Fusion Science and Technology

ISSN: 1536-1055 (Print) 1943-7641 (Online) Journal homepage: www.tandfonline.com/journals/ufst20

Preconceptual Neutronics and Shielding Studies of a Beam-Driven Tokamak Volumetric Neutron Source

Dieter Leichtle, Roman Afanasenko , Christian Bachmann , Aljaz Cufar , Gianfranco Federici , Thomas Franke , Bor Kos , Ivo Moscato , Jin Hun Park , Pavel Pereslavtsev & Alex Valentine

To cite this article: Dieter Leichtle, Roman Afanasenko , Christian Bachmann , Aljaz Cufar , Gianfranco Federici , Thomas Franke , Bor Kos , Ivo Moscato , Jin Hun Park , Pavel Pereslavtsev & Alex Valentine (20 Jun 2025): Preconceptual Neutronics and Shielding Studies of a Beam-Driven Tokamak Volumetric Neutron Source, Fusion Science and Technology, DOI: 10.1080/15361055.2025.2503680

To link to this article: https://doi.org/10.1080/15361055.2025.2503680

FUSION SCIENCE AND TECHNOLOGY

DOI: https://doi.org/10.1080/15361055.2025.2503680

Preconceptual Neutronics and Shielding Studies of a Beam-Driven Tokamak Volumetric Neutron Source

Dieter Leichtle,[a]* Roman Afanasenko,[a] Christian Bachmann,[b] Aljaz Cufar,[c] Gianfranco Federici,[b] Thomas Franke,[b] Bor Kos,[c] Ivo Moscato,[b] Jin Hun Park,[a] Pavel Pereslavtsev,[a] and Alex Valentine[d]

[a]*Karlsruhe Institute of Technology, Eggenstein-Leopoldshafen, Germany*
[b]*EUROfusion Consortium, Fusion Technology Department, Garching, Germany*
[c]*Jozef Stefan Institute, Ljubljana, Slovenia*
[d]*Culham Centre for Fusion Energy, United Kingdom Atomic Energy Authority, Abingdon, United Kingdom*

Received November 15, 2024
Accepted for Publication April 25, 2025

Abstract — *The EUROfusion Consortium is conducting a preconceptual design and feasibility study of a volumetric neutron source (VNS) facility to address perceived needs in the development of integrated breeding blanket and fuel cycle testing and qualification, with aim to mitigate risks stemming from the current low reliability and technical maturity of present design concepts for a DEMO fusion reactor. The main requirements driving the selection of key physics and technical concepts have been identified to cope with a construction and operation schedule consistent with the DEMO design activities.*

The VNS needs to provide a steady-state plasma operation based on reliable beam-target fusion plasmas with a fusion power at 30 MW, generating a peak neutron wall load of at least 0.5 MW/m^2 up to neutron fluences of 30 to 50 displacements per atom. Neutronics assessments are requested to provide essential nuclear loads and shielding performances of this device. To support the technical feasibility of the VNS tokamak, a series of analyses were conducted in support of the architecture and system design concept. The preconceptual design of the VNS was based on ITER-like shielding structures at the inboard side (ca. 70 cm radial depth) and enhanced outboard side shielding (about a 120-cm radial depth).

According to the assumed port configuration and toroidal segmentation of the VNS tokamak, a torus sector model of 60 *deg* has been generated from available computer-aided design models and converted to MCNP geometry descriptions. The primary objectives of the initial nuclear analyses and scoping assessments were the radial build on the inboard side to protect, specifically, the toroidal field coils and the shielding environment around the neutral beam injector (NBI) port duct, as well the divertor and lower ports, which are critical areas for ex-vessel shielding objectives. It could be demonstrated that the protection from a radial build adopting tungsten-based shields is adequate, whereas additional measures on shielding and integration of NBI ports and lower ports are required.

Keywords — *Volumetric neutron source, neutronics, shielding.*

Note — *Some figures may be in color only in the electronic version.*

---

*E-mail: dieter.leichtle@kit.edu0

I. INTRODUCTION

The development of fusion energy and the deployment of future fusion reactors rely heavily on the performance and reliability of the breeding blanket (BB). Its primary mission is to ensure tritium self-sufficiency during reactor operation, combined with high-grade heat extraction and contributions to the overall shielding objectives. The European Union (EU) fusion development strategy is based on main pillars to advance physics and technology through ITER, IFMIF-DONES, and a DEMOnstration fusion reactor.

Recent considerations on rebaselining the ITER project in view of a staged nuclear operation at very low total neutron fluence have stipulated reviews of the BB qualification and the maturization strategy. It has been conceived that the role of ITER, and specifically, its test blanket modules (TBM) programme to de-risk the DEMO design, is questionable and that new risk mitigation strategies need to be implemented.[1] A change in the current strategy in the EU Fusion Roadmap, to rely on the results of the ITER TBM programme and to use DEMO itself as a qualification device, have been advocated. A key element in this approach is the development of a 14-MeV plasma volumetric neutron source (VNS) for full validation of BB concepts up to high fluence operation.

II. MISSION AND CHARACTERISTICS OF A VNS

The critical role of a BB for development of fusion energy, and further challenges due to the current design maturity and technological readiness levels, demands substantial strengthening and tailoring of the required research and development efforts. The risks of attaining tritium self-sufficiency and adequate system and plant availabilities are joined by risks connected to the licensing of a DEMO reactor with incomplete testing and the qualification of a core nuclear system like the BB.

The change in the BB testing and qualification strategy reintroduces the concept of a VNS to be run in parallel both to ITER operation and the DEMO design process.[1] Its primary mission is to provide full validation up to at least 20 displacements per atom (dpa) of BB concepts for DEMO in a relevant fusion environment.

To ensure its mission, several conceptual characteristics have been specified for a VNS to enable, in particular, its construction and operation in parallel to ITER operation and the DEMO conceptual and engineering design activities (see Ref. [2]). Its availability goals require a steady-state plasma operation utilizing a reliable plasma scenario in a compact device.

The nuclear performance objectives aim at peak neutron wall loads (NWLs) of at least 0.5 MW/m^2 and total neutron fluence on the first wall of well beyond 20 dpa, typically up to 50 dpa. As breeding capability is now assumed, the device should operate at low fusion power, typically well below 50 MW. The tokamak option, based on beam-target fusion (at $Q \leq 1$), has been selected as the most promising approach. In this concept, significant neutral beam injector (NBI) heating power (about 40 MW) is required.

III. NEUTRONICS AND SHIELDING APPROACH

As part of the technical feasibility study of a tokamak-based VNS option, physics and engineering challenges have been addressed to identify potential showstoppers and to demonstrate a credible concept for advancing to a conceptual design (see Ref. [3]). Neutronics analyses have been requested to support the engineering assessments of nuclear performance and to address the main shielding issues.

III.A. Objectives

The primary objectives of the neutronics and shielding approach are to verify the main nuclear performance targets and the capabilities to adequately shield the tokamak environment. It should be noted that further nuclear performance related to activation, radiation mapping, plant radiological protection, and radiological waste issues will be conducted in the future.

The shielding of the VNS tokamak needs to be addressed to verify the provisional protection objectives and the assumed radial build, which have been proposed to optimize magnetic field and overall size of the tokamak. Accordingly, the following assessments have been requested within the feasibility study team and are reported in this paper:

1. Nuclear shielding of the toroidal field coil (TFC) at inboard (IB) side.
2. Nuclear shielding around the NBI ducts.
3. Nuclear loads at the divertor and surrounding areas.

III.B. Models and Assumptions

Based on an initial reference architecture of the VNS concept (see Table I), a conceptual computer-aided

TABLE I

VNS Parameters

Pulse	Steady state
Major radius, aspect ratio	2.55 m/5.3
Magnetic field at plasma axis	6 T
Fusion power	30.5 MW
Number of toroidal field coils	12
Peak heat flux on divertor targets	79 MW/m^2

design (CAD) model has been devised by the engineering team. The neutronics modeling and simulation approach started with the selection of a representative 60-deg sector incorporating two TFCs and one NBI duct (see Fig. 1). The torus sector model was developed by deploying modular elements to enable efficient modifications of single system models. Given the scope of the feasibility study and model status, the neutronics model relies on layered representations of blankets and NBI ducts and generally homogenized material compositions.

Structural, functional, and coolant materials have been selected according to the experience gained in the ITER and DEMO designs, as well as considerations regarding the expected total neutron dose and moderate coolant temperatures. Provisional material choices for in-vessel and vessel parts involve 316-L(N) steels for all structures, water coolant at moderate (11 to 31 bar) pressure, and various choices of shield materials for scoping studies. The TFCs are composed of 316-L(N) casings and Nb_3Sn superconductors. The duct liner of the NBI port contains CuCrZr as the heat sink material, and the divertor

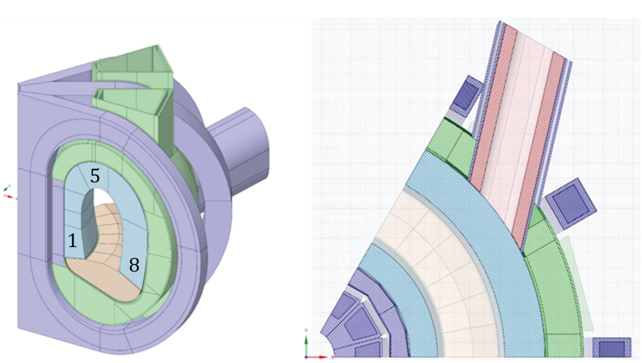

Fig. 1. VNS neutronics CAD model of torus sector with a single NBI duct. Numbers denote the blanket segments used for NWL identification.

adopts plasma facing components (PFC) materials (W, CuCrZr) as in the ITER/DEMO designs. It should be noted that reduced activation behavior currently has not been considered as a material constraint, except for the BB test areas (based on EUROFER as structural material). For this, a material mixture based on the DEMO water-cooled lithium lead (WCLL) concept with PbLi as the breeder/multiplier material has been adopted.

Radiation transport simulations have been performed following the neutronics workflows adopted in DEMO neutronics studies[4] using the MCNP (version 5.0 and 6.2) code and JEFF3.3 nuclear data. Nuclear responses have been tallied according to the specific task. In particular, the IB shielding adopted a one-dimensional mesh (5 cm lateral extent) at the equatorial midplane to capture radial profiles from the first wall to the winding pack of the TFC. Typically, up to 10^9 source particles have been used with weight window meshes generated by the ADVANTG code to obtain the relative errors of the requested responses below 3% to 4%. Global nuclear performance parameters have been assessed versus the criteria given in Sec. II.

The nuclear shielding criteria for the TFC are provided as a target of 450 W/m^3 of nuclear heating in the torodial field (TF) conductor and 50 MGy of absorbed dose to the epoxy insulator within 10 full power years (FPY). Additional shielding criteria with regard to the protection of the vacuum vessel (VV) (2.75 dpa) and concerning the reweldability of the steel pipes [1 atomic parts per million (appm) helium] have been adopted from DEMO.[2]

III.C. Global Nuclear Performances

Coupled neutron and photon transport simulations have been conducted with MCNP to assess the global nuclear responses in the devised VNS torus sector model using a DT neutron emission density of the nominal 30.5-MW fusion power. Shield blankets (SBs) (steel/water mixtures within steel case segments) have been adopted on the IB side and in most outboard (OB) parts, except as the port plug of a testing port and where full poloidal segments could be available for segment testing. In these cases, blanket segments with a WCLL material mixture have been integrated.

As concerns the NWL, a coarse poloidal segmentation (number 1 is lower IB segment and number 8 is lower OB segment, see Fig. 1) has been adopted to resolve its poloidal variation. Peak NWL is at 0.5 MW/m^2 averaged across the OB equatorial level, indicating

TABLE II

NWL and Power Deposition

Segment	NWL (MW/m^2)	Component	Power (MW)
1	0.241	IB blanket	12.30
2	0.413	OB blanket	12.48
3	0.381	Divertor	5
4	0.320	VV shell	0.17
5	0.332	VV shield	0.6
6	0.430	TFC	2.6E-3
7	0.499	Total	30.55
8	0.317		
Average	0.367		

the achievement of the respective performance targets. Over a broad poloidal range, the NWL is generally above 0.3 MW/m^2, thus supporting a still relevant test environment over the full poloidal height.

Global power deposition has been obtained for the full torus, as presented in Table II, with the objective to guide the thermal layout of the VNS tokamak systems. Nearly 24.8 MW are deposited in the blanket system, 5 MW in the divertor cassettes, whereas heating power is very modest within the VV at about 0.8 MW. About 3 kW was assumed to be taken by the TFC. Refined models and assumptions are required in future work to provide detailed spatial heat deposition data.

IV. IB SHIELDING

The original provisional radial build assumed for the VNS was based on an ITER-like shielding structure, i.e., standard steel/water structure with a SB thickness of 40 cm and a VV thickness of about 35 cm. However, as preliminary neutron transport simulations have shown insufficient overall shielding capabilities toward the TFC, but good protection of the VV, a new radial build has been proposed utilizing a thinner blanket (25 cm) and providing margins on the VV thickness (starting point: 46 cm) (see Fig. 2) by the selection of suitable shielding materials across the radial build.

The objective of a series of scoping shielding studies was to evaluate attractive shield material configurations and to suggest an updated and optimized radial build compliant with the TFC protection objectives. The outcome could also suggest a potential reduction in the VV in-wall shield and optional margins for the SB thickness.

The current list of potential shielding materials includes tungsten-based materials (pure W, WC, W_2B_5) and B_4C. Radial profiles of the neutron flux density for three selected configurations are shown in Fig. 3. They are based on screening through combinations of the material choices, highlighting the outstanding neutron attenuation performance of W_2B_5 and the efficient performance of WC in the SB. The cases depicted in the figure are case1 (SB: W_2B_5 VV: B_4C), maximum (SB: WC VV: W_2B_5), and optimum (SB: WC VV: WC + W_2B_5).

The choices and results indicate that sufficient TFC protection at the IB side can be secured by deploying tungsten-based shields with further optimization needs. An example of such optimization is shown in Table III,

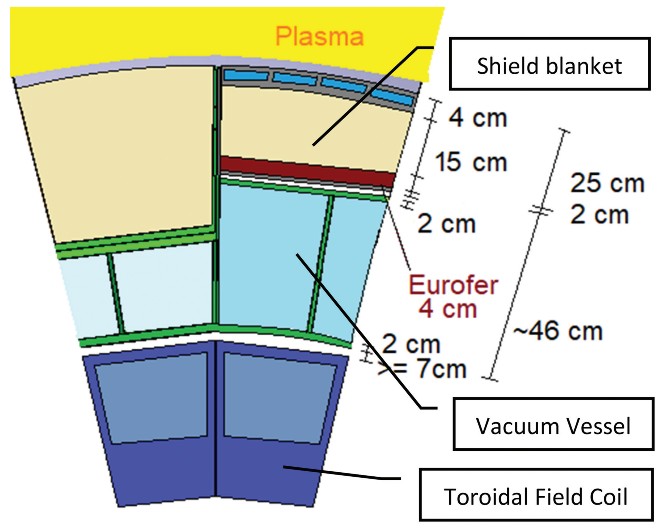

Fig. 2. Radial build at the VNS IB side: (left) original geometry and (right) updated geometry.

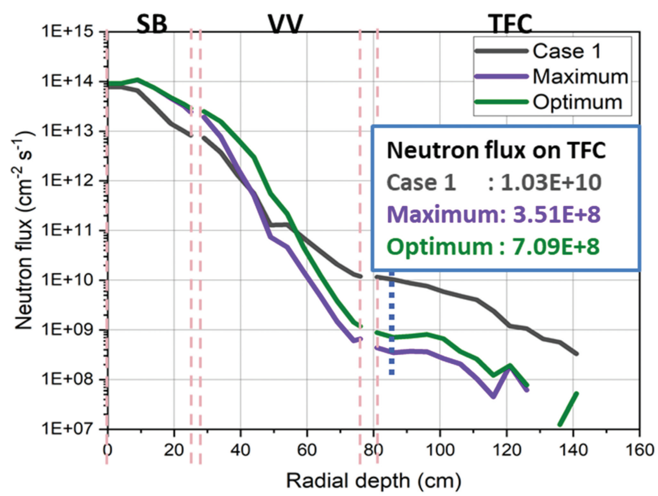

Fig. 3. Radial flux profiles at the VNS IB with three selected blanket/vacuum vessel configurations in the updated geometry.

TABLE III

Nuclear Loads on the TFC Winding Pack*

	Thickness within Vacuum Vessel (t_{VV})	Dose/10 FPY (MGy)			Nuclear Heat (W/m^3)		
		48 cm	43 cm	38 cm	48 cm	43 cm	38 cm
W_2B_5 in VV (cm)	0	31			537		
	0		*53*	*88*		*830*	*1199*
	5		20	35		235	323
	10		13	23		126	212
	20		6	12		54	91

*VV based on B_4C and W_2B_5; SB based on W_2B_5); italic values are above target values.

where the nuclear loads on the winding pack of the TFC at IB are shown varying the radial extent of W_2B_5 in a range of overall VV thicknesses. The results indicate that a VNS VV between a 38-cm and 43-cm thickness could be sensible based on the use of mostly WC or W_2B_5 as the blanket shielding material and only small additions of W_2B_5 (or any other highly optimized neutron absorber) in a boron-containing VV (e.g., as B_4C).

V. NEUTRAL BEAM DUCT SHIELDING

The integration of the NBI heating systems requires large-sized open ports into the VV; in the case of the VNS study, there are 4 such ports out of 12 on the equatorial level. This is generically a challenge for the shielding architecture of a tokamak, as neutrons can leak easily out to ex-vessel areas and into the building (NBI cell). The main objective of the current study was to inform the designers on suitable integration to protect particularly the OB TFC neighboring the ducts. Critical areas concern the limited clearances between the TFC case structure, VV, and NBI port. Accordingly, hot spots of excessive nuclear loads both to the VV and TFC need to be identified and design considerations for mitigation of those issues should be studied.

A neutron flux map in the horizontal plane, as depicted in Fig. 4, helps to understand the challenges. Leakage pathways are prominently seen at the direct line of sight of the plasma neutrons toward less covered areas of the VV. These are affecting the close corner of one TFC. It should be noted that the responses on the TFC at the sector reflective boundary are significantly lower; further tests using periodic boundary conditions confirmed the choice of the other TFC as the most conservative case.

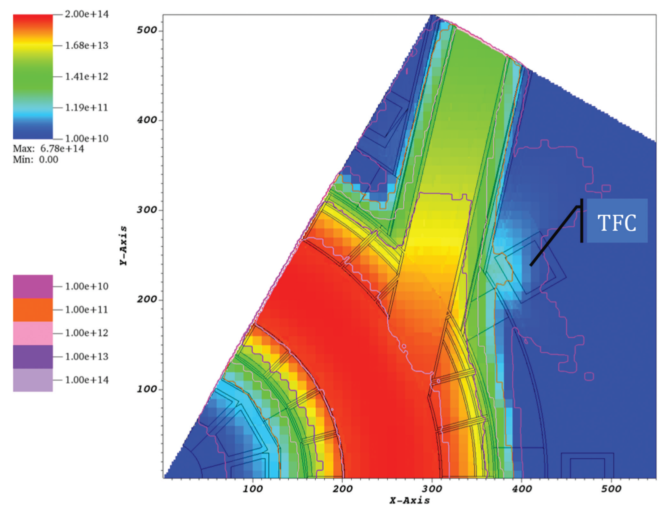

Fig. 4. Neutron flux map at the equatorial level across the NBI port region affecting a close TFC (no additional shielding).

Evaluations of the impact of open upper and lower ports on the responses at equatorial level, i.e., around the NBI ports, have demonstrated that their neutron leakage contribution does not contribute to the responses of interest. This is due to the fact that, indeed, the neutron leakage through the NBI port ducts is dominant. These findings point to a local shielding problem at the NBI duct integration.

To this end, and to mitigate the weak shielding capabilities, an additional outer layer (NBI shield) of 60 mm has been assumed (filled by mixture of 80% W_2B_5 and 20% water). Further studies will address additional measures, such as plug-like inserts between the port and TFC structure. The spatially inhomogeneous loads are clearly visible (see Fig. 5). Whereas the localized peak nuclear heating density (on a 5-cm linear resolution) could be as high as 700 W/m^3, poloidal averages over segments of

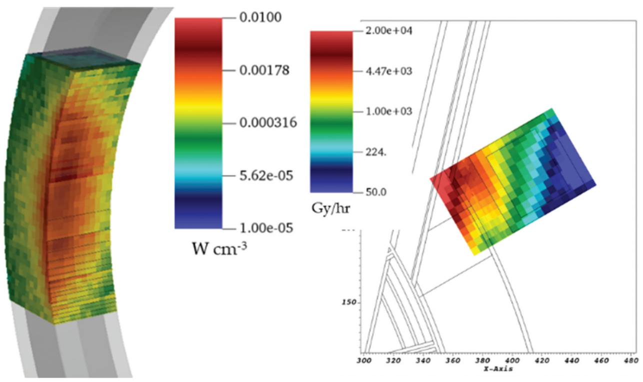

Fig. 5. Distributions of (left) nuclear heat density and (right) absorbed insulator dose rate at the TFC equatorial level (with additional shield layer).

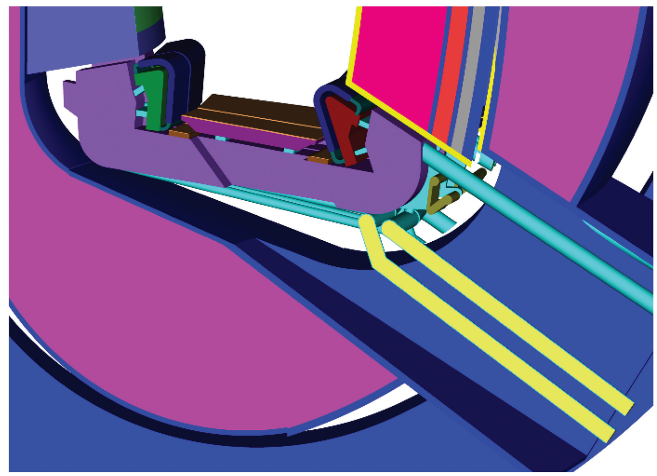

Fig. 6. Divertor cassette integration in the neutronics model showing divertor cooling pipes extending into the lower port.

about a 0.5- to 1-m length are between 70 W/m^3 and 180 W/m^3.

Similarly, localized (1-cm mesh resolution) absorbed dose in the TFC insulator material is as high as 150 MGy (10 FPY). As this is a (local) lifetime criterion, further efforts are required to improve the protection in this respect.

With regard to the protection of the VV in the port integration area, the locally exposed areas at the corner of the port suffer from a lifetime damage of up to 1.8 dpa (10 FPY) in the 316-L(N) steel. This is well below the DEMO design limit of 2.75 dpa.

It was concluded from this study that the NBI port duct integration remains challenging, mainly due to geometrical constraints. TFC nuclear loads require significant enhancements of the shield structures between the VV and TFC, but there are no integration issues concerning the protection of the VV itself.

VI. DIVERTOR INTEGRATION

To study the neutronics and shielding impacts of the divertor integration, the generic model, used in other shielding studies, has been updated with a detailed representation of a divertor cassette design adopted from the DEMO divertor (see Fig. 6). The main objective of this study was to assess integration issues with respect to nuclear loads at the VV and TFC, and also on the divertor coolant pipes guided through the lower port duct. The latter would require cutting and rewelding at positions subjected to potentially high neutron flux levels, which in turn may generate high levels of helium concentration, hindering rewelding.

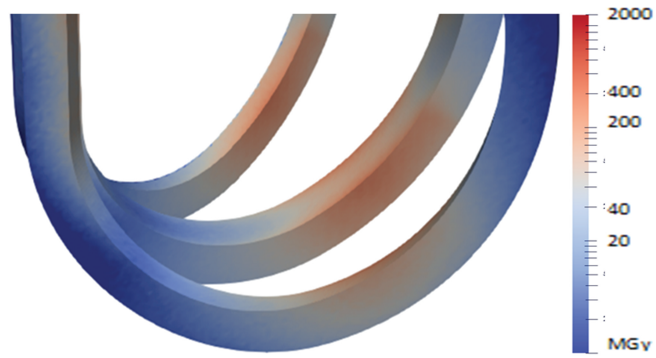

Fig. 7. Distribution of the absorbed insulator dose in the lower section of TFC (10 FPY).

As anticipated, nuclear loads on the TFC are heavily jeopardized by neutron leakage into the lower port, leading to extended regions of response peaking (see Fig. 7). In such completely unshielded ports, the peak nuclear heating range extends to about 3000 W/m^3 and an insulator lifetime dose up to 380 MGy (10 FPY). The results indicate that significant efforts will be required for the mitigation of neutron leakage through the lower port.

The VV below the divertor cassettes is affected mostly by the inclusion of a pumping slot, thus opening a pathway for peaking of nuclear responses in the bottom of the torus. Assessments of nuclear heating and damage in the VV inner shell (see Fig. 8) demonstrate strong peaking, but peak responses are not exceeding the limits. As for damage, the lifetime damage peaks at 1.9 dpa (10 FPY).

Fig. 8. dpa/FPY distribution at the VV inner shell around the divertor cassette.

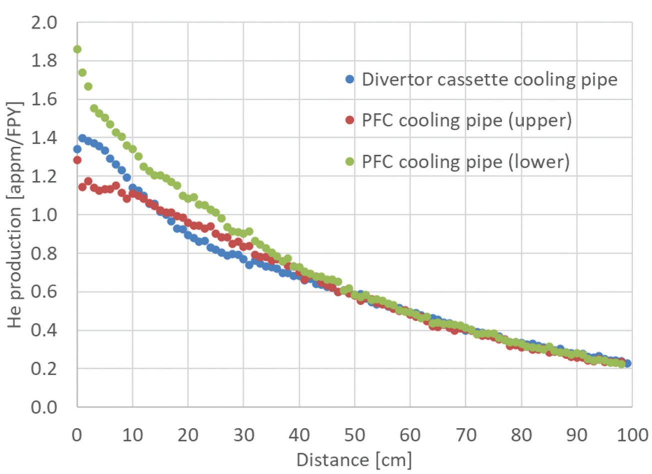

Fig. 9. Radial profile of He production rates (appm/FPY) along the divertor feeding pipes.

One of the key challenges of a regular divertor cassette replacement is the cut-reweld procedure for the feeding pipes. To address its feasibility, helium production rates in pipe steels along the radial extent into the lower port have been calculated (see Fig. 9). A limit of 1 appm applies to thin pipe reweldability, which is jeopardized in positions within the VV close to the divertor cassette. Mitigation measures have been identified, such as optimizing the gap between the blanket and divertor, and prominently, local pipe shielding sleeves at potential rewelding locations.

From the presented results, it can be concluded that divertor integration in conjunction with the lower port configuration requires substantial improvements. This needs to address neutron leakage through the lower port and local pipe shielding for replacement procedures; additional work is reported in Ref. [5]. Lower port shielding is also a prominent task for the further reduction of radiation levels outside the tokamak.

VII. CONCLUSIONS AND FURTHER WORK

Preconceptual neutronics and shielding analyses have been conducted in support of a technical feasibility study of a tokamak-based VNS. In this study, the main shielding and integration challenges inside the tokamak have been assessed. The protection of the TFC, at both the IB and OB sides, is a key issue for the radial build and on the port configurations. Specifically for the IB side, it could be shown that the assumed radial build in favor of a lower magnet field is sufficient to limit nuclear heating and absorbed dose exposure in the TFC to acceptable levels.

The use of tungsten-based materials and its further optimization provides prospect of margins to compensate for the current lack of design details in the SB and VV. Further work in this area will implement preconceptual designs of the SB and VV to verify the current margins. Scoping of moderator materials, such as metallic hydrides, in combination with the experiences of this study should aim at the more efficient use of material configurations.

The integration of NBI ducts at the OB side remains challenging based on preliminary studies reported here. The mechanical constraints in implementing sufficient highly efficient neutron shields between the port duct and TFC require further design considerations. They should aim at optimizing duct-vessel integration versus final sizing of the TFC. Shield plug fillers could be placed between these structures, and optimizing of material configuration will be studied in future work.

Similar challenges for the integration of the NBI ducts are observed in the divertor area and the lower port opening. Whereas nuclear loads to the VV are modest, this does not hold for the TFC segments in the proximity of the open port. Also the cut-reweld procedure of the divertor water pipes, as a prerequisite for replacement procedures of divertor cassettes, is impeded by excessive helium production rates. Further work will address both issues by revising or developing designs, respectively, for the divertor and local shield elements for the protection of water pipes and the mitigation of neutron streaming through the port opening.

In the next phase of the feasibility and preconceptual study of the VNS, further nuclear analyses are required that address the issues of activation, radiation mapping, plant radiological protection, and radiological waste production.

Acknowledgments

The views and opinions expressed, however, are those of the authors only and do not necessarily reflect those of the EU or the European Commission. Neither the EU nor the European Commission can be held responsible for them.

Funding

This work was carried out within the framework of the EUROfusion Consortium, funded by the EU via the Euratom Research and Training Programme [grant agreement no. 101052200–EUROfusion].

Disclosure Statement

No potential conflict of interest was reported by the author(s).

Author Contributions

CRediT: **Dieter Leichtle:** Conceptualization, Project administration, Writing – original draft; **Roman Afanasenko:** Investigation, Visualization; **Christian Bachmann:** Project administration; **Aljaz Cufar:** Investigation, Visualization, Writing – review & editing; **Gianfranco Federici:** Project administration; **Thomas Franke:** Writing – review & editing; **Bor Kos:** Investigation, Visualization; **Ivo Moscato:** Writing – review & editing; **Jin Hun Park:** Investigation, Visualization, Writing – review & editing; **Pavel Pereslavtsev:** Investigation, Visualization, Writing – review & editing; **Alex Valentine:** Investigation, Visualization, Writing – review & editing.

ORCID

Dieter Leichtle http://orcid.org/0000-0003-1741-5061

References

1. G. FEDERICI, "Testing Needs for the Development and Qualification of a Breeding Blanket for DEMO," *Nucl. Fusion*, **63**, 125002 (2023); https://doi.org/10.1088/1741-4326/ad00cb.
2. G. FEDERICI et al., "Overview of the DEMO Staged Design Approach in Europe," *Nucl. Fusion*, **59**, 066013 (2019); https://doi.org/10.1088/1741-4326/ab117i8.
3. C. BACHMANN et al., "Engineering Concept of the VNS —A Beam-Driven Tokamak for Component Testing," *Fusion Eng. Des.*, **211**, 114796 (2024).
4. D. LEICHTLE, U. FISCHER, and C. BACHMANN, "Strategical Approach for the Neutronics in the European Fusion Programme," *Fusion Sci. Technol.*, **77**, *7–8*, 773 (2021); https://doi.org/10.1080/15361055.2021.1887716.
5. A. ČUFAR et al., "Neutronics Simulations for the Lower Port Area of the Volumetric Fusion Neutron Source," *Proc. NENE2024*, Portorož, Slovenia, September 9–12, 2024 (2024).