4th International Conference on Fusion-Fission sub-critical systems for waste management and safety

NOVEMBER 25-27, 2020

Moscow, Russia

PROGRAM & BOOK of ABSTRACTS

PROGRAM

4th International Conference on Fusion-Fission sub-critical systems for waste management and safety

25–27.11.2020, NRNU 'MEPhI', Moscow, Russia

IMPORTANT! The program uses Moscow time.
CET (Rome, Vienna): - 2 hours;
Novosibirsk: + 4 hours;
Hefei: + 6 hours.

Wednesday 25 November 2020

Device comparison I *(Chair: A. Dodaro)*

10.45–11:00 Opening Ceremony

11:00–11:30 **Matteo Barbarino (IAEA)**
IAEA's Coordinated Research Project on Compact Fusion Neutron Sources

11:30–12:00 **Sergey Taskaev (RF)**
High Flux Accelerator-based Neutron Source

12:00–12:30 **Yican Wu (CHN)**
Development and Applications of HINEG High Intensity Neutron Sources

12:30–13:00 Coffee & Tea

Device comparison II *(Chair: Y. Wu)*

13:00–13:30 **Francesco Orsitto (I)**
High field tokamaks as compact neutron sources for FFH reactors

13:30–14:00 **Wenjie Yang (CHN)**
Shielding design and neutronics calculation of the GDT based fusion neutron source ALIANCE

14:00–14:30 **Olov Ågren (SE)**
3D coils for a compact min B mirror field with minimal flux tube ellipticity

14:30–15:00 **Volodymyr Moiseenko (UA)**
Developments for stellarator-mirror fusion-fission hybrid concept

15:00–16:30 Lunch

Device comparison III *(Chair: M. Gryaznevich)*

16:30–17:00 **Dmitry Yakovlev (RF)**
Design and research program of the ALIANCE-T experiment

17:00–17:30 **Vadim Prikhodko (RF)**
Simulation of plasma parameters for ALIANCE project

17:30–18:00 **Sergey Medvedev (RF)**
Tokamaks with external X-points: stability limits and new prospects

18:00–18:30 **Fabio Panza (I)**
Novel Hybrid reactor concept based on Ignitor technologies and physics

18:30–19:00 Coffee & Tea

POSTER session *(Chair: Yu. Gasparyan)*

19.00–20:00

P1 Viacheslav Skokov (RF)
Comparison of lithium divertor options for DEMO-FNS tokamak

P2 Alexey Dnestrovskii (RF)
Neutron source based on circle cross section tokamak

P3 Alexander Sivak (RF)
Diffusion characteristics of radiation defects in iron: molecular dynamics data

P4 Eugenia Dlougach (RF)
BTR code application for Current Drive analysis in FNS-ST

P5 Olga Ogorodnikova (RF)
Radiation defect formation in bcc metals irradiated with high energy protons, self-ions and neutrons with different spectra

Thursday 26 November 2020

Pilot experiments and priorities of demonstrators I *(Chair: B. Kuteev)*

11:00–11:30	**Yury Shpanskyi (RF)** Development and integration study of fusion-fission hybrid systems into Russian nuclear power engineering
11:30–12:00	**Roberto Piovan (I)** Pilot FFHR based on a RFP as a fusion core
12:00–12:30	**Mikhail Gryaznevich (UK)** Progress on ST40 towards optimization for neutron production
12:30–13:00	Coffee & Tea

Pilot experiments and priorities of demonstrators II *(Chair: M. Ripani)*

13:00–13:30	**Fabio Panza (I)** Set up of a deterministic calculation model for the analysis of fusion-fission hybrid systems
13:30–14:00	**Alexander Panasenkov (RF)** Analysis of the scheme and parameters of the neutral beam injector for the FNS-ST tokamak
14:00–14:30	**Sergey Anan'ev (RF)** Integrated modeling of fuel flows in the plasma and in the injection and pumping systems for the DEMO-FNS fusion neutron source
14:30–16:00	Lunch

Materials, Fuel cycle and diagnostics I *(Chair: F. Orsitto)*

16:00–16:20	**Yury Gasparyan (RF)** NRNU "MEPhI" activity towards fusion-fission hybrid systems
16:20–16:40	**Evgenii Kulikov (RF)** Assessment of a possibility to produce non-traditional nuclear fuel in thorium blanket of hybrid thermonuclear reactor. Physical aspects for involvement of thermonuclear reactors into nuclear power systems.
16:40–17:10	**Mikhail Shlenskii (RF)** System studies on the fusion-fission hybrid systems and its fuel cycle
17:10–17:40	**Volodymyr Moiseenko (UA)** Americium and curium burning out in a fusion reactor
17:40–18:00	Coffee & Tea

Materials, Fuel cycle and diagnostics II *(Chair: A. Pisarev)*

18:00–18:30	**Alexander Sivak, Vyacheslav Chernov (RF)** Materials for hybrid systems
18:30–19:00	**Renato Gatto (I)** A reactivity monitoring method for fusion-fission hybrid reactors
19:00–19:30	**Francesco Orsitto (I)** The diagnostics and control of the Fusion-Fission Hybrid tokamak based reactors: the technology for measurements systems

Friday 27 November 2020

Summary and future prospects *(Chair: M. Barbarino)*

11:00–11:30	**Boris Kuteev (RF)** "Device comparison" (summary)
11:30–12:00	**Yican Wu (CHN)** "Pilot experiments and priorities of demonstrators" (summary)
12:00–12:30	**Francesco Orsitto (I)** "Materials, fuel cycle and diagnostics" (summary)
12:30–13:00	Coffee & Tea

Discussions *(Chair: A. Pizzuto)*

13:00–13:30	Device comparison
13:30–14:00	Pilot experiments and priorities of demonstrators
14:00–14:30	Fuel cycle and diagnostics

Final remarks and closing

Instructions for participants

All sessions will be held online using ZOOM. All registered participants will receive a ZOOM link not later than November 20. We kindly request participants to change your ZOOM nickname to your full name and organization:

example — *Alexander Spitsyn (Kurchatov Institute, Russia)*.

A test session will be organized on November 23 from 11.00 to 14.00 msk (all speakers will receive a link to participate in the test session). If you are oral presenter, please try to operate your presentation file via the share screen function as a practice of presentation.

We kindly request all speakers to send their presentations to the organizing committee via e-mail *funfi@mephi.ru* *before or just after* you speech. The presentations will be available for the chairs of the sessions to prepare summaries and not be available for others.

The poster session will also be held online. All posters will be available at cloud storage to all registered participants from November 24th. The link to the cloud storage with the posters will be sent to all registered participants by e-mail not later than November 24th. If you are poster presenter please sent your poster to the organizing committee via e-mail *funfi@mephi.ru* not later than November 23.

Please read the posters interesting for you before the poster session. Discussion of posters will be on November 25 at 19.00 of Moscow time. All poster authors will have to participate in the section and be ready to answer questions from the other participants. The sequence of discussed posters will correspond to it order in the program. We expect that 10 minutes will be spent to discuss of each poster.

ABSTRACTS

IAEA's Coordinated Research Project on Compact Fusion Neutron Sources

Matteo Barbarino

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Keywords

IAEA, fusion, neutron sources

Abstract

The International Atomic Energy Agency (IAEA) is conducting a Coordinated Research Project (CRP) on "Development of Steady State Compact Fusion Neutron Sources" [1] during the period 2018–2022. This CRP is based on a previous project (2012–2016) [2], which investigated a wide range of power options for CFNS confirming the potential for significantly higher intensities and fluxes in the near future. The ongoing activities are organized under the topics: physics basis for steady-state CFNS; tokamak-based CFNS designs; mirror machine-based FNS designs; and dense plasma focus-based FNS designs. The CRP status and objectives will be presented. These include:

- Developing physics basis, numerical models, and simulation tools for plasma, nuclear processes and their interaction.
- Formulating concepts for enabling technologies and associated materials and propose corresponding R&D programmes supporting the transition to engineering design.
- Exploring plasma parameter spaces for optimizing core and edge plasma performance for neutron production at fusion energy gain value Q = 0.1–1.
- Establishing the suitability of steady state CFNS with typical fusion power in the range 1–100 MW (intensity 10^{17}–10^{19} n/s), neutron wall loading in the range 0.1–1 MW·m^{-2}, for dedicated applications, targeted products and services.
- Addressing facility safety issues at plant systems level and integrated level as applicable.
- Bringing together the stake holders and end users (such as the nuclear energy sector including fusion and fission, the basic research sector, biology and medicine sectors) of fusion neutrons for fine tuning specific design requirements for CFNS.

[1] IAEA CRP Website, https://nucleus.iaea.org/sites/fusionportal/Pages/CFNS2.aspx
[2] International Atomic Energy Agency, Conceptual Development of Steady-State Compact Fusion Neutron Sources, IAEA-TECDOC-1875, IAEA, Vienna (2019), https://www-pub.iaea.org/MTCD/Publications/PDF/TE-1875_web.pdf; Supplementary files, http://www-pub.iaea.org/MTCD/Publications/PDF/SupplementaryMaterials/TE-1875%20Supplementary%20Files.zip

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High Flux Accelerator-based Neutron Source

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Keywords

Neutron source, charge particle accelerator, lithium target

Abstract

An accelerator based neutron source is in operation at BINP, Novosibirsk [1]. This source is a state-of-the-art device comprised the Vacuum Insulation Tandem Accelerator (VITA) and an advanced long-life-time solid lithium target resistant to blistering. The accelerator is compact, as having a high ion acceleration rate. The ion beam energy can be varied in a range of 0.6 - 2.3 MeV with maximum beam current of 10 mA. The beam energy is highly stable and monochromatic (0.1%), stability of beam current is 0.5%. The accelerator can be operated with a proton and deuteron beam. The neutron source has been developed mainly for studies of Boron Neutron Capture Therapy [2]. Recently, the device has been actively applied for materials study. In-situ observation of metals blistering under irradiation by 2-MeV protons was performed for the first time [3]. Concentration of hazardous impurities in boron carbide samples developed for ITER was measured [4]. Recently, fast neutrons were generated [5] for testing of fibers for CERN and material samples for ITER. The report presents the recent results obtained in studies of the materials behavior under irradiation and future plans.

[1] S. Taskaev, Physics of Particles and Nuclei 50 (2019) 569–575
[2] Neutron Capture Therapy: Principles and Applications. Eds.: W. Sauerwein et al. Springer, 2012
[3] A. Badrutdinov, et al., Metals 7 (2017) 558
[4] A. Shoshin, et al., IEEE Transactions on Plasma Science 48 (2020) 1474-1478
[5] D. Kasatov, et al., Instruments and Experimental Techniques 63 (2020) 611-615

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Development and Applications of HINEG High Intensity Neutron Sources

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Keywords

Neutron source, HINEG

Abstract

Neutron sources are the important experimental platforms for the R&D of advanced nuclear energy and nuclear technology application. The High Intensity Neutron Generator (HINEG) has been developed in China with different missions including neutronics design validation, material & components irradiation test, nuclear waste burning and nuclear technology application, etc.

HINEG-I has achieved the fusion neutrons with the yield of 6.4×10^{12} n/s at maximum, and has been coupled with the Lead-based Zero Power Critical/Subcritical Reactor named CLEAR-0, which is actually an accelerator-driven Fusion-Fission Hybrid System. Series of typical experiments have been carried out on HINEG-I, including neutronics and code validation, core physics study of advanced reactors, neutron radiography, neutron detector calibration, neutron biological effects, neutron radiation hardening, and so on. HINEG-II is an accelerator-based neutron source with the yield of 10^{14}-10^{15} n/s. It aims to apply to multi-purposes, e.g. neutron capture therapy, isotope production, etc. The design and R&D for key technologies of HINEG-II are performed on-going. HINEG-III is initially conceived as a GDT-based or accelerator-based neutron source with the intensity of 10^{17}-10^{18} n/s. The objectives of HINEG-III are to conduct test of nuclear materials, components test and reliability data collection of nuclear components, nuclear waste burning test, etc. This contribution presents an overview of the series recent activities.

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High field tokamaks as compact neutron sources for FFH reactors

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Keywords

Tokamaks, neutron sources

Abstract

Fusion reactor scaling laws can be used to define plasma parameters of high field compact tokamak for low fusion gain applications as it is the case of Fusion-Fission Hybrid (FFH) reactors [1]. The attractive physics of the high field tokamaks explored on ALCATOR devices [2] and FTU [3] has demonstrated that the tokamaks (at magnetic field B≥7T) can be operated in L-mode with substantial increase of the quality of confinement. In particular the operation with pellets is attractive in this respect because it opens up the possibility of operating the machine in ohmic plasmas at very high density (plasma density $n_e \geq 1\ 10^{20}\ m^{-3}$) with reasonably good confinement properties. The compact high field tokamaks are characterized by a fundamental ohmic confinement with possibly a low RF heating power because of the limited space available.

The paper is dedicated to i) a summary of the physics of high field tokamaks as derived from the database of FT, FTU and ALCATOR devices (including ALC C-MOD); ii) a discussion of their confinement properties in scenarios operated with pellets; iii) the scaling laws for fusion reactors which allow for the evaluation of the geometry and plasma parameters of compact high field (B≥7T) tokamaks at low aspect ratio (A≥2.5); iv) the technical feasibility of these neutron sources and the technology readiness level.

[1] Francesco Paolo Orsitto and Thomas Noel Todd , Tokamaks as neutron sources for Fusion-Fission hybrid reactors:analysis of design parameters and technology readiness levels , Proceedings Conference FUNFI3 , edited by A Pizzuto and F P Orsitto, ENEA 2019, ISBN:978-88-8286-384-5
https://www.enea.it/it/seguici/pubblicazioni/pdf-volumi/2019/iii_international_conf-fusion-fission.pdf
[2] M Greenwald et al , Enegy confinement of high density pellet fuelled plasmas in the ALCATOR C Tokamak , Phys Rev Lett 53(1984) 352
[3] C Gormezano et FTU Team, Highlights of physics studies in the FTU , Fusion Science and Technology 45(2004) 303

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Shielding design and neutronics calculation of the GDT based fusion neutron source ALIANCE

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Keywords

Gas dynamic trap, fusion neutron source, shielding optimization, specific activity

Abstract

The Axisymmetric LInear Advanced Neutron sourCE (ALIANCE) is a GDT based fusion neutron source, aiming to produce 10^{18} of D-T neutrons per second and to provide a large testing volume required for fusion materials and components testing. A similar neutron source could also be valuable in the long run for medical isotopes production and as a driver of sub-critical fission reactor.

This presentation presents the high flux neutron shielding design and extensive neutronics calculations of GDT based fusion neutron source ALIANCE. Neutron distribution of ALIANCE is strongly inhomogeneous along the axis: significant portion of the neutron flux is generated near the two mirrors, while the rest of it is spread over the remaining central volume of plasma. The shielding design includes 40 cm stainless steel as the main shielding layer and an additional tungsten shielding layer at mirror plugs to protect superconducting coils from neutron damage and reduce nuclear heating. The simulations have been carried out by using Monte Carlo transport code SuperMC with nuclear data library FENDL 3.1. Results show that the nuclear heating on the mirror coils can be reduced by more than two thirds with additional tungsten shield, and fast neutron fluence by 30%. The highest nuclear heating and the highest fast neutron fluence zones are located at the mirror coils, and the values are about 300 W/m^3 and 9×10^{18} n/cm^2 respectively, which meets the threshold of ITER superconducting coils. The specific activities of shielding layers are on the order of 10^{12} Bq/kg. Ten years after shutdown, specific activities of structural materials will decrease by more than 98% to meet the safety requirements. The modeling and calculations reported in this paper will be beneficial for the pre-conceptual engineering design of ALIANCE.

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3D coils for a compact min B mirror field with minimal flux tube ellipticity

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Keywords

Mirror machine, Minimum B, Fusion neutron source

Abstract

Several crucial properties can be achieved with a minimum B vacuum magnetic mirror field. First, robust flute stability is still provided when the beta is increased to large values, and the beta enhances the mirror ratio [1]. The flux tube ellipticity is tolerable with proper shaping of the magnetic field [1]. A weak radial electric field, controlled by biased end plates in the low density expander tank region, can force each guiding center to drift in the vicinity of its mean flux surface. The end plate set thus control the positioning of the drift surfaces. The optimal vacuum magnetic field for these purposes has a minimized diamagnetic drift, which corresponds to the SFLM (Straight Field Line Mirror) field. Mirror ratios above 10 can be achieved for a long-thin SFLM with expansion vessels. Previous studies have predicted efficient scenarios for plasma heating by ICRH [2], and core heating by ECRH is expected. Superconducting coil arrangements provide a more than 1 m wide annular region where necessary reactor components could be placed [3]. That geometry is feasible for shielding, tritium breeding, coolant and other crucial ingredients for a fusion neutron source [4]. Previous studies had a focus on a 20 m long 10 MW fusion power device. We are under way to explore scenarios for more short-fat configurations. This will be initialized with determination of 3D super conducing coils for a compact device, less than 8 m long.

[1] O. Ågren, V.E. Moiseenko, "Magnetic shaping for a minimum B trap", accepted for publication in Fusion Engineering and Design (2020).
[2] V.E. Moiseenko, O. Ågren, Phys. Plasmas 12, 102504 (2005).
[3] A. Hagnestål, O. Ågren, V. E. Moiseenko, J Fusion Energ 31, 379 (2012).
[4] K. Noack, V.E. Moiseenko, O. Ågren, A. Hagnestål, Ann Nucl Energ 38, 578 (2011).

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Developments for stellarator-mirror fusion-fission hybrid concept

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Keywords

Fusion-fission hybrid, stellarator, magnetic mirror, transmutation

Abstract

The developments on a stellarator-mirror fission-fusion hybrid concept [1,2] are reviewed. The hybrid consists of a fusion neutron source and a powerful sub-critical fast fission reactor core. The major aim of the hybrid is transmutation of spent nuclear fuel and safe fission energy production. In its fusion part, a stellarator-type system with an embedded magnetic mirror is used. The stellarator confines deuterium plasma with moderate temperature, 1-2 keV. In the magnetic mirror, a hot component of sloshing tritium ions is trapped. There, the fusion neutrons are generated. The magnetic mirror is surrounded by a fission reactor mantle where the transmutation and the energy production take place.

A candidate for a combined stellarator-mirror system is a DRACON magnetic trap. It differs from the classical DRACON with a single (instead of double) magnetic mirror with a relatively short size. For high energy tritium ions, comparative computations of collision-less losses in the mirror part of this specific design of the DRACON type trap [3] predict that high energy ions from neutral beam injection can be satisfactorily confined in the mirror part.

The Uragan-2M stellarator device is used to experimentally check key points of the stellarator-mirror concept. The creation of an embedded magnetic mirror in Uragan-2M is arranged by switching off one toroidal coil. A weak radial electric field, which is foreseen to be spontaneously created by the ambipolar radial particle losses, can make drift trajectories closed, which substantially improves particle confinement [3].

A version of the fuel cycle for the stellarator-mirror fusion-fission hybrid is considered. The material for the fuel is a representative spent nuclear fuel with uranium and fission products depleted. The analysis is based on a scenario that for each transuranic (TRU) fuel load into a hybrid reactor, only a small amount, 10%, of TRU elements are burned. Therefore, to achieve full TRU burnup, the spent TRU nuclear fuel should be repeatedly recycled. Despite that the TRU content varies in each cycle, the reactivity of the system do not change substantially that is tolerable for a sub-critical system.

[1] V.E. Moiseenko, et al., J. Fusion Energy 29 (2010) 65
[2] V.E. Moiseenko, et al., Plasma Phys. Control. Fusion 56 (2014) 094008
[3] V.E. Moiseenko, et al., Plasma Phys. Control. Fusion 58 (2016) 064005

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Design and research program of the ALIANCE-T experiment

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Keywords

Gas dynamic trap, fusion neutron source, superconducting magnets, plasma source

Abstract

ALIANCE-T is a pilot experiment aiming to test the core principles of operation of the ALIANCE [1] volumetric fusion neutron source and to study specific plasma physics problems related to continuously operating gasdynamic mirror traps [2]. Research program of ALIANCE-T focuses on achievement of continuous discharge in axisymmetric magnetic field with high mirror ratio (~ 100) with parameters of plasma suitable for neutral beam injection. The research program includes experimental evaluation of several possible plasma generation techniques, studies of plasma stability against magnetohydrodynamic (MHD) perturbations, testing of stabilization methods most suitable for a continuous discharge, studies on interaction between neutral gas and plasma and its influence on the overall energy confinement time. As a technology demonstration experiment, ALIANCE-T will feature several technical solutions envisioned for ALIANCE-1 [1] and succeeding machines such as modular magnet system with compact superconducting magnetic mirror solenoids, electrode-based plasma stabilization and limiting technologies suitable for a continuous discharge, continuous gas feed for plasma density control and cycling of neutral gas through pumping and gas feed systems. Successful completion of the experimental program is expected to provide well-defined solutions for the ALIANCE-1 experiment with neutral beam injection. The report presents general structure and parameters of the machine's subsystems, including the description of plasma source and its expected performance.

[1]. P.A. Bagryansky et al 2020 Nucl. Fusion 60 036005
[2]. A.A. Ivanov and V.V. Prikhodko 2013 Plasma Phys. Control. Fusion 55 063001

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Simulation of plasma parameters for ALIANCE project

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Keywords

Fusion neutron source, gas-dynamic trap

Abstract

An international project named ALIANCE was jointly started by INEST CAS and BINP SB RAS [1]. The project assumes design of three devices with increasing power. Results of plasma parameters simulation by DOL code [2] will be discussed in this talk.
The first is a prototype of continuously operating beam-driven device. It will have low power of injected beams and deep gas-dynamic regime of confinement. The second is an intermediate step. Beam power will be increased by an order of magnitude reaching the transition between gas-dynamic and kinetic confinement regimes. And the third is a full-scale neutron source. The key parameters of these configuration are presented in Table 1.

Table 1. Results of plasma simulations for ALIANCE project

Parameters	ALIANCE-I	ALIANCE-II	ALIANCE-III
Isotope mix	100% D	100% D	50% D + 50% T
Mirror-to-mirror length, m	4.4	7.7	20
Plasma radius at the central plane, m	0.3	0.3	0.3
Magnetic field in plugs/center, T	12 / 0.2	13 / 0.3	25 / 0.8
Power of injected beams, MW	0.1	4	50
Ion/electron temperatures, keV	0.04 / 0.03	0.5 / 0.4	1.2 / 2.6
Neutron yield rate, 1/s	4·109	5·1013	1·1018

[1] P.A. Bagryansky, et al., Nucl. Fusion, 60 (2020) 036005
[2] D.V. Yurov, et al., Plasma Phys. Rep. 42 (2016) 210

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Tokamaks with external X-points: stability limits and new prospects

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Keywords

Tokamak, divertor, MHD stability

Abstract

Tokamak plasma with negative triangularity is actively investigated both in experiments [1, 2] and in theoretical studies on plasma confinement [3] and power exhaust in demonstration reactors [4]. Calculations of ideal MHD stability for the plasma with negative triangularity and X-points shifted to the outer side of the torus [5, 6] confirmed that, despite the absence of a magnetic well, the stability limits with respect to external kink modes are compatible with the requirements for the normalized beta value in the reactor $\beta_N \geq 3$. At the same time, the plasma pressure in the pedestal at the plasma edge is 4 times lower compared that of in conventional tokamaks with D-shaped cross section and positive triangularity. Therefore, strong ELM (Edge Localized Modes) disruptions, unacceptable in large tokamaks, are robustly avoided. This opens up prospects for considering configurations with an external X-point, exhibiting similar MHD stability properties. The external X-point divertor configuration has been tested in the JT-60 tokamak (1985 - November 1989, upgraded to JT-60U (1991-2010), see review [7]) with limited success: the high confinement operation mode (H-mode) in hydrogen plasma was not properly attained. However, even then it was clear that the presence of an X-point on the outer side of the torus does not contradict high beta values [8]. New calculations of ideal MHD stability for configurations with an external X point show beta limits $\beta_N \geq 3$. Plasmas with external X points and cross section elongation close to unity are passively stable against axisymmetric n = 0 modes. Possible scenarios for the formation of the outer X-point are considered in simulations of the plasma equilibrium evolution with a free boundary by ASTRA/SPIDER code using a simple system of poloidal field coils.

[1] Y. Camenen et al 2007 Nucl. Fusion 47 510
[2] M.E. Austin et al 2019 Phys. Rev. Lett. 122 115001.
[3] M. Fontana et al 2018 Nucl. Fusion 58 024002.
[4] M. Kikuchi et al 2019 Nucl. Fusion 59 056017.
[5] S.Yu. Medvedev et al 2015 Nucl. Fusion 55 063013.
[6] A.Merle et al 2017 Plasma Phys. Control. Fusion 59 104001
[7] M. Kikuchi et al 2018 Eur. Phys. J. H https://doi.org/10.1140/epjh/e2018-90054-2
[8] T.Ozeki et al 1988 Nucl. Fusion 28 1861.

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Novel hybrid reactor concept based on Ignitor technologies and physics

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Keywords

FFH, Tokamak, MSFR

Abstract

A fusion-fission hybrid (FFH) reactor is a coupled system composed by a fusion machine driving a subcritical fission blanket. In this frame the fusion machine is mainly considered as a neutron source for the fission complex which will provide to the man power contribution for the coupled reactor. A similar system could have many advantages, like a good fusion power amplification due to the presence of a fission system but, on the other hand, the system is very complex from the technological point of view. At the moment some countries have considered the idea to stat with a concrete realization of FFH, moreover some of them would consider FFH as step zero.

In this contest we propose a study for a FFH based on Ignitor fusion machine. Ignitor is a unique machine, in the Tokamaks panorama. In particular Ignitor is very suitable for a FFH due to its compactness and its high neutron intensity also in D-D fusion mode.

The main Ignitor characteristics are:

- R_0 = 1.32 m, a = 0.47 m
- B_0 = 13 T
- I_p = 11 MA
- $T_e(0)$=11.5 keV, $T_i(0)$=10.5 keV
- Ohmic heating system + ICRF
- $t_{\text{flat-top}}$ = 4 s
- $t_{\text{cooling}} \approx$ 1h
- P_{fus} ≈100 MW
- $J_n \approx 1\times10^{14}$ n/cm^2 s neutron current on outer FW, Wth≈2.3 MW/m^2 neutron thermal wall load.

Since our aim is to use Ignitor in D-D mode, we don't need the tritium breeding blanket which can be substituted by a fission blanket in order to investigate the main characteristics of this coupled system. A molten salt fission blanket could be particularly adapted for this purpose to avoid mechanical and thermal stresses caused by fusion machine pulses which could represent a complex problem for a traditional fission system. This kind of fission blanket could be more flexible and modeled in order to maximize the system fission yield.

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Comparison of lithium divertor options for DEMO-FNS tokamak

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Keywords

Divertor, lithium, heat loads, DEMO-FNS

Abstract

The issues of heat loads into the tokamak divertor still remain unresolved engineering and physical problems of a thermonuclear reactor development. Several approaches are being discussed and studied to reduce the heat flux onto divertor plates below 10 MW/m^2. Among them, the attractive approaches proposed by Y. Nagayama [1] and R. Goldston [2] are based on a liquid lithium evaporation in a limited amount of divertor boxes. They are relatively simple in practical applications and allow one to achieve necessary steady-state regimes both for heat fluxes and for plasma effluxes from a plasma column to the divertor region. In order to choose one of them for prospective tokamaks the comparative analysis was conducted and presented in this report.

The superconductive tokamak DEMO-FNS [3] that is currently being developed in Russia will operate in the regime of fusion neutron source (fusion power during a steady-state regime up to 40 MW, neutron production rate above $1 \cdot 10^{19}$ s^{-1}). It was shown that the system with "cold" or "condensing" box walls (about 200°C) which can condense Li (like in [1]) looks more suitable compared with "hot" or "evaporating" (up to 800°C) box wall system [2]. Both approaches fit the criterion for heat loads onto the divertor walls. The lithium efflux from the divertor into SOL and into the core plasma is sufficiently small for evaporating box system, while for the condensing box it slightly exceeds the non-contamination limit which can increase heat losses in SOL. The condensing box design is preferable due to an engineering simplicity. It was demonstrated that lithium efflux into the main plasma is suppressed mainly due to the ionization process within SOL which exponentially depends on its width. It is necessary to perform detailed 2D modeling of SOL and divertor construction with lithium for DEMO-FNS plasma parameters determination in divertor and SOL. Engineering and technological study of such design is also required.

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Neutron source based on circle cross section tokamak

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Keywords

Neutron source, circle cross section tokamak

Abstract

The possibility of a neutron source based on a tokamak with a circular cross section is numerically investigated. The circular cross section tokamak has range of advantages such as simple technology as well as a passive plasma vertical stability. Plasma configurations with inductive current ramp up and maintenance of the plasma current with the solenoid is considered. The iron core inductor is suggested to damp the scattered magnetic field and for increase of the volt-seconds supply. ECRH as an additional heating to electrons is considered. For the search of the optimal size of the device numerically a zero-dimensional energy balance model is constructed. The maximal discharge time length and the maximal neutron yield are the criteria for optimization. Calculations are carried out with one-dimensional model with transport code ASTRA for the current ramp-up stage and for the regime of the neutron production. The work also answers the question of the sufficiency of only additional ECR power in these regimes.

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Diffusion characteristics of radiation defects in iron: molecular dynamics data

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Keywords

Neutron irradiation damage, radiation defects, molecular dynamics

Abstract

Diffusion of radiation defects to sinks (dislocations, grain boundaries, sub-boundaries, interfaces, etc.) and their absorption lead to the evolution of the microstructure of structural materials of fusion and fission reactors and, as a consequence, to a change of their physical and mechanical properties (radiation creep, radiation embrittlement, and void swelling of materials). To build physical models of the material properties changes under irradiation, mechanical and thermal loads, it is necessary to know the characteristics of radiation defects, which are the parameters of such models. Fast neutron irradiation causes atomic collision cascades generating not only self-point defects (SPDs) but also their clusters [1]. Molecular dynamics (MD) study of primary radiation damage in iron [2, 3] has shown the higher the cascade damage energy, the higher the clustering of self-interstitial atoms (SIAs): the fraction of single SIAs decreases from 70% to 40% with an increase in the damage energy from 1 keV to 50 keV. In the specified range of damage energies, 30% to 40% of survived SIAs is concentrated in clusters containing from 2 to 5 SIAs. Diffusion characteristics of single SPDs and di-interstitials have been calculated in iron by MD [4, 5]. In this work, diffusion characteristics of clusters containing up to 5 SIAs are determined for iron in the temperature range 300 – 1200 K by MD. The simulation use the same interatomic interaction potential as in [2-5], since it describes well the experimentally known bulk properties of iron and the properties of SPDs [6]. The temperature and size dependences of the SIA cluster diffusion mechanism (1D vs 3D) and their possible consequences for the evolution of the microstructure of the material under irradiation are discussed.

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BTR code application for Current Drive analysis in FNS-ST

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Keywords

FNS-ST, fusion neutron source, spherical tokamak, NBI, neutral beam, injector, beam ionization, shine-through losses, fast ions, slowing-down, current drive, BTR code

Abstract

The fusion neutron sources (FNS) [1] are considered as a promising tool for testing structural materials of future nuclear and fusion reactors, scientific research in the field of nuclear waste disposal, fuel production and control of subcritical nuclear systems, as well as development of hybrid technologies for pilot industrial hybrid power plants. Spherical tokamaks have better steady-state and confinement characteristics among other FNSs, and can be considered as the basis for neutron sources of smaller size and lower cost. For the neutron generation the nuclear fusion reactions occurring with a high-energy beam slowing down seem the most attractive.

The FNS-ST spherical tokamak is of particular interest for the development of a neutron source, since it can provide a high steady-state neutron flux. When choosing the device operation parameters, it is assumed that the non-inductive current drive in the FNS-ST is available by energetic atoms injection. The optimal operating scenarios search of the beam-plasma system is done in the parameters range of maximum values of the neutron yield, the latter corresponds to the maximum values of the plasma current [2].

For fast atoms injection to the FNS-ST deuterium-tritium plasma, a deuterium beam with energy of up to 140 keV and the injected power P_{NBI} = 3.5 MW is supposed. As shown in [2], the reactions on the high-energy "tail" make the largest contribution to the neutron production rate. Taking into account the realistic spatial and angular geometry of the injected beam, calculated by the BTR code [3], and with using the analytical approach for ion slowing-down in plasma [4], the distribution functions of fast ions in plasma volume are obtained for various operating parameters and injection geometry. On the basis of these distributions, the radial profiles of the beam driven current are calculated. Shine-through losses and the corresponding power maps on the tokamak first wall are obtained. With account of the shine-through power restrictions, the optimal parameters of the beam-plasma system are determined, and the attainable values of neutral beam current drive efficiency are estimated.

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Radiation defect formation in bcc metals irradiated with high energy protons, self-ions and neutrons with different spectra

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Keywords

Radiation-induced defects, bcc metals, high-energy neutrons and protons

Abstract

Currently, tungsten and tungsten coatings are the reference materials of the ITER divertor and DEMO reactors and the possibility of using low-activated ferrite-martensitic, RAFM, steels not only as structural materials, but also as the material of the first wall of the fusion reactor is considered. Also, these steels, together with a new generation of RAFM steels with oxide dispersion strengthened by adding Y2O3 nanoparticles, the so-called ODS steels, are considered as promising materials for fast neutron fuel cladding. Neutron damage results in degradation of properties and activation of materials, and therefore affects performance and safety aspects. As a power fusion neutron source does not exist yet, to simulate fusion neutron-induced damage in materials, fission neutrons and charged particles are widely used. In this work, the size and density of radiation-induced defects in bcc metals (W, Mo and Fe) produced by high-energy self-ions, protons and neutrons with different spectrum have been compared by well-established method of positron-annihilation lifetime-spectroscopy (PALS). We found a formation of the larger size of the defects with lower density in the case of irradiation with high-energy neutrons from the p(35 MeV)-Be source compared to fission neutron- and proton- irradiations. New experimental data together with data available from the literature are compared with the dpa theory, including molecular dynamic simulations. Second, He/dpa ratios in different neutron facilities have been compared. We show that He/dpa ratios in the facilities with the hard energy spectra (fusion like) p(35 MeV)-Be source and DEMO are one-two orders larger than in the fission ones LVR-15, HFIR and BOR60. Methods to obtain the best approach to model fusion neutron damage and to overcome the gap between theory prediction of primary defect formation and long-term damage are discussed taking into account the uncertainties.

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Development and integration study of fusion-fission hybrid systems into Russian nuclear power

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Keywords

Conventional tokamak, superconducting magnetic system, hybrid fuel cycle

Abstract

Transition to a closed nuclear fuel cycle supporting operation of thermal and fast power reactors is being carried out in Russia at present time. In this regard, it is important to develop novel technologies for of spent nuclear fuel management and radioactive waste incineration, as well as to develop energy valuable "fusion-fission" hybrid systems. The unification of nuclear fusion and fission reactions in a single design will make it possible to achieve fundamentally new characteristics and parameters of the nuclear energy system. Development of the fusion-fission hybrid facility based on superconducting tokamak DEMO-FNS [1-3] continues in Russia for integrated commissioning of steady-state and nuclear technologies at the power level up to 40 MW for fusion and 400 MW for fission reactions. This activity is included in the ongoing state research program of Russia in the field of nuclear power.

The project aims at achieving the stationary operation of the facility with the neutron wall load of ~ 0.2 MW/m^2 and the neutron fluence over the life cycle of ~ 2 MW·year/m^2, with the subcritical active core and the tritium production blanket surfaces of ~ 100 m^2. This is sufficient for testing materials and components in the spectrum of DT thermonuclear neutrons, as well as for energy production, transmutation technology, production of fission fuel nuclides and tritium.

The analysis of the interaction of the DEMO-FNS facility and further industrial options with the nuclear fuel cycle of nuclear energy is carried out. Such a facility could provide the burning of minor actinides accumulated by Russian nuclear energy fuel cycle during the operation of nuclear power plants in the future. Returning spent fuel to a closed fuel cycle after enrichment in a hybrid industrial facility will reduce the number of nuclear fuel storage facilities and generated radiotoxicity. The report will present contemporary achievements in design and describe operation scenarios for the DEMO-FNS facility.

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Pilot FFHR based on a RFP as a fusion core

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Keywords

Pilot FFHR, RFP, RFX-mod

Abstract

Studies are in progress in order to revisit the status and the potentiality of the Reversed Field Pinch (RFP) as a fusion core in FFHR, taking into account: (i) the recent progress in RFP physics brought by the results of RFX-mod (R=2, a=0.46, I_p=2MA), mainly about MHD modes control, (ii) the expected performance improvements resulting from the ongoing upgrade of the machine.

To cover the gap between the existing experiments and the hybrid reactor, according to the strategy emerged in FUNFI3, an intermediate step with a FFHR pilot experiment in which a RFP is the neutron fusion source is proposed.

Three different categories of issues are considered which require the assessment of:

- the RFP physics and the scaling laws at increased level of current and machine size;
- the technologies allowing increased RFP performances and continuous operation;
- a test bench for a hybrid blanket, combining Tritium production and fission reactions.

In order to optimize the pilot experiment approach in terms of cost reduction, best use of the step-by-step acquired knowledge and clear milestones towards the realization of a low power FFHR, a staged approach with increased complexity and investment is introduced [1].

In the proposed pilot FFHR the aim is the production of D-T fusion power with a RFP configuration (P~30MW, Q~0.4, continuous pulsed operation) and testing the blanket with limited fission fuel. The overall strategy of this approach and the details for each stage of the plant requirements, the tackled issues and the expected results in order to pass to the next phase will be presented in the talk.

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Progress on ST40 towards optimization for neutron production

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Keywords

Spherical Tokamak, CFNS, ST40

Abstract

Tokamak Energy Ltd. is developing a modular design of a fusion power plant [1]. In this new alternative route to Fusion Power, the economically feasible fusion power plant will consist of several low power modules and the auxiliaries are shared between modules. Fusion Module would not aim to be fully self-sufficient in tritium and even in full-scale power plant, self-sufficiency in tritium may be relieved, assuming that one of the modules will be customized for tritium production. CFNS based on a compact ST may be used as a tritium breeder, if neutron output is maximized.

ST40 (design parameters: R=0.4-0.6 m, R/a=1.6-1.8, I_{pl}=2MA, B_t=3 T, k=2.5, □pulse~1-2 sec, 2MW NBI, 2MW ECRH/EBW, DD and DT operations) is a compact spherical tokamak that is operational now and can be a test prototype of such CFNS. There are several ways how to use ST40 to demonstrate optimisation of neutron production. One way that will be discussed in detail – to optimise beam-plasma and beam-beam reactions. This can be done by accelerating fast ions injected by NBI to significantly higher energy, increasing efficiency of beam reactions to make neutron production more efficient. The cross-section of these reactions increases strongly with the energy of fast ions, and if their energy was increased to MeV level, the neutron production may increase by several orders of the magnitude compared to reactions in pure thermal Maxwellian plasma of 10 keV temperature range.

In a high field Spherical Tokamak confinement of such high energy ions is significantly better than in conventional aspect ratio tokamaks or in Spherical Tokamaks with low toroidal field. This effect helps to reduce fast ion losses, also increasing the efficiency of reactions on fast particles. Several approaches for fast ion acceleration will be discussed. The present status of ST40 and the latest results will be presented.

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Set up of a deterministic calculation model for the analysis of fusion-fission hybrid systems

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Keywords

FFHS, Montecarlo codes, deterministic codes

Abstract

A simplified Fusion-Fission Hybrid System (FFHS) deterministic calculation model has been developed in order to study the coupling between a fusion machine and a subcritical fission system.

Montecarlo codes are very flexible and they have to be considered as reference calculation tools for complex systems like FFHS. On the other hand deterministic codes can provide, in shorter calculation time respect to MC codes, parametric and sensitivity analysis, also by exploiting General Perturbation Theory (GPT) methodologies, which can usefully support the global FFHS theoretical analysis.

This work will describe the comparison between the results obtained for a simplified FFHS model by using a Monte Carlo code (MCNP6.1) and a deterministic code (ERANOS). In particular, in this preliminary study, integral parameters like k_{eff} , thermal power, neutron fluxes and some key reaction rate profiles, will be compared.

If the comparison between the two mentioned codes will give encouraging results further, and more refined, studies will be performed in the future in order to consider deterministic codes as a powerful workhorse for the FFHS analysis, including time dependent analysis devoted to transient and reactivity monitoring issues.

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Analysis of the scheme and parameters of the neutral beam injector for the FNS-ST tokamak

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Keywords

Tokamak, source, neutrons, injector, beam

Abstract

The work is devoted to the choice of the scheme and parameters of the fast deuterium atom injector for the neutral injection system of the FNS-ST tokamak [1]. The energy of fast atoms is an important parameter that determines the efficiency of current generation in the tokamak plasma and the neutron yield. As shown in [2], in the range of 100–200 keV with increasing energy, the fraction of neutrons produced by the D-T reaction in the thermal component of the plasma decreases, but the total neutron yield increases, which is provided by an increase in the intensity of the D-T reaction during the beam-plasma interaction. With increasing beam energy, a noticeable increase in the efficiency of current generation is also observed.

The given energy range is transitional for choosing an injector scheme based on either positive ion (PI) or negative ion (OI) sources, since the efficiency of neutralization of PI in this energy region rapidly decreases with increasing energy, while for OI it remains at about 60%. In favor of the OI scheme, the fact that the average angular divergence of a beam with OI is about half as small as in PI sources also works. On the other hand, the emission current density of ions in PI sources is almost an order of magnitude higher than the current density in OI sources, injectors with PI are more compact and their technology has been tested on many installations. In this paper, the analysis of injector schemes for FNS-ST is performed and a variant using PI sources with deuterium ion energies up to 140 keV is selected. Specific energy values will be determined by the operating modes and parameters of the tokamak plasma.

A variant of the injection module with the power of a deuterium atom beam injected into the plasma up to 3.5 MW is considered. The geometry and parameters of an ion source (II) with a deuterium ion beam current of up to 80 A and the geometry of the beam path components from the II grounded electrode to the entrance window to the tokamak chamber are determined. The power loads from the beam onto the components and their cooling conditions are calculated. The gas flow conditions in the injector are determined and the losses of the neutral beam due to the re-ionization of atoms on the background gas are estimated.

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Integrated modeling of fuel flows in the plasma and in the injection and pumping systems for the DEMO-FNS fusion neutron source

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Keywords

Fusion neutron source, fusion fuel cycle, core and divertor plasma modeling

Abstract

To calculate the fuel isotope flows in the DEMO-FNS fusion neutron source [1], the "FC-FNS" fuel cycle model [2] is used, which provides integrated description [3] of the gas, solid-state and plasma flows in the main and divertor plasma regions in conditions of neon injection into the divertor [4]. An approach to modeling the main plasma is developed, which takes into account the difference of the confinement times for the particles originated from different fuel sources. The "FC-FNS" code has been modified to fully match the architecture chosen in the analysis of the fuel cycle (FC) candidate technologies [5] for DEMO-FNS. The model is supplemented with a new scenario of supplying the gas for the heating injectors, which allows D^0+T^0 neutral beam injection with a closed gas cycle. Fuel flow simulations in the FC are carried out, where the parameters of fuel injection are selected to ensure the specified conditions in the main and divertor plasmas for various heating injector scenarios (the isotopic composition of the source gas). The working range is found for the isotopic composition in the divertor plasma, at which the required tritium fraction in the main plasma can be ensured. It is found that the injection of pellets for the ELM control is possible by different isotopic composition in the divertor without negatively affecting the main plasma parameters. It is shown that to ensure a 50% tritium fraction in the main plasma, it is necessary to maintain the tritium fraction in the divertor plasma within the range from 47% to 53% for the D^0+T^0 heating beam and from 54% to 60% for the D^0 beam. The selected value of the tritium fraction in the divertor plasma (that is, in the pumping system) determines the permissible frequency of D_2 pellet injection from the low magnetic field side for the ELM control (from 5 Hz to 110 Hz). The value of the total tritium inventory in the facility is evaluated as 850 to 1150 g.

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NRNU "MEPhI" activity towards fusion-fission hybrid systems

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Keywords

MEPhI, nuclear fusion, plasma wall interaction, tokamak, brazing

Abstract

The development of fusion-fission hybrid systems is a very complex and ambitious task. MEPhI is a key Russian University for students training in many aspects related to this topic, but it is also involved actively in R&D. General information and some key achievements of MEPhI departments in this area will be reported.

MEPhI has a unique complex of plasma and ion beam facilities for H/D and He plasma irradiation of candidate materials in a wide range of parameters and fundamental investigations of plasma-wall interaction [1]. These instruments allows to investigate plasma induced surface modification, erosion and various aspects of hydrogen isotopes retention, including co-deposition and influence of radiation damage. Most of research are currently performed with tungsten and advanced W alloys (in cooperation with FZ Juelich, Germany [2]), steels and aluminum as a proxy for beryllium. In support of concepts of renewable surface of plasma facing components, experiments and theoretical works on resistance of liquid lithium and B4C coatings are carried out. With such a fundamental knowledge, new methods of edge plasma and surface diagnostics, tritium and dust removal are developed for ITER and other tokamaks.

To join dissimilar materials in fusion devices, various methods of brazing using rapidly solidified amorphous alloys are developed for application in ITER [1,2] and DEMO [3] plasma facing components.

Recently, a small spherical tokamak MEPHIST has been built for student training and as a testbed for new methods of plasma and surface diagnostics, and various engineering solutions.

Another activity related to analysis of non-traditional nuclear fuels and involvement of fusion reactors into nuclear power systems will be presented as a separate report [5].

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Physical aspects for involvement of thermonuclear reactors into nuclear power systems

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Keywords

Hybrid thermonuclear reactor, protactinium-231, uranium-232, high fuel burn-up, non-proliferation

Abstract

The paper analyzes a possibility to involve hybrid thermonuclear reactors into the existing nuclear power systems. The possibility is related with production of non-traditional nuclear fuel in thorium blanket of hybrid thermonuclear reactors on D-T plasma. Non-traditional peculiarity of such a fuel consists in significant amounts of some non-traditional isotopes, namely ^{231}Pa and ^{232}U, together with traditional uranium isotope ^{233}U in the fuel. High-energy (14.1 MeV) thermonuclear neutrons can provide accumulation of significant ^{231}Pa and ^{232}U quantities through threshold (n,2n) and (n,3n) reactions. The promising features of the non-traditional fuel composition for nuclear power thermal reactors, basic component of the existing world-wide nuclear power industry, are defined by the following factors. As is known, ^{233}U is able to provide more economical neutron balance in thermal reactors than ^{235}U and reactor-grade plutonium. The better neutron balance can result in higher values of the fuel breeding ratio and, as a consequence, in relaxation of the thermal reactors fuel self-sustainability problem. Isotopes ^{231}Pa and ^{232}U, being fertile and moderate fissionable nuclides, are able to stabilize time-dependent evolution of the thermal reactors power, prolong the thermal reactors lifetime through higher values of the fuel burn-up [1, 2]. Isotope ^{232}U, being intense α-emitter, is able to prevent any attempts for unauthorized usage of ^{233}U in nuclear explosive devices, i.e. ^{232}U can strengthen regime of nuclear non-proliferation. Thus, the hybrid thermonuclear reactors on D-T plasma with thorium blanket can be involved into nuclear power systems for generation of non-traditional, very promising fuel compositions for traditional nuclear power reactors.

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Assessment of a possibility to produce non-traditional nuclear fuel in thorium blanket of hybrid thermonuclear reactor

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Keywords

Hybrid thermonuclear reactor, protactinium-231, uranium-232, high fuel burn-up, non-proliferation

Abstract

The paper analyzes peculiarities in isotopic composition of Th-blanket under irradiation in hybrid thermonuclear reactor (HTR) on (D-T) plasma. The (D-T) plasma was chosen because only high-energy (14.1 MeV) thermonuclear neutrons are able to initiate threshold (n,2n) and (n,3n) reactions. These neutron reactions result in accumulation of non-traditional fuel compositions that include isotopes ^{231}Pa, ^{232}U, ^{233}U and ^{234}U. Such non-traditional fuel compositions may be of interest for nuclear power plant projects with thermal reactors because they can lead to substantial prolongation of the reactor lifetime, higher fuel burn-up and to strengthening of nuclear non-proliferation [1, 2]. Neutron-physical calculations were carried out with application of the computer code Serpent-2 (Monte Carlo method with continuous energy presentation of evaluated nuclear data). The paper presents spectral parameters of Th-blanket, spatial distributions of neutron reaction rates, ^{232}Th burn-up and accumulation of target isotope compositions. Numerical results showed good agreement with experimental data on accumulation of target isotopes in Th-blanket of HTR on (D-T) plasma.

The following results were obtained:

1. The chosen model of HTR allowed us to form high-energy neutron spectrum in Th-blanket with significant fraction of 14 MeV-neutrons.
2. It was evaluated that threshold (n,2n) and (n,3n) reactions are able to produce significant amounts of non-traditional target isotopes ^{231}Pa and ^{232}U.
3. It was shown that accumulation of non-traditional target isotopes weakened substantially in depth of Th-blanket. So, it seems reasonable to seek for optimal thickness of Th-blanket and, thus, to find optimal loading of natural thorium.

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System studies on fusion-fission hybrid systems and its fuel cycle

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Keywords

Fusion-Fission Hybrid System (FFHS), Minor Actinides, Partitioning and Transmutation (P&T), Closed Nuclear Fuel Cycle, Monte-Carlo

Abstract

One of the most priority task for nuclear energetics is to create a closed nuclear fuel cycle that provides utilization of nuclear energy for a long time (more than 1000 years) and to improve the safety and ecological of nuclear technology. This problem includes the problem of spent nuclear fuel (SNF) and radioactive waste (RW) management. Minor actinides (MA) (Np, Am, Cm), that is about 0.1% of SNF mass, have high radiotoxicity and a relatively long half-life. They provide the main contribution to the long-term radiotoxicity of radioactive waste in case of their disposal. The radiotoxicity of fission products (FP) declines much faster: it reaches the radioactive equilibrium with respect to uranium ore in about 300 years. In this way, partitioning & transmutation (P&T) of MA by fission seems reasonable. Although a pretty hard spectrum is required for MA fission due to the threshold character of fission cross-section for most of MA. One of the most perspective ways to solve this problem is an application of a fusion reactor as a high energy neutron source. In this paper, the application of such type of hybrid reactors for MA transmutation has been studied.

The study includes two interconnected investigations. The first is study of fuel inventory evolution in the blanket of fusion-fission hybrid system containing metallic fuel that is zirconium and MA alloy. Three hybrid reactors with fusion power equal to 40 MW supposed by the road map of NRC "Kurchatov Institute" project are investigated: demo, pilot-industrial and industrial reactors. Performance parameters are compared for different coolants (H_2O and CO_2) in the transmutation zone. The design of the reactor with H2O as a coolant is chosen as basic.

The second part of the study is system analysis of the development of nuclear energetics in Russia with involvement of the hybrid reactors. The study conducted by USM-1 model that was created at ITCP «PRORYV»

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Americium and curium burning out in a fusion reactor

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Keywords

Spent nuclear fuel, thermonuclear reactor, MCNPX

Abstract

Today, a large amount of spent nuclear fuel from nuclear power plants has been accumulated in the world, but there is no established strategy for handling this fuel yet. The spent fuel can be partitioned. The majority isotope, Uranium-238, can be used to produce secondary fuel in fast nuclear reactors. Plutonium can be used as a component of the MOX fuel for burning in thermal nuclear reactors. Geological storage of fission products seems feasible since the radiation has decayed after 200-300 years, which is much shorter than the time required for the spent fuel. An important task is the handling of the remaining isotopes, mainly americium and curium, where special arrangements are required. These elements have different nuclear properties and cannot be handled like plutonium.

Fusion neutrons are quite efficient for incineration of americium and curium by means of their fission. The idea in this presentation is to consider fusion power plants for this purpose. The amount of americium and curium is chosen small enough to avoid a strong impact on the fusion reactor design and operation. Nevertheless, a high burn-up degree of americium and curium is possible.

For the calculations, a model of a toroidal thermonuclear reactor with a major plasma radius of 1000 cm and a minor radius of 300 cm is proposed. For such an installation, an addition of 10 tons of fuel (americium plus curium) to the structure will not have a substantial impact on the reactor design.

The burnup rates of americium and curium, which appear to be in within an acceptable range, are calculated by the MNCPX code. The contribution of fission neutrons to the production of tritium is observed. That could be important to achieve self-sufficiency of the fusion reactor for its tritium supply. The calculated reactivity for the studied reactor as a fission system is quite low, which enables a safe operation. Along with the minor actinide incineration and tritium breeding, the fission reactions provide a moderate power output increase, which is rated by tens of percent.

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Materials for hybrid systems

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Keywords

Structural and functional materials, neutron spectra, neutron irradiation damage, helium accumulation, nuclear physical properties

Abstract

Materials of hybrid systems (DEMO-FNS, others [1]) must withstand temperature, mechanical and radiation loads during their operating life without losing their functional properties, and be compatible with their environment (coolants, etc.). The materials should be possibly non-magnetic (not to disturb the magnetic configuration of the installation), low-activated (to ensure the possibility of recycling in a historically short time less than 100 years), and should not accumulate a large amount of tritium. Materials must be industrially developed, certified, and must successfully pass radiation tests. The working conditions of structural materials (SMs) in the DEMO-FNS blanket resemble the conditions of the core of fast reactors BN-600, BN-800 (neutron spectrum, temperatures, mechanical loads). The possibility of using in DEMO-FNS already existing standard and advanced structural materials for fast reactors (austenitic steels ChS-68 and EK-164, ferritic-martensitic steels EK-181 and ChS-139, vanadium alloy V-4Ti-4Cr) is evaluated. In order to reduce nuclear and radiation problems in the use of austenitic chrome-nickel steels, austenitic chrome-manganese steels of the type Fe-(10-18)Cr-(10-30)Mn-W have been proposed [2]. These austenitic steels are low-activated, non-magnetic, relatively low swelling, less prone to high-temperature embrittlement, and less expensive than austenitic chrome-nickel steels. The risks of using copper chrome zirconium alloys in DEMO-FNS are also considered. The work was supported by NRC “Kurchatov Institute” (grant no. 1934a from 28.09.2020).

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A reactivity monitoring method for fusion-fission hybrid reactors

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Keywords

Hybrid fusion-fission reactor, sub-critical multiplication factor, reactivity monitoring

Abstract

Based on the extension of the HGPT (Heuristically Generalized Perturbation Theory) methodology to sub-critical systems [1], a procedure is described for the on-line monitoring of the sub-criticality level of a fusion-fission hybrid system, where the "external" neutron source is generated in a magnetically confined plasma of tokamak type. This procedure, usually referred to as the PCSM (Power Control-based Sub-criticality Monitoring) method [2,3], implies compensating slow, small movements of a dedicated control rod in the fission core with equally slow, small alterations of the independent fusion neutron source. The alteration of the intensity of the fusion source, proportional to its power, can be performed in various ways, such as by modulation of the external plasma heating system (RF wave launcher, for example), or by compression/expansion of the plasma. In this work we consider the latter procedure, which can be realized by varying the equilibrium vertical magnetic field, an action which displaces the bulk of the plasma toward a region of higher/lower toroidal magnetic field, thus inducing compression/expansion. The coupling of the generalized perturbation analysis with the fusion power modulation by plasma displacement may provide an effective way to monitor the sub-criticality level of a fusion-fission hybrid reactor continuously.

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The diagnostics and control of the Fusion-Fission Hybrid tokamak based reactors: the technology for measurements systems

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Keywords

Fusion-fission reactors, diagnostics, blankets

Abstract

The Fusion-Fission Hybrid (FFH) reactor is a complex machine which includes a tokamak fusion neutron source and two blankets: the tritium regeneration zone and the actinide burner zone. These three systems need their own diagnostics and controls. The problem of the implementation and integration of these control systems leads to the necessity of simplification of the technology [1]. In this paper the criteria determining the diagnostics needs of FFH reactors are reviewed following the idea that the measurements systems must be simple and robust, and their number is very limited by the space left by the blankets. The diagnostics for the tokamak neutron source including the machine protection and burn control are the basic equipments. As the fusion and fission blankets can be integrated in a single subsystem their diagnostics must be conceived as an integrated package which includes the measurement of the content of isotopes, the neutron multiplication and effective reactivity of the fission blanket, as well as the tritium regeneration. The requirements on the measurements and the diagnostics and control technology readiness are discussed and compared to that elaborated for ITER [2] and DEMO [3] .

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