

Abstracts for Oral Presentations

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A Practical Way to Fusion Application through Fusion-Fission Hybrid Systems

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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 ongoing efforts to establish fusion as an energy source, there has been a renewed interest in fusion neutron source applications.

The fusion-fission hybrid system is considered to be an intermediate step toward pure fusion application in China. Based on the survey and reviewing of the fusion-fission hybrids related activities previously in the world, a series of hybrid concepts have been designed for re-evaluating the performances of fusion-fission hybrid reactors for energy production, fuel breeding and waste transmutation based on available or very limitedly extrapolated fusion and fission technologies, e.g. at a fusion power level of 50MW, 150MW and 500MW. The performance analyses covering neutronics, thermal-hydraulics and thermo-mechanics, etc. show that the concepts have good feasibility and attractiveness. A practical way to fusion application through hybrid systems has been proposed considering the energy status in China, which includes the pre-testing phase (Phase I) with the accelerator-driven D-T fusion nuclear testing facility HINEG and the superconducting tokamak testing facility EAST, the testing phase with the fusion-fission hybrid multi-functional experimental reactor FDS-MFX (Phase II) and the DEMO phase with the fusion-fission hybrid reactor for fission reactor Spent Fuel Burner FDS-SFB (Phase III). A series of conceptual design and R&D activities are ongoing or planned to support the program.

In this contribution, the proposed development strategy is presented based on the assessment and design analysis of the developed series concepts in China as well as a summary of the fusion-fission hybrid design and R&D activities in China and in the world.

1. Y. Wu, FDS Team, "The CAD-based 4D Neutronics Simulation and Analysis for Fusion/Fusion-Fission Systems". Invited presentation at *the 9th International Symposium on Fusion Nuclear Technology (ISFNT-9)*, Oct. 11-16, 2009, Dalian, China.
2. Y. Wu, J. Jiang, M. Wang, et al., "A Fusion-Fission Reactor Concept Based on Viable Fusion and Fission Technologies," Presentation at *the 23rd Int. Conf. on Fusion Energy (FEC-23)* (Korea, Daejeon, October, 2010) FTP/P1-13, International Atomic Energy Agency.
3. Y. Wu, FDS Team, "Conceptual design activities of FDS series fusion power plants in China," *Fusion Eng. Des.* 81 (2006) 2713-2718.

Development of High-Performance Structural Materials for Fusion

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The fusion nuclear environment presents significant challenges for structural materials to reliably perform their intended function, while maintaining appropriate safety margins against unanticipated failure. Projected atomic displacement damage alone in postulated DEMO type reactors is equivalent to ejecting every atom from its lattice site up to ~200 times. Coupled with displacement damage is production of copious amounts of He and H from transmutation reactions, which is a unique characteristic of the D-T fusion environment, compared to proposed advanced fission reactors. To describe material behavior in this environment requires development of fundamental understanding of the complex interaction between displacement damage and gas production, which causes degradation of a host of material properties, through processes such as hardening, low-temperature embrittlement, phase instabilities, segregation, precipitation, irradiation creep, volumetric swelling, and high-temperature He embrittlement. In addition, plasma-facing components must also tolerate sustained heat fluxes comparable to that experienced by rocket nozzles.

The U.S. Fusion Materials Science Program is developing a fundamental knowledge base of the damage mechanisms controlling the performance limits of structural materials for fusion power systems. Life-limiting materials degradation processes in the fusion neutron environment are being explored through a science-based effort that employs the full range of experimental and computational approaches. The ultimate goal is to develop experimentally validated, physics-based, predictive models of material behavior that can be utilized to improve existing materials or to design superior ones.

In this paper the major life-limiting irradiation-induced degradation phenomena that control the stress-temperature operating window for fusion structural components are reviewed. The possible impact of several of these life-limiting processes on the performance envelope of reduced activation ferritic/martensitic (RAF/M) steels is assessed through key experiments and mechanistically based quantitative models. The latest work to develop 12-14 Cr nanostructured ferritic alloys to expand the operating window beyond that attainable for RAF/M steels is discussed, along with a recently begun activity that employs state-of-the-art computational materials science and key experiments to explore methods to improve the mechanical properties of tungsten and tungsten alloys for divertor applications.

Fusion Technology and Fission Technology – Similarities in R&D Needs

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Collaboration amongst R&D programs, if managed well, is beneficial to all programs involved, and can be a very good way to leverage resources, particularly during lean budget times. Leveraging resources is also a way to leverage influence such as congressional influence – also important during lean budget times. The primary mission of the Office of Nuclear Energy is “to advance nuclear power as a resource capable of meeting the Nation's energy, environmental, and national security needs by resolving technical, cost, safety, proliferation resistance, and security barriers through research, development, and demonstration as appropriate.” The mission of the U.S. Fusion Energy Sciences Program is “to advance plasma science, fusion science, and fusion technology – the knowledge base needed for an economically and environmentally attractive fusion energy source.” At first glance these programs may seem to have very little in common – economic fusion energy has not yet been demonstrated and plasma physics tends to dominate the fusion discussions, while fission energy is currently providing ~20% of US electricity, and the supporting R&D appears more technology-oriented. However it is important to look more closely – there are several areas of R&D common to both programs.

This paper describes opportunities for R&D collaboration, some of which are already being explored, and others which are not yet being discussed. Both programs are focused on nuclear energy – albeit of different types – and this common basis provides the platform for collaborative R&D of benefit to both.

Nuclear Hybrid Energy Systems

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The magnitude and timing of growth in energy consumption likely will create a global imperative to deploy energy production technologies that balance the three pillars of energy security:

- economic stability – related to the affordability of energy products, stability and predictability in their price, and the efficient and effective deployment of global capital resources in their development;
- environmental sustainability – related to minimizing the negative impacts of energy production to air, land, and water systems and advancing the long-term viability of using particular resources in ways that do not limit future generations' ability to prosper;
- resource security – related to the ability to access energy resources and products where and when necessary, in an affordable and predictable manner.

Energy systems that address these elements of energy security will likely require new approaches that more effectively integrate fossil, renewable, and nuclear energy resources into a variety of energy carriers and energy markets. Developing methods and approaches to integrate nuclear energy into a broader range of energy markets and energy carriers (beyond electricity) may be valuable in meeting these challenges.

Nuclear hybrid energy systems (HES) may provide a means to more fully and cost effectively integrate nuclear energy into a variety of energy markets while also enabling the use of renewable and cleaner fossil energy. Broadly described, HES take two or more energy resource inputs (typically including both carbon and non-carbon based sources) and produce two or more energy products (e.g. electricity, liquid transportation fuels, industrial chemicals) in an integrated plant. The concept of using nuclear energy in a variety of non-electrical process applications is certainly not new (e.g. [1]). Additionally, the benefits of loosely-coupled non-nuclear “hybrid” energy plants have also been reported (e.g. [2]). However, tightly coupled HESs, including nuclear hybrids, have not been explored in significant detail until recently. Recent studies have shown that these systems can have appealing attributes in terms of overall process efficiency, enhanced electric grid stability, renewable energy integration, economic performance, and lifecycle greenhouse gas emissions. Ongoing studies at the Idaho National Laboratory [3] offer insight into the potential benefits and architecture of nuclear HESs, suggesting a potentially intriguing approach to enhanced energy security through tightly-coupled nuclear, renewable, and fossil energy conversion systems.

1. Jones, J. E. et al., Assessment of Very High-Temperature Reactors in Process Applications, Oak Ridge National Laboratory Report, ORNL/TM-5410 (1977).
2. Phadke, A. et al., Advanced Coal Wind Hybrid: Economic Analysis, E.O. Lawrence Berkeley National Laboratory Report - LBNL-1248E (2008).
3. Cherry, R. et al., “The Role of Large-Scale Hybrid Energy Systems in Providing Energy Security”, submitted to *Energy and Environmental Science* (2010).

Optimization of Safety Parameters and Accident Mitigation Measures for Innovative Fast Reactor Concepts

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GEN-IV nuclear systems, especially advanced sodium cooled fast reactors (SFRs) are on the horizon and a key issue of their success is the promise of a higher and improved safety level. The time frame for developing these advanced SFRs will cover the next decades. In Europe, safety investigations are currently under way e.g. in the ESFR project of the EU. Both on the prevention and mitigation side significant efforts are invested to fulfill the high safety goals. The route of assurance concentrates on the prevention, the mitigation and elimination of specific severe accident routes and phenomena.

Traditionally in analyzing severe core disruptive accidents (CDA), the accident evolution is broken down into different phases as e.g. initiation phase and transition phase. The first level of optimization and prevention concerns the initiation phase of any transient. The optimization of the void worth and Doppler should lead to a prevention of boiling e.g. for Unprotected loss of Flow (ULOF) situation. In that direction, several attempts were oriented to reduce the positive void effect of the ESFR oxide core. Among all, the increase of the Na plenum height and the addition of an absorber layer above the plenum seem the most promising options. An additional improvement is given by the introduction of a lower fertile axial blanket with 5% vol. of AmO₂; giving a positive contribution in terms of reactivity variation versus burn-up and Minor Actinides balance of the system. Additional attempts (e.g. the introduction of diluents material and the reduction of the height-to-diameter ratio) have been considered within the ESFR project but they cause a strong impact on other core characteristics.

If accident prevention is not achieved the initiation phase leaves behind a neutronic and thermohydraulic instable configuration and inevitably leads to a so-called transition phase generally with higher energetics potential. The transition phase is characterized by a progressive core disruption where local multi-phase fuel/steel pools grow to larger pools with the potential of coherent and compactive fuel motion leading to the occurrence of recriticalities. A competition between fuel losses and in-pool material motion exists deciding over the energetics potential of the transition phase. Entering the transition phase, more event channels are opened and the range scale for reactivity is significantly increased. This leads to special requirements for elaborated codes and tools. The SIMMER-code family has been further developed for this purpose.

To get a control of the transition phase energetics, the idea has been brought forward to install dedicated means in the core that enhance and guarantee a sufficient and timely fuel discharge - a controlled material relocation (CMR) - and influence and 'brake' the recriticality path. Several proposals are under way to accomplish this CMR. Fast reactor developments cover designs from burner to iso-generator to breeder reactors. Each type shows a different event space under severe conditions and requires specific CMR measures. In the current paper new views, concepts and the status of analyses are described and reflected especially focusing on the implementation of such measures without interfering with the basic core design.

Issues on Application of Commercial LWR Core Areas for Irradiation and for Testing of Advanced Fuels

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We have performed a set of dedicated nuclear core calculations related to the behavior of single fuel test rods and of whole fuel bundles loaded in typical fuel element positions in commercial LWRs. The motivation of such studies includes technical and economic assessment of various new nuclear reactor applications, testing and development of common reactor physics codes under new parameter ranges, and out of the ordinary reactor physics topics for education and training purposes. We discuss the use of single Th-fuel bundles both in PWR and BWR cores. Core calculations are performed by Monte Carlo methods (MCNP, Serpent, FLUKA) coupled to burn-up calculations by CASMO, Serpent, Monteburns, and DeTra. As case studies we briefly report the performance of a Radkowsky-type thorium fuel bundle in PWRs and discuss the viability of such fuel options in open cycle from operation, safety, energy production, and waste disposal points of view. During these studies improvements in speeding up the Monte Carlo codes and increasing the accuracy of burn-up calculations have been implemented. Simulation of BWR codes offers more versatile options and we have done core calculations for both single thorium bundles and the effect of their structure within the bundle, control rod and water channels. Use of single test rods in BWRs for isotope production or waste transmutation has been explored, too.

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Neutronic Assessment and Criticality Analysis of the In-Vessel Fuel Storage Facilities in the CDT/FASTEF Reactor Design Project

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The Central Design Team (CDT) project is established in the framework of the EU FP7 EURATOM Programme, under the topic of Radioactive Waste Management. One of the main components of the project is devoted to the design of a Fast Spectrum Transmutation Facility (FASTEF) [1]. The main goal of the CDT project is to establish a preliminary design of a prototype of an industrial-scale Accelerator Driven System (ADS), strongly based on the MYRRHA reactor concept [2]. An important issue regarding the reactor design of the FASTEF facility is the fuel storage, both for fresh and spent fuel, as it entails changes in the criticality of the overall system. The presence of four In-Vessel Fuel Storage Facilities (IVFSF) is foreseen in the present design of the reactor. The four IVFSF will be able to store a maximum of two full core loads, each containing 76 MOX type assemblies. In order to avoid excessive delays between two operation cycles, the project envisages storing the fuel at the reactor periphery. It is then crucial that during the reactor operation the neutronics interaction between the central core and the fuel assemblies in the storage zones is assessed, both in terms of criticality assessment and material damage to the overall structure. Therefore, a neutronics analysis of the IVFSF must be made both for the fresh and for the spent fuel. Whilst the former may entail a dangerous neutronic coupling between the core and storage facility, the latter represents a source of decay heat that can induce damage to the vessel materials.

The state of the art Monte Carlo program MCNPX and ORIGEN 2.2 computer code system were used to determine the buildup, decay and processing of radioactive materials. The criticality behavior (namely the K_{eff}), the neutron fluxes and their variations, the fission power production and the radiation damage (the displacements per atom) were also analysed. Finally, the power heat generated by the fission products decay in the spent fuel was also assessed. This work provides evidence that the current design guarantees the neutronic stability of the facility, in what concerns the fuel storage (both fresh and spent) meeting nuclear safety requirements of the facility.

1. Annex I, "Description of work" to EURATOM Collaborative Project "Central Design Team (CDT) for a Fast Spectrum Transmutation Experimental Facility", Grant agreement no. FP7-232527
2. MYRRHA experiment website, <http://myrrha.sckcen.be>

High-Fidelity Modeling of VHTRs for Performance Evaluations

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This paper will discuss details of the high fidelity modeling of VHTRs. The focus is on the prismatic core configurations targeting performance evaluations. Prismatic block configurations allow for a variety of advanced operation modes including extended operation scenarios for applications with minimized refueling, autonomous operation with minimized control and deep burn scenarios with advanced 3D core shuffling patterns to minimize used fuel TRU inventories. Coupled whole-core reactor modeling approaches and automation algorithms will be presented as they have been developed and are used in several projects focused on advance VHTR performance capabilities. Sensitivity/uncertainty studies will be presented showing potential metrics needed to quantify performance domains and assess fluctuations in decisions with respect to design parameters and predictions of performance characteristics. The specific focus will be on the ability to represent and track individual blocks, identify needs for in-core instrumentations and optimization approaches.

Status, Generic Technical Issues and Prospect of Small and Medium-sized Reactors Development and Deployment

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There is a growing interest in the development and deployment plan of Small and Medium Sized Reactors (SMRs) which can be seen through the numerous concepts that are under design certification and the several units that are under construction, the expanding of potential markets in developing countries, and the increasing demands from the Member States on the supports and role of the International Atomic Energy Agency (IAEA). The SMRs have been studied extensively over the past three decades and several countries have SMR designs ready for deployment. The paper summarizes the status of SMRs designs that are under development globally with the aim to give a broad perspective. Furthermore, it presents generic technical issues and prospect of SMRs, the advantages and challenges that concern SMRs and parameters that potential markets take into consideration with regard to the introduction of the first nuclear power plant. Some of the issues to be addressed are economics, licensability, operational performance, waste management, safety and security, proliferation resistance and financing scheme. In the end, the IAEA program on the common issues and technology of SMRs are highlighted with the purpose to facilitate countries with interest in SMRs. The Division of Nuclear Power of the IAEA is currently developing a set of comprehensive information, evaluation, and methodologies needed for a balanced assessment of SMR design.

The Opportunities and Challenges in Developing New Structural Materials That Will Enable Advanced Nuclear Energy Systems

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Advanced nuclear energy systems will require design, development and demonstration of new materials that provide a balanced set of outstanding performance sustaining properties (strength, toughness corrosion resistance, etc.) under extended service conditions that are uniquely severe. For example, the feasibility of fusion depends on materials and structures that can withstand unprecedented thermal loading over long times, while sustaining high levels of dpa and helium accumulation, reaching 200 dpa and 2000 appm, respectively. Likewise, the prospects for developing advanced fission energy systems that operate at high efficiencies to high fuel burnups (e.g. levels of 40%), and even that provide single 60 year single core lifetimes, hinge on the development and demonstration of cladding-duct materials that can sustain damage up to 400 dpa, or more.

There is little prospect that currently available materials, or even many conceptual candidates, can meet these challenges. Thus it is critical that materials research and development be guided by a combination of state of the art science, combined with very practical and comprehensive experience based engineering insight on what is required to transform a materials concept to a reality in serving in an advanced nuclear energy system reality.

After briefly outlining the need for advanced materials, we focus on the opportunities and challenges for achieving these objectives with so-called nanostructured ferritic alloys (NFAs). NFAs have high strength that permits service to more than 800°C, manifest remarkable resistance to radiation induced displacement damage and can manage high concentrations of helium. However, there are a multitude of challenges to the development and application of NFAs, including: composition-synthesis designs, thermal mechanical processing paths and fabrication-joining methods that optimize properties: reducing costs and creating industrial scale supply sources for large and consistent heats; and, new methods to qualify advanced alloys for nuclear service. A possible roadmap for achieving the goals is described.

Compatibility Assessment of Advanced Stainless Steels in Sodium

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Type 316L stainless steel capsules containing commercially pure sodium and miniature tensile specimens of HT-UPS (austenitic, 14Cr-16Ni), NF-616 (ferritic/martensitic, 9Cr-2W-0.5Mo), or 316L (austenitic, 17Cr-10Ni-2Mo) stainless steel were exposed at 600 or 700°C for 100 and 400 h as a screening test for compatibility. Using weight change, tensile testing, and metallographic analysis, HT-UPS and 316L were found to be largely immune to changes resulting from sodium exposure, but NF-616 was found susceptible to substantial decarburization at 700°C. Subsequently, two thermal convection loops (TCLs) constructed of 316L and loaded with commercially pure sodium and miniature tensile specimens of HT-UPS and 316L were operated for 2000 h each – one between 500 and 650°C, the other between 565 and 725°C – at a flow rate of about 1.7 cm/s. Changes in specimen appearance, weight, and tensile properties were observed to be very minor in all cases, and there was no metallographic evidence of microstructure changes, composition gradients, or mass transfer resulting from prolonged exposure in a TCL. Thus, it appears that HT-UPS and 316L stainless steels are similarly compatible with commercially pure sodium under these exposure conditions.

Operating Experiences of Main Steam Flow-based Power Monitoring to OPR1000 Plants

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We have developed a new secondary calorimetric power calculation methodology based on the main steam generator's steam flow called MSBSCAL as an alternative licensed power monitoring indicator [1]. This MSBSCAL calculation methodology was introduced to resolve the flow venturi or nozzle fouling issue which degrades the thermal performance in Korea Optimized Power Reactor 1000 (OPR1000) plants. Recently, the implementation of the MSBSCAL calculation methodology has been completed and its operating data have been analyzed for some OPR1000 plants. A typical phenomenon of flow venturi fouling issue has occurred in YGN-4 Cycle 12. The main feedwater-based secondary calorimetric power (FWBSCAL) has been increased since the Middle-of-Cycle (MOC), while the MSBSCAL has maintained stably in the cycle. Due to the reduction of electric power caused by flow venturi fouling, the economical efficiency is estimated by one million dollars for the cycle. In UCN-4 Cycle 10, a large standard deviation error of FWBSCAL was observed after replacing flow the venturi and pumps during overhaul periods. Despite the unusual behavior of FWBSCAL during operation, the MSBSCAL did not show any other abnormalities as a power monitoring indicator. Therefore, the MSBSCAL calculation methodology has been verified as an alternative licensed power monitoring indicator in case of FWBSCAL calculation is not reliable or unstable like flow venturi fouling issue.

1. Byung Ryul Jung, et al., "An application of Main Steam Flow-Based Power Monitoring to OPR1000 Plants," *Proceedings of the 17th International Conference on Nuclear Engineering*, ICONE17-75642 (2009).

Investigating the Relationship between Sic Composite Fabrication Parameters and Mechanical Performance

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SiC composites are an enabling technology for next generation high temperature nuclear reactors. These materials maintain excellent material properties at high neutron flux and high temperature (1200oC). General Atomics has established a facility for SiC depositions, and this work is focused on the optimization of thermomechanical properties of SiC-SiC composites. Stoichiometric SiC fiber-SiC matrix composites have been fabricated via chemical vapor infiltration (CVI) from a MTS precursor. Specimens with varying densities (70-85%) were made by varying CVI process conditions (time, pressure, gas mixture), in order to investigate how tensile and flexural strength, as well as the thermal conductivity of SiC composites vary with density. The effect of altering fiber/matrix volume fractions, as well as the dependence of thermomechanical performance on fabric orientation, will be presented. Additionally, initial investigations on the viability and performance of preceramic polymer, and hybrid CVI-preceramic polymer-derived joining technologies will be discussed.

The Impact of Refractory Material Properties on the Helium Cooled Divertor Design

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Present US and European helium cooled DEMO divertor design studies make use of the high temperature strength and good heat conductivity of refractory materials, like for example, tungsten alloys. In such outlines, various materials are used for structural parts. One of the most critical issues of tungsten materials in connection with structural applications is probably the ductile-to-brittle transition (DBT) which starts already at rather high temperatures for tungsten materials. Depending on irradiation and dynamic load conditions, the DBT level could be even higher.

Within a comprising study the influence of microstructure and chemical composition on the impact bending properties of industrially produced tungsten materials was investigated. Also, the effect of different half-finished products like rods, plates, and round blanks were taken into account.

As a result, tungsten materials look quite different compared to those of typical bcc metals. Only specimens of pure tungsten rods show a clear upper shelf area, starting at about 800 °C. All other rod materials don't show pure ductile fracture within the whole test temperature range. On the other hand, all tested rod materials tend to brittle fracture at temperatures below 500 °C. But above that temperature, the specimens show delamination fractures which propagate along the rod axis, that is, parallel to the specimen's long side and perpendicular to the notch. Compared to the rods, all plate and round blank materials show even worse properties: (1) the energies are lower by more than 50 %, and (2) the fracture is dominated by delamination within the whole temperature range. In summary, there are three types of fracture behaviour (brittle, ductile, and delamination) which is closer correlated to the materials microstructure than to the materials composition.

This paper reviews the results and other relevant properties of tungsten materials with respect to possible applications for structural divertor parts. Drawbacks and possible alternatives are discussed.

Evaluation of Heat Transfer in High-Temperature Refractory Foam Heat Exchangers using Computational Fluid Dynamics

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Refractory metallic foams can increase heat transfer efficiency in gas-to-gas and liquid metal-to-gas heat exchangers by providing an extended surface area for better convection, i.e. conduction into the foam ligaments providing a “fin-effect,” and by disruption of the thermal boundary layer near the hot wall and ligaments by turbulence promotion. In this article, we report on progress in computational fluid dynamics (CFD) modeling of actual foam geometries obtained through computerized micro-tomography. The article outlines the modeling procedure from geometry import through meshing and analysis using the 64-bit parallel processing capabilities of common modern workstations and several commercial software packages.

We present the relative contributions of the heat transfer mechanisms stated above, and show how the design of a gas regenerator or liquid metal-to-gas heat exchanger can be optimized for use in high-temperature Brayton cycle applications for nuclear power generation or hydrogen production. Our results include velocity, pressure drop, temperature and thermal stress distributions for several densities of Nb, Mo and W foams compared to Cu. For instance, the simulations reveal that unconnected W foam can increase the convective heat transfer coefficient by almost a factor of two compared to an open rectangular channel and a factor of three if the foam ligaments are thermally connected to the sidewalls under the same flow conditions. The effect of ligament thermal conductivity is also highlighted by comparing the performance of W foams to identical Cu foams and the use of SiC foams in thermal barrier applications. The studies indicate that thermal stresses increase with foam density, but are not clearly correlated with pore cell size. We also discovered that wall-connected refractory foam in the gas side of a liquid metal-to-gas heat exchanger can improve the heat transfer measurably through the wall adjacent to the open channel. This is significant for improving the performance of liquid metal-to-gas heat exchangers. The presence of the connected foam minimizes the thermal stresses in the wall, by concentrating them in the ligaments where the temperature gradients are higher. In addition, the large number of small connected ligaments provides a modest degree of compliance for thermal expansion of the hotter walls in relation to the colder portions of the heat exchanger. These CFD studies have led to design strategies for creating compact, high-temperature, high-pressure heat exchangers that are easily fabricated and perform better than plate-type heat exchangers.

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Conceptual Design of Fusion-Fission Hybrid Multi-Functional Experimental Reactor (FDS-MFX)

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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. A multi-functional fusion-fission reactor concept named FDS-MF simultaneously for nuclear waste transmutation, fissile fuel breeding and thermal energy production based on viable technologies i.e. available or limitedly extrapolated fission and fusion technologies is developed by FDS Team [1-4].

For the FDS-MF concept, two stages, from the experimental stage named FDS-MFX to the DEMO stage named FDS-SFB, have been designed. The final goal of the concept may be a fusion-fission hybrid reactor for fission reactor Spent Fuel Burner named FDS-SFB as a DEMO reactor.

In this contribution, a fusion-fission hybrid multi-functional experimental reactor concept, i.e. fusion-fission hybrid Multi-Functional eXperimental reactor named FDS-MFX, has been proposed for checking and validating the DEMO reactor FDS-SFB blanket relevant technologies based on the viable fusion and fission technologies. The tokamak can be designed based on relatively easy-achieved plasma parameters extrapolated from the successful operation of the Experimental Advanced Superconducting Tokamak (EAST) in China and other tokamaks in the world, e.g. with a tokamak core of the fusion power of ~50MW, the power gain of ~1, the average neutron wall loading of ~0.2 MW/m². The subcritical blanket can be designed based on the well-developed technology of fission power. Based on the different purposes of the FDS-MFX, there are two sub-phases of choosing of the fission fuel: natural uranium for hybrid reactor principle validation phase in prior-period of the experiment, and enriched uranium and spent fuel from PWRs (uranium, plutonium and minor actinides) for engineering validation phase in the late period of the experiment. The design and optimization of fusion plasma core parameters, blanket neutronics, blanket thermal-hydraulics, safety and environmental impact analysis, tritium system and auxiliary system have been presented.

The performance analysis of design and optimization preliminarily demonstrated the engineering feasibility of the design.

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, "The CAD-based 4D Neutronics Simulation and Analysis for Fusion/Fusion-Fission Systems". Invited presentation at the *9th International Symposium on Fusion Nuclear Technology (ISFNT-9)*, Oct. 11-16, 2009, Dalian, China.
3. Y. Wu, J. Jiang, M. Wang, et al., "A Fusion-Fission Reactor Concept Based on Viable Fusion and Fission Technologies." Presentation at the *23rd Int. Conf. on Fusion Energy (FEC-23)* (Korea, Daejeon, October, 2010) FTP/P1-13, International Atomic Energy Agency.
4. Y. Wu, FDS Team, "Conceptual design activities of FDS series fusion power plants in China," *Fusion Eng. Des.* 81 (2006) 2713-2718.

A Fusion Hybrid Reactor Based on The Gasdynamic Mirror (GDM)

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A fusion-fission hybrid reactor whose fusion component is the gasdynamic mirror (GDM) is proposed for power production that could meet the world's energy needs of the next several decades. The choice of the GDM is based on the fact that it is linear, axisymmetric and can operate in steady state. Since the primary role of the fusion component is to supply neutrons to the blanket, it can operate at or near "breakeven" condition, a much less stringent condition than that required for a pure fusion reactor. A large aspect ratio GDM is desirable because of MHD stability considerations, and if we choose such a geometry then a cylindrically symmetric plasma with a surrounding blanket can be treated as semi-infinite cylinders, allowing for the reactor performance to be determined by two, one-dimensional equations: one describing the time evolution of the fissile material density bred in the fertile blanket, and another describing the diffusion of fast neutrons in that region. Our choice for the blanket material is thorium-232 in order to take advantage of the thorium fuel cycle that leads to the breeding of uranium-233. Such a fuel cycle is known to be resistant to proliferation and clandestine operations. We choose to operate the GDM at 0.01 of breakeven, using deuterium-tritium (DT) plasma at a density of 10^{16} cm^{-3} , and a temperature of 10keV. We find that for a reasonable design, such a reactor can generate several megawatts of thermal power per cm "safely" because it is "subcritical", and "securely" because of our choice of the fuel cycle. A systems analysis reveals that about 2% of the net electric power is needed to sustain the fusion component. Moreover, we show that it takes approximately 4 months to reach steady state due to the several steps involved in the breeding cycle.

Overview of Fusion-Fission Hybrid Blankets for Laser Inertial Fusion Energy (LIFE) Engine

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The Laser Inertial Fusion Energy (LIFE) concept is a fusion system for energy production that can also employ hybrid fusion-fission blankets [1]. The present work details designs for various hybrid blanket options and associated missions. Hybrid systems have been designed to generate power, incinerate waste and burn both fertile and fissile nuclear fuel. The blanket designs are modular allowing for periodic replacement of life-limited components and high plant availability. Each module contains moderating and multiplying materials in between the neutron source and the fuel to optimize the neutron spectrum. The fuel blanket is composed of TRISO fuel particles encapsulated in graphite pebbles cooled by a molten salt. In a typical design, low-yield (~25-40 MJ) targets and a repetition rate of ~10-15 Hz produce a 300-500 MW fusion source in the form of neutrons, high-energy photons and ions. Allowing for target injection and laser penetration, the blanket modules form a compact (2-4 m diameter) target chamber surrounding this fusion source. The resulting high neutron flux drives fissile material production and destruction providing a blanket energy gain (total thermal power-to-fusion power ratio) of 4-8, depending on the fuel and mission objective.

We employ a methodology using ⁶Li as a neutron absorber to generate self-sustaining tritium production for fusion and to control power over the lifetime of the engine. This enables deeply subcritical operation while achieving high fuel burnup. In a single pass, fertile uranium and thorium blankets achieve fuel utilization beyond 80% without chemical reprocessing or isotopic enrichment. Fissile blankets can destroy more than 90% of the initial load of weapons grade plutonium or highly enriched uranium while producing 2-4 GW of thermal power.

1. Moses, E. et al., "A Sustainable Nuclear Fuel Cycle Based On Laser Inertial Fusion Energy (LIFE)," *Fusion Science and Technology*, **56**, 2, 566-572 (2009).

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Development of High-Power Laser Based Nuclear Applications

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Since the advent of high-power and short-pulse lasers the focused intensity reaches up to 10^{22} W/cm² which leads to the generation of relativistic laser plasmas. High-power lasers induce electric fields of more than 10^{12} V/m. In consequence, total laser pulse energy is converted to electrons with energies of several MeV, reaching up to 1 GeV in the ultrarelativistic regime. Subsequently, secondary particle beams of protons, heavy ions, neutrons and high-energy photons will be produced by these high-energy electrons that were generated during relativistic laser-plasma interactions. The laser driven particle sources provide an alternative technique to conventional accelerators and reactors in terms of nuclear physics applications and material sciences. To control the laser assisted particle production it is mandatory to determine and characterize the electron dynamics in the relativistic laser-plasma interaction regime.

We present a novel nuclear activation-based method for the investigation of the high energy photons and electron dynamics within the laser-plasma interaction zone. It is based on high density activation targets which are a pseudoalloy of several selected isotopes with different photo-neutron disintegration reaction thresholds. The gamma decay spectrum emitted by the activated targets was used for the deconvolution of the bremsstrahlung spectrum generated by the electrons. This allows the reconstruction of the spectrum of bremsstrahlung photons without any anticipated fit procedures. Furthermore, the characterization of the electrons in the interaction zone is accessible immediately.

The consolidated findings about the interaction mechanisms could be used to realize, control and characterize laser driven particle generation, such as a pulsed neutron source for nuclear and material sciences using special target designs and materials in a pseudoalloyic compound of isotopes. An additional application is the laser assisted nuclear transmutation to produce short-lived isotopes with activities suitable for medical diagnostics and therapy.

Based on first results, future developments of high-power laser based nuclear applications will be discussed in this presentation.

Fusion-Fission Research Facility (FFRF) as a Practical Step Toward Hybrids

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The project of ASIPP (with PPPL participation), called FFRF, ($R/a=4/1\text{m/m}$, $I_{p1}=5\text{ MA}$, $B_{\text{tor}}=4\text{-}6\text{ T}$, $P_{\text{DT}}=50\text{-}100\text{ MW}$, $P_{\text{fission}}=80\text{-}4000\text{MW}$, 1 m thick blanket) is outlined. FFRF stands for the Fusion-Fission Research Facility with a unique fusion mission and a pioneering mission of merging fusion and fission for accumulation of design, experimental, and operational data for future hybrid applications. The design of FFRF will use as much as possible the EAST and ITER design experience. On the other hand, FFRF strongly relies on new, Lithium Wall Fusion plasma regimes, the development of which has already started in the US and China.

The mission of FFRF is to advance fusion to the level of a stationary neutron source and to create a technical, scientific, and technology basis for the utilization of high-energy fusion neutrons for the needs of nuclear energy and technology.

The mission of FFRF is unique, ambitious, and at the same time realistic. It took more than 25 years for the ITER project [1] to approach the construction phase, and still there are numerous uncertainties in the design of its critical elements and systems. The major reason of slow progress in ITER is related to uncertainties in the plasma regime, whose understanding is still evolving without giving sufficient confidence in performance of the next step facility.

Unlike ITER, which has chosen the conventional plasma physics concept for a burning plasma regime, FFRF takes advantage of a new approach to magnetic fusion, which emerged during the last decade (since Dec. 1998) [2,3,4].

For magnetically confined plasma, it is much more efficient to prevent its cooling by neutrals recycled from the walls, rather than to rely, as the conventional approach, on extensive heating power in order to compensate the otherwise essentially unlimited cooling.

The strategy of FFRF is to design the machine in parallel with the supporting experimental and technology development of the LiWF regimes, which is already started in the US and China.

1. Shimada, M., Campbell, D.J., Mukhovatov, et al. "Chapter 1: Overview and summary". 2007, *Nucl. Fusion*, **47**, pp.S1—S17
2. Krasheninnikov, S.I., Pereverzev, G.V., and Zakharov, L.E., 2003. "On lithium walls and the performance of magnetic fusion devices". *Phys. of Plasmas*, **10**, pp.1678--1682.
3. Zakharov, L.E., Gorelenkov, N.N., White, R.B., Krasheninnikov, S.I., and Pereverzev, G.V., 2004. "Ignited spherical tokamaks and plasma regimes with LiWalls". *Fusion Eng. Design*, **72**, pp.149—168
4. Zakharov, L.E., 1999-2009. Lithium: the key to fusion power. On the WWW, at <http://w3.pppl.gov/~zakharov>. Professional web-site.

Comparisons of the Calculations Using Different Codes implemented in MCNPX Monte Carlo Transport Code for Accelerator Driven System Target

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We have analyzed the main quantities determining ADS performance such as spallation neutron leakage, the number of produced neutron yields and their spatial and energy distributions; light charged particle production, protons and alphas, and energy deposition in natural lead target by MCNPX Monte Carlo transport code, which is a combination of LAHET and MCNP codes. The MCNPX code offers options based on physics packages; the Bertini, ISABEL, Cem2k, INCL4/ABLA intra-nuclear codes. Results obtained simulating different model codes implemented in MCNPX are compared with available experimental data.

The system is composed of a natural lead cylindrical target, which has been optimized to produce maximum number of neutrons with a radius of 20 cm and 70 cm of height. Target is bombarded with a high intensity linear accelerator by a 1 GeV proton beam. The protons are assumed uniformly distributed over a cross the beam of radius 3 cm, and entering the target through a 5.3 cm radius hole. The proton beam has an outer radius of 5.3 cm and an inner radius 5.0 cm. The beam window is made of stainless steel, HT9.

An Assessment of Acceleration Techniques in Scattering Source Iterations

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When neutron scattering probability relative to absorption is large, scattering source iteration (i.e. in discrete ordinates calculations) becomes slow and the introduction of acceleration techniques becomes necessary in transport calculations. Two of the most prominent acceleration techniques are the coarse mesh rebalance (CMR) [1] and the diffusion synthetic acceleration (DSA) methods. CMR is based on the principle of neutron conservation in each of the so called coarse mesh regions. The choice of the coarse mesh regions and application frequency are the two independent factors which are critical in the success of the CMR technique. On the other hand, DSA is based on the principle of augmenting the transport calculations by diffusion theory calculations for acceleration of the scattering source iterations. Consistency of the spatial discretization methods in transport and diffusion theory calculations is found to be the critical factor in successful application of DSA to scattering source iterations. Both DSA and CMR have been used in various transport (S_N) codes for the acceleration of scattering source iterations.

In this work, the relative merits of DSA and CMR in the acceleration of the scattering source iterations are tried to be assessed. A discrete ordinates program for spherical geometry transport calculations with both CMR and DSA acceleration options for scattering source iterations have been developed. The spatial discretization is based on diamond differencing (DD). Gauss-Legendre quadratures up to S_{256} are allowed for the treatment of angular variables. Since the objective of the present study is the assessment of the scattering source iterations, the cases considered are mostly chosen from external neutron source driven systems.

Application of the CMR to each coarse mesh region is based on the neutron continuity equation with the partial currents representation. The effect of the size of the coarse mesh regions (spherical shells in the present study) and the frequency of the application of CMR on the convergence rate of the scattering source iterations are studied. The diffusion theory calculations in DSA are based on the cell-centered finite difference method to preserve spatial discretization consistency with the DD method of transport calculations.

The numerical experiments carried out show that both the CMR and DSA techniques are effective in acceleration of the scattering source iteration in discrete ordinates calculations irrespective of the degree of the S_N approximation. The effectiveness of the acceleration techniques becomes more pronounced in problems where the ratio of the scattering to total cross section is high, as expected. In some of the cases considered, the diffusion synthetic acceleration seems to be more effective than the coarse mesh rebalance technique in reducing the number of iterations to convergence in scattering source iterations.

1. A. Yamamoto, et al., "Convergence Improvement of Coarse Mesh Rebalance Method for Neutron Transport Calculations," *Journal of Nuclear Science and Technology*, **41**, 781 (2004).

Study on Hydrodynamic Characteristics of Two-Component Two-Phase Flow in Anti-Siphon Equipment of China Experimental Fast Reactor

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In order to satisfy the requirement of high purity sodium in China Experimental Fast Reactor (CEFR), there sets a sodium purification system in the primary coolant circuit [1]. When the pipe in this system suffers any pipe-break accident, the anti-siphon equipment, which is set in the in-core part of this system, could passively reduce the accident leakage of sodium.

To validate the anti-siphon operation of this anti-siphon equipment, two methods are used: one is the experiment; the other is the numerical simulation analysis.

During the experiment [2], the hydrodynamic characters of the innovated inlet of sodium pickup are certified, and the test results approve that this structure specializes in increasing the leakage of gas and reducing the leakage of liquid. Furthermore, this experiment is served for the engineering validation, so the configuration of its system must be obeyed to the similarity criteria.

During the numerical simulation, a code for one dimensional and unsteady two fluid flowing is written based on the leakage phenomenon in the anti-siphon equipment's pipe line system. This code adopts the two-fluid flow model, carefully takes account of the gas-liquid interfacial friction, and uses the improved finite volume method to numerically resolve this model. Moreover, the two phase flow oscillation state, caused by the special structure of the anti-siphon equipment, is also numerically simulated. Compared to the experimental results, this model can correctly describe the phenomenon of the whole leakage process.

According to the experimental results and the numerical simulations, the function of the anti-siphon equipment, which is passively reducing the leakage of sodium, is obtained good validations. At the same time, this innovated anti-siphon equipment is certificated to be an efficiently and reliably passive facility, as an engineered safety feature, used in CEFR.

1. Peng Yan, Zhang Donghui, Analysis of Sodium/Argon Flow in Anti-Siphon Equipment's Curved Pipeline of China Experimental Fast Reactor, *Chinese Journal of Nuclear Science and Engineering*, Beijing (2008)
2. Ding Zhenxin, CEFR Anti-siphon Equipment Experiment Validation Review Report, *China Institute of Atomic Energy*, Beijing (2005)

Photo-Neutron Source by High Energy Electron on High Z Targets: Comparison Between Monte Carlo Codes and Experimental Data

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We have designed and realized a photo-neutron source at the Beam Test Facility of the e⁻/e⁺ collider DaΦne, at the National Laboratory of Frascati, near Rome (Italy).

Neutrons are produced sending high energy electrons to impinge on a tungsten Target. Electrons interacting with the target lose energy mainly by bremsstrahlung: the ensuing photons can be absorbed by the target nuclei that in such a way are excited. These excited nuclei decay into the fundamental state by boiling-off nucleons, typically a neutron.

In this way we are able to produce neutrons that have an energy spectrum that spans over several decades of energy (from 1.E-9 MeV up to hundred MeV), but the majority of neutron population (more than 80%) is peaked around 1 MeV, with Maxwellian distribution.

This source could be suitably used for calibration of neutron detectors and for material and nuclear science investigations. Moreover photo-neutron processes are encountered in many physics domain: from accelerator to reactor physics, mainly because related to neutron shielding issues in high Z materials, used for gamma shielding.

We will present the Monte Carlo simulations that we performed with three different codes (FLUKA, MCNPX and GEANT4) to estimate the neutron rate and energy spectrum of the expected neutrons, when 510 MeV electrons are sent against an optimized W target. Finally the comparison of the Monte Carlo predictions with experimental measurements, by Bonner Sphere Spectrometer (BSS), will be shown and discussed. Computational details of photonuclear physics implemented in the used codes will be underlined in order to compare qualitatively and quantitatively the three cited Monte Carlo codes.

1. G. Mazzitelli et al., "Neutron Source at the DaΦne Beam test facility," *MOPEB063, Proceeding of IPAC'10, Kyoto, Japan (2009)*.
2. L. Quintieri et al., "Feasibility Study of a neutron source at the DaΦne Beam test facility, using Monte Carlo codes". *2009 IEEE Nuclear Science Symposium Conference Record (N33-6)*

Thermal-Hydraulic Simulation of 80 kA Safety Discharge in the ITER Toroidal Field Model Coil (TFMC) using the 4C Code

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The 4C system code has been recently developed [1] for the simulation of thermal-hydraulic transients in the superconducting coils and related cryogenic circuit of the International Thermonuclear Experimental Reactor (ITER). The code is now undergoing an extensive validation and application campaign, see e.g. [2], of which this paper will present some new results.

4C implements a model of the entire superconducting coil system, coupling 1D compressible flow of the supercritical He coolant along each dual-channel cable-in-conduit conductor (CICC) of the winding, with 3D heat conduction in the solid structures, and 0D-1D conservation laws for the cryogenic circuit components (e.g., pumps, valves, cryolines, heat exchangers).

In this paper present the results of a simulation of a safety discharge of the ITER Toroidal Field Model Coil (TFMC) from its maximum operating current of 80 kA. The TFMC was extensively tested in 2001-2002 in the TOSKA facility at the Karlsruhe Institute of Technology (then Forschungszentrum Karlsruhe), Germany. The thermal-hydraulic transient analyzed here is driven by the time dependent heat deposition in the CICCs, radial plates and coil case, due to AC losses and dissipation of eddy currents induced by the dump of the coil current. The computed evolution of the system (temperatures, pressures and mass flow rates at different locations in the winding, case cooling channels and case surface) will be compared with the experimental results.

1. L. Savoldi Richard, F. Casella, B. Fiori, and R. Zanino, "The 4C code for the cryogenic circuit conductor and coil modeling in ITER", *Cryogenics*, **50**, 167 (2010).
2. R. Zanino, D. Bessette, and L. Savoldi Richard, "Quench analysis of an ITER TF coil", *Fus. Eng. Des.*, **85**, 752 (2010).

Progress in Advanced Materials Under Extreme Conditions for Nuclear Fusion

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Inertial Fusion is an exigent scenario requesting the study and development of new advanced materials able to satisfy extreme conditions. In particular, target design will require detailed information of non-linear wave propagation and response to high shock loading for different materials part of critical components. Atomistic simulations, using advanced Molecular Dynamics and computational methods, will play a crucial role in this task.

We would like to report non-equilibrium Molecular Dynamics simulations, by means of empirical potentials, able to cover similar times and length scales as laser-shock experiments. We are modeling shock-wave generation and propagation in single crystal materials Fe, Au, Ta, W, and Al by means of different MD methodologies. Double layer conformations FeAl, FeV are also being evaluated. New nanostructured materials, like nanocrystalline Fe, Cu, and Ni are being tested under high pressure conditions. In addition, we have developed a method to model nanoscale porous materials in an “*ad hoc*” target scenario.

Nanoscale impacts are also evaluated in first wall materials, both in traditional approaches and nanocrystalline constructions.

Our simulations have proved to be very valuable for improving current target and chamber designs for IF reactor scenarios. Finally, we will evaluate their general behavior and discuss their suitability as future general compounds in the scenario of Nuclear Fusion Technology.

Inertial Fusion Energy (IFE) Programs at General Atomics

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In previous years General Atomics has been involved with all prominent approaches to Inertial Fusion Energy (IFE), including laser driven fusion, heavy ion driven fusion, and Z-pinch based fusion energy. GA's role in these IFE programs has focused primarily on the target technology, including the entire target supply chain ranging from target manufacture to target injection, tracking, and engagement. This role has effectively leveraged off our much larger target fabrication contract for the US national ICF program. While there is currently no IFE program within the US Department of Energy, IFE target technology work continues at GA on internal R&D funding. Our efforts have been re-focused to a nearer term step "on the pathway to IFE" – i.e., targets and insertion systems appropriate for the new generation of powerful rep-rated lasers that are now coming on line. These lasers can be optimally utilized with complex targets, and their ability to shoot at repetition rates approaching once per minute will require substantial improvement in the throughput for target production. In addition, the cost per target will need to be reduced to facilitate experiments with 100's or even 1000's of targets per day, experiments that are meant for gathering of statistically significant data or stochastic behavior. Much faster target insertion and alignment systems will be needed, compared to those that have previously operated. Our activities towards the goal of experiments with 100's to 1000's of shots will be discussed. They include: 1) mass production techniques for 3-D components, 2) automated (robotic) assembly of complex targets, 3) insertion and injection of targets, 4) target tracking and driver beam engagement systems.

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Integrated Chamber Design for the Laser Inertial Fusion Energy (LIFE) Engine

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The Laser Inertial Fusion Energy (LIFE) concept is being designed to operate as either a pure fusion or hybrid fusion-fission system. A key component of a LIFE engine is the fusion chamber subsystem. The present work details the chamber design for the pure fusion option.

The fusion chamber consists of the first wall and blanket. This integrated system must absorb the fusion energy, produce fusion fuel to replace that burned in previous targets, and enable both target and laser beam transport to the ignition point. The chamber system also must mitigate target emissions, including ions, x-rays and neutrons and reset itself to enable operation at 10-15 Hz. Finally, the chamber must offer a high level of availability, which implies both a reasonable lifetime and the ability to rapidly replace damaged components. An integrated LIFE design that meets all of these requirements is described herein.

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Neutronic Studies on the Impact of Wall Shielding Penetrations for Laser IFE Systems

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Building upon the inertial confinement fusion (ICF) technology developed for the National Ignition Facility (NIF) in Lawrence Livermore National Laboratory (LLNL), a conceptual laser-driven inertial fusion energy (IFE) power plant is being designed. In such a conceptual design, as in any laser fusion system, the final optic for each laser beam is exposed to a variety of threats originating from the fusion target. These include prompt neutron and gamma fluxes, x-ray and ionic emissions. In order to limit the consequences of such threats on the second-to-final optic and the rest of the laser systems, a shielding wall is placed between the target chamber area and the laser bay. An optical telescope arrangement is envisioned that allows for the laser beam propagation from the second-to-final to the final optic, through a pinhole in the shielding wall. These pinholes will be the only openings in the wall easing a direct line of sight between the target emissions, including the high-energy neutrons and gamma rays, and the laser's front-end and, hence, they should be small enough to reduce the leakage of radiation. Furthermore, their shape should accommodate the focused laser on its way throughout the wall. An optimum design of this laser pinhole requires a good understanding of the trade-offs that exist between shielding performance, economic considerations and survivability of the laser optical elements.

This paper aims to give a deeper insight on the impact and influence of the pinholes on the radiation doses at the laser bay, located at the other side of the concrete shielding wall. After addressing the difficulties of evaluating shields containing penetrations, it establishes a guideline for the selection of different variables linked to the pinhole's design, and gives a preliminary evaluation of the radiation fields in the laser bay. The study also helps to identify the requirements to enable manual and/or remote maintenance during operation, by analyzing the minimum achievable background levels for different wall shielding designs.

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Licensing Approach and Initial Safety Studies for the LIFE Power Plant

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The Laser Inertial Fusion Energy (LIFE) facility is a laser-based energy system being developed with the goal to deliver fusion power on a timescale consistent with the needs of the energy marketplace. A key factor affecting the delivery timescale is the need to secure appropriate licensing approvals for construction and operations. Although the US Nuclear Regulatory Commission (NRC) has asserted its jurisdiction over future commercial fusion devices, at the present time there are no regulations that are directly applicable to such facilities. It is expected that the LIFE licensing process will follow a risk-informed, performance-based approach to technical requirements rather than the deterministic approach used for the LWR nuclear reactor fleet. That is, risk insights are used to establish requirements that focus licensee and regulatory attention on design and operational issues commensurate with their importance to public health and safety. Early consultation has led to a working assumption that the licensing of LIFE for commissioning and initial operations could take place under an existing part of Title 10 from the Code of Federal Regulations (i.e., 10CFR70), amended as appropriate in order to address fusion-specific hazards. This approach has been used for other facilities for which performance-based regulation is appropriate.

This paper provides an overview of the licensing approach proposed above, which will require a well coordinated effort from the main stakeholders, including DOE, NRC, policy-makers, utilities, vendors and LIFE designers and engineers. Closely tied to the licensing regime are analyses of the expected safety performance of the plant. Initial results from LIFE safety studies will be presented.

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Conceptual Design of Lead-Bismuth Cooled Accelerator Driven Subcritical Reactor (LEBCAR)

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Accelerator driven sub-critical system (ADS) is an important approach to incinerate the long-lived high-level nuclear waste. Liquid Lead–Bismuth eutectic (LBE) has many unique nuclear, thermophysical and chemical attributes that is potential candidate coolant for ADS subcritical reactors. In addition, lead and bismuth can produce copious spallation neutrons when bombarded with energetic protons. This makes LBE one of the top candidates for a high-power spallation target and coolant in an accelerator-driven subcritical system.

The conceptual design of Lead-Bismuth cooled accelerator driven subcritical reactor for transmutation of long-lived high-level nuclear waste named LEBCAR is developed by FDS Team, China based on the neutronics [1], thermohydraulics [2], and mechanics analysis. In LEBCAR, a linear accelerator produces the proton beam of 1.5 GeV with 10mA and the proton impinges on the windowless LBE target in the core central region. The LEBCAR system is rated at 1000 MWth thermal power with a pool-type fast subcritical reactor. Currently, one of the fuel types considered for LEBCAR is the TRU-Zr dispersion fuel, where TRU-Zr particles are dispersed in Zr matrix. The advanced ferritic/martensitic steel is selected as the target guide pipe and fuel clad materials considered the good performance under a high corrosive and radiation environment.

To obtain the acceptable radial power distribution and reduce the reactivity variation with the high burnup, the fuel blanket region was divided into 3 TRU enrichment zones (low, medium, high). The low and high TRU fuels were loaded in the innermost and outermost zones, respectively. The analysis results indicated that LEBCAR has a good performance for transmutation of minor actinides with 300kg per year and has a good proliferation resistance capability.

1. Z. Chen et al., “Preliminary Neutronics Analysis of Lead-Bismuth Cooled Accelerator Driven Subcritical Reactor for Waste Transmutation (LEBCAR),” *presented at the 15th International Conference on Emerging Nuclear Energy Systems (ICENES-15), May 15-19, 2011, San Francisco, USA.*
2. W. Wang et al., “Structure design and analysis for Lead-Bismuth Cooled Accelerator Driven Subcritical Reactor (LEBCAR)” *presented at the 15th International Conference on Emerging Nuclear Energy Systems (ICENES-15), May 15-19, 2011, San Francisco, USA.*

Preliminary Design Study of Fusion Transmutation Reactor based on ITER Platform

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With optimized long pulse low power operation mode in which Q is less than 5 overall system design parameters for fusion transmutation reactor are studied. Once-through spent nuclear fuel (SNF) is chosen as burn-up material which would be deeply burned by 14MeV fusion fast neutron source. One type of the ITER Test Blanket Modules (TBM) with possible different cooling scheme option is to be adopted as fusion-fission hybrid blanket with added fusion transmutation layer. For a given ITER hybrid operation mode with pulse length of 3,000 sec and given duty cycle the amount of SNF burn-up, the heat generated inside of the fusion transmutation blanket, and the tritium breeding ratio are estimated. Cooling scheme of the fusion transmutation blanket is studied by the standard thermal hydraulic analysis.

High Brightness Gamma Ray Generation for Nuclear Transmutation

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Generation of high brightness gamma ray and its application to transmutation of nuclear waste are discussed. Recent laser and optical technology allow us to stored photon in a cavity. Interaction of electron beam with stored photon makes an enhancement of generation of gamma ray.

We have studied a conceptual design of a total system of the nuclear waste transmutation by this way. The system has feasibility for transmutation of FP and TRU with a long life activity by a sharp-shooting of high brightness gamma ray. We also note the energy balance in this scheme by a fission blanket surrounding the target. Economical nuclear transmutation can be considerable in this.

We performed a small size experiment with the low energy electron beam and stored photon in a super cavity. These results corresponded quite well to the predicted one from cavity storage rate and electron beam energy. Preliminary experiments for nuclear transmutation are underway on 1.5 GeV New SUBARU electron storage ring at University of Hyogo.

The issues of this method are the deceleration with a repulsion of the beam electron by Compton scattering and energy balance. We will discuss these items in this presentation.

The nELBE Neutron Time of Flight Facility

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At the superconducting electron linear accelerator of Helmholtz Zentrum Dresden-Rossendorf the nELBE photoneutron source has been built [1]. The research program is dedicated to the investigation of fast neutron induced reactions of relevance for nuclear transmutation and the development of Gen IV reactor systems. Fast neutrons in the energy range from 0.1 MeV to 10 MeV are produced by the pulsed electron beam impinging on a liquid lead circuit as a radiator. As no moderator is used, the short beam pulses of ~ 10 ps provide the basis for a good time resolution for neutron time-of-flight experiments, giving an energy resolution of about $< 1\%$ at 1 MeV with a short flight path of ~ 6 m. The neutron intensity on target is ca. $2 \cdot 10^5$ n/s using an electron bunch charge of 77 pC and 200 kHz pulse repetition rate. Currently a new superconducting RF injector is being built to increase the usable bunch charge.

The nELBE neutron source produces neutrons in the energy range similar to a fast reactor. It is particularly suited for measuring inelastic neutron scattering or neutron induced fission. The experimental area is separated by a 2.6 m concrete wall from the neutron source. A transmission setup using a plastic scintillator with low detection threshold has been developed. For inelastic neutron scattering a double-time of flight setup using an BaF_2 scintillation detector array and fast plastic scintillators has been developed reaching a time resolution of 700 ps (FWHM). By measuring the time-of-flight of the incoming and the scattered neutron separately, the excitation of different levels in inelastic scattering can be identified. First results on $^{56}\text{Fe}(n,n'\gamma)$ will be presented. nELBE is a consortium member of the EURATOM FP7- support action ERINDA (European Research Infrastructures for Nuclear Data Applications, www.erinda.org) and is supported by the German Federal Ministry for education and science BMBF under contract 02NUK13A www.fzd.de/TRAKULA.

1. J. Klug et al., NIM A 577 (2007) 641

Comparison of the Waste Transmutation Potential of Different Innovative Dedicated Systems and Impact on the Fuel Cycle

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The performances of three different types of innovative transmutation systems have been investigated in order to assess in a comparative way their potential to manage nuclear waste arising in a geographical region, where different countries have different policies with respect to nuclear energy development, but share the objective of a common optimized waste management strategy in order to minimize the waste masses sent to a geological repository.

The three systems are 1) a critical low conversion ratio fast reactor (LCRFR); 2) an accelerator driven system (ADS) and 3) a hybrid fission-fusion system (FFH). In order to simplify the comparison, the three systems have been loaded with comparable fuels, in particular with the same Pu to Minor Actinides (MA) ratio.

The first parameter for comparison is the TRU masses transmuted (normalized to the power produced). This parameter is sensitive to the specific physics features of each transmuter system. Successively, a scenario study has been performed, in order to have indications on the needed infrastructures (transmuter reactors, reprocessing and fuel fabrication installations etc.) in each case. Each scenario study has been performed using comparable hypothesis on, e.g. pace of introduction of new facilities deployment, cooling times etc.

The results show that, apart from the technological readiness of each single option, the performances are rather comparable. In particular, there is a significant impact on important fuel cycle parameters (decay heat of the spent fuel, neutron sources at fuel fabrication, radiotoxicity of the residual wastes etc.) in all cases and this feature represents a major challenge.

Areas for further studies are also pointed out.

