NATIONAL RESEARCH CENTRE KURCHATOV INSTITUTE

НАЦИОНАЛЬНЫЙ ИССЛЕДОВАТЕЛЬСКИЙ ЦЕНТР «КУРЧАТОВСКИЙ ИНСТИТУТ»

Kurchatov Complex for Fusion Energy &Plasma Technologies

Общие впечатления о 29-й конференции МАГАТЭ по термоядерной энергии и заседании Международного совета по термоядерным исследованиям (IFRC)

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Заседание Международного совета по термоядерным исследованиям (IFRC)

1. Richard Battery (GA, USA) – председатель конференции проинформировал совет о готовности конференции к реализации.
2. Обсуждены планы следующих конференций
   в Китае (Октябрь 2025) Хаян/Пекин
   в Корее (Октябрь 2027) Сеул

3. Мульти-машинные эксперименты поддерживаются (J-TEXT и HL-2M)
4. Сформированы 7 баз данных по атомарным и молекулярным процессам, Пучкам нейтралов, экранированию парами, проникновению водорода.
5. Вопросы синергии синтеза и деления (атомная наука, отходы, регулирование, D-T топливный цикл) – 20 совещаний запланировано на 2024 год
5. Обсуждена деятельность (Wagner) по Fus-Fis Hybrids, INPRO, ТЯ-электростанциям
6. Внимание к частному сектору (6.2 Мрд долларов) инвестировано 43 стартапа зарегистрированы. Внимание к обмену знаниями и достижению длительной работы технологических систем УТС. Совещания МАГАТЭ по теплоносителям и материалам. (на текущей неделе, я участвую).

Заседание Международного совета по термоядерным исследованиям (IFRC)

7. Проблемы регулирования и лицензирования. Безопасность.
8. Выбор нейтронных источников из радиационных и ядерных вариантов
9. Соотношение гибридных и ядерных установок в энергосистеме

Франческо Романелли (NF главный редактор и руководитель проекта DTT)
Поставил вопрос об обмене информацией между частным и государственным сектором, а также мАГАТЭ.

Watson IAEA отметил выпуск IAEA двух технинческих документов по синтезу

Sophy информировала и работе NF журнала. Он пока не доступен для РФ, но совместные статьи, вроде принимаются. В частности по ССО.
Принята с Б. Кутеев
Технологии и материалы превысили 30% объема журнала.
По инерционному синтезу публикаций мало. Хотя достижение 3 MJ взрыв при 2 MJ вклада лазерного излучения впечатляют.
Публикации частного сектора приветствуются.

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В IAEA группе синтеза с удовлетворением отмечают достигнутое гендерное равенство 4:3   L:M
Движение женщин в синтезе (WiF) достигает 600 человек, работающих в разных областях: Физика, Инженерия, Администрирование и др.

Состояние ИТЭР
Началась сборка первого сектора
С геометрией проблемы (не стыкуются части)
Коррозионные повреждения обнаружены
6,7,8 сектора должны быть отремонтированы
17 из 18 катушек доставлены на площадку
4 сектора вакуумной камеры доставлены
Криостат и катушка PF1 (RF) доставлены

Новая стратегия развития перенесена на 2024 год
DT переносится на вторую фазу
Be заменяется на первой стенке на вольфрам

Заседание Международного совета по термоядерным исследованиям (IFRC)

Новости
Казахстана – I =750 КА, T= 5 с, R= 0.9 м, a=0.45 м

Китай – HL-3 R= 1.78 м, a = 0.65 м I =1 MA, T =1 c
Кофигурация островного дивертора
Стенда технологий синтеза - много! 267000 m2 в SWIP. Развитие материалов!

Коста-Рика Meduse-CR токамак

Франция West phase 2 Полностью интегрированный дивертор. 100 s разряды. Высоко флуенсная кампания 10^26 D/m^2.
Толстые осаждения наблюдены. X точка -хороший радиатор.
Международная коперация: инжекция боровой пыли. Задачи для ИТЭР и ДТТ с литиевой стенкой
Томсоновское рассеяние, ECRH 3 гиротроны с 1 MW и SSO

Германия W7-X стадия настройки 2024 сентябрь-старт. 2 новые катушки. Срывы
T удержание в JET 25 MW/m^2 370 M$ до 2028 Частная Gauss Fusion

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Индия Raji Развитие технологий EH&CD, Технологии бланкета, Crio - насос, Пеллеты, Дистанционное обслуживание и робототехника, нейтронный источник 14 MeV стартовал.
Объемный нейтронный источник SST-2 2/3 от размера ITER
Италия Romanelly Национальная программа разработана и поддерживается правительством. Дорожная карта планирует в следующем году 600 специалистов. Синтез - Деление. DTT является крупным проектом стоимостью 614 млн евро. Общая сумма контракта составляет 1,8 млрд евро. Эксплуатационные расходы составляют 150 млн евро в год. Начало работы в этом десятилетии. Катушки TF находятся в стадии изготовления / потребляемая мощность 150 МВт. Изготавливается инжектор MITIGA. Сотрудничество DTT, SPARK, университетами Испании, IFMIF-DONES

Инновационная стратегия Японии Ешино до 2023 года. Поддержка частного сектора, регулирующих органов, ускорение исследований и разработок для DEMO JP. Инновационный центр/ Образовательные программы JT60SA должны начать работу на следующей неделе.

Корея DEMO программа >500 MW, R/a 6.8 m/2.1m, ST плазма, Vest, дистанционное управление, Товый полупроводниковый центр. Дистанционное обслуживание с США

Заседание Международного совета по термоядерным исследованиям (IFRC)

Великобритания, 2000 сотрудников, бюджет 315 миллионов долларов. Компании получают поддержку правительства, подготовлена следующая программа на 2024-2026 годы. 1 B$ поддержка термоядерного синтеза
Сформирована основная учебная программа МАГАТЭ, которая поддерживает конкретные термоядерные исследования, включая частный сектор, европейскую программу R&D. Евратом запускает с середины 2022 года проект стоимостью 1,38 млрд евро и поддерживает производство энергии до 2050 года. ITER, IFMIF, ДЕМО. ЕС пытается консолидировать общество и пересмотреть дорожную карту УТС в ЕС. Варианты регулирования термоядерных установок и их поддержка, а также оценка безопасности. Euro fusion необходим для лицензирования.

Международное энергетическое агентство Дэвид Мейссонье, агентство поддерживает государственно-частное партнерство. Новые формы (простые) годового отчета FPCC
Продвижение вперед, Синтез сегодня лозунг дня.
Агентство исключило РФ из совместных программ!
Сотрудничество со Стелларатором прекратилось.

Проблемы Голландии остаются в глобальной термоядерной индустрии до 2023 года / Для частных компаний с годовым бюджетом ~ 9 Млн $ рассматриваются различные подходы. Ожидается, что к 2030 году их доля составит 88%.

Следующее заседание IFRC в Июне-Июле 2024

Директора проекта ИТЭР

1.	1998-2000	Paul Rebut	токамачник	Англия
2.	2001-2004	Robert Aymar	ускорительщик	Швейцария
3.	2005-2010	Kiname Ikeda	дипломат	Япония
4.	2010-2015	Osamu Motogima	стеллараторщик	Япония
5.	2015-2022	Bernard Begot	управленец	Франция
6.	2023	Pietro Barabaschi	инженер	

Цитаты из доклада П. Барабаски
«В итоге полностью утеряна инженерная культура»
«Будущее ИТЭР неопределенно. Директор не уверен, что завершит проект»

Actual tasks of Russian Federation in the field of controlled fusion and fusion-fission hybrid systems

B.V. Kuteev

NRC «Kurchatov Institute», Moscow Russia

Thanks to P.P. Khvostenko, et all.

Introduction

General trends of fusion development in RF at the current period cover the activity on the National Strategy for Atomic Energy including Fusion and Fusion-Fission Hybrid Systems

Major participants in this process are SC Rosatom, Ministry of High Education, including Russian Academy of Science, and NRC Kurchatov Institute

Research activity concentrates around Tokamaks, Fusion-Fisison Hybrids and Laser systems

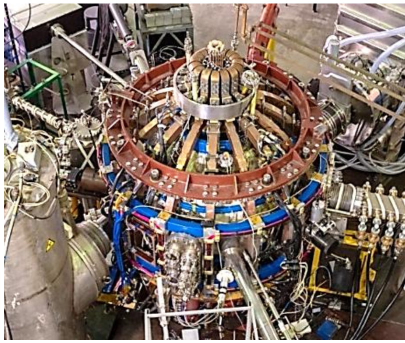
Tokamak based Hybrids are considered as important participants supporting operation of both future Nuclear and Fusion Reactors

Optimization of the nuclear fuel cycle in the AE system is actively considered within UPu and ThU nuclear fuel cycles of fission reactors

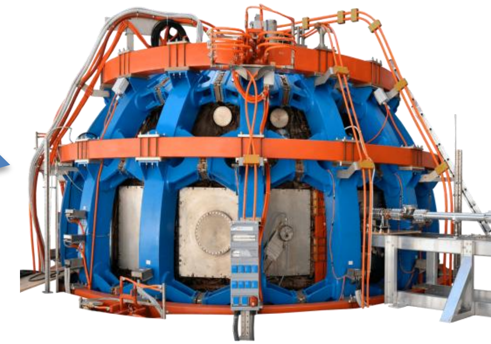
Development of Tokamaks as Fusion and Hybrid Reactors

Basic and Technology Research

Globus M2,T-11M

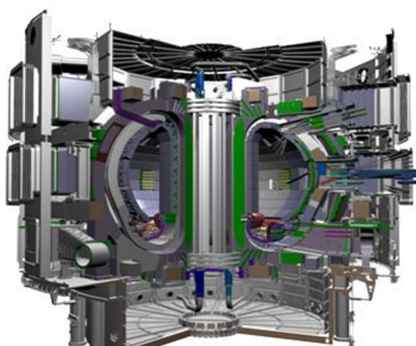


Tokamak T-15MD



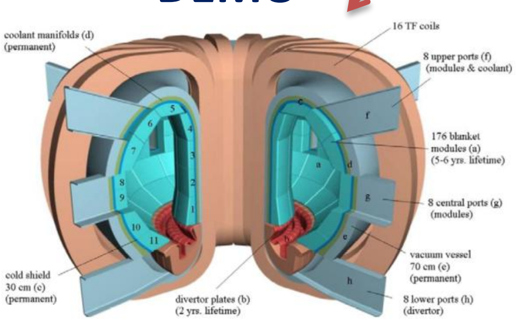
Tokamaks as Fusion Neutron Sources

ITER

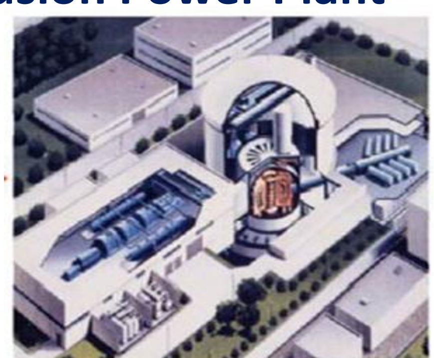


Hybrid facility for:
- fuel nuclides breeding
- transmutation
- energy production

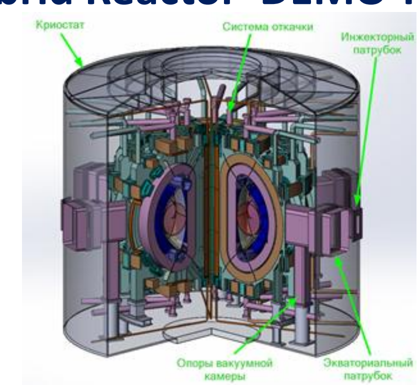
DEMO



Fusion Power Plant



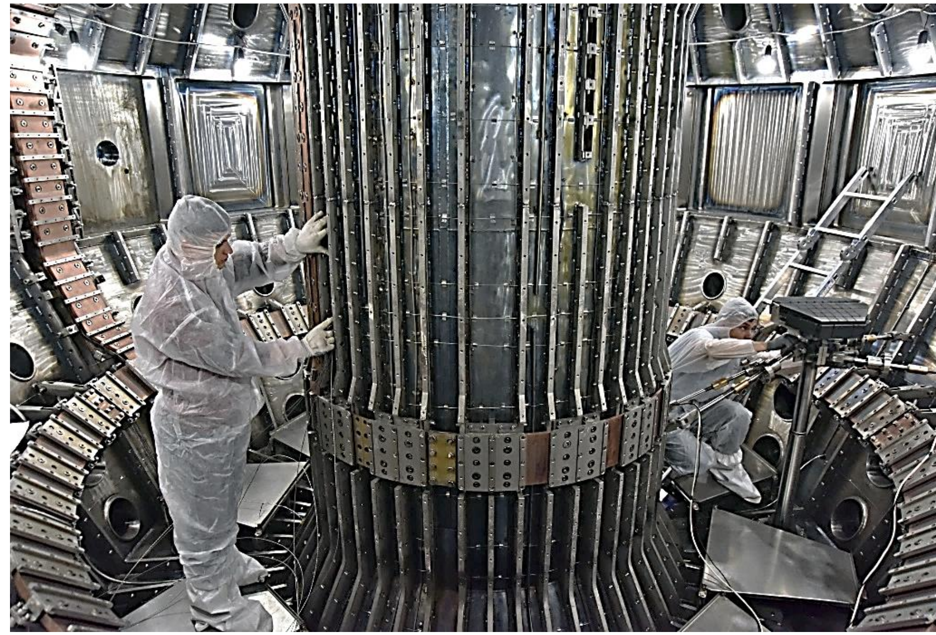
Hybrid Reactor DEMO-FNS



Modern facilities of RF Fusion program

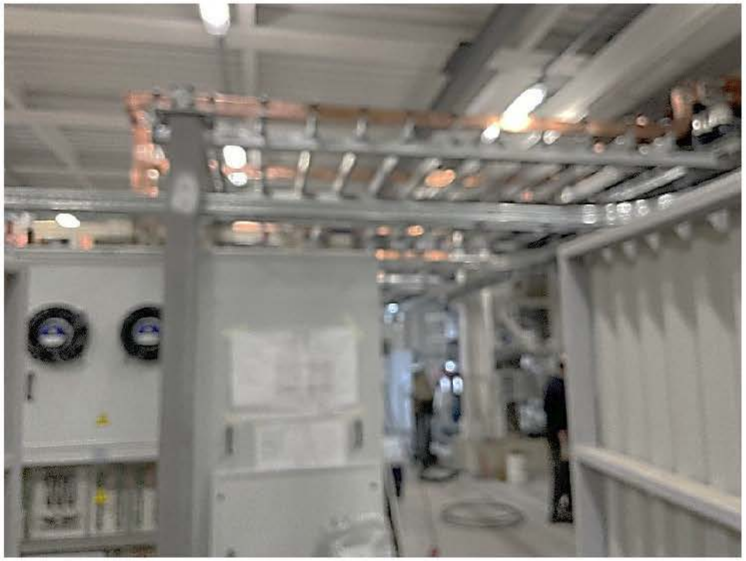
1. **Tokamak T-15MD – the first divertor tokamak of DIII-D and ASDEX-Upgrade scale (H and He discharges only)**
2. **Fusion neutron source FNS-C/FNS-ST – compact SSO device of ST-40 (GB) scale and ~5 MW of DT Fusion for materials study and Steady State Operation and Remote Handling Technology**
3. **Hybrid reactor facility (HRF/DEMO-FNS) – JET and JT-60SA scale device for developing and testing hybrid technologies at an industrial scale of nuclear power ~500-700 MW(t)**
4. **TRT – tokamak with HTSC magnetic system of DTT scale, high magnetic field and heating power up to 40 MW.**
5. **DEMO-scale development within IAEA activity**

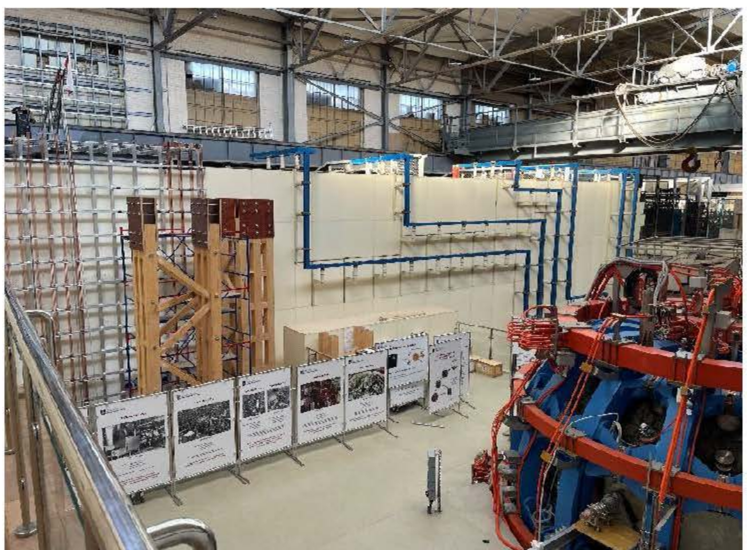
Inside T-15 MD

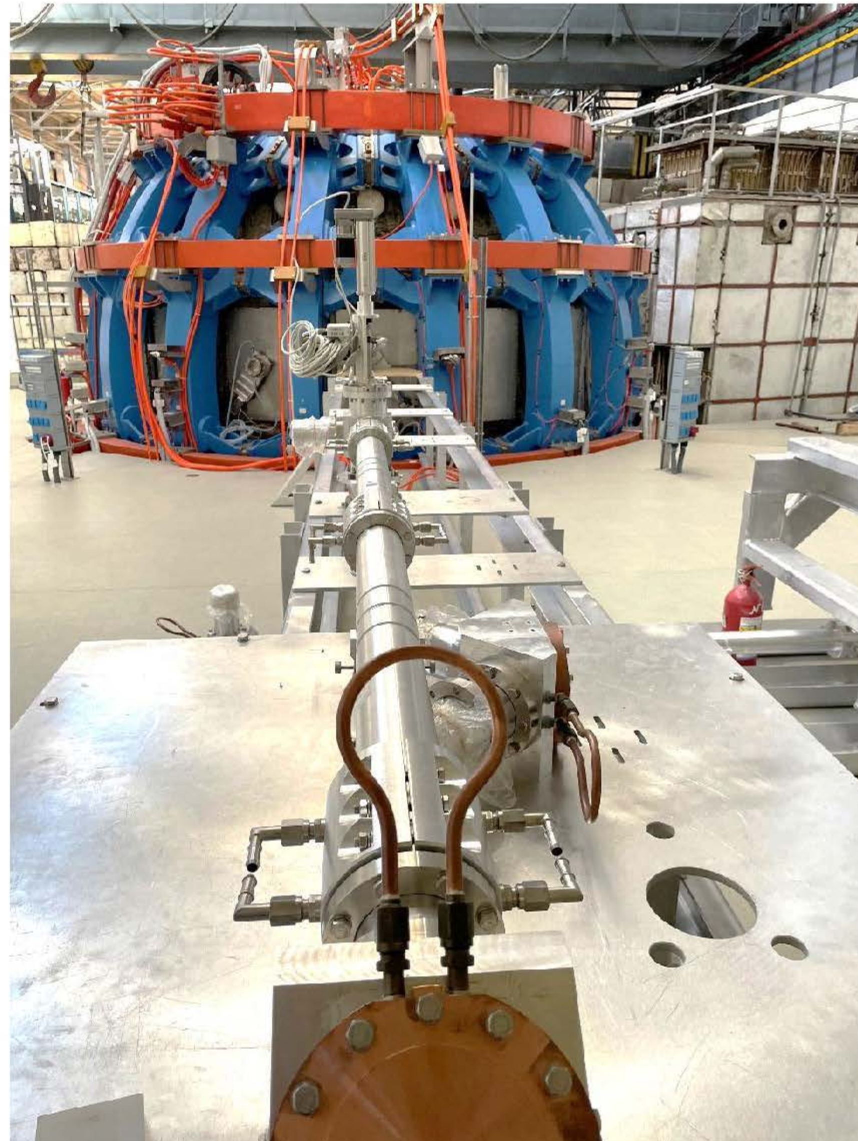


Installation of sensors for electromagnetic diagnostics inside the vacuum vessel

Supporting systems of T-15MD including gyrotrons







Long-term and SSO regimes for tokamaks

- SSO is an important goal in development of modern tokamaks

- T-15MD team hopes to reach tens of seconds for pulse duration
- FNS-C is aimed at operation of days
- DEMO-FNS/HRF - of weeks/month

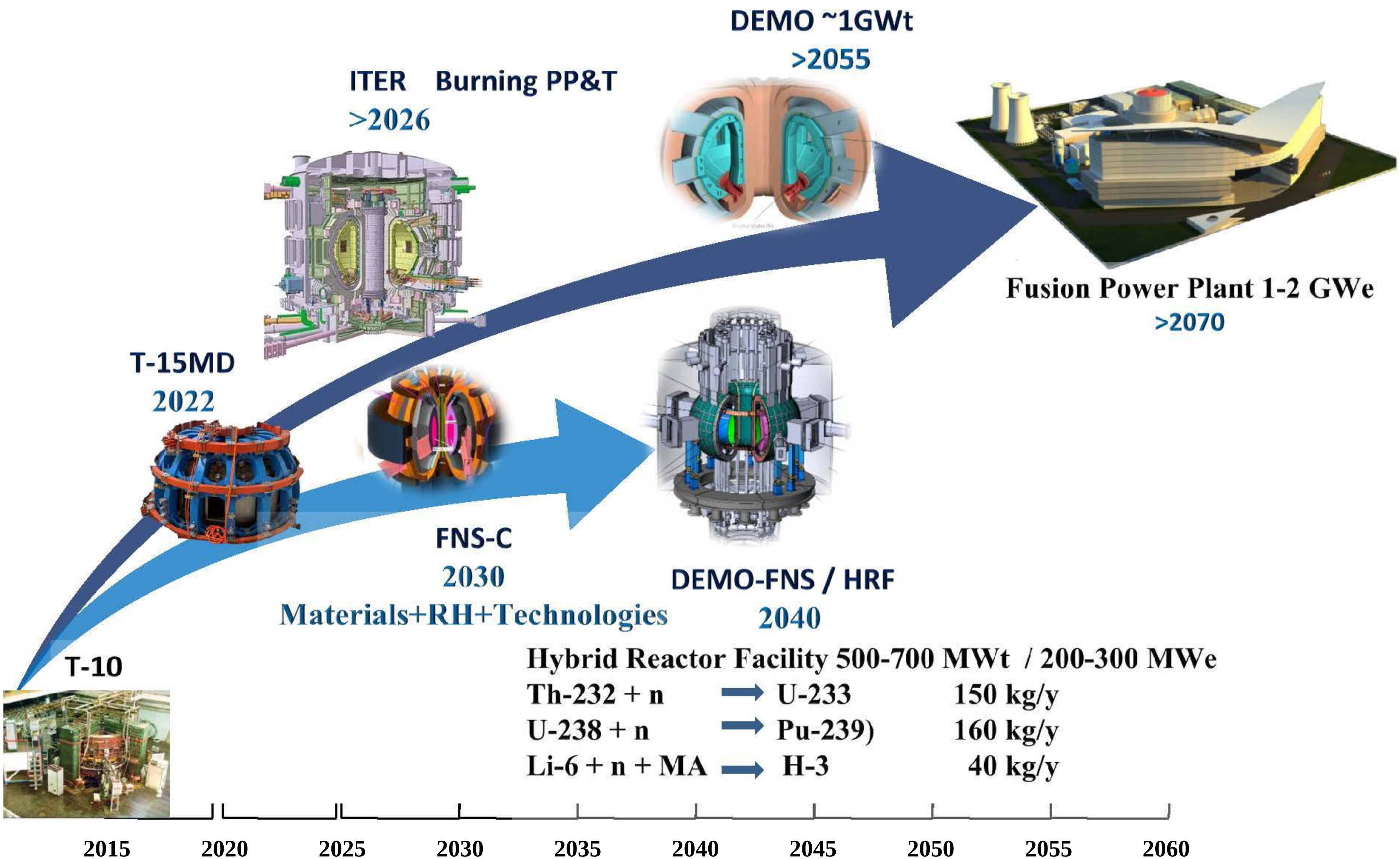
- Critical technologies requiring development include:
  - additional heating and current drive
  - tritium fuel cycle
  - first wall and divertor protection technologies using Li
  - cryogenic electromagnetic systems
  - blankets with fissile materials and tritium breeding
  - remote handling for all systems!

- Available disruption mitigation technique is appropriate for HRF size.

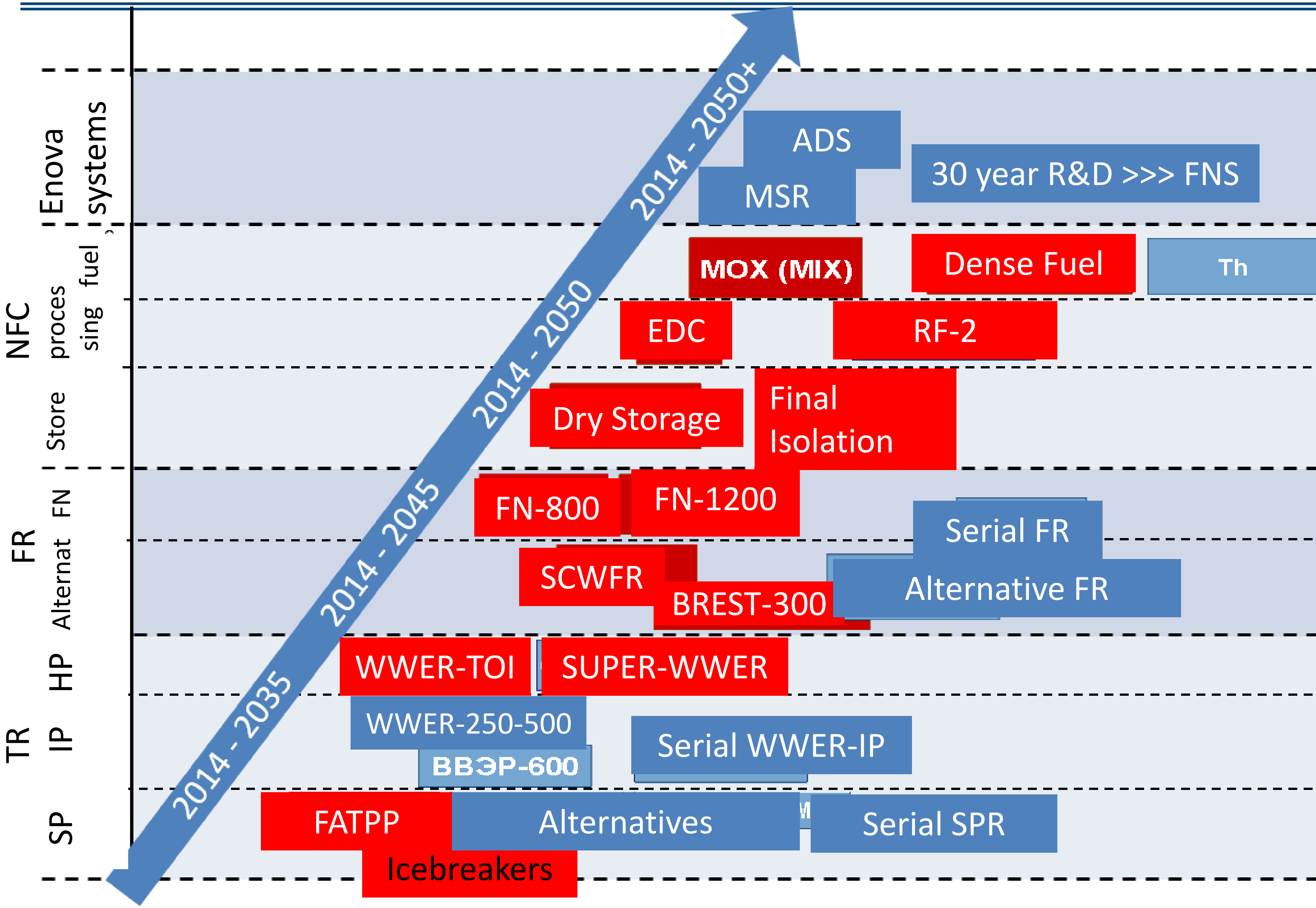
  Materials for 14 MeV neutron environment are to be invented

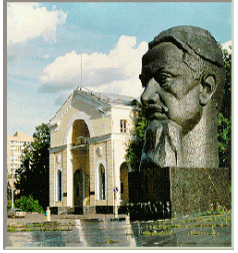
НАЦИОНАЛЬНЫЙ ИССЛЕДОВАТЕЛЬСКИЙ ЦЕНТР «КУРЧАТОВСКИЙ ИНСТИТУТ»

RF-ROAD MAP TO CONTROLLED FUSION

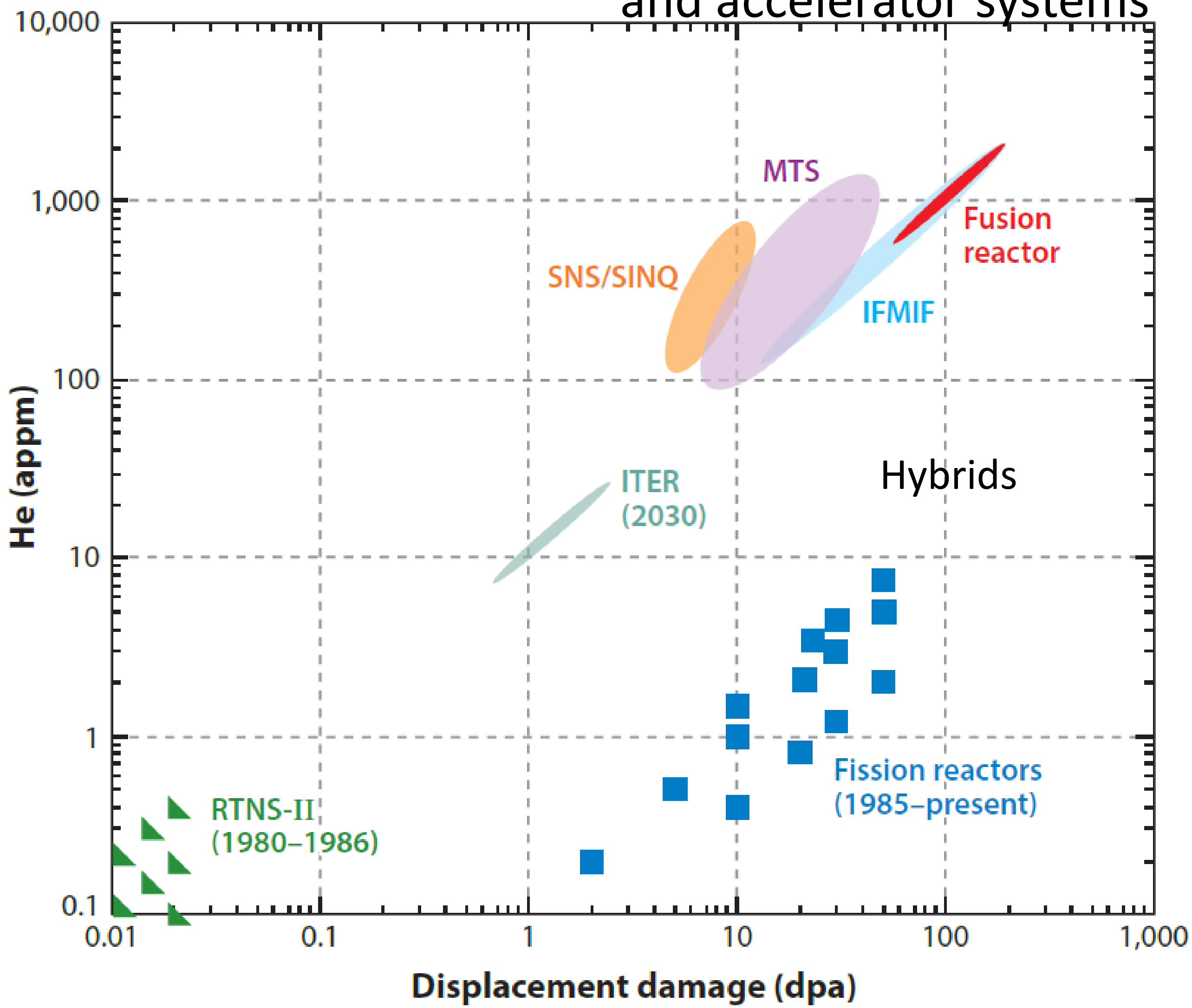


Roadmap of Nuclear Power in RF





Materials damage by fusion and fission neutrons due to He generation(appm) versus displacement (dpa) in fission, fission and accelerator systems



The spectrum of nuclear reactors is 100-300 times softer than in tokamaks

The damage in ITER is close to reactor damage. There are no dangerous problems for materials!

Thermonuclear neutron sources are necessary for the certification of materials

НАЦИОНАЛЬНЫЙ
ИССЛЕДОВАТЕЛЬСКИЙ ЦЕНТР
«КУРЧАТОВСКИЙ ИНСТИТУТ»

Conclusions

- Fusion technologies are actively developed in KI and SC Rosatom being based on tokamak approach within the RF federal project
- Hybrid Reactor Facility makes it possible to fundamentally increase the rate of involvement of uranium-238 and thorium-232 in atomic energy production
- Fusion-fission hybrid systems based on tokamak and molten-salt technologies open up new opportunities for the development of nuclear energy in the 21st century
- They are able to maintain high growth rates, expand the resource base of fuel nuclides due to U-238 and Th-232, reduce the generated radiotoxisity, accelerate the mastery of controlled thermonuclear fusion, but require an innovative approach to solving institutional problems of energy development

DEVELOPMENT OF BASIC THERMONUCLEAR TECHNOLOGIES OF THE FUSION-FISSION HYBRID FACILITY FOR TESTING MATERIALS AND COMPONENT

ID: 2202

Yu.S. Shpanskiy, B.V. Kuteev et al. NRC “Kurchatov Institute” Moscow, Russian Federation
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ABSTRACT

- The aim is to build fusion hybrid facility (FHF) containing a hybrid blanket based on a tokamak with a DT fusion power > 30 MW (corresponds to the generation of ~$1x10^{19}$ neutrons/s) and a fission power of up to 700 MW .
- To achieve this goal, it is planned to develop SSO technologies, as well as a compact fusion neutron source (FNS-C) with a DT fusion power of 3 MW for testing materials and components.
- Design activity was supported by R&D in Neutronics, Optimization of the device layout, subsystems including EMS, VV, divertor, ECR H&CD, NBI.

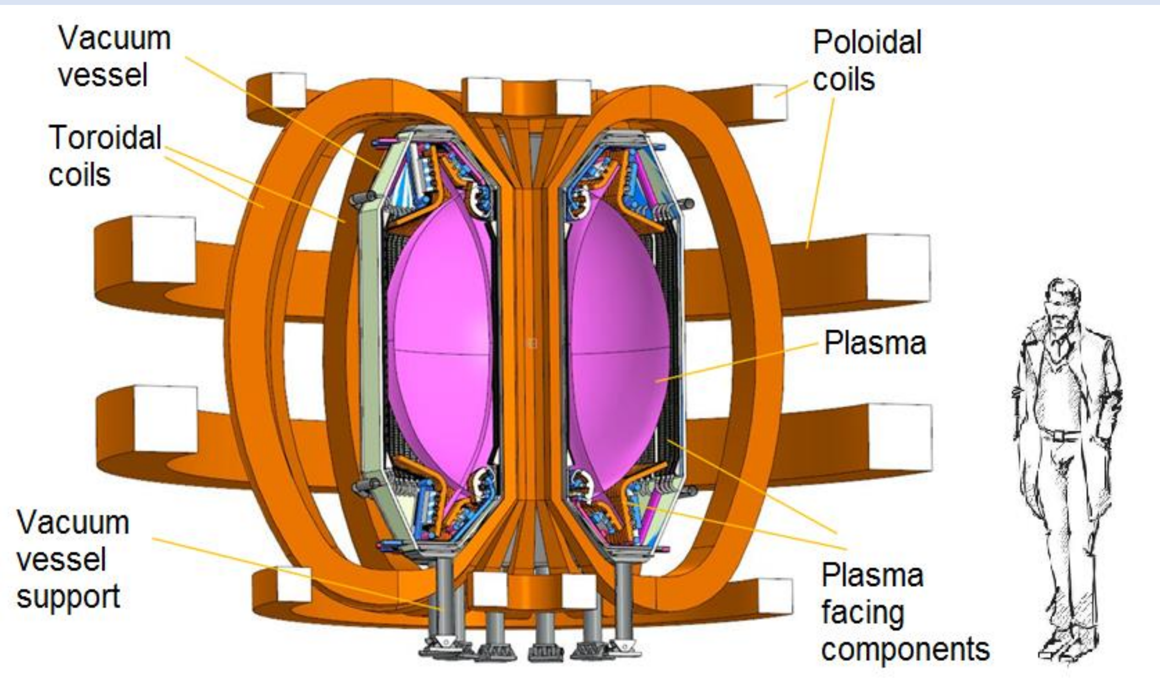
BACKGROUND

- *RF* is developing a closed nuclear fuel cycle that ensures the operation of thermal and fast power reactors. Combining nuclear fusion and fission reactions in a single design will allow achieving fundamentally new characteristics and parameters of the nuclear energy system.
- In Russia, the design of a hybrid fusion plant based on a superconducting tokamak is being developed.
- This activity is included in the current state research program of Russia in the field of nuclear energy.
- It is planned to develop SSO technologies (including lithium), build a complex of fuel and lithium cycle experimental facilities as well as a compact FNS-C ~$1x10^{18}$ neutrons/s).
- Analytical, theoretical and experimental work has been done to support this program.

DEVELOPMENT OF COMPACT NEUTRON SOURCE FNS-C

FNS-C Design

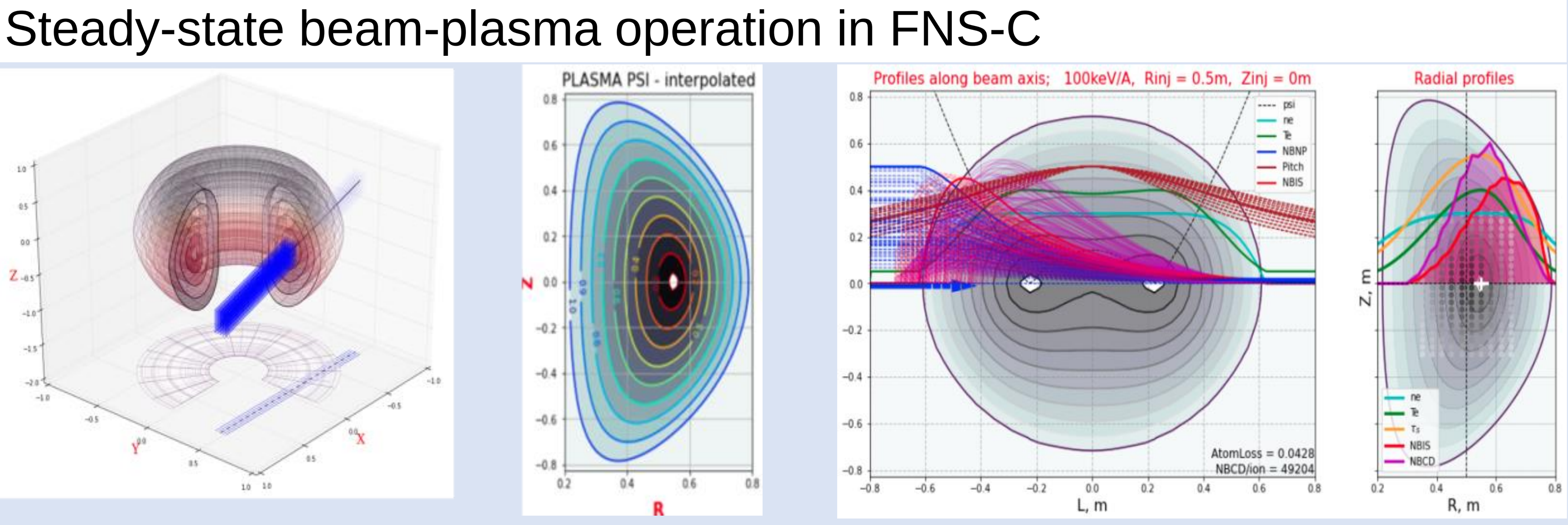
The current aim was to develop technical proposals for the design of the vacuum vessel, the first wall and the divertor target of the FNS-C.



- Thermonuclear power of is 3 MW when using 7.5 MW of additional plasma heating power;
- a compact (R ~ 0.53 m, a ~ 0.3 m, $V_{pl} \leq 2.5$ m^3) tokamak, which potentially allows to obtain a relatively inexpensive neutron source in the shortest possible time;
- the average neutron load on the walls is 0.2 MW/m^2.

Thermohydraulic analyses of the vacuum vessel (VV): it is recommended to use 192 cooling channels. The pressure drop in the cooling channels of the VV at the calculated flow rate of 15.8 kg/s will not exceed 0.1 atm.

Steady-state beam-plasma operation in FNS-C

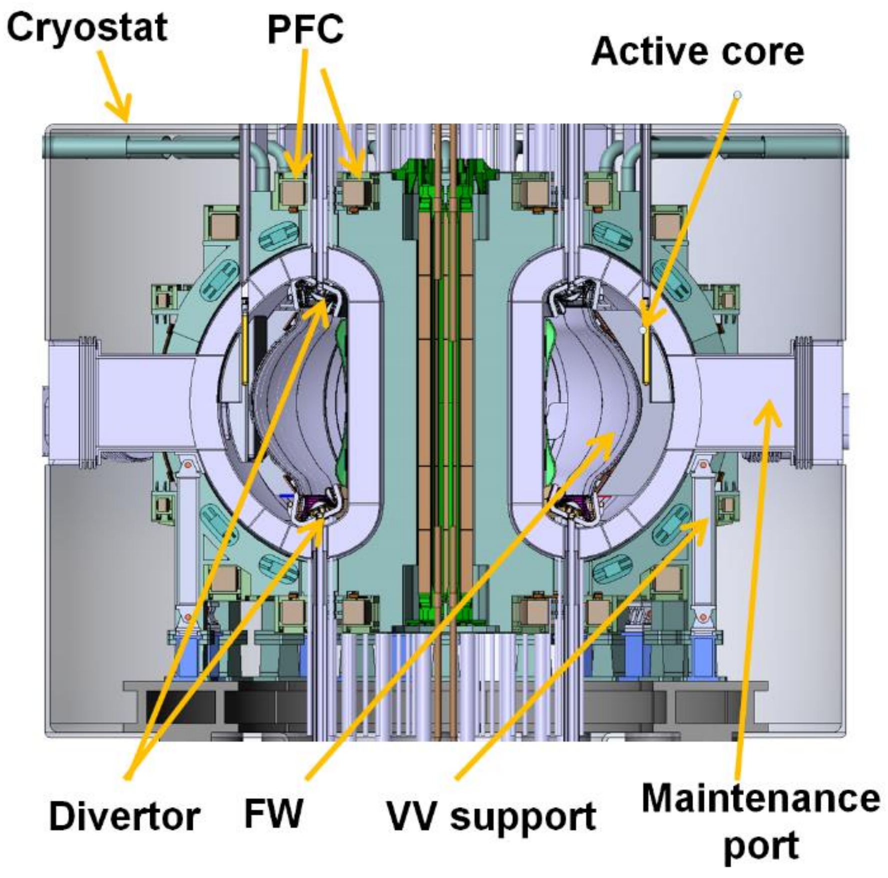


FNS-C magnetic surfaces configuration and injection scheme; Penetration and ionization profiles of a real (0.6 x 0.3) beam into the FNS-C plasma (left), corresponding radial profiles, as well as current generation in a poloidal cross-section (right).

Fusion Hybrid Facility DEVELOPMENT

FHF Design

The FHF design makes it possible to produce electricity, develop technologies for transmutation and removal of ^{233}U fuel nuclides from ^{232}Th



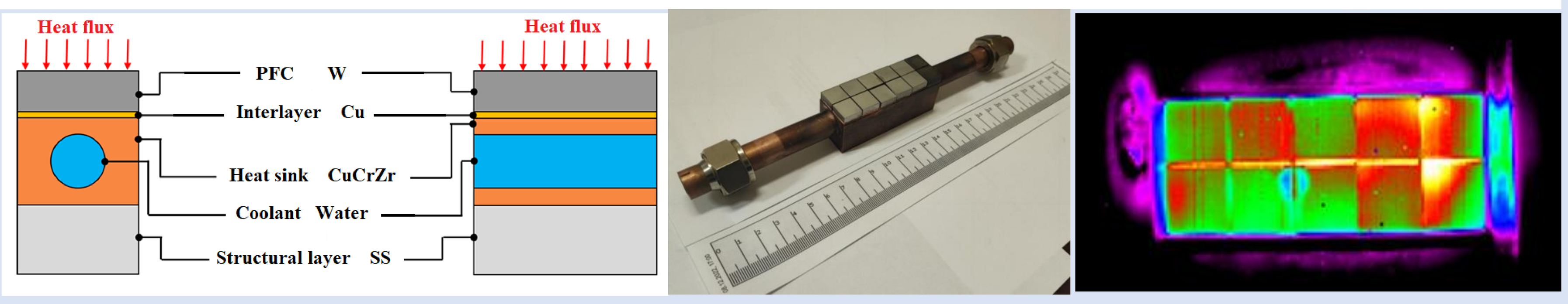
Major radius,	m	3.2
Minor radius,	m	1.0
Toroidal field,	T	5.0
Plasma current,	MA	5
NBI power,	MW	30
ECRH power,	MW	6
Electron/ion temperature,	keV	11.5/10.7
β_N		2.1
β_p		0.96
Neutron yield	n/s	>10^{19}
Consumed/generated Electric power,	MW	up to 200
Thermal power	MW	up to 700
Discharge time,	h	up to 5000
Duty factor		0.3
Life time, years		30

Electron-cyclotron heating and current drive

The calculated efficiency of current generation near the axis of the plasma discharge in the FHF tokamak is close to the values expected for ITER in a similar range of EC wave frequencies and plasma temperatures (n_{cd}=0.064x10^{20}A/(W·m^2)). With input power PEC=6 MW and Te(0)=10 keV, the calculated value of the EC current is I_{cd}=240 kA.

EXPERIMENTAL INVESTIGATION OF PFC PROTOTYPES

Prototypes of multilayer plasma-facing elements were developed (Fig.9) and tested, operating at a thermal load on a scale of 2.5 MW/m^2 (for the first wall) and 10 MW/m^2 (for the divertor target), compatible with lithium at temperatures up to 400 °C and neutron irradiation.



Scheme of multilayer pieces (a), Prototype that has passed thermal tests (b), Thermal imaging at 34.72 MW/m2 load, Tmax(W) =2525°C (c)

A total of 770 loading cycles were carried out with an increase in the absorbed power density from 2 to 35.56 MW/m^2. At a load value of 34 MW/m^2, a sudden increase in temperature appeared up to 2330 °C in local areas. In this case, there was no violation of the tightness of the cooling channel.

CONCLUSION

- Compact neutron source FNS-C with a DT fusion power of 3 MW (corresponding to generation of ~$1x10^{18}$ neutrons/s) for testing materials and components of hybrid systems was designed.
- Enabling systems design of FHF was upgraded including Vacuum vessel, Radiation shield, Divertor, Fueling cycle.
- Design activity was supported by R&D in Neutronics, Optimization of the device layout, subsystems including EMS, VV, divertor, ECR H&CD, NBI.
- Industrial FNS systems are capable of ensuring the equilibrium of the produced and transmuted minor actinides in the Russian nuclear power system, provided that the necessary capacities for spent nuclear fuel reprocessing and fuel fabrication are implemented.

ACKNOWLEDGEMENTS / REFERENCES

Main contributors:

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29-th IAEA Fusion Energy Conference FEC 2023, 16-21 October 2023, London, United Kingdom, ID: 1759.

Neutron Radiation Damage in the Materials of a Compact Hybrid Fusion Neutron Source with a Homogeneous Heavy Water Blanket

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INTRODUCTION

The compact fusion neutron source (FNS-C) with homogeneous heavy-water thorium blanket is designed at the National Research Center ‘Kurchatov Institute’ in Russia as a prototype of a safe fusion-fission reactor (FFHR). The main purpose of FNS-C is the development and debugging of FFHR engineering systems. A key unsolved problem of FNS-C and FFHR is the choice of materials with sufficient resistance to high neutron irradiation.

PURPOSE OF STUDY AND TASKS

The purpose of the research is to calculate radiation damage of the reactor materials in the spectrum of the hybrid fusion neutron source FNS-C to determine the possibility of using these materials in FNS-C.

The following tasks were set:

- to calculate the neutron loading on the first wall in the FNS-C model;
- to calculate energy spectra in FNS-C materials;
- to estimate the doses of radiation damage and accumulation of gas products of transmutantions in the materials of the model;
- to compare the results with obtained in the spectrum of a hybrid reactor machine DEMO-FNS.

METHOD

The Monte Carlo MCNP-4 code was used with ENDF/B-VII nuclear data and the SPECTER code (NRT model).

CALCULATION MODEL

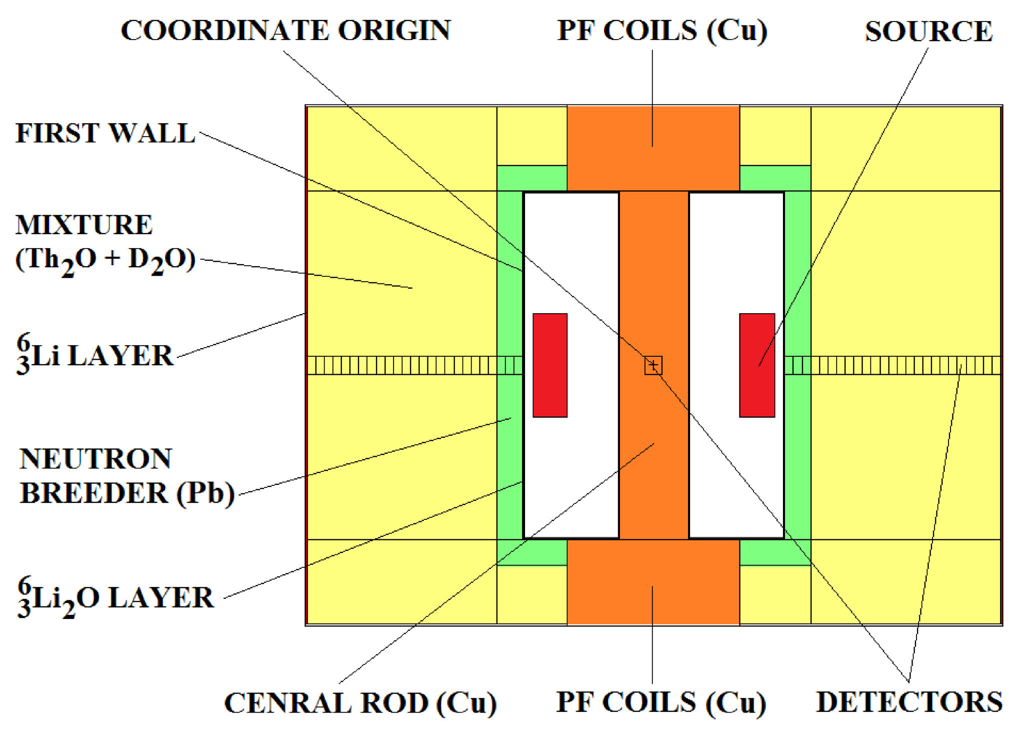

FIG. 1. View of the equatorial horizontal cross-section of the FNS-C model

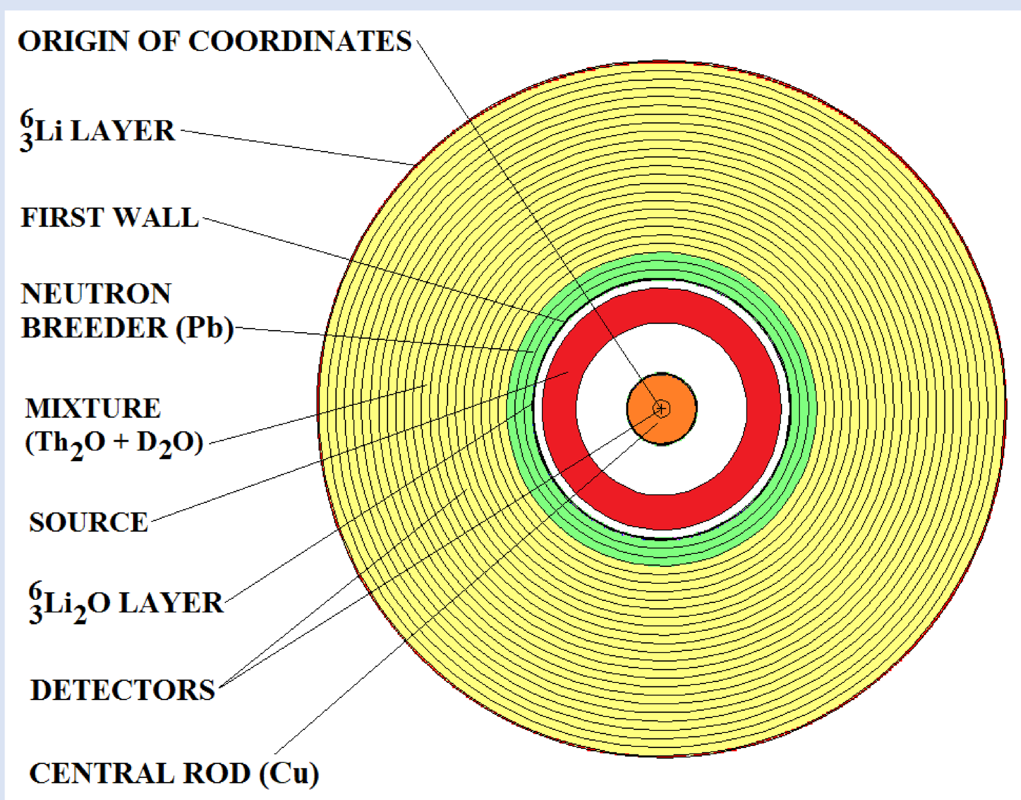

FIG. 2. View of the central vertical cross-section of the FNS-C model

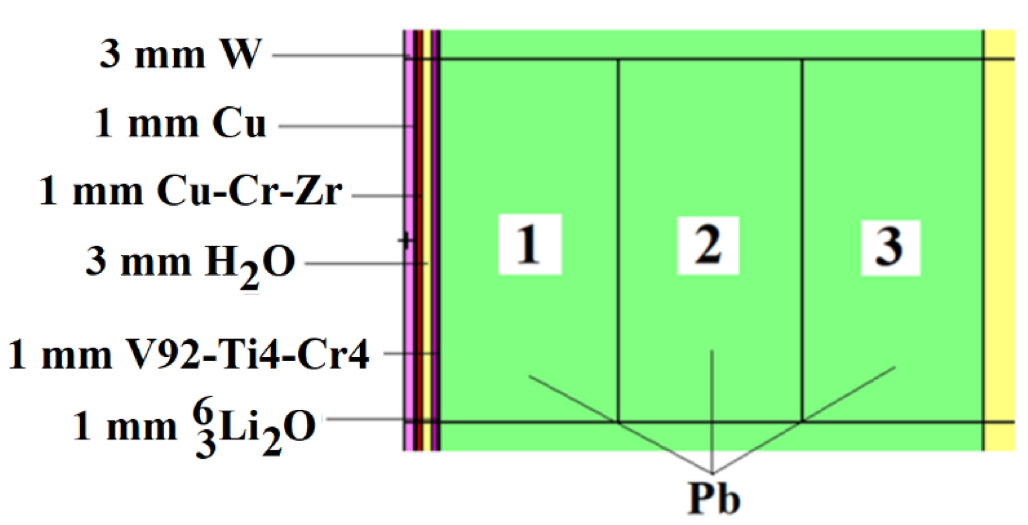

FIG. 3. View of the first wall of FNS-C; 1, 2, 3 are detectors in the lead neutron breeder

OVERAL DIMENSIONS: R = 201.8 cm, h = 301.6 cm;
FUSION NEUTRON SOURSE: 14.1 MEV D-T, isotropic ; R=70 cm r=50 cm, h=60 cm;
EMISSION RATE: 5 MВт ($1,775\times 10^{18}$ n/s);
CENTRAL RODE: Cu, R=20 cm, h=201.6 cm;
VACUUM VESSEL: R=75 cm, r=20.8 см, h=200 cm;
PF COILS: Cu, R=50 cm, h=50 cm;
FIRST WALL: 3 mm W, 1 mm Cu, 1 mm Cu-Cr-Zr; 3 mm H_2O; 1 mm, V92-Ti4-Cr4;
$^6Li_2O$ SHELL AROUND THE FIRST WALL: thickness of 1 mm;
^{6}LI SHELL ALONG THE OUTER CONTOUR OF *(b)* **FNS:** thickness of 1 cm;
SOLID-STATE BLANKET: Pb;
AQUEOUS BLANKET: suspension of ThO_2 in D_2O, enrichment with ^{233}U is possible;
26 DETECTORS: R-r=5 cm, h=10 cm;
DETECTOR IN CENTER: R=5 cm, h=10 cm.
Natural W and Pb were used, as well as W enriched with isotope ^{184}W and Pb enriched with isotope ^{208}Pb.

RESULTS

TABLE 1. The neutron loading on the first wall of FNS-C

Material	Natural			Isotope-enriched		
Amount of ^{233}U	0%	1.37%	1.52%	0	1.37%	1.5%
J^a from blanket (MW/m²)	4.923×10^{-2}	5.433×10^{-2}	8.103×10^{-2}	5.562×10^{-2}	6.400×10^{-2}	$9{,}911\times10^{-2}$
J total (MW/m²)	4.199×10^{-1}	4.282×10^{-1}	4.471×10^{-1}	4.304×10^{-1}	4.447×10^{-1}	$5{,}046\times10^{-1}$
δ^b, %	0.05	0.1	1	0.1	0.3	0.9

[a]Neutron current density;
[b]Statistical uncertainty of calculation results.

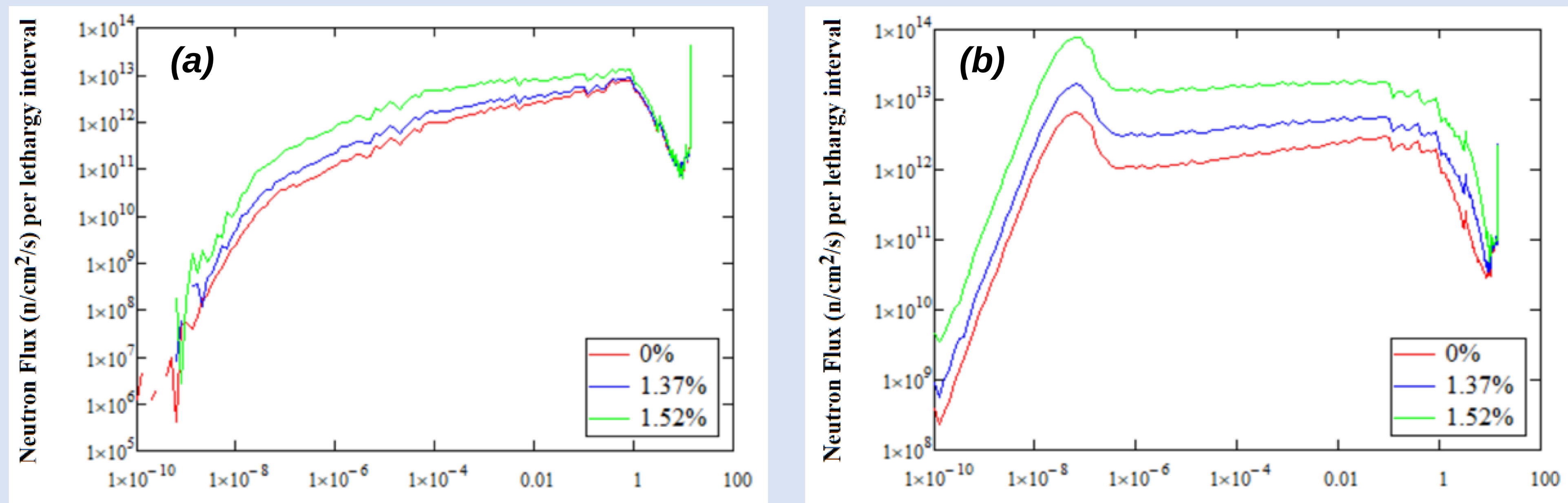

FIG. 4. The FNS-C neutron spectra in the first wall (a) and $^{232}ThO_2$+D_2O (b) with different amount of ^{233}U in the aqueous blanket for natural materials at the fusion source power of 1.775×10^{18} n s^{-1}

TABLE 2. Nuclear heating density from neutrons and secondary photons $Q(n + \gamma)$ in the units of FNS-C with natural materials at full power of 1.775×10^{18} n s^{-1}

Amount of ^{233}U	0%		1.37%		1.52%	
Unit	Q(n+γ) (MW/m³)	δ^a (%)	Q(n+γ) (MW/m³)	δ (%)	Q(n+γ) (MW/m³)	δ (%)
Average value[b] for FW[c]	6.138	0.2	6.896	0.5	1.080×10^{1}	1.4
1 mm Li_2O	6.305×10^{1}	0.3	1.217×10^{2}	0.6	4.449×10^{2}	2.3
Pb blanket[b]	1.434	0.2	2.251	0.5	6.735	2.1
$^{232}ThO_2$+D_2O[c]	1.1	0.2	32.01	1.0	200.4	2.0

[a]Statistical uncertainty of calculation results;
[b]Average value for sum of detector volumes of FW;
[c]First wall;
[d]Maximum value in aqueous blanket.

TABLE 3. The neutron fluence in the materials of the first wall and the radiation displacement dose for one year (365 d) of operation of fns at full power (fpy) of 1.775×10^{18} n s^{-1}

Material	Fluence (n cm^{-2} fpy^{-1})					
	E = 14.1 MeV	δ^a, %	E > 6.7 MeV	δ, %	Total	δ, %
First wall	$1.487 10^{21}$	0.2	1.645×10^{21}	0.2	8.552×10^{21}	0.1
Pb blanket	4.397×10^{20}	0.2	5.394×10^{20}	0.2	8.532×10^{21}	0.1
Li_2O	1.317×10^{21}	0.2	1.528×10^{21}	0.2	6.449×10^{21}	0.1
$^{232}ThO_2$+D_2O[b]	5.470×10^{20}	0.2	1.644×10^{21}	0.2	2.477×10^{21}	0.1
Material	**Dose (dpa fpy^{-1})**					
	E = 14.1 MeV	δ^a, %	E > 6.7 MeV	δ, %	Total	δ, %
Pb бланкет	1.488	0.2	1.778	0.2	4.721	0.1
First wall W	1.386	0.2	1.522	0.2	2.293	0.1
First wall Cu	6.074	0.2	6.679	0.2	1.034×10^{1}	0.1
First wall Cr	4.323	0.2	4.761	0.2	7.945	0.1
First wall Zr	3.909	0.2	4.316	0.2	7.443	0.1
First wall V	4.103	0.2	4.522	0.2	8.124	0.1
First wall Ti	3.803	0.2	4.187	0.2	7.331	0.1
^{6}Li (Li_2O)	6.093×10^{-1}	0.2	7.214×10^{-1}	0.2	1.323×10^{1}	0.1
$^{232}ThO_2$+D_2O[b]	1.731	0.2	5.204	0.8	7.839	0.1

[a]Statistical uncertainty of calculation results;
[b]Maximum value in aqueous blanket.

CONCLUSION

- The doses of radiation displacements and accumulation of gas products of transmutations of the FNS-C are similar to the results obtained for the fusion neutron source DEMO-FNS.
- The contribution of plasma neutrons to the total neutron loading on the first wall is ~ 80–85% for all calculation variants.
- The neutron energy spectra in FNS-C materials are similar to each other in shape, but differ in the total flux. So conversion coefficients for calculating the values of radiation damage are used.
- The heating density from neutrons and secondary photons in the Li_2O layer is large (~ 500 MW/m³) and reaches the thermal power density in the nodes of fast neutron reactors. The heating density in the $^{232}ThO_2$+D_2O aqueous blanket is ~ 32–38 MW/m³ at an enrichment of ~ 1.37% ^{233}U and is half as the power density of thermal neutron reactors of the VVER type.
- The maximum dose of radiation displacements (~ 10–13 dpa/fpy) and the accumulation of gas products of transmutations is obtained for Cu and the Li_2O layer for the tritium production. This requires frequent replacement of the first wall and the Li_2O shell.

Neutral Beam Efficiency in Thermonuclear Neutron Source Tokamak

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29th IAEA Fusion Energy Conference (FEC 2023)

TH-H 2011

Lightweight neutral beam for steady-state tokamak operation

ABSTRACT

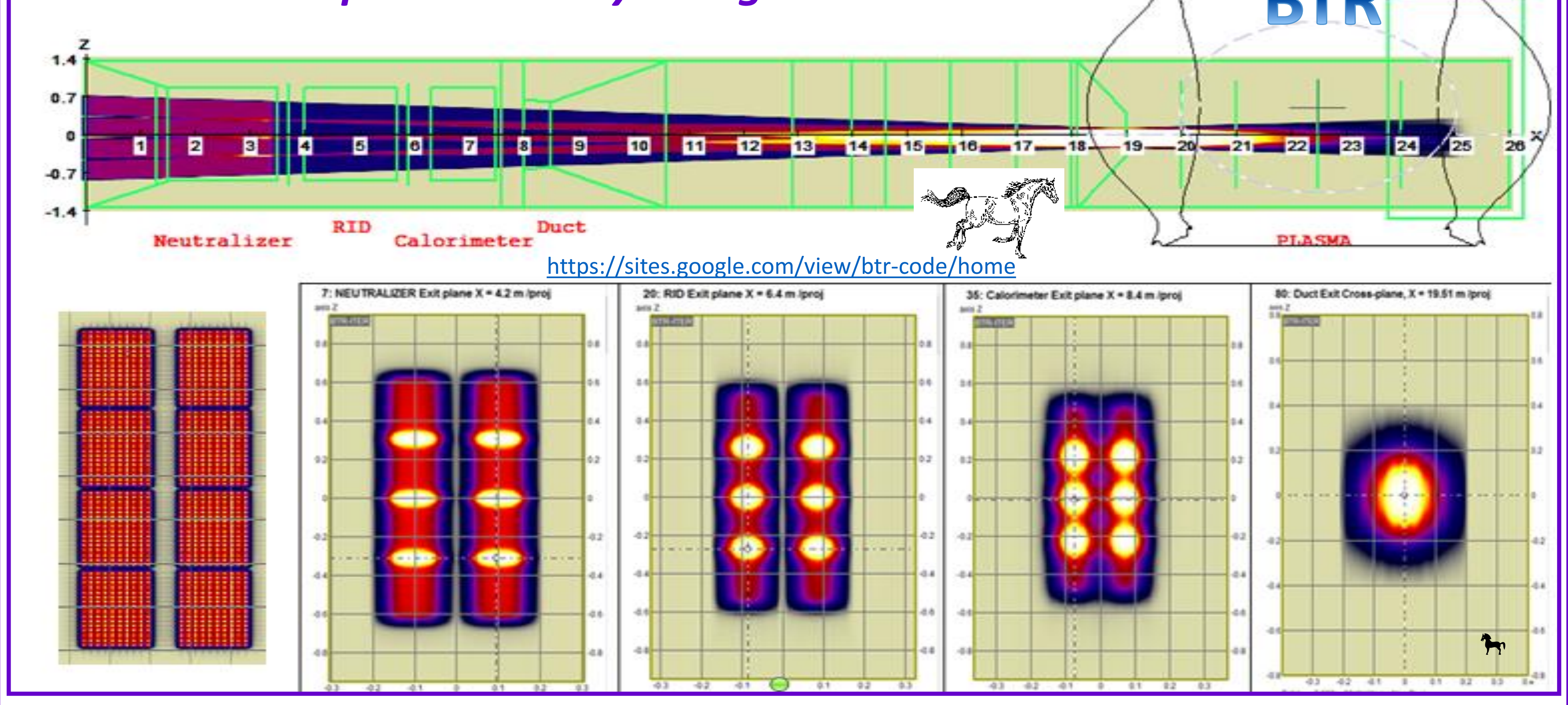
- NBI in FNS: non-inductive CD, steady-state operation, main source of neutrons (beam-thermal reactions), "beam-driven" tokamak operation.
- NBI combined efficiency: beamline performance + NB gain in plasma.
- "Light neutral beam" (LNB): determinism, high level of details, accurate geometry representation – both beam and plasma.
- Combined approach is not only for beam-plasma optimization in two-energy-component tokamak – it is efficient for boosting NBI output in fusion devices with any NBI contribution, including low beam impact (single-energy-component) operation.

BACKGROUND and CHALLENGES

- FNS, tokamak based – for fission, for fusion, for hybride (FF) reactors.
- Testing and validation of materials and struct. components for future.
- FNS experimental results: in nuclear waste treatment, nuclear fuel production, hybrid reactors technologies.
- Two-energy-component tokamak operation (beam + thermal plasma).
  - High neutron generation (10^{18}-10^{19} n/s) at moderate cost and size.
  - Beam controlled steady-state scenario and fusion power density.
  - Low plasma temperature vs pure thermonuclear fusion operation.
  - Low power gain, high neutron yield, rotation and non-inductive NBCD.
  - High sensitivity on NBI energy, spatial geometry and NB aiming.
  - Flexible control of fast ions distributions in phase space by tuning the NBI parameters.

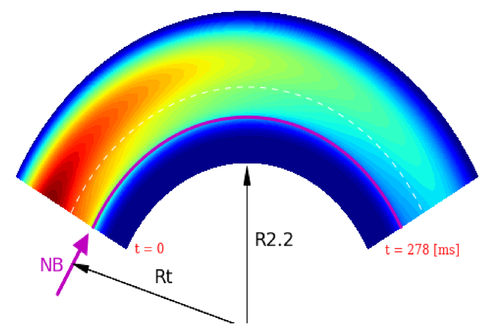
METHODS and IMPLEMENTATION BTR / BTOR codes

Neutral beam power density along the beamline

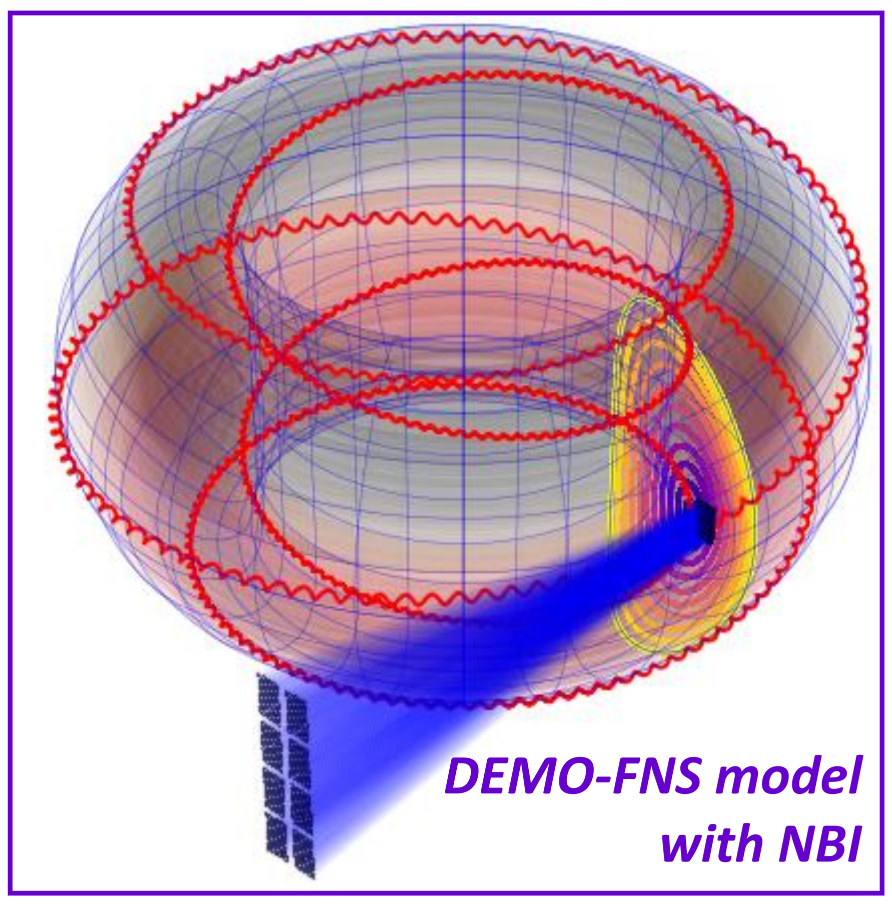


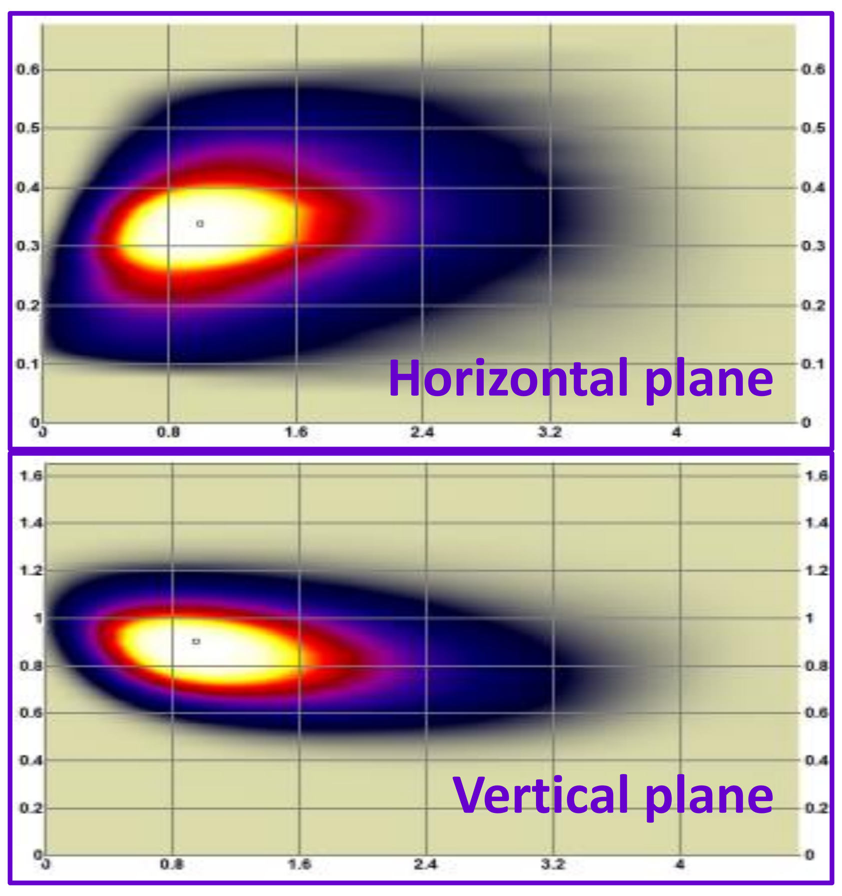
Fast ion thermalisation (slow-down) time

$$\tau_s = \frac{\tau_{se}}{3} \cdot ln\left[1 + \left(\frac{E_b}{E_c}\right)^{3/2}\right]$$



Beam ionization intensity



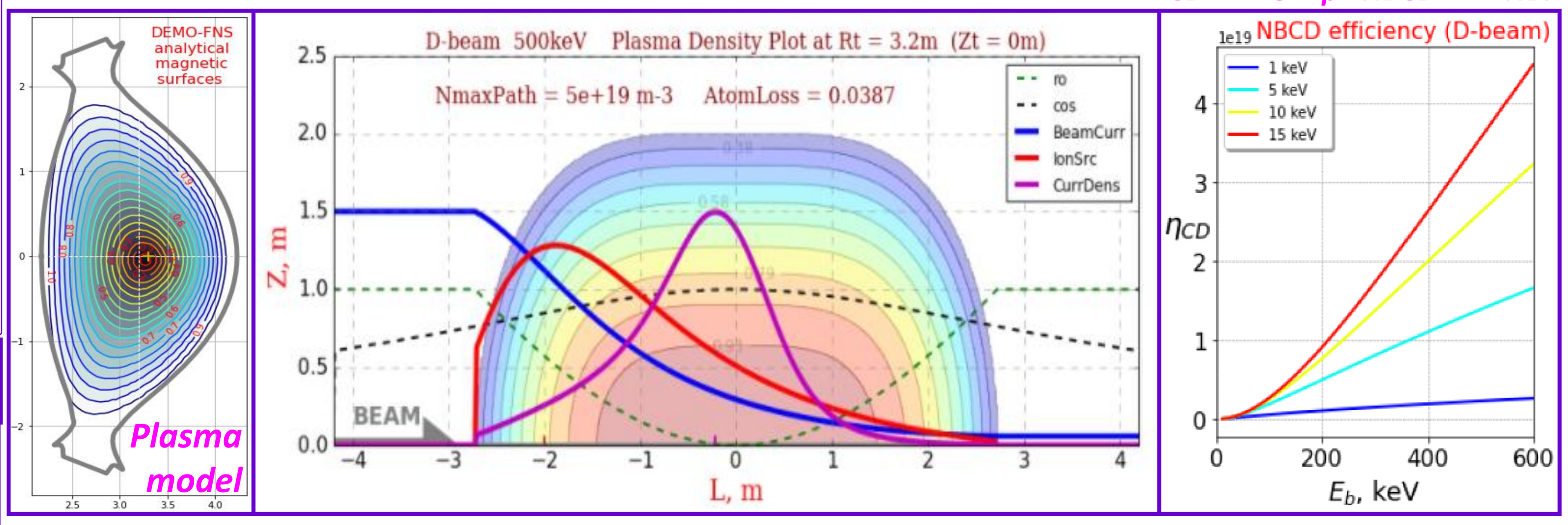


OUTCOME

PENETRATON and SHINE-THROUGH

- Beam capture falls, direct non-ionized losses grow up with E_{NB}.
- Lower E_{NB} fractions are deposited outside hot plasma core region.

Neutral beam penetration and NBCD (on-axis aiming) $\eta_{CD} = n_e R_p I_{NBCD} / P_{NBI}$

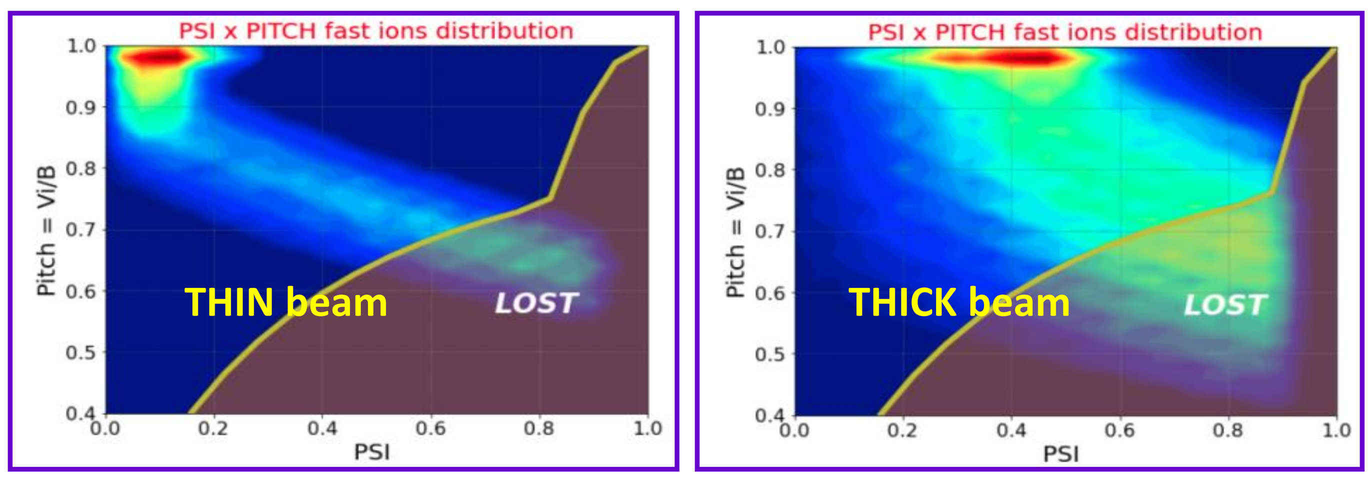


FAST ION ORBIT LOSSES

- Include losses on 1st larmor and banana traps.
- Defined by E_{NB} and pitch-angle (velocity component parallel to MF).
- Highly sensitive to beam and plasma shapes (most critical for ST).

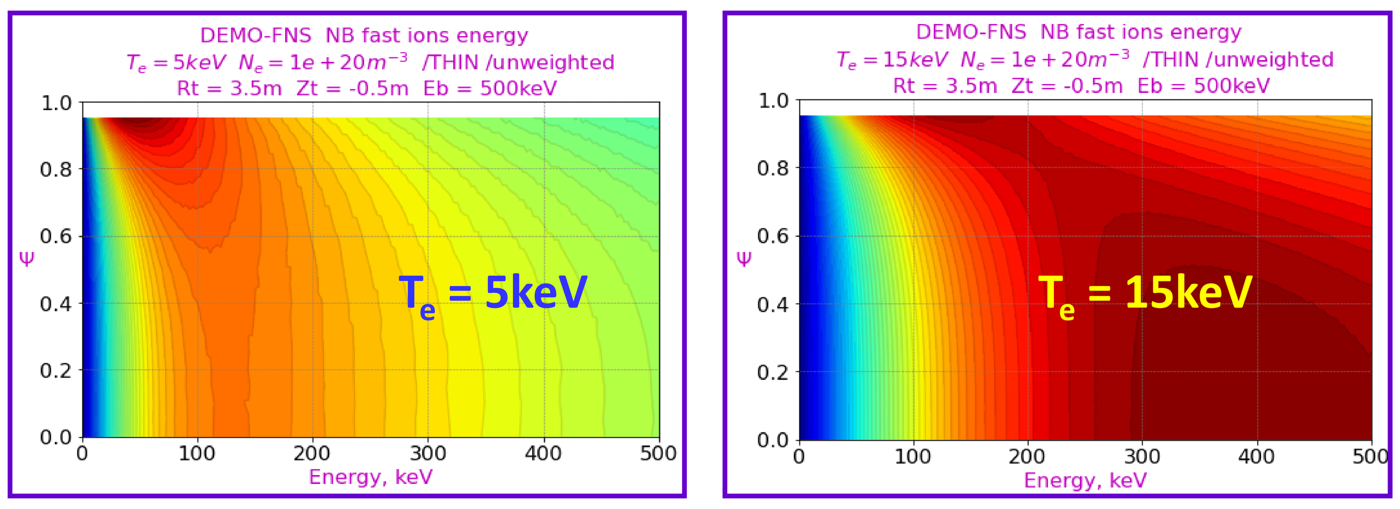
Neutral beam size effect

NB fast ion pitch and losses (from 1st larmor and banana)



FAST ION ENERGY DISTRIBUTION FUNCTION (EDF)

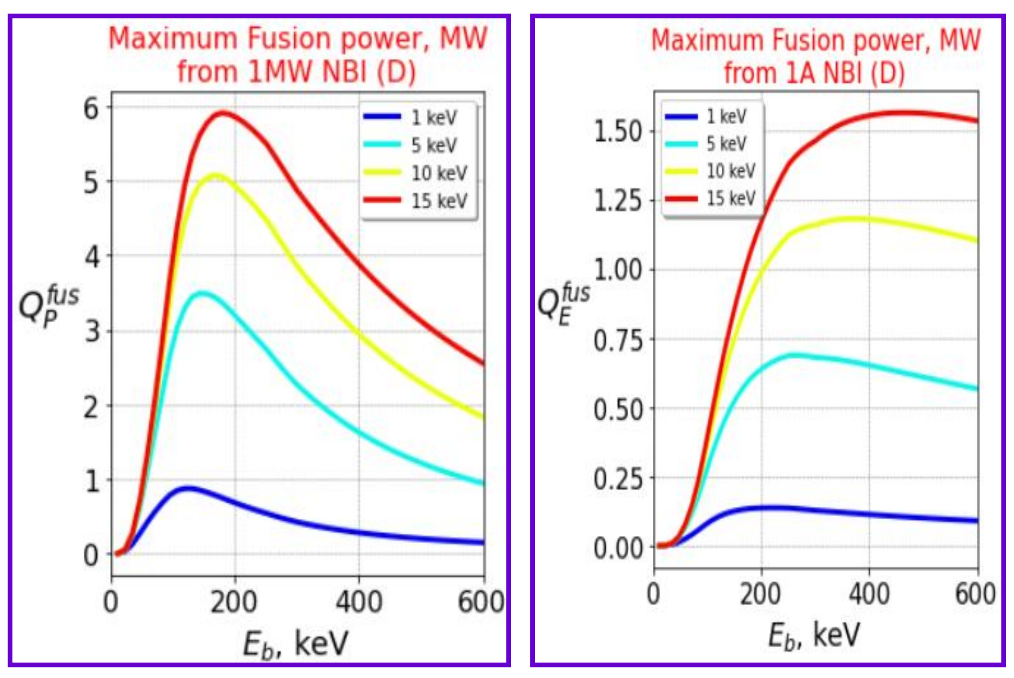
- Depends on $\varepsilon = E_{NB} / T_e$ across radial plasma layers.
- Dominated by hot ion tails (ε < ~15) or by "colder" ions (ε > ~20).
- FI burn-up rates have optimum E range → optimum E_{NB} for given T_e.



NB fast ion EDF across radial coordinate (Ψ) E_{NB} = 500 keV

NBI EFFICIENCY in TOKAMAK

- Best possible confinement regimes are available (Zakharov 2019).
- ∑eff ≈ NB production × NB absorption × NB gain (NBCD + NB fusion).
- NB-thermal fusion gain = $f(E_{NB}, T_e)$.



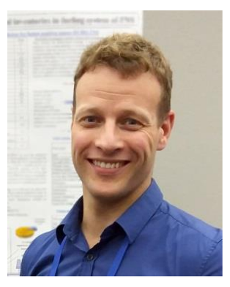
Fusion power gain for optimum NB deposition: left - power gain from P_{NB} = 1MW; right - E_{NB} multiplication = power gain from I_{NB} = 1A

CONCLUSION CHALLENGE

DEMO-FNS design needs to be revised to optimize beam-plasma operation and to intensify beam capture, NBCD and NB fusion

REFERENCES

- John Wesson, TOKAMAKS, 2011,§5.4
- Jassby, D.L., Optimization of fusion power density in the two-energy-component tokamak reactor. Nucl. Fusion 1975, 15, 453



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FEC2023 16-21 October 2023

KURCHATOV INSTITUTE
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Progress in modeling D/T component flows in tokamak-based fusion neutron source fueling system

INTRODUCTION

Concept of DT - fuel cycle (FC) is developed in RF for tokamak based fusion neutron sources (for projects with fusion power of 3-40 MW). Analysis of the FC operation is performed on the basis of advanced SOLPS, ASTRA, and FC-FNS codes combination taking into accounting for interaction of gas flows with plasma. Effects of specific gas mixture flows on plasma optimization and minimizing the tritium inventory at the site are treated in details.

These results will be used for further optimizing the FC of the FNS-ST and DEMO-FNS projects of neutron source and hybrid reactor facility HRF. They are being considered as a part of the comprehensive program of the State Corporation Rosatom "Development of engineering, technology and scientific research in the field of using atomic energy in the Russian Federation for the time period up to 2030" and further up to 2040.

BACKGROUND and CHALLENGES

Fuel cycle systems of a **fusion neutron source (FNS)**, a fusion reactor (CFR), or fusion-fission hybrid systems (FHR or FFHS) based on a tokamak should supply DT fuel to maintain stationary fusion in the plasma and remove excess particles from the divertor regions. Fuel cycle systems (FC) must provide gas evacuation from the vacuum vessel, preliminary purification of and hydrogen extraction from impurity gases, additional purification of chemically bounded hydrogen isotopes, and hydrogen isotope separation to the required concentration level.

These systems must contain the hydrogen isotope inventory and provide the conditions for fusion burning, plasma heating, and current drive (by fast atom beam injection), as well as residual gas detritiation and waste processing.

Particle flows in FC are determined primarily by the requirements for the parameters of the central plasma, as well as by the flows in the pumping system coming from the vacuum chamber. Thus, the modelling of the FC systems and the calculation of the particle flow should be related to the model of the core and divertor plasma [*M. ABDOU et al, Nucl. Fusion 2021, 61, 013001*].

METHODS and MODEL

Earlier, an approach was developed and discussed to simulations of tritium FC systems and calculation of the amount of tritium at the facility, consistently with the core and divertor plasma. In this approach, similarly to [*H.D. PACHER et al, J. Nucl. Mater. 2015, 463, 591*], the state of the core (inside the separatrix) and edge (outside the separatrix) plasma is simulated by a ASTRA and SOLPS4.3 codes combination **and the particle flows in the FC systems are simulated by the FC-FNS code.**

SOL simulation (SOLPS) → **Core plasma simulation (ASTRA+NUBEAM)** → **Fuel flows (in FC) simulation (FC-FNS code)**

Tritium inventory calculation ← **FC design optimization** ← **FC technology selection**

The integration between these three independent codes is carried out by an indirect scheme, where the output data from one code is parameterized and further used as input data in the other code [*A.Y. DNESTROVSKIY, Nucl. Fusion 2019, 59, 096053*]. This allowed us to create an efficient calculation workflow that describes the interaction of various components of the model with very different calculation times.

A model assumes that the core plasma is replenished by particles from three <u>different sources</u> (neutral beam injection, pellet injection, and gas supply to edge/SOL) that have <u>different confinement times</u>.

The flows of particles D and T, which are necessary to maintain a given plasma core density and isotopic composition (D/T), can be described in zero-dimensional form by the expression

$$N_{core} = N_{sep} + S_{NB}\tau_{NB} + \left(S_{pel(LFS)} + S_{pel(HFS)} - \alpha_{ELM} P_{SOL} / 3 \cdot T_{ped}\right)\tau_{pel} + S_{sep}\tau_{sep} - S_{fus}\tau_{tot}$$

here, $N_{core,sep} = n_{core,sep}V_{core}$ is the total hydrogen isotopes particles in the core plasma or at the separatrix, **obtained using the joint SOLPS+ASTRA calculations**, and V_{core} is the core plasma volume inside the separatrix. S_{NB}, S_{pel}, S_{sep}, and S_{fus} are the intensities of the sources (and sinks) of the D and T ions related to injection of neutrals, injection of pellets, gas inflow from the edge, and fusion reactions, respectively, and **τ_{NB}, τ_{pel}, τ_{sep}, and τ_{tot} are the corresponding diffusion confinement times of these ions, which are not equal to each other due to the different particle penetration depths.**

They are calculated with the ASTRA code and are then used in the zero-dimensional FC-FNS calculations. Expression is true if we neglect the change in profiles during intensive injection of pellets and assume that the pedestal parameters do not change.

The further development is the implementation of ions, instead of electrons, in particle transport equations in ASTRA. To model the plasma density, in this work, ion transport equations are used, and the continuity equations are written separately for each of the isotopes of the main gas and each of the plasma fueling systems. Thus, a set of six continuity equations is formed like

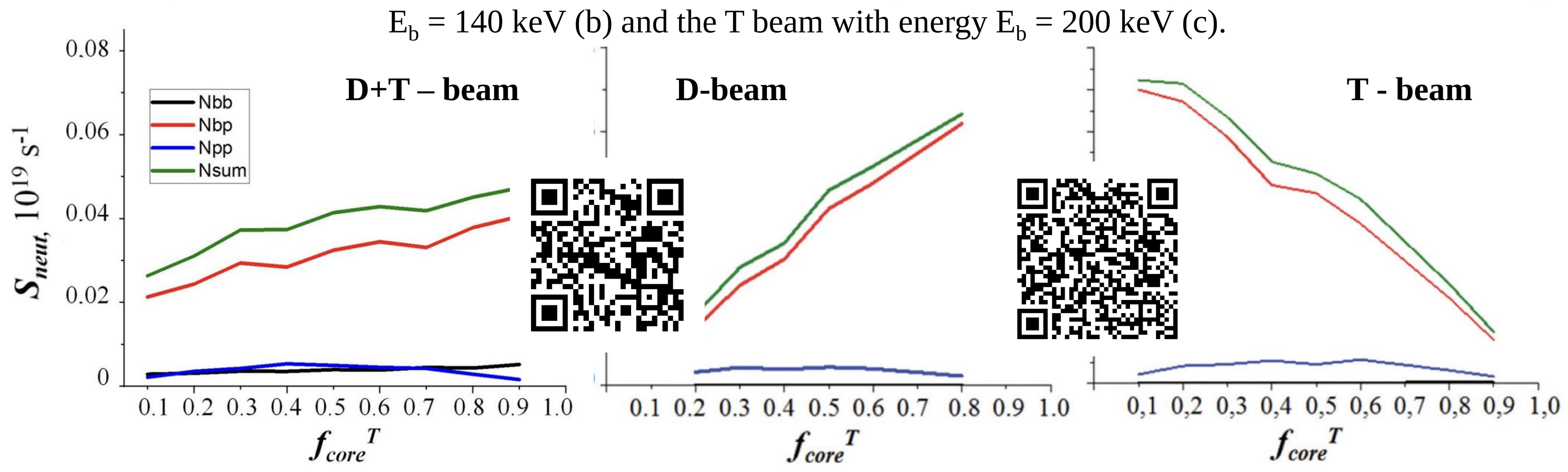
$$D^+ : \begin{cases} n_{D^+}^{fast} : \text{ FP-solver by NUBEAM} \\ \dfrac{\partial}{\partial t} n_{D^+} + \mathrm{div}\left(-D\nabla n_{D^+} + V n_{D^+}\right) = S_{D^+}^{beam} + S_{D^+}^{pel} + S_{D^+}^{sep} - R_{D^+} \end{cases}$$

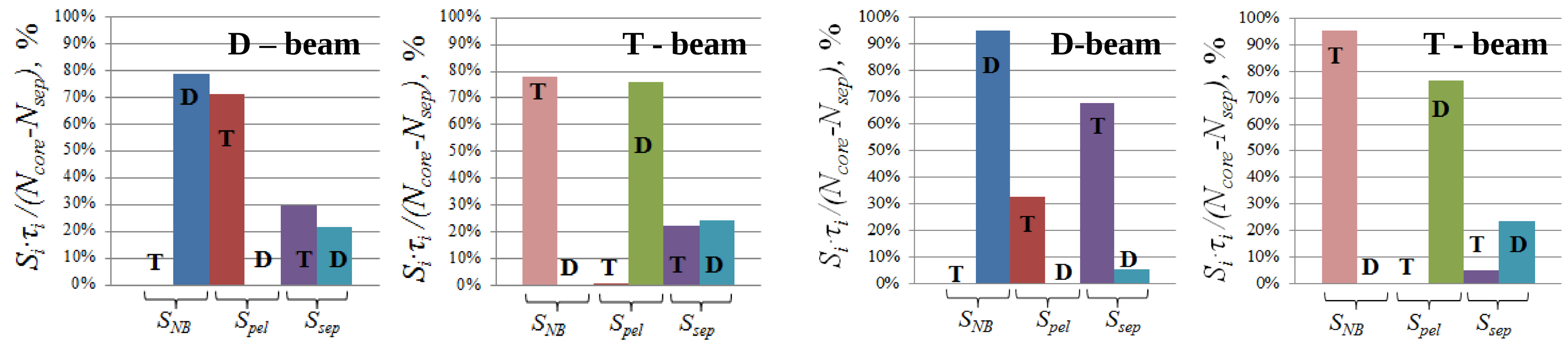
$$T^+ : \begin{cases} n_{T^+}^{fast} : \text{ FP-solver by NUBEAM} \\ \dfrac{\partial}{\partial t} n_{T^+} + \mathrm{div}\left(-D\nabla n_{T^+} + V n_{T^+}\right) = S_{T^+}^{beam} + S_{T^+}^{pel} + S_{T^+}^{sep} - R_{T^+} \end{cases}$$

If you add up the three equations for D or T, you get an equation for the total thermal density D or T. This means that this equation is satisfied.

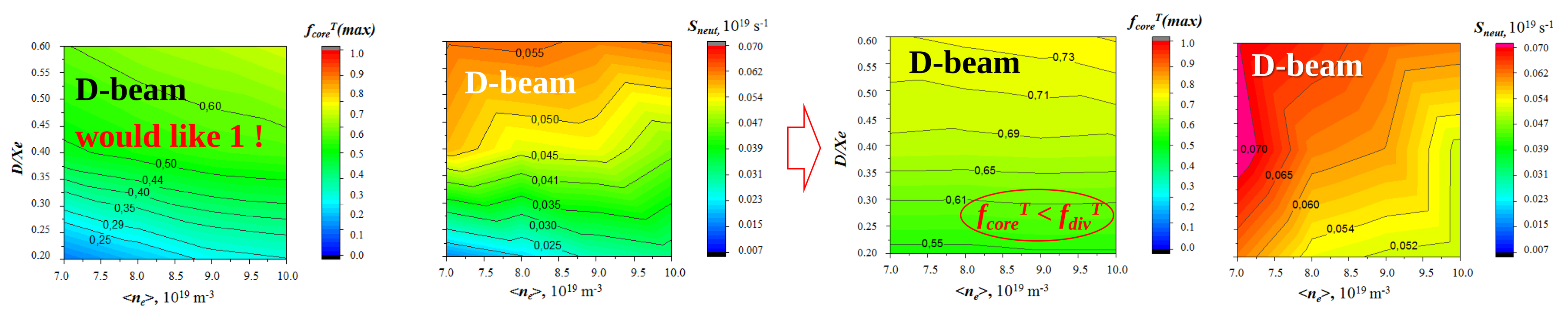
FNS BASED ON SPHERICAL TOKAMAK - FNS-ST PROJECT

Neutron yield as a function of the fraction of tritium in the core plasma for the D+T (a) beam, D-beam with energy E_b = 140 keV (b) and the T beam with energy E_b = 200 keV (c).



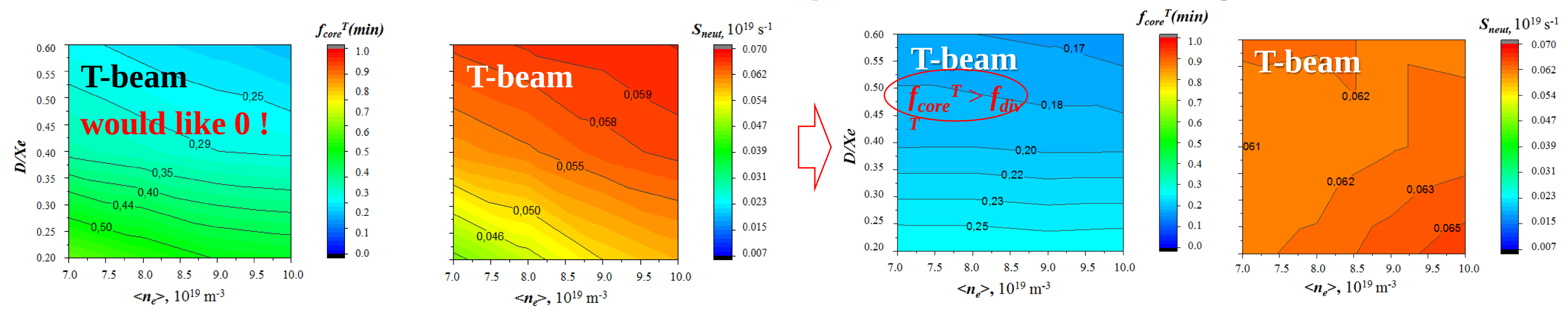


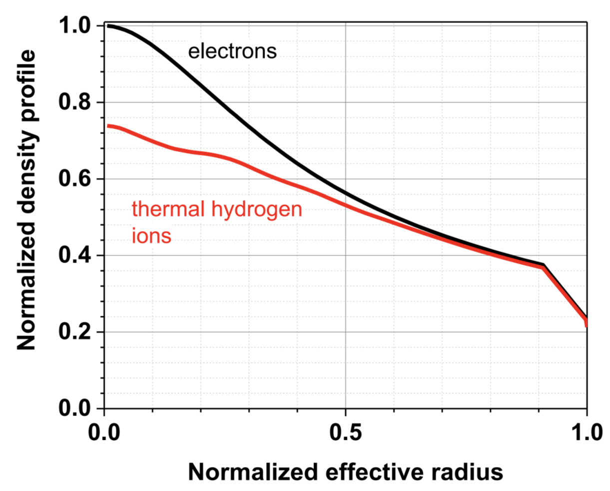
Contribution to the deuterium and tritium densities in the core plasma of the NBI, pellet injection, and neutral flux from the divertor. **For current FC architecture (left) and new (with full exhaust separation) - (right)**



for T-beam: min value of f_{core}^T will be reached at $S_{pel}^T = 0$
for D-beam: max value of f_{core}^T will be reached at $S_{pel}^D = 0$

for T-beam: $S_{pel}^T = 0$ and $S_{GIS}^T = 0$
for D-beam: $S_{pel}^D = 0$ and $S_{GIS}^D = 0$

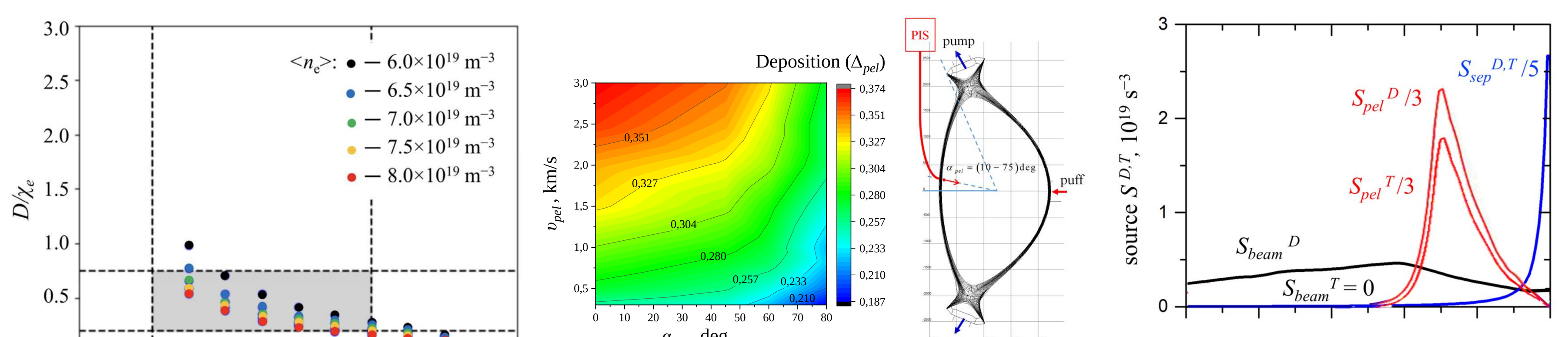




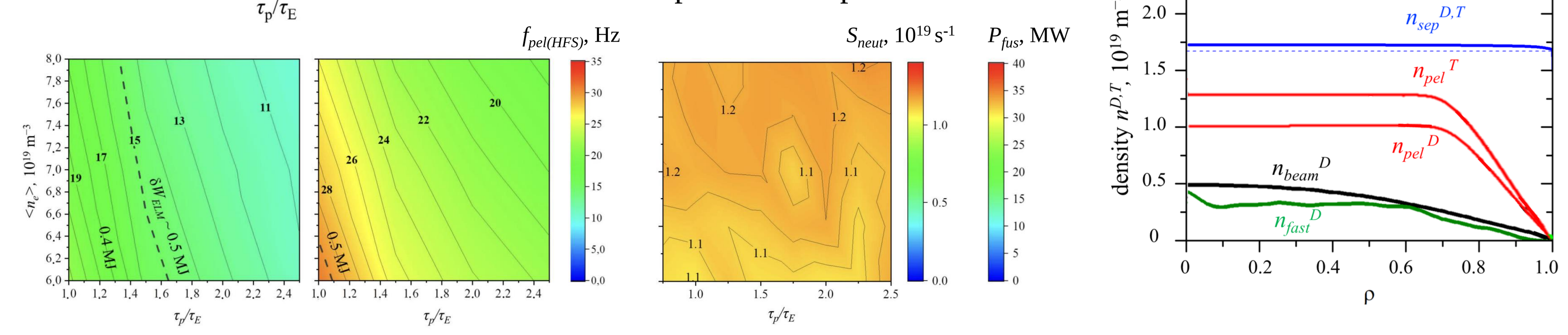
Beam isotope composition	Tritium plant, g	NBI-system, g	Fuelling systems and vacuum chamber, g	Total inventory, g
D+T (current FC architecture)	25	20	180	180-220
D (current FC architecture)	10-25	0	75-160	90-190
D (full exhaust separation)	25	0	180	210
T (current FC architecture)	15	40	60-120	115-175
T (full exhaust separation)	15	40	30	90
T (NBI gas support optimisation)	60	15	30	110

T- inventory distributed differently between at the facility!

FNS BASED ON CLASSIC TOKAMAK – DEMO-FNS

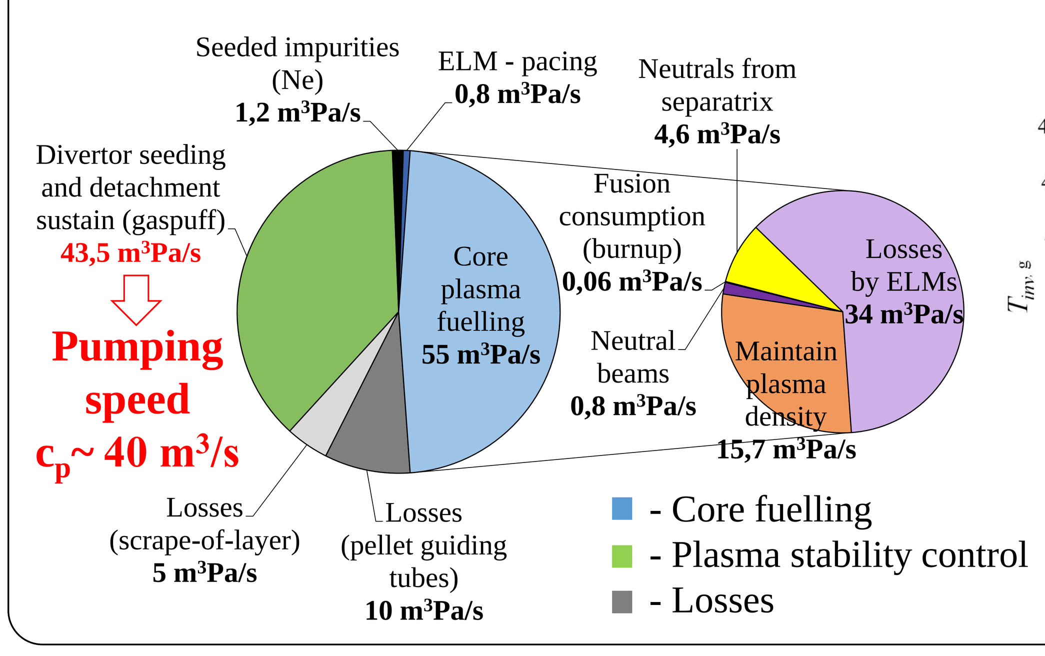


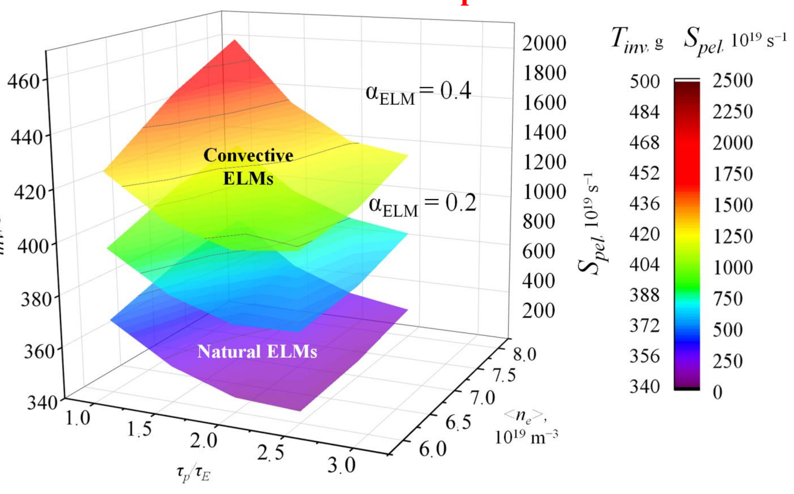
Pellet deposition vs speed and direction



Frequency of fuel pellet injection (HFS) (10% of N_{core} and the required frequency of ELM-pacing pellet injection for δW_{ELM} ~ 0.5 MJ (and below) at (laft) α_{ELM} = 0.2 and (right) α_{ELM} = 0.4

Pellet injection rate $S_{pel} \leq 2{\cdot}10^{22}$ s^{-1}





The tritium inventory at the facility up to 500 g

The range of operation parameters is found for the core plasma, in which the tritium fraction is controlled by pellet injection with a different T/D isotopic composition. Impact of neutral beam heating and fuelling by recycling and neutrals flow from divertor was accounted for as well. The neutral beams with different D/T composition and the core plasma were studied, as well as the total inventory at the facility site and the localization of tritium in FC. The influence of HFS and LFS pellet injection on the plasma core fuelling and ELM triggering is considered.

3D Tokamak Plasma Equilibrium with n = 1 Toroidal Asymmetry

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ABSTRACT

- The productiveness of the approach of how to construct the MHD plasma equilibrium in a tokamak with n = 1 violation of toroidal symmetry is demonstrated.
- For an arbitrary axisymmetric tokamak plasma equilibrium, the small n = 1 deformation of magnetic configuration, which provides plasma equilibrium, is designed analytically.
- An example of asymmetric analogue of the Solov'ev equilibrium with non-degenerated plasma pressure and current density profiles is presented. It demonstrates explicitly the existence of a non-symmetric toroidal plasma equilibrium configurations with no magnetic islands and discontinuities at rational magnetic surfaces.

INTRODUCTION

- In the experimental setups, the conceptual axial symmetry of the tokamak magnetic configuration is inevitably violated by magnetic field errors, coil discreteness, heating asymmetry, assembly defects, etc.
- Low n toroidal asymmetry is typical for tokamaks with ELM suppressing coils, with test blanket modules having non-uniform ferrous steel structures, and for those ones that exhibit the existence of long-lived saturated ideal MHD-modes.
- The theoretical description of three-dimensional equilibria causes certain difficulties related to the presence of rational magnetic surfaces, which can be easily disturbed.
- In this work, we propose and exploit a regular way of how to construct weakly asymmetric n = 1 tokamak plasma equilibrium exhibiting a system of nested toroidal magnetic surfaces.

EQUILIBRIUM MODEL

Extended mixed representation for the magnetic field [Sorokina, Ilgisonis, Plasma Phys. Rep. 45 (2019) 1093]

$$\mathbf{B} = \gamma[\nabla\Psi \times \nabla\varphi] + F\nabla\varphi - \frac{F}{r^2|\nabla\Psi|^2}\frac{\partial\Psi}{\partial\varphi}\nabla\Psi \qquad (1)$$

$\Psi = \Psi(r,\varphi,z)$ – magnetic surface, $\mathbf{B}\cdot\nabla\Psi = 0$, $p = p(\Psi)$

Equilibrium equations for three unknown functions: Ψ, F and γ

- Force balance $\cdot\,\nabla\Psi$ – 3D-analogue of the Grad-Shafranov equation (GSE)

$$\gamma^2\Delta^*\Psi + 4\pi r^2\frac{dp}{d\Psi} + \frac{F}{|\nabla\Psi|^2}(\nabla_0\Psi\cdot\nabla F) + \gamma(\nabla_0\Psi\cdot\nabla\gamma) \qquad (2)$$

$$+\frac{1}{\alpha|\nabla\Psi|^2}\left(\mathbf{B}\cdot\nabla\left(F\alpha\frac{\partial\Psi}{\partial\varphi}\right) - \frac{F^2}{2r^2}\frac{\partial}{\partial\varphi}\left(\alpha\frac{\partial\Psi}{\partial\varphi}\right)\right) = 0$$

$$\alpha = \frac{\partial\Psi/\partial\varphi}{r^2|\nabla\Psi|^2}$$

- $\nabla\cdot\mathbf{B} = 0$ – solenoidality condition

$$\mathbf{B}\cdot\nabla\left(\frac{\gamma|\nabla\Psi|^2}{F}\right) = [\nabla\Psi\times\nabla\varphi]\cdot\nabla\left(F + \frac{\gamma^2|\nabla\Psi|^2}{F}\right) \qquad (3)$$

- $[\nabla\times\mathbf{B}]\cdot\nabla\Psi = 0$ – currents closure condition

$$\mathbf{B}\cdot\nabla F\left(1 - \alpha\frac{\partial\Psi}{\partial\varphi}\right) = 4\pi\frac{\partial}{\partial\varphi}\left(p + \frac{B^2}{8\pi}\right) \qquad (4)$$

Weak asymmetry

Ψ_0 – any solution of axisymmetric GSE

$$\Psi = \Psi_0(r,z) + \epsilon[\Psi_c(r,z)\cos\varphi + \Psi_s(r,z)\sin\varphi]$$
$$F = F_0(\Psi_0) + \epsilon[F_c(r,z)\cos\varphi + F_s(r,z)\sin\varphi]$$
$$\gamma = 1 + \epsilon[\gamma_c(r,z)\cos\varphi + \gamma_s(r,z)\sin\varphi]$$

ASYMMETRIC CORRECTIONS TO STARTING PLASMA EQUILIBRIUM

Seek the partial solution of Eqs. (2)—(4) as a combination of the first spatial derivatives of function $\Psi_0(r,z)$ with unknown functional coefficients:

$$\Psi_c = a_1(z)\frac{\partial\Psi_0}{\partial r} + a_2(r)\frac{\partial\Psi_0}{\partial z},\quad \Psi_s(r,z) = b_1(z)\frac{\partial\Psi_0}{\partial r} + b_2(r)\frac{\partial\Psi_0}{\partial z},$$
$$F_s = f_0(r,z) + f_1(r,z)\frac{\partial\Psi_0}{\partial r} + f_2(r,z)\frac{\partial\Psi_0}{\partial z},$$
$$F_c = g_0(r,z) + g_1(r,z)\frac{\partial\Psi_0}{\partial r} + g_2(r,z)\frac{\partial\Psi_0}{\partial z}.$$

Result. Explicit expression for non-symmetric magnetic surface

$$\Psi(r,\varphi,z) = \Psi_0(r,z) + \qquad (5)$$
$$+\epsilon\left[(c_0\cos\varphi + s_0\sin\varphi)R\frac{\partial\Psi_0}{\partial r} + (c_1\cos\varphi + s_1\sin\varphi)\left(z\frac{\partial\Psi_0}{\partial r} - r\frac{\partial\Psi_0}{\partial z}\right)\right]$$

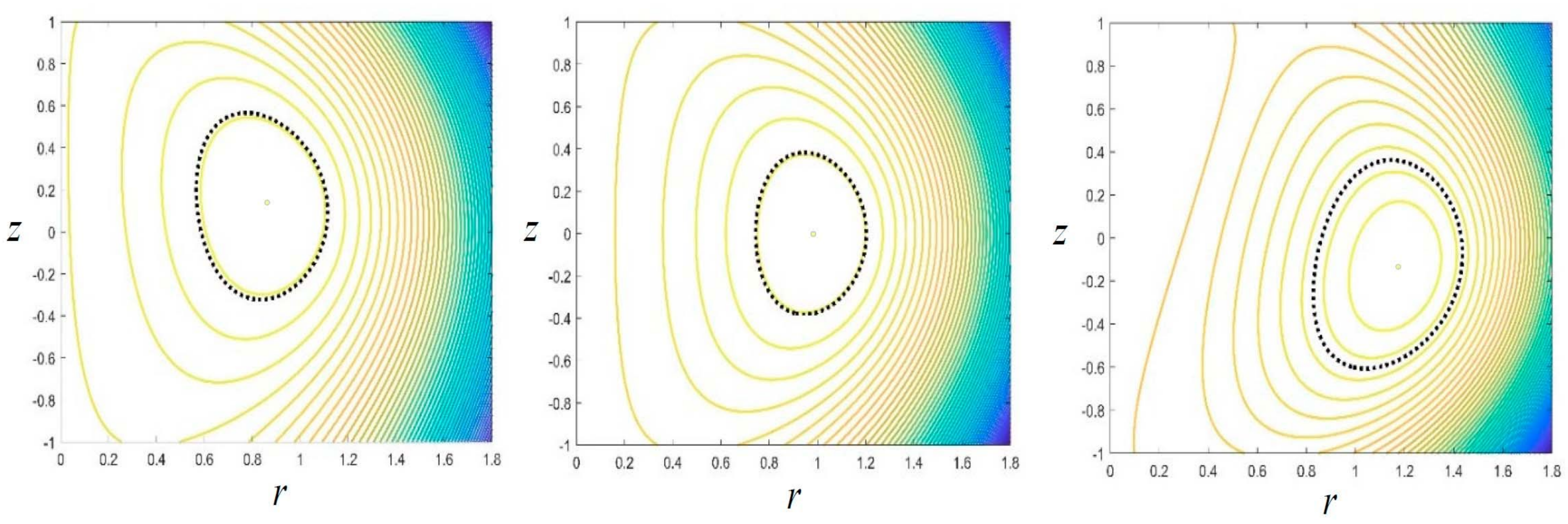
Coefficients c_0, s_0, c_1, s_1 determine the shape of asymmetric disturbances

Example. Symmetry-violated Solov'ev equilibrium

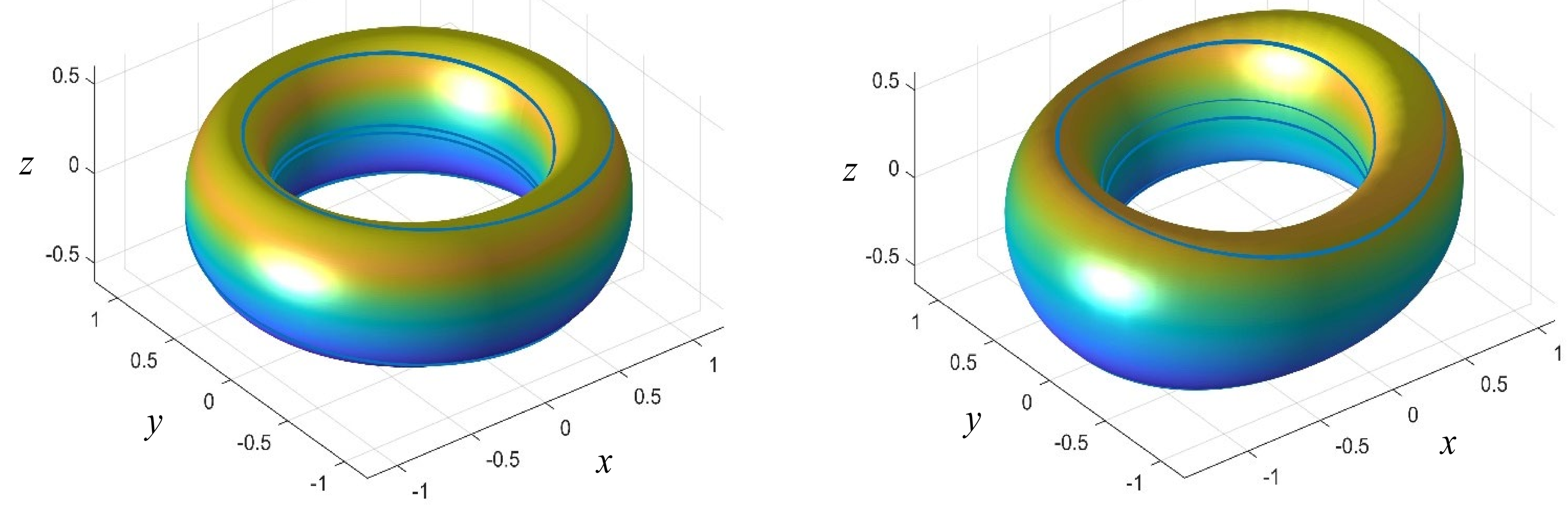
Solov'ev equilibrium – solution of the axisymmetric GSE with constant source functions. Here $\mathrm{d}p/\mathrm{d}\Psi = \mathrm{const} = -A/4\pi$ and $F_0\mathrm{d}F_0/\mathrm{d}\Psi_0 = 0$

$$\Psi_0 = \Psi_a\left(\frac{r}{R}\right)^2\left\{2 - \left(\frac{r}{R}\right)^2 - 4\lambda^2\left(\frac{z}{R}\right)^2\right\} \qquad (6)$$

Perturbed configuration calculated by Eq. (5) with Ψ_0 given by Eq. (6)



Contour-lines of function Ψ in different toroidal cross-sections



3D-view of the magnetic surface described by the axisymmetric Solov'ev solution (left), and its asymmetric modification (right)

CONCLUSIONS

- The use of the extended mixed representation for the magnetic field makes it possible to describe universally three-dimensional equilibrium plasma configurations by means of a system of coupled PDEs
- Obtained equilibrium solution allows one to calculate analytically the asymmetric n = 1 corrections to any two-dimensional configuration satisfying GSE that conserve the equilibrium conditions
- The presence of non-axisymmetric tokamak plasma equilibrium configurations with nested magnetic surfaces is demonstrated explicitly