Neutronics Analyses of EU DEMO 2020 EC Port Configuration

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ABSTRACT

The DEMOnstration fusion power plant (DEMO) is being developed within EUROfusion and one of the challenges is the integration of all the systems into a fusion reactor by designs that meet strict design criteria required for safe and reliable long-term reactor operation. Neutronics analyses are required as an important contribution to this effort.

The work described here has analysed the equatorial port plugs of the electron cyclotron (EC) system in terms of neutronic aspects. The EC port plugs are needed for heating the fusion plasma and for plasma control. Necessary openings in the breeding blanket and port plug structures foreseen for the mm-wave heating beams of microwaves coming from launchers of the EC system in equatorial port plugs are a challenge in terms of neutron shielding. Both the neutron streaming through the EC port and nuclear loads in its critical components like mirrors have to be considered in the system design and integration.

1 INTRODUCTION

The design of a fusion power plant is a complex process where multiple systems with potentially multiple functions need to be integrated into a comprehensive reactor design that performs all of the required functions and is designed to safely operate within the limits of available materials and technology. Part of the integration effort is integration of plasma heating and control systems into ports e.g. integration of the electron cyclotron (EC) system into equatorial ports sitting behind openings in the breeding blanket (BB) segments. Nuclear challenges related to the EC system integration are mainly connected to neutron streaming through the waveguides and gaps needed for the integration and operation of the system. High neutron and gamma fluxes are problematic either directly, e.g. causing excessive nuclear loads in sensitive components like superconductive magnets, or through neutron activation of materials resulting in high shutdown dose rates (SDDRs) in parts of the reactor where maintenance is foreseen either by humans or by remote handling.

Two of the typically most effective strategies for mitigation of neutron streaming through the system are designing gaps and openings in a dogleg-like fashion and strategically adding shielding materials as close to these openings as possible.

The latest pre-conceptual EC port design was assessed in terms of nuclear loads in exposed parts of the system as well as the effect of such EC port plug penetrations on the nuclear heating of the superconducting toroidal field (TF) coils. The current design is a result of design evolution driven by both neutronics and other considerations, e.g. structural requirements, maintainability, and plasma heating and control. The variant examined in the paper is based on a set of optical mirror systems in the EC port plug with fixed and steerable mirrors, powered by fixed frequency gyrotrons in order to perform its functions of plasma heating and control. This variant for DEMO is also called mid-steering launcher concept [1]. It was chosen as a compromise between the previously analysed but abandoned remote-steering concept and the ITER-like front steering concept.

In this paper we describe the neutronic studies for the DEMO EC mid-steering equatorial port concept, i.e. nuclear heating and neutron damage assessments in important parts of the system, nuclear heating in superconducting TF coils, and SDDRs in accessible port areas relevant for maintenance and inspection.

2 NUCLEAR ANALYSES

2.1 Tools used in analyses

MCNP5 v1.6 [2] and ADVANTG 3.0.3 [3] were used for particle transport analyses, and MCR2S [4], MCNP6.2 [5] and WWITER [6] for SDDR analyses. Nuclear data from JEFF 3.3 [7] for neutron transport, MCPLIB84 [8] for photon transport, and EAF2010 [9] for the activation calculations were used. Models for nuclear analyses were converted from CAD to MCNP format using SuperMC [10]. A 10 cm × 10 cm × 10 cm superimposed Cartesian mesh was used to assess quantities of interest when cell tallies were not suitable. Cell and mesh based tallies were of a flux in a cell/mesh cell variety and energy dependent probabilities were taken into account for assessments of nuclear heating and neutron induced damage of the materials. For SDDR calculations the biological-equivalent SDDRs calculated using ICRP 74 photon flux-to-dose rate conversion factors were used [11].

2.2 Nuclear heating and neutron induced damage analyses

Nuclear heating of components is an important part of specification of the cooling requirements. This is important for components close to the plasma, e.g. heating mirrors, port plugs and exposed parts of the vacuum vessel, and for sensitive components like superconducting TF coils. A limit of 50 W/m^3 [12] has been specified for the nuclear heat density on the TF coil winding pack. This limit is found challenging to achieve for various port designs that require significant openings, e.g. in previous designs of the EC port [13], lower port [14], and upper port [15].

Neutron induced damage in components expressed in displacements per atom (DPA) is also an important aspect of the design. Peak values of this quantity are used to define lifetime of components as excessive damage can lead to structural problems.

2.3 Shutdown dose rate analyses

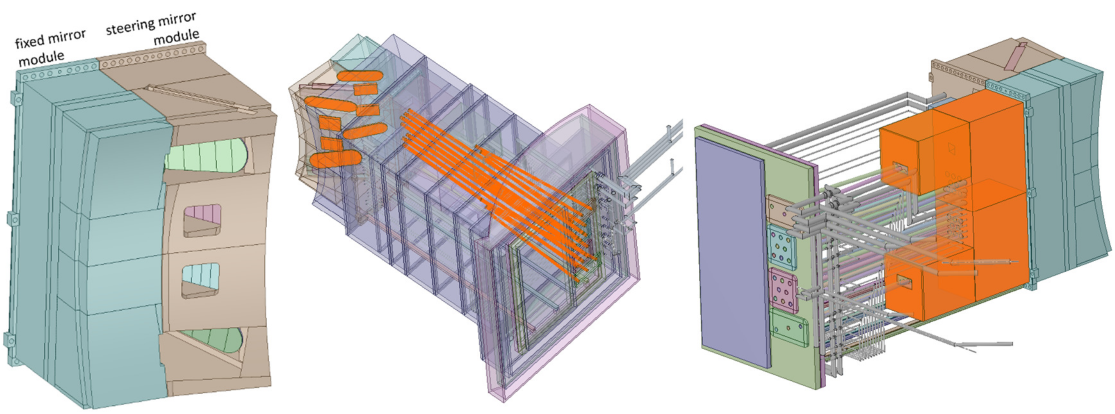
During short- and long-term maintenance the radiation environment due to activated materials, described by SDDR, will guide the maintenance design of systems. This relates to the exposure to ionizing radiation (biological SDDR) as input to assessments of occupational radiation exposure and of the dose (absorbed SDDR) to remote handling equipment, including insulators and electronic equipment. Accordingly, the neutronics tasks are related to the computation of relevant radiation fields and responses to guide the design of systems and to support ALARA assessments of its maintenance.

3 DESIGN AND MODELS

The design of the EC port plug has evolved through several iterations. Due to challenges related to nuclear loads in sensitive components experienced with ITER front-steering EC concept it was first decided that remote steering would be employed. The potential upside of a latter system was that all movable components were positioned far away from the plasma where neutron loads are likely to be less problematic. However, this came at a price of larger openings which lead to high nuclear loads in sensitive components such as TF coils or use of large amount of shielding material [13].

3.1 2020 EC Port Configuration

For 2020 design of the EC port a mid-steering launcher design was proposed (Figure 1). This is a compromise between front- and remote-steering concepts that reduces the sizes of required penetrations by pushing the active steering components closer to the plasma but keeps the nuclear loads in these components manageable by using a dogleg system and designs steering drives for movable mirrors in such a way that they are hidden further inside the port plug [16]. Waveguide shields are positioned around waveguides behind the port plug modules to reduce the streaming of neutrons through the openings.



a b c

Figure 1: The 2020 design of the EC port with a division of the front part into two modules, a fixed mirror and a steering mirror port plug module (further on called SM and FM) (subfigure a) where the mirrors in the SM directly face the plasma and the module is made of water cooled EUROFER while the mirrors in the FM are hidden behind the BB and the module is made of water cooled SS316L(N)-IG. The dimensions of SM and FM are 163 cm (at the thinnest) × 122 cm × 324 cm and 163 cm (at the thinnest) × 92 cm × 324 cm respectively. The optical mirror and waveguides (WGs) system form a dogleg-like mm-wave heating beam path (orange parts in subfigure b) and additional shielding is implemented around waveguides to reduce neutron streaming through the openings (orange blocks in subfigure c).

Main components and their preliminary chosen material composition include:

- Steering mirror module: EUROFER (60 vol%) and water (40 vol%)
- Fixed mirror module: SS316L(N)-IG (60 vol%) and water (40 vol%)
- Equatorial port wall: SS316L(N)-IG (60 vol%) and water (40 vol%)
- Mirror coating (5mm): Tungsten
- Mirror base: EUROFER (60 vol%) and water (40 vol%)
- Waveguide shields: SS316L(N)-IG

3.2 MCNP Model

The geometry for MCNP analyses is based on CAD models. The CAD model used in the conversion into MCNP geometry representation is very similar to the CAD model provided by system designers. The main difference is that all spline surfaces are replaced by simpler surfaces to ensure compatibility with MCNP. In this process of simplification it is essential to ensure that both geometrical features and the amount of materials are not significantly changed. Generally, whenever possible the differences in volumes between the original model and simplified model were kept low, e.g. below 1%, so no density corrections were needed. A single sector model (22.5°) and reflective boundary conditions are used to reduce the modelling and computing effort of the model. It should be noted that these results are seen as conservative due to the use of reflecting boundary conditions, where the calculations simulate a configuration of ECs in every sector, and the off-centre placement reduces the port spacing.

To make design iterations faster, the model makes use of MCNP's universe as shown in Figure 2. In this way for example when the EC port design is modified we can modify and convert only the equatorial port universe instead the whole DEMO sector model.

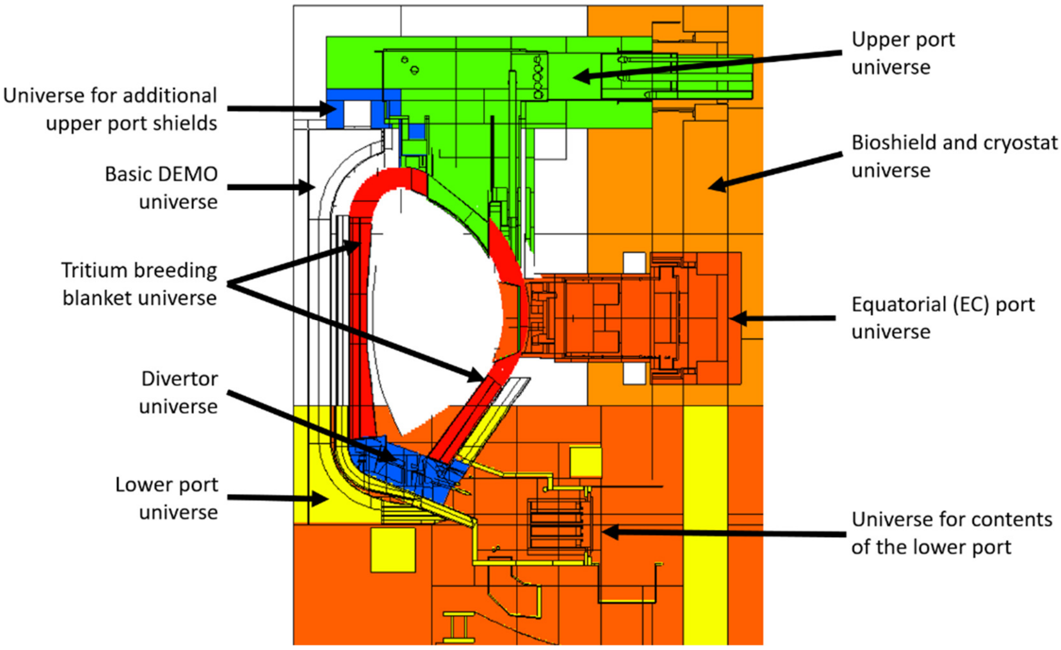


Figure 2: Cross-section view of the MCNP model used in the analyses. The division of the model in terms of universes is clearly visible by use of different colours.

4 SIMULATIONS, RESULTS, AND FUTURE WORK

4.1 Nuclear load results in EC port components

Nuclear heating and neutron induced damage of exposed parts were calculated. The former is important in terms of defining the cooling requirements and the latter can be used to assess lifetime of the component and support the material selection process for exposed components.

Concerning nuclear heating of exposed components both peak and integral values are important to consider. The nuclear heating maps for SM and FM modules are presented in Figure 3 and Figure 4 while maximum values and total nuclear heating are in Table 1. Generally the majority of the heating contribution comes from gamma rays.

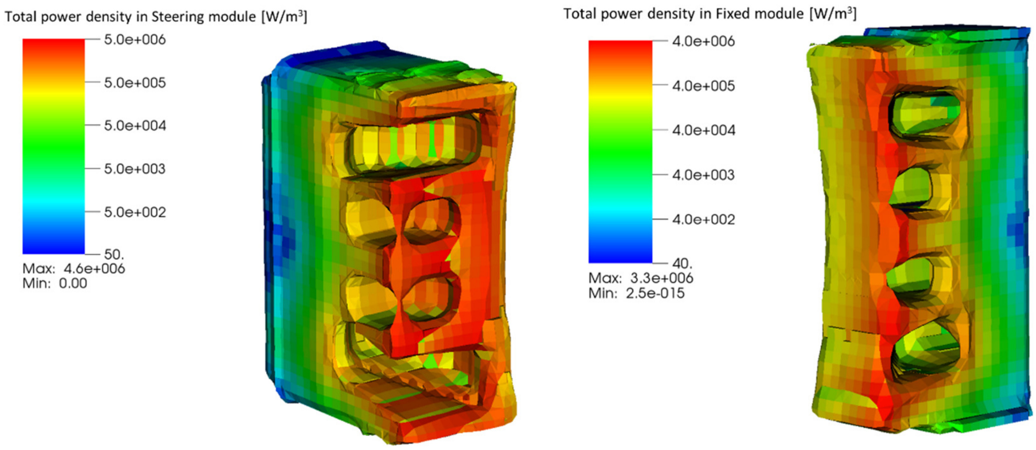


Figure 3: 3D maps of total power density (neutron and gamma) for SM (left) and FM (right) port plug module.

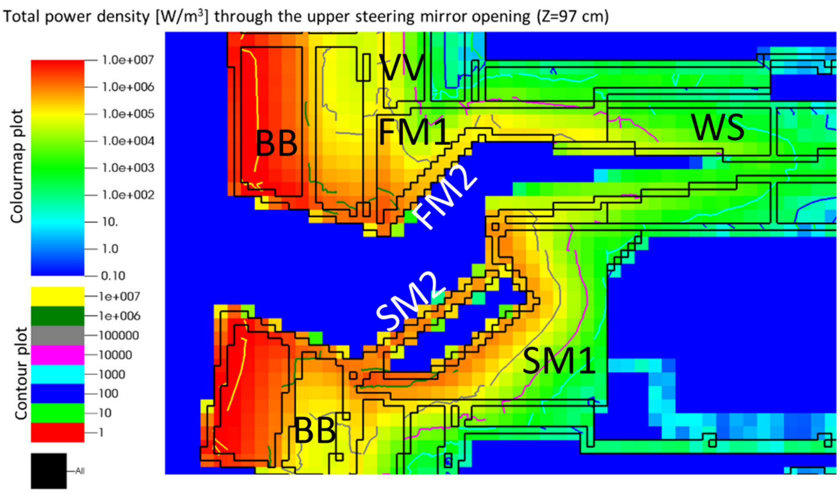


Figure 4: Neutron and gamma power density [W/m^3] through the upper steering mirror opening. Horizontal cross section at Z = 97 cm (for Z = 0 at the port plug midplane). Important components are annotated: BB – breeding blanket, VV – vacuum vessel, SM1 – steering mirror module, SM2 – steering mirror, FM1 – fixed mirror module, FM2 – fixed mirror, WS – waveguide shield.

Neutron induced damage in units of DPA per full power year of reactor operation (FPY) are presented in Figure 5. In its full lifetime DEMO is expected to operate for 6 FPY and calculated DPA/FPY values can be used to assess lifetimes of components. It is crucial that components that are expected to last for the whole lifetime of DEMO are sufficiently protected to comply with their respective DPA limits or they need to be designed to be replaceable. Maximum values for DPA/FPY in port modules are presented in Table 1.

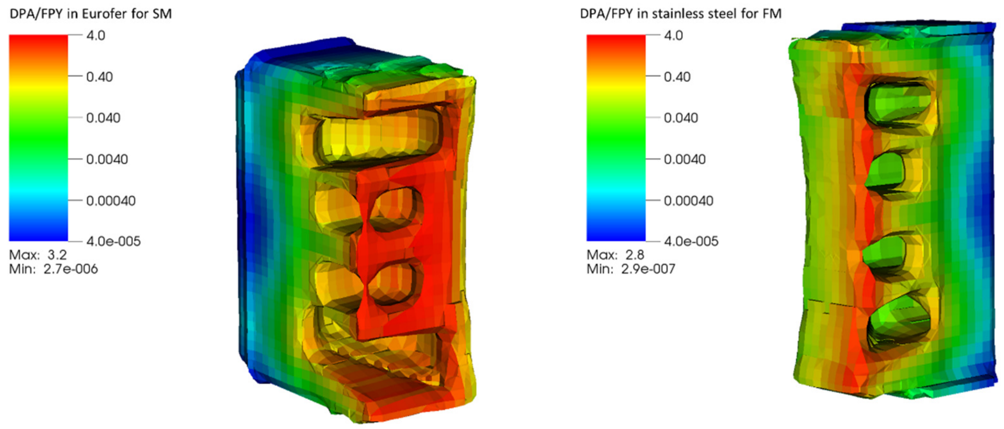


Figure 5: 3D map of neutron induced damage in units of DPA/FPY for SM (left) and FM (right) port plug module.

Table 1: Nuclear loads in port plugs.

	Steering mirror port module	**Fixed mirror port module**
Total nuclear power (neutron and gamma) **[MW]** *1	2.4	0.74
Maximum nuclear power density (neutron and gamma density) **[MW/m^3]** *2	4.6	3.3
Maximum neutron induced damage [DPA/FPY] *2	3.2	2.8

***1** stat. uncertainties < 1%, ***2** Statistical uncertainties of results in the most exposed parts of the geometry < 3% and uncertainties in peak value determination due to discretisation into Cartesian mesh are estimated to be ~10%.

Furthermore, an assessment of peak heating of the most exposed EC mirrors was performed. To this end, mirrors were divided into smaller parts and respective values of nuclear heating and neutron induced damage are represented in Table 2.

Table 2: Ranges of nuclear heating and neutron induced damage in EC mirrors.

Range of nuclear power in mirror [W/cm^3]	Mirror coating (tungsten)	Upper steering mirror	5.1 – 12.8
		Upper heating mirror	8.2 – 10.7
		Bottom heating mirror	8.1 – 10.5
		Bottom steering mirror	5.5 – 13.4
	Mirror base (EUROFER + H_2O)	Upper steering mirror	1.1 – 3.2
		Upper heating mirror	1.6 – 2.2
		Bottom heating mirror	1.6 – 2.2
		Bottom steering mirror	1.1 – 3.2

Range of neutron induced damage [DPA/FPY]	Mirror coating (tungsten)	Upper steering mirror	0.28 – 1.1
		Upper heating mirror	0.48 – 0.68
		Bottom heating mirror	0.47 – 0.68
		Bottom steering mirror	0.24 – 1.1
	Mirror base (EUROFER + H_2O)	Upper steering mirror	0.49 – 2.0
		Upper heating mirror	0.73 – 1.1
		Bottom heating mirror	0.70 – 1.0
		Bottom steering mirror	0.44 – 2.0

4.2 Nuclear heating results in superconducting toroidal field coils

The maximum values of nuclear heating in superconducting TF coils can be a limiting factor when integrating systems with BB openings into ports and other parts of the fusion reactor and therefore were assessed. The requirement is that the peak neutron and gamma heating should not exceed 50 W/m^3.

WG shields are designed based on previous experience and are responsible for a relatively good shielding performance. The peak value of nuclear heating in TF coils was found to be 60 W/m^3 slightly above the limit. However, simulations where 6 cm port walls were replaced by 20 cm double walls (60% SS, 40% H_2O) resulted in peak values of 40 W/m^3 which showed that relatively modest increase in shielding can lead to values within design limits. However, these values are at least somewhat conservative due to the off-centre location of the EC port and reflective boundary conditions. These two factors result in an increase in the maximum values in nuclear heating on the side where the port is closer to the TF coils as effectively two EC ports are close to the more exposed TF coil (Figure 6).

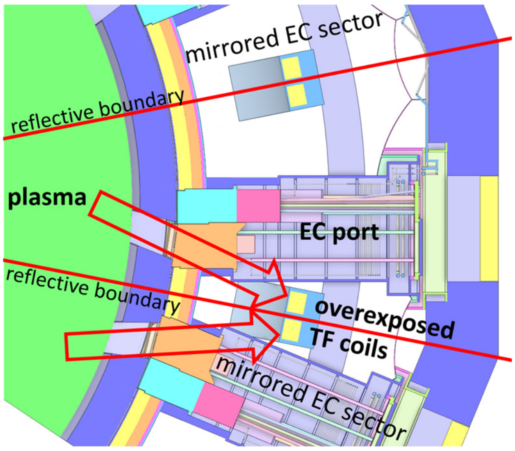


Figure 6: The geometrical anomaly created by using reflective boundary conditions with an off-centre port model. Red arrows indicate an important neutronics pathway towards overexposed TF coil. On the other hand the TF coil on the other side of the port is underexposed (for a model in which all sectors would be EC sectors).

4.3 Shutdown dose rates

The SDDRs were calculated to assess which parts of the reactor building are accessible for human intervention and which will require remote handling. Phase 1 of DEMO operation, i.e. running the machine for 1.57 FPY in 5.2 calendar years, and cooling time of 12 days were assumed. The lower and upper port openings were blocked in this calculation for the purpose of studying the equatorial port. Due to the openings needed for EC heating system, relatively high dose rates inside the port mean that all of the maintenance inside the EC port will have to be performed by the remote handling system. The values inside the port are between 10 and 100 mSv/h. These values indicate the need of remote handling maintenance inside the port, which is why its design is optimized accordingly. Dose rate maps in two cross-sections through the EC openings are shown in Figure 7. The effect of the port on the surrounding area in terms of SDDRs is also visible directly above and below the port. This means that in the current configuration remote handling will be required for maintenance procedures in these areas.

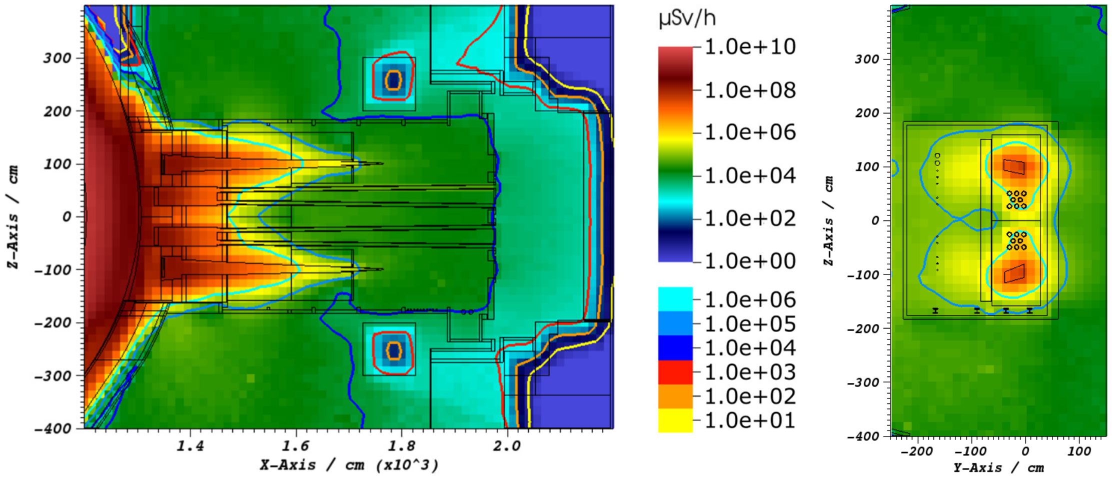


Figure 7: SDDRs in the equatorial EC port 12 days after reactor shut down in µSv/h. Radial-poloidal cross-section through the waveguide openings at Y = -18 cm (left) and poloidal-toroidal cross-section through the port plug modules at X = 1480 cm (right).

The SDDR around the EC port is largely between 10 and 100 mSv/h (blue contours) and is above 1 mSv/h throughout the cryostat in the equatorial port region (red contour). This is above the limit for personnel access. This appears to be caused primarily by radiation leakage close to the join between the port and the vacuum vessel, on the top and bottom of the port and on the side of the waveguide openings. Once again it should be noted that due to the reflecting boundary conditions, the calculations simulate a configuration of ECs in every sector, and the off-centre placement reduces the port spacing. The results are therefore conservative.

5 CONCLUSIONS

This paper presents the results of the neutronics analyses performed for the 2020 model of the EC port in EU DEMO. Nuclear loads in the exposed parts of the port plug were assessed to inform the design process on the cooling requirements and lifetime of components or material selection. The analysis of nuclear heating in superconducting TF coils showed that the maximum value of nuclear heating is currently above the design limit and another design iteration of modest shielding optimisation is required. Further shielding studies with thicker port walls showed that meeting the design limit is possible with such design modification, however, other shielding optimisations might be preferable, e.g. adding shielding material inside the port instead of port walls. SDDR analyses quantified the dose rates in and around the port 12 days after reactor shutdown. As the dose rates within the port interspace are exceeding targets for human intervention by at least one order of magnitude, further shielding efforts are required. This is an integration issue to be addressed by various tokamak systems aiming at globally minimising SDDRs within the cryostat which would result in lower occupational radiation exposure for workers and also somewhat lower other nuclear loads e.g. total nuclear heating of TF coils. As the port itself is designed with remote handling in mind it is not a problem that the SDDR values inside the port significantly exceed limits for the human intervention.

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