Fusion Engineering and Design 211 (2025) 114796

Contents lists available at ScienceDirect

Fusion Engineering and Design

journal homepage: www.elsevier.com/locate/fusengdes

Engineering concept of the VNS - a beam-driven tokamak for component testing

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ARTICLE INFO

Keywords:
VNS
Volumetric neutron source
Breeding blanket testing
Demo
Tokamak
Remote maintenance

ABSTRACT

The qualification of in-vessel components for a fusion power plant requires a test environment with a high flux of 14 MeV neutrons over a sufficiently large surface and volume. Performance testing and qualification of the complex design and technologies of fusion nuclear components is needed, in particular that of the tritium breeding blanket (BB). Testing in relevant conditions over a relevant time will also allow gaining the necessary confidence regarding the build-up and control of tritium inventories inside the BB, which will be an important radioactive source. An option of such a volumetric neutron source (VNS) is a beam-driven tokamak. A feasibility study of the main machine components and associated plant systems is described in this article.

The machine has a major radius of 2.53 m, a single-null divertor configuration, and four tangential 120 keV beamlines that generate a fusion power of approximately 30 MW and provide current drive for a steady-state plasma scenario. The plasma is small with a minor radius of $a = 0.55$ m to maximize the neutron wall load, up to 0.5 MW/m^2, similar to what is targeted in ITER. Approximately 25 m^2 are available for blanket testing including 4 port plugs, which offer flexibility regarding the test module operating conditions and the implementation of instrumentation. Given the small plasma, much of the tokamak's volume is made up by the neutron shielding structures that are similarly sized as in ITER. To reduce the construction risk, ITER-like concepts were adopted for many components. In some cases, however, lessons learned from ITER led to the development of

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https://doi.org/10.1016/j.fusengdes.2024.114796
Received 26 November 2024; Received in revised form 31 December 2024; Accepted 31 December 2024
Available online 23 January 2025

customized or innovative concepts. Due to the modest fusion power the plasma will burn <1 kg of tritium per year, which can be provided from external sources.

1. Introduction

1.1. Mission and basic concept of VNS

VNS is proposed to be built and operated in parallel to other major fusion experiments aiming at enabling a demonstration fusion power plant, ITER [1], which aims at demonstrating burning plasma physics, and DONES [2], which aims at qualifying neutron radiation resistant structural materials. The main purpose of VNS is - complementary to ITER and DONES – namely the testing and qualification of entire in-vessel components built with essential fusion technologies [3]. The need for a VNS was identified as early as 1985 [4] and was then proposed in 1995 to complement ITER [5,6]. The challenge to reach the required plasma performance is much smaller in VNS as compared to ITER, because the energy of the plasma particles required for the fusion reaction is provided through an external neutral beam (NB) system rather than by reaching a high plasma temperature.

A VNS should be operated in a quasi-continuous mode rather than in short pulses, therefore the magnet coils should be superconducting to avoid excessive power consumption. In 1985 it was considered only linear devices would allow the integration of the massive neutron shields to sufficiently protect superconducting coils [4]. The design of such devices is described in [7]. A VNS tokamak device devised in 1985 foresaw Cu-coils, was operated in pulsed mode, and despite the choice of a moderate magnetic field, required a large power consumption [4]. Later however, beam-driven VNS devices with superconducting coils were proposed as tokamaks [5] (as presented also in this article), and as a stellarator [8]. Also, fusion-fission hybrid tokamak devices of even smaller size were proposed [9,10].

In a VNS, relevant testing conditions are achieved through the performance requirements listed in Table 1. These are similar to those defined for the ITER test blanket module (TBM) program, in particular the intensity of the neutron flux i.e., the neutron wall load (NWL), of ≈0.5 MW/m^2 and long plasma pulses to reach thermal equilibrium conditions in the TBMs [1,5]. A significant difference to the ITER TBM program, however, is that VNS aims at reaching a power plant relevant neutron fluence, equivalent to a damage level in the first wall of 30–50 dpa. At the same time, the fusion power was minimized to allow relying on external supplies of tritium. The facility must also allow the test modules to be equipped with instrumentation for monitoring and to carry out in-service inspection. Also, the facility must be designed to enable the removal and replacement of the test modules by remote handling (RH) tools in a reasonably short time and for their transfer to hot cells for post-irradiation examination. Furthermore, it must be possible to operate the test modules with different coolants and with different coolant temperatures.

1.2. Testing options in VNS

Comprehensive assessments have been carried out in the past regarding testing needs and the execution of tests in a VNS focused on the BB [5,6]. These remain an important basis for the definition of the VNS testing strategy. The proposed testing and qualification approach for the BB is based on the Technology Readiness Level (TRL) methodology sequence [11], progressing from single-effect tests to testing of multiple effects / multiple interactions phenomena, and concludes with fully integrated test modules undergoing long test campaigns. It is expected that in VNS the BB can be tested up to TRL level 8 i.e., “system completed and qualified”, including the collection of relevant data to estimate the reliability of the BB.

The VNS presented here offers the relevant testing environment for fusion nuclear components and a large area to install BB TBMs i.e., approximately 25 m^2 on the outboard side (on the inboard side, due to the compact radial build, no test modules can be integrated). This includes four equatorial ports, that would allow test modules integrated in equatorial port plugs and equipped with a large number of sensors and cables for signal transmission much like foreseen in the ITER TBM program [12]. Other concepts for the implementation of TBMs in VNS are being discussed, too, aiming at maximizing the testing options, i.e. as vertical segments or in upper port plugs. In addition to the BB, most other in-vessel components required in a demonstration fusion power plant can be tested in VNS either naturally through their implementation or specifically through dedicated tests. VNS will also demonstrate the safe operation of a nuclear fusion device including a fuel cycle with high tritium throughput and the handling of radioactive components by remotely operated tools.

1.4. Operation and testing strategy

The testing strategy and the related VNS operation plan will be defined during the VNS engineering design phase when testing needs and possibilities are further clarified. Currently, it is proposed to break down VNS operations into three successive phases, see Table 2. To start the construction of DEMO the data obtained from VNS-OP1 is required, to start DEMO operation OP2 should be completed. VNS-OP3 can commence in parallel to the first operation phase of DEMO.

The availability figures assumed in Table 2 are nothing more than abstract projections, which were made based on the following considerations. These can be confirmed only when VNS has been realized. (i) Machine shutdowns for *planned* maintenance will reduce the availability to no less than approximately 70–80 %. (ii) Experience shows that first-of-a-kind machines require frequent *unplanned* shutdowns for maintenance due to failures particularly in the early operation phase. This includes also failures of test modules of the BB. (iii) The overall component reliability increases with time.

An acceleration of the VNS operation schedule is possible through the reduction of unplanned shutdowns. This is generally possible through an increased level of component qualification.

Table 1
VNS project requirements.

Lifetime neutron fluence (corresponding damage in EUROFER)	30 - 50 dpa
Neutron energy	14 MeV
Peak neutron wall load (NWL)	≥ 0.5 MW/m^2
Plasma-facing wall surface	> several m^2
Pulse length (to reach thermal equilibrium condition in TBMs)	> few mins.
Annual tritium consumption	< 1 kg
→ Fusion power limit	< 30 - 40 MW
Electrical power consumption	< 200 MW

Table 2
Tentatively proposed operation phases (OPs) of VNS with expected neutron damage levels and durations based on assumed values of (plasma) availability.

	Max. damage level (in steel)	Assumed availability	Duration
OP1	5 dpa	20 %	5 y
OP2	20 dpa	30 %	13.5 y
OP3	25 dpa	40 %	12.5 y
Total	50 dpa		31 y

1.4. RAMI considerations

A quantitative assessment of the reliability, availability, maintainability and inspectability (RAMI) of VNS is only possible after a conceptual design is developed. During the VNS feasibility study proven and robust technologies were chosen when available and margins were incorporated into the designs as deemed adequate to ensure the required high component *reliabilty*. For several VNS components however, existing technologies cannot be considered proven and R&D programs will be required prior to construction e.g., for high temperature superconductors (HTS), continuously operated NBI, RH tools, and all in-vessel components. Naturally, the various TBMs may be prone to failure. The recovery of the machine from such failures should not have a significant impact on the plant availability. ITER therefore requires a high reliability of its TBMs [12]. While VNS must be designed to recover from such events, the occurrence of such failures must be reduced as reasonably achievable.

To ensure *maintainability*, basic RH schemes were developed for the main IVCs and the NBIs, and the VNS machine and plant were adapted accordingly. Furthermore, the physical separation of plant systems carrying radioactive source terms from other components requiring regular *inspection* and maintenance reduces the exposure of personnel to radiation and should allow the required accessibility.

1.5. Development status of VNS

The outcome of the VNS feasibility study that has been underway in EUROfusion since early 2023 is encouraging. It must be noted, though, that all design descriptions are tentative and may require adjustment. Similarly, most design verifications must be considered preliminary as they are sometimes based on estimates of the loading conditions. Further development iterations will be required to ensure full consistency of the design with load assessments and verifications.

2. VNS tokamak design description

2.1. VNS main parameters

In VNS the fusion power, P_{fus}, is directly dependent on the power provided by the auxiliary plasma heating systems, P_{aux}, which are driving the plant power consumption. This must be reasonably limited in order not to over-constrain the site selection and the operating costs, Table 1. The modest level of P_{fus} was selected also to reduce the fuel consumption and hence the necessary tritium supply to VNS from external sources, which must be compliant with the tritium world market (approximately 2 - 3 kg/year from Canadian and South Korean CANDU reactors [13,14], sold at ≈30 M$/kg [15]). It is noteworthy that VNS could be operated at a power level below nominal by reducing P_{aux}, which would also reduce the NWL (approximately proportionally). Achieving the required high NWL with a modest P_{fus} led to minimizing the wall surface area and hence to a plasma major radius of approximately 2.5 m with a high aspect ratio, A, i.e. a small plasma minor radius, see Table 3. A high A also provides space on the inboard side for the integration of massive neutron shielding structures, the superconducting toroidal field (TF) coils providing a high magnetic field, and the central solenoid (CS) providing the magnetic flux for the inductive ramp up of the plasma current, see Fig. 1. However, sustaining the plasma current inductively for a flat top longer than ≈5 min is not possible due to space limitations for the CS. Hence, VNS relies on the NB system to drive the plasma current, I_p, making it a steady-state device whose pulse length is, in principle, unlimited. A single-null divertor configuration was chosen considering the well-developed physics basis from existing tokamaks. This avoids the need to integrate a second divertor in the upper part of the machine, which can therefore be dedicated to the RH of the blanket, see Section 2.7.

Table 3
VNS main tokamak parameters.

Major / minor radius, R / a	2.53 m / 0.55 m
Aspect ratio, A	4.6
Magnetic field (@ 2.53 m radius), B_0	5.4 T
Plasma elongation, k_{95}	1.35
Divertor configuration	Single null
Plasma current, I_p	1.76 MA
Fusion power, P_{fus}	29 MW
Pulse length	steady-state
Heating & current drive (H&CD)	
- Neutral beam (NB) power	42 MW
- Electron cyclotron (EC) power	10 MW

2.2. Plasma performance

The temperature in the relatively small bulk plasma of VNS is, like in existing tokamak machines such as JET or ASDEX-Upgrade, too low for d-T fusion to occur at a sufficiently high rate. In contrast, machines targeting burning plasmas like ITER [1] and DEMO [16], are much larger than VNS to achieve the necessary plasma confinement and will reach much higher P_{fus}. Therefore, to achieve a similarly high power density, i.e. NWL, VNS relies on a high power injected by the external heating systems, P_{aux}, compared to the size of the plasma. The plasma size of VNS can however not be arbitrarily small. There are at least three high-level limiting factors:

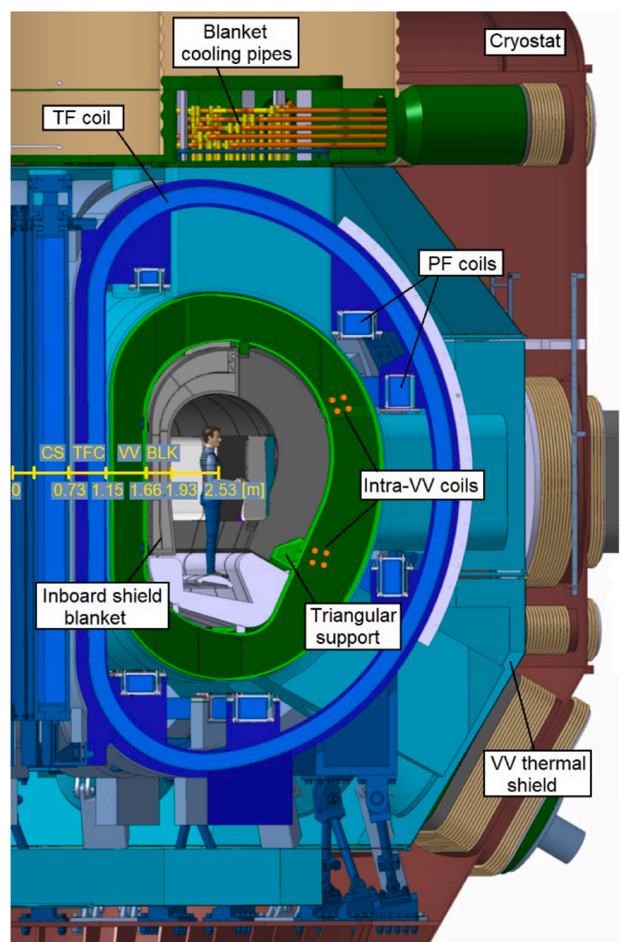

Fig. 1. Section view of the VNS tokamak incl. machine radial build. Abbreviations: TF coil (TFC), blanket (BLK).

- Magnetohydrodynamic (MHD) stability, characterized through β_N. A high-pressure plasma (i.e. a small plasma with large power injection) is prone to instabilities which can terminate the discharge and prevent a robust operation. The MHD stability improves with the magnetic field strength, hence the VNS magnets are designed to provide a B_0 of 5.4 T, which reduces β_N to 3.35 %Tm/MA, which is below the stability limit of 3.5 %Tm/MA [17].
- Power exhaust, i.e. the exhaust of the high auxiliary power, plus the alpha particles from the fusion reaction. A sufficiently large divertor volume is needed to reach detached condition, and the target plates must be inclined to spread the incoming power across a larger surface area [18]. The plasma analyses carried out in [18] show that the design of the VNS divertor allows the exhaust of the plasma power (leading to a heat flux of approximately 6 MW/m^2 on the target plates), with a throughput of the seeding gas (Kr) compatible with the plasma. Since VNS is dominated by external heating, in case of accidental divertor reattachment events, it is conceivable to rely on the swift turning off of P_{aux} to protect the divertor targets from excessive heat loads, but further analysis is needed for confirmation.
- Fast particle confinement. The VNS plasma has a very large population of fast particles i.e., fast D ions from NB injection (birth energy 120 keV) and alpha particles from the fusion reactions (birth energy 14.6 MeV). Their large gyro orbits together with their large drift orbits shifts and the width of their banana orbit are thus comparable to the plasma minor radius. To reduce prompt particle losses [19] and the consequent localized heat loads on the plasma facing components requires a sufficiently large plasma current, I_p, which in turn implies a high B_0 to ensure plasma stability. The approximate peak heat flux due to prompt losses of fast particles is found in [20] to be below 150 kW/m^2, i.e. well within the performance limit of the first wall. Two potentially critical aspects related to the impact of fast particles though require further assessment, namely armor erosion and the interaction of fast particles with MHD local instabilities.

2.3. Consequences of plasma disruptions

VNS does not target the study of plasma physics. It will run repeatedly one or few scenarios that can be expected to be robust against disruptions. Nonetheless, plasma disruptions are events that are expected to occur, and an analysis of possible disruption causes can be found in [21]. Hence, the consequences of disruptions are part of the VNS design basis:

- Thermal loads (in particular due to plasma-wall contact)
- Runaway electrons
- Electromagnetic (EM) loads

2.3.1. Thermal loads

No systematic analysis to determine the thermal loads in different areas of the VNS first wall during disruptions has been carried out so far because these are not expected to be a major issue. Indeed, the energy content in the VNS plasma is, similarly to existing tokamaks, too low to cause major damages (≈3 MJ according to METIS simulations [21]).

2.3.2. Runaway electrons

The VNS plasma current (I_p = 1.76 MA) is comparable to that achievable in ASDEX Upgrade, and significantly lower than that reached in JET. With the exception of dedicated experiments, which are not foreseen in VNS, both devices have been operated without generating significant levels of runaway electrons.

2.3.3. Electromagnetic loads

The vacuum vessel (VV) and IVCs are designed to withstand the most severe EM loads. For the VV the large halo currents are relevant that occur in slow vertical displacement events (VDEs). The most severe EM loads acting on the VNS IVCs occur during fast plasma disruptions. For the scaling of the minimum current quench (CQ) times observed in existing tokamaks to VNS, the plasma cross-sectional area is used as suggested in [22]. The minimum CQ time, t_{CQ}, was defined based on disruption data from ASDEX-Upgrade (AUG) and JET and considering also the wall material. The comparison of the current decay rate in machines with W and C walls, respectively, shows an increase of the shortest CQ time by approximately 33 % [23] because of the carbon contribution to the total radiated power. The fastest area-normalized current decay time, t_{CQ}/S, is approximately 3.3 ms/m^2 in AUG. A similar value is provided in [24] for disruptions in JET with a metallic wall. Given the cross-sectional area of the VNS plasma of 1.43 m^2, the fastest current quench time was defined as $t_{CQ} = 4.7$ ms. The consequent EM loads were determined using the finite element method [25].

2.4. Magnet system and plasma control

Superconductor technology is implemented in VNS to avoid excessive power consumption during the quasi steady-state operation. Consequently, a cryostat is built around the tokamak providing vacuum condition for the superconducting coils. The warm VV and cryostat surfaces are covered by thermal shields which are actively cooled at 80 K and lined with multi-layer insulation sheets to reduce heat radiation to the coils. The thermal shields are mounted to VV and cryostat, respectively. Depending on the readiness of HTS technology either conventional Nb_3Sn or HTS will be selected for the toroidal field (TF) coils, see Table 4.

As a consequence of the massive n-shielding structures the plasma equilibrium and control coils (PF coils) are located at a large relative distance from the plasma. This causes three main issues: (i) To generate the required (poloidal) equilibrium field the PF coils must have currents larger than the plasma current, up to 8 MA in a conventional arrangement with the PF coils located outside the TF coils. This is an indication of the unfavorable PF coil arrangement [26]. (ii) The PF coils have alternating current directions along the poloidal periphery, and effectively fewer degrees of freedom are available for the control of the plasma shape. (iii) The thick outboard blanket increases also the distance of the toroidally continuous VV from the plasma strongly affecting

Table 4
Main engineering features of VNS.

Tokamak	Single-null lower divertor 12 TF coils, 6 PF coils 12 upper, equatorial and lower ports 36 divertor cassettes 24 inboard, 36 outboard blankets
TF coils	NB_3Sn or HTS, welded TF case
CS	NB_3Sn or HTS, square cable in conduit conductor
PF coils	NB_3Sn/NbTi or HTS, square cable in conduit conductor
Vacuum vessel (VV)	Welded double-wall with poloidal ribs, internal n-shielding plates
Cryostat	Welded vacuum chamber with wall thickness ≈15 mm, removable top lid, pedestal ring supporting the tokamak, diameter × height: ≈12.5 *m* × 12.5 m
Divertor	36 cassettes, dome and 2 vertical targets mounted to each cassette, actively water-cooled
Shield blanket	Vertical segments and some outboard modules, internal n-shield blocks, internal ferritic steel plates, actively water-cooled
TBMs	4 equa. port plugs, several outboard blankets, possibly several upper port plugs
Aux. systems	• 3 + 1 NB injectors (42 MW) • EC launcher (10 MW) • Pellet injectors • Up to 8 torus cryopumps • Plasma diagnostics • In-vessel viewing system

the vertical stability performance of the tokamak. As a compensation measure the plasma elongation is relatively low and a triangular support is integrated into the VV structure, see Fig. 1.

Two concepts have been developed to position the PF coils closer to the plasma i.e., inside the TF coils:

a) An unconventional tokamak assembly concept that foresees the PF coils to be arranged around the VV as in some tokamaks with Cu-coils e.g., MAST [27]. This requires the in-situ fabrication of the TF coils, which is described in [28] and, to avoid the need for a heat treatment of the superconductor after winding, is based on HTS or Nb_3Sn with react and wind technology. The maximum current in individual PF coils is reduced with this concept to ≈ 4 MAturns.
b) Superconducting window-frame coils are arranged around the equatorial ports. To enable assembly, these are connected along their poloidal legs to form a ring structure that can be rotated by 15° prior to the attachment of the equatorial port extensions.

Both options are feasible to create the plasma equilibrium and allow the control of the plasma shape. To limit the active power required by the PF coils to control the plasma vertical stability, additional Cu-coils are foreseen in-between the VV shells on the outboard side above and below the equatorial ports, see Fig. 1. These must each carry total currents up to ≈ 100 kAturns and are made of a wound Cu-conductor with MgO insulation inside a stainless-steel pipe. Due to the comparably large Cu cross-section the conductors can be cooled by the VV coolant, which they are submerged in Given the absence of active cooling inside the conductor, the good protection inside the VV, and the relatively low neutron fluence, a very high reliability of the unmaintainable Cu-coils is expected.

The risk of relying on the unconventional coil arrangement is recognized. Therefore, a final decision will be made only after the completion of an ongoing manufacturing and machine assembly feasibility study. Note that all figures in this article refer to the first concept with all PF coils located inside the TF coils.

2.5. Port allocations

The VNS tokamak has 12 upper, 12 equatorial and 12 lower ports providing access to the plasma chamber. Most ports are allocated to systems required to operate the machine i.e., plasma external heating and diagnostics, vacuum pumping, and remote replacement of IVCs. The remaining ports were reserved for testing, see Table 5 and Fig. 2. It is recognized that some required functions i.e., systems, are not yet allocated to VV ports. Some upper ports may be used in addition to blanket RH also for testing, EC heating, or plasma diagnostics. Some of the lower ports may be used in addition to divertor RH and pumping for pellet fueling and/or in-vessel viewing.

Table 5
Tentative allocation of functions to VV ports.

Port	Upper	Equatorial	Lower
#1	Blanket RH	EC	Divertor RH
#2	Blanket RH	NBI	Pumping
#3	Blanket RH	NBI	Pumping
#4	Blanket RH	NBI	Divertor RH
#5	Blanket RH	NBI	Pumping
#6	Blanket RH	blocked	Pumping
#7	Blanket RH	Diagn.	Divertor RH
#8	Blanket RH	TBM	Pumping
#9	Blanket RH	TBM	Pumping
#10	Blanket RH	Diagn.	Divertor RH
#11	Blanket RH	TBM	Pumping
#12	Blanket RH	TBM	Pumping

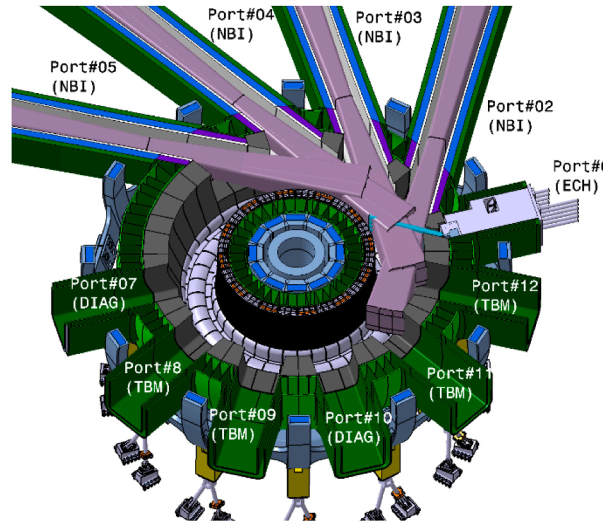

Fig. 2. System allocation to equatorial ports.

2.6. Vacuum vessel

The VV is located inside the cryostat and enclosed in the cage formed by the magnetic coils. The IVCs are mounted to the VV internal surface. The VV is a double wall structure with two shells, poloidal ribs and ports providing access into the main chamber, see Table 6. In-wall shielding plates of B4C and SS304 are bolted in the interspace between the shells. The cooling water that flows from the bottom to the top fills the remaining volume inside the double-wall structure making the VV an effective n-shielding structure. The other main functions of the VV are to provide a high vacuum for the plasma and first confinement to the radioactive source terms inside the VV.

The thickness of the VV shells was defined as 20 mm considering (i) the required spacing of the poloidal ribs, (ii) the cost of plates and welds, (iii) the cooling of the shells that are heated by neutrons, and (iv) the required strength to support IVCs and port structures. Similar to the ITER VV [29] a toroidally continuous triangular support is implemented into the VV structure above the lower port, providing (i) plasma passive vertical stability, (ii) n-shielding of the cutting/rewelding locations of the divertor pipes, (iii) support features of the outboard blankets, and (iv) the corresponding cut-out in the backside of the outboard blankets segment allows their radial extraction through a tilting movement without interfering with the divertor outboard baffle. Therefore, the outboard segments can be remotely replaced without the need for the prior removal of the divertor.

2.7. In-vessel components

The rationale for the selection of the structural materials of the VV and the IVCs undergoing high n-irradiation during the operation of VNS is described in [30] and summarized in Table 7. With the exception of the TBMs the IVCs are made of 316 Ti stainless steel in order to withstand the foreseen end of life neutron fluence without need for replacement. Since the conditions in VNS are comparable to those foreseen in ITER, the entire internal surface of the VV is covered, as in

Table 6
Major dimensions and weight of the VV.

Main VV major dimensions		Weight (tons)	
Outer diameter	8.8 m	Main vessel	260
Height	4.5 m	n-shielding	170
Double wall thickness	0.43 - 0.75 m	Ports	480
Free volume incl. ports	120 m^3	**Total**	**940**

Table 7
Materials of main VNS structures.

Magnet coil structures	Stainless steel (SS)
VV	316L(N)-IG, 304B4, B_4C
Wall armour	W
Shield blanket	316Ti, n/γ-absorbents e.g., W or TiH_2
Divertor	316Ti, CuCrZr, W
Cryostat	316L

ITER, with the blanket and the divertor to reduce the damage and heating of the VV due to the neutron flux. The high heat loads, see Section 2.2, also require all IVCs to be actively cooled. The segmentation of the VNS IVCs follows the same principles as foreseen for DEMO [31, 32] i.e., in each sector the divertor is divided into 3 cassettes and the blanket into 5 segments allowing access to their service pipes and enabling their remote replacement [33]. Because the NB and test equatorial ports cause a division of several (≈12) outboard segments in upper and lower blanket modules, a customized integration concept will have to be developed for these.

The divertor is ITER-like with two vertical targets and a dome mounted onto the cassette body [34]. The targets are tungsten monoblocks joined to CuCrZr tubes flowing high velocity cooling water, see alsoTable 8. In detached condition the maximum heat flux on the target plates is expected to be ≈6 MW/m^2 [18]. Below the dome a pumping slot in the cassette body allows the plasma exhaust gas to flow towards the torus vacuum pumps that are installed in several of the lower ports. An ITER-like integration concept was chosen for the divertor cassettes with the following features: (i) a contact plate with spherical surface on the inboard side constraining translations but not the rotations of the cassette and defining the radial position of the targets, (ii) a support with removable pins on the outboard side constraining the cassette also against rotation about the radial axis, (iii) a radial preload is applied when the cassette is installed ensuring electrical contact on inboard and outboard supports to the VV, (iv) two cooling pipes accessible during RH by orbital tools from the inside of the lower port after removal of local n-shields, see Fig. 1, and (v) toroidal rails below the cassettes for the toroidal transport to the RH ports.

The blanket has the primary function to absorb neutrons and attenuate secondary gammas resulting from neutron interactions; its thickness has been defined accordingly, the minimum being 25 cm on the inboard midplane of the reactor where the radial space is limited, see Fig. 1. On the outboard side the blanket thickness is larger, ≈60 cm. This allows replacing shielding blankets with TBM blankets that exhibit poorer shielding performances. The shield blankets are welded box-structures with internal stiffening ribs. Like the VV, they are operated with cold water, see Table 8. The plasma-facing plate i.e., the first wall, is HIPed and has an internal array of poloidal cooling channels capable of removing heat fluxes on the first wall well beyond the expected local loads during flat top (<150 kW/m^2). The coolant follows a U-path to cool in series the two main zones of the blanket, i.e. first wall and shield zone: water enters the first wall channels through a screen placed at the top of the segment, it flows downward and reaches the bottom of the segment where it accesses a coolant redistribution chamber that redirects the coolant into six shield zone compartments; water returns to the top of the segment flowing upward inside the box structure while cooling the shielding assemblies mounted inside the box compartments. In the midplane to enable a compact radial build of the machine, highly efficient radiation shielding materials are used e.g., W or TiH_2. The integration concept of the blanket segments is an adaptation of the concept described in [35] with the following main features: (i) ferritic materials incorporated inside the inboard segments are attracted towards the machine center due to the gradient in the toroidal field pushing the inboard segments against the VV, hence bolts to mechanically fix the blanket are avoided, (ii) the upper port shield plug constrains all 5 segments of one sector vertically after installation, (iii) each segment has two toroidal shear keys to resist to the large radial moment that occurs in plasma disruptions, (iv) each segment has one electrical strap on the bottom and one on the top to avoid excessive loads due to long halo current paths, and to minimize halo currents flowing through the blanket cooling pipes [25]. Note: the predicted EM loads on the inboard blanket exceed considering the current design the force provided by the currently assumed ferritic materials, hence the mechanical attachment concept of the inboard segments requires further design improvements.

Table 8
Coolant conditions during plasma operation.

Cooling system	Distinctive requirements	Coolant condition
VV	Safety class, decay heat removal	p_{avg} = 1.0 MPa, $T_{in/out}$ = 50 °C
Shield blanket + port plugs	Box structure requires low design pressure	p_{avg} = 1.0 MPa, $T_{in/out}$ = 50/70 °C
Divertor	High heat loads, heat sink of CuCrZr	$p_{in/out}$ = 3.5/2.4 MPa, $T_{in/out}$ = 50/85 °C
NBI	High heat loads, heat sink of CuCrZr	p_{avg} = 1.4 MPa, $T_{in/out}$ = 50 °C/ tbd.
Test blankets:		
- water coolant	see [37]	≈300° @ 155 bar
- gas coolant	see [38]	>300° @ ≈80 bar
TBMs	Individual	Individual

2.8. Tokamak cooling systems

Water in subcooling condition is used as coolant for all tokamak components although some of the outboard test blanket segments can be connected optionally to a gas cooling system. In 180° of VNS the test blankets are water cooled in pressurized water reactor (PWR) condition, in the other 180° the test blankets are gas cooled. The baking condition of VV and those IVCs not normally cooled at temperatures above 300 °C has been tentatively defined as 180 °C @ 20 bar i.e. 30° higher as compared to W7-X [36]. The baking hold time to recover from in-vessel water leaks and for conditioning after venting during in-vessel maintenance is therefore reasonably short, <1 week. Since neither Boron nor Beryllium are foreseen on the VNS plasma-facing components no gas baking system at temperatures ≈350 °C is currently foreseen but could be added if required. Several individual cooling systems are implemented, each customized to the specific requirements of the client components, see Table 8.

In the locations of many of the outboard segments the VNS operator has the choice of installing shield blankets or test blankets. This choice can vary during the VNS operation phase. Each outboard blanket can therefore not only be connected to cooling pipes, but it can optionally be connected also to a tritium removal circuit via two additional service pipes. Depending on the choice the cooling pipes are connected either to the shield blanket or to the respective test blanket cooling loop. The connections are made after the installation of the blankets through valves, which are located in a separated corridor that is protected from the radiation emitted by the activated cooling To minimize the radiation exposure of the personnel, the segment isolation valves installed onto the feeding pipes are equipped with extended reach rods, which allow valve operations from a low radiation area, see Fig. 3. There are valves allowing to isolate each IVC to facilitate draining and drying as well as leak localization.

2.9. Neutral beams

The dependency of VNS on the reliability of its NBIs led to the choice of positive ion beams operated with 120 keV, a technology with decades of operational experience with low downtime from ASDEX-Upgrade (AUG) [39] [40]. Since the main components of the AUG NB are already actively cooled, it can be expected that an upgrade to full

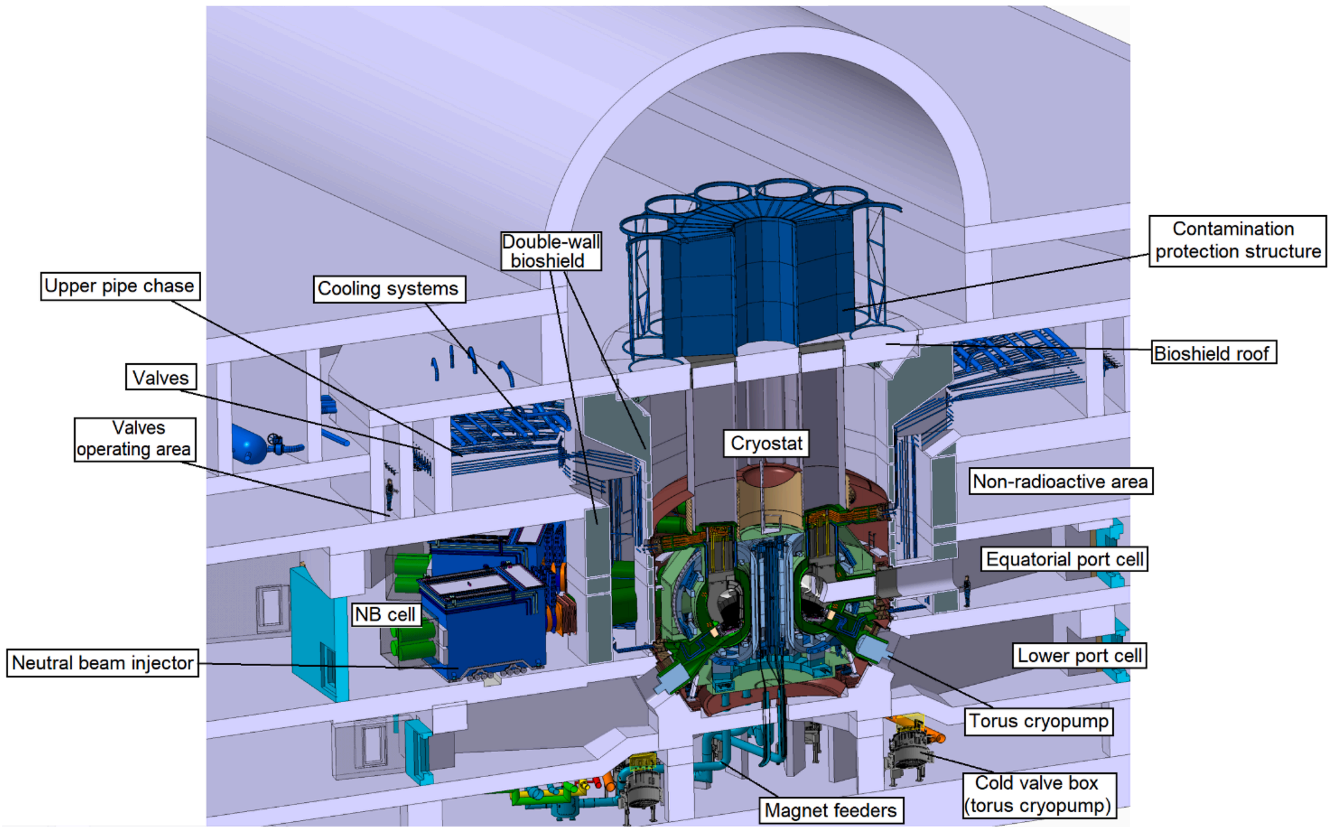

Fig. 3. Configuration of plant systems in the VNS tokamak building.

continuous operation through an adequate R&D program will be successful. Although at constant beam power a higher beam energy would increase the fusion yield in VNS by reducing the D dilution of the T plasma [21], the technical issues and reliability risks associated with negative ion sources were considered unacceptable. Each of VNS's NBIs has four sources and can inject a power of 13.5 MW into the VNS plasma, see Fig. 3 and [41]. During plasma operation three NBIs are in operation, the fourth is in regeneration mode with closed gate valve and warming-up the large cryopanels to release and pump out the absorbed gases [42]. The operation cycle of an NBI is 160 min, alternating 120 min of operation and 40 min of regeneration. A fifth replacement NBI is foreseen to be located in the Active Maintenance facility (AMF).

The dead weight of one of VNS's NBIs (≈80 tons) allows its transfer with a floor-based system to the AMF. During NBI transfer the two valves at the interface to the VV duct are closed, one sealing the VV, the other sealing the NBI box. Since the NBI box is not contaminated on the external side, no cask is needed for the transfer to the AMF. In the AMF a dedicated area allows (i) to open the NBI box to remove NB components requiring maintenance whilst controlling the spread of contamination, (ii) to maintain/refurbish the NB components, (iii) decontamination, and (iv) high-voltage conditioning and re-alignment of sources (no conditioning/re-alignment is foreseen of NBIs installed in the NB cell) [43].

2.10. Electron cyclotron heating

EC heating is required in VNS to keep the electron temperature high and, through this, increase the beam target reaction rate. Other functions are the control of Neoclassical Tearing Modes (NTMs) and the control of tungsten accumulation. Currently, one equatorial port launcher is foreseen providing a maximum power of 10 MW, which integrates 12 waveguides each connected to a 1 MW gyrotron located in an external building allowing for some redundancy. Higher power per beam might be considered in the future allowing for further redundancy. A design study to replace the single equatorial port launcher with several smaller upper port launchers is on-going. This alternative would make an additional equatorial port available for testing.

The equatorial port plug launcher adopts the mid-steering launcher design described in [44], whereby front parts of the launcher reach to the backside of the blankets, see Fig. 4. The launcher therefore does not block the removal of outboard blanket segments and can remain in-situ during blanket RH. Six of the waveguides are "open-ended", i.e. they direct the energy along fixed angles aiming at the plasma center for heating. The other 6 waveguides direct the beams onto steerable mirrors, which launch the beams with fixed toroidal angles, while the poloidal angles are steerable. The steering range may cover the plasma center and the NTMs, since the rational $q = 2/1$ and $3/2$ surfaces in the current reference plasma parameters are not far from the plasma center, in view of the port. The scheme fulfills the power requirement for the center heating and also provides adequate flexibility.

3. VNS plant description

3.1. Tokamak building

The VNS tokamak building is a concrete structure with dimensions of approximately 75 m × 75 m × 50 m. It is the ultimate confinement barrier and has massive roof structures designed to resist to an airplane crash and to provide radiation shielding to the environment during in-vessel maintenance from the top of the machine → skyshine [45]. The following auxiliary buildings are directly adjacent to the tokamak building: assembly hall, AMF, tritium building, & diagnostic building. Other auxiliary buildings such as the cryoplant or magnet power supply building are arranged around the tokamak building and service connections are routed in trenches or on bridges.

The VNS machine defines the layout of the tokamak building. It is

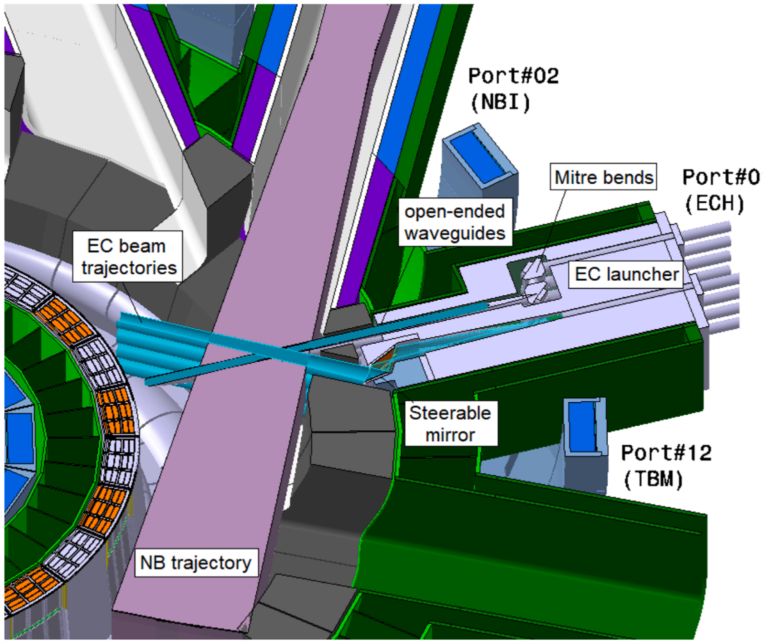

Fig. 4. Design configuration of equatorial port EC launcher.

enclosed by a 2 m thick bioshield, which is double-walled routing radioactive coolant in the interspace and hence segregated from other building areas. The building levels are aligned with the machine ports. The distributors of the cooling systems are integrated in a ring channel, the so-called pipe chase, above the large neutral beam cell, which houses the four NBIs, see Fig. 3.

Outside the bioshield port cells are arranged on the levels connecting to the VV ports. Each can be sealed providing secondary confinement when a cask is located inside and docked to the VV port. On the bioshield roof a steel structure divides the upper maintenance hall for this purpose into 12 "port cells" with segregated pathways to the AMF for the blanket replacement, see also [46]. The casks on the equatorial and lower levels travel to and from the AMF in the galleries outside the port cells.

In the basement level there are no radioactive systems and no in-vessel access ports, hence no contamination is expected. Therefore, many plant systems are integrated on this level to facility accessibility. This includes all services of the superconducting coils, the cryo-distribution, the cold valve boxes and others.

3.2. Fuel cycle systems

Since the VNS plasma is operated in steady-state regime, so is the VNS fuel cycle. At the nominal fusion power of 29 MW the VNS plasma consumes $\approx 10^{19}$ T atoms / *sec*, i.e. 1.6 kg / full power year. It is fueled by the NBIs with D; T is injected through pellets. To increase the probability for the injected high-energy Deuterium neutrals to react with T, the tritium fueling rate and hence its concentration in the plasma is higher by a factor of approximately 10. In addition to D and T the seeding gas Kr is injected [18]. All gases, including the Helium that is created in the fusion reaction, are pumped out either by the torus vacuum or the NB vacuum pumps. Both are cryopumps that are regenerated in cycles, see also Section 2.9. The gases recovered from the cryopumps are pumped to the tritium plant where the isotope separation system (ISS) separates D from T and provides them to NBIs and pellet fueling, respectively. Since both systems require a high purity of D or T, a direct fuel recycling system based on a metal foil pump as proposed for DEMO [47] is not applicable, as it does not separate D from T.

The cryopumps, which accumulate tritium during their pumping phase until they are regenerated, are the main contributors to the in-vessel tritium inventory, in particular the NB cryopumps due their long pumping phase. A rough estimate based on the gas throughput figures predicts a total in-vessel T-inventory below ≈150 g. The main contributors of the tritium inventory in the tritium plant are the cryo-distillation columns of the ISS (≈400 g) and the T storage beds (so far undefined).

3.3. Power consumption

The power consumption of VNS during plasma operation (flat top) is driven by P_{aux}. Secondary clients are the tokamak cooling systems and the cryoplant. A preliminary estimate of the plant power consumption is provided in Table 9.

3.4. Safety and licensing

The basic design of the VNS plant is conceived to satisfy the fundamental safety functions:

- Confinement of radioactive and hazardous materials. VNS implements two confinement barriers in all phases of the plant life. This includes the use of casks for in-vessel interventions, an effective ventilation system, and the consideration of maximum tritium inventories in rooms as defined by International Nuclear Standards e.g. [48]. An air detritiation system allows complying with these limits.

Table 9
VNS steady-state power consumption of main clients during plasma operation.

	Flat top	Standby
NBI [41]	126 MW	0
EC	25 MW	0
Cooling systems	5 MW	< 2.5 MW
Cryoplant		
- Magnet coils (4 K)	2.5 MW	1 MW?
- Thermal shields (80 K)	low	low
- Torus cryopumps (4 K / 80 K)	low	0
- NBI cryopumps (4 K / 80 K)	1.5 MW	0
Tritium plant	1 MW	1 MW
Magnet power	1 MW	0
Safety components	2 MW	2 MW
Others	≈4 MW	≈4 MW
Total	≈ **168 MW**	≈**10 MW**

- Limitation of exposure of workers to radiation and toxic materials. All in-vessel maintenance operations will be carried out by robotic tools. Furthermore, the tokamak building contains a bioshield and separate areas for radioactive systems to protect man-accessible areas from radiation.
- Limitation of impact on the environment and the public (incl. environmental releases, management of radwaste and decommissioning).

4. Summary

The VNS feasibility study concludes with the basic design of a tokamak machine integrated with the main plant systems including provisions for maintenance and able to meet safety & licensing requirements. VNS offers the relevant environment to test and qualify fusion nuclear components and a large area to install BB TBMs i.e. ≈ 25 m^2 on the outboard side. Prior to this study, the following critical aspects were seen as potential showstoppers. The feasibility study identified potential solutions for them and quantified key parameters:

- A *steady-state plasma scenario* compliant with commonly considered feasibility criteria that generates the required NWL [21].
- A *power exhaust concept* together with the configuration of the divertor causing acceptable levels of plasma impurities and divertor heat loads [18,34].
- A *plasma magnetic configuration* together with two options for the configuration of the magnetic coils that allows plasma ramp-up, shaping, and control [26,28,49].
- A concept of *neutron and radiation shielding* structures that meet the critical requirements to protect: (i) the superconducting coils, (ii) the VV and the IVC pipes, and (iii) personnel access areas outside the bioshield.
- *Integration concepts of the main in-vessel components*, incl. their segmentation, electrical and mechanical interfaces, and remote maintenance concepts [25].
- A concept of *positive ion neutral beam injectors* incl. design, operation and remote maintenance [41,42,43].
- An integration concept of tokamak cooling systems protecting other plant areas from radiation emitted by *activated cooling water*.

Although the studies carried out so far do not define the VNS plant at concept level, they indicate that a compact beam-driven tokamak device can be built and operated requiring moderate R&D efforts prior to construction and operation. VNS seems to offer attractive testing opportunities for all fusion nuclear technologies required in a demonstration fusion power plant and could make an essential contribution to the path towards the realization of fusion energy.

CRediT authorship contribution statement

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Declaration of competing interest

The authors declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper.

Acknowledgments

This work has been carried out within the framework of the EUROfusion Consortium, funded by the European Union via 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.

The authors wish to thank S. Chiocchio, A. Cardella and G. Janeschitz for their consultation and guidance. This feasibility study also benefitted significantly from numerous design concepts and technologies that were developed for ITER and from the lessons-learned during the ITER construction.

Data availability

No data was used for the research described in the article.

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