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P01

Assessing Small Modular Reactors for Non-Electric Applications in Kuwait: A Preliminary Reactor Technology Assessment Study

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Small Modular Reactors (SMRs) represent a significant advancement in nuclear energy, offering innovative solutions to contemporary energy challenges. The operational versatility of SMRs is a key focus, accommodating diverse applications such as electricity generation, district heating, desalination, and hydrogen production. This adaptability positions SMRs as vital players in transitioning to low-carbon energy systems.

Furthermore, this study, particularly relevant in regions like the GCC exploring nuclear energy options, aims to identify the most suitable SMRs based on reactor type and operational temperature for electric and non-electric applications. Despite Kuwait's absence of an established nuclear power program, the investigation into the two candidate SMRs reveals their potential to address the country's specific energy needs and sustainability goals.

A preliminary Reactor Technology Assessment (RTA) reveals that light water SMRs are particularly suited for lower-temperature applications, mainly desalination. At the same time, high-temperature gas-cooled reactors within the SMR framework show promise for high-temperature industrial processes, such as the oil industry and hydrogen production. This study examined two SMRs: NuScale (USA) and HTR-PM (China).

The study utilized the IAEA's toolkit for RTA, involving several stages to evaluate the suitability of SMRs for Kuwait's proposed nuclear power program. Initially, the assessment focused on Kuwait's nuclear energy policies, regulations, and public perception of nuclear energy and SMR technology. Due to the absence of extensive public acceptance studies in Kuwait, a preliminary qualitative survey was conducted through expert interviews. The second stage involved evaluating key criteria, including site and environment, nuclear safety, nuclear island design and performance, balance of plant, design for non-electric applications, safeguards, protection, and technology readiness. Comprehensive data on NuScale and HTR-PM were gathered and assessed using a rating scale, with scores weighted based on their importance to determine the most suitable SMR technology for Kuwait.

The final scores for the SMRs considered were 3.50 for HTR-PM and 3.48 for NuScale. The RTA indicated that the HTR-PM reactor is the most suitable for Kuwait's nuclear plan due to its high thermal efficiency, which benefits the oil and gas industry. However, if the focus were solely on electricity output and desalination, NuScale would be more suitable.

In conclusion, SMRs have emerged as a key technology in the sustainable energy landscape. Their scalability, safety features, and efficiency make them an attractive option for countries considering nuclear energy, positioning them as a strategic solution to global energy challenges. The ongoing development and implementation of SMR technology will be crucial in achieving a more efficient, safe, and sustainable energy future, particularly for nations like Kuwait exploring nuclear energy's potential.

P02

An overview of the requirements to new-generation nuclear reactors for the production of medical isotopes and for BNCT

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The production of radioisotopes for diagnostic and therapeutic medical uses and for the supply of medical radionuclide generators is a traditional field of application of nuclear reactors. Furthermore, reactors can constitute a valid source of neutron fields to be used in experimental, clinical and pre-clinical lines of Boron Neutron Capture Therapy (BNCT).

In recent years, the demand for the supply of medical nuclides by hospitals has grown, both in terms of quantity and types of radioisotopes, due to the ever-increasing theranostic applications. On the other hand, the supply has undergone periods of shortage, due to the scheduled decommissioning of some reactors and the prolongation of maintenance operations on others.

The stability of the international radiopharmaceutical market is therefore critically linked to the capacity to produce radioisotopes with the various methods, of which that by neutron bombardment from reactors is the main one.

In this work, the main production pathways, as fission products or through activation reactions, of nuclides of medical interest (^{131}I, ^{90}Y, ^{177}Lu…) or of their progenitors (^{99}Mo, ^{90}Sr, ^{188}W…) are summarized. [1-4]

Regarding BNCT, the IAEA guidelines [5] indicate the following Figure Of Merit (FOM) as requirements for neutron beams: minimum epithermal neutron flux > 10 $cm^{-2}s^{-1}$; minimum degree of collimation as neutron current over epithermal neutron flux > 0.7; maximum gamma dose rate epithermal neutron flux < $2 \cdot 10^{-13}$ (cm^2Gy); maximum thermal neutron and epithermal neutron flux ratio < 0.05; maximum fast neutron dose rate over epithermal neutron flux < $2 \cdot 10^{-13}$(cm^2Gy). To treat a patient in a reasonable time (max 1 h), with a feasible level of boron concentration of about 30 ppm in the lesion, a thermal neutron fluence on the order on 10^{12} cm^{-2} is necessary, meaning an epithermal neutron flux at the entrance of the patient on the order of 10^9 $cm^{-2} \cdot s^{-1}$. When designing new reactors that are expected to be used in the application fields described in this work, it is necessary to take this information into due consideration, in order to be able to predict production yields and optimize purification processes, aimed at reaching the standards required for clinical use.

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P03

The Challenge of Implementing a Geological Final Disposal System in Argentina

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Abstract

There are around 423 nuclear power reactors currently in operation around the world and 200 nuclear reactors are expected to begin the decommissioning process by 2050. This means almost half and the challenge of successfully decommissioning nuclear facilities is set grow.

This article discusses the role of the National Atomic Energy Commission of Republic Argentina (CNEA), in the implementing of geological Final Disposal System called “ConfinaAR Geo”. The problem of waste management has gained relevance with development of electronuclear production and increased social awareness on the need to protect environment. The project responds to two of the unavoidable responsibilities of CNEA which are addressed by the National Radioactive Waste Management Program (PNGRR). One of them is the ethical responsibility linked to the sustainability of nuclear fuel cycle and the principle of not transferring our obligations and decision to future generations. The other is the legal responsibility of the CNEA established by National Law n° 25.018. A network of international, public and private, support the experiences exchanges, the normative harmonization and management searching the excellence and best practices.

Key words: Nuclear energy, nuclear safety, international cooperation, international organisations, radioactive waste, waste management.

Thermal-Hydraulic Transient Analysis of Dedicated Depressurization System for Gen-III PWR in Station Blackout

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Station Blackout (SBO) is one of the most severe accidents in nuclear power plants (NPPs), which poses significant risks to both reactor safety and environmental health. This study focuses on the thermal-hydraulic transient analysis of the Dedicated Depressurization System (DDS) for a Generation-III Pressurized Water Reactor (PWR) in an SBO situation, using the ACP1000 reactor at Karachi as a model. The motivation for this research stems from the catastrophic Fukushima accident, highlighting the necessity for passive safety features in modern NPPs [1].

The primary objective is to develop and validate a transient model for the ACP1000 reactor using the MELCOR thermal-hydraulic code, which simulates the severe accident processes and helps us evaluate the effectiveness of the DDS. The DDS is designed to mitigate severe accident consequences by depressurizing the Reactor Pressure Vessel (RPV), thus preventing high-pressure melt ejection (HPME) and protecting containment integrity [2].

Methodology involves creating a comprehensive nodalization input of the reactor and its primary and secondary loops, then simulating SBO conditions both with and without the activation of the DDS. The resulting transients are analysed according to various figures-of-merit (FOMs) such as core uncover time, maximum core temperatures, system water levels, RPV failure time, in-vessel and ex-vessel hydrogen generation, and pressure inside RPV at time of failure. The analysis indicates significant differences between the scenarios. Without DDS activation, core meltdown and RPV failure occur at high pressure, leading to potential HPME and direct containment heating (DCH). Conversely, with DDS activation, the RPV pressure is reduced below 2 MPa, significantly delaying RPV failure and preventing high temperatures in containment.

The findings underscore the critical role of DDS in enhancing the safety of Gen-III reactors during severe accidents. By rapidly depressurizing the RCS, the DDS minimizes the risk of DCH, thereby safeguarding the containment structure and reducing environmental contamination risks. This research contributes to the ongoing efforts to improve nuclear reactor safety systems, providing a validated model for future applications in different accident scenarios and supporting the development of accident management programmes in NPPs.

In conclusion, the study highlights the effectiveness of DDS in mitigating severe accident impacts, reinforcing the importance of integrating advanced safety systems in modern nuclear reactors to ensure their safe and sustainable operation.

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P05

A Proposal for a Thermal Nuclear Propulsion Engine Using Low-Enriched Cermet-Based Fuel for Spaceships and Generating Electricity using Seeback effect

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Spaceships consume a lot of fuel when launched to space, either during take-off or during their journey. Using thermal nuclear engines instead of chemical engines is better, as nuclear engines would have nearly twice the efficiency of the best chemical engines [1].

Thermal Nuclear Propulsion advantage is that it will use the heat coming out from the fission process to heat up the propellent (H_2) to higher temperatures, causing it to expand. The fuel type to be selected is cermet, as it yields a high melting point of 3695 K [1].

The hydrogen here has two roles at once, which is to act as a propellent and a moderator simultaneously. Here it’s a little difficult to use hydrogen for a third role which is to generate electricity using the usual thermal cycle, so my idea was to use the heat inside the reactor and the cold temperature of space in making use of the seeback effect like what RTGs do, by that we generate sufficient electricity to power the spaceship due to the big difference in temperatures between space and the inside of the reactor, I proposed this idea to my professors as to be my graduation project, to design and model this using simulation codes available.

Simulation codes selected were Openmc for neutronic analysis and moose for thermal hydraulic analysis and also by adding a semiconductor module to moose, we could simulate the seeback effect on it, and then both outputs from both programs can be coupled using Cardinal open source code.

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P06

Radiation Detection and measurement of CT scanner and different environmental sample

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Radiation detection and measurement are critical components in various fields, including nuclear engineering, medical diagnostics, environmental monitoring, and homeland security. The importance of precise measurements in ensuring both effective imaging and safety in environmental assessments [1]. Computed tomography (CT) is a highly effective tool used by radiologists to detect illness in the human body. It was introduced in the early 1970s and was the first computer based medical imaging modality [2]. This research work was based on studies of radiation dose delivered by 16,64, and 128 slice CT scanner from data collected in S.chara hospital, Trento, Italy and CT scanner facilities in Addis Ababa, Ethiopia using phantom (anthropomorphic phantom) . In addition to assess the concentration and life time cancer risk of radionuclides on different environmental sample in Addis Ababa city, Ethiopia using Gamma-ray spectrometer. These radionuclides are the primary source of natural radiation in the environment. Overall, the results show that the radiation dose supplied by 16,64, and 128 slice CT scanner were in a good agreement with the international dose reference level and we observe something difference. So, we were ensuring a patient is not exposed to excessive radiation dose during CT examination while maintaining CT image quality. On the other hand the result implies that the concentration of radium, thorium, and uranium in the environmental sample were within acceptable limits set by regulatory guidelines and the calculated lifetime cancer risk associated with exposure to these radionuclide were lower within a risk of developing cancer . Ultimately, a radiation protection management strategy should be adopted to prevent the radiological negative effects of the environmental sample, and so this research works can be used as a baseline for future related research project.

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P07

Preliminary sustainability assessment of infrastructure for potential future deployment of the SMR CMSR technology in Vietnam

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Small modular reactors (SMRs) are being considered as promising energy sources for supporting the decarbonization target of Vietnam by 2050 [1]. Among the SMR technologies, the Compact Molten Salt Reactor (CMSR) is being considered and investigated as one of potential SMR candidates for future deployment in Vietnam. Namely, a preliminary study on the feasibility of the CMSR power barge deployment in Vietnam was conducted by the PECC2 (Power Engineering Consulting Joint Stock Company 2), Seaborg and Siemens Energy [2]. Subsequently, a nuclear energy sustainability assessment (NESA) for the CMSR technology is being performed jointly between PECC2, VINATOM and Seaborg with the support and coordination of the IAEA's INPRO (International Project on Innovative Nuclear Reactors and Fuel Cycles) Section. The assessment areas to be examined in this NESA work include economics, environmental impact, waste management, proliferation resistance, safety and infrastructure.
This poster is therefore aimed to present the preliminary results of sustainability assessment of infrastructure for potential future CMSR deployment in Vietnam based on the IAEA's Milestones Approach [3] and INPRO methodology [4]. In this assessment, infrastructure considerations for the CMSR cover various aspects related to its unique features, to name a few: the CMSR design and its molten salt fuel; the maritime deployment models of CMSR; and the maturity readiness level of molten salt chemistry and technology, e.g., the corrosion and safety related issues as well as the back end of the fuel cycle. The ultimate goal is to identify the infrastructure benefits and challenges when deploying the CMSR technology, thereby proposing relevant solutions to facilitate its possible future deployment in Vietnam.

Keywords: molten salt reactor, SMR, CMSR, NESA, infrastructure assessment

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P08

Abstract template for … Enhancing Transmutation Rates of Minor Actinides and Long-Lived Fission Products Using Zr and Y Hydride and Deuteride Coatings in an LFR …

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The possibility of transmutation of minor actinides (MAs) and long-lived fission products (LLFPs) in the core of a lead-cooled fast reactor (LFR) offers a promising avenue to reduce the overall repository of radioactive waste. Several transmutation schemes have been proposed and analyzed to enhance the efficiency of transmutation. The present study investigates the feasibility of utilizing moderator coatings around the fuel elements or targets as a means to augment the transmutation rates. This is achieved through the softening of the neutron energy spectrum, which results in increased capture by the selected radionuclides intended for transmutation. Two reference fuel assemblies based on the design concept of ALFRED (Advanced Lead-Cooled Fast Reactor European Demonstrator) were considered: one for the transmutation of LLFPs, in which 12 of the 127 fuel elements in the assembly were replaced by target rods containing a homogeneous mixture of six LLFPs (^{79}Se, ^{93}Zr, ^{135}Cs, ^{107}Pd, ^{99}Tc, and ^{129}I), and the other for the destruction of MAs, in which six MAs (^{237}Np, ^{241}Am, ^{243}Am, ^{243}Cm, ^{244}Cm, and ^{245}Cm) were loaded homogeneously with the fuel in a 95-5 wt.% combination. Four different moderators, namely $ZrH_{1.6}$, $ZrD_{1.6}$, YH_2, and YD_2, with five different thicknesses of 0.01 cm, 0.02 cm, 0.03 cm, 0.04 cm, and 0.05 cm were selected for this study and variation in infinite multiplication factor (k_{inf}), transmutation rates (TRs) and fuel cycle parameters (employing the linear reactivity model) were determined. The results revealed that both the addition of a moderator and an increase in moderator thickness led to improved transmutation rates, but at the expense of fuel cycle parameters. Among the moderators, $ZrH_{1.6}$ resulted in the greatest increase in transmutation rate, but also incurred the highest penalty in cycle burnup, discharge burnup, and cycle length. For $ZrH_{1.6}$ coatings of 0.01 cm, the obtained TRs were 13.57%/y for ^{237}Np, 12.6%/y for ^{241}Am, 9.26%/y for ^{243}Am, 1.78%/y for ^{79}Se, 0.4%/y for ^{93}Zr, 4.23%/y for ^{99}Tc, 5.02%/y for ^{107}Pd, 2.07%/y for ^{129}I, and 0.95%/y for ^{135}Cs. The use of deuteride moderators minimized the decrease in cycle parameters, but compromised transmutation rate. The fuel temperature coefficient (FTC), energy dependent neutron flux, and relative pin power distribution of the reference design and the models were also analyzed. Although the use of moderators did not significantly affect the distribution of power, it did thermalize the neutron spectrum as expected. Finally, a suggestion of the best model was put forth in consideration of both transmutation efficiency and cycle parameters.

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P09

Design and thermal-hydraulic analysis of a microchannel heat exchanger in innovative nuclear reactor application

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The goals set by Generation IV International Forum (GIF) about future nuclear development have inspired the scientific community, leading to the development of different innovative design.
The next nuclear power generation should be economically affordable, sustainable, proliferation resistant and safe. In these years, a strong interest into small modular reactor has grown because of the potential series production and associated economic advantages [1]. Efforts have been spent for economics and safety, leading to the development of integrated reactors leveraging on passive safety systems.
Sustainability and proliferation resistant requirements deflected scientific community attention to different fuels and primary coolants characteristic of advanced modular reactors (AMR). The design selected by GIF are very high-temperature gas-cooled reactor (VHTR), sodium-cooled fast reactor (SFR), gas-cooled fast reactor (GFR), lead-cooled fast reactor (LFR), molten salt reactor (MSR) and super-critical water-cooled reactor (SCWR) [2].
Compatibility challenges between unconventional cooling fluids and most materials for structures, systems and components require research, development, and qualification of existing and new innovative materials, resulting in an increasing cost. In this context, the factor size acquires much more value; thus, the relevance of studying compact heat exchanger that match with new coolants adopted and enhance the performances [3].
Thanks to a high heat transfer area-to-volume ratio and good performance, microchannel heat exchangers can be an attractive compact option for the steam generator and safety-related heat removal system. In addition, the likelihood and the consequences of a channel rupture may be reduced compared to other heat exchanger solutions, with advantages in terms of safety and maintenance costs [4]. Usually employed in industrial applications, the effectiveness of this kind of heat exchangers in advanced nuclear applications is yet to be fully assessed.
This paper aims to perform a preliminary performance evaluation of compact heat exchangers of the microchannel type in transferring heat from the primary to the secondary coolant system of a reactor. Particularly, the selected reactor design belongs to LFR category, foreseeing lead and water in supercritical conditions as primary and secondary coolant, respectively. In literature, the possibility of adopting this technology has been deeply investigated in the hypothesis of using lead and s-CO2 as process fluids.
The heat exchanger has been sized using a Phyton code and the results have been compared through RELAP5 system code. The results are compared between the computational methods highlighting differences which may require further R&D development.

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P10

A package for systematic sensitivity and uncertainty analysis with respect to nuclear data

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Sensitivity and uncertainty analysis has become an important part to justify design of the new generation reactors, known as Generation IV. The main limiting factor in setting appropriate safety margins in their design is nuclear data, which are the chief source of uncertainties. Assessing the nuclear data-induced uncertainty requires covariances in the form of covariance matrices coming from ENDF-6 files [1]. Usually, covariance data are not provided with neutron transport codes and have to be processed from ENDF-6 files by the user limiting the analysis.
To circumvent the limitation, a package has been developed with Python API. It allows not only processing them with appropriately set NJOY parameters but also interacting with the data, fixing mathematically incorrect data such as non-positive semi-definite covariance matrices and correlations with absolute values of over one, and exporting the matrices in a suitable format.
Besides working with the conventional uncertainties for an average number of fission neutrons and cross sections, included in the MF31 and MF33 files of an ENDF-6 file, respectively, a capability of working with the average scattering cosine (MF34) and fission spectrum (MF35) has been introduced.
Based on the generated covariances and sensitivity vectors, calculated with a neutron transport code (currently Serpent [2]), another capability was implemented to automatically propagate the uncertainty similar to the SAMS module of the SCALE code system and other modules of it [3]. Finally, based upon the results of the uncertainty propagation, the developed package permits one to single out the reactions and their energy regions with the highest influence of uncertainty on the functional, which require the most attention to reconsider the data for meeting target accuracy requirements (TARs).
To sum up, the package can completely provide tools to conduct a systematic sensitivity and uncertainty analysis with respect to nuclear data from ENDF-6 pointwise covariance data, taking into account different features of the modern libraries, to the analysis of required nuclear data needs with TAR.

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P11

Start-up core design of RFBB sodium fast reactor with silicide fuel

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The advancement of innovative fast nuclear reactor systems offers a dependable and sustainable solution to address the world's growing energy needs. Breed-and-burn systems, in particular, have the potential to efficiently utilize uranium fuel, which could lead to a reduction in nuclear waste and eliminate the need for expensive fuel reprocessing facilities. In this paper, we design a core for a Rotational Fuel-shuffling Breed-and-Burn reactor with Silicide fuel and Sodium coolant (RFBB-SS). To design the start-up core, first we performed the equilibrium state analysis for the SFR core designed with an RF scheme and with control rod assemblies (CRAs). We then analysed the start-up to the equilibrium state. Then, to reveal its safety features, the control rod assembly worth and the maximum temperatures of the reactor domains were estimated. The results showed that a small start-up core for an RFBB-SS design is feasible and that the maximum temperatures of the fuel cell domains in a hot channel can be kept well below their safety limits. Hence, the reactor could achieve a subcritical state when all control rod assemblies are inserted into the core.

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P12

Poster/Presentation Activity

Critical Minerals for Next Generation Nuclear Reactors

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Abstract:

With an aspiration for sustainable and climate-resilient future, nuclear energy is one of the backbones of clean energy transitions. It provides 10 per cent of the world's total electricity and a quarter of its low-carbon supply. Over the past 50 years, the use of nuclear energy has reduced CO_2 emissions by over 60 gigatonnes (nearly two years' worth of global energy-related emissions [1]. During the period from 1971 - 2018, nuclear power provided nearly 76 000 TWh of zero-emissions electricity which is more than ten times the total output of wind and solar combined. On the flip side, nuclear energy has begun to fade in advanced economies, with plants getting closed and shrinkage in new investments and collaborations.

Amongst others, the reasons for the downside lie in the dependencies and access of critical minerals for the nuclear reactors. For instance, application of Rare Earth Element (REE) - Gadolinium (Gd), with atomic number 64 is used as a neutron absorber for nuclear reactor shielding. Its current production is heavily concentrated in China with availability of more than 70 per cent of the world's rare earth mines and 90 per cent of the complex process of turning them into magnets [2]. In the Sustainable Development Scenario (SDS), total mineral demand from nuclear power (mostly chromium, copper, and nickel) is projected to grow by around 35 per cent compared to 2020 levels, reaching almost 70 kt by 2040 [3]. Its share lowered to just over 60 per cent in 2019, as the United States, Myanmar and Australia started to boost production [4]. And fuels like uranium are no longer recognized as critical minerals since the Energy Act of 2020. Therefore, addressing critical minerals needs and opportunities in nuclear reactors calls for the multidisciplinary approach, involving materials science, engineering, policy and greater international collaboration.

This paper aims to examine role of critical minerals for next generation nuclear reactors and opportunities for collaborations with a fishbone diagram. By showcasing categories, causes and effects from materials, technology and collaboration point of view the analysis will provide a structured approach to distil this complex understanding and enhance collaborative efforts in nuclear reactor R&D. Consequently, it provide recommendations to policy makers, scientists, and academia involving factors like, supply chain vulnerabilities, geopolitical risks and technological dependencies from the economic security perspective.

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P13

DIFFERENT TECHNIQUES FOR REDUCING DLOFC FUEL TEMPERATURES IN A PBMR-DPP-400 CORE

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Abstract: The principle strategies used for reducing the maximum DLOFC temperatures were (a) flattening the peaks in the axial profiles of the maximum DLOFC temperature, in order to increase the surface areas over which effective evacuation of decay heat takes place. This reduces the resulting maximum heat fluxes and temperatures in the hotspots; and (b) “pushing” the radial profiles of the equilibrium power density outward towards the external reflector, thereby decreasing the distance, and thus the thermal resistance, over which the decay heat has to be evacuated towards the external reflector. Easier radial evacuation of decay heat reduces the maximum DLOFC temperatures, which always occur in the inner layers of the fuel core.

These strategies were applied for both 6-pass recirculation fuelling schemes and Once Through Then Out (OTTO) fuelling schemes. The techniques used for flattening the peaks in the axial profiles of the maximum DLOFC temperature were (a) flattening of the peaks in the axial profiles of the equilibrium power density by adding thorium to the LEU fuel in order to improve the breeding and conversion ratios, which slowed the depletion of the enrichment of the fuel with increasing burn-up and thereby increasing the power density in the bottom parts of the fuel core; and (b) placing purposely-designed distributions of neutron poison in the central reflector in order to supress the normal peaks in the axial profiles of the equilibrium power density. The poison in the central reflector simultaneously served the purpose of pushing the power densities outward from the central towards the external reflector.

This strategy was further implemented by creating asymmetric cores in which the enrichment of the fuel in the outer fuel flow channels was higher than in the inner ones, which automatically shifts the fission power out to these higher enriched outer fuel zones.

The result was large reductions in the maximum DLOFC temperatures from 1536°C to 1298 °C for the multi-pass and from 2273°C to 1448°C for the OTTO. The use of neutron poison in the central reflector to flatten the peaks in the axial profiles of the maximum DLOFC temperatures reduced the maximum DLOFC temperature much more effectively than any of the other techniques.

P14

Study on Loss of Coolant Accidents in Future Nuclear Reactors Design

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Next-Generation nuclear reactors [1] are poised to become significant future sources of energy production for local and industrial applications. Their innovative designs aim to reduce the risk of accidents and improve the overall safety of these nuclear power plants. Additionally, next-generation reactors promise to deliver higher thermal efficiencies, making them more suitable for a variety of industrial applications such as hydrogen production, desalination, and district heating.

However, ensuring the safety of these plants is paramount. One of the most severe accidents across all reactor types is the Loss of Forced Cooling (LOFC) accident [2]. In Light Water Reactors (LWRs), an LOFC accident can result in core melt due to temperatures exceeding critical limits. This type of accident also poses a significant threat to next generation reactors. The present research delves into the study of LOFC accidents in new generation reactors. The study employs a combination of theoretical analysis, simulation models, and empirical data to assess the mechanisms of this accident.

This research highlights the importance of robust safety measures and innovative design solutions to ensure the continued safe operation of future nuclear power plants.

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P15

Application of probabilistic safety assessment method in high temperature gas-cooled reactors

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High Temperature Gas Cooled Reactors (HTGR) [1] represent a new generation of reactors capable of achieving outlet temperatures between 750°C and 950°C, making them ideal for cogeneration applications that supply high-temperature heat for industrial processes. These reactors are often designed in multi-unit configurations to enhance scalability, safety, economic efficiency, load management, and site utilization through modular construction, shared infrastructure, redundancy, and simplified licensing. Probabilistic Safety Assessment (PSA)[2] is an essential tool for evaluating the safety of nuclear reactors, including HTGRs. This research explores applying PSA to HTGRs to enhance risk understanding and safety measures.

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Abstract template for 4th Joint ICTP-IAEA Workshop on Physics and Technology of Innovative Nuclear Energy Systems

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The effective handling of used nuclear fuel is a crucial obstacle in the implementation and operation of sophisticated nuclear reactors, namely those classified as Generation IV (Gen4). This review article presents a thorough examination of the many technologies used for reprocessing spent fuel in modern reactors. The work specifically focusses on procedures such as PUREX (Plutonium Uranium Redox Extraction) and pyroprocessing[1][2]. This study investigates the fuel compositions used in Gen4 reactors, namely TRISO, HEU, HALEU, LEU, and ANEEL fuels, and analyses the resulting products during reprocessing. In addition, the article assesses the dangers of proliferation linked to these technologies and examines the consequences for large-scale use in commercial settings[1][2].

The objective of this paper is to provide a comprehensive analysis of the chemical and physical properties of used fuels, the effectiveness of reprocessing techniques, and the possibility of resistance to proliferation[2][4]. Through a comparison of many fuel types and their corresponding reprocessing methods, the research emphasises the compromises between fuel efficiency, waste reduction, and non-proliferation. This study is essential for providing information for the design and policy choices about the future of nuclear energy and guaranteeing the safe, secure, and sustainable utilisation of nuclear power[1][2].

In response to the increasing global energy requirements and the need to address climate change, modern nuclear reactors provide a very promising solution[1]. Nevertheless, the successful implementation of these reactors depends on the development of efficient methods for managing spent fuel that specifically tackle the distinct barriers presented by their fuels and coolants[1]. This review article is an important reference for scholars, policymakers, and industry stakeholders on the intricate management of spent fuel and the development of a sustainable nuclear future.

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P17

ANALYSIS OF ERBIUM TRIFLUORIDE (ErF3) AS A BURNABLE POISON IN MICRO-MOLTEN SALT REACTOR 25 MWt

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Reactivity control in nuclear reactors is critical for safety and efficiency. Using burnable poisons is an effective method for managing reactivity by absorbing excess neutrons and reducing initial reactivity. This study examines the performance of Erbium Trifluoride (ErF_3) as a burnable poison in a Micro-Molten Salt Reactor (MSR), with Erbium (Er) concentrations varying from 0.005% to 0.035% mol. Neutronic analysis was performed using OpenMC with the ENDF/B-VII.0 neutron cross-section library. Key metrics evaluated include the effective multiplication factor (K_eff) and the temperature coefficient of reactivity (TCR), with an initial K_{eff} of 1.08. Results show that increasing Erbium concentration substantially lowers the TCR, from -3.4994 pcm/K at baseline conditions to -9.0273 pcm/K at 0.035% Er concentration. This significant reduction in TCR suggests that higher Erbium concentrations increase the reactor's sensitivity to temperature changes, which may influence its stability and operational performance.

Keywords : Micro-Molten Salt Reactor (MSR), OpenMC, Burnable Poison, Neutronic

P18

Conceptual Design and Safety Research of Heat Pipe-Cooled Space Nuclear Reactors

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Enhancing the capabilities of unmanned space exploration, such as satellite monitoring and space science missions, requires efficient and reliable nuclear power systems. A viable solution is found in the 1-10 kW power level of space nuclear reactor power systems, offering advantages such as a manageable research and development process, and relatively low investment requirements [1]. The author's team proposed a 200 kWe space nuclear reactor power design based on the combination of an integrated UN ceramic fuel, a heat pipe cooling system and the Stirling power generators, meanwhile, the secondary neutrons produced by cosmic rays interacting with reactor materials was evaluated, and it was found that the secondary neutrons can meet the basic source requirement for space reactor passive start-up [2]. Huang evaluated the performance of a heat pipe cooled device with thermoelectric generator for nuclear power application, resutls show that the maximum core temperature under the loss of heat sink accident is 1600 K, which exceeds the copper's melting point temperature 1356 K [3].

This poster introduces a conceptual design for a 5 kWe space nuclear reactor power system, outlining its components and characteristics. The reactor power system as shown in Figure 1, is based on large-size honeycomb uranium alloy fuel, drum-type reactivity control system, high-temperature sodium heat pipe as the heat transfer device, shadow-type shielding, and Stirling power generation. This poster demonstrates a thorough analysis of potential challenges, encompassing heat pipe failure accidents, re-entry scenarios, and weight estimation considerations. The results show that the proposed space nuclear reactor power system concept effectively meets the safety requirements, and the total mass of the power system is estimated at approximately 1.5 tons, with a specific mass of around 300 kg/kW. This research contributes valuable insights for the design of space nuclear reactor power systems operating within a similar power range, providing a foundational reference for future developments in this field.

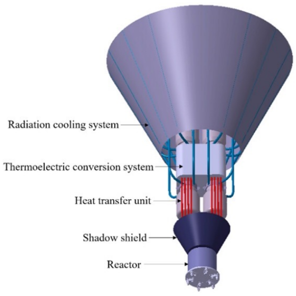

Figure 1: space nuclear reactor power system

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