Study of a Spherical Torus Based Volumetric Neutron Source for Nuclear Technology Testing and Development

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DOE/SBIR ST-VNS Project Publications (1996-1998)

A. Journal and Conference Proceedings

1. E.T. Cheng, et al., "Study of a Spherical Tokamak based Volumetric Neutron Source," Fusion Engineering and Design 38 (1998) 219-255.
2. I.N. Sviatoslavsky, et al., "Engineering Design Issues of a Low Aspect ratio Tokamak Volumetric Neutron Source," Fusion Technology, 30 (1996) 1649.
3. Y.-K.M. Peng, et al., "Design Assumptions and Bases for Small D-T-Fueled Spherical Tokamak (ST) Fusion Core," Fusion Technology, 30 (1996) 1372.
4. E.T. Cheng and R.J. Cerbone, "Prospect of Nuclear Waste Transmutation and Power Production in Fusion Reactors," Fusion Technology, 30 (1996) 1654.
5. R.J. Cerbone, et al., "Neutronics Analysis of a Spherical Torus based Volumetric Neutron Source," Proc. IEEE/SOFE 97, San Diego, California, p. 841.
6. I.N. Sviatoslavsky, et al., "Mechanical Design Considerations of a ST Volumetric Neutron Source," Proc. IEEE/SOFE 97, San Diego, California, p. 253.
7. Y.-K.M. Peng, et al., "Physics and Systems Design Analyses for Spherical Torus (ST) based VNS," Proc. IEEE/SOFE 97, San Diego, California, p. 733.
8. E.T. Cheng, et al., "An ST-VNS Approach for Fusion Technology Development and Demonstration," Proc. IEEE/SOFE 97, San Diego, California, p. 261.
9. C.P.C. Wong, et al., "The Low Aspect Ratio Design Concept – Possibility of an Acceptable Fusion Power System," Proc. IEEE/SOFE 97, p. 1051.
10. Y.-K.M. Peng, "Spherical Torus Pathway to Fusion Power," J. Fusion Energy, 17 (1998) 45.
11. I.N. Sviatoslavsky, et al., "Design of the Centerpost for a Spherical Torus Volumetric Neutron Source," Fusion Technology, 34 (1998) 1061.
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13. E.T. Cheng, "Near-term Applications of Fusion from the ST-VNS Development Path," Fusion Technology, 34 (1998) 489.
14. E.T. Cheng, et al.,"Progress of the ST-VNS Study." Fusion Technology, 34 (1998) 1066.
15. J.D. Galambos, et al., "Systems Analysis and Scaled Costs," submitted to Fusion Engineering and Design (1998).
16. I.N. Sviatoslavsky, et al., "Mechanical Design Considerations of the Centerpost for a Spherical Torus Volumetric Neutron Source," submitted to Fusion Engineering and Design (1998).

B. Meeting Report

1. I.N. Sviatoslavsky, "Minutes of the International Workshop on Spherical Torus Volumetric Neutron Source and 1997 US/Russia Exchange II.3D on the Volumetric Neutron Source Design Study and Review, October 13-15, 1997, Del Mar, California," Fusion Technology, 34 (1998) 163.

Abstract

A plasma based, deuterium and tritium (DT) fueled, volumetric 14 MeV neutron source (VNS) has been considered as a possible facility to support the development of the demonstration fusion power reactor (DEMO). It can be used to test and develop necessary fusion blanket and divertor components and provide sufficient database, particularly on the reliability of nuclear components necessary for DEMO. The VNS device complement to ITER by reducing the cost and risk in the development of DEMO.

A low cost, scientifically attractive, and technologically feasible volumetric neutron source based on the spherical torus (ST) concept has been conceived. The ST-VNS, which has a major radius of 1.07 m, aspect ratio 1.4, and plasma elongation 3, can produce a neutron wall loading from 0.5 to 5 MW/m^2 at the outboard test section with a modest fusion power level from 38 to 380 MW. It can be used to test necessary nuclear technologies for fusion power reactor and develop fusion core components include divertor, first wall, and power blanket. Using staged operation leading to high neutron wall loading and optimistic availability, a neutron fluence of more than 30 $MW\text{-}y/m^2$ is obtainable within 20 years of operation. This will permit the assessments of lifetime and reliability of promising fusion core components in a reactor relevant environment. A full scale demonstration of power reactor fusion core components is also made possible because of the high neutron wall loading capability. Tritium breeding in such a full scale demonstration can be very useful to ensure the self-sufficiency of fuel cycle for a candidate power blankt concept.

1. Introduction

A plasma-based, deuterium and tritium (DT) fueled, volumetric 14 MeV neutron source (VNS) has been considered as a possible facility to support the development of the demonstration fusion power reactor (DEMO). It can be used to test and develop necessary fusion blanket and divertor components and provide sufficient database, particularly on the reliability of nuclear components necessary for DEMO. The VNS device can complement ITER by reducing the cost and risk in the development of DEMO [1].

The spherical torus (ST) is a promising candidate for such a VNS [2,3]. A recent investigation, supported by the USDOE SBIR program, showed that it is cost effective and scientifically feasible to employ the ST based VNS for nuclear technology development [2]. Such a ST-VNS has a major radius of 0.8 m, aspect ratio of 1.33, and an on-axis magnetic field strength of 1.8 tesla. It will be operated with a single-turn (in the central post) normal copper magnet to produce 38 MW fusion power when driven and heated by 21 MW of neutral particles at 500 keV energy. The neutron wall loading at the 10 m^2 test section will reach 1 MW/m^2.

Research continues during Phase II of the SBIR study [4]. The effort focuses on the optimization and technological feasibility of the VNS with higher neutron wall loads. The motivation is to cover the entire range of neutron wall loading (0.5 - 5 MW/m^2) anticipated in future power reactors. A second goal of the high wall loading approach is to test the component performance in an environment approaching or exceeding the lifetime neutron fluence. To satisfy these goals, a slightly larger device is conceived during the Phase II study. The configuration of the device has a major radius of 1.07 m and an aspect ratio of 1.4. The topics investigated during this second phase include physics and mechanical design analysis, materials selection, neutronics and activation analysis, availability assessment, and cost analysis. The status of the Phase II ST-VNS study is reported.

2. ST-VNS Device

The ST-VNS configuration has been determined based on conservative physics assumptions for operating at modest fusion power and neutron wall loading [4]. It has a major radius of 1.07 m, minor radius 0.76 m, aspect ratio 1.4, and plasma elongation 3. Figure 1 displays schematically the configuration of the ST-VNS design. The mechanical design of this device [5] is similar to the smaller size Phase I device as described in detail in Ref. 2. The centerpost is an unshielded single-turn normal conducting copper magnet, made of dispersion oxide strengthened copper alloy, Glidcop. As described in Refs. 2 and 5, it has three sectors with the top and bottom sectors tapered to increase the radius of the conductor in order to minimize the ohmic heat loss, as shown in Fig. 2. A top view of the device ports is given in Fig. 3 showing the two neutral beam injectors (NBI), two maintenance ports, and 8 test ports. Note that each port has a width of 1.05 m at the midplane location. With a module height of 2 m, the 8 test ports will provide a total testing area of 16.8 m^2.

The physics design has considered two cases. One is a device using neutral beams for heating and driving the plasma current, and the other device is heated and driven by ICRF. Note that both devices are designed to have the identical physical configuration as described above. The plasma and power performance parameters for these devices which producing several neutron wall loading values are summarized in Table 1. As shown in Table 1, the ST-VNS produces 151 MW fusion power while achieving a neutron wall loading of 2 MW/m^2. The average toroidal beta is 37% and 27%, respectively, for the NBI and ICRF cases. Due to the lower beta in the ICRF case, the on-axis toroidal field (B_t) is higher in the ICRF case (2.55 tesla) than in the NBI case (2.13 tesla). The von-mises stresses in the center-post copper magnet in the NBI and ICRF devices were estimated to be 96 MPa and 136 MPa, respectively, pushing the ICRF device beyond the presently recommended limit of 100 MPa [6]. Of course, if the toroidal beta in the ICRF device can be improved, the technological requirement in the center-post magnet will be relieved.

Using the ICRF heated and driven, 151 MW fusion power device (2 MW/m^2 wall load) as the reference design, the thermal-mechanical performance of the centerpost was analyzed [5]. Due to nuclear transmutation of copper, the resistivity of copper in the centerpost will increase during operation. An overall of five percent increase in ohmic heating loss for each full power year operation in the 2 MW/m^2 device can be obtained from a detailed distribution of resistivity change in the centerpost as shown in Fig. 4. Table 2 summarizes the thermal hydraulic parameters of the centerpost for up to 3 full power years (FPY) of operation. To minimize the ohmic heating, it is necessary to maintain a low copper temperature during operation. One way to accommodate this is to increase the water mass flow rate. Fortunately, the resultant water pressure drop and increase in water pumping power are not excessive, as shown in Table 2. The centerpost resistive ohmic power was estimated to be 153 MW at start, 161 MW after one full power year, and 178 MW after 3 FPYs. The total supply power needed to operate the reference VNS was estimated to be 313 Mwe. The operating cost figure of merit can thus be calculated to be about 180 $M/y, assuming an availability of 60%. The operating cost will drop to $83M/y if sufficient tritium is bred in the device itself. These results are also tabulated in Table 1.

Design analyses were extrapolated from the reference design to the NBI driven designs and the results are also given in Table 1. Note that as shown in Table 1, the NBI heated and driven device shows lower stresses in the centerpost due to the lower applied magnetic field. At a higher magnetic field of 2.7 tesla, the fusion power can be increased to 377 MW providing a neutron wall loading of more than 5 MW/m^2 at the outboard. The operating cost is $105M/y at 60% availability when tritium self-sufficiency is provided. Tritium breeding is essential in this case, other wise the operating cost will exceed $480M/y.

Scaled capital cost analysis was performed as part of the systems study leading to the current VNS configuration and physics parameters [7]. It was found that the total construction cost of the ST-VNS ranges from \$1.36 to \$1.62B (in 1989\$). The fusion core including the centerpost, the outer return TF legs, divertor, shielding blanket and radiation shield (see Fig. 1) is about \$260M. The NBI heated and driven system for the 2 MW/m^2 (151 MW fusion power) operation is about \$380M. Note that the beam power delivered to the plasma in this case is 56.7 MW. When the heating power increases the NBI cost goes up too. The NBI cost for the highest wall loading case, 5 MW/m^2, is about \$480M because the needed heating power is 72 MW. The remaining cost items include the site (\$440M), instrumentation and control for power supplies (\$79M), diagnostics (\$148M), remote handling systems (\$123M), tooling and assembly (\$89M), etc., adding about \$880M.

3. Fusion Technology Development and Demonstration

Due to the capability of high neutron wall loading in a given device configuration, the ST-VNS can be used to obtain (1) lifetime limiting factors for power core components including the first wall, divertor and blanket, and (2) operating experience of reactor components at increasing levels of neutron wall loading. In other words, the unique capability of the ST-VNS allows the development of needed fusion core components and related technologies for the DEMO in a reactor relevant environment [8]. The possible operational scenario of such a fusion core component development and demonstration device is shown in Fig.5.

Figure 5 shows that the development process consists of staged operations with increasing neutron wall loading levels. Initially operation will be at a very low wall loading, such as 0.5 MW/m^2 shown in Fig. 5, to gain the first experience of significant 14 MeV neutron flux level in an engineering component. The availability can be very low due to lack of knowledge and experience in operating such a device. The ITER developed SS316 based shielding blanket materials can be used for all components except the test blanket section. The core components made of power reactor related materials such as ferritic steel and vanadium alloys will be employed in the test section. Feasible core component concepts can be tested and selected for higher neutron wall loading operation after about 0.5 $MW\text{-}y/m^2$ fluence. Tritium breeding in the shielding blanket region may not be needed at this stage because the needed tritium consumption is only 640 g/y (30% availability). The total tritium consumption in the initial stage is no more than 2.13 kg since it takes about 3.3 years to accumulate the fluence of 0.5 $MW\text{-}y/m^2$. Partial tritium breeding will be in fact available in the power reactor blanket being developed in the test section.

Higher wall loading operations can continue after the initial stage operation, as shown in Fig. 5. At the higher wall loading stages, the outboard shielding blanket region other than the test section can begin to include power reactor relevant core components already tested and qualified in the test section. Another way to interpret it is that the fusion power blankets can be fabricated as blanket segments rather than modules and tested in the entire outboard region. The availability can be increased to 60% after significant experience learned from the operation of a power producing fusion device. During these stages, the qualified power blankets and divertor components will begin to be subjected to mean-time-to-failure testing. Reliability information of the fusion core components start to accumulate [1]. Tritium breeding will be available during these stages. Since the power blanket segment can be employed in the entire outboard region, an overall tritium breeding can be assessed. The self-sufficiency of tritium can thus be demonstrated for the relevant blanket concept.

The staged high wall loading operation of an ST-VNS can provide an accumulated neutron fluence of more than 30 MW-y/m2 in a power reactor relevant environment within about 20 years, as shown in Fig. 5. Compared to a fixed neutron wall load device, the operating time is significantly reduced, as shown in Fig. 6. The ST-VNS can provide a reactor relevant environment for the power core components to demonstrate all functions required for a power plant or DEMO through the start-up, transition, and steady-state operation at the designated power levels. Ultimate performance capability of promising power core components can thus be developed and demonstrated for the power plant application.

4. Conclusions

A low cost, scientifically attractive, and technologically feasible volumetric neutron source based on the spherical torus concept has been conceived. The ST-VNS, which has a major radius of 1.07 m, aspect ratio 1.4, and plasma elongation 3, can produce a neutron wall loading from 0.5 to 5 MW/m^2 at the outboard test section with a modest fusion power level from 38 to 380 MW. It can be used to test necessary nuclear technologies for fusion power reactors and develop fusion core components including divertor, first wall and power blanket. Using staged high wall loading operation and optimistic availability, a neutron fluence of more than 30 $MW\text{-}y/m^2$ is obtainable within 20 years of operation. This will permit the assessments of lifetime and reliability of promising fusion core components in a reactor relevant environment. A full-scale demonstration opower reactor fusion core is also made possible because of the high neutron wall loading capability. Tritium breeding in such a full-scale demonstration can be very useful to ensure the self-sufficiency of fuel cycle for a candidate power blanket concept.

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[1] M.A. Abdou, et al., "Results of an International Study on a High-Volume Plasma-Based Neutron Source for Fusion Blanket Development," Fusion Technology 29 (1996) 1.
[2] E.T. Cheng, et al., "Study of a Spherical Tokamak Based Volumetric Neutron Source," Fusion Engineering and Design 38 (1998) 219.
[3] T. Hender et al., "ST-VNS Study at Culham," Proc. Intl. Workshop on ST-VNS and Applications, San Diego, California, October 1997, to be published in Fusion Engineering and Design.
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[5] I.N. Sviatoslavsky, et al., "Mechanical Design Considerations of the Centerpost for a Spherical Torus Volumetric Neutron Source," Proc. Intl. Workshop on STVNS and Applications, San Diego, California, October 1997, to be published in Fusion Engineering and Design.
[6] I.N. Sviatoslavsky, "Minutes of the International Workshop on Spherical Torus Volumetric Neutron Source and 1997 US/Russia Exchange II.3D on the Volumetric Neutron Source Design Study and Review, October 13-15, 1997, Del Mar, California," Fusion Technology, 34 (1998) 163.
[7] J.D. Galambos, et al., "Systems Analysis and Scaled Costs," ," Proc. Intl. Workshop on STVNS and Applications, San Diego, California, October 1997, to be published in Fusion Engineering and Design.
[8] E.T. Cheng, et al., "An ST-VNS approach for fusion technology development and demonstration," Proc. IEEE/NPSS SOFE'97, San Diego, California, October 1997, p. 261.

Table captions:

Table 1. Summary of Key Operating Parameters for NBI-driven VNS Fusion Core Systems. VNS Device: Major Radius - 1.07 m; Aspect Ratio - 1.4; Plasma Elongation - 3.

Table 2. Summary of Thermal Hydraulic Parameters of the Centerpost at 2 MW/m^2 Neutron Wall Loading (ICRF Driven Device).

Figure captions:

Fig. 1. Side view of the ST-VNS.
Fig. 2. Top view of the ST-VNS.
Fig. 3. Geometry of the centerpost toroidal field copper magnet showing the coolant channel distribution at various elevations.
Fig. 4. Contours of resistivity change in the centerpost copper magnet after 2.5 full power years due to neutron transmutation. The reference ST-VNS produces 151 MW fusion power and 2 MW/m^2 neutron wall load at the test section.
Fig. 5. Illustration of a staged ST-VNS operational scenario from 0.5 to 5 MW/m^2 neutron wall loading. The availability is 60% for all stages except the initial stage.
Fig. 6. Comparison of accumulated neutron fluence for several operational scenarios.

Table 1

Summary of Key Operating Parameters for NBI-driven VNS Fusion Core Systems

VNS Device: Major Radius - 1.07 m; Aspect Ratio - 1.4; Plasma Elongation - 3

Average Wall Load (MW/m^2)	0.5 (NBI)	1.0 (NBI)	2.0 (ICRF)	2.0 (NBI)	3.0 (NBI)	4.0 (NBI)	5.0 (NBI)
Plasma current (MA)	9.7	10.0	13.2	11.1	12.8	13.4	14.3
Toroidal field (T)	1.9	1.91	2.55	2.13	2.44	2.52	2.67
Average toroidal beta $<\beta_t>$	0.24	0.32	0.27	0.37	0.36	0.38	0.39
Poloidal beta β_p	0.61	0.80	0.68	0.94	0.87	0.92	0.91
Normalized beta β_N (Tm/MA)	0.035	0.047	0.040	0.055	0.052	0.055	0.055
Average n_e ($10^{20}/m^3$)	0.91	1.53	2.59	2.16	2.63	3.50	4.57
Average n_i ($10^{20}/m^3$)	0.81	1.40	2.37	1.98	2.40	3.20	4.19
Greenwald n_e limit ($10^{20}/m^3$)	5.31	5.43	7.17	6.03	6.99	7.31	7.79
Borrass ne limit ($10^{20}/m^3$)	1.62	2.15	2.85	2.96	3.37	3.80	4.14
Central temperature (keV)	15.7	14.1	14.0	14.9	15.7	14.0	12.4
Fusion power P_{fusion} (MW)	37.8	75.6	151	151	227	302	377
Fusion amplification Q	1.59	2.04	3.32	2.67	3.96	4.53	5.25
NBI power, P_{aux} (MW)	23.8	37.0	45.5	56.7	57.2	66.7	71.9
Bootstrap current fraction	0.60	0.68	0.79	0.66	0.76	0.81	0.87
TFC total current (MA)	10.2	10.2	13.6	11.4	13.1	13.5	14.3
Mid-plane TFC average J_{tf} (kA/cm^2)	4.28	6.24	8.32	6.96	5.50	5.68	8.71
Centerpost von-Mises Stress (MPa)	75.7	76.5	136	95.6	118	126	141
Inner leg resistive power, P_{tfi} (MW)	86	86	153	107	141	150	168
Fusion core supply power P_{supply} (MW)*	168	192	313	261	320	355	398
Operating cost figure of merit ($M/yr)#	82 (44)	126 (51)	234 (83)	220 (69)	312 (84)	396 (94)	483 (105)

* P_{supply} (MW) = $[1.5 (P_{aux} + P_{tfi}) + 0.1 P_{fusion}]$.

Operating cost figure of merit ($M/yr) = $0.6 [1.67 (1\text{-TBR}) P_{fusion} + 0.44 P_{supply}]$, TBR = 0. Numbers in the brackets are those with TBR=1.0. Assumptions: Tritium cost = $30,000/g; electricity cost = 60 mils/kwh; VNS availability = 60%.

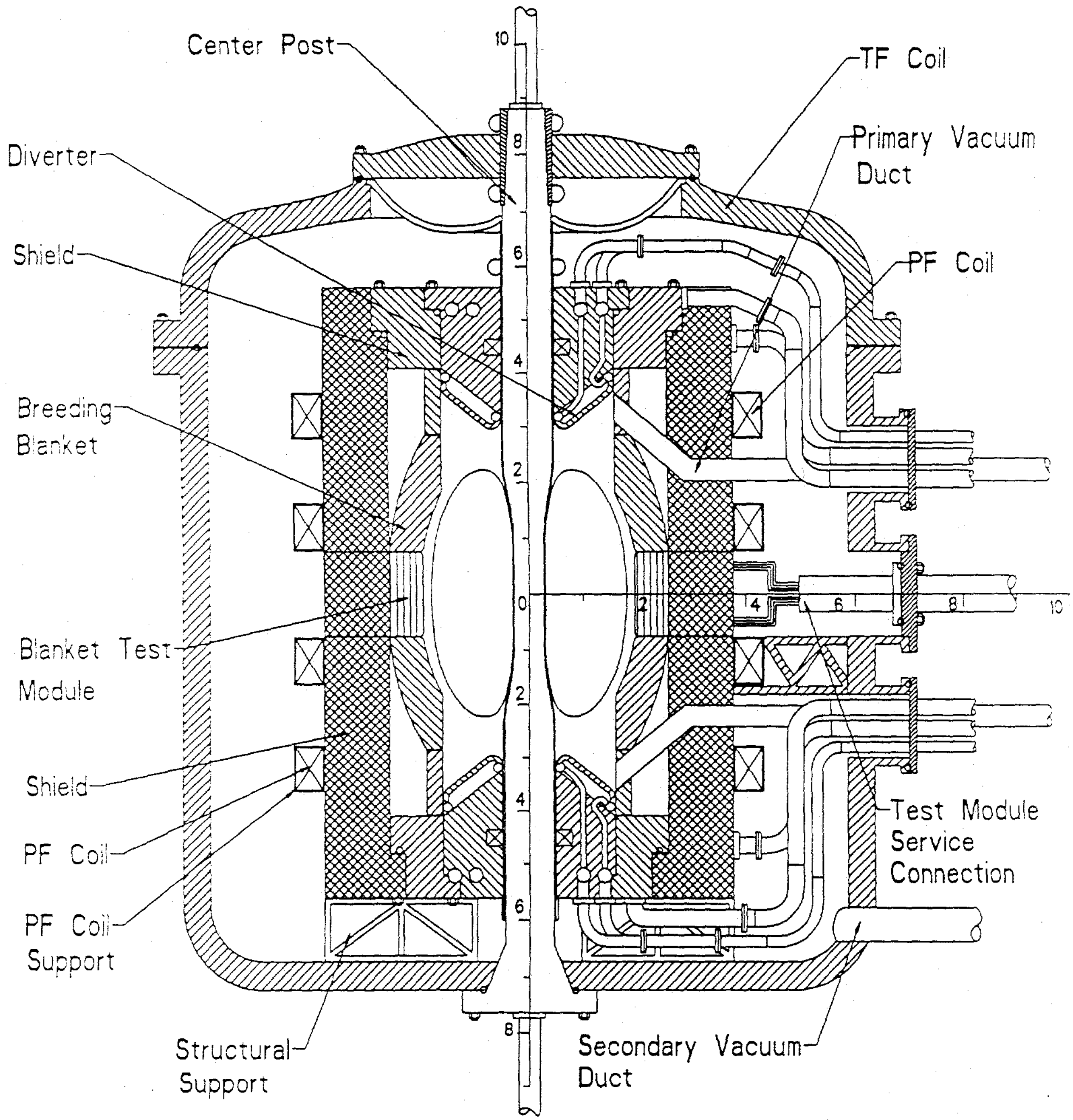
Table 2
Summary of Thermal Hydraulic Parameters of the Centerpost at 2 MW/m2 Neutron Wall Loading (ICRF Driven Device)

	At Start	After 1 FPY	After 3 FPY
Ohmic Heating, MW	153	161	178
Nuclear Heating, MW	11	11	11
Water Mass Flow Rate, kg/s	675	766	937
Maximum Cu Temperature, C	150	151	156
Water Pressure Drop, MPa	0.189	0.255	0.369
Water Pumping Power, kW	128	196	346

Fig. 1. Side view of the ST-VNS.

Elevation View of the ST VNS

(Dimensions in Meter)



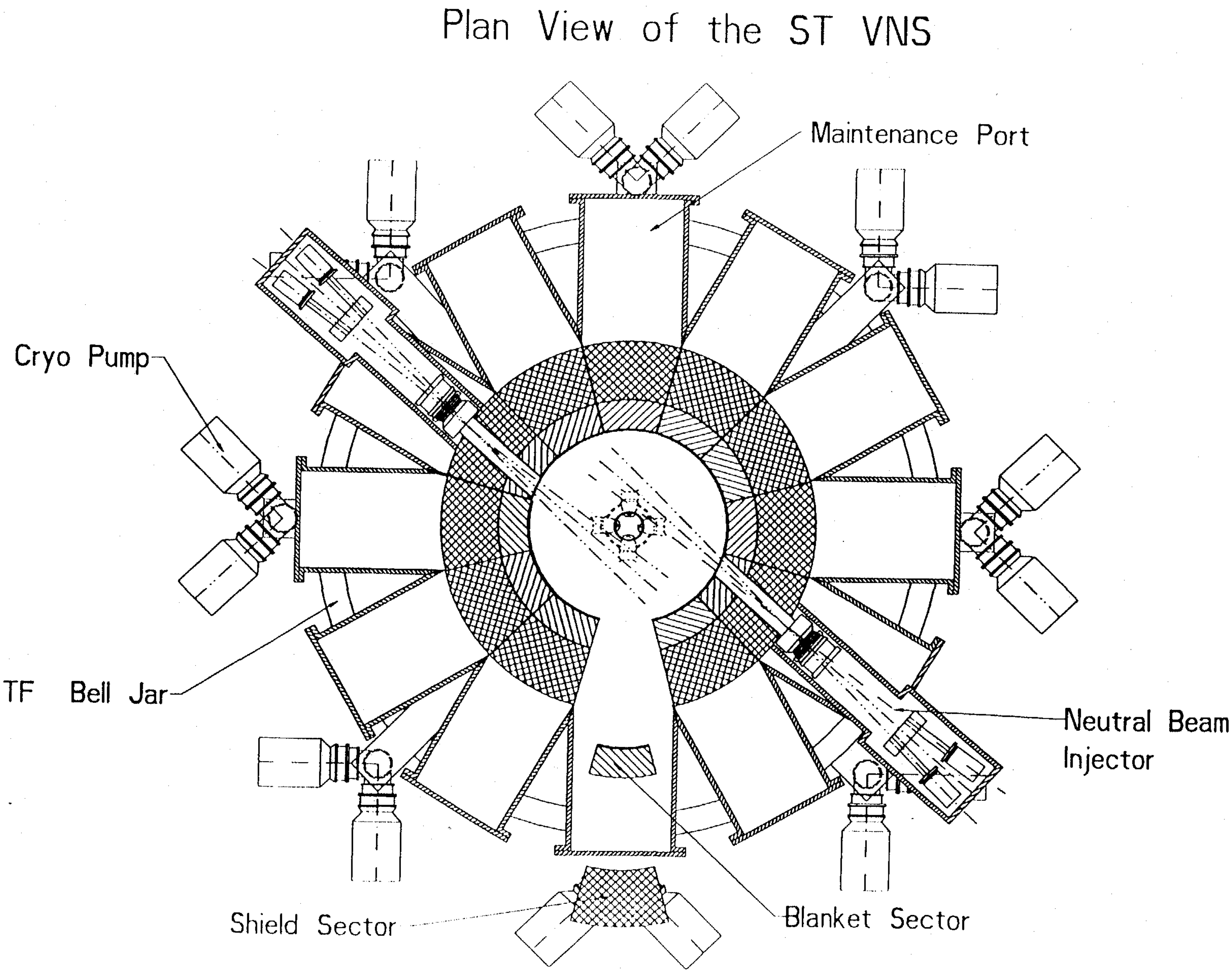


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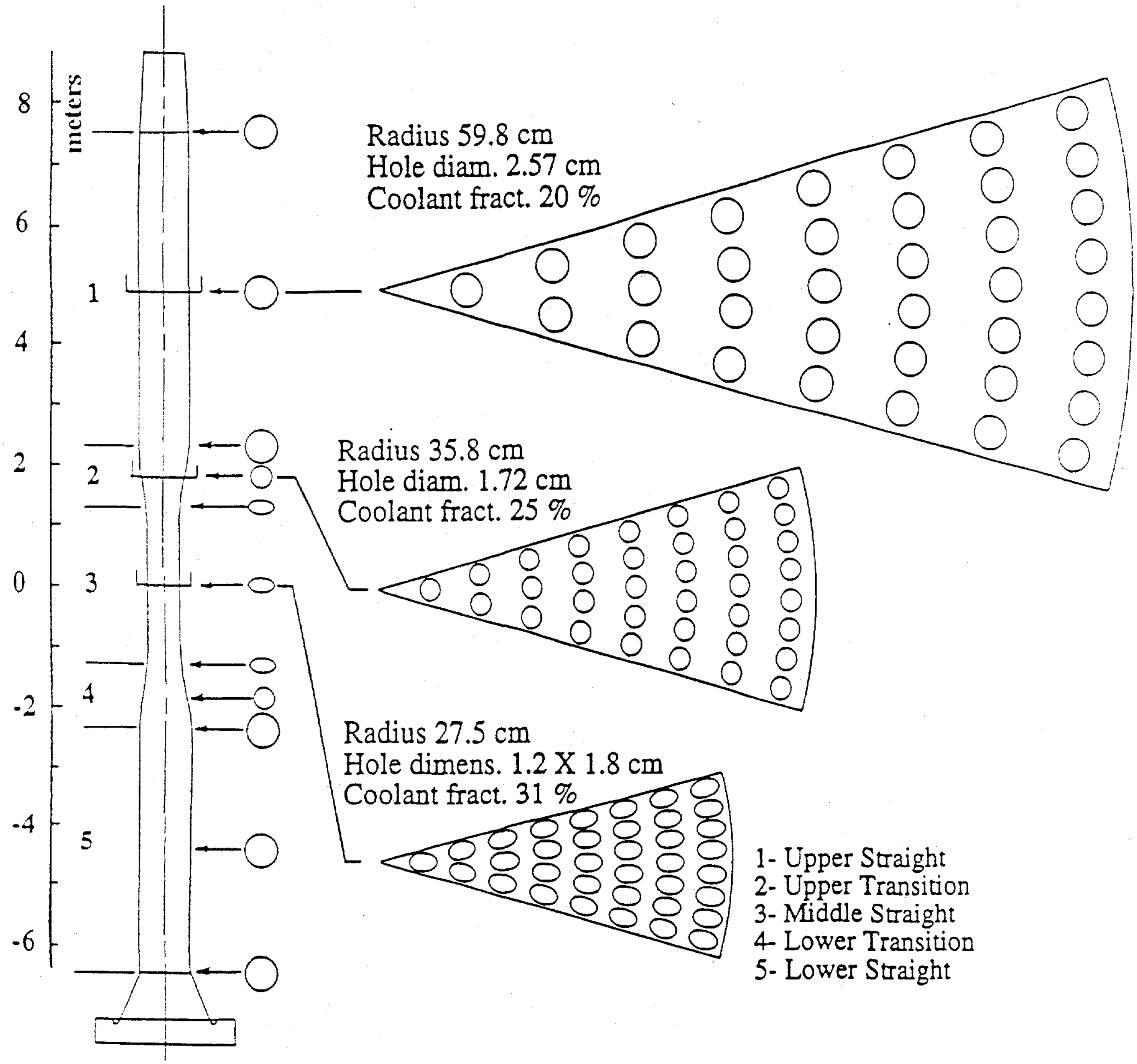
Fig. 3. Geometry of the centerpost toroidal field copper magnet showing the coolant channel distribution at various elevations.



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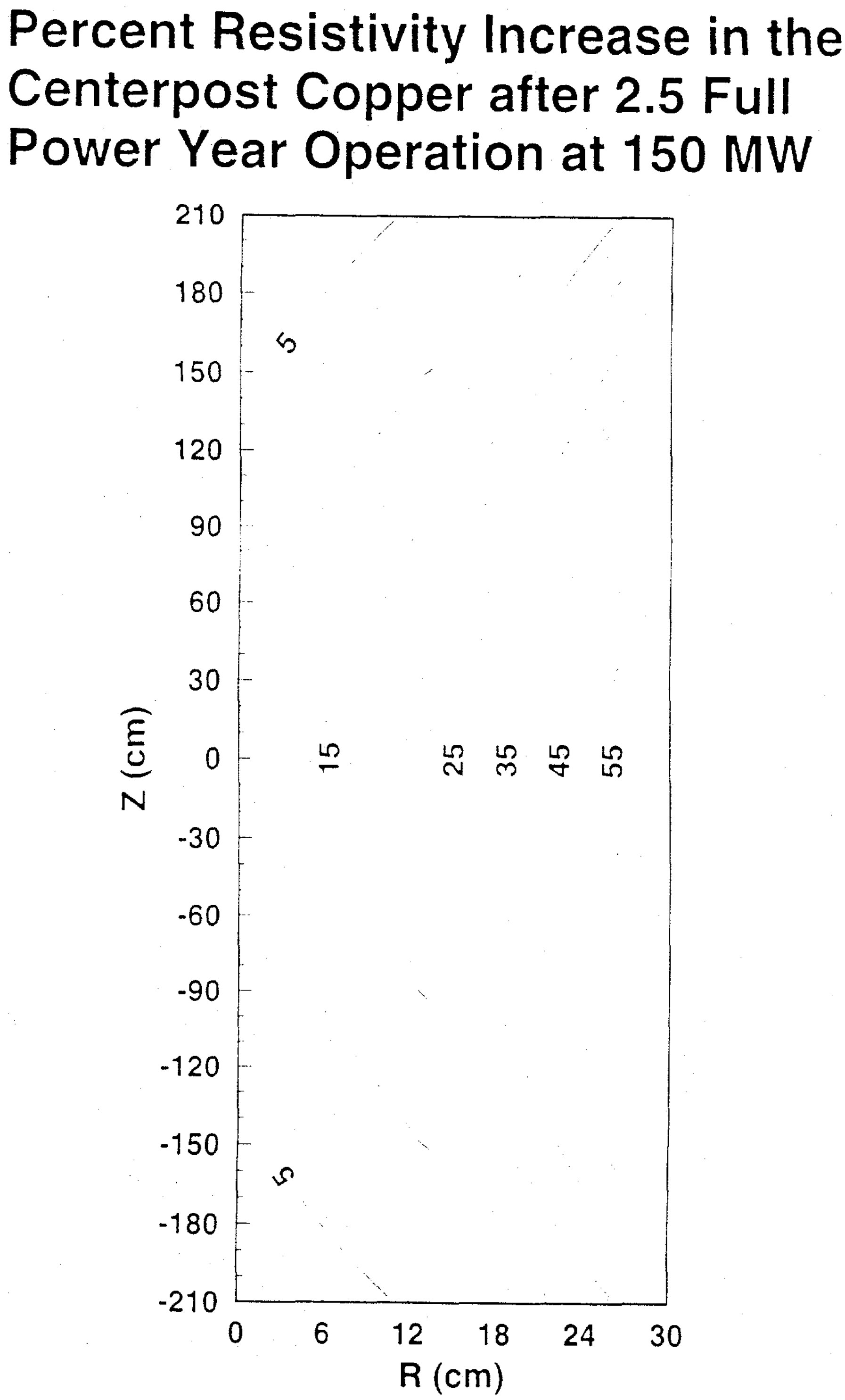

Fig. 5. Illustration of a staged ST-VNS operational scenario from 0.5 to 5 MW/m^2 neutron wall loading. The availability is 60% for all stages except the initial stage.

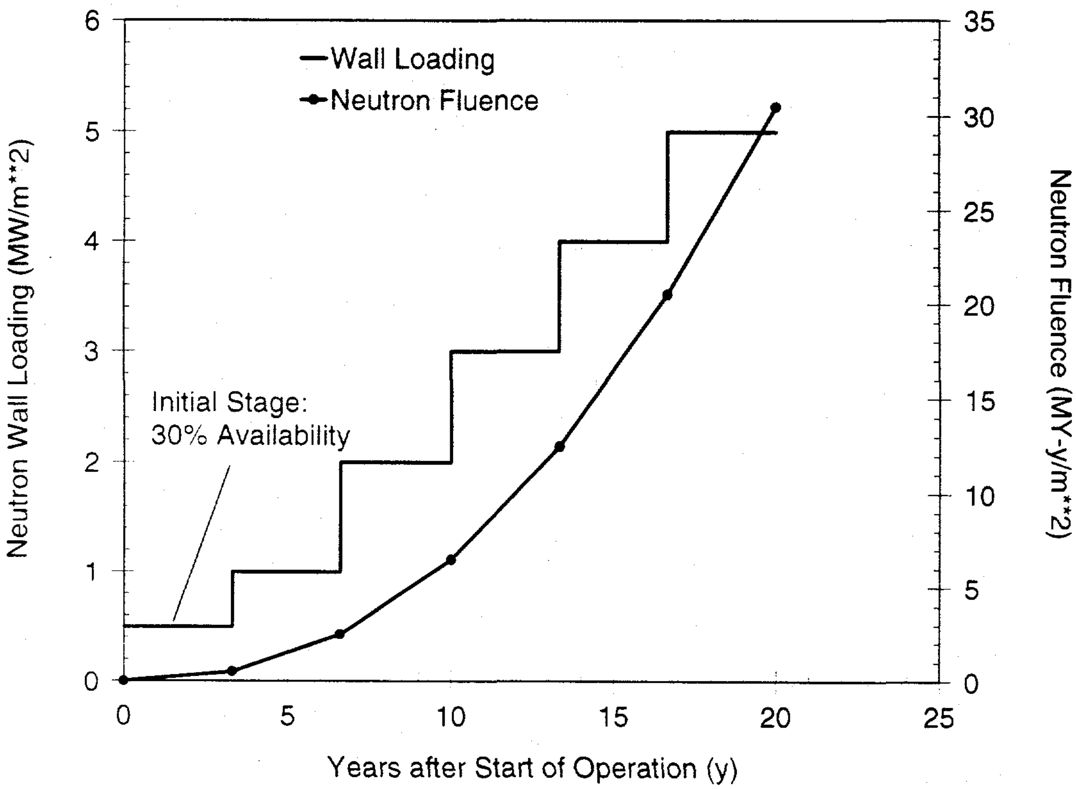

Fig. 6. Comparison of accumulated neutron fluence for several operational scenarios.