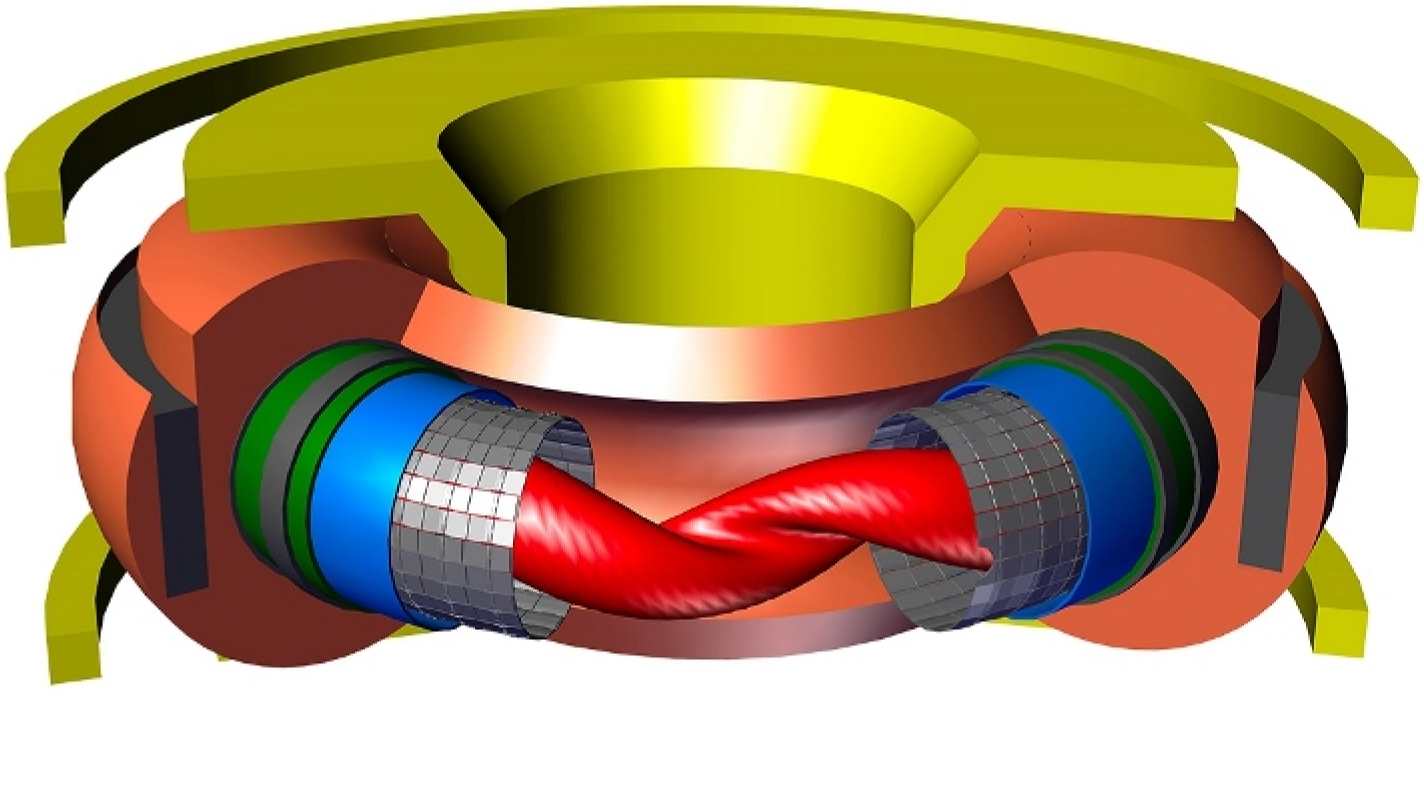
# FUNFI3

3RD INTERNATIONAL CONFERENCE ON FUSION-FISSION SUB-CRITICAL SYSTEMS FOR WASTE MANAGEMENT AND SAFETY

## BOOK OF PROCEEDINGS



Hefei, Anhui, China, 19-21 November 2018

FUNFI3 Third International Conference on Fusion-Fission sub-critical systems for waste management and safety
Book of Proceedings

Edited by Aldo Pizzuto and Francesco Paolo Orsitto

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## FUNFI3

Third International Conference on Fusion-Fission
sub-critical systemsfor waste management and safety

# Book of Proceedings

Following the first successful edition of FUNFI held in Varenna in yr 2011 (Proceedings AIP Conference Nr 1442, 2012, editors J. Kallne, D. Ryutov, G. Gorini and C. Sozzi), and the second edition held in Frascati in October 2016 (proceedingshttp://www.enea.it/it/seguici/pubblicazioni/pdfvolumi/v2017_proceedings_funfi2_2017.pdf, editors Aldo Pizzuto and Francesco Paolo Orsitto), we present the book of proceedings of the third edition( FUNFI3) of the conference (held the 19-21 October 2018 at Institute of Nuclear Energy Safety Technology, Chinese Academy of Sciences (CAS), Hefei, Anhui, China).

FUNFI3 is dedicated to the physics, technology and engineering of machines where fusion reactions drive a fission blanket. Such hybrid devices can be used for energy generation, fissile fuel production and nuclear waste transmutation. The concept of Fusion-Fission hybrid (FFH) systems was introduced in a famous paper by H. Bethe (Physics Today 1979) and recently reviewed by H. Rebut (Plasma Physics Controlled Fusion 2006). The Deuterium-Tritium fusion reactions produce 14MeV neutrons, which are able to induce fission reactions in most of actinides including uranium 238 and thorium. The fission reactions deliver an energy 10 times that of the neutrons produced by fusion: so with one fission reaction per one fusion reaction there is an energy gain of 10. In this condition, the fusion system producing neutrons can work at low minimum gain of $Q_{fusion} \approx 3$, for realizing a global energy gain of the integrated fusion-fission system $Q_{FFH} \approx 30$.
The main aims of FUNFI3 conference are:

- identify the proposals/projects with a high degree of reliability and innovation to make significant progress in the fusion neutron sources and subcritical systems technology;
- trace the path for the definition of the parameters and roles for a PILOT experiment in solving the engineering and physics problems.

Contributions concerning the following arguments are inserted in the programme:

- Mission and priorities of Demonstrators (Tokamak or mirror based systems, stellarators and other configurations including accelerator based hybrid systems);
- Physics, engineering aspects, parameters of a PILOT experiment;
- Level of readiness and development program of the essential technologies and R&D needed.

The FUNFI3 program is divided into six sessions:

Session 1: introduction and tokamak;
Session 2: gas dynamic traps;
Session 3: other confinement systems, mirrors, stellarators, reversed field pinch;
Session 4: subcritical systems;
Session 5: diagnostics and controls of Fusion-Fission hybrid systems;
Session 6: discussion on PILOT experiments.

FUNFI3 is organized by Institute of Nuclear Energy Safety Technology, Chinese Academy of Sciences (CAS), and co-organized by ENEA (Italian National Agency for New Technologies, Energy and Sustainable Economic Development), INFN (Italian National Institute for Nuclear Physics), CNR (National Research Council).

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# FUNFI3

## 3rd International Conference on Fusion Neutron Sources and Subcritical Fission Systems

### 19-21 Nov. 2018, Hefei, Anhui, China

**FUNFI3** is an outstanding international conference being an efficient exchange platform on most recent advancements made in the domains of physics, technology and engineering of fusion neutron sources and subcritical fission systems, which are very suitable for various applications including energy production, fissile fuel production and nuclear waste transmutation to enhance nuclear safety. FUNFI3 is organized by Institute of Nuclear Energy Safety Technology (INEST), Chinese Academy of Sciences (CAS) and co organized by Italian National Agency for New Technologies, Energy and Sustainable Economic Development (ENEA), Italian National Research Council (CNR) and Italian Institute for Nuclear Physics (INFN).

**The main aims** of FUNFI3 are to identify the proposals/projects with a high degree of reliability and innovation to make significant progress in the fusion neutron sources and subcritical systems technologies and trace the path for the definition of the parameters and roles for a PILOT experiment in solving the engineering and physics problems.

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**Contributions concerning the following arguments are expected:**

1) Mission and priorities of Demonstrators (Tokamak or mirror or stellarator based systems and other innovative configurations).
2) Physics, engineering aspects, parameters of a PILOT experiment.
3) Level of readiness and development program of the essential technologies and R&D needed. Proposals of synergetic experience with the present international fusion and fission development programs.

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## Key Dates

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| **Abstract Submission Deadline** | **22 Jun. 2018** |
| **Abstract Acceptance Notification** | **22 Aug. 2018** |
| **Online Registration Deadline** | **22 Oct. 2018** |
| **Conference Convened** | **19-21 Nov. 2018** |

# Contents

# Keynote

# R&D Activities of Neutronics and Lead-based Reactors by INEST/FDS Team

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## Abstract

Advanced nuclear energy system has gained much attention all over the world and neutrons play an important role in advanced nuclear energy systems. In our institution, the research efforts are devoted to the fundamental and applied research on neutronics and advanced nuclear systems.
The fundamental research is focused on the neutron physics and technology, including the development of methodology, simulation software and neutron sources and related experiments. Two representative platforms are highlighted, including the Super Multi-functional Calculation Program for Nuclear Design and Safety Evaluation (SuperMC) and the High Intensity D-T Fusion Neutron Generator (HINEG). The applied research covers advanced fission reactors, fusion reactors and extended nuclear technology applications. In this contribution, the design of China LEAd-based Mini-Reactor (CLEAR-M) for energy generation and advanced external neutron source driven nuclear energy system (CLEAR-A) for multi-purpose are introduced. The technologies and test facilities to support the development of China LEAd-based Reactor (CLEAR) series are presented as well.

## Keywords

Neutronics, Lead-based reactor, SuperMC, HINEG, CLEAR

## 1. Introduction

Advanced nuclear system has attracted more and more attention all over the world for its great superiority of sustainability, safety, and economics. Neutrons play a key role in nuclear energy systems, triggering nuclear reaction of fission system and being the main energy carrier of fusion system.
There are still some challenges for neutron physics methodology and software, especially for simulation of advanced nuclear systems. For instance, the complex geometry structures, material compositions and neutron transport process which involed neutron motion, nuclide transmutation and energy deposition, make it extremely difficult to establish accurate neutronics models. The calculation is time-consuming and accurate results are nearly impossible to obtain with good precisions in receivable time for complex issues. It is unintuitive and time-consuming to reveal physical characteristics from massive calculation data. And the cross sections at high energy, validations for new design of components for advanced nuclear systems are still lacking.
Lead-based reactor has many attractive features and may play an important role in the future energy supply [1]. Chinese government has provided continuous national support the development of lead-based reactors technology since 1986, through the Chinese Academy of Sciences (CAS), the Minister of Science and Technology of China (MOST), the Natural Science Foundation of China (NSFC), etc.
In the last 30 years, Institute of Nuclear Energy Safety Technology, Chinese Academic of Sciences • FDS Team (INEST • FDS Team) has made great efforts on the research of neutron transport physics and technology [2-7] and places more emphases on the China LEAd-based Reactors (CLEAR series) design [8-11], materials [12,13], liquid metal technology [14-17]. In this contribution, the Super Multi-functional Calculation Program for Nuclear Design and Safety Evaluation (SuperMC) [2-6] and the High Intensity D-T Fusion Neutron Generator (HINEG) [2,3,6,7] as the representative work of neutronics methodology & software, neutron sources and experiments, are introduced. The design of China LEAd-based Mini-Reactor (CLEAR-M) for energy production and advanced external neutron source driven nuclear energy system (CLEAR-A) for multi-purpose are presented. The technologies and test facilities to support the development of CLEAR series lead-based reactors are introduced as well.

## 2. Neutronics Studies

Especially to solve the challenges of advanced nuclear systems on the steps of modeling, calculation and visualized analysis, the CAD-based whole-process neutronics modeling and simulation method [2-5] were developed. The CAD-based whole process accurate modeling for neutron transport, feature-accelerated high-efficiency calculation and intelligent visualized analysis are three main parts of neutronics methodology. Firstly, the CAD-based whole process accurate modeling for neutron transport was developed to solve the modeling challenge. Different from the traditional modeling method based on regular geometry elements with a lot of approximations, this method can accurately build irregular geometry model, which can describe the systems using hierarchical structures, various facets and irregular primitives by solid decomposition. Unified model is also established for the whole neutron transport process of 'neutron motion-nuclide transmutation-energy deposition' to avoid errors due to conversion between models of isolated processes in traditional modeling method. Secondly, the feature-accelerated high-efficiency calculation method was proposed to achieve high efficiency calculation with high fidelity. To reduce the unnecessary simulation, the feature of particle location is pre-judged based on space division, and particle tracking is biased during transport according to particles density distribution. Besides, combining the advantages of MC and deterministic methods, MC-deterministic hybrid methods were developed. An adaptive transaction region was established in MC-deterministic hybrid on spatial region, to take into account the nonlinear impact of the secondary particles to interface source. Thirdly, the intelligent visualized analysis is to achieve intuitive analysis of massive calculation data. For multi-style visualized analysis of 3D data, the spatial data is visualized coupled with interested geometry, based on pixel-navigated LOD (level of detail) technology and direct mapping using GPU. And dynamic visualized analysis, such as real-time organic dose assessment by virtual simulation, is achieved based on hybrid hierarchical tree of geometry and with accurate voxel human model.

Based on methodologies above, SuperMC has been developed. It supports the whole-process neutronics simulation and can be extended to perform multi-physics coupling simulation based on unified modeling. SuperMC has been adequately verified and validated by enormous benchmark models and experiments, widely distributed by OECD/NEA Data Bank, and applied in more than 60 countries, and more than 40 mega nuclear engineering projects. Furthermore, taking the neutronics code SuperMC as the core, it is extended to be incorporated system control & safety simulation, to form the Virtual Nuclear Power Plant Virtual4DS, which aims to be a full-scope and full-period safety simulation and emergency decision support platform for future nuclear power plants.

There are three development phases of HINEG. The first phase, named HINEG-I, aimed to develop a neutron source with the intensity of $10^{12}$-$10^{13}$n/s, which has been used to support the R&D of 10 MW reactor for fission, the fundamental research on neutronics and nuclear technology for fusion, and also the applied research for nuclear technology applications. A D-T fusion neutron yield of up to $6.4\times10^{12}$ n/s has been generated by HINEG-I, which is the highest neutron yield among the accelerator-based D-T neutron generators in operation. The second phase, named HINEG-II, aims to develop a neutron sources with yield of $10^{15\text{-}18}$ n/s, including HINEG-IIa and HINEG-IIb. HINEG-IIa is a Mixed-beam Fusion Neutron Source and HINEG-IIb is a Superpower Spallation Neutron Source. HINEG-II can support the development of 100MW fission reactor, and test the materials and component performances for fusion. HINEG-II can also be used for the research of nuclear technology applications. The third phase, named HINEG-III, aims to develop neutron sources with intensity of more than $10^{18}$ n/s, including the GDT-based Fusion Volumetric Neutron Source (FVNS) and Multi-target Spallation Neutron Source. As the fusion/hybrid testing reactor, HINEG-III can support to do integration test of nuclear system engineering. A series of experiments have been carried out on HINEG-I facility, such as the neutronics performance test of dual function lithium-lead (DFLL) test blanket module (TBM) mock-up, measurement of neutron leakage spectra from Pb and LBE, neutron radiography nondestructive testing of aircraft engine blades, neutron biological effect for C.elegans, and so on.

## 3. Design and R&D of Lead-based Reactors

### 3.1 Design

Recently, the CLEAR project has been supported by national/local government and industry investment. We are evaluating, comparing & investigating various concepts of lead-based reactors for different application purposes, such as CLEAR-M, CLEAR-A, et al. In the meantime, three test facilities in parallel named CLEAR-M10a, CLEAR-A10 and CLEAR-I will be built to support CLEAR-M and CLEAR-A projects.

The objective of CLEAR-M project is to develop small modular energy supply system. The main purpose of this system is to provide electricity as a flexible power system for wide applications such as island, remote districts and industrial park etc. The typical design of CLEAR-M is called CLEAR-M10, which is a 10 MW level electric power reactor. The principle of CLEAR-A is an external neutron source driven subcritical lead-based reactor. The main purpose is to make use of depleted uranium, thorium or spent fuel from PWR as fuel to achieve high fuel utilization and nuclear waste minimization while producing energy. In order to validate the engineering technology of the external neutron source driven nuclear energy system, a 10 MW Advanced External Neutron Source Driven Nuclear Energy Experimental System (CLEAR-A10) [3] is proposed to be built in the near future. CLEAR-A10 is a lead-cooled experimental reactor for the technology tests of nuclear breeding, nuclear waste transmutation and energy production. Another accelerator driven lead-bismuth cooled subcritical experimental system named CLEAR-I [11] was also developed supported by CAS ADS project for nuclear waste transmutation research, which was launched in 2011.

### 3.2 Key Technologies and Testing Facilities

Heavy liquid metal coolant technology R&D activities were being carried out to support CLEAR projects, and mainly focused on key components, structural material and fuel, reactor operation and control. The key component prototypes and the integrated operating technology have been validated and tested, which including the main pump, heat exchanger, CRDM, and refueling system [10].

Based on the single engineering technology test and the equipmental prototype development represented above, the Multi-functional lead-bismuth loop KYLIN-II and three integrated test facilities were constructed, including the lead alloy-cooled integrated non-nuclear pool type facility CLEAR-S [17], the lead-based zero power critical/subcritical reactor CLEAR-0 [7], and the lead-based virtual reactor CLEAR-V. The purpose of these test facilities is to meet the integrated testing requirements of the key components and technologies for CLEAR.

## Summary

The INEST • FDS Team have carried out systematic neutronics studies, especially for advanced nuclear systems. On methodology and software, the CAD-based whole-process neutronics modeling and simulation methods were developed. SuperMC, the comprehensive neutronics code, has been applied in more than 60 countries, and more than 40 mega nuclear engineering projects. The neutron yield of HINEG is the highest among accelerator-based fusion neutron sources in operation.

The INEST • FDS Team placed more emphases on design and R&D of CLEAR series reactors for more than 30 years. The concepts of mini-reactor CLEAR-M for energy generation and subcritical system driven by external neutron source CLEAR-A for multi-purpose as well as three experimental reactors has been proposed, and the summary of the technology R&D activities has been given.

## Acknowledgments

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# The ENEA facilities for the physics and engineering study of a fusion-fission hybrid reactor

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## Abstract

ENEA has the role in the Italian research system to build and maintain science facilities. A number of these can be used for studies dedicated to the Fusion-Fission Hybrid (FFH) reactor conceptual feasibility. In particular three experimental facilities located near Rome, a 14 MeV neutron source (Frascati Center) and two research fission reactors (Casaccia Center) can be available for focused experiments on the feasibility of FFH systems. The 14 MeV neutron source (named FNG - Fast Neutron Generator) can be used to investigate the properties of suitable blankets able to provide neutron energy spectrum typical of a fusion reactor. The two research fission reactors are a thermal TRIGA 1 MW reactor (named TRIGA RC-1) and a compact fast 5 kW reactor (named RSV TAPIRO). Both systems, eventually coupled to a suitable fusion-type neutron source, can be used to study some neutronic essential properties of the FFH blankets, like the space/energy shape of the neutron flux inside the FFH blanket and the monitoring criteria to measure and control the subcritical reactivity level of the FFH blanket. Possibly the transmutation of actinides can be studied using fusion-like neutron spectra.
Such studies can take advantage of the experience gained in the past in the frame of the analyses on Accelerator Driven Systems - ADS: i.e. the experimental campaigns MUSE and pre-TRADE.

## Keywords

Fusion Neutron Sources, Subcritical Systems

# Current status of the experiments on the GDT device in support of the GDT-based neutron source project

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## Abstract

The gas dynamic trap (GDT) is a version of a magnetic mirror with a long mirror-to-mirror distance far exceeding the effective mean free path of ion scattering into the loss cone. In the paper, current status of the experiments on the GDT device in Novosibirsk is overviewed. The main goal of the experiments is to develop database for construction of the GDT-based neutron source for fusion materials development and other applications that are briefly discussed in the paper.

## Keywords
Gas dynamic trap

## Introduction

A magnetic mirror trap (`probkotron') is topologically different from tokamaks and stellarators. It was proposed in 1954 by G I Budker (USSR) and, independently, by RF Post (USA) [1,2]. In such devices, the plasma is confined by a transverse magnetic field, and its free axial flow is limited by particle reflection from high magnetic field regions, so-called magnetic mirrors. The reflection of the particles proceeds due to energy conservation and conservation of magnetic moment of the `Larmor circle' of particles, which is an adiabatic invariant when the particles travel in a weakly nonuniform magnetic field.

Early magnetic mirror experiments exhibited several intrinsic problems of this approach. It was observed that in a simple axisymmetric configuration the magnetic mirrors suffered from development of fast MHD plasma instability. Later on it was proposed a magnetic configuration with a minimum B, which is necessary to provide MHD stability and this was experimentally proven. However, the non-axisymmetric magnets, which provide such configuration are rather complex from engineering point of view and expensive. After successful demonstration of MHD stable plasma confinement in the experiments, it was found out that plasma is subjected to micro-instabilities. The natural anisotropy of the ion distribution function in the velocity space (Fig.1) is associated to the fact that only ions with large enough magnetic moment or transverse velocity can be confined by magnetic mirrors. So, in velocity space there are empty “loss cones” that causes excitation and growth of amplitudes of the electrostatic and electromagnetic waves, which at the same time leads to an increase in the

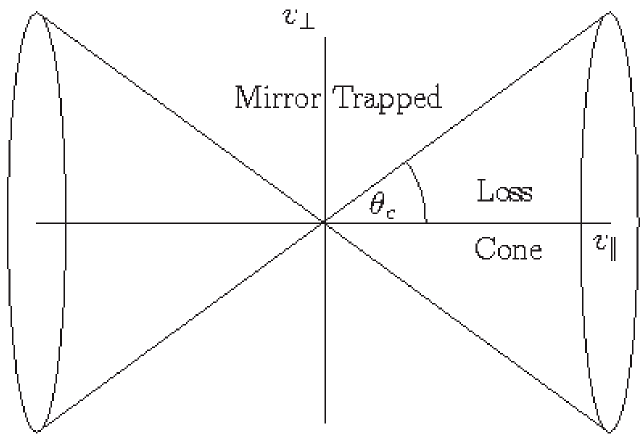


*Fig. 1 - Ion distribution over pitch angle in magnetic mirror machine*

ion scattering rate and decrease the confinement time. The most dangerous electrostatic drift loss-cone instability was suppressed in the experiments by addition of small fraction of warm ions [3]. However, then it was found that electromagnetic Alfven waves can be driven unstable in high-beta anisotropic plasmas.

At the same time, it was generally recognized that even without excitation of MHD or kinetic instabilities axial plasma losses from magnetic mirrors are too high (confinement time close to time between ion-ion collisions). Another drawback of the mirror experiments during the decades was rather small electron temperature. The maximum electron temperature achieved in the experiments was $T_{e\ record}$ = 280 eV (TMX-U, LLNL 1986).

The gas dynamic trap, a version of magnetic mirror is proposed by V V Mirnov and D D Ryutov in the late 1970s [4]. It is basically a long axisymmetric solenoid with magnetic mirrors at both ends for plasma confinement (Fig. 2). The mirror ratio (ratio between mirror field and that in the center) is taken to be high, and the plasma in a solenoid is assumed dense enough, so that the ion mean free path for scattering into the loss cone becomes shorter than the trap length. The collisional plasma confined in the GDT central solenoid exhibits isotropic and Maxwellian velocity distributions of particles, except a small region in the immediate proximity to the magnetic mirrors. Such a design precludes the development of kinetic instabilities inherent in classical open magnetic plasma traps.

## GDT-based neutron source

While the scientific development of the open confinement systems is less mature than for the tokamak, substantial progress has been made in recent years. Particularly, this is true for the experiments at the gas-dynamic trap in Novosibirsk reviewed in [5]. At the GDT device, successful application of several methods for stabilization of most dangerous flute instability in axisymmetric open traps has been demonstrated. It was done by using partial line tying to the end wall, make use of outboard MHD anchors with large favorable curvature of the field lines, and sheared plasma rotation at periphery induced by the biased limiters and segmented end walls [6], resulting in confinement close to that defined solely by axial energy losses. Considerable increase of electron temperature in the GDT experiment (to about 1keV) was obtained using neutral beam heating of central cell plasma in combination with ECR heating [7,8]. Essentially, then it becomes possible to consider GDT for the first practical applications to a fusion neutron source, which can be dedicated, for example, to fusion materials development [9-11], or as a driver for subcritical fission reactors [12]. The axisymmetric configuration of GDT is an attractive option to provide a relatively simple neutron source for fusion-fission hybrid applications.

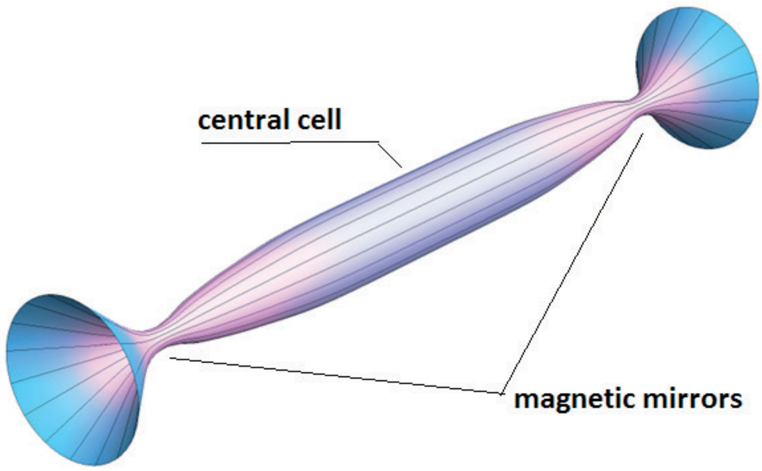


*Fig. 2 - Artist view of magnetic field lines in gas-dynamic trap*

The analysis made in [5] demonstrated that such a 14-MeV neutron source might possess unique characteristics even at plasma parameters close to those already have been achieved in the experiments and, what is of a paramount importance, despite a rather conservative estimation of the prospects for their further improvement. Moreover, the generator can be created based on currently available technologies to exemplify the first practical peaceful application of a thermonuclear device. An important advantage of the GDT-based generator is the possibility of reaching the ß values close to one. The fusion reaction rate per unit volume is $\sim\beta^2 B^4$, which provides a basis for designing a relatively compact machine with a low power and tritium consumption.

The necessity of creating such a neutron source for the purpose of accelerated testing of materials and component units for a future tokamak fusion reactor plant is universally recognized. Its use would be instrumental in addressing such difficult problems of development if radiation-resistant engineering materials and those with low induced activity. The GDT-based 14 MeV neutron source has prospects for application not only in basic research in solid state physics and thermonuclear materials science but also as a device for afterburning the radioactive waste and radionuclide production or a hybrid nuclear power plant with a high degree of internal safety.

The most attractive option is a GDT neutron source with a multicomponent plasma [5] consisting of relatively cold (Te ~1 keV) and dense ($n\approx10^{20}$ $m^{-3}$) plasma confined in the gas dynamic regime and a population of fast anisotropic ions that oscillate back and forth between reflection points near magnetic mirrors. Fast ions are generated by injection of deuterium and tritium neutral beams with an energy of ~100 keV at a small angle (~$20^0$) to the axis.

Due to the relatively low temperature of the target plasma, fast ions are slowing down more efficiently than they are scattered. As a result, the ion angular distribution during the slow-down process remains almost as narrow as at its beginning. The axial velocity of ions near turning points is small and they occupy this region for most of the period of longitudinal oscillations, which accounts for the higher density of fast ions here than in the trap center. The flux of neutrons formed in collisions between fast tritons and deuterons in this region can reach several MW/$m^2$ corresponding to the conditions under which the first wall of the tokamak reactor operates. Such a neutron source is needed to develop materials with extended lifetimes corresponding to the fluence of at least 10-15 MW year /$m^2$ or 100 displacements per atom in the lattice and with a minimum activation under the effect of neutron bombardment.

To conclude this section, the following advantages of the proposed GDT-based neutron source for materials science research should be emphasized:

- The natural continuous mode of operation of the neutron source. The possibility of modulating the neutron flux in a characteristic frequency of a few

kilohertz. The availability of a comprehensive open-ended magnetic traps' database. The data necessary to realize the continuous operation regime of the generator can be obtained with the help of either a hydrogen prototype of the neutron source that does not need special shielding or with a specialized facility.
- The value of $\beta \sim 1$ can be reached in open-ended magnetic traps (i.e., plasma pressure can be close to the pressure of a confining magnetic field), allowing facilities to be created with large neutron fluxes and a plasma volume as small as several liters for materials testing.
- The possibility of creating a source with a 1-2 MW /m$^2$ neutron flux and a large (~100 l) testing zone to form blankets for tritium production.
- Electron temperature can be raised to self-consistent values under conditions of suppressing cold secondary electron fluxes emitted from the end wall. The relevant theory proposed and is confirmed in experiment. A rise in electron temperature would allow the heating power to be decreased at a given neutron flux in the test zone.
- Low tritium consumption with a small amount of this isotope in the facility. It does not need to be produced in the system and can be obtained from a commercial source. A tritium recovery system can be integrated into the facility if appropriate.
- High density of the neutron flux (> 2 MW/m$^2$) permits accelerated testing of materials.
- The small neutron flux outside the test zone and insignificant thermal load spares the facility from critical impacts.
- The primary D-T neutron spectrum corresponds to the first wall irradiation conditions in tokamak reactors. The spectrum has no tail of high-energy neutrons, as in accelerator-based spallation reaction [13] or stripping reaction (D-Li IFMIF type) [14] sources.
- The source makes use of simply designed and therefore inexpensive magnets.
- Well-developed positive ion-based neutral beams can be applied to heat and sustain plasma in the neutron source, as it enters the continuous working regime.
- Expansion of the plasma jet behind the mirror in the flaring magnetic field permits reducing thermal load on the plasma dump surface to an acceptable level of 1 MW /m$^2$ or less.
- Only those technologies that have been developed specially for fusion research are employed to design, construct, and operate the facility, viz. neutral beams or possibly additional ECR- and ICR-heating, superconducting magnets, tritium systems for the neutron source operated in the continuous mode for Q < 1.
- The estimated construction cost of the neutron source is 10% of the ITER cost.
- The neutron source can be put into operation with a hydrogen plasma, which permits activation of engineering materials to be avoided at this stage.

## Experimental GDT device

The experiments at GDT device in Novosibirsk (Fig. 3) are carried out to address the issues of suppression of transverse transport caused by MHD instabilities, influence of micro-instabilities onto the fast ion confinement, development of plasma heating methods, and achievement of the electron temperature necessary for fusion applications.

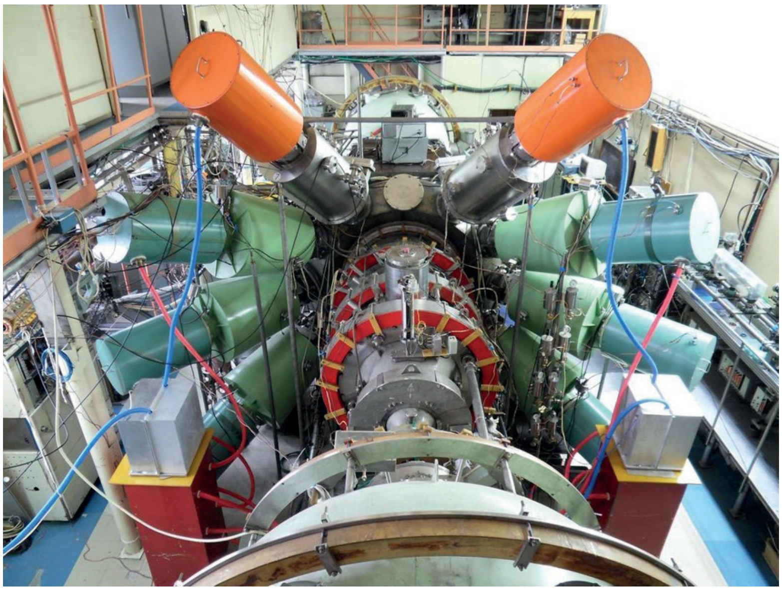

*Fig. 3 - Picture of the GDT device*

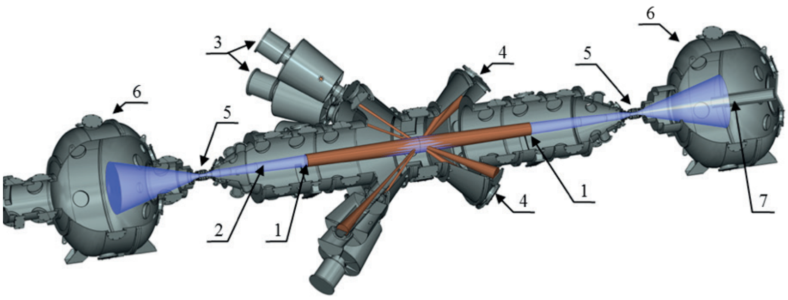


*Fig. 4 - 1-fast ion turning point; 2-warm plasma; 3-neutral beams; 4-beam dumps; 5-mirror coils; end tanks; 6-arc-plasma source*

The magnetic coils of GDT device are axisymmetric as shown in Fig. 4. The GDT consists of a 7-m long central solenoid, a vacuum chamber about 1 m in diameter in the central part, and two end expander tanks 2.6 m in diameter each. The total volume of the vacuum chamber is 15 m$^3$. The working pressure in the central part before a shot ranges (0.5-1)x10$^{-7}$ Torr. The magnetic field is generated by coils placed directly on the central vacuum chamber and by the mirror sections. The trap is preliminarily filled with plasma from a plasma gun mounted in one of the end tanks. The magnetic mirrors of the trap are formed by external large-radius coils serially connected to the solenoid magnetic system and internal coils fed from an independent power supply. Such a design makes it possible to vary the field in the mirrors up to 16 T or the mirror ratio in the range from 12.5 to 100, while the central field varies from 0.1 to 0.35 T. Changes in the mirror field have practically no effect on the field strength and the curvature of field lines in the central solenoid and the expanders. The main parameters of the device is shown in Table 1.

| Parameter | Value |
|---|---|
| Mirror to mirror length | 7 m |
| Magnetic field at midplane | up to 0.35 T (in mirrors 2.5-15 T) |
| Plasma density | $1\text{-}6{\cdot}10^{19}m^{-3}$ |
| Plasma radius at midplane | 6-7 cm |
| Electron temperature | 250 eV (up to 900 eV with ECRH) |
| Injection energy of neutral beams | 20-25 keV |
| Pulse duration | 5 ms |
| Injection power | upto 5.4 MW |
| Injection angle | 45° |
| Fast ion density (turning points) | $\approx 5{\cdot}10^{19}m^{-3}$ |
| Mean fast ion energy | ≈ 10 keV |

*Table 1 - The main parameters of the experimental GDT device*

## Findings from the GDT experiments

The GDT experiments have successfully demonstrated that plasma MHD-driven instabilities in an axisymmetric gas dynamic trap can be stabilized by external cells with a favorable curvature of magnetic field lines. Two types of such stabilizers were studied: the expander, and the cusp; they were filled with the plasma flowing out of the trap. Also studied was plasma confinement with small transverse losses under conditions when the weighted mean curvature of magnetic field lines was unfavorable for stability. In this case, the plasma was confined inside the vortex flow with a large velocity shear at the periphery, maintained by applying potentials to radial limiters and segments of the end plasma dump.

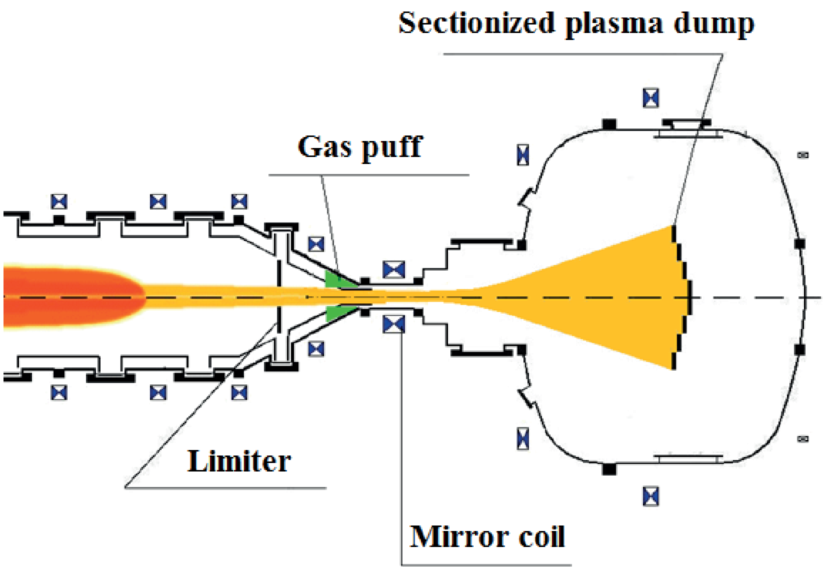

*Fig. 5 - Scheme of vortex confinement*

The scheme of the experiment is shown in this case in Fig. 5.

In this arrangement, electrodes of a sectionized plasma dump and biased limiters are employed to provide a zone of sheared plasma rotation at periphery of plasma column. This results in saturation of large scale MHD modes and suppress the cross-field plasma transport. Shear flows, driven by the biased end-plates and limiters, in combination with finite-Larmor-radius effects are shown to be efficient to radially confine high-β plasma even with magnetic hill on axis. Interpretation of the observed effects as the "vortex confinement", i.e., confinement of the plasma core in the dead-flow zone of the driven vortex, agrees rather well with simulations and experiment. Theoretical scaling laws predict such confinement scheme to be also applicable at higher plasma temperature and density. Note that such a technique has been used in many experiments on suppression of MHD-driven instability in a nonuniformly rotating plasma [15-18]. In [19] also is shown that a large enough rotational flow velocity shear, either spontaneous or generated by end electrodes and local ECR heating, is responsible for the suppression of MHD and drift modes. In this regime, plasma beta measured in the fast ion turning point reaches ~ 0.6 as shown in Fig. 6.

The accumulation of fast ions in high beta regimes was accompanied by saw-like relaxation oscillations of signals from diamagnetic loops. Transverse plasma pressure redistribution was studied with a set of magnetic sensors arranged parallel to the trap axis. The measurements

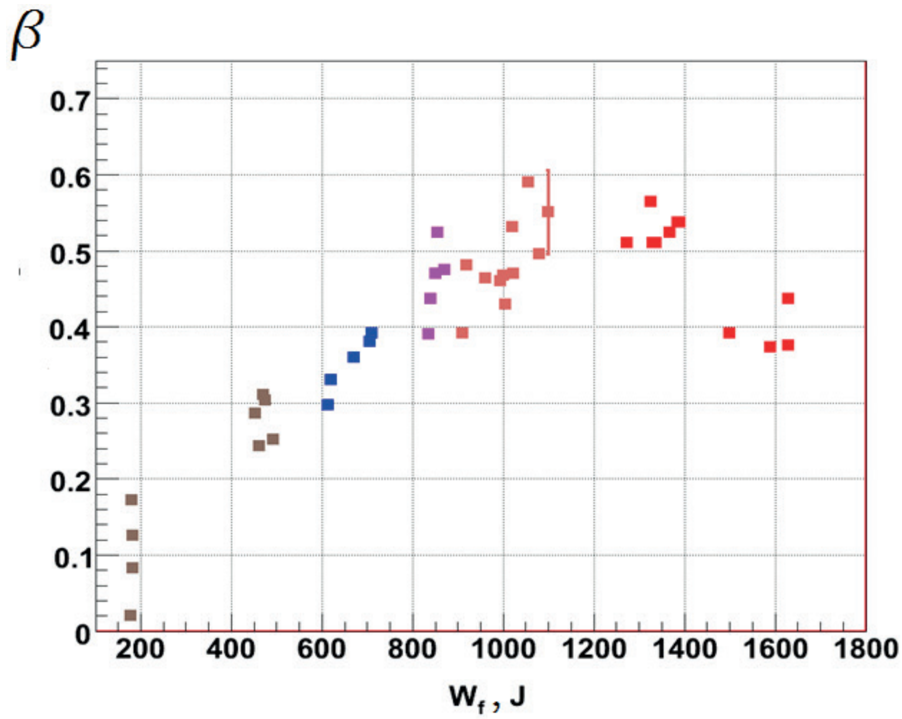


*Fig. 6 - Plasma beta vs fast ion energy content (measured by MSE diagnostics in the turning points region of fast ions at the axis of plasma column)*

indicate a presence of perturbation of the radial component of the magnetic field with respect to its value in a vacuum; the main contribution to the perturbation comes from fast ions. Broadening of its axial pressure profile is most noticeable at a time instant of around 5.5 ms, when the first dip in the diamagnetic signal becomes apparent. At this time, the perturbation amplitude of the radial magnetic field decreases near the center of the ion turning region (the dark, while it grows at the edge of this region. At the same time, the dips in the diamagnetic signals were accompanied by bursts of HF oscillations at a frequency of around 1 MHz (Fig. 7).

The measured characteristics of the plasma HF oscillations suggest that they correspond to bursts of the Alfven ion cyclotron (AIC) instability [20]. A theoretical description of this instability type for open-ended magnetic traps with oblique beam injection is presented in [21]. It is worthwhile to note that the development of AIC instability in GDTs, in contrast to many other systems, does not cause substantial losses of fast ions [22], because only a small group of them having a maximum energy (i.e., fast ions freshly

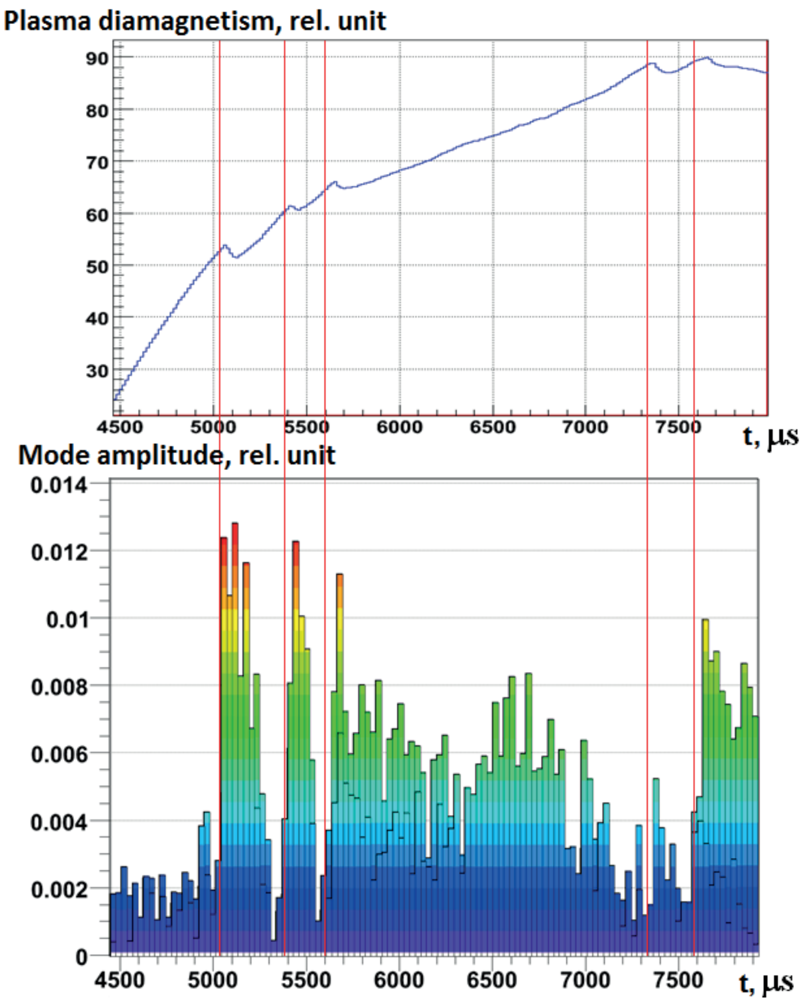


*Fig. 7 - Plasma diamagnetism in fast ion turning point and amplitude of perturbation of radial component of magnetic field*

captured from heating neutral beams) is responsible for the wave buildup. It is enough to only slightly increase (by ~1-2$^0$) the angular spread of such particles to saturate oscillations, whereas the pitch angle must be changed by ~40$^0$ to make such particles enter the loss cone.
The parameters of the experiment are shown in Table 2 for different methods of MHD stabilization.

axis is fairly well described in terms of classical electron drag mechanisms and ion-ion scattering. AIC instability develops in the plasma only in the case of attaining its maximum relative pressure [20]. However, neither enhanced fast ion losses nor an appreciable broadening of the turning point region is observed in the presence of developed instability.

| Parameter | Expander [99] | Cusp [113] | Plasma rotation [121] |
|---|---|---|---|
| Central magnetic field, T | up to 0.22 | up to 0.22 | up to 0.3 |
| Mirror magnetic field, T | 2.5 - 15 | 2.5 - 15 | 2.5-15 |
| Primary plasma density, $m^{-3}$ | (1.5-7) x$10^{19}$ | 4.5 x$10^{19}$ | (3-6) x$10^{19}$ |
| Plasma radius in trap center, cm | 6.5 | 5-10 | 6-7 |
| Electron temperature, eV | 25 | 110 | 250 (~900 with auxiliary ECRH) |
| Energy of injected deuterium or hydrogen beams, keV | 15 | 15-16 | 24-25 |
| Total injection power, MW | - | 4 | 5.7 |
| Maximum local ß | 0.07 | 0.1 | Up to 0.6 |

*Table 2 - Plasma parameters in GDTs obtained with the use of different MHD stabilizers*

Experiments showed that losses of heat from the trap due to electron thermal conductivity can be greatly suppressed by decreasing the magnetic field between the mirror and the trap end. The drop of plasma density in the expanding flux behind the mirror gives rise to a deep potential well for electrons in the central solenoid. As a consequence, most electrons from the trap central section cannot reach the wall. In addition, the expansion of the flux behind the mirror causes the potential profile to flatten in the outside part of the end cells. Under these conditions, cold electrons emitted from the wall where the magnetic field is much weaker than the mirror field can not enter the central solenoid and cool the plasma it contains. Therefore, it can be concluded that axial heat losses from the central solenoid of a GDT are largely determined by plasma flux through the mirrors. The experiments also demonstrated that the measured longitudinal particle and energy fluxes from the trap at low temperatures (T<50 eV) agree with those predicted by the gas dynamic model of plasma flow in magnetic mirrors [23]; at higher temperatures, the observed flows are consistent with the theoretical ones described by the collisionless model [24]. Plasma heating by the injection of neutral beams is associated with the predominance of axial losses in the energy balance, while the fraction of transverse losses is below 15%. In such regimes, electron temperature in the GDT is 250 eV depending on the balance of energy transferred from fast ions to electrons and plasma outflow through the magnetic mirrors. Additional ECR heating of the plasma in the central section of the trap raises the electron temperature to almost 1 keV. Confinement of fast ions injected into the trap in its central section at an angle of 45$^0$ to the

## Conclusions

Successful operation of GDT with high beta, the classical behavior of fast injected ions, and the possibility of heating electrons in the trap to about 1 keV taken together provide a solid basis for considering GDT a prototype of a D-T neutron source with a neutron flux up to 2 MW/m$^2$ annually consuming < 0.2 kg of tritium. Such a source faces no serious physical, engineering, or technological limitations. Importantly, the energy spectrum of GDT neutrons is identical to that of fusion neutrons in the ITER and future DEMO power plants; it satisfies all requirements imposed on neutron sources for materials testing to be used in thermonuclear reactors.
For all that, the plasma was nonstationary under the conditions of the above experiments. Suffice it to say that its electron temperature in stable regimes continued to grow over the entire duration of an injection pulse. This means that the operating time of the GDT facility must be further extended to reach stationary conditions (e.g., to 20 ms or more at the electron temperature of ~ 200 eV). There are plans to realize such regimes in the course of the device modification.
Specifically, the Budker Institute of Nuclear Physics (Novosibirsk) plans to construct a stationary plasma confinement system, a gas dynamic multiple-mirror trap (GDMT) [25]. The GDMT concept is based on the employment of multiple-mirror end solenoids for more efficient suppression of axial plasma losses than in the prototype GDT. Naturally, such a facility must have a superconducting magnetic system and a duration of microwave or beam injection heating of around 100 s if the stationary conditions are to be created. The

primary objective of the GDMT experiment will be to test the concept of a stationary multiple-mirror fusion D-T reactor. Given that the problem of stability at high plasma pressure and temperature is successfully solved, a GDMT- based thermonuclear reactor may be designed in the distant future making use of new `aneutronic' fuels, such as D-$He^3$, $He^3$-$He^3$, and p-$B^{11}$.

## Acknowledgments

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# Overview of fusion development strategy in the United States

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## Abstract

One significant gap in scientific knowledge towards the development of a fusion power system is the development and understanding of material performance and microstructural evolution of candidate materials for plasma facing components under prototypical neutron irradiation conditions. Damage levels will be in 10's of dpa, in the presence of significant helium concentrations. The US domestic fusion program is in the beginning stages of examining possible fusion neutron sources, and the outcomes of a recent workshop will be presented. The multivariate effects of neutron damage and plasma flux on plasma-materials interactions will also be examined in the Materials Plasma Exposure eXperiment (MPEX) which is currently undergoing conceptual design. Samples which are irradiated in the ORNL High Flux Isotope Reactor (HFIR) will be exposed to a steady-state plasma with a fusion power-plant relevant plasma fluence. The presentation will include a discussion of the science program and current plans for MPEX. Finally, considerations of the use of material capsules irradiated in HFIR for international structural materials programs will be presented.

## Keywords

Fusion neutron sources, Linear plasma facilities, material plasma interactions

*Session 1*

# Tokamak

# Requirements to facilities, materials and technologies for FFHs

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## Abstract

Development of innovative fusion and fusion-fission hybrid (FFH) systems needs very clear definition of progress priorities and startup conditions. Despite long-term history, FFHs are still at a conceptual design level and no one either operating or constructing FFH facility. Critical technologies for FFHs are steady state technologies for tokamaks, tritium, hybrid fuel cycle with high-level fuels, molten salt and heat exchange technologies. Remote handling, first wall, divertor, blanket, auxiliary heating and current drive, safety and disruption mitigation systems need substantial up-grade. Target specifications for those are discussed in this paper. Successful development of FFHs within next decades will make an important contribution in development of fusion and fission energy.

## Introduction

FFHs have a long-term history of development [1]. At nowadays they have reached a level of principal decisions due to better understanding the fusion and fission physics as well as feasibilities of enabling technologies and properties of modern materials, both structural and functional, for nuclear devices. To push further FFHs implementations it is necessary to overcome several show stoppers, which prevent transition to practical realization of FFHs positive inputs in sustainable development of atomic energy.

First of all, it is desirable to make higher political support of early fusion applications that is still at a low level in any country in the world. This should be provided together with legislation and regulation of design and operation of FFHs as nuclear facilities. Capabilities of FFHS to operate far from ignition conditions close to practically reached plasma parameters in contemporary tokamaks and stellarators transfer those into realizable facilities in decade-time scales. A smaller size of facilities required for demonstration, pilot and commercial FFHs being compared with pure fusion devices like ITER makes reaching the industrial level possible cheaper and faster that opens new opportunities both in fusion and fission technologies [2].

Special efforts are still needed for the public acceptance of FFHs that may be reached accounting for FFHs inputs in nuclear waste processing and carbon free energy development. Proliferation concerns dealing with FFHs are comparable with fast and thermal reactors development, so this problem will require IAEA guidance and control. Certainly, large scale investment (1-100 $B) will be needed for global implementation of FFHs in the atomic energy field.  Meanwhile, small scale proposals do needed accounting for very low level of current needs in nuclear reprocessing and breeding technologies. This paper tries to clarify where FFHs are and when they could be realized in Russia.

## 1. How to reach FFHs development goals

According to our analysis a few steps may be helpful to make FFHs activation faster.  Those are as follows:

- start fusion neutron devices as fast as possible, as simple as possible, starting from D+D fusion;
- use opportunities of non-Maxwell beam-plasma fusion;
- think about a broader set of products and services;
- evaluate opportunities of fusion systems with power in 1kW-1MW power, compare those with accelerator analogues and immediately start development of the application, if it seems feasible and competitive;
- follow accelerator development strategy as the fusion precursor;
- use FFHs for the materials development for advanced nuclear power including fusion;
- use FFHs for blanket technology development for pure fusion and hybrids;
- search for fusion neutron and ions applications in basic research.

These steps are used in the project "Controlled Fusion and Plasma Technologies for Atomic Energy" proposed by SC Rosatom for realization during 2019-2024 years and further till 2035, which tasks are presented in Table 1.

| Tasks 1-4 | Total/Construction |
|---|---|
| Research and development of enabling fusion technologies | 3.7/2.7 $B |
| Research and development of hybrid reactor technologies and systems | 4.7/3.9 $B |
| Developing innovative plasma technologies including experimental-industrial | 1.25/0.38 $B |
| Developing regulations in the fields of fusion and hybrid systems, assurance of licensing activity, exchange of scientific and technical information, staff education and training | 0.45/0.1 $B |

*Table 1 - Structure and costs of the FP*

## 2. FFHs Development Status and Prospects

Development priorities and technical requirements are to be defined for Facilities, Materials and Technologies of FFHs. Starting conditions for the Program realization should be carefully evaluated.
FFHs continue to remain at the conceptual design level. There is no one operating or being constructed Fusion-Fission facility/system in the world. Neutrons, radio-nuclides, high temperature heat and electricity are expected as final products for FFHS in perspective. Services on subcritical fission of heavy metals, participation in nuclear fuel cycle, generation of intense neutron fluxes with thermal and fast spectra are considered and developed.
Missions of FFHs in Russian Atomic Energy include the following: faster development of fusion technologies and materials; improvement of neutron production economy in AE; diversification of neutrons applications; development of technologies for HFC with high-level radioactive nuclear fuel and implementations of fusion technologies.
Drivers for FFHs development are as follow:

- high efficiency of neutrons production in fusion reactions: energy released per neutron is 3 times less than in accelerators and an order of magnitude is less than that in fission reactors;
- effective fission of any Heavy Metal by 14 MeV-neutrons under currently reached tokamak parameters;
- reduced requirements to materials and fusion power in FFHS compared to pure Fusion option;
- prospective demands of NFC in realization of subcritical fission and energy valuable level for Minor Actinides and for enrichment of spent nuclear with low level of fissile nuclides (fuel of BWR).

Basic requirements to FFHs: safety of nuclear facility to be controlled by RF Law on Use of Atomic Energy 1995 with amendments; SSO for enabling technologies >5000 hours.
Ranges of the performance specifications for FFHs are given in Table 2.
The milestones for FFHS development correspond to 3 levels of DT fusion power 3 (FNS-ST), 40 (ДЕМО-FNS), 500 (ITER) MW and related neutron yields $10^{18}$ n/s, $10^{19}$ n/s and $10^{20}$ n/s. The last one may be not needed for hybrids until materials for pure fusion conditions will not appear.
The neutron-yield of $10^{18}$ n/s allows testing materials and components; $10^{19}$ n/s – control of subcritical active cores of industrial power and transmutation of Minor Actinides; $10^{20}$ n/s – nuclear fuel breeding for thermal and fast reactors in symbiotic fusion fission system [3].
Operation life longer than 10 years is needed to reach 20 dpa and 200 appm for He. This nuclear damage open likely new effects in degradation of properties of structural and functional materials due to neutrons, gas generation (H, D, He), variation of chemical composition and structure, secondary nuclear reactions on artificial nuclides.
Major facilities on the path to Commercial Hybrid Plant were discussed in Ref. [3]. Last design activity confirms

| | | |
|---|---|---|
| Neutron yield of Fusion Source | $10^{15}$-$10^{21}$ | n/s |
| Fusion neutron flux | $10^{11}/10^{14}$ | n/(cm$^2$s) |
| Fission neutron flux | up to $10^{15}$ | n/(cm$^2$s) |
| DT-neutron loading | 0.2 | MW/m$^2$ |
| campaign duration | up to 1 | year |
| Operation life | >up to 20 | year |
| Fusion power multiplication factor Q | 0.1 – 1 | |
| DT-fusion power | $10^{-5}$ - 3 | GW |
| Fission Power | $10^{-4}$ - 3 | GW |
| Consumed electric power for 3 development steps | <20/60/200 | MW |
| Auxiliary heating power | <1/10/40 | MW |

*Table 2 - Performance specifications for FFHs*

the time and cost scales. Progress in DEMO-FNS development is presented in Ref. [4, 5].
A roadmap for development of FFHs and Fusion Power Plant has been proposed (Fig. 2). Major milestones include the following actions: design of test beds, construction of test beds for enabling technologies and materials, integration of SSO tokamak technology in Globus-3, construction of FNS-ST tokamak with 3 MW DT-fusion for attestation of materials and components, Hybrid technologies demonstration in DEMO-FNS opening applications in plasma, neutron, tritium, nuclear fuel cycle (NFC) and other technologies, the Nuclear Power Plant level of operation and commercial levels are expected by 2045 and 2055. It is significant that the size of the tokamak is not changed for DEMO-FNS, pilot PHF and commercial CHP. Technologies and materials developed during FFHs tasks realization will be useful for pure fusion and will support ITER activity on the track to DEMO.
List of principal R&Ds for FFHs includes: tokamak enabling technologies, tritium, remote handling, hybrid blanket, hybrid nuclear fuel cycle, molten salt technologies of continuous nuclide isolation. These technologies must satisfy nuclear regulation and licensing requirements, provide and improve nuclear safety of the facilities, be compatible with neutron environment and with each other, maintain operation campaigns with duration close to a year (5000 hours), ensure RAMI (reliability, availability, maintainability, inspectability).
The state-of-the-art tokamak enabling systems needed for steady state operation (SSO) have very different level of development. Those well-developed are: Electromagnetic system, Pumping, Cryogenics, Tritium, Control, Heat exchange, Heat conversion, Data acquisition. Systems and components, which are far from SSO, include: Remote handling, First Wall, Liquid metals, Divertor, Blanket, Auxiliary heating and current drive, Diagnostics. Critical technologies of Hybrid Nuclear Fuel Cycle include hybrid nuclear fuel cycle (HNFC) employing high-level nuclear fuels and waste, molten salt nuclear technologies, first-loop heat exchange.
Tritium complex of the tokamak impacts on design of Vacuum Vessel, In-Vessel components, Fist Wall, Divertor, Blanket, Heat exchange, Radio monitoring, Data acquisition.
Parameters of tritium complex for different neutron yield of FNS are as follow:

- for < $10^{18}$ n/s and duty factor = 0.3, 50 g/year T-consumption the purchased tritium is possible;
- for $10^{19}$ n/s - $10^{20}$ n/s and duty factor = 0.3 1-20 kg/year T-consumption, the tritium breeding is mandatory;
- for > $10^{20}$ n/s, duty factor ~ 1, 100 kg/year symbiotic operation of FNS with atomic Power Plants providing tritium is possible, if more than 15 kg/year were provided per 1 GW (e).

Critical technologies of tritium fuel cycle are recycling barriers against penetration through structural and functional materials, effective technologies of continuous and cyclical tritium extraction.
Remote handling affects general design of any hybrid facility complex including buildings, site plan and logistics, tokamak, tritium complex, divertor, blanket, hybrid fuel cycle, heat exchange, radio

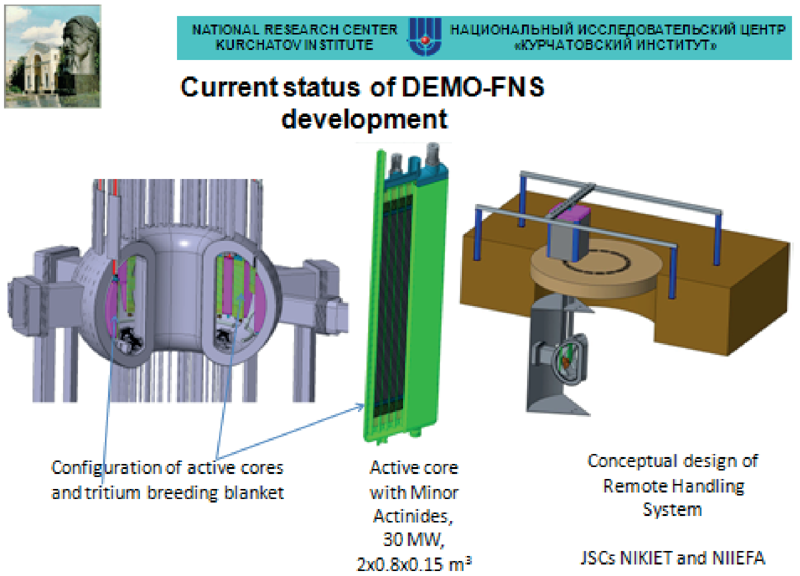


*Fig. 1 - Design of DEMO-FNS facility, subcritical active core and remote handling system*

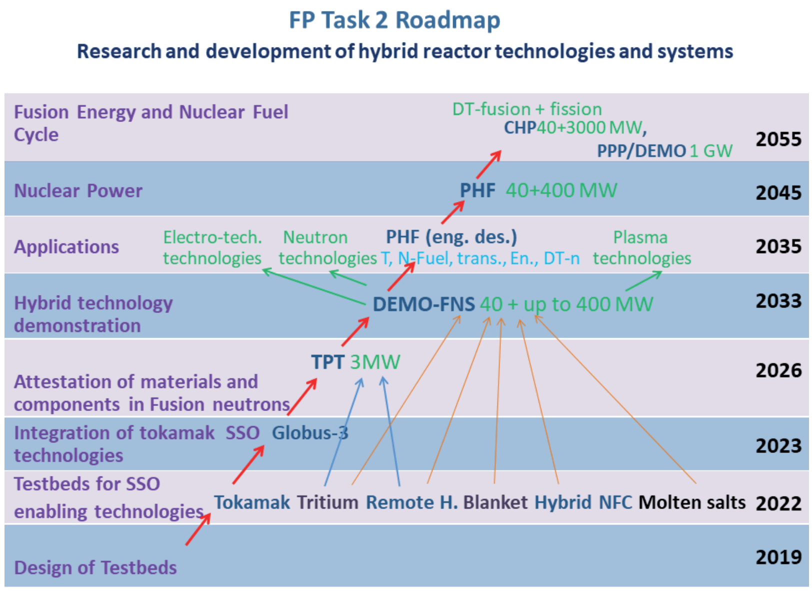


*Fig. 2 - Roadmap for FFHs development up to Commercial Hybrid Plant*

monitoring . Functions of remote handling include: assembling-disassembling of components with a few tons weight (ITER-scale technologies), repair and replacement of activated components maintenance of active cores, blanket modules and high-level fuel and waste, maintenance of the first-loop heat exchanger, in-vessel inspections, transportation of activated components including fuels and coolants. Critical elements of RH are transporters, manipulators, visual control devices. Remote handling must be defined at conceptual design of the facility complex.

Molten salt technologies can provide continuous processing of fuel mixture in FFHs. Russian AE is interested in:

- burning Minor Actinides and heavy metals.
- servicing spent nuclear fuel (SNF) and radioactive waste (RAW).

Objectives & challenges for molten salt technologies include: improved solubility for Th-U (well-known) and U-Pu (recently discovered in FLiNaK), Redox-potential control with accuracy of 5 mV, Heat exchanger inside the tokamak Vacuum Vessel, In-Vessel inspections, Transportation of high-level components, fuel and coolants.

Problems of MS-technologies include: loss of two safety barriers, pure fissile nuclides – «proliferation», Thallium corrosion (for Th-cycle), first-loop heat exchanger, high temperatures enhancing tritium penetration, economics of molten salts, separation of MA at 500-700 °C, residual heat, activation and transport of delayed neutrons through tokamak structures. For MS-technologies R&D cost was evaluated as ~1 B$ and 10 years for developing industrial applications [6].

## Conclusions

Federal project "Controlled Fusion, Hybrid &Plasma Technology" has been recently proposed and submitted for approval to Russian federal jurisdictions. FP tasks will develop Magnetic&Inertial confinement fusion, FFHs, FF enabling technologies and their applications, FNS, nuclear regulation, staff education and training. Development of Fusion and Fusion-Fission Hybrid Systems will require design and construction of facilities, materials and technologies of new generation. General requirements to FFHs correspond to fast fission reactors and systems of Generation 4: sustainable development, competitiveness at industrial level, safety, proliferation. Nuclear science requirements for FFHs are in an intermediate range between pure fusion and fast fission reactors. Successful development of Fusion-Fission Hybrids and technologies is capable to impact on and accelerate development of modern Fusion and Fission Energy.

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# Tokamaks as neutron sources for Fusion-Fission hybrid reactors: analysis of design parameters and technology readiness levels

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## Abstract

DThe aim of the paper is to review models of neutron sources, based on tokamaks, of fusion-fission hybrid (FFH) reactors, using a new scaling law specifically derived for fusion reactors. Starting from the evaluation of the status of knowledge on Q~1 tokamak neutron sources, the criteria for determining the parameters for a MCF (Magnetic Confinement Fusion) neutron source are summarized. Then a short review of the existing tokamak models already studied for FFH is carried out, comparing the performance predicted by those models with that evaluated using the new scaling law for fusion reactors. The technical readiness level of a Q~2 MCF neutron source is evaluated and an R&D line of research is identified based on this evaluation.

## Introduction

As is well known, the world has gained experience in building Q ~1 tokamak devices ( TFTR [1], JET [2], JT60U [3]) with pulse lengths of the order of 10s. Q is the fusion gain factor equal to the ratio between the fusion power and the input heating power (Q=Pfus/Pheating). Determining the parameters of a neutron source for a Fusion Fission application with fusion Q~2-3 based on the tokamak seems a relatively small extrapolation. The present paper starts from a revisited formulation of scaling laws for fusion reactors which includes the concept of Kadomtsev-Lackner similarity extended to fusion plasmas [10,11]. Taking as reference the parameters of a high performance JET discharge (Q~0.55-0.6) [23], the parameters of similar discharges are determined, and then the extrapolation to higher Q appropriate for FFH is attempted. Parameter sets are then determined for a variety of tokamak FFH neutron sources. Possible Figures of Merit for FFH are introduced. The Technical Readiness Level (TRL) of the required tokamak subsystems is presented and discussed, as a consequence of which the higher priority research and development research lines are outlined.
The structure of the rest of this paper is as follows:
Section 2 Existing experience in Q<1 tokamaks
Section 3 Basic requirements for a fusion neutron source for FFH
Section 4 Extension and application of the Kadomtsev-Lackner scaling law
Section 5 Compact tokamak inboard radial build design considerations
Section 6 Possible figures of merit for FFH devices
Section 7 FFH reactor TRL assessments
Section 8 Conclusions and suggestions for future work.
Section 9 References

## Existing experience in Q<1 tokamaks

Readers familiar with magnetic confinement fusion research will be aware that in 1994 the American tokamak TFTR achieved 10 MW of DT fusion power, largely (as always intended) from 100 keV high-energy deuterium ions from the neutral beam injection interacting with the plasma tritons [1]. This was followed in 1997 by JET in the EU producing 17 MW of DT power [2], corresponding to a power gain Q ~ 0.6. Both these results were transient, however, with the time above 90% of those powers respectively 0.8 and 1.2 seconds. Power gain is broadly proportional to the triple product $nT_i\tau$ and the non-nuclear Japanese tokamak JT60U has since improved slightly upon the JET result in terms of the triple product [3]. Presently ITER [4], is under construction in France and is expected to achieve Q≥5, while conceptual design activity is under way in the EU and other countries for a first electricity generating demonstrator, "DEMO" [5], which should have a Q~30.
It is interesting to note that in recent years there has been a burgeoning of interest in privately funded fusion research, such that searching the Internet will reveal about 20 private fusion companies (with widely varying financial support), two of them based on compact steady-state tokamak designs [6,7]. The drive for compactness is due to a desire for reduced capital cost and has also resulted in mainstream fusion institutes creating many designs for small tokamak power plants and neutron sources, [8], some of them pure fusion and some fusion-fission hybrids. In section 4 of this paper, the Q achieved or expected to be achieved in a selection of the machines mentioned in this paragraph is compared with that

predicted by a new energy confinement time scaling law described in that section.

We can say that the world has gained significant experience in the following areas:

- How to build and operate a pulsed tokamak (with short pulses of the order of 10s) Q<1 machine, heated with NBI (neutral beam injection) and RF (radiofrequency), ECRH (electron cyclotron resonance heating) and ICRH (ion cyclotron resonance heating) (~JET(EU))
- How to build a low temperature superconductor device pulsed (of the order of 100s) Q=1 machine, heated with NBI and RF (ECRH) (EAST (China) [12], TORE SUPRA(Fr) [13], JT60SA(JA-EU) [14])
- The MCF community is beginning to learn about High Temperature Superconductor magnets: this technology [6] will give access to high magnetic field fusion neutron sources

## Basic requirements for a fusion neutron source for FFH

A summary of the basic requirements for a low power neutron source useful for a Fusion-Fission hybrid is given in Table 1. As can be noted, the fusion power of the neutron source is relatively low, corresponding to a FFH total power (fusion+fission) of the order of 1 GW. With reference to Fig.3 (see also the discussion in section 4), a Q~2 tokamak can have the following parameters: i) major radius R0=2.3m, magnetic field on axis B=5T, aspect ratio A=3, (labelled as CTNS, 'Conventional Tokamak Neutron Source'); ii) major radius R0=1.5m, magnetic field on axis B=5T, aspect ratio A=1.8,( this will make the magnetic filed on the conductor quite large, labelled STNS, 'spherical tokamak neutron source'). These sets of parameters will be inserted in Table 2, to compare them with the existing FFH models.

It appears that the extrapolation of the technology already available for JET-TFTR devices (Q_Fusion ~ 0.5) to a FFH tokamak based neutron source with Q=2-3 is not that great, compared to the Q~ 5-10 needed for ITER.

| Q<br>Fusion<br>Gain factor | PDT (MW)<br>Deuterium-Tritium fusion power | Pheat<br>Power Heating<br>(MW) | $\beta_N$<br>Normalized beta | n/nGr<br>Greenwald fraction | Pdiv<br>Power flux to the divertor<br>$MW/m^2$ | Blanket<br>Material of the blanket | Pulse duration |
|---|---|---|---|---|---|---|---|
| 2-3 | 60-90 | 30 | <2.5 | <0.8 | <5 | Li+U-238 or Th-232 | >3 hr/steady state |

*Table 1 - Figures for a tokamak based neutron source useful for a Fusion-Fission hybrid reactor*

The real point (as developed in section 4) is related to the possibility of building a device which guarantees a quasi-continuous operation (long pulses or steady state) and a high reliability. This last point (high reliability) is connected to physics operation far from the instabilities which can cause disruptions or affect the neutron production. This means that the plasma operation must be realized far from the q, beta and density limits: in Table I such limits are identified as values of normalized beta βN<2.5 and Greenwald fraction (ratio betwen the plasma density n and the Greenwald density limit nGr) n/nGr<0.8.

The other important limit is the power flux density on the divertor which must be less than the damage limit of the presently available divertor materials, which could be put at a level of ~5 $MW/m^2$, with a plausible erosion rate of the divertor surfaces. The maximum heat flux depends upon the thermal conductivity of the bulk material and the thickness of this material between the plasma-facing surface and the coolant channel (because of the temperature gradient through the material caused by the heat flux), which has to be larger if there is considered to be higher erosion between replacements.

Any such nuclear tokamak must have a full remote handling capability for interventions inside the machine, and to be able to function properly with the plasma control achieved using the minimum possible number of sensors (radiation-tolerant diagnostic systems).

Table 2 shows an overview of the proposed models for a FFH tokamak based neutron source.

Spherical tokamaks with aspect ratio A=1.8 and tokamaks with A>2.7 are considered, with values of fusion gain 2≤Q≤5. These models are intended as prototypes for neutron sources; not all of them respect the ideal figures reported in Table 1. Two potential problems clearly arise in pushing the design of FFH machines to small size. These are the power fluxes to the first wall and divertor target plates, long recognised as a severe problem for ITER, DEMO etc., and the tendency to compensate reduced fusion power gain by increasing the $k_{eff}$ of the fusion-fission blanket. When keff is set very high but ostensibly maintaining a sub-critical assembly, the criticality (naturally unfamiliar to most pure fusion researchers) becomes very sensitive to the blanket distributions of neutron multiplying elements, both fissile (heavy metal and Lithium-7) and non-fissile, and to neutron poisons such as boron, gadolinium and Xenon-135.

These distributions necessarily vary with the degree of burn-up of the blanket. In addition, variations in neutron moderation and reflection back into the blanket due to extraneous assemblies introduced for maintenance activities are likely to become significant issues for the safety case. These potential problems would be more easily avoided if the maximum $k_{eff}$ is set to say 0.95 or lower, ensuring unconditional sub-criticality for all foreseeable plant conditions.

| | FDS-1 (CN) [15] | SABR (USA) [16] | CFNS (USA) [17] | ST135 (UK) [18] | FNS (RF) [19] | CTNS | STNS | JET (EU) | ITER |
|---|---|---|---|---|---|---|---|---|---|
| R(m) Major radius | 4 | 3.75 | 1.35 | 1.35 | 2.75 | 2.3 | 1.5 | 2.9 | 6.2 |
| A Aspect ratio | 4 | 3.3 | 1.8 | 1.8 | 2.75 | 3 | 1.8 | 3.1 | 3 |
| Pfus (MW) Fusion power | 150 | 500 | 100 | 200 | 30 | 60 | 60 | 16 | 400 |
| Q Fusion gain | 3 | 4.6 | 2 | 5 | 1 | 2 | 2 | 0.6 | 10 |
| B(T) | 6.1 | 5.6 | 2.8 | 3.7 | 5 | 5 | 5 | 3.4 | 5.3 |

*Table 2 - Parameters of FFH tokamak based neutron sources compared with JET and ITER*

## Extension and application of the Kadomtsev-Lackner scaling law

The possibility of determining the optimal parameters of future devices is linked to the scaling laws on the basis of the description of a plasma state. In fact the scaling laws for tokamak plasmas were introduced by Kadomtsev noting that the energy confinement should depend upon the dimensionless parameters:
A = major radius / minor radius = R/a
$\beta \sim nT/B^2$ = kinetic plasma pressure / magnetic pressure
$\rho^*$ = Ion Larmor radius / machine minor radius=$(MT)^{1/2}$ A /(R B)
$\nu^*$ = connection length / (trapped particle mean-free path) ~ $n\,R\,T^{-2}\,q\,A^{3/2}$.
q = safety factor ~ $R\,B\,A^{-2}\,I^{-1}\,k$
Where R= major radius, B=magnetic field, I=plasma current, M=ion mass, k=elongation and T=temperature.

Under this premise, devices with equal ($\beta$,$\nu^*$,$\rho^*$,q) at fixed geometry should exhibit the same confinement properties.This means that equivalent devices (plasmas with similar confinement properties) can be obtained by taking fixed the scaling parameter:

$$SK= R\,B^{4/5}\,A^{-3/2} \qquad (1)$$

For reactor plasmas (deuterium-tritium) the $\alpha$-particle power ($P\alpha$) must be introduced as an important contribution to plasma heating. In this case (the reactor plasma) $P\alpha$, the gain factor Q = Pfus/ Pin and the slowing down time of the alpha particles ($\tau_{SD}$) must be introduced as parameters defining the plasma state. In practice, we can define the following set of parameters as a basis for the definition of the scaling laws useful for fusion reactors:

1. Q=Q0 fixed
2. $\tau_{SD} = \Lambda_{SD}\,\tau_E$.($\Lambda_{SD} \leq 1$) (slowing down time of alpha particles ≤ energy confinement time. This is true for JET-DTE1, ITER, DEMO PPCS and EU-DEMO, Te≤20 keV); $\Lambda_{SD}$. is NOT a constant but depends upon the device.
3. $P\alpha = \Lambda_{LH}\,P_{LH}$ ($\Lambda_{LH} > 1.5$), the alpha heating is sufficient to keep the plasma in H-mode.
4. The energy confinement scaling law is ITER IPB98y2 and the scaling for the power threshold for the transition to the H-mode scaling $P_{LH} \approx Alh\,B\,n^{3/4}\,R^2$.

We find that the scaling parameter linking equivalent fusion plasmas is:

SFR =scaling parameter for fusion reactors
$= R\,B^{4/3}\,1/A\,Q0^{1/3} \qquad (2)$

Both scaling laws (1) and (2) give approximately the same weight to the magnetic field and aspect ratio. Fig. 1 shows the dependences of major radius vs aspect ratio of devices having Q0=0.55, similar to JET DTE1 (the first high power campaign with deuterium-tritium at JET).

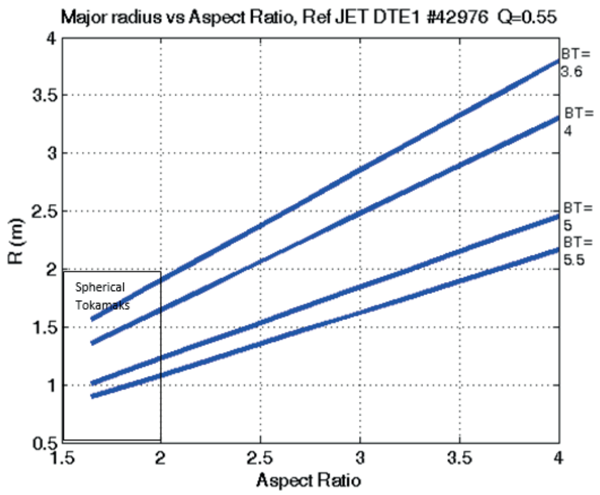


*Fig. 1 - Major radius vs aspect ratio parameters derived from the proposed scaling laws for fusion reactors (eq. 2), taking as reference the parameters of the JET DTE1 experiment*

From Fig.1 we can observe that at fixed magnetic field and Q0:
Q=0.55 is achieved at B=5.5 T for A=3 and R≈1.6m
Q=0.55 can be achieved at B=3.6T for A=1.7 and R=1.6m

Fig. 2 shows the same dependence (major radius vs aspect ratio) using the Kadomtsev-Lackner scaling laws: Q=0.55 is achieved at B=5.5T, for A=3 and R≈2.1m.

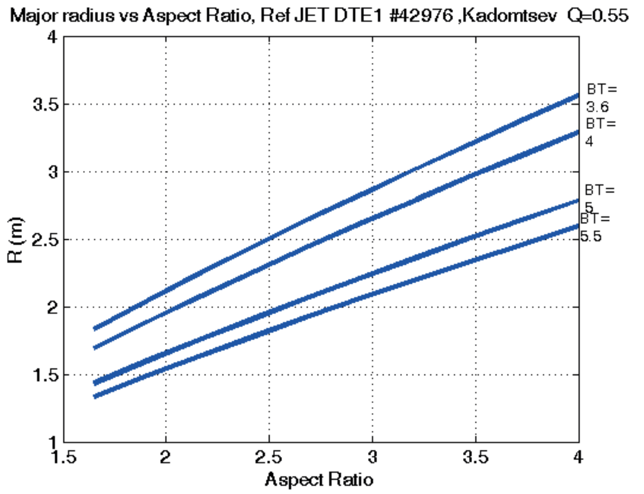


*Fig. 2 - Major radius vs aspect ratio parameters derived from the Kadomtsev-Lackner Scaling laws (eq. 1), taking as reference the parameters of the JET DTE1 experiment*

The stronger dependence upon the magnetic field contained in the new scaling law for fusion reactors (eq. 2) permits a reduction of the device dimensions for the same Q.

The question on how the models shown in Table II are located in a plot derived from the new scaling for fusion reactors (eq. 2) is addressed in Fig. 3: some of the devices proposed in Table II could be compatible with Q0=6 devices, using this scaling law.

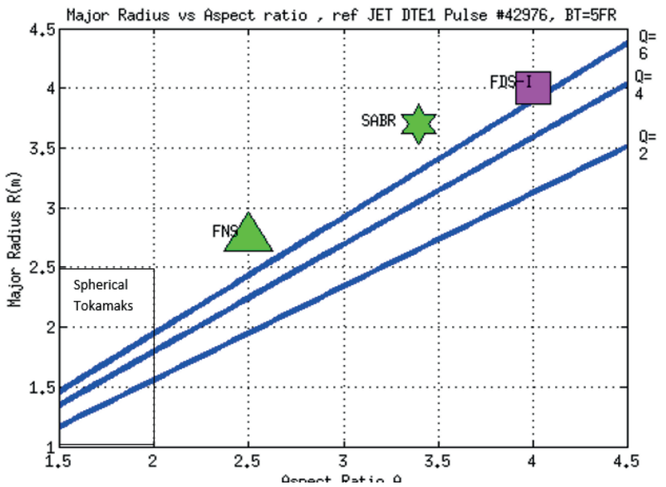


*Fig. 3 - Major radius vs aspect ratio at a fixed magnetic field B=5T, at gain factors Q = 2,4,6 as derived from the new scaling law for fusion reactors (eq. 2)*

## Compact tokamak inboard radial build design considerations

Typical pure fusion DEMO machines are designed with the following significant constraints:

- the need for an adequate triple product $nT_i\tau$
- the avoidance of demountable jointed toroidal field coils
- the peak magnetic field constraint of low-temperature superconductors (typically Nb3Sn)
- conventional operational boundaries in the tokamak physics

This leads to machines with large major radius (~9m) and because the central structure of the machine represents a large fraction of the first wall and hence the neutron load, the necessity for significant tritium breeding capability on the inboard side of the torus. The tritium breeding blanket designs with no heavy metal fission, solely "microfission" in lithium-7 and with neutron multiplication provided by beryllium, lead or bismuth etc., are generally found to require ~1m thickness to be effective for tritium production. The blankets do also provide several orders of magnitude attenuation of the first wall neutron flux so that little additional shielding is needed for the superconducting coils further out in the assembly. However it will be evident from geometry considerations that the minimum plasma aspect ratio of a machine intended to be "compact" would be heavily constrained if the inboard radial build had to include ~1m of blanket as well as a plasma boundary gap, vacuum vessel wall, thermal insulation and a central toroidal field coil conductor assembly carrying up to some tens of MA to ensure that the plasma safety factor q is high enough for stable plasma operation.
However one attraction of the "spherical tokamak" (ST) approach, as noted in [8], is that the total surface area of the centre-stack of an ST is a small percentage of the total first wall area. Thus it becomes feasible to provide only neutron shielding on the inboard wall, permitting tritium breeding modules to be located only in the outboard wall. Clearly introducing heavy metal fissile elements into the breeding blanket raises their neutron multiplication and hence tritium breeding efficiency as well as the power gain of the machine, the basic tenet of FFH machines, further facilitating the choice of only providing neutron shielding on the inboard wall.
But how much neutron shielding is required? Fortunately there have been several studies on this aspect of ST nuclear reactors, including [8, 9] and the conclusion is generally that with readily available elements the most efficient shielding for the neutron spectrum predicted in a DT reactor is a mixture of tungsten, carbon, boron and water (e.g. as tungsten carbide and boron carbide with ~10% water also functioning as a coolant) [20]. The tungsten scatters the neutrons and provides some absorption, the carbon and water are moderators and the boron (especially if placed towards the rear of the shield) is a strong absorber of thermalised neutrons. The attenuation factor per decimetre of this type of neutron shielding varies with design details and whether it is displacement damage, nuclear heating, or neutron flux in some energy band that is being considered, but can approach one decade per decimetre.
The radius of the central conductor has to be sufficient to carry the total toroidal field coil current, ITF, which roughly scales like ITF (MA)= $5R_0B_0 \sim 2qA^2I_p/(1+\kappa^2)$ where $R_0$ is the

plasma major radius (m), $B_0$ is the vacuum toroidal field at $R_0$ (Tesla), q is the magnetic field line pitch or safety factor just inside the plasma edge (usually a little above 2), A is the plasma aspect ratio, $I_p$ is the plasma current (MA) and κ is the plasma elongation (and has a somewhat stronger function than shown when significantly above ~2).

As will be evident from the approximate equation for ITF above, even in an ST the required TF current has to be at least comparable to the toroidal plasma current. Allowing a notional 0.1m for each of the plasma boundary gap, vacuum vessel double wall, and the thermal insulation between the neutron shield and the central conductor, this leaves ($R_0$(1-1/A) – 0.3m) for the sum of the neutron shield thickness tNS and the radius of the central conductor. Accordingly the current density in the central conductor (here meaning the electrical conductor, the cooling channels, any electrical insulation and strengthening structures and ignoring any small central void) is given by $j_{CC} = I_{TF} / \pi(R_0(1-1/A) - 0.3 - t_{NS})^2$. It is beyond the scope of this paper to demonstrate how this basically geometric constraint can be balanced against those of peak magnetic field, mechanical stresses, residual neutron damage rates and nuclear heating and (if a normal conductor) resistive heating in the TF central conductor assembly in order to achieve a self-consistent compact reactor design with an acceptable cryocooling power requirement. Examples can be found in [7,8,9] with some shielding concepts explored in [18]. Clearly if the intention is to demonstrate a type of nuclear tokamak with a relatively short full-power life between central column replacements, the shield thickness and mechanical structure of the central assembly can be significantly reduced, facilitating higher toroidal field in the plasma and/or lower aspect ratio. For instance, in [20,21] this approach results in a central high temperature superconductor assembly of 0.20m outer radius with a shield of 0.32m thickness, for an A=1.8 machine of R0 = 1.35m with Ip = 7.0 MA and B0 = 3.69T. This pure fusion machine was intended to have an energy confinement time 1.88 times that predicted by the usual ITER98Y2 scaling, with Q=5.0 (and P-fusion = 185 MW). Even if this factor was reduced to 1.0, the fusion power would be reduced by approximately its square, 3.53, permitting a Q-fusion ~1.4 which could suffice for a FFH.

## Possible figures of merit for FFH devices

The fusion community is divided on the merits of FFH devices, commonly citing the difficulties of introducing the fusion plasma physics, magnetic field structures and tritium fuel cycle issues to the well-understood environmental problems of even a small fission reactor which could in most applications achieve similar ends. If the application requires a large fraction of the neutron population to be above a particular transmutation threshold (generally above ~8 MeV) then having a 14 MeV DT fusion neutron source may be more attractive than a fission reactor (with its multiple-break-up neutron birth spectrum resembling a Maxwellian distribution of "temperature" ~2.5 MeV), but only if most of the neutrons are from the DT reaction. Spallation neutron sources can have much higher neutron energies but involve high energy particle accelerators, substantial size and investment cost and produce relatively small neutron fluxes. FFH designs intended to make full use of the 14 MeV neutrons for elemental transmutation not readily achievable in a fission reactor may have an economic advantage over spallation neutron sources for this type of application. In this context of competition, the operational ratios listed below are suggested as possible figures of merit to guide the justification for an FFH by consideration of the practicable and economic aspects regarding prospective fusion-fission hybrid concept devices.

- To breed a fissile isotope (or any specific isotope)

1) (The number of intended isotopes removed from the machine over its whole life) - (the whole life cost, from concept design costs to decommissioning and radwaste disposal)
2) (The mass of fissile elements removed from the machine over its whole life) - (the whole life cost described above)
This should be compared to centrifuging natural uranium or producing fissile material in a fast-neutron reactor
3) (The number of intended isotopes removed from the machine) - (the number of DT reactions in the same period)
NB some burn-up of the intended material will occur, leading to a peak or saturation in its inventory and the optimum breeding period will be somewhat smaller than that.

- To destroy waste isotopes

1a) (The number of target isotopes destroyed) - (the number of new radioactive isotopes created in the same period with half-lives >5 years, in the whole machine and its radioactive waste arisings)
1b) (The reduction in Bq of the target isotope) - (the increase in Bq of all other isotopes with half-lives >1 hour both remaining in and removed from the whole machine)
2) (The change over the whole life of the machine in the total Bq of isotopes with half-lives > 5 years both in and removed from it) - (the whole life cost described above)
(And hope the answer is negative!)

- To produce electrical power

(The electrical energy sent to the grid in the whole life of the machine) - (the electrical energy consumed by the machine in its whole life cycle, including construction, producing its fuel and all radioactive waste disposal)

- To produce tritium

The usual Tritium Breeding Ratio: (The number of tritons bred) - (the number of tritons consumed) Here both numbers should be integrated over the whole life. This expression treats the fissile material as just another neutron multiplier aiding the fusion fuel cycle.

- To produce neutrons

1) (The number of neutrons created anywhere in the

machine over its whole life) - (the whole life cost described above).
Some weighting to the desired neutron energy range may be required.
2) (The number of neutrons created in the machine and absorbed in the intended breeding elements, not the rest of the machine) - (the whole life cost described above).
The details such as the choice of one hour or five years for the different classes of undesirable isotopes could be varied, of course, but the intention is to consider the overall purpose of the fusion-fission hybrid solution compared to more conventional and/or fully established techniques, some of which are likely to meet the requirements with smaller environmental and safety concerns.

## FFH reactor TRL assessments

Here the following type of machine is in consideration:
1. Q~2-3 machine with long pulses (say > 3 hrs)/steady state, DT plasma $P_{DT}$~80-100 MW, $P_{in}$~30 MW
1.1. Low level of probability of disruptions: plasma parameters chosen to be away from strong MHD and density limits (for example with $\beta_N<2.5$, $n/n_{Gr}<0.8$)
2. Power on the divertor definitely lower than 5 MW/m$^2$: in this case the problem of the divertor is easier.
3. A blanket for tritium breeding with power gain and neutron multiplication from fission
4. A machine with high reliability, working continuously
5. All maintenance by remote handling
6. Modularity (facilitating rapid interventions on the divertor)
7. Few and simple diagnostics (the acceptable level of complexity of the diagnostics and controls depends on the plasma scenario and on the physics model).

The meanings of the different Technology Readiness Levels are as described in [5].
The Tables 3, 4 of the TRL for the main subsystems of a tokamak neutron source for FFH: it seems that only ECRH ( electron cyclotron resonant heating ) systems are in a certain level of engineering maturity for the insertion in a FFH, while the other main systems need to be demonstrated in a neutron flux environment.
Although the most important developments differ slightly, the steps in TRL are not fine enough to distinguish the 100 and 1000 second FFH concepts.
Table 5 shows the TRL for the plasma scenarios: here only the H-mode demonstrated on JET DTE1 at Q<1 can be considered for FFH reactor designs. The other scenarios need a demonstration at least at low power.

| Subsystem | TRL 100s | Comments |
|---|---|---|
| Superconducting magnets | 4 | Not demonstrated in a neutron flux environment. |
| NBI (100 keV) | 4 | Need to demonstrate immunity to gamma and neutron effects (e.g. grid flash-over due to the ionising radiation or grid insulation degeneration). |
| ECRH (1 MW gyrotron) | 6 | The gyrotrons are not in any radiation field and steady state operation has been demonstrated at the developer's works, for hours if not months, but only on test-beds. |
| ICRH (1 MW) | 4 | As NBI but for antenna operation; also parasitic currents may inject antenna material into the plasma. |

*Table 3 - Technology Readiness Level for prototype with 100-second pulses*

| Subsystem | TRL 1000s | Comments |
|---|---|---|
| Superconducting magnets | 4 | Not demonstrated in accumulated neutron fluence. |
| NBI (100keV) | 4 | Need to show long-term reliability and immunity to large neutron fluence (e.g. grid distortion). |
| ECRH (1 MW gyrotron) | 6 | As NBI but for antenna damage. |
| ICRH (1 MW) | 4 | As NBI but for antenna damage. |

*Table 4 - Technology Readiness Level for prototype with 1000-second pulses*

| Scenario | TRL | Comments |
|---|---|---|
| H-Mode | 6 | OK in JET at Q~0.6, needs demonstration at Q~2 |
| Hybrid mode | 4 | Needs demonstration in relevant Q>1 environment – possibly JET DTE2 |
| Advanced mode | 3 | To be demonstrated in a near steady-state machine |

*Table 5 - Technology Readiness Level for possible operational scenarios*

## Conclusions and suggestions for future work

An FFH machine would need only a modest improvement in Q-fusion over that of JET, preferably with a smaller plasma than JET to improve economic acceptability. In this paper a set of parameters for FFH tokamak based neutron sources are derived from a new scaling law for Fusion Reactors [10,11] which extends the Kadomtsev-Lackner similarity theory to fusion reactors, taking into account the fact that the alpha power must be introduced for defining the plasma state of a fusion reactor. An example of this evaluation is the set of parameters for a (compact tokamak) neutron souce at Q=0.55, which is: magnetic field on axis B=5.5 T, aspect ratio R0/a= major radius/minor radius= A=3 and major radius R0≈1.6m. Several compact Q~1 tokamak designs have been developed by different groups world-wide, e.g. as neutron sources for materials testing, and currently the High Temperature Superconductor (HTS) ST approach would appear to be most promising, whether for pure fusion or an FFH. HTS continues to be developed for non-fusion applications, steadily improving the attractiveness of this option for MCF devices. As indicated in the section on Technology Readiness Levels (TRLs), further devlopment is still required to achieve reliable high efficiency steady state NBI, divertor and first wall modules, equilibrium control of steady-state tokamak plasma at high elongation, and long-lived diagnostics for control of the nuclear plasma.

The inboard radial build of any compact tokamak reactor needs to accommodate an affordable central toroidal field conductor and adequate neutron shielding for the central conductor lifetime between replacements (if any), in many designs leaving adequate tritium breeding to be obtained only from the outboard side of the plasma. If the coils are cryocooled, whether normal or superconducting, the electricity consumed by the cryoplant needs to be included in the recycled power of the installation. Care should be taken in evaluating the usefulness of FFH machine proposals (perhaps guided by the creation of suitable figures of merit) to account for whole life costs (design work through to decommissioning and radwaste disposal) and the production of the desired output (power or special isotopes of some kind) weighed against the production of undesired radioactive waste created from previously benign isotopes in the blanket and the other parts of the machine exposed to the neutrons.

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# Radiation Damage Studies on the First Wall Material of Fusion Reactors

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## Abstract

Different types of candidate structural materials have been developed and characterized for fusion energy reactors. Among them, steels (austenitic stainless steels and ferritic/martensitic steels), vanadium alloys, refractory metals and alloys (niobium alloys, tantalum alloys, chromium and chromium alloys, molybdenum alloys, tungsten and tungsten alloys), and composites (SiCf/SiC and CFC composites) have primary importance. Steels have unique advantages with respect to extensive technological data base and significantly lower cost compared to other candidates. Furthermore ferritic steels and modified austenitic stainless (Ni and Mo free) have relatively low residual radioactivity. However, steels cannot withstand high neutron wall loads to build a competitive fusion reactor. Some refractory metals and alloys (niobium alloys, tantalum alloys, molybdenum alloys, tungsten and tungsten alloys) can withstand high neutron wall loads. But, in addition to their very limited technological data base, they have serious disadvantages due to the high residual radioactivity and prohibitively high production costs.

SiCf/SiC composite as structural material in a fusion reactor is attractive based on its low induced radioactivity, low afterheat, high temperature properties and excellent corrosion resistance. However, the improvement of both thermal conductivity and stability of thermo-mechanical properties after irradiation remain the main issue of SiCf/SiC research and development. Also they are limited with low neutron wall loads despite high temperature resistance up to 1000 °C.

Innovative concepts with a protective liquid wall inside the fusion plasma chamber can unify several advantages, namely ①achieving very high neutron load values, ② along with low maintenance costs due to the largely extended lifetime of the first wall structure (the most sensitive and very expensive component of a fusion reactor), using ③ low cost steels structures, ④ based on wide technological data base, and ⑤ with a low residual radioactivity.

## Introduction

The growing world energy consumption represents one of the major challenges of the 21st century with respect to economy and environment. At present, ~90% of world energy is supplied by fossil fuels. However, logistical problems, such as fuel transport and distribution, and environmental problems, such as particulate pollution and excessive $CO_2$ in the atmosphere, could limit the growth of fossil energy. Nuclear energy production can be considered as an important alternative to relax the chemical and thermal pollution of the environment.

As a preliminary goal of the energy strategy in a technologically advanced country, a nuclear energy production ratio of 50% of the total electricity could be defined. It is worth to comment here that the nuclear program in a country serves much more than just providing electricity to industry and the general public. It opens a way for the country to make large advances in many fields, including design engineering, manufacture, construction, and project management. In addition to that the promotion of R&D aimed at the advancement of nuclear technology will greatly help in the formation of intellectual assets for society in the 21st century, not only in the nuclear field but also in science and technology in general. Deep space research and future colonization of the solar system cannot be considered without a widespread use of nuclear energy.

At present, ~ 17% of world nuclear electricity is produced by light water reactors (LWRs), which require low enriched (3-4%) nuclear fuel. This type of an energy strategy will lead soon or late to a very serious bottleneck in the provision of the nuclear fissile fuel. The very long doubling time of a fast breeder (10-30 years) seems not very promising in supplying the growing world energy needs.

Controlled fusion energy appears to have potential in providing unlimited energy for mankind. A fusion energy system has attributes of an attractive product with respect to safety and environmental advantages compared to other energy sources [1] and it has clear safety and environmental advantages over fission energy. Fusion fuels are abundantly available in the nature, contrary to relatively scarce fission fuel resources. Hence, growing efforts have been invested in fusion energy research in the past 40 years.

Selection of structural materials plays a key role in enhancing the economic competitiveness of fusion reactors. Structural materials for fusion reactors are subjected to thermal, mechanical, chemical and radiation loads. A selection study for candidate materials may be extrapolated based on the experiences gained from fission reactors only to a very limited degree. The expected conventional loads appear higher for economically competitive fusion reactors. This includes ① higher operating temperatures, ② chemically aggressive coolants as energy carrier, such as molten salts, liquid lithium metal or eutectic lithium-lead, lithium-tin, and ③ furthermore magneto-hydro-dynamic effects. In addition to that, nuclear radiation loads for fusion reactors differ greatly from fission reactors. The latter are subjected to fission neutron flux with an average energy ~ 2 MeV and to gamma-ray radiation. In a fusion reactor, first wall around the fusion chamber must withstand to high energetic charged particle fluxes, Bremsstrahlung and gamma-ray radiation, and most importantly to unconventionally high energetic intense neutron fluxes with a mean energy ~ 14 MeV. The latter are expected to lead to much higher material damage than observed by fission reactors, not only due to higher neutron kinetic energy, but also, and even more important due to detrimental threshold reactions for structural materials in MeV range. *Any maintenance and repair work on fusion chamber first wall will cause a long-term plant shutdown and will be very costly.* Hence, a selection study for structural fusion reactor materials must be conducted under consideration of various unconventional aspects.

## Competitiveness of fusion reactors

Fusion reactors must be economically competitive for a successful energy market penetration. A commercially competitive power plant with low cost of electricity (COE) requires high power density (HPD), high power conversion efficiency (> 40%), high availability (lower failure rate, faster maintenance) and finally simpler technological and material constraints. These represent primary goals for fusion power technology (FPT) [1,2]. FPT is concerned with all components in the immediate exterior of the plasma, commonly called 'in-vessel system', which include first wall, divertor, blanket, and the vacuum boundary. The two most important requirements for obtaining practical HPD systems are:

(a) High power production per unit volume of the plasma;
(b) FPT in-vessel components that can handle the high surface heat flux and high neutron wall load (NWL) on the first wall in such HPD systems [1]. In that context, the neutron flux load on the first wall becomes a key issue.

For a breakthrough into the energy markets, fusion reactors must compete primarily with fission reactors. In an extensive analysis, Abdou and the APEX team have evaluated the key elements for a competitive fusion power system [1,2], which is briefly summarized below.

The average core power density for a fusion power reactor based on the "ITER-type" traditional tokamak is compared with the average core power density in MW/m$^3$ in several types of fusion reactors: Pressurized Water Reactor (PWR), Boiling Water Reactor (BWR), High-Temperature Gas-Cooled Reactor (HTGR), and Liquid-Metal Fast Breeder Reactor (LMFBR).

One can see that the average core power density in a fission reactor is higher than in an International Thermonuclear Experimental Reactor (ITER) [3] type reactor by a factor of ~ 80, 7.5, and 200 for PWR, HTGR, and LMFBR, respectively. If fusion reactors are to achieve the same average power density, the NWL will need to be in the range 22-600 MW/m$^2$. Such high wall loads may be impossible to achieve and handle in current magnetic fusion concepts.

On the other hand, fusion has clear safety and environmental advantages over fission. Therefore, fusion can be expected to be acceptable at a somewhat higher cost than fission. Fusion research should set a preliminary goal for the NWL to be greater than 10 MW/m$^2$ in order to enhance the potential of economic competitiveness for fusion power systems. An overview of current design concepts can be summarized as follows:

Next European Torus (NET) is planned for an average wall loading in the 0.5 to 0.7 MW/m$^2$ range and the fusion power between 400 to 600 MW [4]. The ITER power plant design is pursued with a modest average NWL of 1 MW/m$^2$, a fusion power of 1100 MW in physics phase and 860 MW in technology phase [5]. The wall load in the high-magnetic-field ARIES-I tokamak reactor study is 2.5 MW/m$^2$ by a fusion power of 1925 MW [6,7]. TITAN-I high-power-density reverse-field pinch reactor is supposed to operate with a competitive NWL of 18 MW/m$^2$ by a fusion power of 2300 MW [6].

For a (D,T)-, (D,D)- and (D,He-3) reactor, the optimal operating plasma temperatures are around 15, 20 and 55 keV, respectively. Lower optimal temperature is another significant factor for an earlier commercialization prospect of a (D,T) reactor.

Highly energetic charged particles allow one to use direct converters for the production of electricity with a higher efficiency than a heat engine. This situation becomes extremely favorable in a (D,He-3) fusion reactor because all the major reaction products will be charged particles. Neutrons will be produced only through same side reactions. In fact, none of the fusion fuel cycles will be absolutely free of neutrons, shown in table A1 of refs. [8] and table 1 [9]. In the primary fusion reactions, the majority of advanced fuel cycles may yield charged particles only, but there will be always at least one secondary or side reaction with neutron production.

The residual radioactivity in the structure of a fusion reactor results from the interaction of neutrons with the material. A neutron poor fusion reactor will have also lower residual radioactive contamination for a given energy output.

## Material damage under neutron irradiation

All metals have a crystalline microstructure. Crystalline materials have their atoms arranged in a well-defined lattice where each atom has a fixed rest position. Ideally, there should be no imperfections in the atomic arrangement. However, imperfections exist due to presence of impurities and alloying elements. Further irregularities can be introduced by plastic deformation. Any alteration in the regularity of the lattice will alter the properties of a material.
Material damage types under neutron irradiation can be classified in two groups:

*Microscopic radiation damage effects*

- Atomic Displacement under Neutron Irradiation (DPA): The displacement of an atom from its lattice position results from transferring the threshold energy, typically of the order of few dozens of electron volts, to the target. Atomic displacements are the fundamental process of radiation damage in metals.
In fusion blankets, displacements of the atoms from their lattice sites as a result of collisions with highly energetic fusion neutrons will be a damage mechanism for structural materials at much higher levels than in conventional fission reactors.
- Gas production "(n,p); (n,d); (n,t); (n,$\alpha$)" : In fusion blankets, another very serious damage mechanism for structural materials will be gas production, mainly through (n,p) and (n,$\alpha$) and to some extent through (n,d) and (n,t) reactions above a certain threshold energy. Materials suffer from embrittlement due to gas bubble formation.
- Nuclear transmutation: Foreign atoms production.
- Micro Melting: Local Formation of Hard, Brittle Martensite!
- Ionization Effects of Gamma Rays and Charged Particles.

*Macroscopic radiation damage effects*

- High-Temperature Embrittlement: Dimensional Changes; Swelling and Irradiation Creep. High temperature embrittlement is mainly caused by the nucleation and growth of bubbles filled with (n,$\alpha$)-produced helium.

The maximum operating temperatures for the materials with low residual radioactivity are:

- 550 °C for ferritic steel,
- 700 °C for ODS, V-Cr-Ti,
- 1000 °C for SiC-SiC composites.
- 1100 °C for Nb1Zr,
- 1200 °C for TZM,
- 1300 °C for T-111,
- 1500 °C for tungsten.

- Low-Temperature Embrittlement: Low-temperature embrittlement is due to hardening by radiation-induced defect agglomerates that act as obstacles for dislocation movement. This leads to an increase in the yield stress and, particularly in body-centered cubic alloys, to a shift in the "Ductile-Brittle Transition Temperature" (DBTT).

Table 1 shows the temperature range of the main macroscopic radiation damage effects according to the melting temperatures $T_M$ of the structural material.

- Degradation of Material Properties

| Effect | Temperature | Important for |
|---|---|---|
| Segregation and changes in precipitation structure | $T > 0.2\ T_M$ | Corrosion, weldability |
| Increase of DBTT | $0.1\ T_M < T < 0.3 T_M$ | BCC steels and refractory alloys for pressure vessels |
| Irradiation creep under mechanical load | $0.2\ T_M < T < 0.4\ T_M$ | Most nuclear materials |
| Irradiation growth | $0.1\ T_M < T < 0.3\ T_M$ | Non-cubic materials (Zr and its alloys, U, graphite) |
| Void swelling | $0.3\ T_M < T < 0.5\ T_M$ | Austenitic steels |
| Helium high temperature embrittlement under creep and fatigue loads | $T > 0.45\ T_M$ | First wall structures |

*Table 1 - Temperature range of the main macroscopic radiation damage effects*

Upper temperature limit for structural materials in fusion reactors can be controlled by four different mechanisms in addition to safety considerations:

- Thermal creep,
- High temperature helium embrittlement,
- Void swelling,
- Compatibility/corrosion issues.

## Structural materials

Principal candidate materials for fusion reactor first wall and other structures with low residual radioactivity are listed as follows:

STEELS
- Austenitic Stainless Steels
- Ferritic/Martensitic steels

VANADIUM ALLOYS

REFRACTORY METALS AND ALLOYS
- Niobium Alloys
- Tantalum alloys
- Chromium and Chromium Alloys
- Molybdenum Alloys
- Tungsten and Tungsten Alloys

COMPOSITES
- $SiC_f$/SiC Composites
- Carbon Fiber Reinforced Carbon Composites (CFC)

Among the candidate materials, steels seem to be the primary structural material for fusion energy reactors due to their unique large technological database and experience. Ferritic steels and modified austenitic steels containing Mn instead of Ni and W and/or V and Ta instead of Mo reveal relatively low residual radioactivity giving clear environmental advantages. The maximum operating temperature of ferritic steels can be increased up to 700 °C by using oxide dispersoids. On the other hand, steels have a very low neutron wall load limits causing low plant efficiency.
Vanadium alloys are attractive for fusion applications because of their low activation and good mechanical and thermal properties. They provide safety and environmental advantages associated with low activation characteristics, high temperature properties, and low-decay heat-generation rate that are resistant to irradiation-induced swelling and embrittlement over a wide temperature range. A V-4Cr-4Ti alloy appears to be near optimum composition, although further development and optimization is required to evaluate effects of nonmetallic elements and other alloying additions on the properties. Vanadium alloys are readily fabricable, they can be welded and operate at high temperatures and accommodate high-surface fluxes. The primary issues for vanadium alloys that require further research involve effects of high helium concentrations on the properties of neutron-irradiated alloys, effects of nonmetallic element concentrations on properties, their fatigue properties, operating limits and weld development including the effects of irradiation on weld elements. Moreover, electrically insulating coatings for lithium-cooled systems also require further development.
Refractory metals and alloys offer much higher operating temperatures and higher NWL capabilities than the low activation materials, namely, ferritic steels, vanadium alloys and $SiC_f/SiC$ composites. Therefore, niobium, tantalum, chromium, molybdenum and tungsten alloys with pure chromium and pure tungsten are considered as potential candidates for high performance in fusion reactors. However, they do not satisfy the 'low activation' criteria except chromium and some chromium alloys and available database for their irradiation properties and lifetime is very scarce. They need much more research and development facilities to supply enough databases to use in fusion reactors.
The use of low activation $SiC_f/SiC$ composite as structural material in a fusion reactor is attractive based on its low induced radioactivity, low afterheat, high temperature properties and excellent corrosion resistance. The improvement of both thermal conductivity and stability of thermo-mechanical properties after irradiation remain the main issue of $SiC_f/SiC$ research and development. The constant progress in fiber quality which led to the fabrication of almost stoichiometric fibers is a good premise for reducing such concern. However, extensive research efforts are needed to develope the matrix-fiber interface and matrix processing itself in order to reduce the differences in performances with respect to the bulk CVD SiC. This objective may probably be achieved by using processing parameters. Joining and coating techniques and hermeticity need further developments to study their compatibility with fusion environments and to improve their performance. Furthermore, lifetime of advanced $SiC_f/SiC$ composites that requires the necessary database has to be determined.
Carbon fiber composite (CFC) is another composite considered for fusion reactors. It has a very limited data base so that it needs more research and development for qualification for fusion reactors.
At presence, the lack of intense 14 MeV fusion neutron source is the main handicap in determining materials behavior in realistic fusion reactor environment. To get realistic data about the irradiation behavior of structural materials, development of high intensity source of 14 MeV neutrons remains as a crucial issue.

## Conclusions

Neutron transport calculations with MCNP6 code have lead to the following conclusions:

*Performance: Tritium breeding and energy multiplication*
- Coolants with Li have higher tritium breeding performance than Li containing ceramics.
- Coolant selection is the key element for the neutronic performance. TBR and M are highest with Li, followed by LiPb and FLIBE.
- First wall materials ALSO affect neutronic performance. Highest TBR and M values are obtained for V, lowest TBR with W and lowest M with Mo.
- Sandwich structure of Li coolant and graphite reflector increases TBR for LiPb and liquid Li.
- Material damage increases for hybrid mode due to the neutron multiplication in the fissionable component.

*Damage on the first wall: First wall material selection is the key element for material damage*
- DPA is highest by V, lowest by W.
- He production is high for steels, lowest by W.
- Coolant type has significant effects on material damage.
- DPA is highest with LiPb, lowest with liquid Li.
- He and H productions are identical and lower with LiPb and liquid Li, higher with FLIBE. They are high for steels and negligible for W.
- Li and FLIBE coolant thicknesses don't affect the material damage. DPA increases with LiPb thickness, but gas production remains unchanged.

High intensity fusion neutron sources are needed to have reliable data on material damage limits for reactor design.
ITER operating in pulsed mode with a first wall load of 0.5 to 1 MW/m$^2$ will have only limited capability of material testing experiments for commercial fusion reactors.

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# Compact, high field spherical tokamaks for non-electrical fusion applications

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## Abstract

Tokamak Energy is a privately funded UK company aiming to accelerate the development and commercialisation of nuclear fusion by combining two emerging technologies: spherical tokamaks (STs) and magnets made from high temperature superconductors (HTS). As part of the development process Tokamak Energy are designing and constructing a series of devices, some of which will have potential non-electrical applications.

The currently operational device in this series – ST40 – is a high field, compact ST with main parameters: major radius, $R_0$=0.4-0.6m; aspect ratio, A=1.7-2.0; elongation, $\kappa$=2.5; on-axis toroidal field, $B_T$=3T; and plasma current, $I_p$=2MA. It will have pulse lengths of 1-2s and 2 MW of neutral beam heating, with a further 2 MW of RF heating under consideration. For start-up, ST40 uses a solenoid free method called merging-compression (MC) [1], pioneered on START and MAST and currently under investigation at the University of Tokyo [2]. Confinement studies on MAST [3] and NSTX [4] have indicated that in STs confinement times display a favourable trend with decreasing collisionality. When fully operation, ST40 will be able to access low collisionalities and extend the ST confinement database. ST40 will operate with DD and potentially DT, so will be a prototype intense compact neutron source. At densities of $10^{20}$ m$^{-3}$ and temperatures of 10keV, the neutron production rate is of the order $10^{15}$n/s in DD and $10^{17}$n/s in DT.

Tokamak Energy are also designing a next step device, ST-F1, that will use HTS magnets, be able to operate in steady-state and produce fusion power on a scale needed to prove the high field ST concept.

This paper will:

1) present the results from the first phase of ST40 operations that took place in early 2018

2) outline performance predictions for future ST40 operations

3) describe the potential non-electrical applications of a next step device.

## Keywords

Fusion Neutron Sources, Spherical Tokamak, High Temperature Superconductors, Compact Fusion.

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# RAMI analysis of China test blanket modules for ITER

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## Abstract

In China, two kinds of test blanket module concepts were developed for ITER. One is the Helium Cooled Ceramic Breeder (HCCB) TBM, which uses. Li4SiO4 pebble bed as tritium breeder, Beryllium pebble beds as neutron multiplier and the Reduced Activation Ferritic/Martensitic (RAFM) steel as structural material. The other one is DFLL (dual-function Lithium Lead)-TBM, which has two types design: the Dual-cooled Lithium Lead (DLL) blanket and the Quais-Static Lithium Lead (SLL) Blanket. The DLL blanket is a dual-cooled lithium lead (Pbli) breeder system with helium gas to cool the first wall and main structure and Pbli eutectic to be self-cooled. The SLL Blanket is designed to use quasi-static Pbli flow instead of fast moving Pbli with the similar structure as of the DLL module. The CLAM (China Low Activation Martensitic) steel is selected as the structural material. The. TBM and its associated ancillary systems are called TBS.

The RAMI (Reliability, Availability, Maintainability, Inspectability) analysis was adopted in order to verify whether the design of those two TBMs and auxiliary systems meet the availability objectives of ITER, which was done by a reliability and probabilistic safety assessment program named RiskA developed by the FDS Team, China. The functional breakdown was conducted according to those two TBS conceptual designs. The DFLL TBS was divided into 3 main functions and 72 basic functions with 1 support function. The HCCB TBS was divided into 3 main functions and 50 basic functions. The relationships between these functions were described by IDEF0 method. Then the reliability models for each function for those two TBSs were established with RBD (reliability Block Diagrams) method. The availability of each function was calculated based on the published reliability data from FCFR-DB, Riskbase etc. The analysis result was used to support the design optimization. FMECA (Failure Mode, Effects and Criticality Analysis) was performed on those two TBSs. A critical list of all the possible function failure modes was established and quantitatively evaluated by the quantifying the severity of the effects and the occurrence of the causes. The comparisons of those 2 TBMs were discussed in this paper and several issues of concern for these two TBMS are also proposed. With the RAMI analysis, some mitigation actions were given to remission the unacceptable failure mode with high critical. The work also provided some reference for improving the availability of TBM.

*Session 2*

# Gas Dynamic Traps (GDT)

# Physical processes in the expander and their effect on energy confinement in a GDT-based neutron source

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Presented paper is the next step in the research of axial transport in Gas Dynamic Trap. Experiments dedicated to the neutral gas role in the expander of mirror device were carried out. Ion current density distribution measured at the end plate does not depend on neutral gas density in the expander. Experimental indications of neutral gas extrusion from the axis of the expander to its periphery were observed. Numerical model describing such extrusion by elastic collisions of neutrals with plasma ions is in agreement with experimental data.

## Introduction

An important feature of open magnetic traps is an existence of direct contact of hot plasma along magnetic field lines with cold surface of plasma absorbers, which inevitably must be placed in area with expanding magnetic field beyond magnetic mirrors. This is the reason why we should investigate physical mechanisms defining energy transport along magnetic field lines and build theoretical and numerical models, which can get reliable extrapolations of longitudinal energy and particle fluxes in reactor-like machines.

The major attention should be payed on phenomena, taking place in magnetic expander of the trap (region between magnetic mirror and the surface of plasma absorber).

Even in case of no magnetic expansion, electron heat flow in collisionless plasma is limited by ambipolar potential barrier that appears near the surface of plasma absorber and reflects the majority of electrons. There is a danger, that in real thermonuclear plasma potential drop in Debye sheath near the wall will be higher than the threshold of appearing unipolar arc, and when arcs emerge, potential drop possibly disappears. Secondary electrons in expanding magnetic field will be partly reflected by magnetic mirror back to the wall, and it is possible to decrease electron flow much more by increasing mirror ratio. The theoretic limit for longitudinal losses is close to $8T_e$ for every electron-ion couple, leaving the trap.

This simple model can be improved while considering electron scattering in the volume of the expander. Indeed, secondary electrons that cannot penetrate to the mirror is confined as in an adiabatic trap. Population of trapped electrons can be formed in the expander in the presence of weak scattering. In this case an ambipolar field does not concentrate in Debye sheath but distributes in the volume of the expander [1]. This field gives a possibility to avoid unipolar arcs on absorbers in plasma with fusion temperatures. According to the theory, if mirror ratio for magnetic field in the expander exceeds 40 (for hydrogen plasma), the majority of secondary electrons cannot penetrate to the mirror throat and electron heat flow saturates on the level of theoretic limit $8T_e$ per electron-ion couple.

All these regimes were predicted theoretically [1,2] and realized in the experiments with conditions close to collisionless regime at Gas Dynamic Trap (GDT) in Budker Institute of Nuclear Physics [3,4]. Those experimental researches showed that there is a population of cold electrons confined in the expander by the effective Yushmanov's potential. In case of magnetic field expansion (magnetic field in the mirror related to magnetic field on the end plate) K > 40, potential drop in the Debye layer at the plasma collector and energy of confined electrons are much lower than $T_e$ in the center of the magnetic trap. Also, at K ≈ 40 it's possible to achieve stable plasma confinement with high electron temperature (about 0.7 keV).

The theory implies the plasma flow into the expander is close to collisionless. This imposes stringent restrictions on the vacuum conditions in the expander. It is not clear what level of residual gas we can afford and what happens when there are significant number of neutrals. It seems quite possible that residual gas will be ionized thereby the population of trapped electrons will increase and begin to affect significantly on the plasma in the trap. It is also obvious that it will be very difficult to satisfy the requirements of high vacuum conditions in the expander of the operating fusion reactor.

Influence of neutral gas on processes in GDT expander is the key issue of this paper.

## Neutral gas in the expander

Experiments described in the paper were carried out at GDT, which is an axially symmetric magnetic mirror machine [5]. The main part of the GDT device is a 7 m long solenoid, with a magnetic field at the midplane up to 0.35 T and a mirror ratio R = 35. The GDT facility is intended for the confinement of plasmas with two ion components. One component is deuterium plasma with an isotropic Maxwell velocity distribution. This plasma has electron and ion temperatures of up to 250 eV and a density of ~ $1\text{–}3 \cdot 10^{19}$ $m^{-3}$ and is confined in a gas dynamic mode. Confinement of such plasma in the GDT is similar to that of a gas in a vessel with a small hole. The particle lifetime in the GDT is about $\tau_{||} = L \cdot R/V_i$, where L is the trap length, R is the mirror ratio, and $V_i$ is the ion thermal velocity. Another component consists of fast deuterons with an average energy of ~ 10 keV and density up to $5 \cdot 10^{19}$ $m^{-3}$ and is produced by intense deuterium neutral beam injection (NBI) of 5 ms duration, 22–25 keV particles energy and 5 MW power. This component is confined in adiabatic mode.

The theory mentioned above implies the plasma flow into the expander is close to collisionless. This imposes stringent restrictions on the vacuum conditions in the expander. It is not clear what level of residual gas we can afford and what happens when there are significant number of neutrals. It seems quite possible that residual gas will be ionized, thereby the population of trapped electrons will be increased and it will begin to affect significantly on the plasma in the trap. It is also obvious that it will be very difficult to satisfy the requirements of high vacuum conditions in the expander of the operating fusion reactor.

Simple estimations based on analysis of elementary processes taking place in plasma show that in the region of GDT near the mirror (K = 10, plasma diameter 15 cm, n = $10^{12}$ $cm^{-3}$) neutrals should be ionized with probability close to the unity. Therefore, ion current to the end plate should increase essentially and we can register it directly.

It's possible to make an upper-bound estimate: if every gas molecule gives an electron to the plasma, and current of these "cold" electrons becomes equal to the ion current from the trap, the situation should be very unfavorable for plasma confinement. Using such estimation the critical gas density appears to be $n_{crit} = 10^{12}$ $cm^{-3}$.

However, main plasma parameters such as electron temperature and neutron yield remain constant in much wider range – up to n = $10^{14}$ $cm^{-3}$ (Fig. 1). Fast ions energy content is constant as well in this range.

Measured by gauge head PMM46 gas density (hydrogen was puffed) in the expander is hundred times higher than upper-bound estimate, but there is no degradation of plasma confinement. To find out the mechanism of such behavior we used six ion current probes (three electrodes, collector biased by –1600 V) mounted radially on the end plate (Fig. 2).

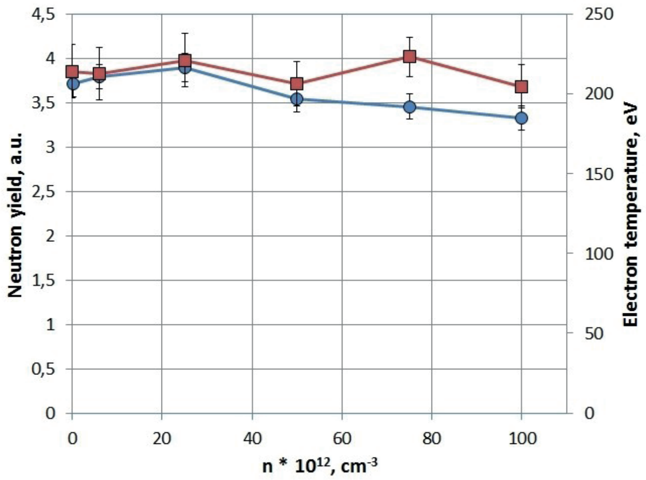


Fig. 1- Neutron yield (circles) and electron temperature in the central cell (squares) on neutral gas density in the expander

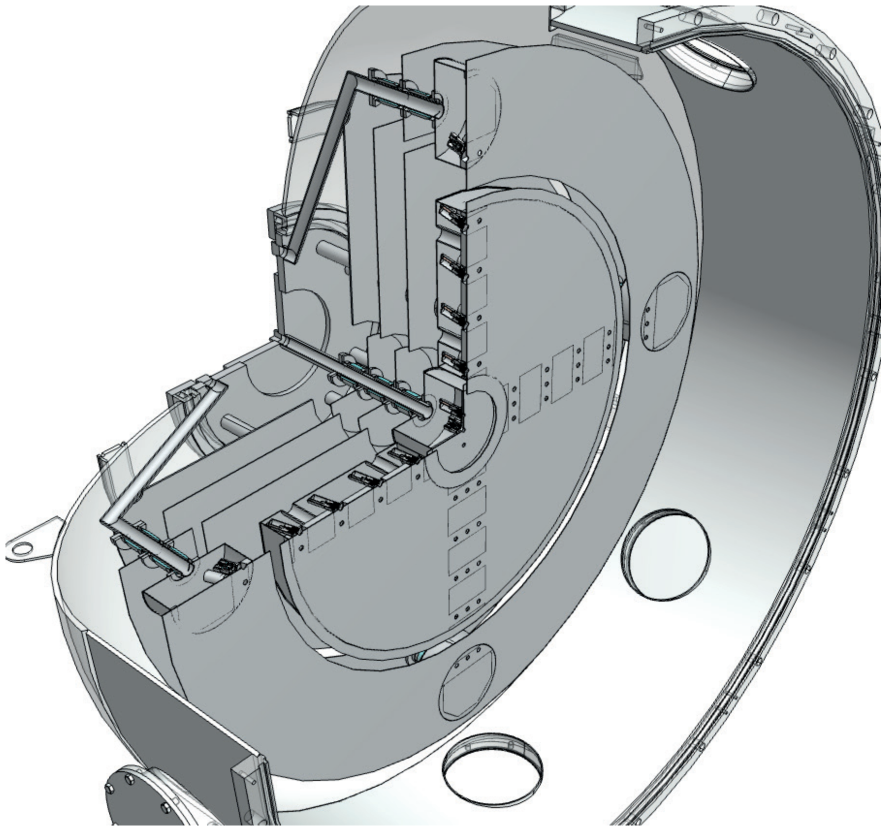

Fig. 2 - Scheme of end plate in western expander of GDT

To investigate neutral gas behavior in the expander the optic tomography is now being developed at GDT (Fig. 3). This is a system consisting of 42 channels, which can register radiation of H$\alpha$ and D$\alpha$ lines in the expander using narrow-banded interference filters. Avalanche photodiodes with broadband amplifiers are being used as detectors of radiation. This system allows investigating plasma dynamics in range of frequencies up to 1MHz. We can estimate radial profile of radiation in the expander by one-dimensional code constructed for the moment.
Typical results on radiation profiles are shown at Fig. 4. NBI pulse starts at 4 ms from GDT impulse beginning and finishes at 9 ms. As far as radiation intensity indicates gas density profile, therefore from the Figure 4(a) it's obvious that gas moves from the axis to the periphery during the impulse. Herewith the radiation at the periphery is rising at higher values of puffed gas density (Fig. 4(b)).

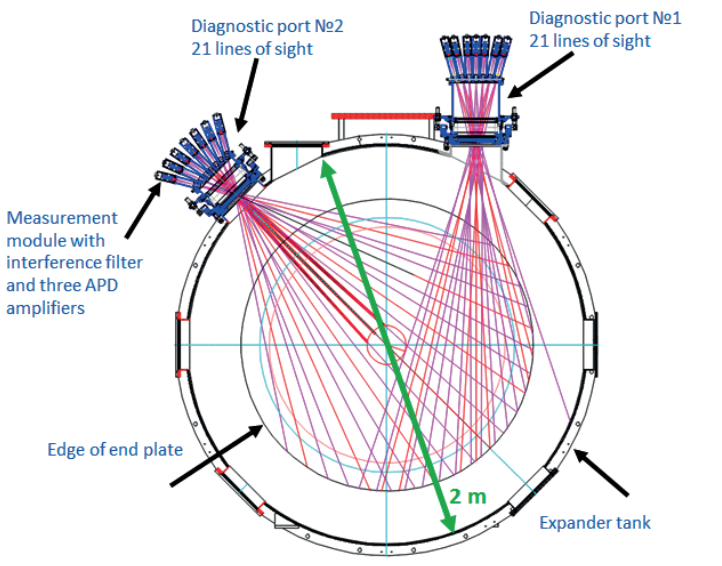

*Fig. 3 - Layout of optic tomography system in GDT expander*

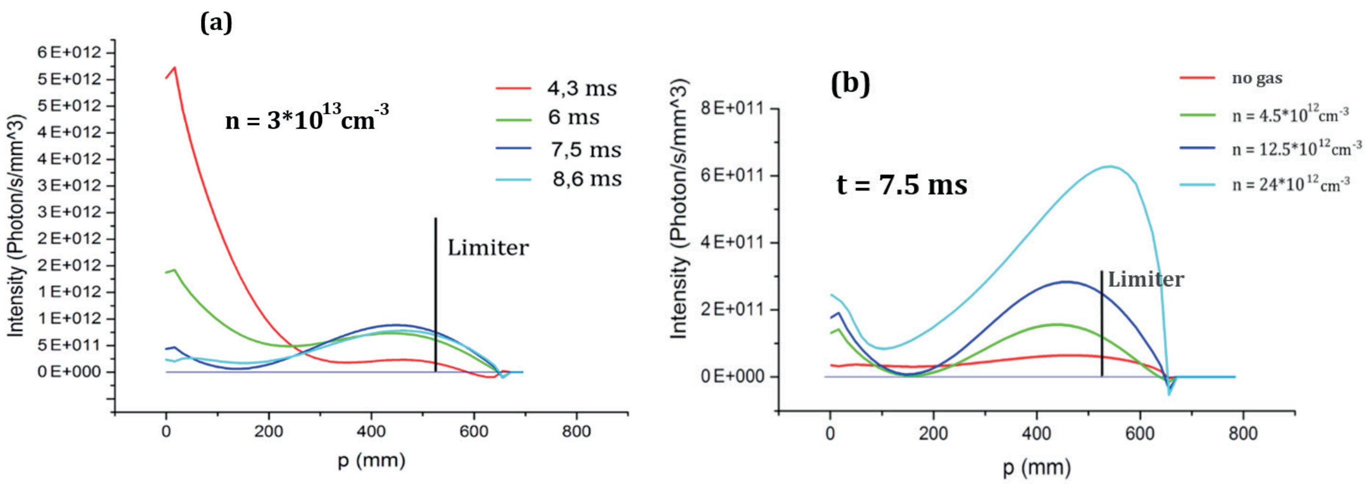

*Fig. 4 - Radial profiles of H$\alpha$ line intensity in GDT expander: (a) for different moments of the GDT impulse at neutral gas density of n = $3{\cdot}10^{13}$ cm$^{-3}$, (b) for different densities of neutral gas at the moment of 7.5 ms*

## Numerical model

Results obtained can be interpreted as absence of gas ionization in the expander region and extruding of neutral gas from the axis of the expander to its periphery. To describe gas behavior computational model had been created. This model is based on the numerical solution of kinetic equation for neutral gas, which has initially Maxwellian distribution function and interacts with plasma ions by elastic collisions $H_2$ + D+ → $D^+$ + $H_2$ (0.5 eV), cross section $\sigma = 3{\cdot}10^{-15}$ cm$^{-2}$ [6]. Kinetic equation for gas particles had been solved in the region inside the cylindrical plasma column with fixed parameters; the collisions with large transmitted momentum had been taken into account.
Figure 5 represents radial distribution of gas density calculated by means of described model for various collision frequencies ($\gamma = a/\lambda$ – collision parameter, a – plasma radius, $\lambda$ – electron mean free path, initial gas density $n_0 = 10^{13}$ cm$^{-3}$).

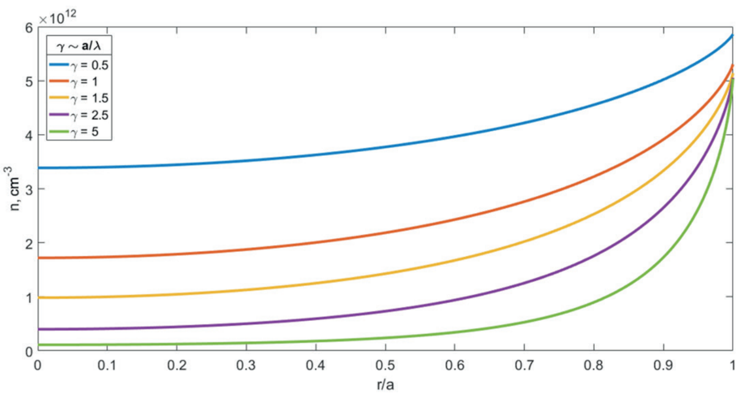

*Fig. 5 - Radial distribution of gas density for various collision frequencies calculated by kinetic numerical code*

Numerical results on gas density profile inside plasma column also show gas extrusion from the plasma, which is in a principal agreement with experimental data. This model is now under development. It has planned to include in it a gas dynamic part to calculate gas behavior in region between plasma and chamber and to consider some inelastic processes.

## Conclusions

Experiments to study the influence of neutral gas in the expander on plasma confinement in the central part of the GDT were carried out. It is shown that the key parameters of the plasma remain constant over a wide range of gas densities in the expander: from $10^{10}$ to $10^{14}$ $cm^{-3}$.

The assumption that the gas is extruded from plasma due to elastic collisions has experimental basis and is confirmed by preliminary numerical calculations; a corresponding computational model is being developed. In the first approximation, it can be argued that in a fusion reactor based on an open trap, the requirements for vacuum systems for expanders can be significantly softened compared to those originally planned.

## Acknowledgements

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# High field neutral beam injection for improving Q of GDT-based fusion neutron source

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## Abstract

Gas Dynamic Trap (GDT) is very attractive as a kind of fusion neutron source (FNS) for testing fusion materials and components as well as driving fusion-fission hybrid reactor due to its linear and compact structure, low physics and technology requirement, relatively low cost and tritium consumption. These years, the conceptual designs of GDT-based FNS for above two purposes, named FDS-GDT, have been designed by Institute of Nuclear Energy Safety Technology (INEST), Chinese Academy of Sciences · FDS Team in China. However, the fusion energy gains (Q) in current designs are still far lower than one, even lower than 0.05. A new method was proposed that using high field neutral beam injection (HFNBI) for improving the Q of GDT, and the preliminary analysis show that the Q could possibly be improved 2-3 folds.

The parameters of two designs of GDT-based fusion neutron source (FDS-GDT) were updated with new simulation models by updating plasma power balance and particle balance of GDT in case of HFNBI. One (GDT-FVNS) is for testing fusion materials and components which can provide 2.57 MW of fusion power, up to 2 MW/m$^2$ of neutron flux density. The Q will be 0.13 and the neutral beam injection power only requires 20 MW in condition of that the maximum magnetic field in the mirror throat is about 15 T and mirror ratio of 100. The other (GDT-hybrid) is for driving fusion-fission hybrid reactor with 14.72 MW of fusion power and 0.16 of Q. The subcritical blanket for transmuting the minor actinides (MA) was also designed by using Super Multi-functional Calculation Program for Nuclear Design and Safety Evaluation (SuperMC) developed by the FDS Team. The simulation results shown that the transmutation system can transmute about 95.1 kg minor actinides per year and produce about 450 MW of electricity power.

## Keywords

Gas Dynamic Trap, Fusion Neutron Source, Fusion Energy Gain

# Optimization of stability conditions for the DCLC mode in a neutron source based on the gas-dynamic mirror trap

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## Abstract

At the a gas-dynamic trap (GDT) facility operating at the Budker Institute of Nuclear Physics, parameters have already been reached that allow creating a neutron source for materials science applications. With some extrapolation of the GDT parameters, it is possible to create a subcritical reactor driver. However, on the way to such extrapolation, there is a danger of the growth of the Drift-Cyclotron Loss-Cone (DCLC) oscillations. This instability arises because of the empty loss cone in the population of hot ions in combination with the radial density gradient of the plasma. It generates potential oscillations, which are stretched along the lines of the magnetic field, run along the azimuth and have a frequency near the ion cyclotron one. The oscillations lead to anomalous scattering of ions and can provoke losses of particles and energy from the source of neutrons. A method of stabilizing the DCLC instability is known by filling the loss cone with warm ions. However, as the density of warm ions increases, a Double-Humped (DH) instability develops. To stabilize it, an increase in the temperature of warm ions is required, which contradicts the conditions for suppressing the DCLC instability as the distribution of warm ions acquires empty loss cone.

The present paper is devoted to the search for conditions for the stabilization of a plasma with several species of ions, in particular, a mixture of deuterium and tritium. We chose neutron generator parameters and hot ion distribution functions based on the simulation results using the DOL numerical code. To calculate the increments of unstable oscillations, we used a dispersion equation derived in the approximation of smallness of the transversal wavelength in comparison with the radius of the plasma. Based on the results obtained, we formulated a rule for selecting the parameters of a population of warm ions. It states that for effective stabilization of the DCLC and DH, the temperature of warm ions should exceed a certain value, and the spectrum of harmonics of cyclotron frequencies of warm ions should overlap all cyclotron harmonics of hot ions.

## Keywords

Fusion Neutron Sources, Subcritical Systems, Drift-Cyclotron Loss-Cone mode

# Optimization of GDT-based neutron source: results and plans

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## Abstract

Interest to hybrid reactors is rising now worldwide. The key part of hybrid reactor is an intensive neutron source (NS). One of the proposed NS concepts is based on gas-dynamic trap (GDT). This paper is focused on the optimization of GDT-NS parameters performed by DOL code [1]. Previous simulations [2] lacked micro-stability analysis. The next step of optimization will take into account stability criteria for Drift-Cyclotron Loss-Cone and Double-Humped modes.

## Keywords

Fusion Neutron Sources, Subcritical Systems, Gas-Dynamic Trap

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# Magnetic system of GDMT – the next-generation multipurpose experimental facility for evaluation of plasma performance of GDT-based fusion neutron source

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## Abstract

Gas-dynamic multi-mirror trap (GDMT) is an ongoing project in development at Budker Institute of Nuclear Physics, which aims to demonstrate a leap in performance for linear axisymmetric magnetic plasma confinement systems and lay the groundwork for a full-scale development of fusion neutron sources (NS) based on gas-dynamic plasma confinement. As a multipurpose facility, GDMT is being designed to experimentally evaluate a number of concepts, which promise a dramatic improvement in plasma confinement in linear systems.

Magnetic system of GDMT consists of a confinement region with magnetic field strength up to 3 T which is terminated by either single high-field (up to 18 T) magnetic mirrors or advanced multi-mirror modules, which provide necessary plasma flux suppression. Owing to its modular design, the system length can change from 6 m for studies of high-$\beta$ plasma regimes to several tens of meters to accommodate for NS studies with maximized confinement zone length and most advanced multi-mirror modules. The design goal is to build a modular superconducting magnetic system, which enables an easy reconfiguration of the confinement zone and attachment of mirror sections, while being mechanically robust and cryogenically efficient.

The report outlines the scientific program of GDMT and related requirements on the magnetic system limited to first-stage experiments with single magnetic mirrors. Starting with an optimal subdivision of solenoid into coils, which provides required magnetic field homogeneity and inventory of vacuum ports, the report focuses on several types of cryogenic modules which make up the magnetic system. The paper details the mechanical structure of the modules and presents the design of the cryogenic system including cryostat supports, radiation shield, current leads and the selection of cryorefrigerators, which enables operation with minimal refills of liquid helium.

## Keywords

Fusion Neutron Source, Gas-dynamic Trap, Magnetic Mirror, Superconducting Magnets, Cryogenic System

*Session 3*

# Other Confinement Concepts

# Fusion-fission hybrid reactor system based on a minimum B field with guiding center motion on magnetic surface

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## Abstract

The SFLM (Straight Field Line Mirror) concept consists of single cell minimum *B* mirror field with expanders beyond the confinement region. Motivations for the SFLM studies are to identify a steady state device design where obstacles for application would not be ruled out by insufficient plasma confinement, material problems, accessibility for diagnostics, tritium consumption and breeding, as well as reactor safety issues. A radial constant of motion is required for confinement. That is arranged by magnetic shaping combined with a weak plasma rotation controlled by biasing plates at the end tank. Some key results are on the plasma heating, on material loads (from neutrons and plasma bombardment), on a high power amplification by fission (preferably as high as possible within safety constraints) and on reactor safety. Critical problems are avoided by the compact super conducting coil design, the flux tube expander, the openings for accessibility, the plasma heating methods and reactor blanket arrangements. The SFLM geometrical arrangements address these challenges.

## Keywords

Hybrid reactor, fusion neutron source, magnetic mirror, SFLM, biasing plates
PACS: 52.55

## Introduction

Fusion neutron sources [1] offer a possibility for application of fusion in a not too distant future. Fusion material testing is an area with quite urgent needs, and energy production may be achieved in a hybrid reactor where the main power is provided by fission in a reactor surrounding the fusion neutron source. Steady-state operation on a time scale of a year without interruption is essential for such applications. In response to this, we analyze a mirror machine concept, i.e. the SFLM (Straight Field Line Mirror) [2-15]. Several critical issues for applications are addressed in our studies.

In figures 1 and 2, a compact 10 MW fusion power device (25 m long with an outer coil radius of 3 m and a mid-plane plasma radius of 0.4 m) is outlined [8,9]. The design is aimed for a 1.5 $GW_{th}$ hybrid reactor, where fission provides a power amplification by about a factor of 150. Damage of equipment is reduced by avoiding ports for diagnostics and plasma heating [5-7] in sensitive areas.

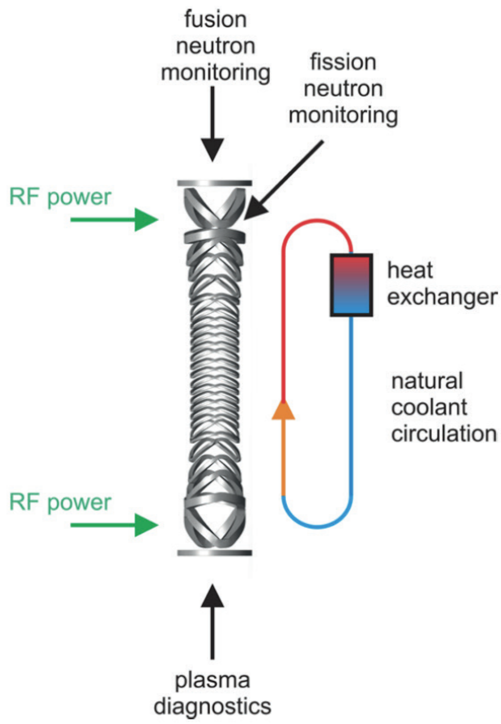


*Figure 1 - Outline of SFLM reactor arrangements with the 3D coils shown. Diagnostics and monitoring equipment are located on the top and bottom of the device to protect sensitive equipment from neutron and particle bombardment. RF heating is fed through openings near the magnetic mirror ends. A fission reactor core is located in between the plasma confinement region and the coils, where the arrangement with minimized holes for diagnostics is beneficial for neutron economy. A vertical orientation enables passive coolant circulation*

The top and bottom of the device is intended for access, with feeding of plasma heating and diagnostics. On opposite sides of the plasma confinement region, the magnetic flux tube expands and directs plasma to large receiving plates with a radius of 4 m, which replace the "divertor plates" in a toroidal device. Electric potential control can be installed at sectioned end tank plates to produce a weak radial electric field [15] with an associated slow plasma rotation, which is predicted to eliminate collision-less radial drifts. Each particle then moves in the vicinity of a single magnetic surface, and from this we identify *a radial constant of motion* for the SFLM magnetic mirror system [15].
In figure 2, the compact 3D superconducting coils [10, 12] have 2 m inner radii and only 3 m coil outer radii. Sufficient space is available for the fusion device and the fission reactor core. Strong mirror field gradients have imposed a particular challenge in the coil design.
Antennas for ICRH (ion cyclotron resonance heating) are located on both sides of the mirror ends of the device [5, 6, 7] in regions protected from neutron bombardment. ICRH heating can be applied over long times, even for a time scale of a year.

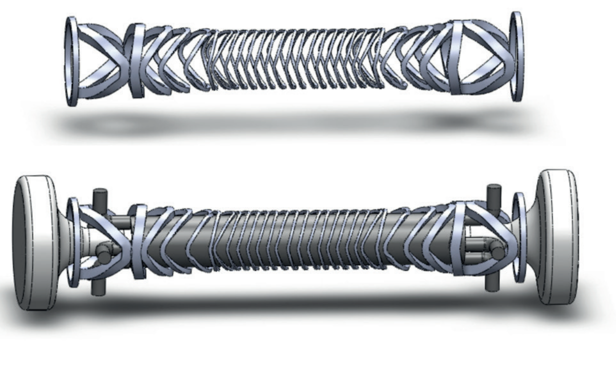

*Figure 2 - Superconducting coils for the SFLM. The upper figure shows the superconducting coils. The lower figure indicates location of fission reactor part between the coils and the vacuum chamber for the plasma and the expander plates which are intended as plasma receiving "divertor plates"*

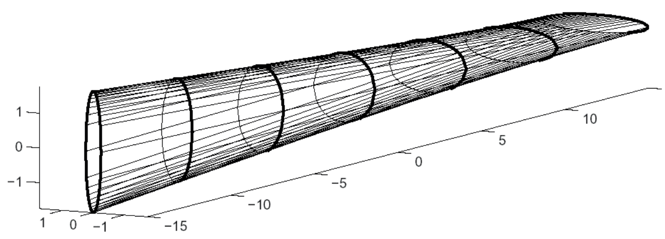


*Figure 3 - The magnetic field lines of the SFLM are straight and non-parallel. With straight field lines, there is no guiding center drift, and each guiding centers move on a single magnetic field line. The stronger magnetic field modulus at the opposite ends provide longitudinal confinement by the mirror effect*

Neutrons produced from fusion reactions can be utilized for fission and incineration in a surrounding reactor [8, 9, 13]. Avoidance of holes in this region is important to achieve a high power amplification by fission (up to a value as high as 150 in the SFLM-geometry). Even a fusion Q factor as low as Q=0.15 may be sufficient for efficient power production. This is about ten times lower than predicted for tokamaks.

## Radial constant of motion

We need to identify constants of motion to assure confinement along the longitudinal and radial directions [15] in a collision-free idealization. As is well known, the longitudinal confinement in a mirror machine relies on the mirror effect, which is governed by two constants of motion, i.e. the energy $\varepsilon = q\phi + m\mathrm{v}^2/2$ of the particle and its magnetic moment $\mu = m\mathrm{v}_\perp^2/(2B)$, which yield confinement to a region $B \le (\varepsilon - q\phi)/\mu$. Radial confinement is not guaranteed from first principles in a mirror field where the asymmetry is broken. Collision-free radial drifts into regions not intended for confinement must be avoided since it would quickly ruin the confinement. We have shown that the guiding centers could be forced to move on its mean magnetic surface, apart from minor oscillatory radial excursions which could be neglected, with a properly shaped minimum *B* field combined with a weak plasma rotation. The plasma rotation could be controlled by biasing potential plates placed at the end tank. Voltage requirements are modest, in the range of only 200 V in a reactor scenario. Since a magnetic surface has a constant value of the radial Clebsch coordinate $r_0$, a guiding center motion on a mean magnetic surface implies a constant value of the guiding center radial coordinate $\bar{r}_0(\mathbf{x},\mathbf{v})$. We then identify the *radial constant of motion*

$$\rightarrow \qquad \bar{r}_0(\mathbf{x},\mathbf{v}) = r_0 - r_{0,g} = constant$$

Here, $r_0$ is the radial coordinate of the particle and $\bar{r}_{0,g}(\mathbf{x},\mathbf{v})$ is the small but fast "gyro ripple" associated with the gyro motion (which is responsible for the diamagnetic current). For a fusion reactor, confinement for about 10 000 longitudinal bounces are required for power production. This restricts the tolerable radial net drift in each longitudinal bounce to less than 0.1 mm in a 100 m long device with 1 m plasma radius, which is a challenging demand for any confining device. Here, the existence of the radial invariant $\bar{r}_0(\mathbf{x},\mathbf{v})$ assures a perfect radial confinement in a collision-free idealization. Vlasov equilibria could then be described by distribution functions which depend only on constants of motions [19], i.e.

$$\rightarrow \qquad F(\varepsilon,\mu,\bar{r}_0) = F_c - r_{0,g}\frac{\partial F_c}{\partial r_0}$$

where $F_c \equiv F(\varepsilon,\mu,r_0)$ is evaluated at the radial position of the particle (not the guiding center) and the term $-r_{0,g} \cdot \partial F_c / \partial r_0$ give rise to the diamagnetic plasma current. This class of distribution functions for the ions and the electrons leads to a quasi-neutral electric potential

$$\rightarrow \qquad \hat{\phi}(r_0, B) \approx \hat{\phi}(r_0)$$

in the plasma region, where the dependence on B reflects the mirror effect. In typical cases, the electric potential is nearly constant on a magnetic surface and $\phi \approx \hat{\phi}(r_0)$. That provides a possibility to control the radial electric field by biasing plates outside the confinement region.

The idealized SFLM field has straight nonparallel magnetic field lines [2], where each gyro center is restricted to move on a single magnetic field line [4], see Figure 3. A concern for radial confinement is deviation by field errors from this ideal field. That cause radial excursions from the initial magnetic surface, which often corresponds to oscillatory radial drifts for the imperfect SFLM field, see Figure 4, and then poses no major threat [11]. However, some particles quickly escape the confinement region by a net radial drift [14]. A weak plasma rotation could cure the situation [19]. Radial excursions in Fig. 4 would shrink to nearly zero by applying a weak electric potential variation across the plasma cross section.

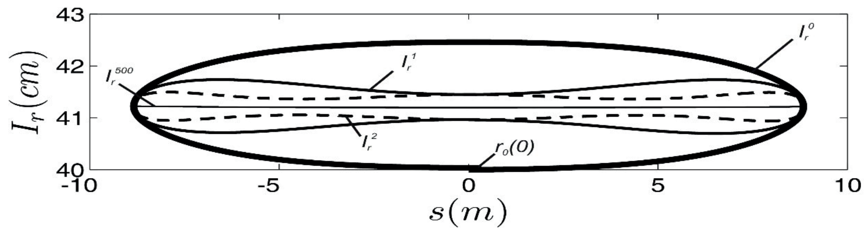


*Figure 4. Variations of the lowest order radial invariant describe "banana" excursions from a magnetic flux surface during a longitudinal bounce. In the radial invariant, banana widths are accounted for by bounce harmonic terms. Notice that the longitudinal scale is much larger than the radial scale, which demonstrates demonstrates small radial excursions of the particle. The "banana" widths can be reduced further by applying a radial electric field*

Several experiments have demonstrated confinement improvement by a radial electric field. That is applied in the GDT (Gas Dynamic Trap) axisymmetric mirror [16] as well as in the anchor cells of the Gamma10 tandem mirror. A very high ellipticity (around 50) is required in the Gamma 10 anchor cells for plasma stabilization by the quadrupolar field, which has a drawback of introducing strong radial magnetic drifts. For this reason, it has been necessary to introduce biased electric potential plates to reduce the radial drift loss in Gamma10. One suggested interpretation [16] for the experimental results in GDT is that a shear plasma rotation "chops" large plasma structures originating from flute instabilities into smaller structures near the region of opposite plasma rotation, which may produce improved overall confinement in a similar manner as an ITB (internal transport barrier). A complete stabilization of the large scale flute mode is however preferable, and this is a motivation to consider an average minimum B field such as the SFLM field. The ambition for the SFLM is quiescent elimination of non-oscillatory radial drifts, with MHD stability maintained.
The complete magnetic field has expanders beyond the confinement region. A detailed magnetic field shaping, to minimize flux surface ellipticity and maintain average minimum B property for plasma stability, have confirmed that the magnetic field is almost identical to the SFLM field in the major part of the confinement region. Flux surface footprints near the expander tank walls are nearly circular, which enables a large number of biasing potential plates with short-circuiting avoided [19] (each flux surface must at most intersect one potential plate). With a mirror ratio of four, the maximal flux surface ellipticity is found to be

$$\rightarrow \qquad \varepsilon_{ell} \approx 16$$

with a strict minimum B field in the confinement region for stability. Although an increased ellipticity could be an acceptable price for flute mode stability, a too high ellipticity means an impractical "needle-like" shape of the plasma cross section. The obtained ellipticity seems to be well within a tolerable range.

## Plasma heating

We intend to investigate if higher voltages of the biasing plates (exceeding 100 kV) could be a method to heat the plasma from a controlled $E \times B$ rotation. Centrifugal confinement experiments make use of a strong plasma rotation for plasma heating. The Alfven critical speed, which corresponds to a rotation energy around 3 eV for the electrons, have been a threshold in several experiments. One reason may be that the ionization of neutrals may trigger short-circuiting events between the biasing plates. Since the Alfven critical speed corresponds to a 10 keV rotation energy for deuterium ions, heating the ions to a few keV may be possible without challenging the Afven critical speed. An attractive feature of centrifugal confinement schemes, with heating mechanism like in Penning discharge [20], is the steady-state option and the possibility to control the radial deposition of the heating. Control of the longitudinal heating deposition requires a different method. The SFLM studies have considered minority deuterium heating at the fundamental cyclotron frequency and second harmonic heating of the tritons [5, 6, 7]. The two antennas could be placed on opposite ends of the mirror. Good coupling between the antennas and the plasma with efficient heating is predicted, with sustaining of a sloshing ion distribution [5, 6, 7]. Placing the antennas near the mirror ends is favorable for protection from particle bombardment.

## Neutron computations, power amplification and reactor safety

Results from Monte Carlo simulations for the neutrons have previously been reported in the FUNFI2 proceedings. A 25 m long device aimed for 10 GW fusion power has been considered. Omitting details, the geometrical arrangement is the following: Space is available for a fusion neutron source in the region $r < 1\,\mathrm{m}$, and 3D superconducting magnets could be placed in the region $2\,\mathrm{m} < r < 3\,\mathrm{m}$. An annular reactor region between $1\,\mathrm{m} < r < 2\,\mathrm{m}$ contains a buffer region, core expansion zone, fuel, coolant loops, reflector, shielding region, tritium breeding etc. The computations predict a tritium breeding ratio above unity, good shielding of superconducting magnets and antennas, and a 200 dpa rate for the first wall exceeding 30 years, where it is necessary to account both for the neutrons originating from fission as well as fusion. The design with a buffer has a favorable impact on the dpa rate of the first wall. Detailed material choices and geometrical arrangements could be found in [9].

The computations are carried out for a subcritical device, with the neutron multiplicity $k_{eff}$ below unity. As high a value of $k_{eff}$ as possible is desirable for power production, but safety requirements impose an upper bound on $k_{eff}$. Our computations focused on the value

$$k_{eff} = 0.97,$$

which gives a 3% margin to a critical state [9]. All preliminary studies carried out this far, which includes LOCA (loss of coolants) etc, suggest that reactor safety could be maintained with this value. Although a supercritical scenario so far has not been identified [8, 9], it should be emphasized that there remains a need to deepen the reactor safety studies. Reactor safety should be a first priority for any nuclear installation.

With $k_{eff} = 0.97$, the power amplification by fission, i.e. ratio $Q_{PAF} = P_{fis} / P_{fus}$ between generated fusion power and fission power, can be surprisingly high for the SFLM [9, 13]:

$$Q_{PAF} \equiv \frac{P_{fis}}{P_{fus}} \approx 150$$

This implies that a fusion power of only $P_{fus} = 10\,\mathrm{MW}$ would correspond to 1.5 $GW_{th}$ total power production. Even a fusion Q factor as low as [8]

$$Q = 0.15$$

may be sufficient for efficient power production. With such a low value for the $Q$ factor, the "divertor plates" should be capable of receiving 60 MW power from plasma leaking to the end walls. For a 4 m end tank radius, the power deposition is w0.6 MW/m$^2$, within a tolerable range for the heat load. The heat load reduces further if the fusion Q could be increased.

A safety arrangement is the option to quickly turn off the fusion neutron source. There is also a need to remove decay heat, with can be of the order of 10% of the full power on a short time scale after the reactor is switched off. With a vertical orientation of the reactor, passive coolant circulation is predicted to be capable of removing a decay heat of 150 MW [18]. External pumps would be required to remove heat when the reactor is switched on. The calculated pumping power [18] well below 50 MW is within a tolerable range.

Although the tritium consumption is only about 1% of the consumption in a fusion reactor, it is still necessary to avoid the cost of the tritium fuel. A tritium breeding ratio above unity, where 1.5 is a representative number, is conveniently obtained for the SFLM hybrid reactor.

## Discussion and conclusions

Inspired by the progress made in the Russian GDT device, Chinese researchers have launched an initiative to construct an axisymmetric mirror fusion neutron source with superconducting magnets. Superconducting magnetics would expand the experience of mirror plasmas, since all mirror devices up to now have been operated in short pulses, limited by the heating of the magnets. Although mirrors have a "natural divertor" which expel impurities from the confinement region, avoidance of impurity accumulation with longer pulses needs to be experimentally verified. Other system challenges with prolonged plasma discharges, for instance plasma heating, could be tested with a superconducting device. Axisymmetric mirror machines with planar circular coils can provide high mirror ratios with circular plasma cross sections. Simplicity and flexibility is a strong argument for axisymmetric systems. An uncertainty with axisymmetric systems is flute mode stability. Improved confinement by expanders, combined with a shear plasma rotation, may be insufficient in certain parameter regimes. If adequate confinement quality could not be reached, a reserve plan may be to switch to a minimum $B$ field, which in several experiments have demonstrated a robust stabilization of the flute mode.

Mirror machines have a high $\beta$ (ratio of plasma pressure to magnetic field pressure) and flexible geometrical arrangements. The major obstacle for mirror machines have been a too low electron temperature, compare [3]. However, progress with increased electron temperatures have been reported in several mirror experiments, i.e. the multimirror GOL 3 and GDT experiments at the Budker Institute in Russia and at the Gamma10 experiment in Japan. The GDT results have a particular relevance for our studies [16]. In a sequence of experiments, the GDT electron temperature have been increased, where the present limitation seems to be around 900 eV, verified in a few shots. This is approaching a range where the electron temperature would be sufficient for power production in a hybrid reactor.

Several of our mirror results can be transferred to the steady-state stellarator-mirror experiments at KHIPT

at Kharkiv in Ukraine [17]. A local mirror region is established in the Uragan 2M stellarator device by switching off a toroidal field coil, resulting in a stellarator-mirror configuration. A radial electric, which may be spontaneously generated by a small initial escape of charges, is predicted to have a favorable influence on radial confinement also for that toroidal system, although the option to use electrical biased end plates is not available for a toroidal system.
Results from the SFLM are summarized in this paper. Major predictions are:

1. MHD stability with a high $\beta$ is expected from the minimum *B* property.
2. Collision free radial drift loss is predicted to be eliminated by applying a radial electric field controlled by biasing potential plates outside the confinement region. Each guiding center motion then approaches a motion on a single magnetic surface. A radial constant of motion exist in such situations.
3. With a mirror ratio of 4, the maximal flux surface ellipticity could be reduced to 16.
4. Steady-state operation for a time scale over a year is not ruled out.
5. Plasma heating by ICRH is predicted to be efficient. Centrifugal confinement may be an additional mean to control the heating deposition.
6. A compact design is possible with superconducting 3D coils.
7. The geometry of the SFLM design is aimed to satisfy reactor requirements for accessibility and avoidance of damage of sensitive equipment.
8. Preliminary reactor safety studies are favorable, but deepened studies are necessary.
9. The tritium breeding ratio is above unity. The first wall 200 dpa rate exceeds 30 years.
10. A small fusion power (10 MW) may be sufficient for a power generation of 1.5 $GW_{th}$.

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# Subcritical systems and stellarator-mirror fusion-fission hybrid

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## Abstract

A subcritical nuclear reactor must have an embedded neutron source to cover shortage of neutrons. The fully controlled neutron source then gives full control on fission reaction inside the reactor. This control is undoubtedly necessary when there is lack of the delayed neutrons, which are the only that give an opportunity for control of critical reactors. Such a situation appears in transmutation of spent nuclear fuel by subcritical systems. The transmutation is, in fact, usage the fuel made of spent nuclear fuel for energy production. Future of the nuclear energy is associated with the fast reactors which burn the synthetic isotopes U-233 and Pu. These nuclei offer much less delayed neutrons than U-235 that is currently used for energy production. There is not much experience with critical fast reactors, and the question whether they are within safety margins is open. Subcritical systems can compete at this market too. There is a new initiative of "hybrid nuclear-renewable energy systems" [1] in which the nuclear part could be also a subcritical system.
Among fusion based sub-critical systems two different approaches could be separated. The first one is integration of the fission blanket into a fusion machine, normally into a tokamak. This approach is inherently costly since a fusion reactor is much more complicated than a fission one. Another approach is based on neutron generation inside a cavity in the nuclear reactor by magnetically trapped fast tritium ions that are sloshing inside the deuterium plasma. Such a scenario could be organized using a magnetic mirror or stellarator-mirror combined plasma machine. The major advantage of such an approach is a possibility to place mostly all plasma systems into a neutron-free zone. This greatly reduces the complexity and cost of the system and simplifies its maintenance. The mirror plasma device does not show excellent plasma confinement. For this reason it is good to consider its usage with reactors of high neutron multiplication factor $k_{eff} \leq 0.995$.
The stellarator-mirror fission-fusion hybrid [2] may operate with lower keff. In this report, the major developments for this concept are discussed along the future prospects.

## Keywords

Fusion Neutron Sources, Subcritical Systems

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# Preliminary integrated design of a RFP fusion core and a hybrid reactor blanket

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## Abstract

The Reversed Field Pinch (RFP) could be an attractive fast neutron source from D-T fusion in hybrid reactors due to the possibility to reach fusion condition with ohmic heating only and to the simplified design (no large toroidal field coils, no additional heating systems, no divertor) with respect to Tokamak. In previous studies, a pulsed operation, able to guarantee a quasi-continuous 14.1 MeV neutron production (dwell time of a few seconds), has been identified, utilizing a purely inductive plasma current rise and sustainment. Scaling laws, derived from the experimental results obtained in RFX-mod, allow for predicting the plasma temperature and the loop voltage vs. the machine sizes and plasma current level. Considering these laws and on the basis of a preliminary poloidal coil design, the relationship between the machine size and the attainable stored volt-second were investigated. The achievable plasma parameters (current, loop voltage, pulse duration and temperature) with reasonable machine sizes, R=6 and a=1 for examples, match very well the performances required for an hybrid reactor in terms of neutron flux and machine stresses. Based on this configuration a blanket, surrounding the torus, composed of a lithium-lead eutectic mixture for tritium production and a three fission sectors fuelled by steel rods containing Pu+MA (60%)-Zr (40%) embedded in liquid lead was studied and designed. The nuclear analysis of this simple configuration shows the possibility to operate at $k_{eff}$ ~ 0.97 corresponding to a total fission power of about 1.2 GW. Improvements of the RFX-mod machine are underway, introducing a new load assembly with reduced distance between plasma and conducting shell which will provide a smoother magnetic boundary. On the basis of present experimental data and model simulations, this is expected to improve the plasma confinement properties.

## Introduction

The Reversed Field Pinch (RFP) is a plasma confinement configuration that, despite of the present lower confinement properties as compared to the Tokamak and Stellarator which make it less attractive for fusion energy production, has potentially other advantages as a fusion reactor due to its more intrinsic simplicity (no superconducting toroidal field coils, no additional heating, no divertor) and the possibility to reach fusion condition by ohmic heating only.
Even if fusion condition with high gain (Q > 10) is still quite far because of the currently achieved confinement time, on the basis of the scaling laws derived from RFX-mod - the largest RFP device in operation - the Q ~ 1 condition, required for a fusion core in a hybrid reactor could be reached in a device with high plasma current (about 20 MA).
The contribution is divided into three sections:

- A short review of the main RFX-mod results and the related scaling laws;
- A preliminary conceptual design of a RFP machine acting as fusion core for a hybrid reactor;
- A preliminary outline for the blanket and first nuclear analyses.

In the conclusions, the future work related to the improvements of RFX-mod is reported, which could further support this solution.

## RFX-mod results review and scaling laws

The RFP magnetic configuration has the peculiar feature that the toroidal component of the magnetic field is self-generated internally by the plasma itself through a process called "dynamo mechanism", the toroidal field winding being devoted only to control a low reversed value at the plasma edge. The reduced level of current in the toroidal windings allows for the use of copper instead of superconductors.
Since no intrinsic limits exist for the plasma current, the burning regime can be achieved by ohmic heating only, without the need of additional heating systems.
In RFX-mod (R=2, a=0.46) plasma current up to 2 MA with electron temperature of 1.6 keV and relative density of $n/n_G$=0.15 have been obtained because a spontaneous helical equilibrium, called Quasi Single Helicity is reached above 1 MA (see [1]).

Data from discharges in H and D show that the plasma electron temperature increases with plasma current, as shown in Fig. 1:

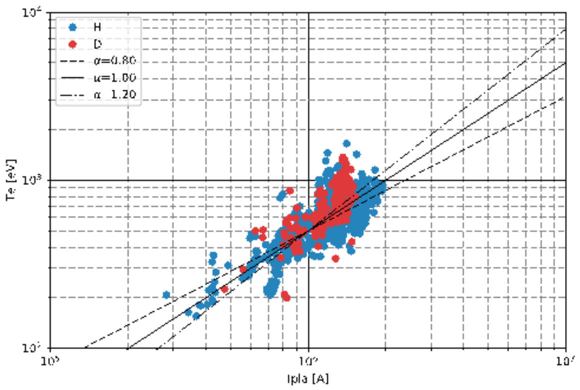

Fig. 1 - Results from RFX-mod: electron temperature vs. plasma current

The resulting scaling is:

$$T_e \propto j^{1.1}$$

where $j$ is the plasma current density.

With further conservative assumptions on the dependence of the temperature vs. minor radius, the following relation can be derived [2]:

$$V \propto R\, a^{-0.35}\ I_p^{-0.65}$$

where $V$ is the plasma resistive loop voltage, $R$ the major radius, $a$ the minor radius and Ip the plasma current.

## Preliminary conceptual design

In Fig. 2 an outline scheme of the fusion-fission hybrid reactor is represented with the main components. The reduced size of the toroidal winding allows, in principle, increased machine accessibility, as shown in Fig. 3.

The loop voltage for plasma current rise and sustainment can be generated inductively, by means of a magnetizing flux variation from its maximum initial value up to the opposite one (double swing operation) as shown in Fig. 4.

When the magnetizing flux reverses its value, the plasma current cannot be further sustained and the decreases to zero but a new pulse is ready to start with a plasma current flowing in the opposite direction with respect to the previous pulse. This type of operation is called "continuous pulsed mode" and a duty cycle with long burning phase can be obtained if sufficient Volt-second are stored in the machine.

In order to verify the Volt-second capability of a RFP machine, a preliminary design of the magnetizing winding with different values of $R$ and a was carried out under the assumption of keeping the magnetic field into the superconductor lower than 12 T and of guaranteeing enough space for the blanket, with a minimum thickness of 35 cm in the inner blanket section to screen the superconductor from neutron flux.

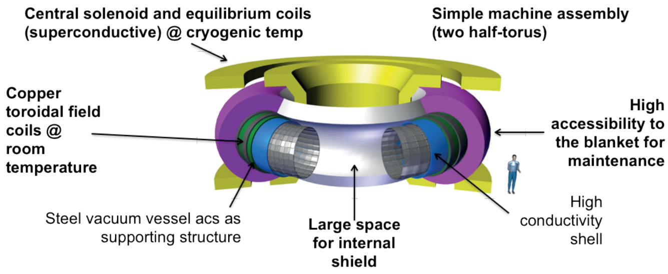


Fig. 2 - Outline scheme of the RFP fusion-fission hybrid reactor with the main components

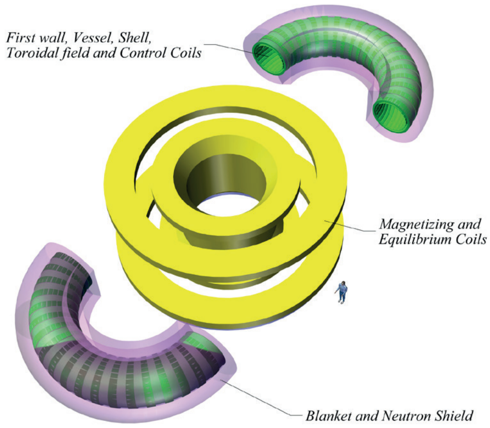


Fig. 3 - Machine accessibility: the torus could be separated into two halves

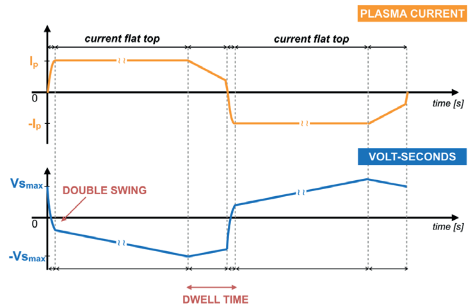


Fig. 4 - Continuous pulsed operation with double swing of magnetizing flux

Taking into account the Volt-second consumption during setting-up related to the plasma internal/external inductance, the generation of the toroidal field by the plasma itself and the resistive losses, the plasma scenarios reported in the Table 1 have been derived with a 20 MA plasma current during the flat top phase.

| | | | | |
|---|---|---|---|---|
| $R$ | [m] | 4 | 6 | 6 |
| $a$ | [m] | 1 | 1 | 1.5 |
| Magnetizing flux | [V s] | 350 | 1050 | 830 |
| $I_p$ | [MA] | 20 | 20 | 20 |
| $T_e$ | [keV] | 9.6 | 9.6 | 6.1 |
| $V_{loop}$ | [V] | 5.1 | 7.6 | 6.6 |
| Flat top | [s] | 10 | 75 | 65 |
| Duty cycle | [%] | 62 | 90 | 87 |
| Fusion power | [MW] | 90 | 135 | 20 |
| $Q$ | | ~0.9 | 1 | 0.15 |
| Thermal wall loading | [MW/m$^2$] | 0.76 | 0.76 | 0.38 |
| Neutron wall loading $\Gamma_n$ | [MW/m$^2$] | 0.46 | 0.46 | 0.04 |
| $\phi_n$ | [$10^{13}$n/cm$^2$/s] | 2.1 | 2.1 | 0.17 |

*Table 1 - Plasma scenarios with different machine sizes*

A significant 14.1 MeV neutron production, in the order of 110 MW is generated in the case R=6, a=1, with a neutron flux of about $2x10^{13}$ neutrons/cm$^2$/s with a pulse duration over a minute and a duty cycle of 90%.

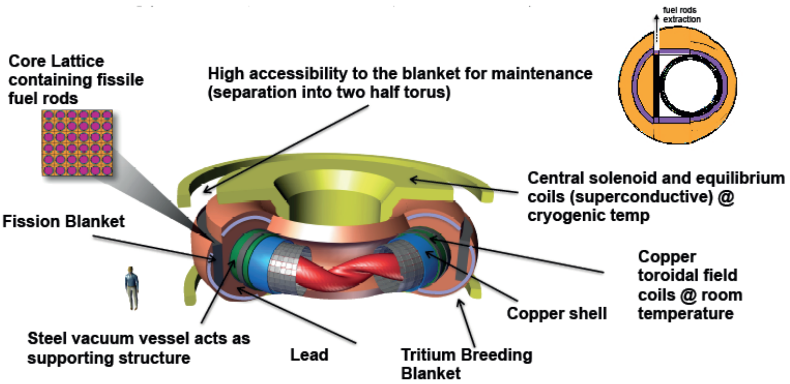


*Fig. 5 - Layout of a preliminary blanket*

## Preliminary blanket design

The blanket has to comply with tritium breeding and fission reactions induced by fast neutrons.
The layout of a preliminary solution for the blanket is shown in Fig. 5.
The blanket contains:

- Lead/lithium eutectic mixture for tritium production
- Liquid lead as coolant
- A core lattice containing fissile fuel roads (road radius 0.450 cm, height 234 cm) with alloy of Plutonium-Minor Actinides (60%) and Zirconium (40%) [3], [4], [5], [6].

To test different aspects (energy production, transmutation and radiopharmaceutical production) the blanket is divided into three sectors, according to Fig. 6.

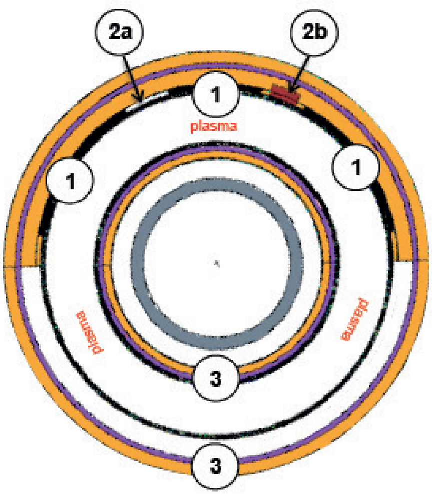


*Fig. 6 - Top view of the blanket showing: 1) the fissile zone with the core lattice; this zone covers about half of the torus; 2) the zones a) and b) are with high neutron flux used for experiments of transmutation for nuclear waste reduction and radiopharmaceutical production through fission reactions induced by slow neutrons and 3-purple) the lead/lithium eutectic mixture for tritium production*

In order to evaluate the effect of the fast neutron produced by fusion in the bulk of the blanket, a schematic layout of the inner part of the torus, which contains the first wall, the conducting shell, the vessel, the toroidal field winding, the saddle coils for MHD control and the cooling system has been assumed as shown in Fig. 7.
With these assumptions a total thermal power of about 1.2 GW is generated inside the blanket with an energy gain of 9 with respect to the generated fusion energy; inside the fissile fuel $k_{eff}$ = 0.973.
The neutron flux inside the core lattice containing the fissile fuel roads is shown in Fig. 8, where a multiplication of a factor of 10 is evaluated with the code MCNP6 in A, the zone closer to the fusion core, and a multiplication of a factor 100 in B, the farther zone from fusion reactions.
An advantage in utilizing this type of fissile fuel is due to the possibility to produce a large amount of thermal power (1.2 GW) with reduced increase of radiotoxicity.
The total fissile fuel radiotoxicity vs. time after 1 year of irradiation and energy production, followed by natural decay periods, is reported in Fig. 9 (blue line) and it is compared with the same fissile fuel maintained with the natural decay (red line). The figure clearly shows that the long term fuel radiotoxicity has a similar trend and its value is not significantly increased after a high energy production amount during a year of reactor operation.

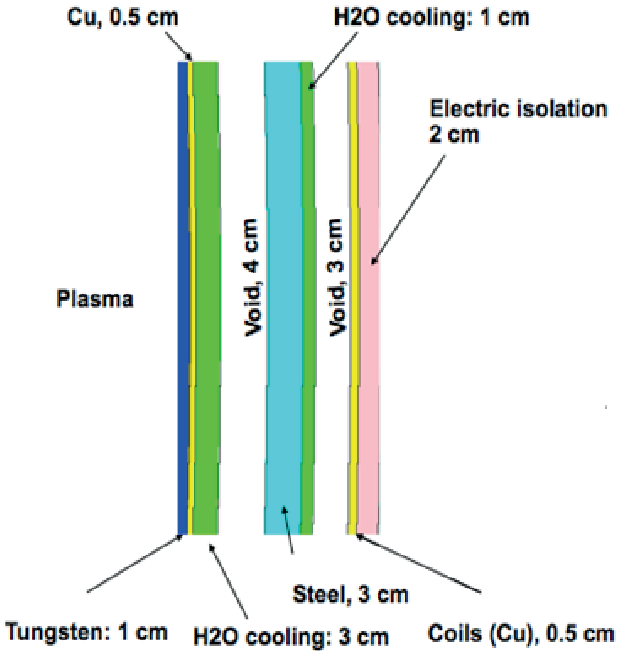


*Fig. 7 - Schematic layout of the inner part of the machine for nuclear analysis*

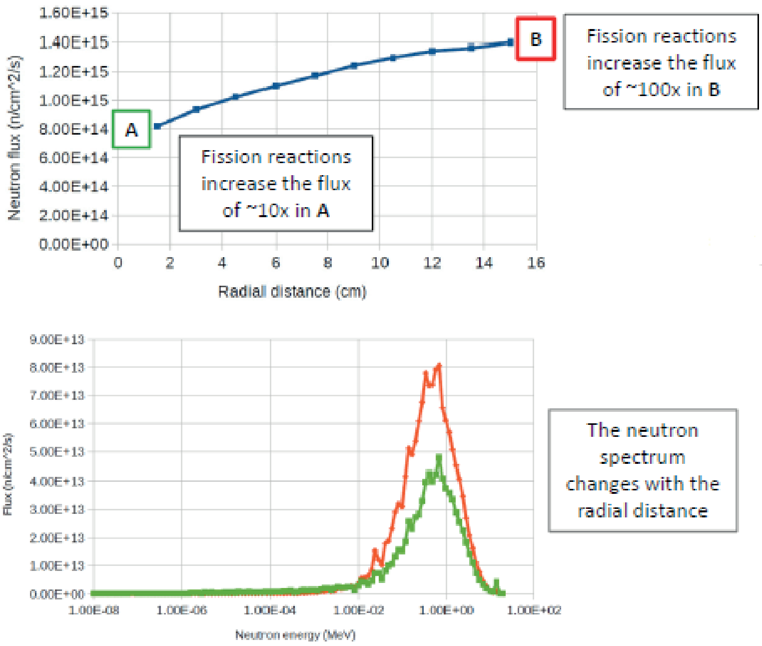


*Fig. 8 - Evaluation of the neutron flux and spectrum with the code MCNP6 in the fissile fuel road*

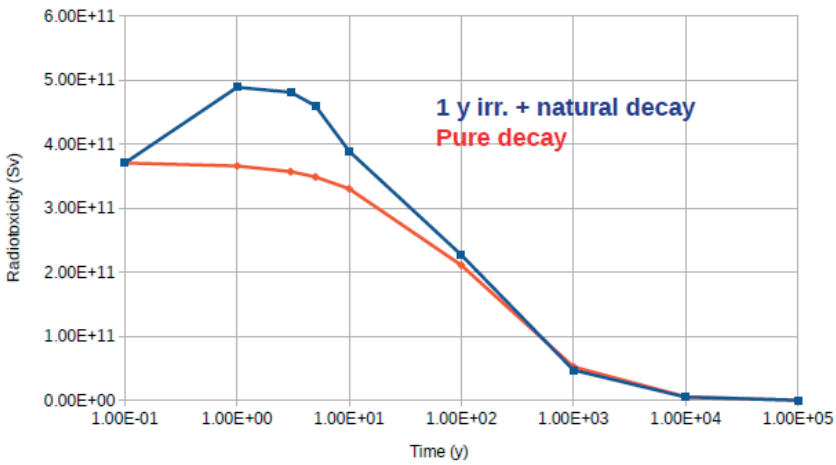


*Fig. 9 - Radiotoxicity of the fissile fuel after a year of irradiation (and energy production) followed by natural decay (blu line) and the same fuel maintained in a natural decay condition (red line)*

## Conclusions

The performance of a hybrid fusion-fission reactor using a RFP configuration for the fusion core was here presented and based on the experimental results of RFX-mod. The machine is presently in a shut-down phase in order to introduce machine improvements which aim at further enhancements of the RFX-mod performances.

The main modification concerns the reduction of the distance between the plasma and the conducting shell in order to minimize the secondary MHD modes at the edge. In fact, a strong correlation of these modes with the plasma confinement time has been experimentally observed, as shown in fig. 10.

From this study the RFP configuration looks as a viable and promising option to generate neutrons for inducing fission reactions and a tentative electromagnetic design of a RFP reactor shows that 20 MA plasma current could be reached, producing up to 140 MW from fusion reactions.

The nuclear analysis confirms that the proposed configuration is tritium self-sufficient and that 1.2 GWt of power is generated inside the blanket with an energy gain of 9 with respect to fusion power.

The underway improvements on RFX-mod are expected to further increase the RFP performances.

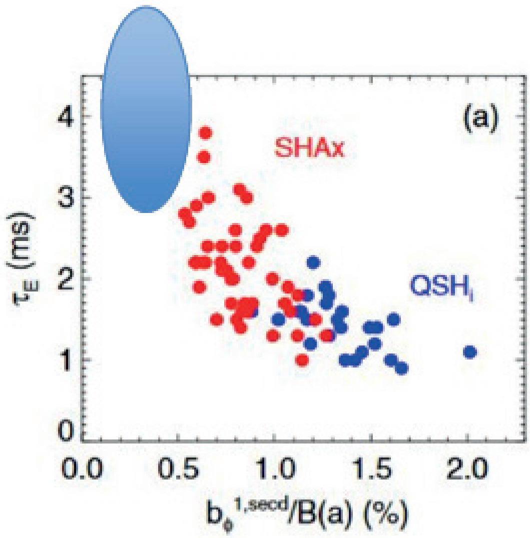


*Fig. 10 - Experimental correlation between the MHD secondary modes and confinement time. Strong reduction of these modes (of a factor 3) is expected with the planned machine improvements; the blue elliptic domain corresponds to the new operating zone*


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*Session 4*

# Subcritical Systems

# The validation of concept and of application performances for source-driven sub-critical systems


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## Abstract

The nuclear fuel cycle and the potential role of FFH will be reviewed. Challenges and the need for validation approaches will also be discussed. Uncertainty quantification, sensitivity analysis and experimental validation with representative experiments will be suggested for the case of the fission component of a hybrid fission-fusion system. Finally, an experimental approach for the validation of the fission component of an FFH will be proposed, using an existing TRIGA reactor, external neutron source driven and operated in a subcritical mode.


## The nuclear fuel cycle and the potential role of FFH

The fission process used in nuclear reactors and the overall nuclear fuel cycle (Fig. 1) produces a number of isotopes (Fig. 2) that can be toxic to human beings and the environment.
Since the start of the large scale deployment of nuclear energy, disposal of the long lived isotopes has been an issue that has had a priority in most nuclear countries.
A typical issue associated to the spent fuel management is represented by the spent fuel radiotoxicity.
In Fig. 3 it is shown the evolution of the radiotoxic inventory, expressed in sievert per tonne of initial heavy metal (uranium) (Sv/ihmt) of UOX spent fuel unloaded at 60 GW d/t, versus time (years).
Numerous studies performed in the last three decades have shown that three major spent fuel and nuclear waste management options can be envisaged as indicated in Fig. 4 (see e.g. Ref.1).
The P&T option refers to the "Partitioning and Transmutation" strategy (see, among many others, Ref.2), where the FFH system, as other source driven fission systems, can play a significant role.

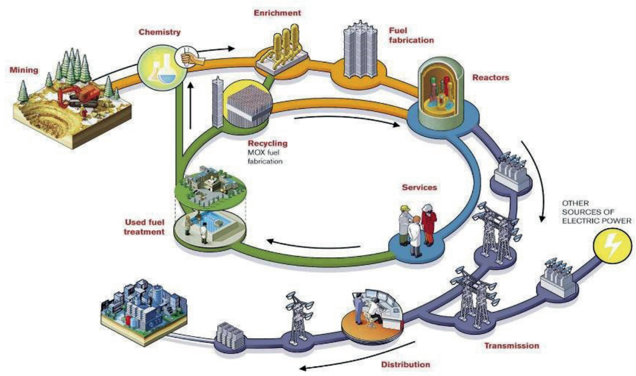



*Fig. 1 - A schematic view of the nuclear fuel cycle*

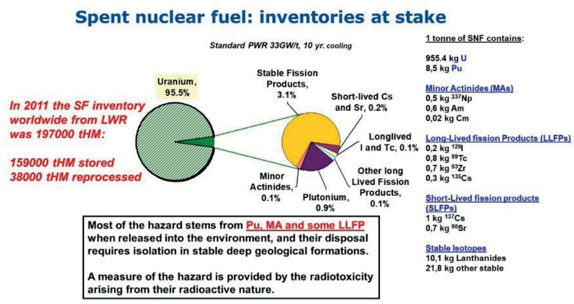



*Fig. 2 - The spent fuel issue: inventories at fuel unloading*

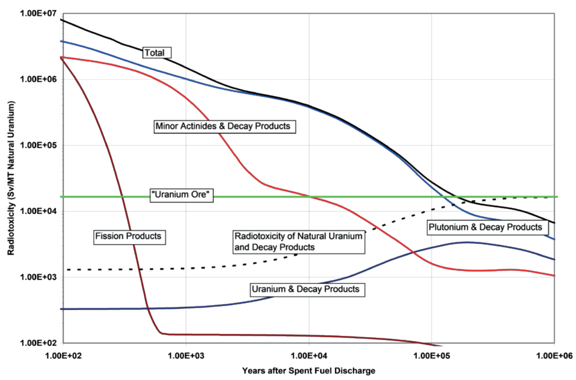



*Fig. 3 - The evolution of the spent fuel radiotoxicity inventory with time*

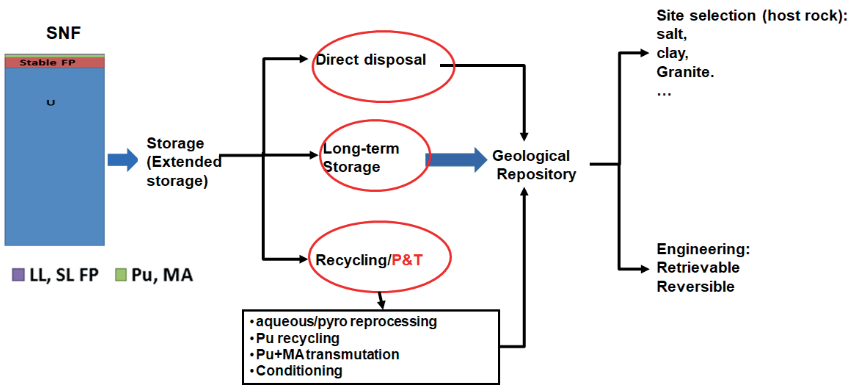



*Fig. 4 - Three major options for the spent fuel management*

## Key R&D issues in P&T

As far as key issues/challenges in P&T, one can summarize (Ref.3) them as follows:

*Transmutation-Fuel Fabrication*

- High Minor Actinide (MA) content
- Inert Matrix fuel (no Uranium support) e.g. for ADS and targets
- Performance under irradiation

*Transmutation systems (Fast Reactors; Source driven systems: ADS, FFH, Fig. 5)*

- Safety assessment (e.g. in case of very high MA content)
- Neutronics (e.g. reactivity coefficients) and transmutation yield
- Sub-criticality control (in subcritical systems)

*Partitioning of actinides from spent fuel*

- High separation efficiency
- Aqueous methods with newly developed molecules
- Pyrochemical methods (e.g. for highly active spent transmutation fuel)

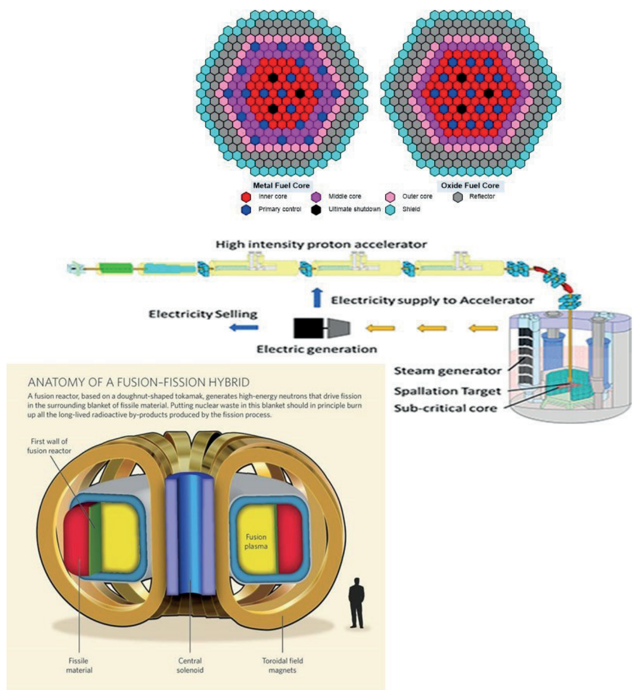



*Fig. 5 - A schematic view of an ADS and of a FFH system*

Some major challenges are related to the use of source driven subcritical systems, namely:

- The physics of the system where a delicate balance between the source neutron production and the neutron multiplication of the nuclear component determines e.g. the dynamic behavior of the system
- Source reliability (continuous operation required)
- The safety of the source driven systems remains of utmost concern (residual heat related accidents, potential overpower transients etc)
- Engineering of the barriers between the source and the nuclear component
- The energy "cost" of the neutrons (characterized by ratio of the energy that should be spent to produce the source to the total fission energy to be used e.g. for transmutation) that should be kept to a minimum while keeping the fission blanket sub-critical (e.g. K~0.95)

Experimental validation of each concept is definitely required. Examples of subcritical core parameters R to be validated are as follows:

- Sub-criticality and its monitoring (e.g. evolution in time)
- Reactivity coefficients (i.e. reactivity variations due to system parameter variations: coolant void coefficient, temperature and Doppler effect etc.)
- Power (neutron flux) spatial distributions
- Neutron spectrum assessment and tuning.

For each parameter R, uncertainties have to be assessed together with target accuracies for *safety and system optimization.*

Strategies have to be envisaged for *uncertainty reduction* in order to meet design target requirements.

As an example of concept features validation in this field, one can mention the experiments for the validation of the Accelerator Driven System concept by component (Ref. 4) as represented schematically in Fig. 6a and 6b.

In practice, the MUSE experiment at the MASURCA facility in Cadarache (Ref. 5) has been the first experimental validation of the neutronics of a source driven sub-critical system (at zero power, C) by the coupling of a deuton accelerator to a (d,T) target at the center of a fast neutron sub-critical core, see Fig: 7.

A similar approach can be envisaged for a FFH system as indicated in Fig. 8.

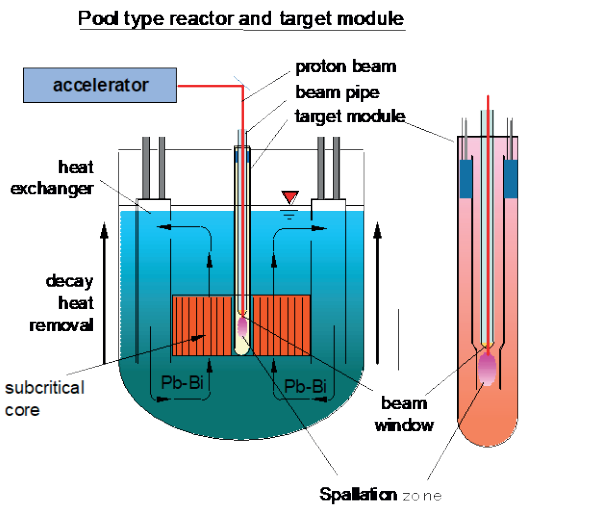



*Fig. 6a - The validation "by component" of an ADS system: the components*

~1 GeV protons, current ~10-20 mA
Subcritical core:
Subcriticality monitoring
MA-dominated fuel
Spallation target:
solid or liquid metal:
W, Pb-Bi, Pb
Coolant:
Pb-Bi, Pb, He, Na

**Schematically:**

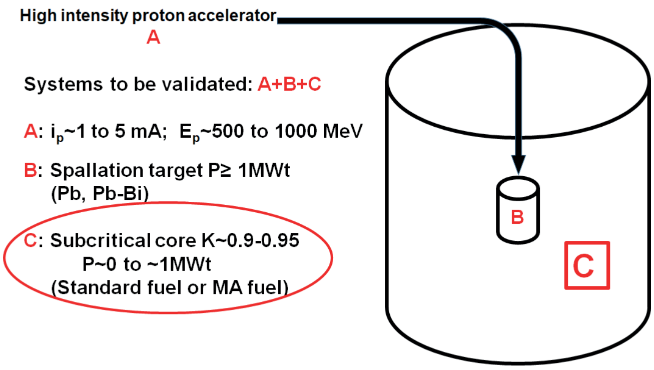



*Fig. 6b - The validation "by component" of an ADS system: a schematic example*

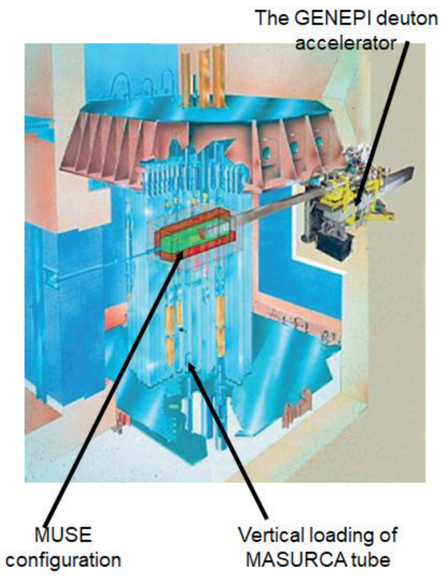



*Fig. 7 - The MUSE experiment at MASURCA*

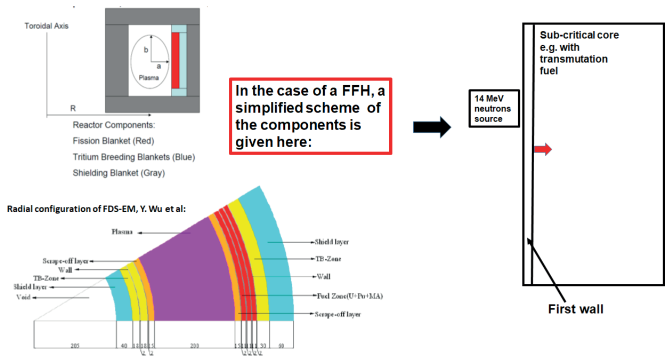



*Fig. 8 - The concept of validation by component applied to a FFH system*

## Sensitivity/Uncertainty Analysis

Sensitivity/uncertainty analysis is a necessary tool to allow the most effective validation of a system, in particular in terms of uncertainty reduction (Ref.6).
Sensitivity coefficients S to input parameters p are the key quantities that have to be evaluated for each integral parameter of interest R, as indicated previously:

$$S = \frac{\partial R}{\partial p}$$

They are determined using different methodologies e.g. Generalized Perturbation Theory (GPT) or direct derivatives evaluation
a) Uncertainty Quantification
In uncertainty assessment, the sensitivity coefficients are multiplied by the uncertainties of the input parameters p in order to obtain the uncertainty of the targeted parameter of interest.
Recently science-based approaches have been developed to produce covariance data for nuclear parameters, and correlations in energy and among the different input parameters, like reactions and isotopes, can also be provided.

$$\Delta R^2 = S_R^+ D S_R$$

where ΔR is the uncertainty, SR are the sensitivity coefficients arrays, and D is the covariance matrix.
b) Experiment « representativity » factors (Ref.6)
To make the best use of experiments, one can use the concept of "representativity" to better assess the potential for uncertainty reduction of any specific experiment. In particular, "representativity factors for each experiment have been defined in Ref.6.
c) "Representative" experiments for validation
For each core integral parameter or for a set of integral parameters, sensitivity profiles can/should be evaluated. If uncertainties (variance-covariance data) are available for the "input" parameters (cross sections, decay data etc), a rigorous procedure allows:

- to define quantitatively the "representativity" of a (or a series of) experiment(s) with respect to a (or a series of) reference design parameter(s), and

- to define the expected reduction of uncertainty on each design parameter as a result of the performance of the experiment(s). This can be achieved if experimental uncertainties are as low as possible.
- to define the domain of applicability of the validation.

## Possible validation strategy for the fission subcritical component of a FFH

The idea is to couple an existing reactor capable of demonstrating crucial features of power operation (notably power coefficients of reactivity) in a subcritical mode with an external 14 MeV neutron source.

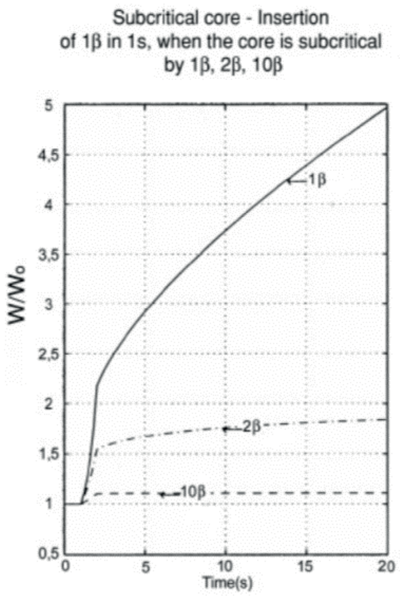
It is felt that such experiment should be able to demonstrate and test many features of operability of a FFH system such as relations among reactivity (including feedbacks), source spectrum, source importance and behavior at power.

A key requirement is to keep experimental uncertainties as low as possible.

The *evolution of the power* with time and the related variation of the temperature is associated to the *variation of the reactivity* (Doppler reactivity effect, fuel expansion reactivity, reactivity due to the material concentrations in the core, including the coolant etc.). These reactivity effects (*feedback reactivity effects*) are essential for the safety of a critical reactor. In a subcritical core, the feedback reactivity effects are of different relevance according to the level of subcriticality (Ref. 7).

In fact for a core deeply subcritical, the dynamic behavior is dominated by the external source and its variation in time.

Closer to criticality, the feedback effects become more important and the behavior of the core is approaching that of the corresponding critical core as shown in Fig. 9 (Refs. 7 and 8).



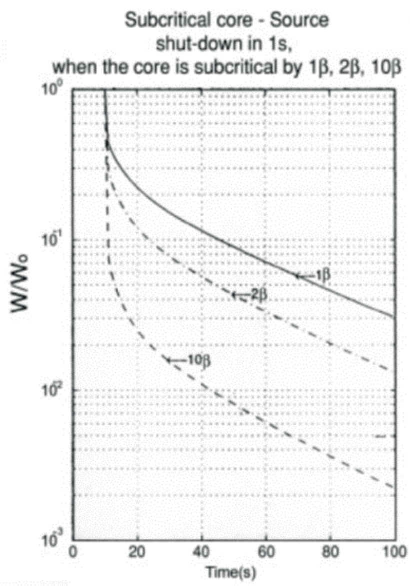




*Fig. 9. Time behavior of the power level with different reactivity insertions at different subcriticality levels*

In a very simplified way, if the core is subcritical by -10 β, a feedback reactivity equal to ± 1 β, induces a ± 10 % variation of power and a ± 50 % variation of power if the system is subcritical by - 2 β.

In a critical reactor + 1 β reactivity insertion makes the reactor prompt critical and - 1 β stops the chain reaction.

In view of the definition of an "optimal" level of subcriticality, it is of high relevance to verify the transition of the behavior of the subcritical system from a "source-dominated" to a "feed-back dominated" regime.

One can envisage a parametric series of experiments, e.g. in a TRIGA reactor (see paper by M. Carta et al at this workshop, Ref.9), investigating e.g.

- Different levels of subcriticality and the presence of a Tritium Breeder (TB) layer simulation
- Different materials surrounding a (D,T) source
- Different type of transients (reactivity insertions)
- Etc.

And it is possible to characterize/validate:

- The neutron spectrum at different positions
- the flux (power) spatial distributions
- o the reactivity effects induced by selected core perturbations
- Etc……………..

« Representativity » studies as indicated before, can be performed, using any preliminary FFH system as reference system.

## Conclusions

The experimental validation of a FFH presents challenges and an approach « by component » can be envisaged. This type of validation is complementary to global type of validation experiments, since possible source of uncertainties will be clearly pointed out and associated to specific components.

In particular, several aspects of the fission component of a FFH need careful experimental validation as summarized here below:

➢ The choice of the subcriticality level plays an important role and has impact on the safety case.

➢ the tritium breeding effectiveness should be carefully assessed

➢ Uncertainty quantification can suggest the most representative experiments to be performed in order to meet design requirements

Finally, the principle of a possible and realistic experimental approach to the validation of the source driven fission component of a FFH has been indicated.

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# The experimental validation of the Fusion-Fission concept using a 'tokamak fusion blanket' coupled with a standard fission system

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## Abstract

Fusion-Fission Hybrid devices (FFH) could have a potential role in the management of fission reactor wastes, with in principle some advantage over the comparable Accelerator Driven systems, devoted to the same objective. The validation of the concept poses major challenges in the area of fusion "source" development. However, the physics of the multiplying and transmutation blanket has also to be carefully understood and validated. Simple experiments can be envisaged for that purpose. Some of these experiments have already been performed in a different research framework (i.e. validation of the ADS concepts) and could constitute a very first data base. Moreover, a new series of similar experiments could be planned. In practice in a 'new' formulation of the FFH concept validation while maintaining the basic approach, the Accelerator (used in the ADS based systems) can be substituted by a system generating suitable neutron spectrum similar to the spectrum generated by a tokamak fusion neutron source. The paper describes a possible experiment (TRIGA-FFH) along the lines of MUSE4, but with a different 'tokamak neutron source', and possibly using the TRIGA facility at ENEA-Casaccia. The neutron spectra obtainable in a tokamak based fusion reactor are simulated, and the FFH system evaluation consist in two variants :

- type I experiment : 'tokamak fusion blanket' + standard fission system ;
- type II experiment: 'tokamak fusion blanket' + standard fission system + a blanket containing some fuel sample for transmutation.

The diagnostics system for the measurements of subcriticality successfully used in MUSE-4 experiment can be used in this TRIGA-FFH experiment, together with power spatial distributions, spectrum indexes for spectrum characterization at different positions and possibly neutron "importance" measurements. Parametric experimental studies with different materials and arrangements, are also foreseen.

## Keywords

Fusion Neutron Sources, Subcritical Systems

# Simple FFH pilot experiment model based on DTT machine

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## Abstract

A Fusion-Fission Hybrid reactor (FFH) has two main subsystems: a fusion reactor, acting as a neutron source and a sub-critical fission blanket, composed by an assembly of nuclear fuel, acting mainly as a power amplifier.
In this context, a pilot experiment could be considered as a relatively low power FFH, this means that the fusion reactor characteristics could be less demanding, because a significant part of thermal power would be produced by the fission blanket thanks to its amplification properties. This fact could imply, for example, a higher fusion duty cycle, which is fundamental to reduce the thermal material stress in the fission blanket components, due to the fast power variation.
In this work, we have considered a tokamak reactor with the same geometric and materials characteristics of DTT (Divertor Tokamak Test facility).

## Starting point: fusion machine

In order to study a possible hybrid reactor configuration, we have considered, as a starting point, the DTT tokamak [1] which will be built in the ENEA Frascati research center as a divertor test machine as a support for ITER project [2].
DTT is a tokamak inducing D-D fusion reactions with the following main characteristics shown in Fig. 1a and Fig. 1b:

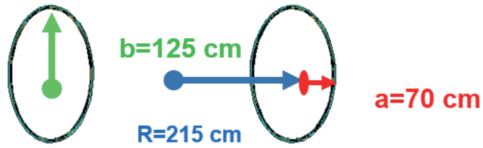
- major radius R=2.15 m
- aspect ratio A=3.1 (A=R/a)
- elongation 1.8
- toroidal field B=6 T
- plasma current $I_p$=6 MA
- additional power $P_{Tot}$=45 MW

The first DTT wall has been simulated by using the following effective material thicknesses:

- Tungsten: 0.5 cm
- Copper: 2 cm
- Water: 0.5 cm
- Steel: 3 cm

The emerging neutrons after the first wall have the energy spectrum reported in Fig. 2.

(a)




(b)

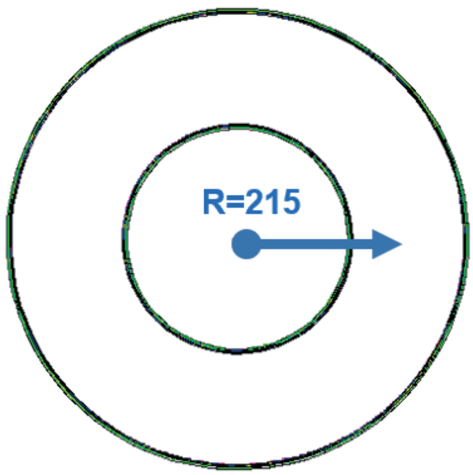



*Fig. 1 - Side (a) and top (b) schematic views of DTT tokamak with the relative geometric characteristics and dimensions*

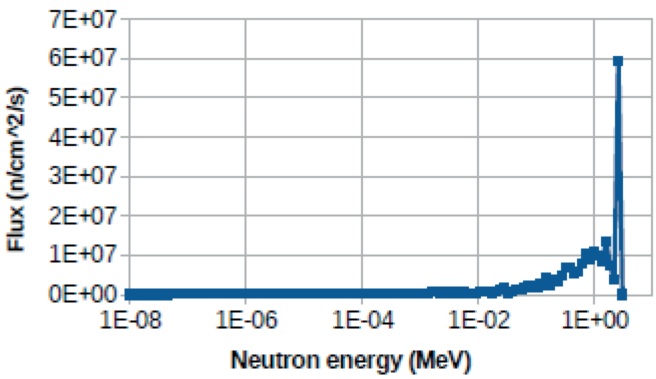



*Fig. 2 - Energy spectrum of neutrons emerging from first DTT wall*

## Fusion-fission pilot experiment

Starting from the above tokamak characteristics, a modular fission blanket generating a thermal power of about 20 kW and surrounding only a part of the external section of the torus has been designed (Fig. 3) by considering three different configurations as described in the following paragraphs. Since this is a pilot experiment model, the considered hybrid system is based on a DD fusion machine. The fission blanket is based on a relatively simple and known system with a low power level (20 kW) and a low effective multiplication factor (between 0.83 and 0.86 in the different configurations) to ensure a high safety level. Moreover, the fission blanket gives a power amplification factor of about 4 since only half of the total power produced in the tokamak (10 kW) comes from the d +d → $^3$He + n production branch. The other equiprobable production channel, d + d → $^3$H + p, is not useful for our purposes, because no source neutrons are produced.

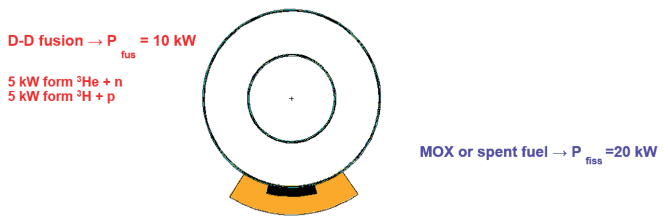



*Fig. 3 - Schematic view of the fusion-fission hybrid system. The fission core is represented by the black region (fuel rods) and the reflector by the yellow region*

### *MOX lead configuration*

This fission blanket configuration (Figure 4) is formed by 0.357 cm radius, 262 cm height and 0.068 cm steel cladded [3] MOX fuel [4] rods (table 1), completely embedded in a solid lead matrix. The cooling system is provided by steel water pipes (R=0.25 cm, 0,05 cm thickness) [5]. The 25 cm thick and 150 cm long fission core is surrounded by a 50 cm thick lead reflector. The effective multiplication factor for this configuration is $k_{eff}$=0.86.

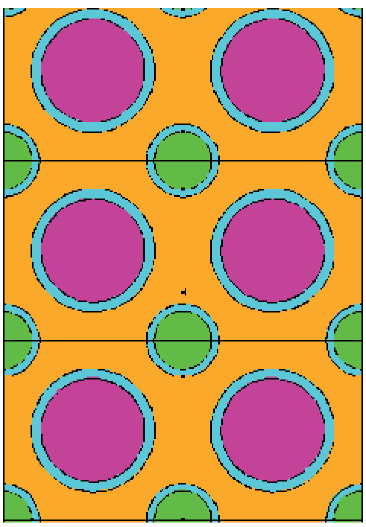


*Fig. 4 - Core lattice for MOX lead configuration: MOX fuel (purple), water (green), steel cladding (light blue) and lead (yellow)*

| Element | Concentration (%) |
|---|---|
| Uranium | 78 |
| Plutonium | 22 |

*Table 1 - MOX fuel vector*

### *MOX water configuration*

This fission blanket configuration is formed by 0.45 cm radius and 262 cm height 0.068 cm steel cladded MOX fuel rods (Table 1) embedded in water acting both as a neutron moderator and as a coolant as shown in Fig. 5. The fission core 7 cm thick and 150 cm long, is surrounded by a 100 cm thick graphite reflector. The effective multiplication factor for this configuration is $k_{eff}$=0.83.

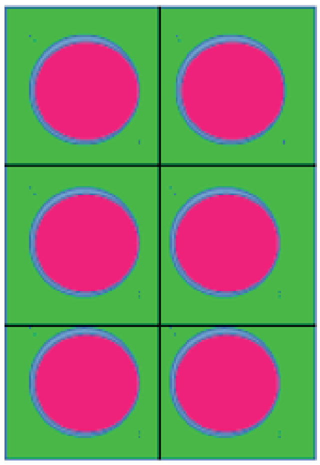


*Fig. 5 - Core lattice for MOX water configuration: MOX fuel (purple), water (green), steel cladding (light blue)*

### *Spent fuel water configuration*

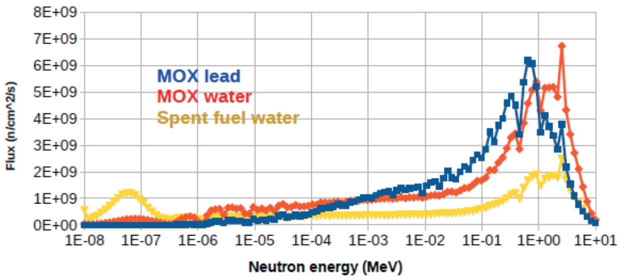
This fission blanket configuration is formed by 0.45 cm radius and 262 cm height 0.068 cm steel cladded spent fuel rods [6] (Table 2) embedded in water acting both as a neutron moderator and as a coolant. The core lattice has the same geometry of the MOX water configuration shown in Fig. 5. The 35 cm thick and 150 cm long fission core is surrounded by a 100 cm thick graphite reflector. The effective multiplication factor for this configuration is $k_{eff}$=0.86.

| *Element(s)* | *Concentration (%)* |
|---|---|
| Uranium | 95.53 |
| Plutonium | 0.83 |
| Minor actinides | 0.11 |
| Long lived fission products | 0.19 |
| Medium lived fission products | 0.16 |
| Stable isotopes | 3.18 |

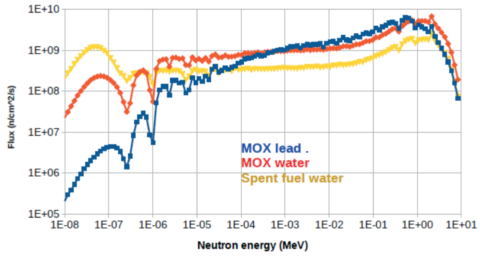
*Table 2 - Spent fuel vector (standard PWR 33 GWd/ton after a cooling time of 10 yr)*

### *In-core neutron energy spectra*

In Figures 6 a, b the energy flux distributions for the three previous mentioned configurations are reported.




(a)




(b)

*Fig. 6 - Neutron energy distribution (log-lin (a) and log-log (b)) for MOX-lead configuration (blue line), MOX water configuration (red line) and spent fuel-water configuration (yellow line)*

The previous figure clarifies the differences between the three considered configurations: the ration between the fast flux component (above 0.5 MeV) and the slow component (below 1 eV) gives an idea of the "hardness" of each spectra. This ration is higher in the MOX-lead configuration while is lower in spent fuel-water configuration because the slow component is higher with respect to the others one as clearly shown in Table 3.

| Configuration | Integral $\Phi$ (n/cm $^2$/s) | $\Phi$ < 1eV (%) | $\Phi$ > 0.5 MeV (%) | $\Phi$ > 0.5 MeV/$\Phi$ < 1eV |
|---|---|---|---|---|
| MOX-lead | 1,45 x $10^{11}$ | 0.10 | 37.28 | 361.7 |
| MOX-water | 1,52 x $10^{11}$ | 2.59 | 45.20 | 17.4 |
| Spent-fuel-water | 7.15 x $10^{10}$ | 21.33 | 34.01 | 1,6 |

*Table 3 - Integral flux values and percentage below 1 eV and above 0,5 MeV for the three considered configurations*

## Conclusions

A low fusion-fission hybrid pilot experiment based on DTT machine has been studied. The aim of these calculations is to investigate the feasibility of a preliminary hybrid configuration on a fusion machine that represents a typical Tokamak with the presently manageble level of complexity. The subcriticality of the fission part gives the possibility to choose among a large number of fission fuels as shown in the calcutions where the most representative fuel combinations are simulated. Three low power fission blanket configurations with different kind of fuel (MOX/spent fuel) have been considered in order to evaluate the effects of different materials (lead and water) from a neutronic point of view. The value of the energy amplification factor, even in a research machine, not devoted to energy production, is about 4 for each considered configuration.

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# Impact of the neutron energy spectrum variation on fission and delayed neutron yields in hybrid reactors

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## Abstract

Fusion-fission reactors hybrid (FFH) are studied since promise an almost inexhaustible source of clean and safe energy. Among the many technical problems to be studied the knowledge of the power produced in the different fissile regions represents one of the most important issues. However, the different zones forming the reactor characterize for large neutron energy spectra variation (i.e. from the fusion one up to degraded neutron energy spectra). Since the mass chain fission yields (CFY), that is the probability to get a certain fission fragment of mass A from the fission event, depend upon neutron energy these neutron spectra variations impact on the production of CFY. The variation of the CFY, in turn, affects the production of the delayed neutrons emitters and thus the reactivity. It is thus important to consider the effects that neutron spectra variation can have on the reactivity when the $K_{eff}$ is close to unity. In this work, some experimental data concerning the CFY variation with the neutron energy spectrum, measured at TAPIRO fast source reactor of ENEA Casaccia, are first presented. These measurements show the CFY variation with the neutron spectrum is well measurable. To follow, the delayed neutron parameters (using the Keeping six groups approximation) are calculated for the case of the JET tokamak operated with DD and DT plasma and compared to the parameters routinely used for a fission reactor. Important differences are envisaged which can impact on the $K_{eff}$ of FFH systems.

## Keywords

Hybrid fusion-fission systems, fission yields, delayed neutrons parameters, $K_{eff}$, reactor reactivity

## Introduction

Fusion-fission hybrid reactors (FFH) are studied since promise an almost inexhaustible source of clean and safe energy as well as to burn the nuclear waste (actinides) of the present fission reactors fuel-cycle. The issues to be considered and solved for realizing FFH are many and complexes. It is necessary, e.g. the availability of a reliable fusion neutron source (tokamak) able to operate at least in long lasting mode (i.e. > 3000 sec). Furthermore, owing to the heterogeneity of materials and zones typical of FFH systems, the variation of the neutron flux energy spectrum is to be considered. In fact, if the FFH reactor is operating with $K_{eff}$ close to unity the reactor stability has to be studied because the neutron energy spectrum variation affects the delayed neutron emission which, in turn, modify the $K_{eff}$. It worth recalling that in FFH systems the neutron spectrum varies from the fusion one, next to the first wall of the tokamak (nominally 14.1 MeV neutrons), up to very degraded neutron energy spectra, e.g. in the breeding blanket zone. The total *delayed neutron yield* ($\nu_{Tot}$) of most of fissile isotopes varies up to a factor of two from thermal neutron energy up to 14 MeV. To better understand the discussion above let consider the basic point of nuclear fission.

The fission process produces two fragments and two or more so called prompt neutrons ($\nu_p$) emitted at the time of the fission event. The main features of the fission event are: a) the two fragments are (usually) unequal in mass, that is their distribution is *asymmetric* (Fig. 1a); b) we can distinguish in between *Heavy Fragments* with mass number AH in the range $130 \le A_H \le 145$ and *Light Fragment* with $A_L$ in the range $90 \le A_L \le 105$. At low or thermal neutron energy the mass chain or cumulative fission yield (CFY) curve is represented by the characteristic *saddle* curve in Fig. 1a with two peaks at atomic mass $A_L \sim 90$ and $A_H \sim 135$. Therefore, already moving from thermal neutron energy up to 14 MeV (which is the energy range of interest for FFH systems) the differences in the FY production are not negligible (see Fig. 1a). The curves in Fig. 1a can be explained by modern fission theories [1]. For the sake of the present paper, we will focus our attention to the nuclides (also called *precursors*) which are responsible for the *delayed neutron emission* that, as it is well known, plays a fundamental role in the control of a nuclear reactor.

It is known that at neutron energy typically above 4-5 MeV an important role is played by the onset of the so-called *second, third, etc. chance* fission (Fig. 1b), that is the emission of two, three, etc. neutrons prior the fission event (within $10^{-15}$ s from the fission event), this is called

*n-chance* fission. The fission event occurring at energy lower than the threshold of the second chance fission (e.g. at thermal neutron energies) is called *first chance fission*. From experiments, it is found that the CFY curve changes with the neutron energy (Fig. 1a); this is correlated to the onset of the n-chance fission [1]. The onset of the various multi-chance fission events is well evident in the fission cross section ($\sigma_f$), it is observed that regardless of the fissile isotope, $\sigma_f$ increases with the onset of the n-chance fission (e.g. see Fig. 1c).
The CFY variation with the neutron energy can have important consequence for the reactor stability since the production of *precursors* is modified. This, in turn, will affect the delayed neutron emission in terms of both emission time (delay time) and emission intensity or yield. Since the delayed neutrons (DN) are responsible for the reactor reactivity a variation in their production yield and decay rate will impact on the $K_{eff}$ of the reactor. The neutron energy spectra variation is thus affecting the reactivity of the different zones into which the FFH reactor is divided. Furthermore, these effects are not homogenously distributed in the reactor volume. It is important to study the effect of neutron energy spectra variation on the production of precursors and thus on the delayed neutron emission especially for FFH reactor design assuming $K_{eff}$ very close to unity (e.g. 0.98).
The paper reports experimental evidence of the CFY variation reporting the CFY data measured at TAPIRO fast source reactor [2] in different fast neutron spectra. The paper then calculates the variation of the delayed neutron emission as consequence of the FY variation with the neutron energy. An example of calculation of delayed neutron parameters at different neutron energies relevant to FFH systems is also addressed.

## Fission Yields versus neutron energy

Experimental CFY data measured at different monoenergetic neutron energies are available in literature [3-6] for the various fissile actinides in between 2.0 MeV and 15 MeV (range of interest to FFH).Variation from 10% up to 40% respect to the CFY measured in thermal neutron flux are observed for the most important nuclides which results to be delayed neutron (DN) emitters (also called precursors). For a reactor system the *neutron energy spectrum averaged mass chain fission yield* (CFY) is of interest since a weighting effect due to the neutron spectrum is affecting the production of precursors. Measurements of CFY at TAPIRO Fast Neutron Source Reactor were performed for $^{235}U$, $^{238}U$, $^{232}Th$ and $^{239}Pu$. However, owing to the limited number of pages available, only some representative results are reported here after.

### *Fission Yield measurements at TAPIRO reactor*

TAPIRO reactor has a core made of metallic $^{235}U$, 93,5% enriched, a power of 5 kW and maximum neutron flux, at the core centre, of 7.0E+12 ncm$^{-2}$s$^{-1}$. At the core centre an almost $^{235}U$ fission-like neutron spectrum is present. The core is crossed by an 8 mm diameter channel and

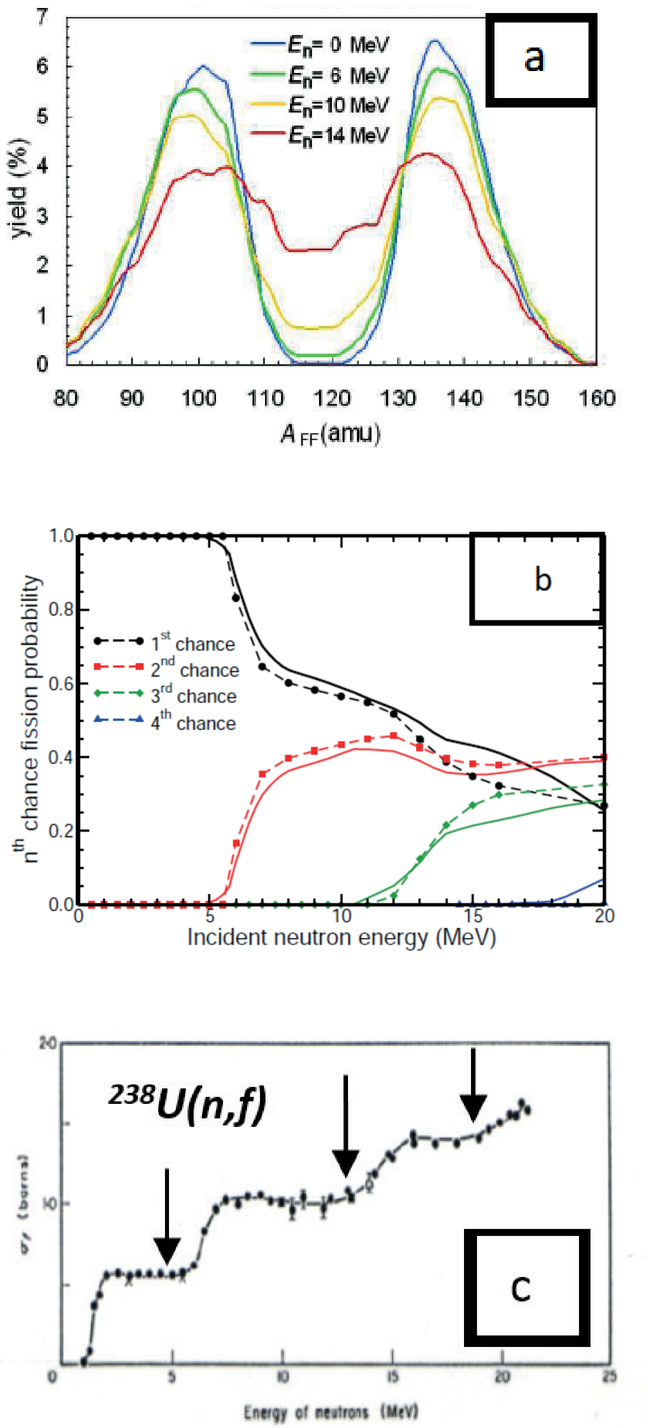


*Fig. 1 - a) Mass chain fission yields for $^{239}U(n,f)$ at different neutron energies; b) n-th chance fission probability versus incident neutron energy; c) Fission cross section for $^{238}U(n,f)$. The onset of the first, second, etc. chance fission is shown by the arrows*

is surrounded by a copper reflector 30 cm thick in which shim and control rods are present. Metallic fission foils ($^{235}U$, $^{238}U$, $^{239}Pu$, $^{237}Np$) were located in different positions in the copper reflector as well as in the core centre and the reaction rate (Rj) of some selected fission products was measured by radiometric technique using absolutely calibrated HPGe and total uncertainty lower than ±3%. Absolutely calibrated micro fission chambers (FC), with fissile deposits as the used fissile foils, were employed to determine the fission rate (Ffc) in each experimental position with an accuracy lower than ±2%. For a given fissile isotope, the measured cumulative mass chain fission yield for the j-th isotope is defined as:

$$CFY_j = R_j / F_K \qquad (1)$$

where:
$CFY_j$ = Cumulative Mass Chain Fission Yield of j-th isotope (e.g. $^{140}$Ba, etc.)
$R_j$ = Reaction rate of the j-th isotope (e.g. $^{140}$Ba, etc.)
$F_K$ = Fission Rate measured by the micro-fission chamber for the K-th fissile isotope

The neutron spectrum energy in each experimental position was identified by the *spectral index F8/F5* as measured by the micro FC. For the sake of this work, as an example of the results obtained at TAPIRO, Fig. 2 shows the variation of the CFY for some measured fission products of $^{235}$U(n,f) respect to literature CFY values measured at thermal neutron energy. The vertical bars are the errors on the fission yield ratio.

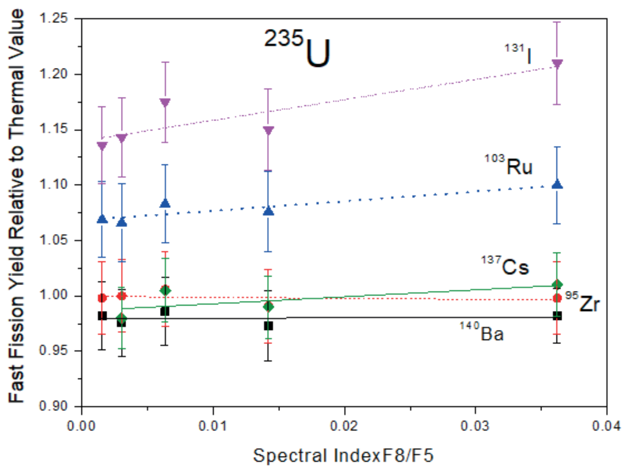


*Fig. 2 - Example of CFY variation with the neutron energy (relative to CFY in thermal flux). The higher the F8/F5 ratio the harder the neutron spectrum is*

As shown in Fig. 2 an almost linear variation of the CFY with the neutron energy is observed. Indeed, this linear variation is also found when measuring the CFY using monoenergetic neutrons [3,4]. If monoenergetic CFY data are available, we define the spectrum averaged CFY for a given fission product j:

$$CFY_j \frac{\int_0^{14.9\,MeV} Y_j(E)\sigma_f(E)\phi(E)dE}{\int_0^{14.9\,MeV} \sigma_f(E)\phi(E)dE} = \qquad (2)$$

here $Y_j(E)$ is the mono-energetic CFY for the *j-th* nuclide, $\sigma_f(E)$ is the fission cross-section of the given fissile material, $\Phi(E)$ is the neutron flux spectrum and E the neutron energy. By using literature data [3,4] for the mono-energetic fission yields, we used eq. (2) to calculate the CFY for the fission products measured in the TAPIRO neutron spectra and we compared these calculation with the CFY measured at TAPIRO. The comparison is reported in Fig. 3. The agreement is very good.

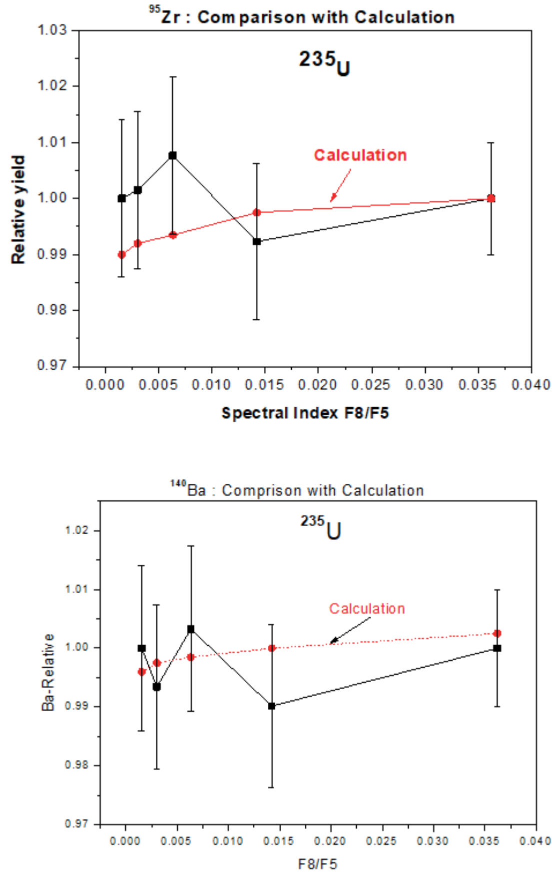


*Fig. 3 - Comparison between fast fission yields measured at TAPIRO (black points) and fast fission yields calculated using monoenergetic data weighted over the Tapiro neutron spectrum (red points). Data refer to $^{95}$Zr and $^{140}$Ba for $^{235}$U(n,f)*

## DN yield versus neutron energy

Once measured the effect of the neutron energy spectrum variation on the production yield of the fission products, we can use this information to evaluate the variation of delayed neutron emission related to the neutron spectrum variation. To do this we will calculate the *delayed neutron parameters* versus neutron energy. The delayed neutron decay of a precursor *j* is described by means of the *delayed neutron emission probability* $P_{nj}$ (i.e. the probability of delayed neutron decay per disintegration of the nucleus) which is known from measurements or can be calculated [7].
Another parameter for a precursor is the delayed neutron yield $\nu_j$, which represents the partial contribution of the j-th precursor to the total delayed neutron yield $\nu_{Tot}$. Among the hundreds of fission products, there are dozens of precursors with half-lives in the range ranging from a few ms up to ≈ 55 s. Despite the spread of half-lives, after Keepin [8], it is possible to describe the delayed neutron emission through six groups of precursors, each group being represented by its *decay constant* $\lambda_k$,

*group yield* $\nu_k$ and a proper *averaged neutron emission spectrum*. These six groups' parameters are derived by an appropriate weighting procedure:

$$\upsilon_k = \sum_{j=1}^{N} \upsilon_{kj} = \sum_{j=1}^{N} YC_{kj} P_{nkj} \qquad (3)$$

$$\upsilon_{Tot} = \sum_{k=1}^{6} \upsilon_k \qquad (3a)$$

and

$$\lambda_k = \frac{\sum_{j=1}^{N} YC_{kj} P_{nkj}\, \lambda_{kj}}{\sum_{j=1}^{N} YC_{kj} P_{nkj}} \qquad (4)$$

N is the number of precursors per each group k, $P_{nkj}$ is the delayed neutron emission probability [7], $YC_{kj}$ is the *cumulative fission product yield* for the j-th precursor in group k and $\lambda_{kj}$ *the decay constant* of precursor j in group k. Considering eq. (3) and (3a) we can define $\alpha_k = \nu_k/\nu_{Tot}$ which represents the *fractional delayed neutron yield* for the k-th group.

Eq. (3) can be explained intuitively. The $\nu_{kj}$ value for each precursor in the k-th group is given by the probability of finding that particular isotope given by $YC_{kj}$ multiplied by the associated neutron emission probability. $YC_{kj}$, in turn, can be written as $YC_j = Y_j{*}FCY_j(Z,A)$, that is the product of the *mass chain fission yield* $Y_j$ and the *fractional cumulative fission yield* $FCY_j(Z,A)$. $Y_j$ was discussed above and measured at TAPIRO, it depends on the neutron energy. $FCY_j(Z,A)$ represents the probability that within a particular isobaric chain, the delayed neutron emitter j is formed with atomic number Z. Thus $FCY_j$ is the sum of the probabilities of forming nuclei with charge < Z for a given mass A immediately after the fission. This point is important since the fission process and thus the fission products are depending upon the neutron energy, thus also FCY depends on the neutron energy. In the general case we can write FCY(E,Z,A).

The discussion above let us to calculate the delayed neutron parameters quite easily for monoenergetic neutrons and provides the method for calculating these parameters also in any neutron spectrum (as for the case of a hybrid system) if the needed quantities are known. In the case of a neutron spectrum, the eqs. (3) and (4) must be rewritten taking into account that $YC_{kj}$ is now averaged over the neutron spectrum (the k index is omitted):

$$< YC_j > = \frac{\int_0^{E_0} YC_k(E)\Phi(E)\sigma_f(E)dE}{\int_0^{E_0} \Phi(E)\sigma_f(E)dE} \qquad (5)$$

How to calculate eq. (5) is discussed here after. From literature, the $Y_j$ (E) data are available for almost all the relevant fission products being measured in monoenergetic neutron beams [3-6] as well as in thermal flux. From the available experimental data it is known that Y*j*(E) is almost linear with the neutron energy E. More complex is the calculation of the $FCY_j(E,Z,A)$ function for each precursor in a given neutron spectrum. The calculation of FCY(E,Z,A) requires the application of the *theory of nuclear systematic* (TNS) [9]. Experimentally, the charge distribution after fission is Gaussian [10] so its standard deviation $\sigma$ and its most probable charge $Z_p$ (central Z value) are the parameters of interest. Experimentally it is known that $\sigma$ is rather independent from the energy of the neutrons inducing fission while $Z_p$ is related to the neutron energy, the mass A and the Z value of the fission products. The calculation of $Z_p$ is possible using the TNS theory.

### Theory of nuclear systematic

TNS correlates the $Z_p$ values measured for the thermal neutron induced fission of $^{235}U$ to those for the other fissile materials. The correlation is established through the ratio $A_c/Z_c$ that is the ratio between the mass and the charge of the compound nucleus undergoing fission. Here after we present the results based upon the Nethaway method [11].

As already said, the charge distribution after fission is Gaussian and can be written as:

$$P(A,Z) = \frac{1}{(2\pi\sigma)^{1/2}} exp\,\langle -\frac{[Z-Z_p(A)]^2}{2\sigma^2} \rangle \qquad (6)$$

The fractional cumulative yield is thus:

$$\mathrm{FCY(E,Z,A)} = \int_0^{(Z+0.5)} \mathrm{P(A,Z',E)dZ'} \qquad (7)$$

According to Nethaway $Z_p(A,E)$ can be written as:

$$Z_p = 0.4153A - 1.19 - \beta_0(236 - 92\frac{A_c}{Z_c} + \Delta Z_p(A,E) \qquad (8)$$

and

$$\Delta Z_p(A,E) = a_0(\mathrm{Z_c} - 92) + \mathrm{b_0}(\mathrm{A_c} - 236) + \mathrm{c_0}(\mathrm{E}^* - 6.52) \qquad (9)$$

$\beta_0$, $a_0$, $b_0$, $c_0$, are constants, E* is the excitation energy of the fissile nucleus.

Example of the results for some calculated FCY(E,Z,A) values versus neutron energy are shown in Fig. 4.

Once eq. (7) is solved for the different fissile isotopes and the needed FCY(E,Z,A) data are available the next step is to calculate the $YC(E)_j = Y(E)_j{*}FCY_j(E,Z,A)$ data to be used in eq. (5) for calculating the neutron spectrum averaged $FC_j$ values ($<FC_j>$). The latter data are needed to solve eq. (3) and (4). In Table 1, as an example, the $FCY_j$ values calculated at 14 MeV for $^{238}U$(n,f) and $^{232}Th$(n,f) and referring to the first two groups of the Keepin scheme are reported and compared to literature data. The agreement is good.

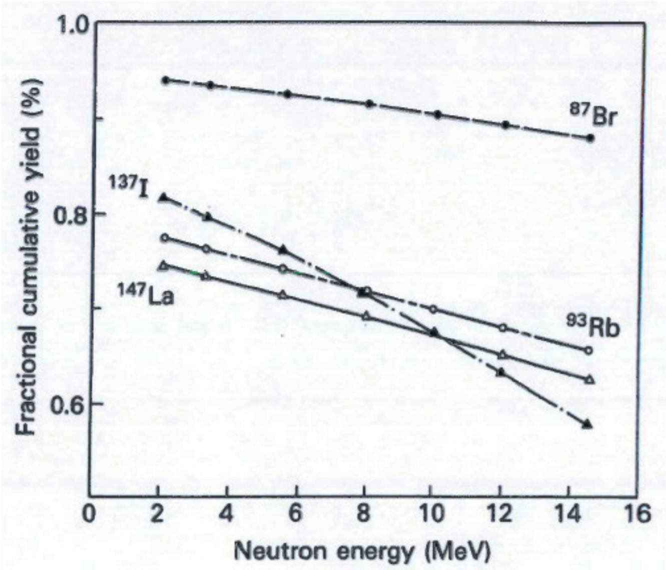


*Fig. 4 - Example of FCYs Vs. neutron energy calculated for*

*232Th using the Nethaway method*

To complete this work in Table 2 we report the delayed neutron parameters for $^{232}$Th(n,f) calculated for 14 MeV neutrons as well as for DD and DT neutron spectra using the DD ad DT first wall neutron spectra calculated for the JET tokamak. To note the important variation of the total delayed neutron yield $\nu_{Tot}$ with the neutron energy, in a DT spectrum it is almost half of the value at 2.0 MeV. It is known that $\nu_{Tot}$ is almost constant from thermal energy up to about 2-3 MeV [12,13] to reduce with the neutron energy increase. This is demonstrating the importance to consider the delayed neutron emission variation for an hybrid system owing to the importance that the total delayed neutron emission has on the reactivity of the system and thus on the $K_{eff}$.

The results in Table 2 and the discussion above let us to conclude that the neutron spectra variation in a FFH system should be considered if the system is operating with $K_{eff}$ close to unity and its effect on the delayed neutron emission evaluated.

| Group # | Isotope | $^{238}$U This Work | Ref.1 | Ratio | $^{232}$Th This Work | Ref.1 | Ratio |
|---|---|---|---|---|---|---|---|
| 1 | $^{87}$Br | 0.857 | 0.842 | 1.02 | 0.877 | 0.868 | 1.01 |
| 2 | $^{88}$Br | 0.682 | 0.726 | 0.94 | 0.687 | 0.622 | 1.10 |
| | $^{137}$I | 0.587 | 0.719 | 0.82 | 0.577 | 0.715 | 0.81 |
| | $^{136}$Te | 0.251 | 0.434 | 0.58 | 0.208 | 0.346 | 0.60 |
| | $^{134}$Sb | 0.158 | 0.39 | 0.41 | 0.118 | 0.293 | 0.40 |
| | $^{141}$Cs | 0.791 | 0.789 | 1.00 | 0.806 | 0.885 | 0.91 |

*Table 1 - Calculated FCY for $^{238}$U and $^{232}$Th compared to Literature data (first two groups)*

| Energy | Group | 1 | 2 | 3 | 4 | 5 | 6 | $n_{Tot}$ |
|---|---|---|---|---|---|---|---|---|
| 2.0 MeV | n | 16.5 | 75 | 82.6 | 259.1 | 44 | 40.1 | 517.5 (90.0) |
| | $\lambda$ (s$^{-1}$) | 0.0125 | 0.0349 | 0.1403 | 0.33 | 0.918 | 2.2927 | |
| DD spectrum | $\nu$ | 16.8 | 73.7 | 85.5 | 262.8 | 41.2 | 37.5 | 517.5 (90.0) |
| | $\lambda$ (s$^{-1}$) | 0.0125 | 0.0349 | 0.1392 | 0.3323 | 0.918 | 2.2927 | |
| 14 MeV | $\nu$ | 12.2 | 45.2 | 52.8 | 141.6 | 29.7 | 16.5 | 298.0 (47.0) |
| | $\lambda$ (s$^{-1}$) | 0.0125 | 0.0338 | 0.1382 | 0.3231 | 0.9038 | 0.2426 | |
| DT spectrum | $\nu$ | 13.5 | 50.8 | 60 | 164.3 | 22.6 | 12.7 | 324.0 (52.0) |
| | $\lambda$ (s$^{-1}$) | 0.0125 | 0.0344 | 0.1389 | 0.3254 | 0.9038 | 0.2426 | |

*Table 2 - $^{232}$Th calculated delayed neutron parameters for different neutron energies and spectra*

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*Session 5*

# Diagnostics and Controls for FFH

# Gamma ray based measurement of the fusion power in a Fusion-Fission Hybrid reactor

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## Abstract

A possible concept for a fusion-fission hybrid reactor is based on the combination of a high power tokamak as fusion neutron source and a surrounding blanket of subcritical fission material for neutron multiplication. A reliable measurement and monitoring of the power of the fusion source is essential for measuring the performance of the fusion neutron source and for the overall measurement of the Fusion Fission Hybrid (FFH) reactor power. On today's tokamak, the fusion power is measured by an absolute counting of the 14 MeV and 2.5 MeV neutrons in DT and D plasmas, respectively. This requires a calibration of the neutron detectors by placing a neutron source (either a radioisotope or a portable neutron generator) at different locations in the tokamak vessel and collecting data with the relevant neutron diagnostics. On a FFH the absolute measurement of the fusion power will be more complicated due to the more complex neutron field, that includes a contribution from neutrons born in the fission blanket. The latter is difficult to be told apart, unless absolute neutron counting measurements are accompanied by the determination of the neutron spectrum with a dedicated spectrometer. In this work, we will present instead an alternative method based on measurements of gamma rays for determining the DT fusion power in a FFH reactor. The D+T--> 4He+n is the main reaction in a DT plasma and also features a low (about 5·10-5) branching ratio reaction D(T,γ)5He, which emits gamma rays of mean energy equal to 16.6 MeV. From an absolute measurement of the 17 MeV gamma ray flux, and from a good knowledge of the corresponding cross section, one can in principle infer the 14 MeV neutron yield and thus the DT fusion power. This novel method has been recently proposed for high power DT burning plasmas as a second independent method to benchmark the traditional one based on neutron counting. It is promising also for a FFH since the background contribution induced by neutrons on the 17 MeV gamma ray measurements is expected to be negligible. In this presentation, we will present and address the issues that need to be solved in order to make the gamma ray method for the measurement of the fusion power a reliable alternative to the traditional one based on neutrons.

## Keywords

Fusion Neutron Source, Fusion power, Gamma rays

# The neutron detection techniques of Fussion Fission Hybrid blanket

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## Abstract

The neutron measurements play an essential role in understanding characteristics of the neutron fields of a Fussion Fission Hybrid (FFH) system, with the wide range of neutron intensities and energies of interest from thermal neutrons to above 14 MeV that have to be covered. The safe operation of a FFH blanket requires the continuous on line measurement of neutron flux levels at various locations. With fission reactors as reference, here we review the several main neutron detection techniques for ex-vessel and in-core power monitoring for fission reactors such as pressurized water reactors (PWRs) and materials testing reactors:

- gas proportional counters using BF3 fill gas as source range detectors;
- the boron-lined gamma-compensated ionization chambers with current mode for intermediate range;
- the boron-lined ionization chambers in the full power range.

Some kinds of self-powered detectors (SPDs) can be used to measure neutron flux levels above 1012 nv with good accuracy and a good signal-to-noise ratio. In recent years, as an excellent potential candidate for neutron detection and spectrometry, new silicon carbide (SiC) semiconductor neutron detectors based on a Schottky diode design have achieved rapid development thanks to their good performances in harsh environments characterized by high temperatures and intense neutrons and gamma radiation fields. We will discuss whether the above-mentioned technology can be adopted to characterize the neutron field in the FFH blanket.

## Keywords

Neutron detection, fission reactors, power monitoring, silicone carbide.

# Fusion devices as neutron sources for FFH (Fusion Fission Hybrid): readiness level and concept validation experiments

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## Abstract

The fusion neutron sources needed for FFH (Fusion-Fission Hybrid) devices are not available so far, and the blankets integrating the fusion and fission characteristics need to be projected and validated. The concrete validation of the FFH concept is needed. This paper is devoted to the definition of areas where it is urgent the work for the validation of the FFH concept. Starting from a definition of a general figure of a neutron source needed for FFH, the paper is devoted to: i) analysis of the technology readiness level (TRL, see ref.1) of Fusion sources and Fusion-Fission blankets and ii) possible experiments for the validation of the FFH concept. Limiting the analysis to tokamak FFH neutron sources, there are two important technologies in the early stage of development: i) the demonstration of a discharge many hours long (at least three hrs) at relevant plasma parameters, and tokamak continuous operation for many months; ii) study of divertor geometry and control in a high radiation environment. Moving to general FFH concept validation, the neutron spectra obtainable in a tokamak based fusion reactor can be optimised , and the concept validation of the FFH system can be carried out in : i) type I experiment: "tokamak fusion blanket"+ standard fission system; ii) type II experiment: 'tokamak fusion blanket' + standard fission system + a blanket containing some fuel sample for transmutation.

## Keywords

Fusion Neutron Sources, Subcritical Systems

## References

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# Comprehensive neutronics simulation program SuperMC for fusion and fission applications

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## Abstract

Super Multi-functional Calculation Program for Nuclear Design and Safety Evaluation, SuperMC, is a full-function neutronics simulation software system, including radiation transport depletion, activation and: dose calculations. The advanced capabilities of the latest: version, SuperMC3.2, include: CAD/Image-based accurate modeling, intelligent data analysis based on multi-D/multi-style visualization and network collaborative nuclear analysis on cloud computing platform

The whole process-of neutronics calculation, radiation transport, depletion, activation, and dose calculations are: inner-coupled. Series of advanced methods has been developed to accelerate radiation transport calculation, including hybrid Monte Carlo (MC) and deterministic methods, global weight window generator (GWWG), etc. By these methods, the calculation efficiency for ITER analysis: is enhanced by 637 times. In order to- solve the challenge of large memory consuming (TB-level) in radiation transport and depletion coupled high fidelity simulation, thread- level data decomposition parallel computing technique is developed.

SuperMC supports automatic conversion between CAD models or images and MC simulation models, and can accurately describe complex irregular geometries. The software supports multiple styles of data visualization, data and geometries mixed visualization, and results quantitative analysis. Integration nuclear analysis service of modeling, calculation and visualization on cloud computing platform is provided by which complex nuclear analysis can be conveniently performed

SuperMC has been verified and validated by: more than 2000 benchmark models and experiments including ICSBEP, IRPhEP, SINBAD, PWRS (BEAVRS, HM, TCA, etc.), fast reactors (BN600, IAEA-ADS, etc.), fusion reactors (ITER benchmark model, FDS-II, etc.), etc. The results of SuperMC agree well with reference calculation results and experiment results. It has been applied in over 40 mega nuclear engineering projects, such as HPR1000, ITER, etc.

## Keywords

SuperMC, whole process, neutronics calculation.

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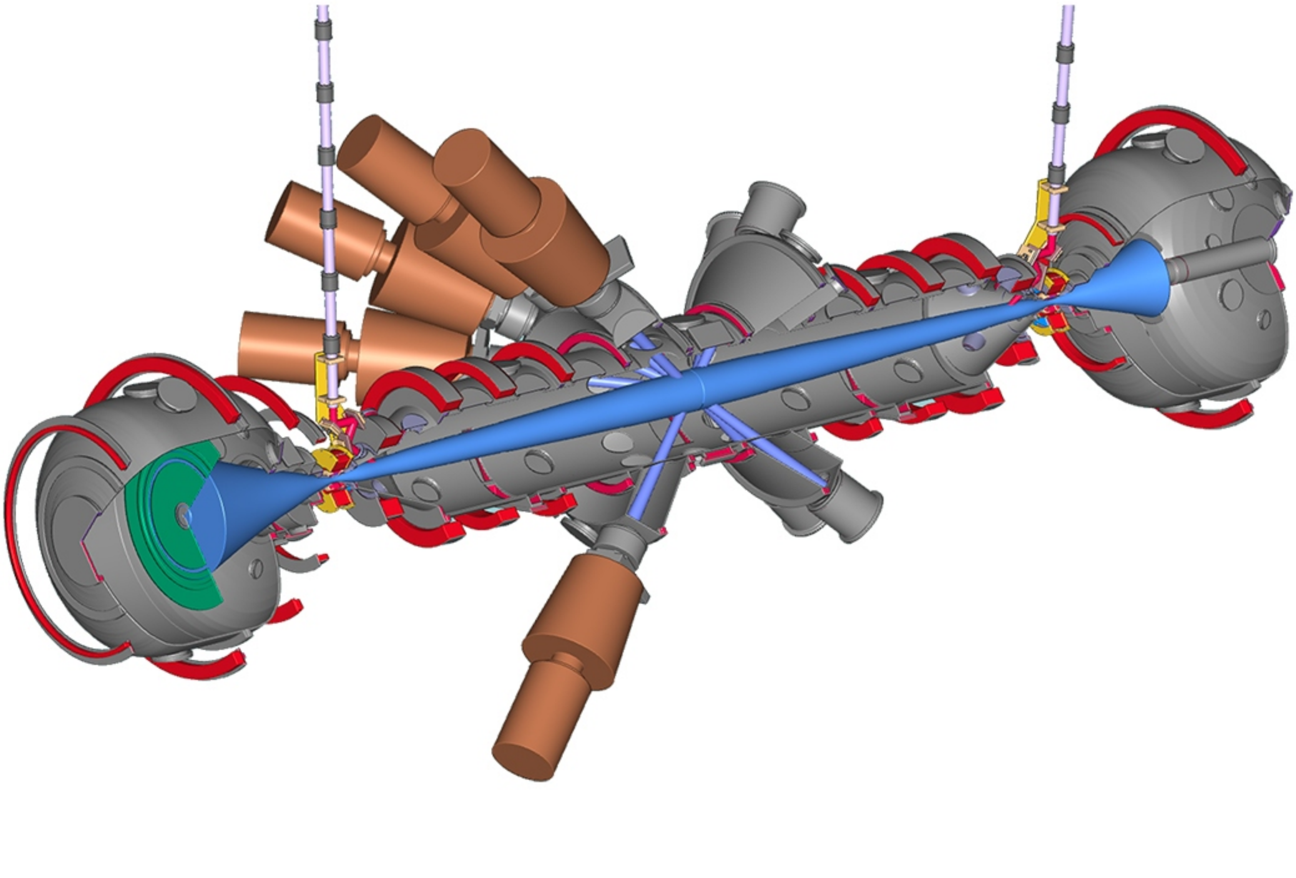

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