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Subcritical nuclear reactor with Thorium and NG for BNCT

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Nuclear power is a reliable solution to produce heat and electric energy without the emission of greenhouse gases. Generation IV nuclear reactors is a set of reactor designs recommended to improve the nuclear power performance; aiming to increase the security, to reduce the environment impact and to avoid the nuclear weapon's proliferation. In this set, are included the Molten Salt Reactors (Th-MSR). Subcritical nuclear reactors are used to learn the reactor physics, to test materials and systems to measure radiation, to carried out neutron activation, etc. To startup and run these reactors is necessary to use an external neutron source [1-3]. Features of subcritical nuclear reactors depend on the fuel-moderator array, the fuel and moderator type and the external neutron source. To determine the neutron features inside the reactor, and to define radiation protection protocols around the reactor is necessary to determine the neutron spectra [4,5]. The objective of this work was to determine the neutron amplification, the effective multiplication factor and the neutron spectra outside the reactor in a homogeneous Th-salt subcritical reactor (Th-HSR) using 2.45 MeV neutrons from a neutron generator (NG) as neutron source. The aim is to produce neutron spectra suitable for Boron Neutron Capture Therapy (BNCT). Reactor calculations were carried out with the MCNP5 code. Outside the reactor the neutron spectrum varies from 10^{-6} to 20 MeV. Most of the neutrons are between 10^{-2} y 3 MeV, which with proper filters can be modulated for BNCT. The neutron spectra are also feasible for their use for dosimetry phantom studies, material analysis and in-vitro cell irradiation focused on BNCT. A limitation of this work is the need of further research to design the neutron beam shaping system.

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P02

Safety Analysis for Severe Accident in Nuclear Fuel Facilities

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Experimental Fuel Element Installation (EFEI) is one of Indonesia's nuclear fuel facilities. EFEI is consisted of three sub-facilities which are Pilot Plant Conversion (PCP), Fuel Fabrication Laboratory (FFL), and Quality Control (QC) [1]. Some severe nuclear and non-nuclear accidents could happen in EFEI. The examples of nuclear accidents are criticality, radiation exposure, contamination, and the release of airborne radioactivity. While non-nuclear accidents are consisted of explosions caused by H_2 gas, overpressure and fires caused by short circuits, flammable gases or chemicals and the toxicity of other chemicals [2,3,4,5]. The most severe conditions are those that can release significant amounts of radioactive substances into the environment [6]. The main purpose of safety analysis is to control hazards and reduce risks, supporting the goal of nuclear safety to protect workers, society and the environment from radiological hazards caused by the operation of the EFEI. The HIRADC (Hazzard Identification Risk Assessment Determining Control) method was used in accordance with BATAN Standard (SB 006 on 2019) [7]. The basic input required to be analyzed by HIRADC is the system design data and the hazard data from the system. The analysis was carried out in the form of sequence of events analysis, analysis of causality, and the dispersion of radioactive substances. Possible accident scenarios that may occur from system operations in IEBE are dissolution process failure in PCP, explosion and contamination of H2 Reduction Tank in PCP and FFL, UO2 contamination in PCP and FFL, Pelletizing Process failure in FFL, explosion and chemical fire in QC, contamination of waste management system, and VAC system operation failure. Based on the HIRADC method it can be inferred that the severest type of accident, which scored A, is explosion and contamination of H2 Reduction Tank in PCP and FFL which can cause explosions caused by H2 gas, contamination, and fire.

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P03

Prospects of development of nuclear energy in Kazakhstan

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The most popular issue today is the use and consumption of energy in the world. The non-renewability of such resources as oil, gas, coal and the volatility of their prices on the world market are encouraging mankind to seek other sources of energy. For objective reasons, the solution to the problem can be the active development of the nuclear industry.

The purpose of this work is to substantiate the relevance of the development of nuclear power in Kazakhstan, the need for the construction of the NPP, as well as the analysis of risks that may arise during the implementation of the project.

Arguments for the development of the nuclear industry in Kazakhstan:

1. Practically unlimited fuel resources, thanks to the use in the future of a closed fuel cycle. It is this strategic perspective that justifies the development of potentially dangerous technology.
2. Negligible low cost of nuclear fuel transportation [1].
3. Spent nuclear fuel has relatively small volumes and can be «refined» in fast reactors [2].
4. Nuclear power plants are environmentally friendly as they contribute to the reduction of greenhouse gas emissions.

It should be concluded that nuclear power has the necessary characteristics for the gradual replacement of enterprises operating on organic fuels, as well as for becoming the dominant industry in Kazakhstan.

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P04

AN INVESTIGATION OF NEUTRON YIELD IN ADS SPALLATION TARGETS

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Between traditional fission reactors and the promise of nuclear fusion, Hybrid Systems such as Accelerator Driven System (ADS) have been studied as a viable alternative for the safe and efficient production of clean and sustainable energy. Accelerators Driven Systems (ADS) are an innovative type of nuclear system, which is useful for long-lived fission product transmutation and fuel regeneration. The ADS consist of a coupling of a sub-critical nuclear core reactor and a proton beam produced by particle accelerator. These particles are injected into a target for the neutrons production by spallation reactions. The spallation target constitutes the physical and operational interface between the accelerator and the subcritical core and it is the neutron source term for ADS, so it is of great importance to know the number of neutrons emitted by the incident proton (n/p), the energy deposited on the target, the angular, neutron and the of spallation products energy distribution of spallation products [1]. Such quantities will influence the neutron flux that arrives in the fission system and that will directly affect the transmutation [2]. Transmutation requires the minor actinides to be irradiated in a very intense neutron field and with a hardened spectrum such as can be attained in a high-power fission system such as ADS [3]. The proposed system uses a source of protons with energies between 0.3 GeV and 1.6 focusing on different targets [4]: liquid targets as mercury (Hg), natural lead (Pb) and a eutectic lead-bismuth alloy (LBE) and solid targets (depleted uranium (U), tantalum (Ta) and tungsten (W). The dimensions of the spallation target (radius R and length L) are modified. The neutron flux on the surface of the targets is evaluated for different energies of the incident protons and different target geometries considering the neutron escape, absorptions and production by the *(n,xn)* reactions. Simulations were performed using MCNPX and MATLAB and Python programming tools were used for data processing and modelling. The idea is to obtain a high neutron flux with a hardened neutron spectrum for transuranic transmutation in a ADS-fission systems purposes [5].

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P05

Assessment of radiation shielding competence of TeO_2 and Eu_2O_3 incorporated borosilicate glasses

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Interaction of high-energy radiation photons with matter is an important property that is exploited in areas of radiobiology, nuclear and space technology. Borosilicate glasses containing heavy metal oxides (HMOs) such as Bi_2O_3, BaO, and ZnO exhibiting the dual properties of excellent transparency and radiation absorption, can function as radiation shields for scientists and observers working in these fields [1, 2]. Though HMOs contribute to greater density and effective atomic number (Z_{eff}), the combined effect of TeO_2 and Eu_2O_3 in improving the quality of radiation shields must be explored.
In this research, traditional melt-quench technique [3] was utilized to get five glasses of two compositions (50-x)B_2O_3- 17.5SiO_2 -0.5CeO_2- xTeO_2-12Bi_2O_3- 12ZnO- 8BaO and 12B_2O_3-16SiO_2 - yEu_2O_3- (40-y) TeO_2-12Bi_2O_3- 12ZnO- 8BaO with x= 0, 20, 40 mol% and y= 2, 4 mol%, later coded as BiTe-0,20,40 and BiTeEu-2,4. Their densities determined by applying Archimedes theory showed increment of value from 3.9084 g/cm^3 for x=0 to 5.0875 g/cm^3 and 5.4377 g/cm^3 for x= 20 and y= 4 mol%, respectively. Further, experimental values of linear attenuation coefficient (LAC), radiation protection efficiency (RPE), and transmission factor (TF) were procured for these glasses by exposing them to ϒ-rays from three radioisotopes namely, ^{241}Am (0.06 MeV), ^{137}Cs (0.662 MeV), and ^{60}Co (1.173 and 1.33 MeV). The results were then compared and validated with the values computed from Phy-X/PSD software [4]. We found that experimental and theoretical LAC values were close to each other with a deviation of <5%, which proved the reliability of using the theoretical approach for determining shielding parameters of high- energy photons. All the prepared glasses showed the same trend of LAC values with rising photon energy and at each energy, LAC improved with the consecutive increase in TeO_2 and Eu_2O_3 mol%. The estimated RPE and TF concluded that BiTeEu-4 provided maximum protection against radiation compared to all other glasses. PSD software also generated Z_{eff} and half value layer (HVL) values for the energy spectrum of 0.015-15 MeV. Z_{eff} of BiTeEu glasses was found to be greater than BiTe glasses with BiTeEu-4 having the highest Z_{eff} values of 57.19, 25.77, 23.25, and 23.05 at energies 0.06, 0.662, 1.173, and 1.33 MeV respectively. Existence of minimum HVL is a trait of good shielding material and in this case, BiTeEu-4 gave the lowest HVL of 1.458 cm which is even lower than that of ordinary concrete, steel-magnetite concrete [5], RS-323 and RS-360 glasses[6] at 0.662 MeV.

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P06

Evidence of Fission Gas Bubble in High Burn-up Structures in Nuclear Target

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Abstract: The MeV-range energetic ions beyond Coulomb's barrier of a target-beam combination lead to nuclear reactions, subsequently evaporation's leave or submerge the target materials. This is through extreme high temperature referring as high burn up structure(HBS). The Erbium (Er) is used as burnable, neutron absorber with nuclear fuel matrix. Its quite interesting to observe pure Er under reaction environment. In this study, I present the evidence of HBS of

1. Elongated and vermicularinter-granular bubbles in Te.
2. Clustered spherical inter-granular bubbles and fuel cracks in Er and Tm.
3. Carbonaceous deposits on Sn.

via SEM observation. The RBS, EDS and XRD characterization results of these nuclear targets will also be presented.The details of the experiments are in Table 1 with reference's.

Sl No	Target	Beam	Energy (MeV)	Target Type	Thickness	Burnup Details
1	^{130}Te	^{36}Cl	121-155	Carbon backed	400g/cm2	1
2	^{164}Er	^{28}Si	185	Free standing	0.83mg/cm2	2
3	^{169}Tm	^{18}O	94	Free Standing	6.5 mg/cm2	3
4	^{118}Sn	^{7}Li	15-29	Free standing	400g/cm2	4

Table 1: Details of target and beam for burnup condition

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P07

Development of the VVER-1000 alternative power changing mode for operation in the load-following mode

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The share of nuclear power in the most countries' energy mixes is small, thus, there is no need to operate them in the load-following mode (LFM). However, the situation is different in several countries. The share of nuclear power in the national energy mix of some of them had become so important (e.g., France and Ukraine), that the nuclear operators had to implement or improve the load-following capabilities at their NPPs. Before the war, Ukrainian operator completed two stages (different amounts of cycles during each) of the LFM pilot operation at Unit 2 of the Hmelnytska NPP. Currently, this issue is even much important due to power shortage for load-following mode because of constant russian attacks on infrastructure.

One of the key aspects of the LFM implementation is the definition of the most optimal mode for reactor changing power for a common type of the Ukrainian reactors: VVER 1000 MWt type B-320. In the reactors, the power density field and thermal power during LFM operation are controlled by the combined impact of the boron regulation (BR) and control rods (CR).

BR in the reactor is not automatized for the LFM and requires additional time for core interaction to start (transport time). The approach used to deal with boric acid in radioactive waste streams is particularly important and has significant financial, technical and environmental impact [1]. Hence, the mode is not desirable for the LFM industrial operation.

The main objective of the study is the boric acid volume reduction with ensuring acceptable axial and radial power density distribution stability for the LFM industrial operation at Ukrainian NPPs and the alternative power changing mode development.

The proposed mode is an analogue (adaptation) of the "X" mode being operated at the French NPPs for PWR. The mode was developed to control the AO and power simultaneously by using CRs. The control banks are positioned in the overlapping configuration: one group in the lower part and four groups in the upper part of the core. Simultaneous movements of the banks allow efficient controlling of the power and AO. [2]

One of the main issues in the case of the "X" mode adaptation is the difference in the CRs groups type. In PWR opposite to VVER type reactors special developed control banks (called «grey») are operated together with standard «black». The grey banks allow decreasing the deformation of the power density field, which is particularly important for maneuverability of the reactor. [2] All the CRs groups of the VVER 1000 MWt reactors are «black», so it is important to choose or even reconfigure the available groups.

The results of the research provide the investigation of the alternative changing power mode in accordance with the safety requirements in the context of the boric acid volume used to decrease the power level, the axial offset stability and power density field nonuniformity coefficient (Kq) maximum values. The calculation data of the parameters are based on the simulation data obtained by the BIPR-7A calculation program. The program uses the VVER reactors mathematical model. Also, the results provide the most optimal control rods configuration in accordance with the chosen calculation parameters.

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P08

Transmutation Analysis Of Reprocessed Fuel In An Subcritical Fast Reactor With Fusion Neutron Source

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One of the major issues to be solved by nuclear technology is the management of radioactive waste from nuclear power reactors. Currently, an alternative to the closed cycle of nuclear fuel is the use of MOX fuel in LWRs. However, the consumption of transuranics - among them the minor actinides (AM) - in these types of conventional reactors is still low, due to the low availability of fast neutron flux. The study of hybrid systems such as fusion-fission reactors [1] and Accelerator Driven Reactor Systems (ADS) [2] show promise regarding the issue of transuranic transmutation. These systems, there is a production of highly energetic neutrons that can be associated with fast systems, such as a blanket of fissile material around these neutron sources thus allowing the transmutation of the fuel [3]. Therefore, models of hybrid systems developed and under continuous studies at DEN/UFMG [4,5], have a relevant role for the future scenario of nuclear technology. As an example, a semi-heterogeneous subcritical fast reactor with an external source based on ADS system designed by Nifenecker et al. [6] with thermal power of 515 MWt has been modelled using a fusion neutron source as an external source and lead as coolant material. The fusion neutron spectrum was characterized by a Tokamak homogeneous model based on ITER with a Divertor system inserted in the lower region of the plasma chamber. This model was simulated using the MCNP5 code. The reactor designed by Nifenecker et al. [6] was modified to insert reprocessed fuel spiked with thorium and depleted uranium [7] evaluated when the neutron source from the evaporation spectrum of spallation reactions for ADS is substitute by a fusion neutron source. After that, using MCNPX and SERPENT codes, neutronics and transmutation parameters during the burnup has been evaluated. The library used for the fuel transmutation analysis was ENDF/B-VIII.0.

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P09

How Can the Computational Methods Address the Challenges of Modelling and Licensing for Next-Generation SMRs?

According to the carbon reduction politics and Net-Zero targets, Small modular reactors (SMRs) as a component of future integrated energy systems have demonstrated a growing interest worldwide. Despite SMRs' advantages, the developing reactors are possessed several challenges stemming from their various innovative designs. For example, some SMRs don't use water as a coolant; they use Helium, Sodium, Lead-Bismuth, or Molten salt as a reactor coolant. Thus, a proper numerical simulation as a sample challenge is essential for the design and analysis of these reactors. Due to the high computational cost of high-fidelity simulations (e.g., multi-physics, turbulence models in CFD, neutron transport, etc.), multi-query applications can often be prohibitively expensive where many repeated evaluations are needed like uncertainty quantification, design optimization, and control. Reduced-Order Model (ROM) as a mathematical technique is a solution for computational challenges. This approach is like a getaway option that seeks to alleviate numerical modeling challenges for new generation SMRs by employing a surrogate model instead of the full model analysis. ROM leverages a computed set of basis functions in order to significantly reduce the number of degrees of freedom in the system. As an important application, the method called the Best Estimate Plus Uncertainty (BEPU) approach is proposed for licensing issues dealing with uncertainty quantification. According to the costly computational process in the uncertainty quantification, ROM can be the best solution (or just solution) to address the BEPU implementation challenge for analyzing new generation SMRs. Although ROM consists of several intrusive and non-intrusive methods and techniques, few studies have been conducted in the nuclear-related fields. Hence, ROM is a new concept that can meet the challenges of modeling and licensing the new generation SMRs in the future. It should be noted the researchers made an attempt to implement the ROM in several case studies related to Sodium Fast Reactors (SFRs), Molten Salt Reactors (MSRs), and Lead Fast Reactors (LFRs) during the last years, but an effort is needed to achieve a comprehensive approach and framework and to accelerate the deployment process.

Supercritical Water Cooled Nuclear Reactor

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Supercritical Water Cooled Nuclear Reactor (SCWR) accepted as one of the prospective Generation IV reactors in the proposed Generation IV Forum (GIF). Under this program, SCWR conceptual proposals are being developed by more than 45 organizations in 16 countries with developed nuclear power. The idea of SCWR is based on the implementation of a once-through single-loop scheme of a nuclear power plant cooled by supercritical pressure water. The introduction of a nuclear power plant of this type will increase the efficiency up to 45%, increase the fuel breeding ratio, reduce metal consumption and construction volumes, and increase the degree and environmental performance. At the development stage, if problems arise with the storage of spent fuel and minor actinides, a transition to a fast neutron spectrum reactor, MOX fuel and a closed fuel cycle is possible.
Within the framework of the MFP-4, various variants of SCWR are being developed, differing in the parameters of the coolant and the schemes of its circulation in the cores. Conceptual designs of reactors have been developed, in which the possibility of their economic efficiency by 20-40% in comparison with Generation 3+ reactors has been found.

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P11

NAA study for major and minor elements determined in the soils and phosphate rocks of the prospective phosphate mining area in the Hinda district, Republic of Congo.

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Abstract

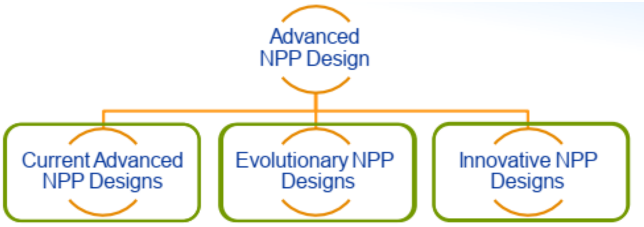
The concentration of major and minor elements was determined using the k0-method of NAA through the Moroccan Triga Mark II research reactor. The results obtained have been compared with the international values of these elements. The environmental pollution indexes were determined. Base of that, different possibility pathways of contamination have been described. Pearson coefficients were also determined in order to understand relationships between elements. The acquired data will serve as a baseline data for the follow-up studies to assess environmental contamination and the agronomic interest of the phosphate rocks of the prospective phosphate mining area in Hinda.

Keywords: major and minor elements, phosphate rock, soil, NAA, k0-IAEA, Republic of Congo.

P12

"Innovative Reactor Concepts and Fuel Cycle Options"

Advanced reactors are the design of current interest with substantial improvements over previous/existing designs to achieve a goal reduced cost and improved safety.



Evolutionary Design:

Advanced designs that achieve improvements over existing designs thorough small modifications (maintaining proven designs, (Generation III/III+))

Innovative Design:

Advanced designs with conceptual changes in design approaches or configuration comparing with existing practices (Generation IV)

Innovative Design: Safe, Secure, Sustainable, Competitive, Versatile

- GFR gas-cooled fast reactor
- LFR lead-cooled fast reactor
- MSR molten salt reactor
- SFR sodium-cooled fast reactor
- SCWR supercritical water-cooled reactor
- VHTR very high temperature reactor

GFR: Advantages (Potential for new fissile breeding due to fast neutron spectrum, transparent and inert coolant, high efficiency) **Challenges** (Safety demonstration and in particular decay heat removal in case of loss of flow and depressurization accidents, high-temperature materials and fuel qualification)

LFR: Advantages (Potential for new fissile breeding due to fast neutron spectrum, high density, high thermal inertia, high thermal conductivity and expansion coefficient, efficient heat removal at low velocities and high natural circulation level, passive with water and air → no intermediate circuit, large margin to boiling (1740° C) → no pressurization required **Challenges** (High density → erosion, seismic refueling issues, at high temperature structural materials (such as iron or nickel) are slowly dissolving in lead flow → protection needed, high void reactivity effect (e.g. gas entry), low margin to freezing (327ºC) → special safety measures need)

MSR: Advantages (Potential for new fissile breeding due to fast neutron spectrum, large margin to boiling → no pressurization required, strongly negative fuel salt density (void) reactivity effect, high efficiency due to high temperatures, no structural materials → no radiation damages, possibility to add or remove fuel salt and simpler reprocessing, continuous removal of insoluble fission products. **Challenges** (Strong corrosiveness of molten salt fuels, lack of usual barriers (fuel cladding) → new safety approach needed, high fluence on vessel, Part of fuel always outside core → larger fuel inventory needed; reduced β, low margin to freezing, low or unknown solubility of compounds formed during operation)

SFR: Advantages (Potential for new fissile breeding due to fast neutron spectrum, excellent thermal conductivity of sodium, very efficient cooling, large margin to boiling → no pressurization required, Significant operational experience (300+ reactor-years) **Challenges** (Chemically active in contact with water or air → intermediate circuit needed, significant scattering cross section, spectrum hardening when removed, positive reactivity effect → special safety measures needed)

SCWR: Advantages (Based on Gen-III+ reactor technology, merges it with advanced SCW technology used in coal plants, higher efficiency than Gen-III+, both thermal and fast spectrum possible) **Challenges** (Materials, water chemistry, and radiolysis, thermal hydraulics to fill gaps in SCW heat transfer and critical flow databases, safety demonstration (positive void effect for fast spectrum option), fuel qualification)

VHTR: Advantages (High temperature enables non-electric applications, "Walk-away" safe, inert gas coolant, high efficiency). **Challenges** (Reach temperature of ~1000ºC (for hydrogen production), coupling with process heat applications, graphite as a waste)

Various vital issues associated with nuclear power are coupled primarily to the choice of fuel cycle.

Nuclear Fuel Cycle (NFC):

- Procurement, preparation, utilization and ultimate disposition of the fuel from the reactor.
- Resource limitations, non-proliferation, and waste management are primarily fuel cycle issues.
- Selection of fuel cycles and reactors depends upon the requirements for sustainability, safety, and economics.

Current fuel cycles:

i. *Open cycle*: Spent Nuclear Fuel (SNF) is removed from a reactor and stored; current approach in Canada, Finland, Germany, Sweden and the USA.

ii. *Closed or partially closed cycle*: Actinide material is recovered from SNF in a reprocessing facility and fabricated into new fuel.

Innovative fuel cycles

i. ***Full fissile material (Pu,^{233}U) recycling***: All SNF is reprocessed for recovery and recycling of U, Pu and/or ^{233}U, Minor actinides and fission products are sent to the waste stream, preserves fissile material resources.

ii. ***Actinide and long-lived fission product recycling***: All SNF is processed and actinides are recycled multiple times to fully consume the fissionable material and to transmute the minor actinides, allows the preservation of resources, the minimization of the radiotoxicity of the final waste, and the optimization of repository space, which are the main criteria of innovative fuel cycles.

Partitioning &Transmutation (P&T): Potential route in the management of SNF, new avenues for long term waste management by eliminating long term radionuclides and their thermal effects and thus reducing the necessity or capacities of disposal facilities.

Major scenarios to implement P&T:

a) Waste minimization (Homogenous TRU recycling)
b) Reduction of MA inventory
c) Reduction of TRU inventory as unloaded from LWRs

Potential benefits of P&T:

- Reduction of the potential source of radiotoxicity.
- Reduction of the heat load and high-level waste volume
- Improved proliferation resistance is expected if TRU are not separated.

Table 1: The basic technologies and applicable fuel cycles

Reactor type	Neutron spectrum	Fuel cycle	Reprocessing
SFR	Fast	Closed	Aqueous, Pyro
LFR	Fast	Closed	Aqueous, Pyro
GFR	Fast	Closed	Aqueous, Pyro
MSR	Fast	Closed	Pyro
MSR	Thermal	Closed	Pyro
VHTR	Thermal	Open	-----
SCWR	Thermal	Open	-----
AHWR[1]	Thermal	Closed	Aqueous
ADS[2]	Fast	Closed (Double strata)[3]	Aqueous, Pyro

1: Advanced heavy water reactor; 2: Accelerator driven systems Accelerator driven systems (ADSs) are innovative systems designed primarily for the final burning of plutonium and minor actinides at the end of thermal and fast reactor operations; 3. Double strata: Two separate fuel cycles: i) commercial reactors with Pu utilization ii) separate MA management.

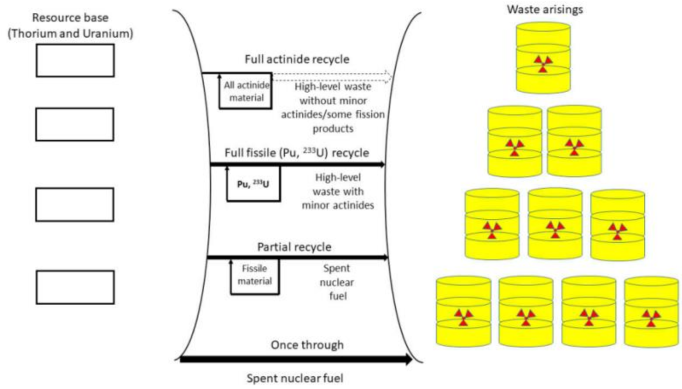


Fig.1.Total waste arising from alternative nuclear fuel cycles

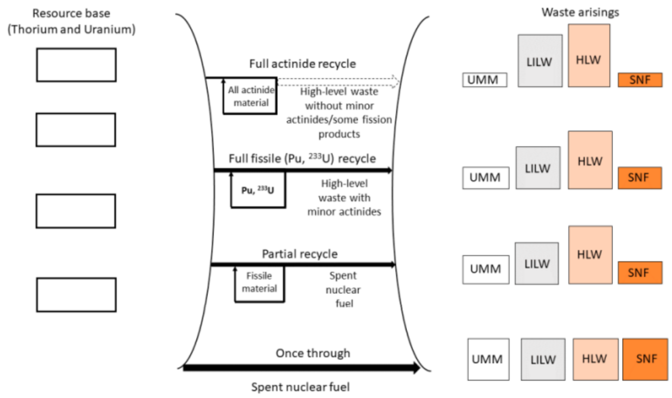


Fig.2. Relative waste arising from alternative nuclear fuel cycles

Acronyms

TRU = Transuranic; MA = Minor Actinide, LLW = Low Level Waste, HLW = High Level Waste, UMM = Uranium Mining and Milling; LWR = Light Water Reactor

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SAFARI-1 FUEL CYCLE

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SAFARI-1 was commissioned in March 1965. It is an MTR type tank in a pool research reactor, normally operation at full power of 20 MW. The operational cycles are usually 3-4 weeks long and follows a scheduled shutdown for five days to perform necessary maintenance and reloading of the core. The reactor core layout consists of 9 by 8 grid which houses 26 fuel assemblies, 6 control rod (which are fuel follower type) assemblies, 7 solid aluminium elements, 11 solid beryllium elements, 8 plugged beryllium elements, 4 lead reflectors, 9 aluminium water box and one hydraulic rabbit (which consists of two positions).
Historically SAFARI-1 was operating using highly enriched uranium (HEU) plate-type fuel which was produced on-site. In 2009, SAFARI-1 completed its conversion from HEU to low enriched uranium (LEU) and is currently fully operational on LEU plate-type fuel.

SAFARI-1 LEU fuel element has 19 fuel plates which consist of uranium silicide, the control rod also has 19 fuel plates with cadmium as a neutron absorber. During each scheduled maintenance shutdown, a fuel burn-up is performed in order to determine which fuel and or control rods needs to be replaced for a new core cycle. Typically about 4 new fuel elements are added to the new core for a 4 weeks cycle run. In case control rods needs to be replaced, usually 2 are replaced at a time and only 2 new fuel element will be added for that 4 weeks cycle. The fuel elements that were replaced during a scheduled shutdown can be reused again in an operational core cycle if they were not fully burnt. With control rods, once they are replaced, they cannot be reused again and are declared spent fuel.

The spent fuel at SAFARI-1 is temporarily stored in the reactor pool for a short period until it is transferred to an underground dry storage facility within Necsa grounds. This transfer is usually done once a year to create more space for more spent fuel in the reactor.

[1] Ramatlhatse,PC (2022).

P14

Preliminary neutronic study for VHTR with several fuel designs

Preliminary neutronic study for VHTR with several fuel designs Khukhsuvd Batsaikhan1, Odmaa Sambuu 1,2 1Nuclear Research center, National University of Mongolia, Peace avenue 122/1, Bayanzurkh District, Ulaanbaatar 13330, Mongolia 2 Department of Chemical and Biological Engineering, School of engineering and Applied Sciences, National University of Mongolia, Ikh surguuliin surguuliin gudamj 3, Sukhbaatar District, Ulaanbaatar 14201, Mongolia The purpose of the present work is to determine the neutronic characteristics of annular cylindrical prismatic Very High Temperature Gas-cooled Reactor (VHTR) cores with different fuel designs and to compare their features. The study includes the detailed neutronic investigations of annular VHTR cores with eight different types of TRISO (contained SiC layer) and TRIZO (contained ZrC layer) fuel particles. All calculations were conducted using continuous Energy Monte Carlo Code MVP2.0 and MVPBURN with the JENDL4.0 nuclear data library. According to the obtained results, as for annular cylindrical prismatic core designs of which were fueled with

P15

A proposal of a hybrid fusion-fission system based on ARC

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Studies are being carried out looking for technologies for a compact, robust and affordable reactor that aims to reduce the size, cost and complexity of a fusion reactor installation. One promising fusion reactor underway at the Massachusetts Institute of Technology (MIT) is the Affordable Robust Compact Reactor (ARC) [1]. This reactor uses high temperature superconductors (HTS), which allow the generation of large magnetic fields on the shaft and therefore reduce its size. The component irradiation tests, in the ARC reactor, are performed using the same deuterium-tritium (DT) fusion reaction of the ITER reactor, whose neutron energy produced is 14.1 MeV. Researches developed at the Department of Nuclear Engineering (DEN/UFMG), investigated the insertion of the transmutation layer in the fusion reactor based on ITER, to constitute a hybrid fusion-fission system [2],[3]. The transmutation layer inserted into the reactor allows operation in subcritical mode and induces transuranic transmutation. This transmutation uses the hardened flux of neutrons originated from the fusion reaction, through fission reactions [4]. Based on this idea, this work aims to carry out a study of the effect of inserting a transmutation layer in the ARC reactor, and its implications on the neutron flux and, consequently, on other related quantities, for the region of the transmutation layer. It is expected that as the ARC reactor is smaller than the ITER reactor, the neutron flux in the hybrid system based on the ARC reactor will be higher, due to the lower scattering and leakage in this system. For this, simulations will be carried out in different models, using the code based on the Monte Carlo method (MCNP) [5], both to investigate the best system configuration for insertion in the transmutation layer, and to characterize these models.

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P16

Characteristic of Loss of Forced Cooling Accident in High Temperature Reactors

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The High Temperature Reactors [1] as Generation IV reactors are considered as a heat and power production sources for the industrial applications. The Probabilistic Safety Analysis (PSA) [2] is aimed to identify of initiating events during accident scenarios. The Loss of Forced Cooling (LOFC) accident is one of the severest accidents in all reactors. In LWRs, LOFC accident leads to the core melt as a result of exceeding the temperature from limited criteria. However, this event can lead to release of the fission products into the environment in HTRs. The present research focuses on characteristic of LOFC accident in HTRs in point of view of the accident progression. The preliminary study on PSA of HTRs will be presented within in this work.

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P17

Recirculating Flow Analyses in a Lead-Cooled Small Modular Reactor

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Computational Fluid Dynamics (CFD) have been used to advance studies on thermal-hydraulics of advanced nuclear reactors, especially those cooled by liquid metal. Thus, in order to demonstrate the capabilities of using CFD codes in the Department of Nuclear Engineering at the Federal University of Minas Gerais, this work presents an analysis of the flow patterns in a lead-cooled reactor under normal steady state operating conditions by the means of 3D CFD simulations. To carry out the studies, a Small Modular Reactor (SMR) was chosen mainly due to the available reference data in the literature. The SEALER (Swedish Advanced Lead Reactor) [1] is one of the most relevant liquid-metal cooled reactor proposals, for its small build size and very low power (8 MW_{th} - 3 MW_e), it is suited for deployment in remote regions with poor electric grid infrastructure. Because of SEALER's pool-type design, thorough investigations on natural circulation and convective flow patterns must take place, especially in order to identify regions of coolant stagnation and recirculation. For such, the CFD model was based on published data from [1, 2], including geometry and operating conditions. The core region was divided in all of the fuel, control, reflector and shielding assemblies, each modeled as an individual porous region, using the Darcy-Forchheimer model [3], according to their respective pressure drops estimations. The axial and radial thermal power distributions were taken into account and applied as volumetric heat sources in the fuel assemblies.
For the sake of computational resources economy, the domain was reduced to a symmetric $1/4$ of the whole. Thus, symmetry boundary conditions were applied to side faces and free fluid surfaces were modeled as slip walls. The liquid lead coolant was modeled with temperature-dependent thermophysical properties [4]. The case was run on ANSYS Fluent. The results, though representative of a steady state condition, were based on transient calculations, due to the natural convection and geometry implications in computation stability. Although necessary and more expensive, the transient approach can capture possible cyclic flow behavior and improve the quality of the analysis.
In conclusion, by employing CFD in this study, detailed natural convection patterns can be analysed, which would not be possible if using traditional thermal-hydraulic system codes. With the velocity and thermal fields mapped, recirculating flow regions and excessively hot or cold spots can be identified and actions against possible accident-inducing conditions, such as coolant freezing, can be taken [5] and further safety design modifications in geometry and operating conditions can help optimize the coolant flow and heat transfer within the reactor.

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