Fusion Power Plant Optimisation Through System-Code Assessment of Advanced Technologies

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Abstract

The 2025 EUROfusion Low Aspect Ratio (LAR) Gate Review established a validated reference configuration for a European demonstration reactor [1] through an integrated physics-engineering design process. As the European fusion roadmap evolves towards a **Pilot Plant** and, ultimately, a **Fusion Power Plant** (FPP) [2], future stakeholder requirements and design assumptions are expected to evolve beyond those adopted for DEMO. The present work therefore uses the DEMO LAR configuration as a validated benchmark to quantify the **system-level benefit** of emerging technologies, rather than as a proposed FPP design.

Updated neutronic, thermohydraulic and structural models [3] were implemented within the PROCESS systems code to investigate advanced cryogenic structural materials for superconducting magnets, alternative neutron shielding materials, and concept-specific optimisation enabled by different breeding blanket solutions. Optimisation around a selected breeding blanket concept illustrates the additional flexibility that could be enabled by blanket qualification in a dedicated Pilot Plant (e.g. VNS [4]). Under the current stakeholder requirements [5], the combined assessment enables reactor configurations of major radius $R_0 \leq 8$m. Beyond the specific design solutions obtained, the proposed workflow provides a reusable methodology that can be directly applied as stakeholder requirements evolve, enabling emerging technologies quantitative assessment, supporting FPP design and R&D prioritisation.

1. Introduction

The 2025 EUROfusion LAR Gate Review established a new reference configuration for a European demonstration reactor through an integrated physics-engineering design process, satisfying the current stakeholder requirements while significantly improving operational margins with respect to previous DEMO design iterations [1]. The resulting configuration, developed under the present stakeholder requirements ($300 \leq P_{el,net} \leq 500$ MWe, $\tau_{pulse} \geq 2$ [5]), has a major radius $R_0 = 8.6$ m and aspect ratio $A = 2.8$ [1]. It represents the current European reference for integrated reactor studies, while its operational space is being further consolidated through ongoing high-fidelity plasma physics analyses.

At the same time, the European fusion roadmap is evolving towards a Pilot Plant followed by a commercial Fusion Power Plant (FPP) [2]. Future stakeholder requirements are therefore expected to evolve, reflecting different operational objectives, technology maturity and economic considerations. Consequently, the present work does not attempt to define a future FPP baseline. Instead, the DEMO LAR configuration, which integrated physics and engineering solutions, is adopted as a common benchmark for assessing how emerging technologies influence reactor design at system level. An important distinction between demonstration reactors and future FPPs concerns the breeding blanket strategy. DEMO must preserve flexibility to accommodate different blanket concepts for technology qualification, whereas blanket qualification in a dedicated Pilot Plant could enable future FPPs to optimise the reactor architecture around a selected breeding blanket concept or future innovative blanket solutions,

removing demonstration-driven design compromises.
Within this framework, three representative technology areas expected to provide significant reactor-level impact are investigated:

- advanced cryogenic structural materials for superconducting magnets;
- alternative neutron shielding materials;
- concept-specific optimisation enabled by different breeding blanket solutions.

The resulting workflow separates technology assumptions from stakeholder requirements, allowing future FPP design spaces to be explored consistently as both technologies and stakeholder requirements evolve.

2. System-level assessment of emerging technologies

2.1 Advanced cryogenic structural materials for superconducting magnets

The DEMO LAR Gate Review adopted SS316LN as the structural material for both the Toroidal Field (TF) and Central Solenoid (CS) coils, following a conservative approach prioritising technology readiness. However, advanced cryogenic steels currently under development for next-generation fusion magnets offer substantially higher allowable stresses and fracture toughness, enabling reductions in structural dimensions while increasing the operational capability of the magnetic system.
Two representative materials were investigated: **JK2LB** developed for the ITER Central Solenoid conductor jackets, and **N50-H**, a high-strength nitrogen-alloyed austenitic steel currently considered for high-field magnet applications, including ITER, where the nitronic series has been used for the Central Solenoid (CS) tie plates, and CFS studies for SPARC, among others. The interest in such high-performance steels is therefore shared across international R&D efforts, with ongoing activities in Europe, Japan, China, and the US. Updated material properties were implemented within the PROCESS systems code.
Two complementary assessments were performed under the current stakeholder requirements. First, the reactor geometry was maintained unchanged to quantify the improvement in pulse capability resulting solely from the enhanced structural performance. Secondly, the reactor major radius was re-optimised while maintaining the prescribed stakeholder requirements.
At constant geometry, replacing SS316LN with **JK2LB** in the CS increases the achievable pulse duration from approximately **2.2 h** to **2.8 h,** while adopting **N50-H** in the TF structures increases the pulse duration to approximately **2.7 h.** Combining both materials extends the pulse duration to approximately **3.4 h,** corresponding to an increase exceeding **50%** with respect to the reference design. This improvement results from the reduced structural fraction required to sustain electromagnetic loads, allowing additional magnetic flux to be stored within the CS.
When reactor size is minimized, the improved structural capability enables a reduction of the reactor major radius from **8.6 m** to approximately **8.45 m**. Beyond this point, the limiting design constraint shifts from the magnetic system to the divertor protection criterion, indicating that further compactness requires improvements in other reactor systems. Advanced cryogenic steels therefore provide significant operational benefits while contributing to reactor compactness and increasing overall design flexibility.

2.2 Alternative neutron shielding materials

Neutron shielding represents one of the dominant contributors to the reactor radial build. Consequently, improving shielding effectiveness directly increases the design freedom available for future FPP, either by reducing reactor dimensions or by reallocating recovered radial space to neighbouring systems.
The DEMO LAR reference configuration adopted conservative ITER-like shielding assumptions. In the present work, updated neutronic analyses were used to assess alternative shielding solutions based on **tungsten** (W), **boron carbide** (B4C) and **titanium hydride** (TiH2) [6]. These materials are well established in the fission nuclear industry. A neutronic analysis was performed scanning material/coolant fraction from 40% to 10%, keeping 6% steel as structural

part. The requirements on inner TF magnets shielding were established for a DEMO reactor as:

- Max volumetric power density in winding pack ≤ 50 W/m^3 [7] (ITER value is 1000 W/m^3)
- Max insulator dose at end-of-life ≤ 10 MGy (as ITER [8])
- TF neutron fluence at end-of-life ≤ 10^{22} neutrons/m^2 (as ITER [8])

Simplified models have been implemented within PROCESS while maintaining the same irradiation limits adopted for the superconducting magnets.

Among the investigated options, tungsten provides the largest improvement in shielding effectiveness. The achievable benefit depends strongly on the selected breeding blanket concept. For **WCLL,** where the blanket itself contributes significantly to neutron attenuation, the improved shielding enables recovery of approximately **200 mm** of radial build while preserving the adopted TF irradiation limits. For HCPB, the corresponding gain is approximately **40 mm**, reflecting the lower shielding capability of helium-cooled blanket concepts.

The recovered radial build can be exploited according to different optimisation strategies. It may be directly converted into a reduction of the reactor major radius, allocated to increase the available space for the CS, and for a reduction of the TF peak magnetic field by reducing the plasma-to-coil distance. At fixed reactor size, for example, recovering approximately **200 mm** of radial build in the WCLL configuration reduces the peak TF field by about **0.6 T**, while simultaneously providing approximately **90 Wb** of additional CS magnetic flux, corresponding to nearly **50 min** of extra pulse duration (at ≈30mV).

These results identify neutron shielding as one of the highest-impact technology areas for improving reactor compactness. More importantly, they demonstrate that shielding optimisation cannot be considered independently of the selected breeding blanket concept, naturally motivating the combined technology assessment presented in the following section.

3. Combined technology assessment

The individual technology developments presented in the previous sections were subsequently combined to evaluate their cumulative impact on reactor performance. The optimisation was performed starting from the current stakeholder requirements (**300 ≤ $P_{el,net}$ ≤ 500 MW, τ_{pulse} ≥ 2 h** [5]), considering representative breeding blanket concepts rather than seeking a unique optimum reactor configuration. The resulting design points are summarised in **Table 4.1**.

Inputs	WCLL	HCPB
CS steel	**JK2LB**	
TFs steel	**N50-H**	
VV thickness	40 cm (**-20 cm**)	56 cm (**-4 cm**)
BB thickness	80 cm	
Plasma-wall distance	15 cm (**-7 cm**)	
ECRH heating power	15 MW (+5 MW)	
ECRH current drive	0 MW (-40 MW)	
Energy multiplication factor	1.20	1.34
Thermal-to-electric efficiency	0.32	0.37
Results		
R_0 [m]	**7.9**	**7.9**
A [-]	**2.6**	**2.8**
P_{fus} [MW]	1458	1176
$P_{el,net}$ [MWe]	350	350
q_{95} ($q_{95_{min}}$)	3.6 (3.3)	3.6 (3.3)
τ_{pulse} [hr]	2.27	2.5
$P_{sep}B_{T,0}/q_{95}AR_0$ [MW.T/m]	5.5	5.5
$\dot{q}_{div,outer}$ [MW/m^2]	42.1	44.81
I_p [MA]	19.60	17.12
$B_{T,0}$ [T]	4.13	4.35
$B_{T,TF_{max}}$ [T]	10.09	10.47

Table 4.1 PROCESS results for a combination of advanced steels for the magnetic cage and reduction of VV thickness due to alternative shielding materials optimised for the two breeding blanket models WCLL, and HCPB.

Owing to its superior neutron shielding capability, the WCLL concept allows substantial reductions in dedicated shielding thickness, resulting in reactor configurations with $\mathbf{R_0 \approx 7.9m}$. Conversely, the higher thermal conversion efficiency of HCPB reduces the fusion power required to achieve the prescribed electrical output, enabling reactor configurations approaching also $\mathbf{R_0 \approx 7.9\ m}$.

A preliminary thermohydraulic study [9] assessed that the cooling requirements of the TF winding pack can be achieved until a vol. load of **450W/m^3,** instead of the 50W/m^3 [7] used above. This model was also implemented within PROCESS, resulting in a shielding thickness reduction (using W as the case above) of **≈21cm** for the WCLL and **10cm** for the HCPB (fluence becoming the limit). Applying the same optimization as in Table 4.1 allows a further decrease in major radius down to $\mathbf{R_0 = 7.9m}$ for WCLL, and **7.8m** for HCPB.

These results should not be interpreted as indicating the superiority of one blanket concept over another; rather, this methodology and the implemented integrated models, demonstrate that the optimum reactor configuration depends on the selected technology assumptions.

More generally, the study shows that reactor compactness cannot be attributed to a single technology development. Instead, it results from the cumulative contribution of complementary advances in structural materials, neutron shielding and concept-specific optimisation. The proposed workflow therefore enables quantitative exploration of future FPP design spaces while simultaneously identifying the technologies providing the largest reactor-level benefit.

4. Conclusions

The validated EUROfusion DEMO LAR configuration was adopted as a consistent benchmark for assessing emerging technologies relevant to future FPP. Rather than proposing a new reactor baseline, the present work demonstrates a technology-assessment methodology capable of quantifying the **system-level benefit of emerging technologies** under a prescribed set of stakeholder requirements.

The assessment shows that **advanced cryogenic structural materials**, **alternative neutron shielding solutions** and **concept-specific breeding blanket optimisation** provide complementary reactor-level benefits. Under the current stakeholder requirements, their combined implementation enables reactor configurations with $\mathbf{R_0 \approx 7.8\text{-}7.9\ m}$, while increasing engineering margins and operational capability, using the same assumptions as in [1].

More importantly, the methodology separates stakeholder requirements from technology assumptions. As future FPP requirements evolve, the same workflow can be directly reapplied to evaluate updated design assumptions, identify new reactor design solutions and quantify the reactor-level impact of emerging technologies. In addition to supporting future FPP design, the methodology provides a quantitative basis for prioritising fusion technology R&D by identifying those developments that deliver the largest system-level benefit.

5. References

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