"*PIERO CALDIROLA*" *INTERNATIONAL CENTRE FOR THE PROMOTION OF SCIENCE*

AND INTERNATIONAL SCHOOL OF PLASMA PHYSICS

INTERNATIONAL WORKSHOP ON

FUSION FOR NEUTRONS
AND
SUB-CRITICAL NUCLEAR FISSION
FUNFI

VILLA MONASTERO, VARENNA, ITALY

SEPTEMBER 12-15, 2011

PROGRAMME AND ABSTRACTS

Monday, 12 September

8.00	Registration
08:45	Assembly
	Welcome & Introduction & Information
Chairman: Lontano	
09:30	R Goldston (T1)
10.30	COFFEE
11:00	M Salvatores (T2)
11.30	Y Wu (L1)
12.00	Q Huang, (O1)
12:55	Group Photo
13.00	LUNCH
Chairman: Wu	
15.00	Y Zheng (O2)
15.25	A Rineiski (T3)
16.15	M Gryaznevich (L2)
17.00	COFFEE
17.30	Poster presentations
19.30	Reception at the Hotel Royal Victoria

Science and Technology Challenges in the Development of Magnetic Fusion Energy Systems

Robert J. Goldston, Princeton University

Reading from the plasma core outwards, the key areas of science and technology that require development for practical magnetic fusion energy can be identified as:

1. **Confinement**
   As the plasma size increases to produce > 2 GW of thermal fusion power, will energy confinement simultaneously increase, supporting high fusion gain?

2. **Stability and consequences of instability**
   What instabilities will be encountered, how can they be controlled, and what will be their consequences, in large plasmas?

3. **Plasma heating, fueling and current drive**
   What are the most practical means to heat and fuel a fusion plasma and to sustain its electrical current in steady state?

4. **Plasma – material interactions**
   What plasma operational conditions and plasma-facing materials can accommodate the high heat load and particle fluence of a fusion system?

5. **Neutron – material interactions**
   What materials are needed to handle the anticipated high-energy neutron fluence, and what technologies will allow efficient breeding of tritium fuel?

6. **Magnet technology**
   Can magnet technology be developed that will allow higher power densities, and so more cost-effective fusion systems?

ITER is designed to provide the critical information in areas 1 and 2, and indeed key information should be available on these topics as soon as ITER's plasma current exceeds 10 MA with heating power exceeding ~30 MW. The key issues of integrating heating, fueling and current drive technologies with steady-state plasma operation (area 3) should be provided by ITER, EAST, JT-60SA and KSTAR. Area 4 will gain key insight from ITER, and other superconducting devices, but a strong R&D effort in technology development and plasma-testing outside of ITER will also be required. In area 5 ITER will provide information on initial operation of a variety of blanket technologies, but both a point neutron source and more aggressive component testing will be needed to prepare for commercial fusion. Since fusion power density varies as B^4, continuing R&D in magnet development holds considerable promise.

The resolution of the science and technology questions above must lead to a plasma configuration that is cost-effective, remotely maintainable, and capable of operation at high duty factor.

Neutronics for critical fission reactors and sub-critical fission in hybrids

Massimo Salvatores (CEA, Cadarache, France)

This tutorial lecture will introduce first a number of basic neutronics concepts:

- Fission and other neutron-nuclei interaction reactions
- Neutron flux and reaction rates
- Neutron mean free path
- Multiplication factor
- Delayed neutrons and neutron chain reaction control
- The transport of neutrons and the Boltzman equation

It will be successively discussed the notion of thermal and fast neutron reactors:

- Rationale for thermal or fast neutron reactors.
- Neutron moderation and cross section comparison
- Neutron balance for thermal and fast neutron reactors
- The concept of neutron consumption/fission
- The case of fast neutron reactors: neutron balance, neutron surplus, burners and breeders

Finally, the neutronics caracteristics of sub-critical reactors will be introduced and discussed:

- Neutron and energy balance in sub-critical systems
- Role of external source
- Importance of source neutrons

Overview of Hybrid Nuclear System Research Activities in China

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Abstract: Although the recent experiments and associated theoretical studies of fusion energy development have demonstrated the feasibility of fusion energy science, there will still be a long way to realize fusion energy application commercially and economically.

Along with the achieved and ongoing efforts to establish fusion as an energy source, especially first as neutron source, as there is a renewed interest in fusion-fission hybrid reactors, especially based on the latest progress in the construction and operation of the EAST (Experimental Advanced Superconducting Tokamak) in China and the International Thermonuclear Experimental Reactor (ITER). Three functions of fission blanket concepts based on viable technologies, i.e. the energy multiplier (EM) with the goal of energy production, the fuel breeder (FB) with the goal of fissile fuel breeding to support development of fission power plants, waste transmuter (WT) with the goal of transmutation of the long-lived nuclear wastes to solve the potential problem of fission power plants, had been used for the re-evaluation of feasibility, capability and safety & environmental potential of fusion-fission hybrid systems. On the basis of the results of the re-evaluation activity, a Fusion-Driven hybrid subcritical System (FDS) concept has been designed and proposed as Spent Fuel Burner (SFB) based on viable fusion and fission technologies. The plasma fusion driver can be designed based on relatively easy-achieved plasma parameters extrapolated from the successful operation of the existing fusion experimental devices such as the EAST in China and other tokamaks in the world, and the subcritical fission blanket can be designed based on the well-developed technology of fission power plants. To achieve the final goal of FDS-SFB application, an intermediate stage to develop a Multi-Functional eXperimental reactor (MFX) is designed to test multi-types of blanket modules before a FDS-SFB application is realized.

In the meanwhile, the Accelerator Driven System (ADS), a fast sub-critical reactor coupled to a proton accelerator, which would allow large quantities of nuclear waste to be burned through the transmutation process. Chinese Academy of Sciences has launched a project to develop ADS to demonstrate the technologies of transmutation of nuclear waste of commercial ADS. The development of experimental and DEMO reactor concepts named CLEAR and related R&D work for construction and operation of the reactors are underway. The project objective of China LEad-bismuth cooled Experimental Accelerator-driven Reactor (CLEAR-I) is to build a lead-bismuth cooled experimental reactor, which can be operated optionally on critical way (5-10MWth) or on subcritical way driven by spallation neutrons in about 7 years.

In this contribution, the proposed roadmap and testing strategy of fusion application and the planned roadmap of ADS development are presented based on the assessment and design analysis of the developed series hybrid system concepts as well as a summary of the hybrid system design and R&D activities in China.

Key words: Fusion; Hybrid; Subcritical; Accelerator Driven System

[1] Y. Wu, J. Jiang, M. Wang, et al., "A Practical Way to Fusion Application through Fusion-Fission Hybrid Systems". Invited presentation at the 15th International Conference on Emerging Nuclear Energy Systems (ICENES-15), May 15-19, 2011, San Francisco, USA.

[2] Y. Wu, J. Li, Y. Li, et al. An integrated multi-functional neutronics calculation and analysis code system: VisualBUS, Chinese J. Nuclear Science & Engineering, 27 (2007) 365-373.

[3] Y. Wu, FDS Team, "Conceptual design activities of FDS series fusion power plants in China," Fusion Eng. Des. 81 (2006) 2713-2718.

Design, Construction and Experiment of Liquid LiPb/PbBi Eutectic Loops for Advanced Nuclear Reactors in China

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Abstract: The liquid metal cooled reactor has been proposed as one of the promising advanced nuclear system. Lithium lead eutectic (LiPb) has been proposed as tritium breeder and coolant for fusion reactor, and lead bismuth eutectic (PbBi) as the candidate spallation target material and coolant for accelerator driven sub-critical systems (ADS) as well as coolant for fast neutron reactor. The liquid metal loops are the necessary experimental platforms to study the related technologies of liquid metals.

In order to validate the key technology for the liquid LiPb blanket of FDS series reactors, series of liquid LiPb loops named DRAGON series has been designed and built by FDS team. The thermal convection loop DRAGON-I (500ºC) and DRAGON-II (700ºC) have been built and operated for more than 10,000hrs and 5,000hrs, respectively. And the muntifunctional forced convection loop DRAGON-IV has been designed to investigate the compatibility of materials with liquid LiPb, the thermal-hydraulic behavior and the purification technology of flowing LiPb, etc.. Moreover, the forced convection thermo-hydraulic LiPb loop DRAGON-V has been designed to validate the DFLL-TBM technology before ITER. Besides, DRAGON-VI and DRAGON-VII have been designed for the auxiliary system for EAST-TBM and ITER-DFLL-TBM, respectively.

Base on the experience on the design and construction on DRAGON loops, series of liquid PbBi loops named KYLIN series has been designed and built or under construction. The first thermal convection PbBi loop KYLIN-I has been operated for more than 2,000hrs and some preliminary corrosion results for candidate nuclear materials were got and analyzed. And the muntifunctional forced convection loop KYLIN-II has been designed to investigate the compatibility of materials with liquid PbBi, the thermal-hydraulic behavior, the purification of PbBi, oxygen measurement and control technology for the flowing PbBi etc.. Moreover, the large forced convection PbBi loop KYLIN-III has been designed to serve as the thermal-hydraulics verification facility for ADS core and the thermo-mechanical verification facility for high power liquid PbBi target of the PbBi cooled reactor. Besides, DRAGON/KYLIN-ST and DRAGON/KYLIN-RT were also built for static and flowing corrosion experiments, respectively.

Based on these platforms, the corrosion experiments and the corrosion behavior of martensitic steel CLAM, T91 and austenitic stainless steel 316L were presented and analyzed as well in this paper.

Key words: Advanced nuclear reactor; LiPb loop; PbBi loop; Corrosion

The neutronics studies of fusion fission hybrid power reactor

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Abstract

The current fission energy technology faces up to the challenges of uranium resources shortage, long-lived wastes accumulation, nuclear safety and proliferation. The fusion energy technology seems to be much better and capable of finally solving such problems. However, it has been generally recognized that it will be a long way and hard work before fusion energy could be commercially and economically utilized. To find an economical way to utilize the fusion neutron, the fusion fission hybrid reactor is proposed and becomes more and more attractive.

In this paper, the neutronics conceptual design of hybrid power reactor is proposed. The ideas of three different types of fission blankets have been discussed, considering different fusion parameters of current experimental device, device can be easily obtained in the future and ITER level device. The sub-criticality, energy multiplication factor and tritium breeding ratio are evaluated as the criterion of design.

The neutronics analysis is implemented by a new code developed based on the DRAGON code, containing a new hybrid multi-group data library using WIMS-172 format, the DRAGON-based self-shielding, neutron transport and burnup calculation modules, and an extension calculation module for the evaluation parameters of hybrids. A one-dimensional spherical model is established to perform the conceptual design and analysis.

The conceptual design aims to find a feasible 1000MWe hybrid power reactor patterns. Based on it, the energy multiplication criterion varies with the change of fusion parameters, i.e., for the three type of fission blankets are 75, 25 and 8, respectively. The cycle length aims 5 years. In the cycle length, the TBR is required to be no less than 1.05. The mature oxides fuel and pressurize water coolant technology is adopt. The Li_2O with enriched ^6Li is applied as the tritium breeding material.

Three patterns are analyzed: 1) For the current fusion technology, i.e., the fusion power reaches 50MW and power gain approaches 1, only the reprocessed fuel is satisfying. Plutonium must be separated and the mixed oxide fuel with high plutonium (18%w/o) is required to achieve over 75 times energy multiplication. The burnable poison is applied to compensate the extra reactivity at the beginning of life. 2) For the device can be easily obtained in the future, i.e., the fusion power is 150MW and the power gain can reach 3, the spent fuel of PWRs can be used in the hybrids with only simple reprocessing like removing the fission product gas. The burnable poison and large uranium-water ratio are both effective to obtain the stable energy multiplication. 3) For the ITER level device, i.e., the fusion power can reach 500MW and power gain is no less than 5, the nature uranium can be directly used as the fuel and about 10 times energy multiplication can be achieved.

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FAST NEUTRON REACTORS - PRINCIPAL FEATURES AND EXPERIENCE

Andrei Rineiski
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Currently, nuclear energy is produced mainly in reactors with essentially thermal neutron spectrum. They contain a significant amount of “moderator” materials (such as light water, heavy water, graphite) that slow down most of fission neutrons to energies comparable to the energy of thermal motion of atoms before the neutrons interact with the fuel. The advantages of thermal reactors (compared systems with a fast neutron spectrum, without moderator) are: lower fuel enrichment and lower (for most design options considered in the past) construction cost. The operating thermal reactor fleet is a net consumer of fissile nuclides, such as U-235 and, therefore, is not sustainable.

Fast reactor (FR) designs have been developed in major nuclear countries for a several decades to combine energy production with breeding of fissile nuclides. This strategy assumes recycling of nuclear fuel. Breeding is still the principal mission of FR projects in countries relying on nuclear energy in the long term. More recently, fast spectrum systems and associated fuel cycle options attracted attention due to their capabilities to burn transuranic nuclides (TRUs) and, to less extent, fission products contained in nuclear waste. Fast spectrum burners, both critical and subcritical are considered in many countries for management of spent reactor fuel. Due to their ability to breed fissile materials and/or deal with spent fuel, the majority of Gen-IV systems are those with a fast neutron spectrum.

FR designs are quite different from thermal ones. A conventional FR is a liquid-metal cooled system, with a core containing highly enriched by TRUs fuel inside a low pressure vessel. Alternatively, gas-cooled and super-critical-water-cooled systems are considered. Molten salt reactors investigated currently in Europe are also fast spectrum systems, without graphite moderator. The lecture includes a comparison of fast and thermal systems as concerns reactor physics, fuel cycle, design, economics and safety.

High TRU content and employment of non-conventional coolants pose challenges while developing safe and economically attractive fast reactor designs. Choice of particular fuel, structure and coolant types for a fast system is associated with advantages and drawbacks as concerns reactor characteristics in a considered fuel cycle and its economics. It is also associated with specific design features, such as presence of an intermediate coolant loop and influences the safety, because different initiators, events, and phenomena may be relevant to a different extent for different FR options. Essential features of systems with different fuels and coolants, proposed design solutions and safety approaches are highlighted in the lecture.

Research on Fusion Neutron Sources.

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The world magnetic confinement fusion programme is focussed on ITER construction and exploitation, in pursuit of the ultimate goal of commercially viable fusion power for clean electricity generation. Materials studies on IFMIF and construction of a DEMO reactor are essential steps towards this long term goal. To accelerate the programme and reduce risks it will be necessary to continue research on present facilities (to progress fusion science and technology) and to add more facilities, e.g. steady-state Fusion Neutron Sources (FNS) as component test facilities (CTF) for material studies complementing IFMIF and low-power neutron sources.

Several options of a Fusion Neutron Source have been studied during the last decades. They include the UKAEA Volumetric Neutron Source (VNS) [1], CCFE Component Test Facility (CTF) [2,3], Oak Ridge CTF [5], University of Texas CFNS [5], TSUK/Kurchatov Institute Super Compact Fusion Neutron Source (SCFNS) [6,7] and others [6]. Their main features and feasibility are analyzed and will be discussed in this talk. Auxiliary heating, tritium consumption and magnetic systems largely determine the capital and running cost of a FNS, and it is shown that a small or medium size Spherical Tokamak (ST) provides significant advantages. The main physics and technology challenges, e.g. stability and confinement properties at high fast ion fraction and low collisionality, optimisation of an operating regime for the beam-plasma dominant neutron production, wall and divertor load at steady-state conditions, magnet design and power consumption, divertor design options and other issues will be discussed in detail.

An important advantage of the ST path to Fusion Power is the possibility to progress from a prototype to a bigger device just by increasing the linear dimensions with no significant changes in design and technologies [8]. This suggests that the demonstration of a reliable application of Fusion as a steady state Neutron Source in a compact ST even at the level of a few MW neutron output will significantly advance not only the mainstream Fusion for Energy research, but also the commercial exploitation of fusion power in nuclear industries and in a wide range of other neutron applications, which will be overviewed.

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6. M. P. GRYAZNEVICH et al, “Options for a steady-state compact fusion neutron source”, to be published in *Transactions of Fusion Science and Technology*, 2011
7. B. V. KUTEEV et al., *Nuclear Fusion,* **51,** 073013 (2011)
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Tuesday, 13 September

Chairman: Ryutov

09.00	M Salvatores (T4)
09:50	W Stacey (T5)
10.40	COFFEE
11:00	M Kotschenreuther (L3)
11.40	J Jiang (O3
12.05	M Reed (O4)
12.30	I Ďuran (O5)
13.00	LUNCH

Chairman: Srinivasan

15.00	A Ivanov (L4)
15.40	D Ryutov (O6)
16.05	F Crisanti (O8)
16,30	F Orsitto (O9)
16.55	COFFEE
17.25	Poster discussion

Fuel cycles and envisioned roles of fast neutron reactors and hybrids

Massimo Salvatores (CEA, Cadarache, France)

In this tutorial lecture, it will be given a definition of the nuclear fuel cycle and a summary of its caracteristics, physics parameters and basic equations.

Specific fuel cycles will be discussed, in particular in the frame of of the waste management strategies :

- Open (« once-through ») fuel cycle
- Reprocessing and partial recycle
- Full fissile material recycle
- Full actinide recycle

Successively it will be introduced the simulation of nuclei evolution under irradiation and decay outside the reactor: the Bateman equations.

Among the fuel cycle parameters, the irradiated fuel decay heat will be singled out for a short discussion.

Finally, a full example of a regional fuel cycle scenario will be presented, based on different types of transmuter reactors : critical fast reactors, accelerator driven systems and fission-fusion hybrids. The major features of the comparison will be briefly discussed.

Summary of FUNFI Tutorial Talk

"Principles of the Fusion-Fission Hybrid

and Its Role in Closing the Nuclear Fuel Cycle"

Weston M. Stacey, Georgia Tech

June 16, 2011

The first part of this talk will describe the basic physics principles of the fusion-fission hybrid and of closing the nuclear fuel cycle (transmuting actinides in spent nuclear fuel and breeding plutonium or uranium-233). The physics basis for certain potential advantages of a fusion-fission hybrid reactor relative to a critical nuclear reactor for these "closing the fuel cycle" missions will be identified.

The second part of the talk will describe an example of the use of a fusion-fission hybrid based on the leading magnetic fusion tokamak physics and technology (as is being demonstrated in ITER) and the use of the leading fast reactor physics and technology (as was developed for the Na-cooled, metal fuel Integral Fast Reactor program). A conceptual design of the SABR fusion-fission hybrid reactor for actinide transmutation will be summarized, and outstanding technical issues will be identified. The fuel cycle analyses of several scenarios for deploying SABRs to support the US LWR fleet will be described, and comparisons of SABR with ADS and Fast Reactor transmutation reactors for this mission will be presented. Preliminary dynamic safety analyses of SABR will be presented.

The third part of the talk will describe how the nearer-term fusion-fission hybrid mission would fit into the long-term magnetic fusion mission of developing electrical power reactors.

NEW APPROACHES TO TOKAMAK BASED HYBRIDS

M. Kotschenreuther, S. Mahajan, P. Valanju, B. Covele

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Fission-fusion hybrids are described with unique advantages for the fission fuel cycle missions of waste transmutation or fuel production. For waste destruction, subcritical fusion-fission hybrids burn the intransigent transuranic residues (with most of the long lived bio-hazard) of a new fuel cycle that uses cheap light water reactors (LWRs) for the easily burned majority of the TRU. In the new fuel cycle, the number of hybrids needed to destroy a given amount of original LWR waste is 5-10 times less than the corresponding number of critical fast reactors. (The low quality residue is extremely challenging for fast reactors to destroy.) The time needed for > 97% waste destruction would be reduced from centuries to decades.

For fuel production, hybrids can produce fuel for 3-4 times (and quite possibly more) as many LWRs and with no fuel reprocessing. Fertile fuel rods exposed to neurons in the hybrid can reach fissile concentrations that enable conventional fuel burning in an LWR, without any fuel reprocessing or fuel re-fabrication, and the attendant proliferation risks and costs.

The LWRs produce several times the thermal power of the hybrid, and many countries possess indigenous thorium deposits that could fuel their entire electricity supply for centuries. Variants of this cycle that include reprocessing can support about 8 times as many LWRs, while reducing throughput of weaponizable material in the reprocessing step by several dozen times. These cycles allow sustainable fuel cycles where the large majority of reactors are commercially successful thermal spectrum systems.

The centerpiece of the fuel cycle is a high power density compact fusion neutron source (CFNS, with major radius + minor radius ~ 2.5-3.5 m), which is made possible by a new magnetic geometry, called the super-X divertor, to solve the formidable power exhaust problem. The physics and technology requirements of the CFNS are far less than the requirements of a pure fusion power source. The CFNS is light enough to be used as a replaceable module inside of a mechanically separate fission assembly. Advantages of the system as part of a timely strategy to combat global warming are described.

Conceptual Design of Fusion-Fission Hybrid Reactor for Spent Fuel Burning (FDS-SFB)

Jieqiong JIANG[1], Minghuang WANG[2], Ming JIN[1], Muyi NI[2], Yan CHEN[1], Jun ZOU[1], Yunqing BAI[1], Qin ZENG[1], Gang SONG[1], Liqin HU[2], Yong SONG[1], Hongli CHEN[2], Qunying HUANG[1], Yican WU[1,2], FDS Team

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Abstract: Fusion has the long-range potential to serve as an abundant and clean source of energy. It is commonly realized that it needs hard work before pure fusion energy could be commercially and economically utilized. The fusion-fission hybrid systems/reactors driven by fusion neutron source can be used to transmute long-lived radioactive waste and to produce fissile nuclear fuel, and to produce energy as a way for early application of fusion technology. The proposed roadmap and testing strategy of fusion application was presented based on the assessment and design analysis of the developed series hybrid system concepts in China as well as a summary of the hybrid system design and R&D activities in China and in the world by FDS Team [1-3]. The final goal is to achieve the demonstrator reactor for spent fuel burning.

In this contribution, a Fusion-Driven hybrid subcritical System (FDS) concept has been designed and proposed as Spent Fuel Burner (SFB) based on viable technologies. The plasma fusion driver can be designed based on relatively easy-achieved plasma parameters extrapolated from the successful operation of the existing fusion experimental devices such as the EAST tokamak in China and other tokamaks in the world, e.g. with a tokamak core of the fusion power of ~150MW, the power gain of ~3, the average neutron wall loading of ~0.5 MW/m^2 .The subcritical blanket can be designed based on the well-developed technology of fission power.

The simulation calculations and performance analyses of plasma physics, neutronics, thermal-hydraulics, thermo-mechanics, safety, and fuel cycle have been present.The results have shown that the proposed concept can meet the requirements of tritium self-sufficiency and sufficient energy gain as well as effective burning of nuclear waste from fission power plants and efficient breeding of nuclear fuel to feed fission power plants. The design and optimization preliminarily demonstrated the engineering feasibility of the design.

Key words: Fusion; Hybrid; Subcritical; Spent fuel;

[1] Y. Wu, J. Jiang, M. Wang, et al., “A Practical Way to Fusion Application through Fusion-Fission Hybrid Systems”. Invited presentation at the 15th International Conference on Emerging Nuclear Energy Systems (ICENES-15), May 15-19, 2011, San Francisco, USA.

[2] Y. Wu, FDS Team, “Conceptual design activities of FDS series fusion power plants in China,” Fusion Eng. Des. 81 (2006) 2713-2718.

[3] Y. Wu, J. Li, Y. Li, et al. An integrated multi-functional neutronics calculation and analysis code system: VisualBUS, Chinese J. Nuclear Science & Engineering, 27 (2007) 365-373.

A Fusion-Fission Hybrid Reactor in Steady-State L-Mode Tokamak Configuration with Natural Uranium

Mark Reed, Ronald R. Parker, Benoit Forget

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This work develops a conceptual design for a fusion-fission hybrid reactor operating in steady-state L-mode tokamak configuration with a subcritical natural or depleted uranium pebble bed blanket. A liquid lithium-lead alloy breeds enough tritium to replenish that consumed by the D-T fusion reaction. The fission blanket augments the fusion power such that the fusion core itself need not have a high power gain, thus allowing for fully non-inductive (steady-state) low confinement mode (L-mode) operation at relatively small physical dimensions. A neutron transport Monte Carlo code models the natural uranium fission blanket. Maximizing the fission power gain while breeding sufficient tritium allows for the selection of an optimal set of blanket parameters, which yields a maximum prudent fission power gain of 7.7. A 0-D tokamak model suffices to analyze approximate tokamak operating conditions. This fission blanket would allow the fusion component of a hybrid reactor with the same dimensions as ITER to operate in steady-state L-mode very comfortably with a fusion power gain of 6.7 and a thermal fusion power of 2.1 GW. Taking this further can determine the approximate minimum scale for a steady-state L-mode tokamak hybrid reactor, which is a major radius of 5.2 m and an aspect ratio of 2.8. This minimum scale device operates barely within the steady-state L-mode realm with a thermal fusion power of 1.7 GW. Basic thermal hydraulic analysis demonstrates that pressurized helium could cool the pebble bed fission blanket with a flow rate below 10 m/s. The Brayton cycle thermal efficiency is 41%. This reactor, dubbed the **S**teady-state **L**-mode non-**E**nriched **U**ranium **T**okamak **H**ybrid (**SLEUTH**), with its very fast neutron spectrum, could be superior to pure fission reactors in terms of breeding fissile fuel and transmuting deleterious fission products. It could operate either as a breeder, producing fuel for pure fission reactors from natural or depleted uranium, or as a deep burner, fissioning heavy metal and transmuting waste with a cycle time of decades. Its primary mission is that of a deep burner producing baseload commercial power with a once-through fuel cycle. The reactor breeds plutonium that could actually be more proliferation-resistant than that bred by fast reactors. Furthermore, it can maintain constant total hybrid power output as burnup proceeds by varying the neutron source strength.

PROSPECTS OF STEADY STATE MAGNETIC DIAGNOSTIC OF FUSION REACTORS BASED ON METALLIC HALL SENSORS

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Hall sensors with their small dimensions, simple principle of operation, and linear dependence of output voltage on measured magnetic field, offer an attractive non-inductive method of magnetic field measurements for ITER tokamak and for future fusion reactors operating in steady state regime, including possible hybrid systems based on magnetic confinement fusion. The use of Hall sensors for steady state magnetic diagnostics is presently limited by several issues related to their radiation and thermal stability. Previous studies on semiconductor type Hall sensors showed that the commercially produced Hall sensors are insufficient as they cannot satisfy even ITER ex-vessel environment requirements [1]. Specially produced semiconductor-based Hall sensors made by MSL Lviv, Ukraine seem to be more suitable for this environment. These InSb based sensors showed good radiation resistance up to the total neutron fluence of at least 10^{18} cm^{-2} [2]. Despite the optimistic forecast for ITER ex-vessel applications, the semiconductor-based radiation hard Hall sensors will probably not be capable to satisfy requirements posed by future DEMO-like reactor. Alternative option to semiconductor-based Hall sensors could be those based on metals. The supposed higher radiation hardness of the metal based Hall sensors is paid for by somewhat lower sensitivity compared to the semiconductor ones. As a result, metals are only very rarely used for producing commercial Hall sensors and, the knowledge of properties of such sensors is rather limited.

Consequently, this proposed contribution will focus on metal Hall sensors design, construction, as well as description of parameters of such sensors, e.g., offset voltage, sensitivity and its dependence on temperature, input and output resistance, and performance of the sensors during temperature cycling. If experimental schedule allows, also comparison of sensors' characteristics before and after irradiation at LVR-15 experimental fission reactor will be presented. The present sensors prototypes are designed as thin ($< 1\mu m$) metallic (copper, titanium, tantalum ...) active layer deposited on AlN or Al_2O_3 Direct Bond Copper substrates with etched 0.127 mm thick copper contact areas. Feasibility of alternative concept of sensors preparation based on thick film technology will be also discussed.

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Fusion neutron research at Novosibirsk including experiments

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In recent years, there has been a renewed interest in magnetic mirror plasma confinement for fusion technology applications, in particular, such as volumetric neutron sources. The reason is that the mirrors offer many potential advantages for the applications: high beta values, topological simplicity, the absence of axial plasma current and disruption-like events, intrinsically steady-state operation, natural He-ash removal out of the ends and low heat loads on plasma-absorbing surfaces in the end-tanks. Axial symmetry adds to the general advantages of mirror devices by illumination of neoclassical transport, by making feasible much higher mirror fields, by enabling relatively simple and flexible magnets, etc.
Among other proposed schemes of the mirror-based neutron sources, the gas dynamic trap (GDT) is, probably, the most far developed, both theoretically and experimentally. It would produce high neutron flux in the localized regions inside a long, axially symmetric, high-β, magnetic solenoid. The plasma confined in the solenoid has two ion components strongly differing in energy. The lower energy collisional component is confined in the solenoid due to a strong increase of the magnetic field at the end mirrors. Another component comprises the energetic D and T ions with anisotropic angular distribution, which are produced by the angled injection of about 100 keV , D/T neutral beams at the center of the solenoid. Then, a quite moderate electron temperature of 0.5 ÷ 1 keV is sufficient to generate a neutron flux as high as 2 MW/m^2 even with rather moderate power consumption. For the given temperature of a warm plasma and the beam energy, the fast ion angular distribution remains rather narrow, and centered on the initial value of the pitch angle during their slowing down to considerably lower energies. The fast ion density is then strongly nonhomogeneous along the solenoid with sharp peaks near the turning points where the ion axial velocity is small. The neutron flux density is also strongly peaked in the same regions that house the testing zones. As a mirror machine, a gas dynamic trap has the advantage of confining high-β plasmas. This results in a higher neutron flux density than would produce other plasma based sources. Possible application of the GDT-based neutron source for fusion materials testing and as a driver for fission-fusion hybrids is discussed in the paper. The paper also reviews recent results obtained from the GDT experimental device in Novosibirsk.

Modulating the Neutron Flux from a Mirror Neutron Source

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A 14-MeV neutron source based on a Gas-Dynamic Trap (e.g., [1] and references therein) will provide a high flux of 14 MeV neutrons for fusion materials and sub-component testing. Its design is well supported by the existing experimental data from the on-going mirror experiment at Novosibirsk [2]. In addition to its main goal, the source has potential applications in condensed matter physics and biophysics. In this report, the author considers adding one more capability to the GDT-based neutron source, the modulation of the neutron flux with a desired frequency and amplitude up to 100%. The modulation may be an enabling tool for high-precision, low-signal experiments favoring the use of the synchronous detection technique, as well as for the assessment of the role of non-steady-state effects in fusion devices. Several approaches to creating a modulated neutron flux in some limited zones of the GDT facility have been assessed, including the local modulation of the magnetic field strength, periodic gas puffs, modulation of the injection energy, and deliberate triggering of the velocity-space instabilities. A conclusion is drawn that modulation frequency of up to 1 kHz is achievable relatively easily. Limitations on the amplitude of modulations at higher frequencies are discussed.

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A Fusion-Fission test experiment proposal, based on the standard H-mode scenario.

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The so called Hybrid (Fusion-Fission) Reactors seem to be a realistic alternative to the drawbacks of both a pure Fusion and/or a pure Fission reactor; this alternative could also guarantee a "substantial" destruction of the transuranic nuclear waste or production of nuclear combustible. So far, several different hypothesis have been proposed for this type of reactor, all of them based on very advanced and hypothetic Fusion scenarios. In the present paper we propose two possible compact (R = 2.5-2.9m – a = 0.73-0.8m) high toroidal field (B_T = 8.5-7.0 T – I_p = 8-6.8 MA) Tokamaks (based or not on superconductors) to perform a robust and "economic" experiment, propaedeutic to the definitive design of the reactor, where the problems connected with the integration of a Tokamak device with a Fission device can be tested on a realistic base. On both solutions the total fusion power will be of the order of 100 MW and the14MeV neutron density power of the order of 0.5-1MW/m^2

Diagnostic systems for hybrid reactors

FP Orsitto
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The hybrid reactor(HR) can be considered an attractive actinide-burner or a fusion assisted transmutation for destruction of transuranic nuclear waste [1,2]. The HR has two subsystems :i) a break-even class (Q~1-2) tokamak, as 14 MeV neutron source and ii) the transuranic burner. Two examples of hybrid load assembly are taken as reference in the present study having a JET- like device [2] and a spherical torus [1] for core devices. Plasma parameters are similar: electron density $ne=10^{20}m^{-3}$, electron temperature Te=10keV, neutron flux in the range of 10^{19}-10^{20} n/(m^2 s). The beta normalized is similar in both core devices β_N~2.5-2.7 . Diagnostic equipment of HR must complement the tokamak measurements plus blanket diagnostics including Tritium regeneration and fission characterization.The diagnostics for tokamak are usually divided into the following categories:i)machine protection;ii) basic plasma control;iii) advanced control;iv)physics evaluation [3] . The tokamak of a HR must have a simple equipment for basic plasma control , machine protection and some fundamental plasma measurement. The minimum set of diagnostic systems is matter of a careful choice. In the present study the systems given in tab I are discussed, together with some basic plasma measurements like electron temperature and alpha losses, ITER-like requirement on measurements are assumed [3], for the core neutron source. The blanket diagnostics aim to measure the tritium produced and recirculated , and the transmutation (i.e. fission by fast neutrons of transuranic elements with long lifetime) of fuel. The blanket will have Pu242,Cm244, Am 243 as primary long-lived transuranic elements to be transmuted [4,5]. The gamma ray spectroscopy is used to detect Am243. The main decay channel of Am243 , Pu242 and Cm244 is in alpha particles with energies between 4.9MeV(Pu242), 5.3MeV(Am243) and 5.8 MeV(Cm244). The technology for alpha spectroscopy measurement is available. The study is dedicated to a detailed analysis of the choice of the minimum set of diagnostics and requirements on measurements for the core device and blanket.

Tab I Tokamak Diagnostics	
machine protection	basic plasma control
neutron monitors	line average electron density
neutron activation measurements	fusion power
Dα emission monitors	Zeff measurements
Divertor Langmuir probes	impurity and D,T influx
Infrared TV camera(divertor)	resistive wall mode
runawys electrons	Halo current
nT/nD in plasma core	divertor erosion monitors
Gas pressure (divertor and ducts)	
Dust	
surface temperature(divertor and first wall)	
plasma shape and position	
plasma current	

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Wednesday, 14 September

Chairman: Mirnov

08.45	E Gonzalez (T6)
	R Moir (L5)
10.15	COFFEE
10.45	O Ågren (L6)
	K Noack(O10)
	V Moiseenko (O11)
	H Anglart (O12)
12.45	Lunch

Chairman: Gorini

14.45	A Stanculescu (T7)
15.35	R Srinivasan (O13)
16.00	S Taczanowski (O14)
16.25	J Källne (O15)
16.50	COFFEE
17.30	Assemble outside Hotel Royal Victoria
	Boat trip and banquet dinner in Bellagio

Principles of ADS hybrids and the EU research program

Enrique González (CIEMAT)

Abstract

Climate change and worries about fossil fuel utilization for electricity generation on one side and the quest for a definitive solution for the nuclear waste with better public acceptance, promoted the research on a more long-term sustainable nuclear energy. Partition and Transmutation was identified as a technological route to enhance this sustainability by multiplying by 30 natural resources utilization and reducing the final long-lived wastes by factor larger than 100. To enhance its efficiency, transmutation devices will prefer nuclear fuels with high fraction of plutonium and minor actinides and low uranium contain. The low intrinsic safety characteristics of this type of fuels and the need for flexibility in the advanced fuel cycles with transmutation motivated the research on subcritical nuclear systems, where the subcriticality and the external neutron source provided the flexibility and allowed to compensate for the missing safety margin of the transmutation fuels. The ADS concept, where the external neutron source was provided by and spallation source activated by a high intensity proton accelerator with proton beams of ~1 GeV, soon become the preferred technology in the EURATOM framework program because its technological availability.

The ADS concept rises a large number of scientific and technological issues to optimize its design and asses its safety and performance. For example, using large content of minor actinides in the fuel will make more important the precise knowledge of their reaction cross sections. Another example is that, being subcritical, the ADS will operate in source following mode with much less effects from the shelf regulation via the delayed neutrons from fission. This will result, on one hand, on the need to continuously monitor the level of reactivity, with new techniques derived from the specific reactor physics of subcritical systems. On the other hand, it will result on a different dynamics, as compared to critical reactors, with corresponding new safety issues.

During more than one decade a large research program has been promoted by EURATOM to develop the concept and required technology, to validate experimentally its main features and physical principles and to evaluate the feasibility and potential impact of ADS, within Partition and Transmutation technologies, in the nuclear fuel cycle.

This tutorial will provide a brief overview of: the motivation for ADS hybrids and possible roles in nuclear fuel cycle; the concept and main components of ADS for transmutation; the main technological challenges in fuel, materials and accelerators; the need for additional nuclear data; the ADS kinetics and associated reactivity monitoring techniques, the specific dynamic and safety issues, and the final performance from the fuel cycle point of view, always with references to the main related EURATOM research projects and the corresponding experimental validation.

Mirror-based hybrids of recent design

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Early application of the simple axisymmetric mirror, requiring intermediate performance between a neutron source for materials testing $Q=P_{fusion}/P_{input}$ ~0.05 and pure fusion Q>10, are the hybrid applications. The Axisymmetric Mirror has attractive features as a driver for a fusion-fission hybrid system: geometrical simplicity, as well as the typical mirror features of inherently steady-state operation, and natural divertors in the form of end tanks. This level of physics performance has the virtue of being low risk with only modest R&D needed; and its simplicity promises economy advantages. Operation at Q~0.7 allows for relatively low electron temperatures, in the range of 3 keV, for the DT injection energy ~ 80 keV from existing positive ion neutral beams designed for steady state. A simple mirror with the plasma diameter of 1 m and mirror-to-mirror length of 40 m is discussed. Simple circular steady state superconducting coils are based on 15 T technology development of the ITER central solenoid. Three groups of physics issues are presented: axial heat loss, MHD stability, and microstability of sloshing ions.

Burning fission reactor wastes by fissioning transuranics in the hybrid will multiply fusion's neutron energy by a factor of ~10 or more and diminish the Q needed to overcome the cost of recirculating power for good economics to less than 2 and for minor actinides with multiplication over 50 to Q~0.2. Hybrids that obtain revenues from sale of both electricity and production of fissile fuel with fissioning blankets might need Q<2 while suppressing fissioning might be the most economical application of fusion but will require Q>4.

Producing ^{233}U from thorium has both proliferation advantages and concerns. ^{232}U that inevitably accompanies ^{233}U production makes the material undesirable but not impossible for use in fission weapons. Fusion's 14 MeV neutron being well above the threshold for making ^{232}U can enhance the $^{232}U/^{233}U$ ratio from its usual value of ~0.1% to >>1%. This enhances the generation of both 2.6 MeV gamma rays and decay heat that facilitates detection of stolen material and makes for weapon design problems.

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The hybrid reactor project based on the straight field line mirror concept

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The straight field line mirror (SFLM) concept is aiming towards a steady-state compact fusion neutron source [1-6]. Besides the possibility for steady state operation for a year or more, the geometry is chosen to avoid high loads on materials and plasma facing components.

A comparatively small fusion hybrid device with "semi-poor" plasma confinement (with a low fusion Q factor) may be developed for industrial transmutation and energy production from spent nuclear fuel [1,5]. This opportunity arises from a large fission to fusion energy multiplication ratio, $Q_r = P_{fis}/P_{fus} >> 1$. The upper bound on Q_r is primarily determined by geometry and reactor safety. For the SFLM, the upper bound is $Q_r \approx 150$, corresponding to a neutron multiplicity of $k_{eff} = 0.97$ [1,5]. Power production in a mirror hybrid is predicted if the electron temperature reaches 500 eV, substantially lower than the requirement $T_e \approx 10$ keV for a fusion reactor. Power production in the SFLM seems possible with $Q \geq 0.15$, which is about ten times lower than typically anticipated for hybrids (and 100 times smaller than required for a fusion reactor) [5]. This relaxes plasma confinement demands, and broadens the range for use of plasmas with supra-thermal ions in hybrid reactors.

Some beneficial features with the axisymmetric Gas Dynamic Trap (GDT) are the simple circular coils, omnigenuity and the high mirror ratio. Confinement of high beta plasmas is demonstrated in GDT, but MHD stability requires a plasma flow into the expanders. This flow has a negative effect on the possibilities to achieve a sufficiently high electron temperature. In the SFLM concept, where additional coils provide a quadrupolar field, a plasma flow into the expanders is not required for MHD stability. The SFLM expanders distribute axial plasma loss over a larger area, and are also aimed to achieve a higher electron temperature.

A brief presentation will be given on basic theory for the SFLM [2] with plasma stability and electron temperature issues [1], RF heating computations with sloshing ion formation [3,4], neutron transport computations with reactor safety margins and material load estimates [5], magnetic coil designs [6] as well as a discussion on the implications of the geometry for possible diagnostics. Reactor safety issues are addressed and a vertical orientation of the device could assist natural coolant circulation. Specific attention is put to a device with a 25 m long confinement region and 40 cm plasma radius in the mid-plane [1]. In an optimal case ($k_{eff} = 0.97$) with a fusion power of only 10 MW, such a device may be capable of producing a power of 1.5 GW_{th}.

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Safety and power multiplication aspects of mirror fusion-fission hybrids

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Topic: plasma Q, fission k and energy gain of fusion-fission hybrids

Abstract

In last decade the Georgia Institute of Technology has developed several design concepts of commercial tokamak based fusion–fission hybrids for burning the transuranic elements of spent nuclear fuel from Light-Water-Reactors [1-2]. The general design objectives were the production of 3 GW_{th} fission power, a maximal value of k_{eff}=0.95 and at least self-sufficiency in power generation and tritium production.

In a research project at Uppsala University a simplified neutronic model for a straight field line mirror hybrid which fulfils the same constraints has been devised and its most important operation parameters have been calculated [3]. The model can also be considered as representative for hybrids driven by axi-symmetric mirrors. In order to reduce the demand on the fusion power of the mirror, a modified option of the hybrid has been examined that generates a reduced fission power of 1.5 GW_{th} with an increased maximal value k_{eff}=0.97.

In the first part of the presentation, the main features of both mirror hybrid options are briefly illustrated and their key parameters are summarized. The cores are cooled by Pb/Bi-eutectic (LBE) and sustain fast neutron fields. Fast systems are known to show positive reactivity effects in the cases of coolant loss and coolant boiling. Therefore, the reactivity effects of fully and partly voiding the LBE-coolant loop were calculated. In addition, the case of filling the coolant loop with water instead of LBE was calculated. The results are given and their consequences on the reactor safety of the hybrids are discussed.

In the second part, different effects which can be initiated by the driver and influence the power multiplication of a mirror hybrid are considered. Possible negative impacts on its safe operation are discussed and conclusions regarding the operation of the mirror are concluded.

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Plasma heating and hot ion sustaining in mirror based hybrids

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In the report the possibilities of plasma heating and sloshing ion sustaining in mirror based hybrids are briefly reviewed. The sloshing ions, i.e. the energetic ions with a velocity distribution concentrated to certain pitch-angle, play an important role in plasma confinement and generation of fusion neutrons in mirrors. First the neutral beam injection is discussed as a method for sloshing ions generation. The experimental results and the results of numerical modelling are reviewed. The sloshing ions could be also sustained by RF heating. The fast wave heating schemes, i.e. magnetic beach, minority and second harmonic heating, are addressed and their similar and different properties are described. The characteristic features of wave propagation in mirrors and overview of possible scenarios of RF heating in hybrid devices including both fundamental harmonic minority heating and second harmonic heating are examined.

The minority heating scenario is effective for mirror plasmas. In small mirrors single global resonance excitation can be used, but the global resonance overlapping scenario has an evident advantage. For large mirrors the minority heating is efficient in a wide range of minority concentrations and plasma densities; it allows one to place antenna aside from the hot ion location; a simple-design strap antenna suitable for it has a good performance. However, this scenario is appropriate only for light minority ions.

The second harmonic heating can be applied for heavy minority. It could be realized in the regime of overlapping of the global resonances. The arrangements for it are similar to the minority heating. The conversion of the fast magnetosonic wave to the ion Bernstein wave may distort the heating pattern. However, the calculations show that this may be avoided. The efficiency of second harmonic heating is determined by weaker wave damping than for minority heating.

The numerical calculations show that in the hybrid reactor scale mirror the deuterium sloshing ions could be heated in the minority heating scheme, while the tritium ions could be sustained by the second harmonic heating.

Numerical models for calculation of antenna excited wave field distribution in mirror traps and problem of spurious solutions are addressed. The attention also is paid to the antennas for fast wave excitation in mirror devices and to the problem of suppressing of slow wave excitation.

The electron cyclotron heating cannot sustain the sloshing ions. However, it could be used for initial plasma production and for different methods of plasma control.

Workshop on Fusion for Neutrons and Subcritical Nuclear Fission – FUNFI 2011, Villa Monastero, Varenna, Italy, 12-15 September 2011.

Principles of passive and active cooling of mirror-based hybrid systems employing liquid metals

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ABSTRACT

Nuclear reactors require reliable and efficient cooling systems that remove the heat generated by fission both during the normal operation and after the reactor shut-down. The cooling systems need to be reliable to provide heat removal in all possible circumstances. The reliability is usually achieved in two different ways: either by providing a functional redundancy of active systems or by employing inherently safe passive processes. The cooling systems need to be efficient to assure that temperatures of fuel and construction materials do not exceed the allowable limits using as low as possible effort. The efficiency is usually achieved by proper design of the cooling systems.

The purpose of this paper is to present the principles of design of passive and active cooling systems employing liquid metals that are suitable to mirror-based hybrid systems. Active cooling systems are in general more efficient than the passive ones. Due to that such systems are employed in the majority of currently operating nuclear reactors. This is particularly true for all commercial LWRs operating worldwide, which employ active cooling systems both during the normal operation and after the reactor shut-down when removing the decay heat.

When applying active cooling principles to mirror-based hybrid systems one has to take into account that the required pumping power will dramatically increase when liquid metals, and in particular lead alloys, such as the lead-bismuth eutectic, are employed. Due to that an efficient and reliable pumping system is required. In this paper a review of existing pumping systems with liquid metals, as well as their comparative analysis, will be provided.

Recently there is a growing interest in employing passive cooling principles in nuclear systems. Such systems have been investigated since several decades; however, only two reactors (Doderwaard in the Netherlands and VK-50 in Russia; both already removed from operation) were using boiling water passive cooling systems during normal operation. In mirror-based hybrid system employing passive cooling principles, a natural circulation of liquid metal has to be considered. This paper will present a review of current experience with natural circulation of liquid metals in nuclear reactors. In particular, differences between natural circulation of water and liquid metals will be discussed.

OVERVIEW OF NUCLEAR ENERGY PRESENT AND PROJECTED COMMERCIAL USE

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Current renewed interest in nuclear energy is driven by the need to develop carbon free energy sources, demographics and development in emerging economies, as well as security of supply concerns. The pace at which the nuclear energy option is embraced seems to be accelerating worldwide[1], with the existing marked imbalances[2] in energy availability causing more and more emerging economies to give it serious consideration.

Worldwide, there are 440 reactor units in operation, with a total installed capacity of 374 GW_{el}[3]. There are also 64 reactor units under construction, totaling 63 GWel[4]. Nuclear Power will probably grow, but its share of global electricity may fall. Some countries have phase-out policies, others see advantages for energy security, and some plan to increase nuclear capacity. The IAEA predicts growth of nuclear power generation from 2558 $TW_{el}h$ in 2009 to (low-high estimates) 3314-4006 $TW_{el} \times h$ by 2020, 4040-5938 $TW_{el} \times h$ by 2030, and 4342-10436 $TW_{el} \times h$ by 2050[5].

Several factors will influence the contribution of nuclear energy to the future energy mix. Among them, the most important are the degree of global commitment to greenhouse gas reduction, continued vigilance in safety and safeguards, technological advances, economic competitiveness and innovative financing arrangements for new nuclear power plant builts, the implementation of nuclear waste disposal, and, last but not least, public perception, information and education.

The paper will present an overview of the current nuclear energy situation, possible development scenarios, an overview of next generation reactors and fuel cycles, and of some non-electric applications of nuclear energy.

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1 An indication for the so-called "nuclear renaissance" can be found in the new construction start numbers, *viz*. 10, 11, and 15 in 2008, 2009, and 2010, respectively. However, these numbers are far from the 25 average construction starts per year in the 1970s, and are very localized (China, India and Russia).

2 The OECD average yearly per capita electricity consumption is ~8000 kWh, about two orders of magnitude higher than in many African countries, ~13 and ~3 times higher than in India and China, respectively (these three areas add up to almost 50% of the world's population). There is a strong correlation between energy consumption and poverty, with more than ¼ of the world's population without access to electricity and ~40% relying on traditional biomass as primary energy source.

3 IAEA, Power Reactor Information System (PRIS), www.iaea.or.at/programmes/a2/

4 Ibid

5 IAEA RDS-1/30 REFERENCE DATA SERIES NO. 1, 2010 Edition

Indian fusion test reactor

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Abstract

The fusion reactor as a volumetric neutron source can serve many applications needed for realizing fusion power reactor. For the Indian energy scenario, such a device can also produce fissile fuel for accelerating the nuclear power production. The Indian Fusion Test Reactor (FTR) is a low fusion gain (Q = 3-5) device to be used as component test facility for qualifying future reactor materials as well as for demonstrating the production of fissile fuel. FTR will be a medium sized tokamak device with a neutron wall load of 0.2 MW/m^2. The presently available structural materials can be used for this device and such a device can be realized in ten years time from now. This device should produce about 25-50 kg of fissile fuel in one full-power-year and also produce the tritium needed for its operation. In this paper, a detailed discussion about physics parameters of this device and the required technologies for realizing FTR will be presented.

ACTINIDE INCINERATION IN FUSION-FISSION HYBRID – A MODEL NUCLEAR SYNERGY

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The alliance of fusion with fission is a cause worthy of great efforts, as being able to ease (if not even to solve) the serious problems that face both these forms of nuclear energy.
1) **Fission**. The most troublesome component of the High Level Waste (HLW) i.e. Minor Actinides (MA) shows disadvantageous physical properties: intense radioactivity, minute fraction of delayed neutrons, intense heat release, positive reactivity coefficients etc. Such features affect safety of critical systems so much that exclude incineration of the MAs in them. As a remedy – the MAs abatement in safer subcritical systems has been proposed, namely, in Fusion-Driven Actinide Incinerators (FDI) [1]. Besides the 14MeV neutron component may better incinerate all hardly fissionable actinides. Though the fusion technology is not yet ready, the problem of actinide waste seems last until the fusion hybrid systems are available.
2) **Fusion**. Very high investment costs caused by tokamak sizes and difficult technology put in doubt whether alone the minute demand for fuel (Li) and lack of danger of uncontrolled super-criticality prove sufficient for making it economically competitive. For illustration, the material consumption of tokamaks in some design (PPCS-AB) is exceeding 120 000 tons, while the mass of Li-Pb eutectic (tritium breeding material) amounts to 35 000 tons [2]. Tritium, due to its high solubility in most metals thus resulting in inventory attaining 10 kg [3] is a problem too.
All the above problems may be solved with synergic union of fission with fusion.
The performed preliminary evaluations demonstrated that a radical shift of energy production i.e. the energy gain from plasma to fissionable blanket is feasible. A reduction of the fusion component to about 2% at given power of the system, brings a radical drop in the value of plasma Q down to the level ~0.3 achievable in small systems (e.g. mirror devices – Gas Dynamic Traps – GDT). The sizes of the latter are rather comparable to those of fission reactors whose life cycle mass of contaminated steel amounts to a few thousand tons.
In an FDI all the radiations from the plasma: corpuscular (i.e. neutrons, alphas and other ions) and electromagnetic ones are drastically reduced. Thus, proportionally – the radiation damage: its plasma-wall component and the neutron induced ones: gas production, DPA and to a degree – transmutations. Finally, last but not least, the fundamental safety of the system has been proved by simulation of its collapse that has shown preserving its subcriticality in this extreme state.
In spite of all the above encouraging remarks a number of questions still await solution. E.g. a trade-off dilemma: system size vs. reduction of radiation damage, while the latter is quite unevenly distributed, is unsolved. From the point of view of fission the efficiency of MAs incineration requires further investigations, as well as the safety properties of the system.
Summarizing, the concept of Fusion-Driven Actinide Incinerator – small, simple and cheaper deserves consideration also as an intermediate step towards the Pure Fusion that should bring near the development of Fusion Energy.

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NEUTRON DIAGNOSTICS FOR MIRROR HYBRIDS

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Fusion-fission (FuFi) hybrids will need instrumentation to diagnose the deuterium-tritium plasmas of which neutron measurements are essential also to monitor the fission rate. A central task for the neutron measurements is to determine the fusion and fission neutron yield rates Y_{FU} and Y_{FI}. For instance, yield rate information is the basis to set the desired operating conditions of the hybrid such as neutron multiplication factor (k) of the fission blanket, i.e., its sub-criticality value.

The FuFi hybrid will represent a hostile environment to perform diagnostic measurements. Such conditions have individually been encountered in present (magnetic) fusion experiment operating with DT plasmas (for instance, JET and plans for ITER) and fast neutron fission reactors. This experience can be used in exploration of diagnostics for hybrids but the interface issues add complexity. This is true, especially, for the measurements of fusion neutrons which is a minority fraction of the total neutron yield; the fusion component is distinguishable, mainly, by its energy spectrum characteristics. Moreover, the access for measurement is severely restricted by the layers of blankets surrounding the fusion neutron source.

In this contribution we will discuss the interface restriction of fusion plasma diagnostics in the FuFi hybrid based on the straight field line mirror (SFLM). The need for some basic neutron measurements will be discussed along with the possibilities for performing them.

Thursday, 15 September

Chairman: Källne

09.00	R Moir (O16)
09.25	R Goldston (O17)
09.50	S Mirnov (O18)
10.30	COFFEE
11:00	H Blix (SL1)
11.50	R Schenkel (SL2)
12.40	Concluding discussion
13.10	CLOSE of FUNFI 2011

Fission-suppressed fusion, breeder on the thorium-cycle, and nonproliferation

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Abstract

Fusion reactors could be designed to breed fissile material while suppressing fissioning thereby enhancing safety. The produced fuel could be used to startup and makeup fuel for fission reactors. Each fusion reaction can produce typically 0.6 fissile atoms and release about 1.6 times the 14 MeV neutron's energy in the blanket in the fission-suppressed design. This production rate is 2660 kg/1000 MW of fusion power for a year. The revenues would be doubled from such a plant by selling fuel at a price of 60/g and electricity at \$0.05/kWh for $Q=P_{fusion}/P_{input}=4$. Fusion reactors could be designed to destroy fission wastes by transmutation and fissioning but this is not a natural use of fusion whereas it is a designed use of fission reactors. Fusion could supply makeup fuel to fission reactors that were dedicated to fissioning wastes with some of their neutrons. The design for safety and heat removal and other items is already accomplished with fission reactors. Whereas fusion reactors have geometry that compromises safety with a complex and thin wall separating the fusion zone from the blanket zone where wastes could be destroyed.

Nonproliferation can be enhanced by mixing ^{233}U with ^{238}U. Also nonproliferation is enhanced in typical fission-suppressed designs by generating up to 0.05 ^{232}U atoms for each ^{233}U atom produced from thorium, about twice the IAEA standards of "reduced protection" or "self protection" set at a dose rate of 100 rem/h (1 Sv/h) 1 m from 5 kg of ^{233}U with 2.4% ^{232}U one year after chemical separation of daughter products. With 2.4% ^{232}U, high explosive material is predicted to degrade owing to ionizing radiation after a little over ½ year and the heat rate is 77 W just after separation and climbs to over 600 W ten years later.

The fissile material can be used to fuel most any fission reactor but is especially appropriate for molten salt reactors (MSR) also called liquid fluoride thorium reactors (LFTR) because of the molten fuel does not need hands on fabrication and handling.

Key words: Hybrids, thorium, U-233, fusion fuel breeder, nonproliferation

Climate Change, Nuclear Power, and Nuclear Proliferation: Magnitude Matters

Robert J. Goldston, Princeton University

Integrated energy, environment and economics modeling suggests that worldwide electrical energy use will increase to ~12 TWe in 2100. Due to limitations of other low-carbon energy sources, nuclear power may be required to provide ~30% of world electrical energy by 2100. Calculations of the associated stocks and flows of uranium, plutonium and minor actinides indicate that the proliferation risks at mid-century, using current light-water reactor (LWR) technology, are daunting. There are institutional arrangements that may be able to provide an acceptable level of risk mitigation, but they will be difficult to implement.

A transition may be made to fast-spectrum fission reactors, due to constraints on uranium supply or on waste storage. If fast reactors burning transuranics (TRU) with conversion ratio 1.21 are commercialized by ~2040, they could replace light-water reactors completely by 2100. However the global proliferation risks in this scenario are much greater and more resistant to mitigation; for example the yearly amount of plutonium being loaded into the fast reactors in 2100 would be enough for one million nuclear weapons. The standard for Material Unaccounted For at reprocessing plants is currently 1%.

In an alternative scenario, pure-fusion power plants are commercialized in ~2050 and begin to replace light-water reactors. If electric power production from fusion grows at less than 0.9%/year of the world's total electrical power production (fission's rate was 1.2% from 1975 – 1990) all fission systems could be replaced with fusion systems by 2100. In this case the proliferation risks are much smaller. It is not credible for a clandestine fusion system to be used to produce significant quantities of fissile material. Covert production of fissile material in a fusion power plant under safeguards is also not credible. In a scenario of breakout from safeguards, a fusion power plant has no fissile material at the time when inspectors are expelled. It can be rendered incapable of producing such material through a missile strike, with no risk of release of radioactive material.

In the case where pure-fusion plants follow LWRs, the TRU in the legacy waste from the LWRs remains as a long-term proliferation risk. If 10% of the fusion plants are instead fusion-fission hybrids, burning TRU, then it is possible to load all of the legacy TRU into the hybrids and their fuel cycle by 2100. The quantity of TRU in the system decays by a factor of two approximately every 31 years. Whether this approach, which puts Pu into active process, is more or less of a proliferation risk than simply guarding the buried waste will require judgment.

What we should do for transition from current tokamaks to fusion-fission reactor (From fusion romance to reality)

S.Mirnov

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Annotation

The Russian fission community places several heavy demands on quality of fusion neutron source for the first step of investigation of minority burning and breading of nuclear fuel. They are: the steady state regime of neutron production (not rare 80% of main operation time), the total power on neutron flux should be not lower than 10MW with surface neutron load not lower than $0.2MW/m^2$. Between the current fusion devices: mirror traps, reverse field pinches, stellarators, spherical torus and tokamaks only lasts have today the some probability to fulfill in the near future these hard demands. Two well known DT-tokamaks with neutron power production higher 10MW - TFTR and JET- had maximal neutron load approximately $0.1MW/m^2$ only and only in transient (with time scale lower 1s) regimes. The quasi steady state neutron emission (~5MW, 5sec) was performed in JET with mean surface neutron load lower than $0.025MW/m^2$ only. In this communication it will be discussed the main needs of JET scale tokamak improvement for increase on neutron load up to $0.2MW/m^2$. They are: decrease of Zeff by ECRH and lithium use as plasma facing components, the increase of energy of steady state neutral injectors up to 150-170keV (tritium), acceleration of ion beam by use ICRH on fundamental frequency, the He removal and creation of closed loop circulation of DT fuel.

GEOPOLITICAL AND STRATEGIC ASPECTS OF PRESENT AND FUTURE NUCLEAR ENERGY

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The present and the potential future role of NE is a vital issue for the world. Most people see the choice of energy sources as a practical matter where you compare factors such as cost, safety, waste and allow yourself to be guided by the result. However, everywhere there are also people who are terrified by the thought of **nuclear radiation** and who will oppose nuclear power with tooth and nail. A populist response is to exploit the anguish for vote getting. A serious response is to say that if the anguish is justified we must abstain from nuclear power. If not, we must explain why not and make use of nuclear power. Factors of relevance in serious assessments and comparisons are:

- **No use of energy is at zero risk.** However, objective examinations have shown the environmental and health risk of nuclear power to be at about the same level as renewables and much lower than the risks of fossil fuels. While big nuclear accidents – as at Fukushima – have had terrible consequences they have contributed to a de-demonization of nuclear power: core melts do not go through the earth (as suggested in the China Syndrom). Damages may be extensive and long lasting but not forever.
- **Nuclear power** causes very low emissions of CO2 and has a waste that may be safely isolated until it is harmless. With new types of nuclear reactors and fuel waste may be further reduced in volume and radioactivity. **Fossil fuels** that now dominate may contribute decisively to the existential threat of **global warming.**
- A popular cry is for **renewable sources** of energy—like wind and solar power – and a rejection of fuels that are finite – like coal, oil and gas. However, cost is not irrelevant and if spent nuclear fuel is reprocessed and used in fast reactors and thorium is used, nuclear fuels will last many hundreds of years. Is that not enough?
- Reliance on nuclear power offer states stable base load power and a measure of **energy independence.** Uranium and thorium can be bought in politically stable areas. Fuel can be stored for years of use.
- The risk of use of highly enriched uranium and plutonium for **nuclear weapons** exists regardless of an expanded use of nuclear power but measures should be taken to discourage uneconomic and unnecessary construction of enrichment and reprocessing plants.

FUNFI workshop, Sept 12-15,2011 Varenna

Nuclear Energy Acceptance and potential Role to meet Future Energy Demand. Which technical/scientific achievements are needed?

Roland Schenkel

Abstract

25 years after Chernobyl,the Fukushima disaster has changed the perspectives of nuclear power. The disaster has shed a negative light on the reliability and rigour of the national nuclear regulator and plant operator and the usefulness of the international IAEA guidelines on nuclear safety. It has become clear that, in the light of the most severe earthquake in the history of Japan, the plants at Fukushima Daiichi were not adequately protected against Tsunamis.

Nuclear acceptance has suffered enormously and has changed the perspectives of nuclear energy dramatically in countries that have a very risk-sensitive population, Germany is an example. The paper will give a survey on the reactions in major countries and the expected impact on future deployment of reactors and on R&D activities.

On the positive side, the disaster has demonstrated a remarkable robustness of most of the 14 reactors closest to the epicentre of the Tohoku Seaquake although not designed to an event of level 9.0.

Public acceptance can only be regained with a rigorous and worldwide approach towards inherent reactor safety and design objectives or upgrades of existing facilities that limit the impact of severe accidents to the plant itself (like many of the new GenIII reactors). A widespread release of radioactivity and the evacuation(temporary or permanent) of the population up to 30 km around a facility are simply not acceptable.

Several countries have announced to request more stringent international standards for reactor safety. The IAEA should take this move forward and intensify and strengthen the OSART mission scheme.

The paper will examine which nuclear energy options could lead to safer reactors with lesser burden to the environment under severe accident conditions. Needless to mention that these reactors need to be proliferation resistant and should ideally produce less long term radiotoxicity. This will also cover fusion-fission hybrids which have recently been proposed for energy production.

Scientific/technical achievements that are required in the different areas will be highlighted including the the nuclear waste management and disposal area.

Posters Monday-Wednesday

From Monday 10.30	On display
Tuesday 17.45	Discussion session
Wednesday 17:00	Posters removed

Number and author	Institution	Title
1. Ågren	Uppsala Univeristy	Radial drift waves invaraiant in long mirror and thin mirrors
2. Hagneståł	Uppsala University, Uppsala, Sweden	Coil system for a mirror-based hybrid reactor
3. Anikeev	Novosibirsk State Univ, Novosibirsk, Russia	Optimisation of the neutron source based on gas dynamic trap for transmutation of radioactive wastes
4. Yurov	Budker Inst, Novosibirsk, Russia	Parameters optimization in a fission-fusion system with a mirror machine based neutron source
5. Beklemishev	Budker Inst, Novosibirsk, Russia	GDT-based Neutron Source with multiple-mirror end plugs
6. Ciotti	Associazione ENEA-Euratom, Frascati (Rome), Italy.	Italian hybrid and fission reactors scenario analysis
7. Fomin	Akhiezer Inst Theor Physics, Nati Sci Center, Ukraine	Physical Basis of Advanced Fast Reactor Working in Nuclear Burning Wave Regime
8. Zeng	Inst Plasma Phys, Chinese Acad Sci, Hefei, China	CAD-based 4D Neutronics Simulation Software for Fusion, Fission and Hybrid Systems
9. McNamara	Leabrook Computing, UK	NEST: A Nuclear Energy Security Treaty. Separating Nuclear Energy from Nuclear Weapons.
10. Kotenko	Inst Plasma Physics, Kharkiv, Ukraine	Magnetic field of combined plasma trap
11. Moiseenko	Inst Plasma Physics, Kharkiv, Ukraine	A fuel for sub-critical fast reactor
12. Moiseenko	Inst Plasma Physics, Kharkiv, Ukraine	Fusion neutron generation computations in a stellarator-mirror hybrid with neutral beam injection
13. Giacomelli	Univ Milano-Bicocca, Milano, Italy	Diamond detectors for beam monitors of fast neutron sources
14. Croci	IFP-CNR, Milano, Italy	A new GEM based neutron diagnostic concept for high power deuterium beams
15. Nocente	Univ Milano-Bicocca, Italy	Neutron sensitivity and γ-ray measurements in a fusion environment

RADIAL DRIFT INVARIANT IN LONG-THIN MIRRORS

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In omnigenous systems, the guiding centers are constrained to move on magnetic surfaces. Conditions for specific such systems have identified and studied in several papers [1-3]. Since a magnetic surface is determined by a constant radial Clebsch coordinate, omnigenuity implies that the guiding center radial coordinate (the Clebsch coordinate) is constant. Near omnigenuity is probably a requirement for high quality confinement and in such systems only small oscillatory radial banana guiding center excursions from the average drift surface occur. The guiding center radial coordinate r_{gc} is then the leading order term for a more precise radial drift invariant I_r , i.e. $I_r = r_{gc} + r_{ban}$, where the last "banana" term is oscillatory [4]. An analytical expression for the radial invariant is derived for long-thin quadrupolar mirror equilibria. The formula for the invariant is then used in a Vlasov distribution function. Comparisons are first made with Vlasov equilibria using the adiabatic parallel invariant. To model radial density profiles, it is necessary to use the radial invariant (the parallel invariant is insufficient for this). The results are also compared with standard fluid approaches. In several aspects, the fluid and Vlasov system with the radial invariant give analogous formulas. One difference is that the parallel current associated with finite banana widths could be derived from the radial invariant.

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Coil system for a mirror-based hybrid reactor

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A vacuum magnetic field from a superconducting coil set for a single cell minimum-B mirror-based fusion-fission reactor [1] is computed [2]. The magnetic field is optimized for MHD stability, ellipticity and field smoothness, using expressions derived to first order in a long-thin approximation. The field is modeled and optimized using a local optimization method. It is similar to the Straight Field Line Mirror Field. A recirculation region and wide magnetic expanders on both sides are added to the central mirror confining region [1]. The coil set producing this field consists of circular and quadrupolar coils [2]. The coils are modeled using filamentary line current distributions. Basic scaling assumptions are made for the coil dimensions, based on a maximum allowed current density of 1.5 kA/cm^2 for superconducting coils. Sufficient space is available for a fission mantle in between the coils and the plasma chamber and antennas for the RF heating [3]. After determination of the field by the optimization, the coils are optimized using a local optimization method. The resulting field is finally checked for MHD stability, maximum ellipticity and other criteria. The resulting confinement region is 25 m long with a 40 cm midplane plasma radius.

The coil set is large compared to the confinement region due to the space occupied by the fission mantle. This has economic disadvantages, and some approaches to simplify the coil set and make it cheaper are discussed. The coil set typically contributes to a substantial fraction of the total cost for fusion devices, predicted to be in the order of 30% for ITER [4]. Means to substantially reduce the cost for the coils (by using weaker fields, more compact coils etc) could therefore be important and seem possible.

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OPTIMISATION OF THE NEUTRON SOURCE BASED ON GAS DYNAMIC TRAP FOR TRANSMUTATION OF RADIOACTIVE WASTES

A.V.Anikeev

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For a number of years the Budker Institute of Nuclear Physics in collaboration with the Russian and foreign organizations develop the project of 14 MeV neutron source, which can be used for fusion material studies and for other application [1]. The projected neutron source of plasma type is based on the plasma Gas Dynamic Trap (GDT), which is a special magnetic mirror system for plasma confinement [2]. Presented work continues the subject of development the GDT-based neutron source (GDT-NS) for hybrid fusion-fission reactors. The paper presents the results of recent numerical optimization of such neutron source for transmutation of the long-lives radioactive wastes in spent nuclear fuel. The calculations of the neutron source's parameters have been made by Monte-Carlo methods with using of the last improved version of the Integrated Transport Code System (ITCS). ITCS was developed for GDT and GDT-NS simulation and includes different modules for plasma, particles transport and neutron production modeling. The new physical phenomena (as a vortex confinement, ambipolar plugging, high β etc.) were included into account in these simulations. The experimental and theoretical grounds of these phenomena were made in the GDT-U experimental facility in the Budker Institute.

As a result, a new improved version of the GDT-NS is proposed and numerically simulated. The proposed neutron source has two n-zones of 2 m length with a neutron power of 1.6 MW/m and a neutron production rate up to 1.5×10^{18} n/s each. This source can be used for application to a fusion driven system for the burning of MA in spent nuclear fuel. We use one plasma-fusion GDT driver for two sub-critical burners placed around the neutron emission zones. The considered sub-critical burner configuration is based on the reactor design of the European Facility for Industrial Transmutation (EFIT) [3]. The EFIT reactor was designed for the demonstration of the transmutation of minor actinides in an ADS facility on the industrial scale EFIT has a power of about 400 MWth and is cooled by lead. Its fuel is uranium free CERCER fuel 50% MgO +50% (Pu,MAO_2) in volume, containing a large quantity of americium. The plutonium content is ~ 37% leading to k_{eff} ~ 0.97. In [4] several variations of EFIT design parameters were studied by using ENDF/B-6.5 based 69 group cross sections in the deterministic S_n code TWODANT. The extended version of the EFIT reactor was used for application with the GDT-NS. Results of the calculations for such fusion-fission system for incineration of radioactive wastes will be presented.

This research was supported by a Marie Curie International Incoming Fellowship Return within the 7th European Community Framework Programme.

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Parameters optimization in a fission-fusion system with a mirror machine based neutron source

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Long-lived fission products utilization is a problem of high importance for the modern nuclear reactor technology. It is possible to use sub-critical fission systems with neutron drivers for the resolution of this problem. BINP jointly with IBRAE RAN develops a conceptual design of a hybrid sub-critical system with a gas dynamic mirror machine based neutron source for the incineration and transmutation of minor actinides. Source neutron flux spatial variability, improved multiplicity and continuous regime of the driver operation are the main advantages of the proposed system in comparison with similar accelerator driven systems [1]. A number of numerical experiments was carried out in order to maximize energy efficiency factor and multiplicity of the developed fusion-fission device. The report suggested is dedicated to the results obtained.

Neutron source parameters calculation was performed by means of GENESYS code, a zero-dimensional model of plasma evolution in the Gas Dynamic Trap. Energy and particle balance of the machine is considered in accordance with processes described in [2]. Approximation of test particles is used for calculation of fast ions distribution dynamics. Charge exchange with neutral heating beams, deceleration and angular scattering on the warm plasma components are considered in related Fokker-Planck equation. Linear neutron emission intensity in relation to the position along the trap axis is evaluated by Monte-Carlo method. Figure 1 presents verified neutron emission intensity distribution for one of GDT experiments with H-D mixture [3].

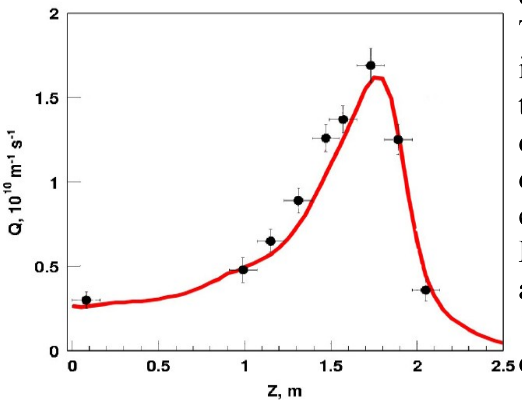

Figure 1. Linear neutron emission intensity for D-D reaction along the machine axis

Sub-critical blanket characteristics were investigated by NMC (Neutron Monte-Carlo Code) – code for simulation of neutron processes in sub-critical systems. Software is developed in IBRAE RAN and presupposes usage of straight modeling for neutron transport calculation along with energy multigroup approximation. ENDF/B -VII neutron data library is utilized as a primary source for the following processing.

Modeling was performed for a GDT-based neutron source with 75 MW neutral beam injectors and EFIT-like homogeneous sub-critical blanket. Dependence of the system multiplicity on Pb-Bi buffer thickness and neutron emission intensity of the neutron source according to the emission zone length were calculated. Single- and double-zone neutron source configurations were considered.

The research was supported by RFBR grant 09-08-13746-ofi_c and P1193, P1580, P969 state contracts with Federal Education Agency.

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GDT-based Neutron Source with multiple-mirror end plugs

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Axially symmetric mirrors, such as the gas-dynamic trap (GDT) in Novosibirsk, have low transverse particle and heat losses. If injected at an angle to the magnetic axis, fast particles form the “sloshing ion” distribution. Its density is peaked at turning points and may be higher than that of the “warm” background plasma, which is generally needed just to provide the microstability. As a result, such traps are ideal for the beam-target or beam-beam fusion, making the GDT-like devices good Neutron Source candidates. However, energy cost of neutrons in pure GDT scheme is rather high due to axial losses and low electron temperature.

Better axial confinement at a high plasma density can be achieved in multi-mirror configurations [1]. Namely, if the scattering length of particles out of the loss-cone is less than the trap length, the axial loss processes become diffusive. If it is roughly equal to the cell length, the loss rate is reduced by the factor of the number of cells. Classical Coulomb scattering is too weak at fusion temperatures and containable pressures, but the collective scattering is just as good. Evidence of improved confinement due to collective scattering of ions in cells of the multiple-mirror trap was observed in GOL-3 experiments. Even if such collective scattering is not present due to inherent instability of the outflow as in GOL-3, suitable oscillations can be induced by external sources, such as a pulsing electron beam.

We present a new project to be built at the Budker Institute. It combines GDT-type central cell with multiple-mirror “tails” and sufficient energy content with low axial losses. It will be 30m long, run in 1s pulses, and have effective $Q_{DT} \sim 0.1$ (Fig. 1). The neutron flux will be generated in a single 5m-long active zone. Primary heating consists of 8 1MW neutral beams, while two 5MW 50kV electron beams, injected axially, provide auxiliary electron heating and collective scattering in multiple-mirror plugs. Vortex confinement based on charge injection by electron beams is planned as the primary mechanism for suppression of flute modes.

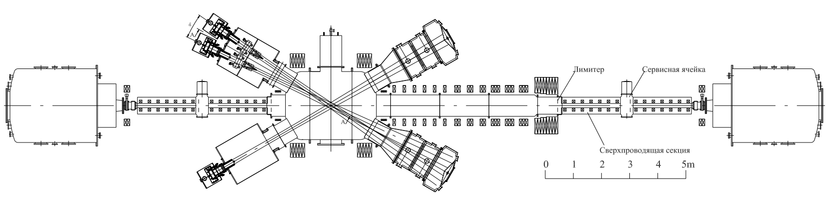

Figure 1: Design of the new-generation trap in Novosibirsk.

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Italian hybrid and fission reactors scenario analysis

M. Ciotti, J. Manzano, M. Sepielli

It is very well believed that electricity production from fossil fuels should be replaced in the next time by environmentally more sustainable sources, a strong expansion of third generation nuclear fission reactors is the best option to meet this goal. This phase will be followed by a fourth generation (4G) reactors that must offer better solutions concerning some aspects of the nuclear technology. A Fusion-Fission hybrid reactor is a valid 4G candidate and represent an interesting development for both technologies.

Fission can gain both from the ability of such a system to reduce spent fuel activity , a good opportunity to smooth public opinion disaffection, and to produce innovative abundant fuel that could assure to this technology the role of becoming the main energy production choice for a long term scenario . On the other hand fusion can try to have in sight a more realistically achievable application in a predictably time period.

In this paper different scenarios are analyzed starting with a basic one where a few hybrid reactors are coupled with a classical fission reactor fleet in order to withstand a national foreseen electrical energy production percentage, up to more fancy ones with the possibility to take into account Th-U cycles and innovative small and modular reactors.

Physical Basis of Advanced Fast Reactor Working in Nuclear Burning Wave Regime

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A great interest for the future power engineering presents the development of new concepts of nuclear fission reactors with the so-called intrinsic safety, in which the development of uncontrolled chain nuclear reaction is impossible due to the physical principles of their operation. One of such concepts is based on the self-sustained nonlinear regime of the nuclear burning wave in a fast reactor (the Feoktistov wave [1]). The critical state in such a reactor is sustained automatically without external control due to a negative reactivity feedback. Moreover, the realization of this concept can help to solve several actual problems of nuclear power at once, such as the close fuel cycle organizing, the long-live radionuclide transmutation and non-proliferation of fissile materials. These unique features of such reactor attract interest of many groups of investigators, among them Edward Teller [2], Hiroshi Sekimoto [3] and others.

We present here the results of further development of this concept with the purpose of creation of physical basis of the safe fast reactor working in the nuclear burning wave regime. Our investigation is based on the deterministic approach, which includes the numerical solution of non-stationary non-linear diffusion equation for neutron transport together with a set of the burn-up equations for fuel components and the equations of nuclear kinetics for precursor nuclei of delayed neutrons [4]. The calculations were carried out for U-Pu, Th-U and for the mixed Th-U-Pu fuel cycles. The obtained results show that the usage of the mixed Th-U fuel gives an opportunity to reach quite acceptable for practical utilization values of neutron flux and power production density in such a reactor. A special attention is paid to the analysis of the nuclear burning wave regime stability towards distortions of the neutron flux and fuel inhomogeneity in the system. The reactor behavior at the transition regimes such as its starting and re-starting after its forced shutdown is also analyzed.

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CAD-based 4D Neutronics Simulation Software for Fusion, Fission and Hybrid Systems

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Abstract: Neutronics methods and programs are important analysis tools and essential to nuclear reactor system design and research. For advanced nuclear devices with large-scale and complex geometries and material distributions, such as fusion and sub-critical fission, it is very difficult to obtain the exact three-dimensional (3D) spatial distribution and time-dependent neutronics parameters, because of complex neutron energy spectrum structure and time-dependent physics properties. Under the study of advanced neutronics simulation methodologies and the modern computer technologies, a CAD-based 4D (3D geometry and 1D time) integrated multi-functional neutronics simulation software named VisualBUS has been developed by the FDS team devoted to research and development of advanced nuclear energy systems in China. It contained four main subsystems and supported nuclear data library HENDL. These subsystems are CAD-based modeling subsystem, coupling calculation subsystem, visualized analysis subsystem and multi-objective optimization subsystem. In addition, some interfaces have been developed to couple other physics simulation programs, e.g. for mechanics and thermal-hydraulics design analysis. The object-oriented concept, network-based architecture, cloud computing and collaboration technology are adopted in the development of VisualBUS.

VisualBUS can perform multiple neutronics analyses of fusion, fission and hybrid systems (fusion-driven sub-critical systems and accelerator-driven sub-critical systems). This system and its subsystems have been tested successfully with the ITER benchmark model and the IAEA-ADS benchmark model. Series of applications have preliminarily demonstrated its efficiency and maturity. Some advanced functions are under development.

Key words: neutronics simulation; four dimensional; fusion; fission; sub-critical fission

NEST: A Nuclear Energy Security Treaty.

Separating Nuclear Energy from Nuclear Weapons.

Brendan McNamara

Leabrook Computing, August 2010

Nuclear energy ought to rise to over 3000 reactors worldwide by 2050 to beat global warming and replace the energy losses from the decline of oil and gas. With over 100 countries involved, with vividly different politics, the NPT is clearly insufficient to ensure global safety & security. A new approach is suggested which will completely cut off Civilian Nuclear Energy from any involvement in Nuclear Weapons. The approach shows how graduated, proportional, and non-military penalties can work for mutual security on a region by region basis. The Baltic Region is well suited to development of the NEST agreements and the conflicted Eastern Mediterranean is used to evaluate the robustness of the proposals. In the absence of an NPT position, Fusion and other emerging nuclear technologies are raising new concerns which would be readily solved in NEST agreements.

MAGNETIC FIELD OF COMBINED PLASMA TRAP

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The MCNPX numerical code has been used to model the neutron transport of a mirror based fusion-fission reactor. The fission reactor part has a cylindrical shape with an inner radius 1.94 m and a 26 m length. The fuel consists of “standard” spent nuclear fuel, the major part of which consists of plutonium isotopes.

Inside the fission reactor core is a vacuum chamber containing a 25 m long hot plasma producing fusion neutrons. Such a scheme of the subcritical system is proposed and studied in the article [K. Noack, V.E. Moiseenko, O. Ågren, A. Hagnestål Annals of Nucl. Energy **38**, Issues 2-3, (2010) p. 578.]. To sustain the hot ion plasma which is responsible for the fusion neutron production, radio-frequency heating could be used. The antenna should be placed near the plasma edge and near the axial end of the neutron radiation zone. In this case, the calculated radiation damage of the antenna is 3.4 DPA per year (311 days of operation).

To reduce the radiation damage of the antenna the plasma column is extended up to 26.2 m, and the antenna location is moved correspondingly outwards. Since plasma extends slightly beyond the fission reactor core in this case, this part of the plasma column is surrounded by a vessel filled with borated water to absorb the outcoming neutrons and protect the antenna. In this case, the calculations indicate that the antenna located near the plasma edge is subjected by a radiation damage of 1.9 DPA per year (311 days of operation). The reduction of the damage is achieved by reducing the neutron flux on the antenna.

In addition, preliminary calculational results for the radial leakage of neutrons through the mantle surface of the fission reactor are presented. These preliminary calculations predict that the power load on the magnetic coils would be small, i.e. lower than 10 kW, for a hybrid reactor producing a power of 1.5 GW_{th}.

A fuel for sub-critical fast reactor

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In a sub-critical fission reactor, the requirements for fuel composition are less restrictive as compared to a critical reactor. For this reason the minor actinides that are the most harmful components of the nuclear waste can be included into fuel. Along with the problem of the nuclear waste transmutation, the problem of minimization of waste production is of current interest. It is not possible to eliminate production of waste at a nuclear power plant, but, as ~~it~~ is shown in this report, it is in principle possible to arrange a fuel composition with no net production of transuranic elements.

The idea is to find the transuranic elements composition in a mixture with the U^{238} isotope, which acts as a source for neutrons and transuranic elements, in which, during fission neutron irradiation, for each transuranic element an exact balance between the transuranic elements production and burnup could be met. The production is due to neutron capture by the neighboring isotope or beta-decay of a lighter isotope. The burnup includes fission, neutron capture and decays. The goal is to construct a solution with a continuous supply of the U^{238} isotope , and with a steady-state content of the transuranic elements.

For the calculations a simplified burnup model which accounts for 9 isotopes of uranium, neptunium, plutonium and americium is used. The calculated fuel composition consists mainly of uranium with minority of plutonium isotopes.

Such a fuel, after usage in a sub-critical fast reactor, should be reprocessed. The fission product content increases during burnup, representing a net production of waste, while the transuranic elements and U^{238} should be recycled into a new fuel. For such a fuel cycle, the net consumption is only for U^{238}, and the net waste production is just fission products.

Fusion neutron generation computations in a stellarator-mirror hybrid with neutral beam injection

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In the paper [Moiseenko V.E., Noack K., Ågren O. "Stellarator-mirror based fusion driven fission reactor" J Fusion Energy 29 (2010) 65.], a version of a fusion driven system (FDS), i.e. a sub-critical fast fission assembly with a fusion plasma neutron source, is proposed. The plasma part of the reactor is based on a stellarator with a small mirror part. In the magnetic well of the mirror part, fusion reactions occur from collisions of an RF heated hot ion component (tritium), with high perpendicular energy with cold background plasma ions. The hot ions are assumed to be trapped in the magnetic mirror part. The stellarator part which connects to the mirror part provides confinement for the bulk (deuterium) plasma.

A more conventional method to sustain the hot ions is ~~the~~ neutral beam injection (NBI). NBI is here studied numerically for the above-mentioned hybrid scheme. For these studies, a new kinetic code, KNBIM, has been developed. The code accounts for Coulomb collisions of the hot ions with the background plasma. The geometry of the confining magnetic field is arbitrary for the code. It is accounted via a numerical bounce averaging procedure. Along with the kinetic calculations the neutron generation intensity and its spatial distribution are computed. The energy spent for one neutron is used to make estimates for the energy efficiency of the system.

DIAMOND DETECTOR FOR HIGH RATE MONITORS OF FAST NEUTRON BEAMS

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High rate neutron monitors are important for application such as Fusion-fission hybrid reactors. Fast neutron counters are being developed for applications such as the study of neutron-induced single event effects (SEE) in microelectronics components, which ahs been recognized as a key threat to the reliability of advanced electronic systems. A new beamline (CHIPIR) dedicated to neutron SEE testing is being built at the ISIS spallation source (UK) to provide fluxes of 10^5-10^7 neutrons s^{-1} cm^{-2} in the 1-800 MeV energy range. This defines new requirements for spectrocopic measurements of neutron beams.
A fast neutron detector is being developed based on a commercial high purity single crystal diamond (SDD) coupled to a fast digital data acquisition system. The detector was tested at the ROTAX beam line which features a pulsed neutron beam generated by proton induced spallation on a Tungsten target at 50 Hz repetition rate. The SDD event signal is digitalized at 1 Gsample to reconstruct its deposited energy (pulse-height) and arrival time; the event time of flight (ToF) is obtained from the recorded proton beam signal t_0. Despite a low average count rate, a fast acquisition is needed since the peak count rate is very high (~800 kHz) due to the pulsed structure of the neutron beam.
Measurements at ROTAX indicate that three characteristics regions exist in the biparametric (ToF, pulse-height) spectrum: i) background gamma events of low pulse-heights; ii) low pulse-height neutron events in the energy range E_n = 1.5-6 MeV ascribed to neutron elastic scattering on ^{12}C; iii) large pulse-height neutron events with E_n > 6 MeV ascribed to $^{12}C(n,\alpha)^9Be$ and $^{12}C(n,n)3\alpha$. This type of detector is proposed as fast neutron beam monitor for CHIPIR.

A NEW GEM BASED NEUTRON DIAGNOSTIC CONCEPT FOR HIGH FLUX NEUTRON BEAMS

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Fusion-fission hybrid reactors will need high flux neutron detectors to diagnose the deuterium-tritium fusion plasmas as well as the fission reactions. In the context of high flux neutron detectors, new GEM based detector are being developed for application to the ITER neutral beam test facility under construction in Padova. Two experimental devices are being built: SPIDER, a 100 kV negative hydrogen/deuterium RF source, and MITICA, a full scale, 1 MeV deuterium beam injector. A number of diagnostics will be deployed in the two facilities to qualify the beams. This paper reports the development of a neutron diagnostic called CNESM (Close-contact Neutron Emission Surface Mapping [1]) which aims to provide the map of the neutron emission from the beam dump surface and that is placed right behind the SPIDER beam dump.
The CNESM uses nGEM as neutron detectors [2]. These are Triple GEM (Gas Electron Multipliers) detectors, equipped with a cathode that also serves as neutron-proton converter foil and ensures the directionality property of this detector. The nGEM readout pads (area 20x22 mm^2) will record a useful count rate of ≈5 kHz, providing a time resolution better than 1 s and a space resolution better than 5 mmThe CNESM was designed on the basis of simulations of the different steps from the deuteron beam interaction with the beam dump to the neutron detection in the nGEM. The CNESM does not resolve the neutron emission profile within one beamlet footprint but is able to detect a 10% change in the neutron emission from individual beamlets. The main information output from the CNESM diagnostic is the map of the deuterium beamlets intensity. This is derived from the neutron emission map with a suitable unfolding algorithm. The final dimension prototype for SPIDER (active area 35.2 x 19 cm2) is currently being built and it will be tested in 2012.

[1] G. Gorini et al, RFX_SPIDER_TN_141 Technical Note, 2011
[2] B.Esposito et al, Nucl. Instr. and Meth. A (2009), doi:10.1016/j.nima.2009.06.101

Neutron sensitivity and γ-ray measurements in a fusion environment

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Fusion-fission hybrid reactors will need dedicated instrumentation to diagnose the deuterium-tritium fusion plasmas, in particular the behaviour of α particles produced in the d + t → n + α reaction. The self sustainment of a fusion reaction depend in fact on efficient energy transfer from 3.5 MeV born α particles to the bulk plasma through multiple Coulomb collisions, which requires a sufficiently long confinement time. A certain degree of concern is however raised by the prediction that α particles may drive unstable certain magneto-hydrodynamic modes which could severely deteriorate the fusion performance or even cause plasma disruption. This motivates the need to study the behaviour of α particles in a dedicated burning plasma experimental device that – to date – has not yet been built.

On the diagnostics side, dedicated experimental techniques are required for this challenging task. A promising option is given by gamma ray emitting reactions between α particles and impurities in the machine, where the observation of Doppler broadened γ-ray peaks provide information on the α particle energy distribution as demonstrated in recent experiments on the Joint European Torus. The implementation of a gamma spectrometry system in the harsh environment of a burning plasma experiment poses stringent technical requirements. Among these, insensitivity to neutron background and capability to operate at high counting rates are mandatory. In this work we present a solution based on a $LaBr_3(Ce)$ detector. High rate operation is obtained through modified photomultipliers, together with a fully digital data acquisition, and demonstrated by gamma ray spectroscopy measurements in the MHz range both at accelerators and in vicinity of a tokamak. Sensitivity to neutron background is determined from 14 MeV neutron irradiation measurements at the Frascati Neutron Generator. The results are discussed in view of requirements for γ-ray spectroscopy observations on the International Thermonuclear Experimental Reactor. Prospects for similar measurements on a fusion-fission hybrid device are also addressed.