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To cite this article: Jin Hun Park , Pavel Pereslavl'tsev , Roman Afanasenko & Dieter Leichtle (25 Sep 2026): Neutronics Analysis of Shielding Performance for VNS Inboard Toroidal Field Coil with Various Material Options, Fusion Science and Technology, DOI: [10.1080/15361055.2026.2735684](https://doi.org/10.1080/15361055.2026.2735684)

To link to this article: <https://doi.org/10.1080/15361055.2026.2735684>



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Published online: 25 Sep 2026.



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Neutronics Analysis of Shielding Performance for VNS Inboard Toroidal Field Coil with Various Material Options

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Received June 16, 2026

Accepted for Publication September 9, 2026

Abstract — *The Volumetric Neutron Source (VNS) is a crucial facility for the engineering validation of breeder blanket systems, which are critical components of future demonstration power plants (DEMOs). In a compact tokamak device such as the VNS, the inboard region has extremely limited radial space, making the design of a highly efficient shielding system a key challenge that determines the device's lifetime and performance. This system must protect the superconducting toroidal field coils (TFCs) from neutron and gamma radiation. In particular, degradation of the epoxy-resin insulation due to radiation is a primary factor limiting the lifetime of the magnet system. This study conducts a comparative analysis of the shielding performance of various candidate materials within the VNS inboard shield blanket and vacuum vessel using MCNP6.3 with a three-dimensional computer-aided-design-derived model of the VNS. The key performance indicators are the neutron flux and nuclear heating density in the first TFC winding pack layer and the absorbed dose in the TFC epoxy insulator.*

Keywords — *Volumetric Neutron Source, MCNP, neutronics.*

I. INTRODUCTION

The realization of a future fusion power plant requires not only the elucidation of plasma physics phenomena but also the assurance of engineering reliability for key components operating long term in extreme fusion environments. The Volumetric Neutron Source (VNS) is a key research facility proposed to achieve this goal, serving as a bridge to close the technological gap between the currently under-construction ITER and a future demonstration power plant (DEMO). While ITER focuses on investigating burning plasma physics, the VNS is tasked with creating a high neutron flux and fluence environment, similar to that of DEMO, to test and validate the performance and lifetime of

internal fusion reactor components, such as the tritium breeding blanket [1]. To this end, the VNS is designed as a compact tokamak that drives fusion reactions using high-power neutral beams, aiming to maintain a competitive neutron wall load to primarily test breeding blanket concepts as port plugs or eventually full segments at the vessel outboard side.

The design philosophy of the VNS marks a critical turning point in the paradigm of fusion engineering. It represents a shift from the historical focus on achieving plasma performance to addressing viable technical solutions for material science and nuclear engineering challenges essential for the operation of an envisioned power plant. The analysis of high-performance shielding candidate materials in this study aims to contribute directly to solving a core engineering problem faced by the VNS: ensuring long-term operational reliability within an economically viable compact design.

The most critical design constraint for compact tokamaks, particularly the VNS and spherical tokamaks, is the extremely limited radial space available in the inboard region of the torus [2]. This narrow radial build must accommodate the first wall (a plasma-facing component), shield blanket (SB), vacuum vessel (VV), thermal shields,

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and radiation shielding, while simultaneously providing sufficient radiation protection for the tokamak's core systems: the central solenoid and the inboard leg of the toroidal field coils (TFCs). It is noteworthy that although the VNS's mission is to provide a test bed for the qualification of DEMO-relevant tritium breeding blankets, the inboard side adopts SBs due to the prevailing shielding issues.

This spatial constraint makes shielding efficiency, or shielding performance per unit thickness, the most critical design variable. Conventional shielding methods, such as the combination of relatively thick assemblies of stainless steel and water used in larger devices like ITER, are difficult to apply in the inboard region of a compact tokamak. Therefore, the development and application of high-performance shielding materials with capabilities superior to conventional ones are indispensable to achieve the required radiation attenuation within the limited space.

The TFC is one of the most critical and costly components in a tokamak, and its operational lifetime is directly limited by radiation damage. The inboard shield must effectively suppress three primary damage mechanisms that impact TFC longevity.

First is the degradation of superconducting performance. The 14.1-MeV fast neutrons generated by the D-T fusion reaction collide with the superconductor lattice, causing displacement cascades. Initially, these defects can act as flux pinning sites, partially increasing the critical current density (J_c). However, exceeding a certain neutron fluence leads to a sharp and irreversible decrease in both critical current density and critical temperature (T_c), permanently degrading the magnet's performance [3].

Second is insulator breakdown. The organic insulating materials, particularly epoxy resins or epoxy-cyanate ester blends, which provide electrical insulation between the TFC windings, are the most radiation-sensitive components in the magnet system. Neutrons and gamma rays cause bond scission in the polymer chains of the epoxy, leading to gas generation, swelling, and mechanical embrittlement. This severely degrades the interlaminar shear strength and dielectric strength. The radiation dose limit of the insulator often becomes the most critical factor, particularly in a high-fluence operation regime, determining the overall lifetime of the fusion magnet and, consequently, is the primary criterion for determining the required shield thickness [4]. The integral absorbed dose rate, a key evaluation metric in this study, directly addresses this issue.

Third is nuclear heating and cryogenic load. The nuclear heating generated by neutrons and secondary gamma rays in the superconducting coils, which operate at cryogenic temperatures of around 4 K, imposes a significant

thermal load on the cryogenic refrigeration system [5]. Because removing 1 W of heat from the coil requires several hundred watts of power from the cryoplant, excessive nuclear heating not only increases operational power demand but also threatens magnet stability and can induce a quench if local neutron- and gamma-induced heating exceeds the available cooling margin in any part of the superconducting winding. The nuclear heating and energy deposition evaluated in this study quantify this thermal load.

In conclusion, designing the VNS inboard shield is a complex engineering challenge that extends beyond merely blocking radiation; it must ensure the long-term structural and functional integrity of the TFC within a constrained space. This study therefore seeks an optimized shielding solution for this specific VNS inboard configuration by quantitatively comparing the shielding performance of candidate materials under identical modeling assumptions.

II. NEUTRONICS MODEL AND METHODOLOGY

II.A. VNS Computational Geometry and Source Definition

The neutronic analyses in this study were conducted using the VNS neutronic model shown in Fig. 1. The VNS MCNP geometry was not generated through a single, uniform computer-aided-design (CAD)-to-constructive solid geometry (CSG) conversion of a complete engineering CAD model. Instead, component-level neutronic geometries were prepared by the teams responsible for the respective components using McCAD [6] and GEOUNED [7], with the conversion route selected according to the individual component and its source CAD/model requirements.

The resulting CSG component models were organized as MCNP universes and integrated into the 120-deg-sector model through universe/FILL definitions. This modular component-wise workflow is consistent with the integrated VNS neutronic model development described in Leichtle et al. [8] and allows independently prepared component models to be assembled and updated within a common transport model.

Geometry consistency was checked after integration and before transport calculations. The model represents the multilayered inboard region from the SB, through the VV to the TFC. As no finalized VNS test blanket concept has been established yet, the breeder blanket slice visible in Fig. 1 is a homogenized water-cooled lithium lead-based representation provided by the EUROfusion neutronic coordination for the integrated VNS model, which was

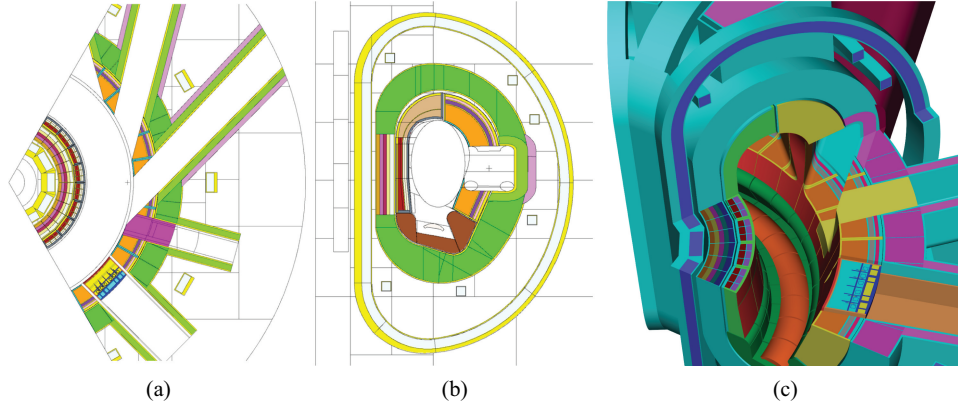


Fig. 1. VNS neutronics models: (a) X-Y plan view, (b) X-Z plan view, and (c) cut through the assembled MCNP CSG model rendered with Oak Ridge National Laboratory Radiant [11]. Apparent overlaps in the outboard region of (c) are rendering artifacts of the transparently displayed neutral beam injection liner and homogenized electron cyclotron launcher and do not indicate overlaps in the transport geometry.

retained unchanged in all candidate-material cases; its contribution to neutron backscattering is therefore consistently included in the comparative calculations.

Calculations were performed with MCNP version 6.3 [9]. The JEFF-4.0 continuous-energy ACE nuclear data library at 300 K was used for both neutron and photon transport [10]. JEFF-4.0 at 300 K is the standard nuclear data library of the EUROfusion VNS neutronic activity and was adopted here on that basis. In each material case, the shielding material constituent of all SB and VV layers in the inboard equatorial midplane region (Fig. 2) was replaced by the same candidate material, while the stainless steel and water fractions and all other components were kept unchanged. The neutron source was represented by nominal 14.1-MeV D-T neutrons with a spatial emission distribution derived from plasma simulations. The monoenergetic source representation is a scoping approximation that neglects finite D-T spectral broadening; because the same source definition (SDEF) was used for

every candidate, it is considered appropriate for the relative material screening performed in this work.

Space- and energy-dependent weight windows for variance reduction were generated using ADVANTG 3.2.0 [11]. The SDEF was normalized to unit probability, and the tally results per source particle were scaled to the D-T source strength corresponding to 29-MW fusion power.

II.B. Candidate Shielding Materials

The 10 candidate materials evaluated in this study were selected based on their unique nuclear and physical properties, reflecting various strategies considered in fusion shielding design. The key properties and characteristics of the candidate shielding materials are summarized in Table 2.

1. *Tungsten boride (W_2B_5), tungsten carbide (WC), tungsten (W)*: These W-based materials feature a high-

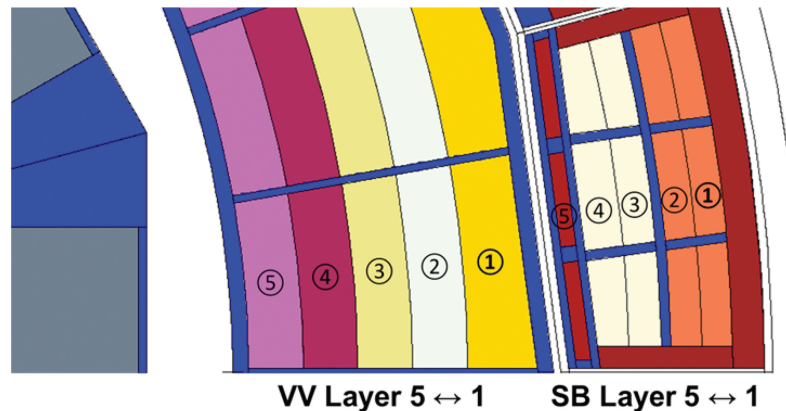


Fig. 2. Homogenized SB and VV shield layer structure at the inboard equatorial midplane.

atomic-number (high- Z) element. Tungsten is highly effective for gamma-ray shielding due to its high electron density and also contributes to fast neutron energy degradation via inelastic scattering [5]. Pure W serves as a reference material, while WC offers favorable thermomechanical properties for environments requiring structural stability. W_2B_5 is a multifunctional shielding material that combines the neutron and gamma attenuation of W with the high thermal neutron absorption cross section of boron (B). Under neutron irradiation, W can transmute into rhenium (Re) and osmium (Os), potentially contributing to long-term hardening and embrittlement. Cost and supply chain stability are also relevant for bulk application in fusion devices.

2. *Stainless steel [Type SS316L(N)]:* Austenitic or ferritic-martensitic stainless steels are standard materials extensively used for structural and shielding purposes in fusion devices. They offer moderate shielding performance, excellent structural properties, and the advantage of proven manufacturing technology. However, their shielding efficiency per unit thickness is lower than that of the other specialized materials considered in this study. It is often used in the form of borated steel to enhance its thermal neutron absorption capabilities; however, it should be noted that B itself is ineffective at attenuating fast neutrons, so borated steel acts primarily on the thermalized portion of the spectrum rather than the fast neutron component.

3. *Boron carbide (B_4C):* A lightweight ceramic material characterized by very high hardness and thermal stability. Its primary shielding function is the absorption of thermal neutrons via the $^{10}B(n,\alpha)^7Li$ reaction, which has a very large cross section. Like borated steel, B_4C is ineffective for fast neutron shielding on its own. Furthermore, being composed of low-atomic-number elements, its gamma-ray shielding capability is very weak. Consequently, it is typically employed in composite or multilayer structures alongside materials capable of moderating fast neutrons and attenuating gamma rays.

4. *Metal hydrides (TiH_2 , ZrH_2 , and HfH_2):* Complementary to thermal neutron absorbers like B-based materials, metal hydrides provide fast neutron moderation. Their high hydrogen (H) atom density enables significant neutron energy loss through elastic scattering. However, maintaining hydride stability under high-temperature and high-irradiation conditions presents a technological challenge. Hafnium hydride (HfH_2) is particularly interesting as a multifunctional shielding material, where H provides moderation while

hafnium (Hf) contributes gamma-ray attenuation and thermal neutron absorption. Hafnium is produced in limited quantities as a by-product of zirconium (Zr) production, contributing to its high cost and supply limitations.

5. *Tantalum (Ta):* A high-melting-point refractory metal similar to W, boasting excellent corrosion resistance. Because of its high atomic number, it exhibits superior shielding performance against gamma rays and fast neutrons. However, a major concern is its high activation potential, producing the radioactive isotope ^{182}Ta upon neutron absorption, as well as its high cost.

6. *Gadolinium oxide (Gd_2O_3):* Gadolinium, particularly the isotope ^{157}Gd , possesses the highest thermal neutron capture cross section (approximately 254 000 b) among all stable isotopes, making it an exceptionally effective thermal neutron absorber. As an oxide of a high-atomic-number rare earth element, it also contributes to gamma-ray shielding. It is primarily used in composite shielding materials, dispersed within a matrix such as polymers.

The diverse properties of these materials stem from the fact that an effective shielding system against mixed neutron and gamma fields must perform multiple functions rather than a single one. The key challenge in shielding design is how to optimally combine the three core functions—fast neutron moderation, thermal neutron absorption, and gamma-ray attenuation—within a limited space. The candidate materials selected in this study represent various approaches to solving this challenge.

II.C. Shielding Layer Structure and Composition in the Inboard Equatorial Midplane Area

The inboard shielding system in the VNS model is a multilayered structure composed of the SB and the VV. The baseline design of these components consists of several layers with a homogeneous mixture of materials, primarily stainless steel and water, designed to provide initial radiation attenuation. The specific composition of each layer is detailed in Table 1. In Table 1, 95% of theoretical density (TD) is applied uniformly to the solid candidate materials. This value represents a practical packing factor together with the assembly and manufacturing tolerances typical of bulk shielding structures, avoiding an overestimation of the attenuation performance.

For this comparative study, the shielding material constituent of all 10 SB and VV layers listed in Table 1, located in the inboard equatorial midplane region shown in Fig. 2, was replaced by the same candidate

TABLE 1
Material Fractions of Each SB and VV Shielding Layer in the Baseline Design

Layer Identifier		Shield Blanket					Vacuum Vessel				
		1	2	3	4	5	1	2	3	4	5
Composition by volume percentage	Shielding material	72.73	72.73	73.83	73.83	94.67	85.94	87.87	87.60	87.29	83.27
	Stainless steel	12.21	12.21	12.00	12.00	5.33	5.90	6.03	6.20	6.39	7.24
Percentage of TD	Water	15.06	15.06	14.17	14.17	0	8.16	6.10	6.20	6.32	9.49
	Shielding material ^a	95	95	95	95	95	95	95	95	95	95
	Stainless steel	100	100	100	100	100	100	100	100	100	100
	Water	100	100	100	100	100	100	100	100	100	100

^aThe 95% of TD applied to the solid shielding materials is the assumed packing factor; stainless steel and water are taken at their nominal densities.

material in each simulation case. The stainless steel and water volume fractions of every layer were kept at the values given in Table 1, and all components outside this region were left unchanged. This procedure isolates the relative influence of the candidate material without changing the geometry or the volume fractions of the baseline design.

II.D. Performance Evaluation Metrics and Radiation Limits

The three metrics and the VNS screening limits used here follow the magnet and neutronics design guidelines established for VNS and reported in the published device studies [8,12]. ITER insulation data [13] are cited as additional design context.

1. *Neutron flux ($n/cm^2 \cdot s$)*: The total neutron flux in the first TFC winding pack layer. This is a fundamental indicator of the overall intensity of the radiation field reaching the magnet.

2. *Nuclear heating/power density (W/m^3)*: Energy deposited in the TFC winding pack by neutrons and secondary gamma rays. The VNS screening target for the first winding pack layer is $444 W/m^3$ [12]. Published VNS neutronics studies provide consistent supporting context [8].

3. *Dose on TFC epoxy ($MGy/10 FPY$)*: Accumulated absorbed dose in the TFC insulation over the 10 FPY reference full-power exposure. A VNS screening limit of $50 MGy/10 FPY$ is adopted for the TFC insulation [12]. For comparison, the insulator specification for the ITER TFC is based on $10 MGy$, the more conservative of the two [13].

III. COMPARATIVE ANALYSIS OF SHIELDING PERFORMANCE

The shielding performance of the 10 candidate materials was compared across neutron flux, nuclear heating, and TFC insulator absorbed dose. The results reveal distinct differences stemming from the unique nuclear properties of each material and underscore the importance of mixed-field shielding behavior in the compact VNS inboard region.

The responses were obtained from coupled neutron-photon MCNP calculations using superimposed mesh tallies restricted to the inboard equatorial midplane

TABLE 2
Summary of Key Properties for Candidate Shielding Materials*

Material	Chemical Formula	Density (g/cm ³)	Primary Shielding Mechanism	Key Thermomechanical Properties	Key Radiation Effects and Operational Issues
Tungsten boride	W ₂ B ₅	13.10	Inelastic scattering (n), thermal neutron capture (n, α), γ shielding	Melting point: 2365°C; high hardness	Transmutation, He/Li production (B)
Tungsten carbide	WC	15.63	Inelastic scattering (n), photoelectric effect/Compton scattering (γ)	Melting point: 2870°C; high hardness, excellent thermomechanical stability	Transmutation, Co activation if Co binder is used
Tungsten	W	19.25	Inelastic scattering (n), photoelectric effect/Compton scattering (γ)	Melting point: 3422°C; high thermal conductivity	Transmutation to Re/Os, embrittlement
Stainless steel	SS316L(N)	8.00	Elastic/inelastic scattering (n), γ shielding	Melting point: 1375°C to 1400°C; excellent structural properties, proven manufacturing technology	Activation, irradiation embrittlement
Boron carbide	B ₄ C	2.52	Thermal neutron capture (n, α)	Melting point: 2450°C; high hardness, lightweight, thermal stability	He production, low γ shielding performance
Titanium hydride	TiH ₂	3.76	Elastic scattering (n, moderation)	Dissociation limit: ~400°C; relatively low thermal stability	Hydrogen dissociation at high temperatures
Zirconium hydride	ZrH ₂	5.60	Elastic scattering (n, moderation)	Dissociation limit: ~550°C; higher thermal stability than TiH ₂	Hydrogen dissociation at high temperatures
Hafnium hydride	HfH ₂	11.40	Elastic scattering (n), thermal neutron capture (n), γ shielding	Dissociation limit: ~700°C; higher thermal stability than ZrH ₂	Hydrogen dissociation at high temperatures, high cost, limited supply
Tantalum	Ta	16.69	Inelastic scattering (n), photoelectric effect/Compton scattering (γ)	Melting point: 3017°C; excellent corrosion resistance	High activation potential (¹⁸² Ta), high cost
Gadolinium oxide	Gd ₂ O ₃	7.41	Thermal neutron capture (n), γ shielding	Melting point: 2420°C; very high thermal neutron absorption cross section	Secondary gamma-ray production

*Note on density values: All entries are nominal room temperature TDs (100% TD). As stated in Table 1, a uniform packing factor of 95% TD is applied to these nominal values in the MCNP shielding models. Note on thermal limits: Standard melting points are reported for refractory metals, ceramics, and steels. For metal hydrides (TiH₂, ZrH₂, and HfH₂), thermal dissociation limits (dehydrogenation thresholds) are listed instead of melting points, as H outgassing occurs well below the melting temperature of the base metal.

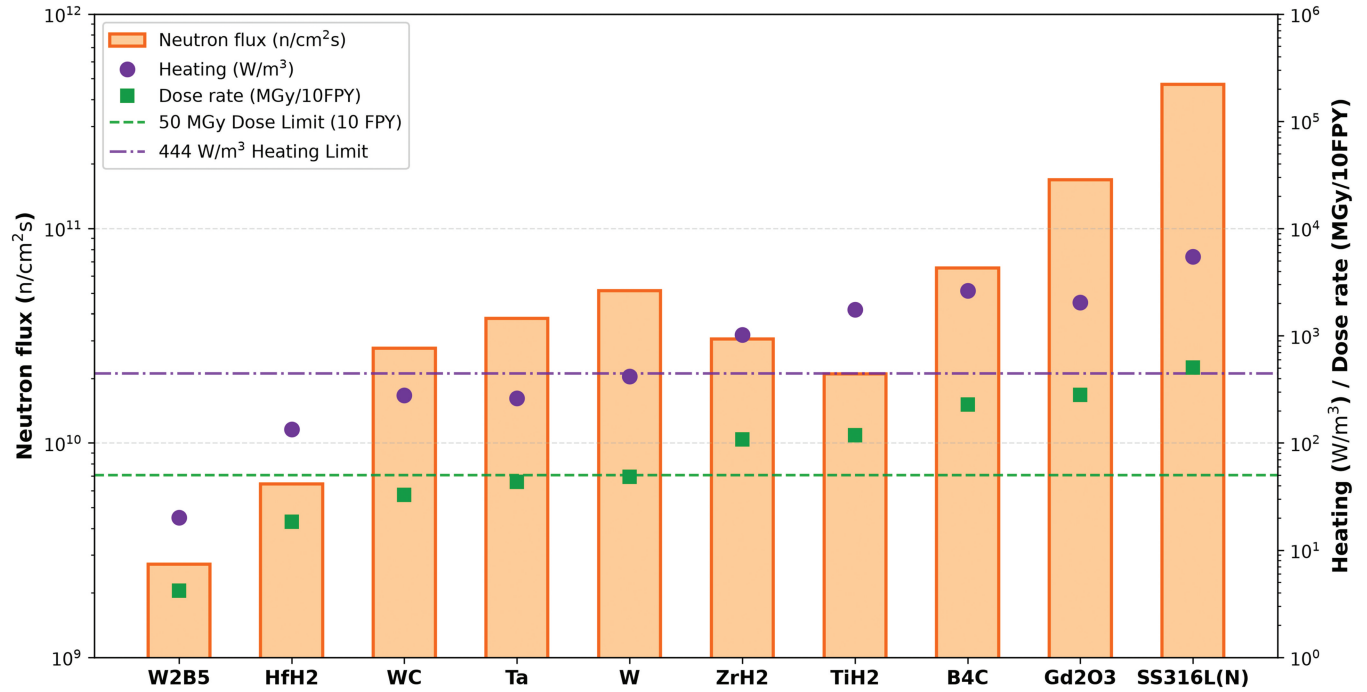


Fig. 3. Comparative shielding performance of candidate materials: neutron flux, nuclear heating, and accumulated absorbed dose in the TFC epoxy insulator. Horizontal lines indicate the 444 W/m³ nuclear heating and 50 MGy/10 FPY dose screening limits. Materials are ordered by increasing epoxy dose.

region, i.e., the shielding path investigated in Fig. 2. The neutron flux, the TFC nuclear heating, and the epoxy insulator dose were scored on a mesh with a voxel size

of the order of 1 cm³, and the values reported in Fig. 3 and Table 3 are the peak voxel values within the first TFC winding pack layer.

TABLE 3

Summary of Shielding Material Performance for the VNS Inboard Region

Material	Chemical Formula	Neutron Flux (n/cm²·s)	Nuclear Heating (W/m³)	Absorbed Dose in TFC Epoxy (MGy/10 FPY)	Potential to Meet 50-MGy Limit ^a
Tungsten boride	W ₂ B ₅	2.725×10^9	20.1	4.2	Very high
Hafnium hydride	HfH ₂	6.434×10^9	134.0	18.4	Very high
Tungsten carbide	WC	2.771×10^{10}	277.5	32.7	High
Tantalum	Ta	3.806×10^{10}	261.6	43.2	High
Tungsten	W	5.131×10^{10}	418.0	48.3	High
Zirconium hydride	ZrH ₂	3.055×10^{10}	1016.5	107.6	Moderate/low
Titanium hydride	TiH ₂	2.100×10^{10}	1749.8	117.6	Moderate/low
Boron carbide	B ₄ C	6.551×10^{10}	2632.2	228.7	Very low
Gadolinium oxide	Gd ₂ O ₃	1.687×10^{11}	2046.7	279.6	Very low
Stainless steel	SS316L(N)	4.690×10^{11}	5442.0	503.5	Very low

^aNote on potential to meet 50-MGy limit rating:

Very high (≤ 25 MGy/10 FPY): The calculated dose is at or below half of the 50-MGy screening limit, providing at least a factor of 2 dose margin in the tested configuration.

High (> 25 to ≤ 50 MGy/10 FPY): The calculated dose satisfies the 50-MGy screening limit, but with less than a factor of 2 margin.

Moderate to low (> 50 to ≤ 150 MGy/10 FPY): The calculated dose exceeds the stand-alone screening limit but remains within a factor of 3 of the target; such materials may still have a role in a deliberately optimized multimaterial configuration.

Very low (> 150 MGy/10 FPY): The calculated dose exceeds three times the 50-MGy screening limit in the tested stand-alone replacement configuration.

All tally outputs were normalized to the D-T source strength corresponding to 29-MW fusion power, and the insulator dose rate was integrated over the 10 FPY reference exposure. The relative statistical error of the reported values is below 1% for the neutron flux and below 3% for the coupled neutron-photon responses. Accordingly, the values in Fig. 3 and Table 3 characterize the peak local response at the inboard equatorial midplane and should not be interpreted as whole-coil averages or as global maxima over the entire TFC. This spatial restriction is an intentional screening assumption that enables a consistent material-to-material comparison under the same geometry and source conditions. Candidate materials in Fig. 3 and Table 3 are ordered by increasing accumulated epoxy dose, whereas material lists in each subsection are ordered from best to worst according to the metric discussed.

III.A. Neutron Flux Attenuation Performance

Total neutron flux in the first TFC winding pack layer serves as an indicator of the overall intensity of the radiation field penetrating the shield. As illustrated in Fig. 3 and detailed in Table 3, W_2B_5 , HfH_2 , TiH_2 , WC, ZrH_2 , Ta, and pure W proved to be the most effective materials for attenuating neutron flux. This superior attenuation stems from two distinct physical mechanisms:

1. *Inelastic scattering (high-Z elements)*: Materials such as W_2B_5 , WC, Ta, and W contain high-atomic-number elements with large inelastic scattering cross sections, which effectively degrade the energy of 14.1-MeV fast neutrons.

2. *Elastic scattering (hydrogenous moderators)*: Notably, the metal hydrides (TiH_2 , ZrH_2 , and HfH_2) exhibited remarkable neutron flux reduction due to their high H atom densities, which moderate high-energy neutrons through elastic scattering collisions with H nuclei.

In contrast, materials such as SS316L(N), B_4C , and Gd_2O_3 resulted in a relatively high transmitted neutron flux. Although B_4C and Gd_2O_3 possess exceptionally high absorption cross sections for thermal neutrons, their interaction cross sections for fast neutrons are comparatively small. Consequently, at a fixed thickness, they cannot sufficiently moderate or attenuate the hard fusion neutron spectrum without complementary fast neutron moderators.

III.B. Reduction Effect of Nuclear Heating in TFC

Nuclear heating, which occurs as neutrons and secondary gamma rays deposit energy in the TFC, is a critical constraint for the cryogenic cooling system. When ranking

the cases by nuclear heating, the most effective materials were W_2B_5 , HfH_2 , Ta, WC, and W, in that order. Only these five candidates remain below the 444 W/m^3 screening target introduced in Sec. II.D; the remaining materials exceed it by factors of approximately 2 to 12.

It is noteworthy that neutron flux attenuation and nuclear heating reduction are not directly proportional. The physical rationale for these discrepancies lies in secondary-gamma-ray production and photon attenuation capabilities:

1. *Why Ta outperforms WC and W in heating reduction*: Although Ta transmitted a marginally higher total neutron flux than WC and W, it achieved lower nuclear heating (261.6 versus 277.5 and 418.0 W/m^3 , respectively). This is due to Ta's high atomic number and dense electronic structure, which provide superior photoelectric and Compton scattering attenuation against secondary gamma photons generated within the shield.

2. *Why metal hydrides (TiH_2 , ZrH_2) exhibit higher heating despite low neutron flux*: Although TiH_2 and ZrH_2 successfully moderated fast neutrons, the resulting thermalized neutrons were captured by structural metals, emitting high-energy secondary gamma rays. Lacking high-Z elements, TiH_2 and ZrH_2 could not efficiently absorb these secondary gammas, leading to elevated heating levels in the TFC.

3. *Why Gd_2O_3 and B_4C exhibit high heating*: Gd_2O_3 undergoes intense radiative capture, including $^{157}\text{Gd}(n,\gamma)$ reactions that generate secondary gamma cascades, while B_4C has limited capability to attenuate penetrating photons.

These findings underscore that an effective fusion shielding material must act as a mixed-field shield, balancing fast neutron moderation, thermal neutron capture, and secondary gamma attenuation.

III.C. Absorbed Dose and Lifetime Assessment of the TFC Epoxy Insulator

Radiation-induced degradation of the epoxy insulator is one of the primary lifetime constraints for the TFC. In this study, 10 FPY is the reference full-power design exposure used for the TFC lifetime assessment. Fig. 3 and Table 3 show the accumulated absorbed dose in the epoxy insulation over this period. Rather than repeating all numerical values already shown in the figure and table, the candidates are grouped into screening tiers as follows:

1. *Top-performing group (W_2B_5 , HfH_2 , WC, Ta, and W)*: All five materials keep the accumulated insulator dose below the 50 MGy/10 FPY screening limit in the tested configuration. W_2B_5 provides the largest dose margin, followed by HfH_2 , WC, Ta, and W.

2. *Moderate- to low-performing group [TiH_2 , ZrH_2 , B_4C , Gd_2O_3 , and $SS316L(N)$]*: These materials exceed the 50-MGy threshold when used as the tested replacement material. Their performance should not be interpreted solely from total neutron flux, as the epoxy dose includes energy deposited by both neutrons and secondary photons.

Consequently, neutron flux and dose rankings may not coincide. Hydrogenous materials like TiH_2 and ZrH_2 effectively reduce fast neutron flux but shift the spectrum toward lower energies, increasing the probability of capture reactions and associated secondary gamma production in surrounding structures. Gd_2O_3 is an efficient thermal neutron absorber but can generate intense capture gamma emission, while B_4C has weak gamma attenuation and limited fast neutron moderation as a stand-alone shield. In contrast, W-containing and other high-Z candidates combine neutron energy degradation with stronger attenuation of secondary photons, helping to reduce the total absorbed dose in the TFC insulation.

Interpretation of the Table 3 screening assessment is as follows. The qualitative labels are based solely on the calculated 10 FPY epoxy dose and its margin relative to the 50 MGy VNS screening limit; they are not stand-alone material qualification criteria and do not account for thermomechanical, activation, manufacturability, or cost considerations. A material exceeding 50 MGy in this single-configuration screening is therefore not automatically excluded from a future multimaterial design, but it fails to meet the dose target as a stand-alone replacement in the configuration tested here.

IV. CONCLUSION AND IMPLICATIONS

The neutronics screening narrows the candidate pool for the VNS inboard shield. W_2B_5 , HfH_2 , WC, Ta, and W provide the most favorable combined performance across neutron flux, nuclear heating, and TFC insulator dose in the configuration investigated. The results also demonstrate that no single nuclear property is sufficient: Effective compact shielding requires an appropriate balance of fast neutron energy degradation, neutron absorption, and secondary gamma attenuation.

Final material selection cannot be based on nuclear performance alone. Engineering implementation must also account for thermomechanical behavior, manufacturability, activation, material availability, cost, compatibility with cooling and structural functions, and the achievable packing or joining concept.

1. *W_2B_5 , W, and WC*: Among the W-based candidates, W_2B_5 provides the strongest overall nuclear shielding performance in the current screening. WC also performs well and can offer favorable structural strength and stability [5]. Pure W remains an effective high-Z shield, but in the tested configuration, it results in higher TFC nuclear heating and epoxy dose than W_2B_5 and WC.

2. *HfH_2* : It exhibits remarkable versatility by performing neutron moderation, absorption, and gamma-ray shielding simultaneously. However, its use at the required scale is significantly hindered by operational challenges, namely, potential H release at high temperatures, alongside supply constraints and high cost as a by-product of Zr production [14].

3. *Tantalum*: It exhibits shielding performance comparable to W, but its high activation potential, producing long-half-life radioactive isotopes (^{182}Ta) upon neutron irradiation, could impose significant burdens on maintenance and waste disposal [2].

Accordingly, W_2B_5 , HfH_2 , WC, Ta, and W are retained as priority candidates for the next VNS shielding design iteration. Their use will be determined through a trade-off study that combines the neutronics results presented here with engineering and material constraints.

The present study was deliberately structured as a comparative material-screening exercise: The same allocated shielding material fractions in the SB and VV were replaced by one candidate at a time to isolate material effects. The results suggest that a future optimized solution may benefit from a multimaterial or graded arrangement, but the present calculations do not by themselves demonstrate a final two-material SB/VV configuration. In particular, the poor stand-alone performance of B_4C illustrates that a strong thermal neutron absorber is not necessarily an effective bulk shield for an initially hard fusion neutron spectrum unless sufficient moderation and photon attenuation are provided.

The rationale for a graded arrangement is not nuclear performance alone. Deploying a single high-performing W-based material throughout the inboard build would

shield well beyond the screening targets but at a penalty in cost, material availability, and manufacturing complexity. Distributing the shielding functions across the radial build allows the required margins to be met while limiting the amount of the most demanding material. On this basis, and following the shielding mechanisms observed above, the candidate functional roles carried forward to the subsequent design iteration are the following:

1. *SB region (plasma facing)*: A dense W-based material such as W_2B_5 or WC can provide fast neutron energy degradation through inelastic scattering together with substantial gamma attenuation.

2. *VV/magnet-facing region*: Once the spectrum has been degraded by the plasma-facing layer, moderation and thermal neutron absorption become significantly more effective. Materials such as B_4C or metal hydrides are therefore considered as components of an engineered mixed-material arrangement, combined with sufficient high-Z photon attenuation, rather than as stand-alone layers.

These functional roles identify the candidates carried forward into an improved VNS shield design rather than configurations demonstrated in this paper. A full coupled neutron-photon assessment of that updated design is currently underway. Because the geometry, layer thicknesses, and material distribution are being optimized simultaneously, those results fall outside the scope of the present comparative screening and will be reported separately. That analysis will determine whether the favorable material trends identified here translate into adequate shielding margins at an acceptable engineering cost in a realizable VNS design.

Acknowledgments

This work was carried out within the framework of the EUROfusion Consortium and has received funding from the European Union through the Euratom Research and Training Programme (Grant Agreement No. 101052200-EUROfusion). Views and opinions expressed are however those of the author (s) only and do not necessarily reflect those of the European Union or the European Commission. Neither the European Union nor the European Commission can be held responsible for them.

Author Contributions

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Disclosure Statement

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

Funding

The work was supported by the EUROfusion Consortium [101052200].

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