and the second of a single column of eight WGs (EC#2, Figure 2). The inner dimensions of the copper WGs were in both cases 63.5 mm × 63.5 mm and the thickness of their walls 2 cm.

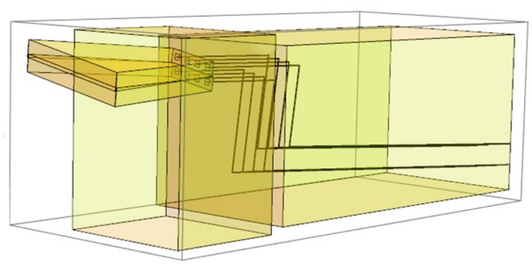

Figure 1. EC#1 waveguide configuration (2×4 WGs).

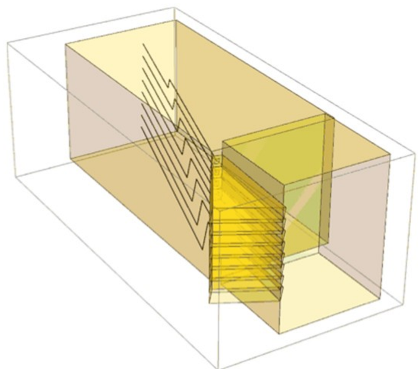

Figure 2. EC#2 waveguide configuration (8×1 WGs).

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2.2 DEMO models

The DEMO model used in the neutronic analyses was based on the generic model [2][3] with modifications needed for the equatorial port plug containing the EC heating system (Figure 3). For both configurations the part of the tritium breeding blanket in front of the equatorial port was modified to include penetration for EC WGs. McCad [4] and SuperMC [5] CAD to MCNP conversion tools were used for model preparation and ADVANTG [7] was used to speed up the simulations.

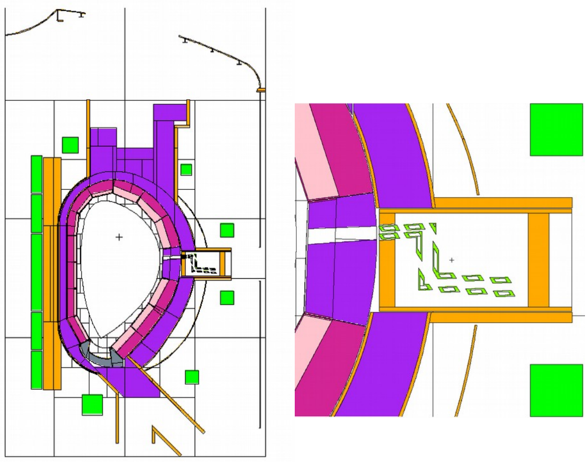

Figure 3. Vertical cross-section of the MCNP model of DEMO used in analyses (left) and a closer view of the initial EC#1 port plug configuration (right).

2.2 Relevant design limits

The design limits relevant for this work are presented in Table 1 and the lifetime of DEMO is considered as 6 full power years (FPY) of operation.

Table 1. Design limits for the assessed factors [8][9].

Parameter	Value
Tritium breeding ratio (TBR)	≥1.10
Peak nuclear heating of TF coils (NH TFC)	<50 W/m^3
Peak displacement damage in vacuum vessel (VV dpa)	<2.75 dpa/lifetime

2.3 First results

The effect on the TBR for the single module segment WCLL blanket concept [10] was assessed. The value for the reactor without EC port plug (baseline) was 1.156 and the values for both EC configurations are presented in Table 2 for a case with 5 EC systems providing up to 50 MW of plasma heating during operation [11][12]. This assessment used full sector (20°) model and took into account that only 5 out of 18 sectors include the system. Analysis also included an estimation of the effect on the T breeding performance in the sectors next to the EC system. The tritium breeding performance of the sectors next to sectors with EC port plug were assessed as a mean value between the EC and the baseline while the sectors not neighboring on the EC sectors were taken as unaffected (baseline values) [11]. The resulting TBR value for EC#1 is just below the acceptable 1.1 while the EC#2 is sufficiently high to allow some margin needed for integration of other systems. The reason for the difference in TBR performance is that in the case of EC#1 the whole part of the TBB in front of the port plug was replaced with a shield while the position of WGs in EC#2 configuration close to the edge of the blanket module allowed for significantly smaller displacement of the breeding material with shield.

Table 2. First results with initial design of the EC port plugs.

Parameter	Value EC#1	Value EC#2
TBR	1.099	1.126
NH TFC [W/m^3]	400	900
VV dpa [dpa/lifetime]	0.5	10

Due to the large openings of (at their narrowest part) approx. 67 cm × 32 cm and 26 cm × 122 cm in the tritium breeding blanket and the port plug for EC#1 and EC#2 respectively, the shielding performance was insufficient for both cases. The peak values of TF coil heating (Table 2) were thus 8-times (Figure 4) and 18-times too high for EC#1 and EC#2 respectively. Additionally, it was found that while the peak values of the neutron damage in VV increased for both cases (Table 2), the position of the opening in the blanket for EC#2 close to the edge of the port especially increased the neutron damage in the exposed part of the VV above the limit of 2.75 dpa/lifetime. On the other hand, the peak values for EC#1 were found to be well below this limit (Figure 5).

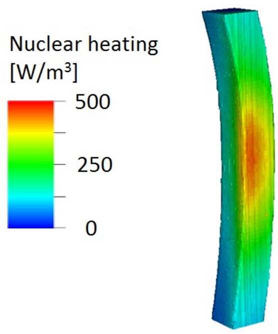

Figure 4. An initial determination of the nuclear heating in TF coils for EC#1. The profile of nuclear heating is peaked with higher values closer to the opening for the EC WGs.

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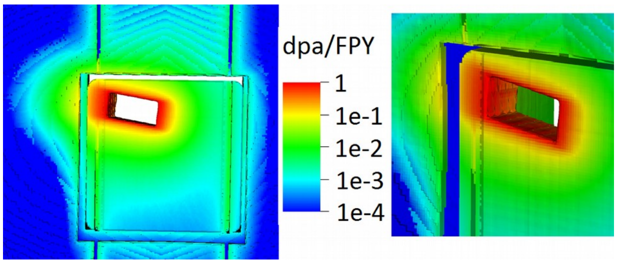

Figure 5. Neutron displacement damage in the stainless steel of the VV and port plug for the EC#1. Parts around the opening for EC WGs show significantly elevated values due to streaming through the opening in the TBB. However, the distance between the opening and the VV means that the peak values in the VV are 0.09 dpa/FPY or 0.5 dpa/lifetime, well within the design limits.

As the plasma facing part of the WGs is exposed to neutrons and neutron induced gamma radiation the neutron damage and nuclear heating in the Cu WGs were also assessed. Results are presented in Table 3.

Table 3. Peak values of neutron induced displacement damage and nuclear heating in plasma-facing part of the Cu waveguides.

Parameter	EC case	Peak value
Neutron damage	EC#1	1.5/FPY or 9.1/lifetime
[dpa]	EC#2	3.6/FPY or 21.6/lifetime
Nuclear heating	EC#1	2.3
[MW/m^3]	EC#2	4.8

Additionally, nuclear heating maps for the port plug region were assessed (Figure 6). The effect of the large opening in the blanket is clearly visible as elevated heat loads extend all the way to the port plug. The values are within the technical limits.

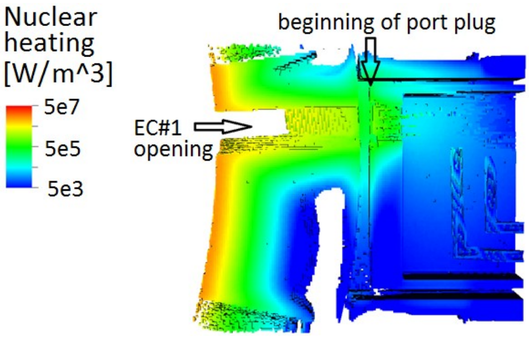

Figure 6. Nuclear heating maps for EC#1 configuration. Maximum heating loads of 1.3×10^7 and 1×10^6 W/m^3 were found for the first wall and front part of the port plug respectively. Values in the opening are between these two values.

3. Shielding optimization

Clearly the shielding performance of both original EC configurations is insufficient and additional shielding is required. This was expected due to the size of the penetrations and the amount of shielding material displaced by the EC systems and their port plug. Different shielding strategies were tested and possible solutions found. In these analyses, additional shielding blocks (60% stainless steel, 40% water) were positioned in different locations inside the port plug, the effect of the increased port walls tested, and the size of the gap between the port plug and the port varied.

3.1 Nuclear heating in TF coils

Adding sufficient amount of shielding to the port plug can reduce the peak values of nuclear heating in TF coils below the design limits. However, putting a lot of material into the port plug increases its weight and can make such design impractical. Because of that, different shielding strategies were considered for EC#1 to find different possible design configurations:

- 40 cm shield in front part of the port plug (Figure 7, left) and 20 cm shield right after the dogleg (Figure 7, right).
- Doubling the thickness of the port wall (DPW configuration) from 20 cm to 40 cm in order to reduce the peak nuclear heating of TF coils without adding weight to the port plug.
- Combination of thicker port wall and 20 cm shield after WG dogleg
- Reduction in the size of the opening in the port plug around the WGs (Figure 8).

Results of these shielding cases for EC#1 are presented in Table 4. While multiple strategies for decreasing of the nuclear heating in the TF coils were found, a combination of multiple (e.g. shield + thicker port wall) is likely more suitable than simply adding a large amount of shielding material to the port plug. However, other analyses such as shut down dose rate (SDDR) analyses in relevant regions are needed to further confirm the suitability of the design choices as for example the size of the gap between the port plug and the port wall was found to have minimal effect on the peak TF coil heating but could potentially significantly increase the SDDR behind the port plug. After all, in the final design all the design requirements must be met.

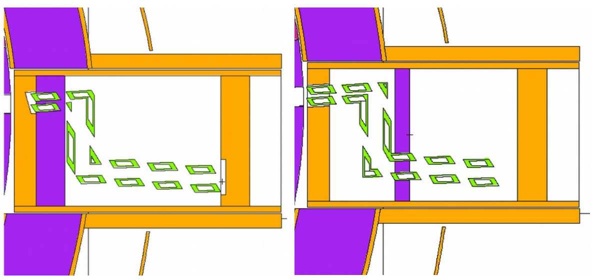

Figure 7. Two shield options in the port plug. 40 cm shield before the WG dogleg (40 cm shield, left) and 20 cm shield after dogleg (20 cm shield after dogleg (DL), right).

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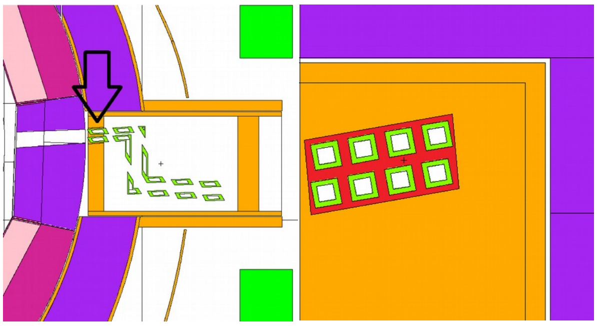

Figure 8. The location of the opening around WGs (left) and a (red) cell used to close it up (right)

Additionally some analyses were performed for EC#2. It was found that a 70 cm shield in the front part of the port plug results in a nuclear heating of 11 W/m^3 in the TF coil – similar levels to those obtained for a 40 cm shield in EC#1. Other shielding strategies, e.g. thicker port wall and reduced opening for WGs, are expected to produce similar relative reductions to the ones from EC#1.

Table 4. Peak values of nuclear heating in TC coils for EC#1 and different shielding strategies.

Shielding option	NH TFC [W/m^3]
Original	400
40 cm front shield	12
20 cm shield after dogleg	70
Double port wall (DPW)	130
No opening around WGs	170
DPW + 20 cm shield after DL	20

3.2 Neutron damage in vacuum vessel (EC#2)

Different approaches were considered in order to reduce the neutron damage to the vacuum vessel due to the opening for the EC#2 configuration (Figure 9).

While adding some shield might be possible it would be difficult to sufficiently reduce the damage due to geometrical requirements of the opening – the opening around WGs is determined based on the operating requirements of the EC system. An alternative solution would be to exclude the exposed area from the VV. This might be achieved through increasing the size of the opening for the port beyond the region exposed to excessive neutron radiation and then reducing the opening by stainless steel attachment which is not part of the VV. In this way, the requirements for the exposed part of the stainless steel is less strict than for the VV which is a crucial lifetime component. It was found that excluding ~25 cm of the VV edge reduces the peak value below the limit (exclusion zone is shown as white square in Figure 9).

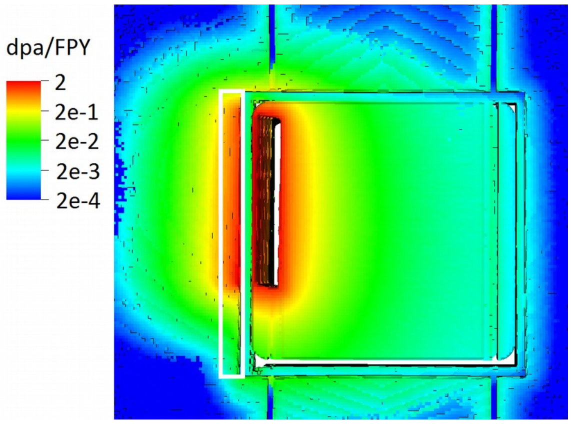

Figure 9. Map of neutron damage in stainless steel parts of the EC#2 port plug and VV.

4. Conclusion

Neutronic analyses were performed in support of the development and integration of the equatorial port plug for the remote steering EC heating system. Various potentially design limiting parameters were analyzed and shielding configurations optimized when necessary. The analyzed parameters include the TBR, nuclear heating in TF coils, and neutron damage in VV. These analyses gave insight into the problem, identified some potentially suitable shielding solutions for the analyzed cases, and showed some shielding strategies useful for future system designs – while simply filling up large parts of the port plug is a valid shielding strategy it can result in excessive weight of the port plug so alternative strategies have to be considered. The reduction in the size of the opening around EC waveguides and the increase in the thickness of the port wall are two examples of such alternative options. To meet all of the design requirements in the final design it is likely that multiple shielding strategies will have to be implemented at the same time.

Acknowledgments

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