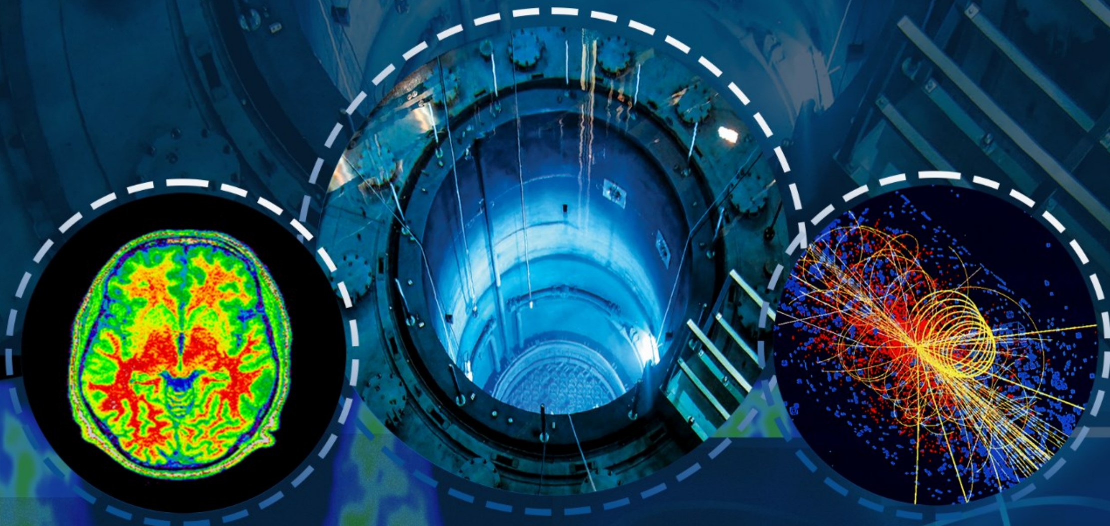
BOOK OF ABSTRACTS

ICANSE 2024

International Conference on Advances in Nuclear Science and Engineering (ICANSE) 2024



2-4 SEPTEMBER 2024

East Aula - ITB
Ganesha Campus
Bandung, Indonesia

ICANSE 2024 International Conference on Advances in Nuclear Science and Engineering

FMIPA·ITB

DPITm

104 TAHUN ITB

65TH DIES NATALIS Institut Teknologi Bandung

Kampus Merdeka INDONESIA JAYA

LEMBAGA PENELITIAN DAN PENGABDIAN KEPADA MASYARAKAT INSTITUT TEKNOLOGI BANDUNG

PERTAMINA

ROSATOM

PREFACE BOOK OF ABSTRACT ICANSE 2024

In the era of rapidly developing research and technology and research, studies in nuclear science and engineering have become increasingly important. Faced with global challenges related to energy, health, and the environment, knowledge in these fields is crucial for addressing complex problems. The International Conference on Advances in Nuclear Science and Engineering (ICANSE) serves as a vital forum for scientists, researchers, students, and practitioners to share the latest ideas, knowledge, and findings in nuclear science and engineering. Indonesia faces a growing demand for clean and reliable energy. Nuclear energy presents a potential solution to meet this demand while reducing greenhouse gas emissions. However, public perception and knowledge about nuclear energy can be limited. Utilization of nuclear energy have been used not only for energy used application but also for non-energy application. In term of energy application which is based on the heat which is produced from the reactor that can be used to generate the turbine for electricity production as well as co-generation applications. Co-generation system comes from the excess heat from the reactor which can be used for water desalination, hydrogen production, for coal gasification and liquefaction applications and enhanced oil recovery (EOR) applications. Energy business of NPP can drive the economy of the country especially by producing enough electricity to supply the industrial demand and other services. Another business which is related to nuclear energy is non-energy applications for health and medicine, pharmaceuticals, nuclear batteries, metal industry, agricultural environment, livestock, and so on based on the radioisotope utilization. This application for medical and radio pharmacies application that requires nuclear application based on radioisotope utilization which is produced from a reactor as well as accelerator facilities.

The 7th International Conference on Advances in Nuclear Science and Engineering, (ICANSE) 2024 as the series international conference since 2007, aims at summarizing recent research activities relevant to the advanced development of application in the nuclear science and engineering and facilitate communication among relevant experts. After join symposium between Bandung Institute of Technology (ITB) and Tokyo Institute of Technology (Tokyo Tech) at 2005, the conference of ICANSE was established based on one of two years event 2007, 2009 and 2011. It was organized by Bandung Institute of Technology (ITB), National Nuclear Energy Agency (BATAN) and supported by ministry of education of Indonesia and COE-INES and CRINES of Tokyo Institute of Technology (Tokyo Tech). Participants, invited speaker and contributor speaker are invited from academia in university, research center and governmental organization as well as industry and public who interested in nuclear science and technology.

In this 7th ICANSE 2024, as an international conference after long break after covid-19 pandemic, the event is held In conjunction with Annual Meeting and Business Nuclear Society, organized by Bandung Institute of Technology - Department of Physics, Department of Nuclear Science and Engineering (ITB) Faculty of Mathematics and Natural Sciences (FMIPA), Center For Nuclear Science, Technology and Innovation (PSTIN), Bandung Institute of technology. The event is sponsored by LPPM ITB, PT Pertamina (Persero), Indonesia and Rosatom Russia and some supporting organizations Institute for Science and Technology Development (LPIT), Bandung Institute of technology, Indonesian Nuclear Society Association (HIMNI), Indonesian Nuclear Society (HIMNI), Research Organization for Nuclear Energy, National Research and Innovation Agency (BRIN),

The theme of the International Conference on Advances in Nuclear Science and Engineering (ICANSE) 2024 is "Optimizing Strategy of Nuclear Sciences and Technology Utilization on Global NZE, Energy Security, Health, and Economy Growth". This theme focuses on the role of nuclear technology in addressing global challenges related to energy, health, and the environment, and emphasizes the importance of innovative research and applications in promoting sustainable development. The topics of the conference are listed but not limited to:

- Innovative nuclear energy systems
  - Nuclear utilization systems based on fuel cycle
  - Simultaneous solution for safety, radioactive waste and proliferation problems
  - Advanced Small Reactors without on-site Refueling
  - Small Modullar Reactors
  - Generation IV Nuclear Reactors
- Innovative transmutation systems
- Innovative separation and fuel cycles/radioactive wastes
- Nuclear nonproliferation issues
- Innovative energy systems
  - Hydrogen energy system
  - Cogeneration system
  - Thermal energy utilization system
- Material and process for innovative energy systems
- Radiation Physics and Protection
- Biophysics, medical physics and nuclear medicine
- Nuclear Physics and Nuclear Data
- Accelerator and Sub-critical Assembly.
- Theoretical and Computational nuclear physics and particle physics
- Nuclear education, social and public acceptance
- Energy management and planning, Policy and Regulation

Chairperson of ICANSE 2024

CONFERENCE PROGRAM AND PRESENTATION SCHEDULE

First Day
September 02, 2024

No	Time	Activity	Topics
1	08.00 - 09.00	Registration	
2	09.00 - 09.10	Opening Ceremony	
3	09.10 - 09.20	Opening Remarks from HIMNI/Indonesian Nuclear Society	
4	09.20 - 09.30	Opening Remarks and Opening of Annual Meeting of HIMNI and ICANSE 2024 (ITB Officials)	
5	09.30 - 09.50	Plenary Session and QnA: Ministry of Energy and Mineral Resources of the Republic of Indonesia*	Indonesia Nuclear Policy and Application on Nuclear Energi Implementation for NZE in Indonesia
6	09.50 - 10.10	QnA	
7	10.10 - 10.30	Indonesian Nuclear Association (HIMNI)	Policy and Regulation
8	10.30 - 10.50	National Energy Council (DEN)*	
9	10.50 - 11.10	National Research and Innovation Agency (BRIN)*	
10	11.10 - 11.30	Nuclear Energy Regulatory Agency (BAPETEN)*	
11	11.30 - 12.00	QnA	
12	12.00 - 13.00	Lunch Break	
13	13.00 - 13.30	Plenary Session and QnA: Ministry of Health of the Republic of Indonesia*	Health reform policy and implementation of nuclear science and technology for health and medicine
14	13.30 - 14.00	QnA	

15	14.00 - 14.20	Plenary Session: President Director of PT Pertamina (Persero) *	Discussion with Nuclear Stakeholder and Business Community
16	14.20 - 14.40	Plenary Session: EVP PLN	
17	14.40 - 15.15	QnA	
18	15.15 - 15.45	Coffee Break	
19	15.45 - 16.05	Plenary Session: Deputy Director of ROSATOM	Discussion with Nuclear Stakeholder and Business Community
20	16.05 - 16.25	Plenary Session: Director of Kakiatna Enjiniring	
21	16.25 - 16.55	QnA	
22	16.55 - 17.00	Closing Day 1	
23	19.00 - 21.00	Welcome Dinner and Business Meeting	

Day Two
September 03, 2024

No	Time	Activity	Topics
1	09.00 - 09.20	Plenary Session 1 : ITB (Zaki Su'ud) (Offline)	Human Resources and Research and Development (RnD) in the nuclear sector for energy and health
2	09.20 - 09.40	Plenary Session 1 : BRIN (Head/Deputi SDM)*	
3	09.40 - 10.00	QnA	
4	10.00 - 10.20	Plenary Session 1 : ITB (Freddy Haryanto) (Offline)	
5	10.20 - 10.40	Plenary Session 1 : Bapeten (Head/Deputi)*	
6	10.40 - 11.00	QnA	
7	11.00 - 11.15	Paralel Session 1	
8	11.15 - 11.30	Paralel Session 1	
9	11.30 - 11.45	Paralel Session 1	
10	11.45 - 12.00	Paralel Session 1	
11	12.00 - 13.00	Lunch Break	
12	13.00-13.20	Plenary Session 2 : Artisiuk (Online)	Small Modular Reactor
13	13.20-13.40	Plenary Session 2 : Liem Peng Hong (Offline)	

14	13.40 - 14.00	QnA	
15	14.00 - 14.15	Paralel Session 2	
16	14.15 - 14.30	Paralel Session 2	
17	14.30 - 14.45	Paralel Session 2	
18	14.45 - 15.00	Paralel Session 2	
19	15.00 - 15.15	Paralel Session 2	
20	15.15 - 15.30	Paralel Session 2	
21	15.30 -15.45	Coffee Break	
22	15.45 - 16.00	Paralel Session 3	
23	16.00 - 16.15	Paralel Session 3	
24	16.15 - 16.30	Paralel Session 3	
25	16.30 - 16.45	Paralel Session 3	
26	16.45 - 17.00	Paralel Session 3	

Day Three
September 04, 2024

No	Time	Activity	Topics
1	09.00 - 09.20	Plenary Session 3 : Jinsuo Zhang (Online)	Advance Nuclear Reactor and Nuclear Material
2	09.20 - 09.40	Plenary Session 3 : Kinoshita (Offline)	
3	09.40 - 10.00	Plenary Session 3 : Indarta Kuncoro Aji (Offline)	
4	10.00 - 10.30	QnA	
5	10.30 - 10.45	Paralel Session 4	
6	10.45 - 11.00	Paralel Session 4	
7	11.00 - 11.15	Paralel Session 4	
8	11.15 - 11.30	Paralel Session 4	
9	11.30 - 11.45	Paralel Session 4	
10	11.45 - 12.00	Paralel Session 4	
11	12.00 - 13.00	Lunch Break	
12	13.00 - 13.20	Plenary Session 4 : Topan Setiadipura (Offline)	NPP for Cogeneration and Industry

13	13.20 - 13.40	Plenary Session 4 : M. Ilham (Online)	
14	13.40 - 14.00	QnA	
15	14.00 - 14.20	Plenary Session 5 : Alexander Tsibulya (Offline)	Alademik Lomonosov FPU: Operating Experience
16	14.20 - 14.40	Plenary Session 5 : Rosenergoatom (Online)	Virtual Tour for The First Floating Nuclear Power Plant (NPP)
17	14.40 - 15.00	QnA	
18	15.00 - 15.30	Coffee Break	
19	15.30 - 15.45	Paralel Session 5	
20	15.45 - 16.00	Paralel Session 5	
21	16.00 - 16.15	Paralel Session 5	
22	16.15 - 16.30	Paralel Session 5	
23	16.30 - 16.45	Paralel Session 5	
24	16.45 - 17.00	Paralel Session 5	
26	17.00 - 17.15	Closing Ceremony	

PARALLEL SESSION SCHEDULE

Paralel 1 - September 3, 2024				
Time	**Room 1**	**Room 2**	**Room 3**	**Room 4**
11.00 - 11.15	ABS-43	ABS-24	ABS-69	ABS-30
11.15 - 11.30	ABS-51	ABS-33	ABS-26	ABS-37
11.30 - 11.45	ABS-53	ABS-1	ABS-34	ABS-42
11.45 - 12.00	ABS-52	ABS-38	ABS-12	ABS-57

Paralel 2 - September 3, 2024				
Time	**Room 1**	**Room 2**	**Room 3**	**Room 4**
14.00 - 14.15	ABS-20	ABS-44	ABS-40	ABS-35
14.15 - 14.30	ABS-64	ABS-45	ABS-4	ABS-66
14.30 - 14.45	ABS-50	ABS-58	ABS-19	ABS-22
14.45 - 15.00	ABS-56	ABS-63	ABS-2	ABS-10
15.00 - 15.15				
15.15 - 15.30				

Paralel 3 - September 3, 2024				
Time	**Room 1**	**Room 2**	**Room 3**	**Room 4**
15.45 - 16.00	ABS-21	ABS-9		
16.00 - 16.15	ABS-3	ABS-13		
16.15 - 16.30	ABS-48	ABS-39		
16.30 - 16.45	ABS-59	ABS-60		
16.45 - 17.00	ABS-11			

Paralel 4 - September 4, 2024				
Time	**Room 1**	**Room 2**	**Room 3**	**Room 4**
10.30 - 10.45	ABS-23	ABS-46	ABS-18	ABS-7
10.45 - 11.00	ABS-49	ABS-61	ABS-41	ABS-47
11.00 - 11.15	ABS-5	ABS-62	ABS-25	ABS-55
11.15 - 11.30	ABS-6	ABS-67	ABS-28	ABS-71
11.30 - 11.45	ABS-16	ABS-68	ABS-29	ABS-36
11.45 - 12.00			ABS-65	ABS-80

Paralel 5 - September 4, 2024				
Time	**Room 1**	**Room 2**	**Room 3**	**Room 4**
15.30 - 15.45	ABS-8	ABS-77	ABS-79	
15.45 - 16.00	ABS-17	ABS-78	ABS-76	
16.00 - 16.15	ABS-27	ABS-75	ABS-70	
16.15 - 16.30	ABS-32	ABS-74	ABS-72	
16.30 - 16.45	ABS-31	ABS-73		
16.45 - 17.00				

ICANSE 2024 ROOM MAP
EAST HALL, BANDUNG INSTITUTE OF TECHNOLOGY

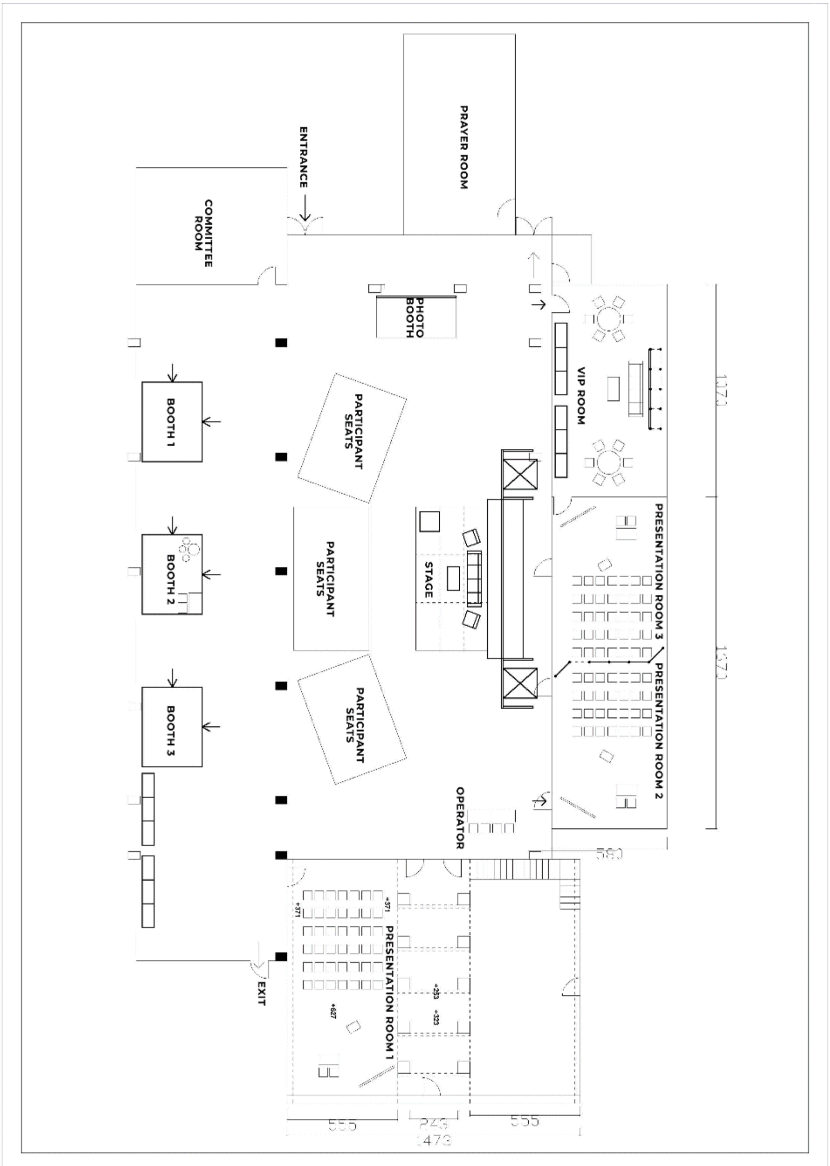


TABLE OF CONTENT

[ENERGY MANAGEMENT, REGULATION, AND POLICY]

[INNOVATIVE NUCLEAR ENERGY SYSTEMS]

[INNOVATIVE TRANSMUTATION SYSTEMS]

[MATERIAL AND PROCESS FOR INNOVATIVE ENERGY SYSTEMS]

[NUCLEAR NONPROLIFERATION ISSUES]

[RADIATION PHYSICS]

[ABS-40]

Measurement and Identification of Human Brain Wave Signals while Listening to Al-Quran Recitation using Electroencephalogram (EEG) and Fourier Analysis

Elin Yusibani, Ayu Mahara, Fahyumi Senye, Fashbir, Adi Rahwanto, Evi Yufita

Physics Department Faculty of Mathematics and Natural Sciences Universitas Syiah Kuala Banda Aceh Indonesia, 23111

Abstract

Brain signals are not entirely random- they exhibit periodic properties in their components. Various studies have been conducted to determine the clinical, physiological, and psychological effects of these signals. However, the underlying mechanisms behind these signals have not been widely explored. The recorded Electroencephalogram (EEG) signal patterns are associated with mental activity, levels of consciousness, physiological conditions, and brain pathology. Generally, the human brain produces five basic signals: delta, theta, alpha, beta, and gamma. These signals function optimally in different situations, and disruptions in their production can lead to various issues. An increase in the amplitude of the theta and alpha waves can indicate a person^s comfort level. Individuals with higher alpha levels tend to be less easily agitated, which indirectly supports the immune system. Fourier Analysis was applied to identify EEG signal patterns while listening to Quran recitation. Electrodes were placed using the 10 - 20 system, and brain wave signals were recorded for approximately 3 minutes. The signals were then analyzed to assess the impact of listening to Quran recitation.

Keywords: EEG, Fourier analysis, brain wave signal, al quran recital

Topic: Biophysics and Medical Nuclear Physics

[ABS-43]
Composition the macro-micro mineral and heavy metals content in processed indigenous fish species: implications for food security

Fatmawati Nurcahyani, T R Mulyaningsih, Firda Amalia, S Yusuf, Nadya Nurdini, Alfian*

Research Center for nuclear beam analysis technology - Research Organization for Nuclear Energy - National Research and Innovation Agency
Kawasan Puspiptek Serpong, Tangerang Selatan, Banten, Indonesia
a fatm003@brin.go.id

Abstract

This study was conducted to identify the composition of macro-micro minerals and heavy metals in processed indigenous fish species because data on mineral content and heavy metal contaminants are still limited. Processed fish is marine fish that has been processed with the aim that the fish is not easily damaged. Processed fish were sampled from traditional markets in Pemalang Central Java on June 2021. The mineral content in the sample was analyzed using neutron activation analysis. Sample irradiation was performed on the Rabbit system of G.A Sywabessi Multipurpose Reactor in Serpong, at a neutron flux of &#8776-3 x 1013 n cm-2s-1. Quantitative analysis was carried out using the INAA comparison method and INAA-ko-IAEA software. The results showed that processed fish contained macro minerals of Na, Ca, Mg, and K- micro mineral of Co, Cr, Fe, Mg, Mn, Se and Zn, and toxic elements of Hg and As. The concentration of essential elements Na, K, Ca, and Mg >500 mg/kg- The elements Fe and Zn are in the order of 20-267 mg/kg, while the concentrations of the Cr, Mn, Se, and Co elements are <33 mg/kg. Hg was detected in all fish species, but not detected in striped mackerel and anchovy. The concentration of Hg is 0.2-2.08 mg/kg and As concentration is 2.85-39.46 mg/kg. This value exceeds the maximum limit of the value issued by the Indonesian Food and Drug Supervisory Agency, 0.25 mg/kg for As and 1.0 mg/kg for Hg in processed fish, respectively. Target Hazard Quotient (THQ) for individual elements Zn, Cr, Mn, Co, Hg, and As respectively, and Hazard Index (HI) for combined elements gave a value<1. These indicated that these processed fish species are safe for consumption even though the study results contain heavy metals. This is because the level of fish consumption in Indonesia is still low. The data obtained from this study can be used to complement the nutrient food database and provide information about potential food safety hazards. Future research needs to be continued for food safety control.

Keywords: macro-micro mineral, processed indigenous fish, food security
Topic: Biophysics and Medical Nuclear Physics

[ABS-51]
Design and testing of the oblique and V-shaped Cu-ITO microelectrode arrays to generate dielectrophoretic force on red blood cells

Edwar Iswardy, Khazanna, Elin Yusibani, Mursal, Kurnia Lahna, Sri Fitriyani

1 Department of Physics Faculty of Mathematics and Natural Sciences, Banda Aceh, Indonesia, 23111
2 Master Program in Physics, Department of Dental Materials, Faculty of Dentistry, Banda Aceh, Indonesia, 23111
3 Department of Dental Materials, Faculty of Dentistry, Banda Aceh, Indonesia, 23111

Abstract

Dielectrophoresis (DEP) is a phenomenon in which a force is applied to a particle induced by a gradient of an electric field. The dielectrophoretic technique is popular for manipulating bioparticles, because it requires only a small sample, is label-free, rapid, and inexpensive. Manipulation of biosample can be in the form of monitoring, separation, sorting, capturing, etc., so that the DEP method can be applied as a biosample analysis tool. However, research on the application of the DEP method is still developing on various electrode arrays and bioparticles. In this work, a lab-on-chip device with an oblique and V-shaped 3D microelectrode array has been developed to manipulate red blood cells using the dielectrophoresis (DEP) method. The microelectrodes were fabricated with copper and indium tin oxide films on a glass substrate, while the microchannel was constructed using double-sided tape insulators. Red blood cell samples were prepared in deionized water and EDTA medium with an electrical conductivity of 1.5 S/m. The test of dielectrophoretic force characteristics on red blood cells was carried out by applying an AC signal to the microelectrode, and the phenomenon was observed using a microscope with a CCD camera. The results showed that negative DEP forces were observed at frequencies of 5-7 MHz, 3.5-5 MHz, and 2-4 MHz in the oblique electrode spacing area and in the middle area of the V-shaped electrode curve. While positive DEP forces were observed at frequencies of 8-14 MHz, 6-13 MHz, and 5-11 MHz in the edge area of the oblique electrode and in the inner tip area of the V-shaped electrode curve, respectively at voltages of 5 Vpp, 10 Vpp, and 15 Vpp. The results of this work show the promising potential of lab-on-chip devices with oblique and V-shaped microelectrode arrangements to manipulate bioparticles so that biosamples can be further analyzed.

Keywords: Oblique and v-shaped microelectrodes, Dielectrophoresis, Red blood cells, Non-uniform electric field

Topic: Biophysics and Medical Nuclear Physics

[ABS-52]

Comparison of Dose Distribution Between Proton and Photon Radiotherapy on Organ at Risk (OAR) Based On The Number of Beams

Jesse Owen, Rena Widita

Department of Physics, Faculty of Mathematics and Natural Sciences, Bandung Institute of Technology, Jalan Ganesha 10, Bandung 40132, Indonesia

Abstract

Proton and photon radiotherapy have become essential treatment methods in treating cancer. The number of radiation angles in therapy planning significantly influences the success of the therapy. Therefore, the aim of this study is to investigate the dose distribution of proton and photon radiotherapy on Organs at Risk (OAR) based on the number of radiation beam angles. This study involves dosimetry planning using matRad to create a Treatment Planning System (TPS) to observe and compare the dose distribution on the tumor target and OAR. By varying the number of radiation angles, this study will evaluate the effectiveness of both radiotherapy methods in delivering an optimal dose to the target while minimizing the dose to OAR. The results of the study are expected to provide further insights into the impact of the number of angles on dose distribution, providing a basis for selecting a more accurate radiotherapy method and improving the quality of cancer care.

Keywords: Dose Distribution, Photon, Proton, Radiation Angle

Topic: Biophysics and Medical Nuclear Physics

[ABS-53]
Addition of Repopulation on Tumor Control Probability (TCP) Model in Prostate Cancer

Salma Aqilah Zahroh (a), Rena Widita (a)*

a) Department of Physics, Faculty of Mathematics and Natural Sciences, Institut Teknologi Bandung, Jalan Ganesha 10, Bandung, 40132, Indonesia
salmaqilahz@gmail.com

Abstract

When designing fractionation schedules for radiotherapy, one critical factor to account for is overall treatment time. Accelerated repopulation of tumor cells can significantly impact biological effectiveness of treatment. To address this, incorporating a repopulation factor into tumor control probability (TCP) model becomes essential. In this study, we analyzed data from 9,690 patients treated with external beam radiotherapy. Patients were categorized into three risk groups: 28% low risk, 53% intermediate risk, and 19% high risk. TCP model was fitted to clinical outcomes, specifically 5-year biochemical relapse-free survival (5y-bRFS). We employed maximum likelihood estimation (MLE) with Nelder-Mead simplex algorithm to maximize likelihood function produced by TCP model. Our results show that dose required to counteract daily repopulation is 0.54 Gy/day, with a kick-off time of 23 days. Our analysis reveals that there was no significant difference in kick-off time and repopulation rate across different risk groups. Incorporating repopulation factor into the TCP model yielded a good fit to data, as indicated by Akaike Information Criterion (AIC). Strategies to mitigate tumor cell repopulation during radiotherapy include employing accelerated fractionation, which reduces overall treatment time and minimizes opportunities for accelerated tumor repopulation.

Keywords: Maximum likelihood estimation, Nelder-Mead simplex algorithm, Prostate cancer, Repopulation, Tumor control probability model

Topic: Biophysics and Medical Nuclear Physics

[ABS-22]
Periodic Safety Review of RSG-GAS: Strategy and Evaluation to Manage Research Reactor Aging

Endiah Puji Hastuti(a,b), Topan Setiadipura(b), Dwi Irwanto(a), Abdul Waris(a)*

a) Bandung Institute of Technology, Master of Computational Science, Master of Nuclear Science and Engineering, and Doctor of Nuclear Engineering Division, Faculty of Mathematics and Natural Sciences Dept., Bandung-Indonesia-401323
* endi001@brin.go.id
b) Research Center for Nuclear Reactor Technology, Research Organization for Nuclear Energy, National Research and Innovation Agency (BRIN), Building No. 80 BJ Habibie Science and Technology Area, Setu, South Tangerang, Indonesia-15314

Abstract

IAEA^s research reactor data base indicates that ~50% of 225 units of research reactor in the world have been in operation for more than 50 years and about 20% of them are more than 60 years, including research reactors in Indonesia. The latest research reactor in Indonesia is multi-purpose reactor RSG-GAS 30 MW that has been operating for 37 years (Its first criticality was in 1987). In accordance with the regulation of BAPETEN, the regulatory body, to ensure sustainable safe reactor operation, a periodic evaluation on reactor safety should be performed. This periodic evaluation is one technique to evaluate nuclear reactor safety whether the reactor design, the status of the structure system and component (SSC), equipment qualification, component aging, safety performance and operation feedback, safety management and nuclear preparedness program, and radiological impact on environment still fulfil the existing safety criteria. This paper is aimed to describe the strategy and evaluation on the periodic safety review (PSR) of RSG-GAS using the reactor operation report in year 2005 - 2015, which is 10 years as regulated. The results of the reactor operation evaluation based on these data show that RSG-GAS still meets the reactor operation safety criteria and the reactor is allowed to operate until 2030. Henceforth, evaluation on the reactor operation safety for 2016 - 2025 period is necessary. In addition, safety analysis report, aging management program, and the latest operation report are also required.

Keywords: RSG-GAS, research reactor, PSR, aging management

Topic: Energy Management, Regulation, and Policy

[ABS-26]
Integrating New and Renewable Energy in Indonesia^s Post-Pandemic Energy Landscape: The Role of Nuclear Power

Sunarko

Polytechnic Institute of Nuclear Technology
Babarsari, Yogyakarta

Abstract

The past COVID-19 pandemic has profoundly impacted the global economy and altered electricity use patterns, including in Indonesia. Government policies enforcing social restrictions and remote work led to increased household energy consumption while reducing commercial energy use. The pandemic caused a sharp economic downturn, with growth plummeting to -5.32%. Although the economy is recovering, some changes in energy use patterns persist. Utilities and energy companies have adjusted their infrastructure and investment plans in response to these shifts. The pandemic underscored the need for resilient and sustainable energy systems. The recent initiative to phase out coal in the energy sector paves the way for New and Renewable Energy (NRE) to provide more environmentally friendly electricity, helping to reduce greenhouse gas (GHG) emissions. This paper examines how NRE, particularly nuclear energy, can be integrated into the national energy mix, considering the recovering demand and the shift to new electricity consumption patterns post-pandemic.

Keywords: Sustainable energy- Coal phase-out- NRE- Nuclear energy

Topic: Energy Management, Regulation, and Policy

[ABS-34]
Preparing Decommissioning Regulatory Infrastructure for Embarking Countries with Ageing Nuclear Facilities: A Case Study of Indonesia

Reno Alamsyah (a), Sidik Permana (b) dan AbdulWaris (b)*

a) BAPETEN. Nuclear Energy Regulatory Agency
b) Bandung Institute of Technology

Abstract

Indonesia is considered as among nuclear embarking country. Besides that, in Indonesia currently a government research institute operates three research reactors, and six fuel cycle facilities in addition to one facility operated by a state-own company. All of these installations are more than 36 years of operating ages. Decommissioning issue for the two areas, i.e., embarking and ageing facility, is a very important challenge for the regulator to be prepared.

What national policy available and what to be prepared? Has pertinent international obligation fulfilled? What regulation and law existed and what to be prepared? How regulator build their capacity and what HRD Plan should be prepared? This study answered all of the above typical questions with a comparative, gap analysis and descriptive method. Data and information from reliable publications, such as from the IAEA and NEA or from scientific papers of recognized publication, were used in this study. Interview to senior re-searchers and managers of the engaged organizations was also carried out to identify potential impediments and solution with sufficient risk assessment discussion.

The description of the existing installations, current status of legislation, international instruments, and regulation and guides were provided together with the regulatory frame-work and updates of their capacity, training and education arrangements, and human re-sources. Regulatory approach chosen by the regulator and the application graded approach were evaluated, including with the availability of the TSO and its competency on decommissioning. This paper reviewed as well the regulator^s integrated management system, regulatory policies, infrastructure for consultation and communication to interested parties in developing regulation and decision making or licensing, and the infrastructure for oversight and enforcement. Finally, the paper appraises the core challenges in developing a better capacity building and human resources plan. Acceptance criteria in evaluating of these regulatory topics were developed from national common legislation and law, international agreements, international standards, especially from the IAEA, and good international practices.

This study concluded that Indonesia has been the parties of the required international agreement related to decommissioning. Indonesia could solve its challenges at the same time with the establishment of national policy in waste management, decommissioning and environmental remediation, in addition to the stablished nuclear safety policy. Furthermore, there are a plenty room for improving the regulation, regulator capacity and it^s HRD plan in some specific areas. This study also suggest that the regulator body may use many international and bilateral cooperation to help improve their regulatory infrastructure.

Keywords: Decommissioning, Regulation, Embarking Country, Ageing Facility

Topic: Energy Management, Regulation, and Policy

[ABS-66]
Comparison of the Use of Fissile Material U-233 and Plutonium in Thorium Nitride (ThN) Fuel in Small Long-Life Modular PWR Cores at 300 MWth Power Level

Boni Pahlanop Lapanporo (a,b), Zaki Su^ud (a), Asril Pramutadi Andi Mustari (a)*

(a) Department of Physics, Faculty of Mathematics
and Natural Sciences, Institut Teknologi Bandung
Jl. Ganesa 10, Bandung 40132, Indonesia
(b) Department of Physics, Faculty of Mathematics
and Natural Sciences, Universitas Tanjungpura, Jl. Prof.
Dr. H. Hadari Nawawi, Pontianak 78124, Indonesia
* boni8poro@physics.untan.ac.id

Abstract

In this study, neutronic calculations were conducted on thorium nitride (ThN)-based fuel with different fissile materials (U-233, reactor-grade plutonium (RG-Pu), weapon-grade plutonium (WG-Pu), and a mixture of RG-Pu and WG-Pu) in a small long-life modular PWR core with 300 MWth power. We calculated using the Standard Reactor Analysis Code (SRAC), employing the PIJ module for fuel cells and the CITATION module for the reactor core. The 3D XYZ reactor core model consisted of 89 fuel assemblies, each with 17x17 pins (264 fuel pins and 25 guide tubes). We composed the fuel pins of Th-232, fissile material, and Pa-231 as a burnable poison. We divided the core into three regions (F1, F2, F3) with increasing fissile material percentages from the center outward, differing by 1-2%. Results showed that U-233 and the mixture of RG-Pu and WG-Pu performed best, maintaining excess reactivity below 1.00% dk/k for 20 years. Power density distribution at a 65% fuel volume fraction was similar for all fissile materials, with plutonium being more uniform than U-233. All fissile materials had low and safe PPF values for PWRs. U-233 had the highest Doppler coefficient, RG-Pu the lowest, and WG-Pu increased at high volume fractions. The mixture of RG-Pu and WG-Pu had a moderate Doppler coefficient. Burnup levels were slightly lower for RG-Pu at 60% fuel volume and lower for the mixture of RG-Pu and WG-Pu at 65% fuel volume fraction.

Keywords: Long-life PWR, Neutronics, SRAC, Thorium, Uranium, Plutonium, BP

Topic: Energy Management, Regulation, and Policy

[ABS-69]
Analysis of weather field data at the TRIGA 2000 reactor site to enhance public safety measures

Haryo Seno (a), Muhamad Hilmi Haidar (b), Nabila Putri Rihan (b), Djoko Prakoso Dwi Atmojo (a), Jibran Alfandi Rachman (c), Prasetyo Basuki (d)*

a) Nuclear Energy Research Organization, National Research and Innovation Agency (BRIN)
*haryo.seno@brin.go.id
b) Telkom University
c) Gadjah Mada University
d) National Research and Innovation Agency (BRIN)

Abstract

Planning for a radiological safety and emergency preparedness program in the nuclear reactor facility requires a comprehensive understanding of environmental impact assessment. The radioactive plume dispersion emitted from nuclear reactors is becoming the most significant source for environmental risk in the atmospheric pathway. Many factors affect the radioactive plume dispersion into the environment. However, weather data, in particular wind speed and direction, significantly influence the dispersion of radioactive plumes into the environment. Since it is difficult to determine the wind speed and direction immediately after an emergency situation, a thorough grasp of previous wind data with a machine learning approach would be helpful to forecast subsequent and forthcoming wind data. This could predict the movement of the radioactive plume as precisely as possible, thus being able to facilitate proper countermeasure processes and protect the nearby population. This study examines a dataset of weather sensors from the last three years of the TRIGA 2000 nuclear reactor site. The characteristics of the wind field blowing from the reactor site to the surrounding heavily inhabited places are particularly investigated. In conclusion, suggestions for developing countermeasure strategies are offered.

Keywords: radiological safety, wind data, machine learning, emergency countermeasure

Topic: Energy Management, Regulation, and Policy

[ABS-79]

Study of the Impact of Nuclear Power Plants (NPP) and NuScale Reactors on Economic and Environmental Aspects

Kevin Wijaya(a) and Sidik Permana(a,b)

a) Nuclear Physics and Biophysics Research Division, Physics Department, Faculty of Mathematics and Natural Science, Bandung Institute of Technology, Jalan Ganesha 10 Bandung 40132, Indonesia.
b) Nuclear Science and Engineering Department, Faculty of Mathematics and Natural Science, Bandung Institute of Technology, Jalan Ganesha 10 Bandung 40132, Indonesia.

Abstract

Nuclear Power Plants (NPPs) are one type of power plant that can produce large amounts of electricity and do not produce carbon. Although when operating, NPPs can compete with power plants from other energy sources, they require quite large capital costs. In addition, there have been cases of nuclear accidents that have an impact on the environment. Therefore, this article discusses the economic and environmental analysis of the existence of NPPs. In addition, the NuScale Reactor is also analyzed from the design and development, seawater desalination, and the impact of NuScale on the economy and the environment. From the results of the study, it was found that NPPs require large capital costs but when operating they tend to be smaller, NPPs can also open up jobs from various sectors, have a significant impact on environmental objects such as water quality, soil, air, ecosystems, and habitats. Also obtained are design features that are only owned by NuScale and can answer the Fukushima disaster problem. Then, information was obtained that there are three desalination and reverse osmosis (RO) processes that are more efficient in terms of economy and produce clean water, while if water quality is desired, the multi-effect distillation (MED) process can be used.

Keywords: Desalination, economy, environment, nuclear power plant, NuScale

Topic: Energy Management, Regulation, and Policy

[ABS-7]
Preliminary Numerical Investigation of Cooling Fins Feature in the Frozen Salt Melting Performance

Anni Nuril Hidayati (a), Asril Pramutadi Andi Mustari (b), Yulia Mifftah Huljanah (c), Nina Widiawati(a)*

a) Research Center for Nuclear Reactor Technology, Research Organization for Nuclear Energy, National Research and Innovation Agency, Building No. 80, BJ Habibie Integrated Science Area, South Tangerang 15310, Indonesia
*anni030@brin.go.id
b) Nuclear Physics and Biophysics Research Division, Physics Department Faculty of Mathematics and Natural Science, Institut Teknologi Bandung, Jalan Ganesha 10, Bandung, 40132, Indonesia
c) Bachelor Program of Physics Department, Faculty of Mathematics and Natural Science, Institut Teknologi Bandung, Jalan Ganesha 10, Bandung, 40132, Indonesia

Abstract

The opening time of the freeze plug safety feature in a molten salt reactor is a crucial aspect that must be carefully considered. This relates to the reactor core^s drainage during a blackout or abnormal temperature rise due to a LOCA, Loss of Coolant Accident, while in operation. This study analyzes the impact of varying the geometry of the distance between the freeze plug and cooling fins on the opening time. Calculations were performed using a particle-based Lagrangian numerical simulation called MPS, Moving Particle Semi-implicit. The calculations were conducted using two-dimensional geometry, with the initial design being an in-line arrangement. The approach involves forced convective heat transfer between the liquid fuel, the freeze plug, and the cooling fins, while conductive calculations were performed among the freeze plug particles during melting. This study is essential to determine the variations that yield optimal values for supporting the safety system of molten salt reactors.

Keywords: Forced Convection, In-line arrangement, Molten Salt, Particle, Plug

Topic: Innovative Nuclear Energy Systems

[ABS-8]

Comparative Analysis of Molten Salt Reactor (MSR) Using FLiBe, FLiNaK, and FNaBe with Power Output of 100 MWe

Cici Wulandari(a,b), Sidik Permana(a,b), Dwi Irwanto(a,b), Abdul Waris(a,b)*

a)Nuclear Physics & Biophysics Research Division, Department of Physics, Faculty of Mathematics and Natural Sciences, Institut Teknologi, Bandung, Indonesia
b)Department of Nuclear Science and Engineering, Faculty of Mathematics and Natural Sciences, Institut Teknologi, Bandung, Indonesia
*cici.wulandari01@itb.ac.id

Abstract

This study presents the neutronic design and analysis of a Molten Salt Reactor (MSR) with an electrical output of 100 MWe, specifically tailored for deployment in a low populated regions such as West Kalimantan or small islands where electricity grid connectivity is unreliable. The selection of a 100 MWe power level aims to meet the energy needs of these areas while minimizing the infrastructural and environmental impacts. The comparative analysis was conducted on various molten salt types, including FLiBe (LiF-BeF2), FLiNaK (LiF-NaF-KF), and FNaBe (NaF-BeF2), to determine the most efficient salt for reactor operations. Neutronic calculations were carried out using the SRAC (Standard Reactor Analysis Code) program, employing the PIJ (cell calculation module) and CITATION (core calculation module) to evaluate reactor performance. The results demonstrate that FLiBe molten salt offers significant advantages in terms of neutron economy, contributing to higher reactor efficiency and improved fuel utilization. These findings suggest that FLiBe is a superior choice for MSRs in the regions, providing a reliable and sustainable energy.

Keywords: FLiBe, FLiNaK, FNaBe, MSR, Neutronic

Topic: Innovative Nuclear Energy Systems

[ABS-10]
DESIGN AND ANALYSIS OF VODO-VODYANOI ENYERGETICHESKIY REAKTOR-1000 (VVER-1000) WITH MONTE CARLO N-PARTICLE (MCNP)

Kevin Wijaya(a), Dwi Irwanto(a,b), and Sidik Permana(a,b)*

a) Nuclear Physics and Biophysics Research Division, Physics Department, Faculty of Mathematics and Natural Science, Bandung Institute of Technology, Jalan Ganesha 10 Bandung 40132, Indonesia.
*kevinwijaya3001@gmail.com
b) Nuclear Science and Engineering Department, Faculty of Mathematics and Natural Science, Bandung Institute of Technology, Jalan Ganesha 10 Bandung 40132, Indonesia.

Abstract

Currently, electrical energy is still one of the sources of human needs in everyday life. With the depletion of fossil energy, nuclear power plants can become one of the suppliers of electricity that can be used. In Indonesia, there is still no nuclear reactor for nuclear power plants, only for research purposes.
One type of nuclear reactor is the Vodo-Vodyanoi Enyergeticheskiy Reaktor-1000 (VVER-1000).VVER-1000 is a generation III fission reactor with a thermal power of 3000 MWth. In this final project, we will discuss the design and analysis of the VVER-1000 using a program based on Monte Carlo N-Particle (MCNP) which is a license code from the Los Alamos National Laboratory (LANL). The purpose of this study was to determine the fuel assembly model, determine the multiplication factor, and determine the multiplication factor as a function of burnup and burnup rate with variations of uranium enrichment and gadolinium variations using MCNP. Broadly speaking, this research is divided into three stages, namely the literature study stage, the MCNP simulation and modeling stage, and the processing and analysis stage. Based on the experiments carried out, obtained a model of the fuel assembly along with the fuel cell, guide tube cell, and central tube cell. Then, it was obtained information that variations in uranium and gadolinium enrichment affect the value of the multiplication factor which affects the amount of neutron production. Then the results of the multiplication are obtained which changes the value of each factor and the variations used do not affect the burnup rate value.

Keywords: Burnup, fuel assembly, MCNP, multiplication factor, VVER-1000

Topic: Innovative Nuclear Energy Systems

[ABS-11]

STUDY OF THE EFFECT OF THORIUM AS A FUEL AND PROTACTINIUM-231 AS A BURNABLE POISON ON VODO-VODYANOI ENYERGETICHESKIY REACTOR-1000 FUEL ASSEMBLY USING MONTE CARLO N-PARTICLE (MCNP) SIMULATION

Kevin Wijaya(a), Dwi Irwanto(a,b), and Sidik Permana(a,b)*

a) Nuclear Physics and Biophysics Research Division, Physics Department, Faculty of Mathematics and Natural Science, Bandung Institute of Technology, Jalan Ganesha 10 Bandung 40132, Indonesia.
*kevinwijaya3001@gmail.com
b) Nuclear Science and Engineering Department, Faculty of Mathematics and Natural Science, Bandung Institute of Technology, Jalan Ganesha 10 Bandung 40132, Indonesia.

Abstract

One type of nuclear reactor is the Vodo-Vodyanoi Enyergeticheskiy Reactor-1000 (VVER-1000). VVER-1000 is a generation III fission reactor with a thermal power of 3000 MWth. In this final project, we will discuss the effect of giving thorium as a fuel and protactinium as a combustible poison in the VVER-1000 fuel assembly using a simulation program based on Monte Carlo N-Particle (MCNP) which is a license code from the Los Alamos National Laboratory (LANL). The purpose of this study was to determine the multiplication factor of the fuel assembly with variations in ThO_2 and variations in the fractions of gadolinium and protactinium-231, as well as the burnup rate for 5 years. Thorium-232 is predicted to be a substitute for UO_2 fuel in reactors because of its abundant material and can produce uranium-233 which is a better fuel than uranium-235. While protactinium-231 has a function other than absorbing neutrons, generating it can further produce fissile material that aids in chain reactions. Simulation of using MCNP to determine the effective multiplication factor with variations of ThO_2 fraction for 3 cases of administration of combustible poison, as well as variations of the fraction of combustible poison itself. Results Based on the simulation, the fuel assembly with Pa-231 as burnable poison has a higher value than gadolinium. In addition, the larger the burnable poison fraction, the smaller the multiplication factor value is. In addition, the multiplication factor and burnup rate were also obtained for 5 years.

Keywords: Burnable poison, burnup, MCNP, thorium, uranium

Topic: Innovative Nuclear Energy Systems

[ABS-17]
ANALYSIS OF BURNABLE POISON DOPING ON FUEL MOLTEN MIXTURE FOR MOLTEN SALT REACTOR (MSR) FUJI-12

Hilarius Adiwarna(b), Cici Wulandari(a,b), Syeilendra Pramuditya(a,b), Sidik Permana(a,b), Syaiful Bakhri (c)*

a) Nuclear Physics & Biophysics Research Division, Department of Physics, Faculty of Mathematics and Natural Sciences, Institut Teknologi, Bandung, Indonesia
b) Department of Physics, Faculty of Mathematics and Natural Sciences, Institut Teknologi, Bandung, Indonesia
c) Research Center for Nuclear Material and Radioactive Waste Technology (PRTBNLR), Research Organization for Nuclear Energy, National Research and Innovation Agency (BRIN), South Tangerang 15314, Indonesia
*psidik@itb.ac.id

Abstract

Nuclear Power Plants (NPPs) are a type of power generator that supports Sustainable Development Goals (SDGs) by producing clean, affordable, and climate-friendly energy. This study simulated the MSR FUJI-12 to determine suitable SRAC 2006 parameters with a PIJ module for cell calculation and CITATION module for core calculation. The simulation results showed an initial keff of 1.013046 for a U-233 F4 concentration of 0.22%, with a deviation of 0.0354% from the target of 1.0134. However, the final keff remained below 1, at 0.9455, prompting an increase in the U-233 F4 concentration to 0.28%. One of the challenges identified was the high Power Peaking Factor (PPF) of 3.49, that may damage the inside coating of reactor chamber. Using Protactinium as burnable poison and actinide producer to stabilize the keff, it was concluded that the doping on molten mixture is 0.003% to avoid uncriticality on MSR FUJI-12. However, the impact of the doping is unsignificant to the PPF. It is then proposed using other type of burnable poison, such as Americium (Am) and Gadolinium (Gd). However, as it was known that only Protactinium that produces burnable actinide to fuel the core, the impact is negligeble except for the Gd gas, as its doping is enough to decrease the initial keff and to increase the conversion ratio (CR) in MSR from 0.835 to 0.933.

Keywords: Burnable poison, MSR FUJI-12, Protactinium, U-233 F4

Topic: Innovative Nuclear Energy Systems

[ABS-27]

Analysis of Uranium-Plutonium Based Fuel Performance in Gas-Cooled Fast Reactors (GFR) With Various Core Geometries Using OpenMC

Ratna Dewi Syarifah (a), Ahmad Haris Rasidi (a), Nurul Kholifah (a), Fajri Prasetya (b), Ahmad Muzaki Mabruri (b), Mohammad Ali Shafii (c*), Dian Fitriyani (c)*

a) Department of Physics, Universitas Jember
b) Department of Nuclear Science and Engineering, Institut Teknologi Bandung
c) Department of Physics, Universitas Andalas
*Corresponding Author e-mail: mashafii@sci.unand.ac.id

Abstract

Analysis of thorium-based fuel performance in Gas-cooled Fast Reactors (GFR) with various core geometries has been conducted. GFR is a type of Generation-4 nuclear reactor with a fast neutron spectrum and helium gas coolant, enabling it to operate at high temperatures and effectively facilitate the breeding of U-233 from Th-232. This research aims to analyze the performance of thorium-based fuel in GFR core geometries: balanced, tall, and pancake shapes. The performance analysis includes neutronics aspects such as criticality, neutron behavior, and material transmutation resulting from burn-up. The analysis was conducted using the Monte Carlo approach with the OpenMC code, which is an open-source tool for neutron and photon transport analysis. The research findings indicate that the balanced geometry variation yields the best performance, requiring less fuel to achieve criticality over the same period, providing a more uniform neutron flux distribution, and achieving the most optimal breeding ratio. These results highlight the potential of the balanced design as the preferred choice to enhance efficiency and reliability in GFR reactors.

Keywords: GFR, Thorium fuel, Variation Core Geometries

Topic: Innovative Nuclear Energy Systems

[ABS-30]
ANALYSIS OF POWER PEAKING FACTOR REACTOR CORE TRIGA 2000 BANDUNG USING OPENMC PROGRAM

Fajri Prasetya (a), Nuri Trianti (b), Ratna Dewi Syarifah (c), Zaki Su^ud (a), Efrizon Umar (b)*

(a) Department of Nuclear Science and Engineering, Institut Teknologi Bandung, Ganesha Street No. 10, Siliwangi, Coblong Subdistrict, Bandung City, West Java, Indonesia, 40132
*prasfajri@gmail.com

(b) Nuclear Reactor Technology, National Research and Innovation Agency (BRIN), Tamansari Street No. 71, Siliwangi, Coblong Subdistrict, Bandung City, West Java, Indonesia, 40132

(c) Department of Physics, Universitas Jember, Kalimantan Street, No 37, Krajan Timur, Sumbersari Subdistrict, Jember, East Java, Indonesia, 68121

Abstract

Currently, the TRIGA reactor has been reshuffling several fuel elements in order to maintain its criticality during operation. The reshuffling includes the filling of heterogeneous fuel elements with an enrichment of 8.5%wt- 12%wt- 20%wt for the isotope U-235. The heterogeneity in the preparation of fuel elements has the opportunity to produce a fairly high PPF (Power Peaking Factor) value. This is the main focus that needs to be discussed in this study. The purpose of the study is to conduct a more comprehensive neutronic analysis of fuel elements by calculating the criticality value (k-eff) and PPF which are then compared with the results of the reference calculation (MCNP). The results of the validation of OpenMC and MCNP k-eff values showed accurate values with an error percentage (%&#8710-k/k) of < 1%, namely 0.1159% (60% of control rod withdrawal) and 0.1988% (100% control rod withdrawal). The results of the PPF calculation for the axial direction (APF) show a similar trend pattern for each variation of power used. In the middle position, the active fuel element has the highest PPF value than other regions. This shows that the fission rate in the central area of the fuel element is very dominant. The power used in this calculation is varied in the range of 223-640 kW because it still meets the safety limits of the current TRIGA 2000 Bandung reactor operation.

Keywords: K-eff- OpenMC- PPF

Topic: Innovative Nuclear Energy Systems

[ABS-31]
Analysis of Criticality and Fissile Material Production in Pebble Fuel of HTR-10 Using OpenMC Code

Ahmad Muzaki Mabruri (a), Nuri Trianti (b), Zaki Suud (a,c), Efrizon Umar (b), Ratna Dewi Syarifah (d)*

a) Department of Nuclear Science and Engineering. Bandung Institute of Technology, Bandung, Indonesia
*amuzaki073@gmail.com

b) Nuclear Reactor Technology, National Research and Innovation Agency (BRIN), Bandung, Indonesia

c) Department of Physics, Bandung Institute of Technology, Bandung, Indonesia

d) Department of Physics, University of Jember, Jember, Indonesia

Abstract

Analysis of Criticality and Fissile Material Production in Pebble Fuel of HTR-10 Using OpenMC Code has been conducted. Pebble fuel is a type of nuclear reactor fuel consisting of TRISO particles randomly dispersed within a kernel and surrounded by several layers, forming a unified pebble. The analysis was performed using Monte Carlo calculations with OpenMC. The pebble fuel model is assumed to operate at a constant power of 0.54 kW and 4.00 kW for 12 months, representing the average operating power of the HTR-10 design and the maximum operating power for a single pebble. Calculation results with various nuclear data libraries indicate that at maximum power, the pebble can only last for 7 months. Meanwhile, at 0.54 kW power, the pebble can maintain criticality until the end of the burn-up period, with the production of fissile plutonium material amounting to 0.8% of the total U-238 present at the beginning of the burn-up. This shows that the pebble fuel design has good proliferation resistance and is able to maintain criticality longer under actual operating conditions

Keywords: HTR-10- Pebble fuel- Proliferation- Triso.

Topic: Innovative Nuclear Energy Systems

[ABS-32]
OPTIMIZATION OF ACTIVE CORE DESIGN ON NEUTRONIC ASPECTS FOR MOLTEN SALT REACTOR (MSR) FUJI-12

Hilarius Adiwarna(b), Cici Wulandari(a,b), Marisa Variastuti(a,b), Syeilendra Pramuditya(a,b), Sidik Permana(a,b)*

a)Nuclear Physics & Biophysics Research Division, Department of Physics, Faculty of Mathematics and Natural Sciences, Institut Teknologi, Bandung, Indonesia
b)Department of Physics, Faculty of Mathematics and Natural Sciences, Institut Teknologi, Bandung, Indonesia
*psidik@itb.ac.id

Abstract

The Sustainable Development Goals (SDGs) are aided by nuclear power plants (NPPs), a form of power generator that generates economical, clean, and climate-friendly electricity. To find appropriate parameters for the SRAC-2006 and JENDL-4.0 modules with PIJ module for cell calculation and CITATION module for core calculation. this study is simulated for MSR FUJI-12 as one of the fourth generation of reactor. This study continued to optimize the core reactor to achieve a Conversion Ratio (CR) approaching 1 and a Power Peaking Factor (PPF) approaching 2. Variations in the number of core regions-two and three regions-with different fuel percentages were explored. By variating the composition of fuel and moderator or Moderator to Fuel Ratio (MFR) for each region, it was found that the MSR FUJI-12 with three regions and fuel percentages of 50%-30%-40% achieved a CR of 0.962 and a PPF of 2.579. However, this variation impacted the life cycle of the reactor from its adjustment on fuel and moderator to only 1400 days.This limited lifespan could be addressed with the concept of online refueling, applicable to MSR technology.

Keywords: MFR, MSR FUJI-12, Protactinium, U-233 F4

Topic: Innovative Nuclear Energy Systems

[ABS-36]
Computational Fluid Dynamics for Irregular Pentagon Natural Circulation Loop

Rahmad Dahana Saputra (a), Sidik Permana (a,b), Duwi Hariyanto (c), Syeilendra Pramuditya (a,b), Syaiful Bakhri (d)*

(a)Nuclear Physics and Biophysics Research Division, Physics Department, Faculty of Mathematics and Natural Science, Bandung Institute of Technology, Jalan Ganesha 10 Bandung 40132, Indonesia
(b)Nuclear Science and Engineering Department, Faculty of Mathematics and Natural Science, Bandung Institute of Technology, Jalan Ganesha 10 Bandung 40132, Indonesia
(c) Industrial Technology Faculty, Institut Teknologi Sumatera, Jl. Terusan Ryacudu, Lampung, Indonesia
(d) Research Center for Nuclear Material and Radioactive Waste Technology (PRTBNLR), Research Organization for Nuclear Energy, National Research and Innovation Agency (BRIN), South Tangerang 15314, Indonesia
*psidik@itb.ac.id

Abstract

Electricity is an important need for everyday life. Nuclear Reactor is a power plant that have low CO2 emission and the energy is available at any time. Generation IV nuclear reactor have a passive safety system that allow it to run without external power source or any human intervention. One of the designs of generation IV nuclear reactor is Molten Salt Reactor (MSR) that have natural circulation as its passive safety system. A study of irregular pentagon natural circulation loop using water as its coolant has been done previously. In this study the natural circulation loop is simulated using ANSYS Fluent. A mesh independence study is done for obtaining mesh with accurate result but at effective computational cost. After that some flow parameters is obtained, namely cold leg temperature, hot leg temperature, and the flow Reynolds number with a value of 34.29 C, 39.93 C, and 86.8-132.6 respectively. Those parameters is compared to a reference study and an error less than 3.5% are obtained. Those result mean that this simulation model can be used for further study.

Keywords: Natural Circulation, Ansys Fluent, Pentagon Loop

Topic: Innovative Nuclear Energy Systems

[ABS-37]
Study of Prospecting NPP Cogeneration Potential as District Heating in Indonesia

Muksin Aji Setiawan(a,b), Sidik Permana (a), Topan Setiadipura (b)*

(a) Nuclear Science and Engineering, Faculty of Mathematics and Natural Sciences, Bandung Institute of Technology, Basic Science Center A Building, Jalan Ganesa 10, Bandung, 40132, Indonesia
(b) Center for Nuclear Reactor Technology, Research Organization of Nuclear Energy, National Research and Innovation Agency, Jl. Kawasan PUSPIPTEK Serpong building 80, Tangerang Selatan, Banten, 15310, Indonesia

Abstract

Metropolitan City, such Jakarta as always linked to large population and huge energy consumption challenges. During working hours, up to 4 million people move around this city. These phenomena make energy demand pattern quiet slightly different compare to the satellite. The demand for energy spikes, especially in the summer months when air conditioning use surges, necessitating a reliable electricity supply. Currently reliant on coal as its primary energy source, Jakarta faces issues when coal availability is disrupted or economically unfeasible. This dependence becomes problematic when coal supply is disrupted or when coal is no longer economically viable. The use of coal also has negative long-term environmental impacts. To support the government^s goal of achieving net-zero carbon emissions, there is a need to increase the use of renewable energy sources. Nuclear energy emerges as a potential option for adoption in Indonesia due to its substantial production capacity. Utilizing Nuclear Power Plants (NPPs) cogeneration aspect for district heating and cooling can reduce dependence on fossil fuels, particularly for cooling needs. This will lead to a reduction in air pollution resulting from fossil fuel combustion, resulting in improved air quality in Jakarta. This study evaluates the potential utilization of NPPs in the context of large cities in Indonesia, taking into consideration factors such as geography, population, economy, and environmental impact. Consequently, Jakarta is expected to move toward cleaner and more sustainable air quality.

Keywords: Energy Consumption- Energy Demand- Cogeneration- District Heating

Topic: Innovative Nuclear Energy Systems

[ABS-42]
INFLUENCE OF PIPE DIAMETER IN ADVANCED REACTORS NATURAL CIRCULATION SYSTEM

Amna Yasya Mubarok, (a), Sidik Permana (a), Dwi Irwanto (a)*

a) Faculty of Mathematics and Natural Sciences, Bandung Institute of Technology
Jalan Ganesha 10, Bandung 40132, Indonesia
*amnayasya@gmail.com

Abstract

4th gen reactors are focused on developing 4 aspects- highly economical, enhanced safety, minimal waste, and proliferation resistance. 4th gen reactor using natural circulation on coolant as an enhanced safety system. Natural circulation makes sure coolant is still able to release heat since the coolant is still flowing. This research is done to study the natural circulation using COMSOL Multiphysics and experiment. Model used in this research is a reactor coolant which is 100 cm in height and 50 cm in width. In this research the diameter of the pipe varied at 1/2, 1/4, and 1 inch. Simulation and experiment able to perform natural circulation. The result shows changes in heat dissipation on each configuration. Bigger pipes have more water volume, meanwhile the heater and cooler give and take the same amount of heat, the result is bigger pipes have higher difference on water lowest and highest temperature.

Keywords: Natural circulation, Pipe diameter, Reactors

Topic: Innovative Nuclear Energy Systems

[ABS-44]
Akademik Lomonosov FPU: operating experience

Ivan Renev (a), Alexander Renev (a), Alexander Tsibulya (a)*

Rosatom Energy Projects JSC, Leninskaya Sloboda 26 bld. 5, Moscow, Russian Federation

Abstract

Over four years of operation, the world^s only floating nuclear power plant Akademik Lomonosov generated more than 860 million kWh of electricity. Moreover, every year the share of carbon-neutral electricity from floating nuclear power plants in the overall energy balance of the region is growing steadily. For the entire 2023, the operation of the floating nuclear power plant made it possible to generate 28.5% of all electricity produced in Chukotka. In the future, after 2025, when the Bilibino NPP is shut down and the restoration of the floating nuclear power plant is completed, the station load will reach peak values and approach the maximum power of 70 MW, which will have a positive impact on reducing power tariffs. The purpose of the report is to familiarize students with the experience of operating the world^s only floating nuclear power plant Akademik Lomonosov.

Keywords: Akademik Lomonosov, SMR, FNPP, FPU

Topic: Innovative Nuclear Energy Systems

[ABS-45]
Civil marine reactor evolution. From OK-150 to RITM-200

Ivan Renev (a), Alexander Renev (a), Alexander Tsibulya (a)*

Rosatom Energy Projects JSC, Leninskaya Sloboda 26 bld. 5, Moscow, Russian Federation

Abstract

The RITM-200 reactor unit (RP) with a capacity of 175 MW is a fundamentally new step in the development of the icebreaker fleet. The reactor plant has a unique energy-efficient integral layout, which ensures that the main equipment is placed directly inside the steam generating unit housing. This allows to classify the RITM-200 reactor as the 3rd generation of civil ship-class reactor unit. In contrast to the 2nd generation of reactor plants (OK-900 and KLT-40), in the 3rd generation a transition was made from a block layout to an integral one. New engineering solutions made it possible to implement the strict restrictions on weight and size characteristics laid down in the technical specifications. Thanks to the integrated layout, the RITM-200 reactor turned out to be twice lighter, one and a half times more compact and 25 MW more powerful than the KLT-40 reactor units. The RITM-200 nuclear power plant is capable of ensuring more economical operation of the new nuclear icebreaker compared to existing ones with increased reliability and safety. The improvement of the reactor plant proceeded in the following directions: Reducing the composition of equipment and its weight and size characteristics, Increased maneuverability, Increasing equipment life, Reduction of own energy consumption.

Keywords: SMR, FPU, RITM, KLT-40

Topic: Innovative Nuclear Energy Systems

[ABS-47]
Enhancing Safety in Molten Salt Reactors: Understanding How Design Choices Affect Freeze Plug Performance

Muhammad Ilham

Research group for Reactor Physics and Thermal Hydraulics Technology, Nuclear Science and Engineering Center, JAEA Tokai.

Abstract

Molten salt reactors (MSRs) represent a promising advancement in nuclear technology, offering potential benefits such as improved safety and efficiency. A critical component of MSRs is the freeze plug, which acts as a safety mechanism installed between the reactor vessel and the drain tank. During an emergency, external cooling stops, and the salt inside the freeze plug melts, allowing the fuel salt to be discharged. The melting time (opening time) and the drain time, which are key factors in ensuring MSR passive safety, depend on the drain tube design and the initial shape of the frozen salt formed in the drain tube. Given this, a simulation of the solidification process was first performed, followed by melting and drain time approximation. Systematic numerical simulations were conducted to explore the effects of wall thickness, inner diameter, and tube inclination on the freeze plug used in MSRs. Furthermore, a novel jacket design for melting acceleration and a conical tube for plug structural integrity were proposed. It was found that in the solidification process, the tube design significantly impacted the equilibrium shape of the frozen salt. Melting simulations showed that a small tube shortened the opening time. However, it took a long time to drain the liquid salt from the reactor core into the drain tank after opening. The results indicated that all aspects of the solidification, melting, and drainage processes should be sufficiently understood to utilize the freeze plug as an effective passive safety system in MSRs. Accordingly, a simplified analytical model was developed for a rough estimation of the opening time that reasonably agreed with the full simulation results.

Keywords: Molten salt reactor, Freeze plug, Solidification, Melting, Opening time, Drain time, Inclination, Numerical simulation

Topic: Innovative Nuclear Energy Systems

[ABS-55]
Neutronic & Dynamic Analysis of TMSR-500 Reactor Core

Alessandro Widjati (a), Sidik Permana (a,b,c), Syeilendra Pramuditya (a,b,c), Cici Wulandari (a,b,c)*

a) Department of Physics, Faculty of Mathematics and Natural Science, Bandung Institute of Technology, Bandung 40132 Indonesia
b) Department of Nuclear Science and Engineering, Faculty of Mathematics and Natural Science,Bandung Institute of Technology, Bandung 40132 Indonesia
c) Nuclear Physics and Biophysics Research Group, Faculty of Mathematics and Natural Science,Bandung Institute of Technology, Bandung 40132 Indonesia
alswidjati@gmail.com

Abstract

TMSR-500 designed by ThorCon is a generation IV nuclear reactor. This reactor has potential to provide energy that are sustainable, safe, and low in CO2 emissions. Therefore, study about TMSR-500 will have a significant impact on the development of nuclear energy. Neutronic and dynamic analysis will be conducted in this study. For neutronic analysis, influence of reactor design parameters against neutronic aspects will be evaluated. The design parameters evlauated are fuel composisiton, fuel volume fraction, coolant quantity, and material temperature, while the neutronic aspects are effective multiplication factor (keff), conversion ratio (CR), and delayed neutron fraction. For dynamic analysis, response of thermal power, fuel salt temperature, and graphite temperature to positive reactivity insertion will be evaluated. The method used to obtain the neutronic aspects of the reactor is neutronic calculation by computer simulation using the SRAC2006 code system and the JENDL-4.0 nuclear library. While to obtain the dynamic response, dynamic calculation is performed by solving the reactor point kinetics equation and the reactor heat transfer equation numerically. The study shows that the concentration of fissile material is directly proportional to keff and delayed neutron fraction, while the concentration of fertile material is directly proportional to CR. The fuel volume fraction will be inversely proportional to the keff at the beginning of operation, but will be directly proportional to CR and delayed neutron fraction. TMSR-500 requires a minimum U-235 concentration of 1.28% with an optimal fuel volume fraction of 27%. TMSR-500 has a negative coolant nuclide density and temperature reactivity coefficient, while also has a good safety response to positive step reactivity insertion in the 50 pcm to 500 pcm range because the temperature of the materials in those dynamic conditions are still within a safe range.

Keywords: coolant, dynamic, fuel composition, fuel volume fraction, material temperature, neutronic, reactivity insertion, TMSR
Topic: Innovative Nuclear Energy Systems

[ABS-57]
Parameter Justification to Validate Neutronic Calculation Using MCNP6 for CANDLE Reactor Case

Maryam Afifah(a), Zaki Su'ud(a), Nuri Trianti(b) , Dwi Irwanto(a)*

a) Department of Nuclear Science and Engineering, Faculty of Mathematics and Natural Sciences, Institut Teknologi Bandung, Indonesia 40132
* 34922002@mahasiswa.itb.ac.id

b) Research Centre for Nuclear Reactor Technology, Research Organization for Nuclear Energy, National Research and Innovation Agency, PUSPIPTEK, Indonesia 15314

Abstract

This paper presents the survey parameters to justify several factors with the minimum parameters that should be applied to validate neutronic calculation for CANDLE reactor case. In particular, we investigate the effect of neutron number history, inactive-active cycle, the burnup time step length and the recommendation of those factors as a way to improve the computing efficiency using MCNP6 code. This study presented the minimum number of neutron history and the total cycle to minimize the oscillation. However, the keff value with larger length step has a difference of 1% compared to smaller length step application. In this case we explore the methodology that used in MCNP6 to calculate burnup to make sure the numerically stable method was applied. Lastly, this study concluded some recommendations that should be applied for complex core geometry to achieve accuracy and efficiency for neutronic calculation using Monte Carlo method especially using MCNP6 for CANDLE reactor case.

Keywords: CANDLE reactor, Monte Carlo, Depletion, time step length, MCNP6

Topic: Innovative Nuclear Energy Systems

[ABS-58]
Effect of the Number of Freeze Valve Channels on Core Discharge Rate and Pressure in Core and Drain Tank

Ahmad Muzaki Mabruri (a), Amna Yasya Mubarok (a), Bagja M B Kertasafari (a), Asril Pramutadi Andi Mustari (a), Sidik Permana (a)*

a) Department of Nuclear Science and Engineering. Bandung Institute of Technology, Bandung, Indonesia *amuzaki073@gmail.com

Abstract

Analysis of the Influence of the Number of Freeze Valve Channels on Core Discharge Rate and Pressure in Core and Drain Tank has been conducted. This research employed a 2D freeze valve design representation using the Moving Particle Semi-implicit Method. This method utilizes the behavior and movement of particles within a predefined geometry, influenced by various physical parameters within the system. The aim of this study is to observe velocity profiles and pressure profiles during the core discharge process with variations in freeze valve geometries V1, V2, and V3. The results indicate that the core discharge times for V1, V2, and V3 are 4.75 s, 6.40 s, and 6.15 s respectively. V3 exhibits a lower average system pressure compared to V1 and V3, suggesting that V3 is more effective in managing pressure during the core discharge process.

Keywords: Core Discharge- Drain Tank- Freeze Valve- MPS- MSR

Topic: Innovative Nuclear Energy Systems

[ABS-63]
Nuclear Energy: A Strategic Path Towards Indonesia^s Sustainable Energy Future

Nining Yuningsih(a), Indah Rosidah Maemunah(b), Fisca Dian Utami(c)

a)SMAN 5 Sukabumi, Indonesia
b)National Research and Innovation Agency, Indonesia
c)Politeknik Transportasi Sungai, Danau, dan Penyeberangan Palembang, Palembang, Indonesia

Abstract

Indonesia^s evolving energy landscape highlights nuclear energy as a viable solution for future energy demands. Nuclear power offers a reliable and sustainable alternative, crucial for balancing increasing urban energy consumption and variable rural needs. This study underscores nuclear energy^s potential role in meeting Indonesia^s future energy requirements, stressing the necessity of strategic energy policy planning to align with these projections. Comparing energy sources like oil, gas, and nuclear power plants reveals nuclear energy^s robustness as an alternative. A 1000 MWth nuclear power plant can fulfill approximately 46-53% of the energy demands in Nusa Tenggara, Maluku, and Papua. Moreover, nuclear power boasts minimal carbon emissions, approximately 0.029 kg/kWh, positioning it favorably in achieving Indonesia^s 2060 energy targets and fostering a sustainable energy future.

Keywords: Energy, Emission, Nuclear, Sustainable

Topic: Innovative Nuclear Energy Systems

[ABS-73]
Fundamental Principles in the Development of MSR Technology

Indarta Kuncoro Aji (a)*

a) PT Kakiatna Enjiniring, District 8 - Treasury Tower 7th floor, Jl. Jend. Sudirman Lot 28, Kav 52-53 SCBD, Jakarta, Indonesia

**indarta@kakiatna.co.id*

Abstract

The advancement of Molten Salt Reactor (MSR) technology requires a comprehensive understanding of the fundamental principles of transport phenomena in molten salts. This paper explores critical aspects of these principles, which are essential for optimizing reactor performance and enhancing safety, with particular emphasis on the solid-liquid interface during melting and solidification processes (such as mushy regions, latent heat, etc.), as well as natural convection. By elucidating these transport phenomena, we aim to improve the design and operational reliability of MSR systems. A better understanding of these principles is crucial for advancing MSR technology and ensuring its effective application in sustainable energy production.

Keywords: molten salt, solid-liquid interface, natural circulation

Topic: Innovative Nuclear Energy Systems

[ABS-76]
PERFORMANCE ANALYSIS ON THORIUM FUEL BASED SMALL MODULAR REACTOR (SMR)

Amila Amatullah (a), Sidik Permana (b,c), Dwi Irwanto (b,c), Akfiny Hasdi Aimon (c), Cici Wulandari (b,c)*

a) Nuclear Laboratory in Nuclear Science and Engineering, Faculty of Mathematics and Natural Sciences, Bandung Institute of Technology Ganesha No. 10 Bandung, Indonesia
*amilaamatullah@gmail.com
b) Graduate Program in Nuclear Science and Engineering, Faculty of Mathematics and Natural Sciences, Bandung Institute of Technology Ganesha No. 10 Bandung, Indonesia
c) Graduate Program in Physics, Faculty of Mathematics and Natural Sciences, Bandung Institute of Technology Ganesha No. 10 Bandung, Indonesia

Abstract

Small Modular Reactors (SMRs) and micro-reactors show some advantages smaller size and power, accessibility to remote regions, and reduced land requirements. Thorium as an alternative fuel to enhance SMR performance and safety gives some interesting feature such more abundance of resource as well as its superior neutron characteristics and potential for higher fuel efficiency. The study aims to compare uranium and thorium fuel cycles based on small modular integral PWR with thorium base utilization and its comparison with uranium and its combination with additional burnable poison. Initial neutron multiplication factor obtains 1.134 for thorium cycle, 1.129 for uranium fuel and 1.058, uranium-thorium cycles. In addition, Conversion ratio has been analyzed that thorium cycle have a higher CR (Conversion Ratio). with an average CR of 0.765, compared 0.692 for uranium cycles. Furthermore, thorium fuel cycle has significantly lower production of plutonium isotopes compared to other fuel cycles. The study show high power peaking factor (PPF) of 1.923 at the Beginning of Cycle (BOC), requiring further research for mitigation although its still less than 2. Analysis on performance comparisons at different power levels necessitate modifications in terms of geometry, operational duration, or fissile nuclide enrichment has been analyzed for optimization process. This research shows the potential of thorium-based fuel utilization in SMRs, by improving fuel efficiency and reactor safety.

Keywords: SMR- Conversion ratio- Neutron multiplication factor- Conversion Ratio- Thorium to Submit This Sample Abstract

Topic: Innovative Nuclear Energy Systems

[ABS-77]
Evaluating 10 MWt Pebble Bed Reactor with Different Nuclear Data Libraries and Monte Carlo-based code OpenMC

Kharisma A.D. Rahmany(a), Dwi Irwanto(a,b,c), Syaiful Bakhri(d),*

a)Department of physics, Faculty of Mathematics and Natural Science, Bandung Institute of Technology, Jl. Ganesa No.10, Lb.Siliwangi, Coblong District, Bandung City, West Java 40132,Indonesia
b)Department of Nuclear Science and Engineering, Faculty of Mathematics and Natural Science, Bandung Institute of Technology, Jl. Ganesa No.10, Lb.Siliwangi, Coblong District, Bandung City, West Java 40132,Indonesia
c)Nuclear Physics and Biophysics Research Group, Department of Physics, Faculty of Mathematics and Natural Sciences, Bandung Institute of Technology, Jl. Ganesa No.10, Lb.Siliwangi, Coblong District, Bandung City, West Java 40132, Indonesia
d) Research Center for Nuclear Material and Radioactive Waste Technology (PRTBNLR), Research Organization for Nuclear Energy, National Research and Innovation Agency (BRIN), South Tangerang 15314, Indonesia
dirwanto@itb.ac.id

Abstract

HTGR, as a gas-cooled VHTR reactor, is one of the reactors that has been widely researched and developed. It is even considered to have mature technology. Based on the fuel, HTGR has a pebble bed type where this system could perform online refuelling. This study aims to determine the neutronic aspects of the 10 MWt Pebble Bed Reactor, which is based on HTR-10 design, with different nuclear data libraries. The analysis was conducted by reviewing the simulation results of the effective multiplication with JENDFL 3.3, JENDL 5.0, ENDF/B-VII.I, and ENDF/B-VIII.0 nuclear data. The simulation uses the Monte Carlo code, OpenMC, by simulating the entire reactor in the initial criticality state. This study investigates how different nuclear data impact the effective multiplication factor, with a detailed analysis focusing on neutronic parameters. The findings aim to offer a more comprehensive understanding of the development of the Pebble Bed reactor, contributing to advancing nuclear reactor design and safety.

Keywords: Pebble Bed Reactor, Nuclear Data Library, Effective Multiplication Factor, Monte Carlo Method

Topic: Innovative Nuclear Energy Systems

[ABS-78]
Neutronic and Thermal Hydraulic Analysis of Gas Cooled Fast Reactor in Indonesia - An Overview

Kevin Wijaya(a) and Dwi Irwanto(a,b)*

a) Nuclear Physics and Biophysics Research Division, Physics Department, Faculty of Mathematics and Natural Science, Bandung Institute of Technology, Jalan Ganesha 10 Bandung 40132, Indonesia.
b) Nuclear Science and Engineering Department, Faculty of Mathematics and Natural Science, Bandung Institute of Technology, Jalan Ganesha 10 Bandung 40132, Indonesia.

Abstract

Gas-cooled Fast Reactor (GFR) is one of the fourth-generation reactors that is still being developed. The GFR system combines the advantages of a fast spectrum system for long-term sustainability of uranium resources and waste reduction (through fuel reprocessing and long-lived actinide fission) with a high-temperature system (industrial use and high thermal cycle efficiency). Several types of reactors, research methods have been carried out such as the type of output power produced, fuel variations, fuel composition variations, ring geometry variations, and core configuration variations have been carried out. Research has been carried out related to the thermal hydraulic analysis of GFR using various numerical methods. This article discusses the results of several neutronic and thermal hydraulic analyzes from research that has been carried out and discusses things that can be developed from previous GFR research.

Keywords: Burn up- fast reactor- GFR- hydraulic thermal- multiplication factor

Topic: Innovative Nuclear Energy Systems

[ABS-80]
Parametric Analysis of Molten Salt Natural Circulation Loop using Computational Fluid Dynamics

Rahmad Dahana Saputra (a), Sidik Permana (a,b), Duwi Hariyanto (c), Syeilendra Pramuditya (a,b), Syaiful Bakhri (d)*

(a)Nuclear Physics and Biophysics Research Division, Physics Department, Faculty of Mathematics and Natural Science, Bandung Institute of Technology, Jalan Ganesha 10 Bandung 40132, Indonesia
(b)Nuclear Science and Engineering Department, Faculty of Mathematics and Natural Science, Bandung Institute of Technology, Jalan Ganesha 10 Bandung 40132, Indonesia
(c)Industrial Technology Faculty, Institut Teknologi Sumatera, Jl. Terusan Ryacudu, Lampung, Indonesia
(d)Research Center for Nuclear Material and Radioactive Waste Technology (PRTBNLR), Research Organization for Nuclear Energy, National Research and Innovation Agency (BRIN), South Tangerang 15314, Indonesia

Abstract

Natural circulation is a form of passive safety in nuclear reactor. Natural circulation is important to prevent another accident like fukushima nuclear disaster that happen because the loss of power to the pump. An experiment on irregular pentagon natural circulation loop has done before using water as its coolant. In this study, the water coolant is replaced with molten salt to analyze its flow parameters. ANSYS Fluent which is a finite volume based computational fluid dynamics software is used to simulate the loop. First the loop heater and cooler temperature is varied to see the respons of the Reynolds number (Re), cold leg temperature (t1), and hot leg temperature (t3) for FLiBe, FLiNaK, and FNaB. For all molten salt, the Re is proportional to heater temperature when the cooler temperature is constant. For constant heater temperature condition, Re does not have a particular trends when the cooler temperature increase. t1 and t3 always increase when the heater or cooler temperature increase. FNaBe is added for second analysis that focuses on comparing Re, t1, and t3 for constant heater condition and constant cooler condition. It is observed that FNaB have the highest Re due to its low viscosity. A correlation on the trends of t1 and t3 to the velocity trends is also observed.

Keywords: Natural Circulation, Ansys Fluent, FLiBe, FLiNaK, FNaB, FNaBe

Topic: Innovative Nuclear Energy Systems

[ABS-2]
A Brief Comparison of Indonesian Prospective Nuclear Fuel Cycles

Dany Mulyana

Center of Nuclear Security Science and Policy Institute, Texas Engineering Experiment Station, Texas A&M University, College Station, TX, USA.

Research Center for Nuclear Reactor Technology, Badan Riset dan Inovasi Nasional, Jakarta, Indonesia.

Abstract

A brief comparison study on fuel cycles of four prospective Indonesian power reactors has been completed. The study oversees 160 MWt NuScale Small Modular Reactor (SMR), 10 MWt Reactor Daya Eksperimental (RDE-10), 40 MWt Pembangkit Listrik dan Uap Panas Industri (PeLUIt-40) and Thorcon^s TMSR-500. Both RDE-10 and PeLUIt-40 were fueled with 17wt% HALEU while the other two were fueled by 4.95wt% LEU, without thorium. The study was based on a set of data generated through Monte Carlo reactor physics simulation using OpenMC. The study found that NuScale SMR is the least 235U consumer by 0.84 g.MW-1 d-1 while the RDE-10 is the most 235U consumer by 0.97 g.MW-1 d-1. The simulation results show that TMSR-500 is the most uranium consumer by 1.41 g.MW-1 d-1 while the uprating of RDE-10 to PeLUIt-40 makes PeLUIt-40 as the least uranium consumer by 1.14 g.MW-1 d-1. As the least uranium consumer, both RDE-10 and PeLUIt-40 provide the highest attainable discharged fuel burnup of extractable energy of about 146 GWd/MTU. Assuming a one batch depletion scheme, NuScale SMR, RDE-10, PeLUIt-40 TMSR-500 has fuel consumption cycles in 2,210 days, 2140 days, 550 days, and 320 days, respectively.

Keywords: Fuel Cycle, Indonesian Nuclear Power Plant, NuScale SMR, PeLUIt-40,Reaktor Daya Eksperimental, TMSR-500

Topic: Innovative Separation and Fuel Cycles

[ABS-12]

Obtaining the Optimum TRU Layer Configuration for Transmutation in Molten Salt Reactor Fuel Channel

R. Andika Putra Dwijayanto (a), Andang Widi Harto (b), Azizul Khakim (a)

a)Research Centre for Nuclear Reactor Technology, Research Organisation for Nuclear Energy, National Research and Innovation Agency, 80 KST BJ Habibie, South Tangerang, Indonesia 15343
b)Department of Nuclear Engineering and Physics Engineering, Faculty of Engineering, Universitas Gadjah Mada, Jl. Grafika No. 2, Yogyakarta, Indonesia 55281

Abstract

Incineration of transuranic (TRU) elements can be performed in various types of nuclear reactor. Thermal molten salt reactor (MSR) can be utilised for TRU incineration, typically by blending TRU elements with the carrier salt. This method has issues in low trifluoride solubility and lower TRU burnup than fast spectrum MSR. This study proposes an innovative method of TRU incineration in MSR as a heterogeneous configuration by inserting tubular TRU layer into MSR fuel channel. To obtain optimum configuration, TRU layers comprised of reactor grade plutonium (RGPu) and minor actinide (MA) were inserted in an MSR fuel channel in central, middle, and outer configurations. Neutronic analysis was performed in the MSR fuel channel using MCNP6.2 code, and decay analysis was done using ORIGEN2.1. The analysed parameters include infinite multiplication factor, temperature coefficient of reactivity (TCR), transmutation efficiency (TE), and post-irradiation decay activity. It was observed that TRU layer with a volume fraction of 5% in a narrow fuel channel and outer configuration exhibit the optimum TE and improved the TCR. After 1460 days of irradiation, the TE was found to be 30.98% and lower radioactivity than unirradiated TRU after 400 years of decay. This result is promising as an initial stage prior to whole core calculation.

Keywords: SD-TMSR, MCNP6.2, TRU incineration, transmutation efficiency

Topic: Innovative Transmutation Systems

[ABS-3]
Computed Radiography Method for Non-Destructive Testing to Characterize Cooling Piping in Research Reactor G.A. Siwabessy

Jepri Sutanto, Baskan Hanurajie, Iwan Sumirat, Sidik Permana, Asril P.A Mustari

ITB
BRIN

Abstract

A radiographic test is one of the NDT methods used to verify the material (non-destructive testing). It utilizes ionizing radiation, such as X-rays or gamma rays, to create detailed images that reveal hidden flaws, defects, discontinuities, or anomalies. A radiographic test method commonly checks welding connections, fabrication, forging, and casting. Radiography NDT technique is used across diverse sectors, including manufacturing, aerospace, infrastructure, energy, and even research reactors or nuclear power plants (NPP). This study will discuss piping in a research reactor in South Tangerang, i.e., G.A. Siwabessy. Industrial digital radiography substitutes conventional film radiography and is replaced by imaging plate (IP). Parameters such as SNR and CNR and basic spatial resolution (SRb) must be considered and understood. Digital radiography is currently more widely used in addition to the short process and interpretation of results with up-to-date software developments. The linear indication is attached to the result of the interpretation. The linear indication^s orientation, size, and shape recommendations can provide valuable information to the nuclear energy research organization while replacing or substituting the piping. Image interpretation results were obtained to provide recommendations for improvements by standards and regulations. Other NDT methods are needed to compare the results.

Keywords: Non-destructive testing, industrial radiograph, x-ray, material, computed radiography

Topic: Material and Process for Innovative Energy Systems

[ABS-5]
Combined Gamma Irradiation and Sodium Lauryl Sulphate (SLS) Surfactant Methods as Modified Activated Carbon (g-SLS/AC) for Fe2+ Wastewater Adsorbent

Alvino Andreas Lumban Tobing, Sherina Massayu Putri, Angelica Isabella Christian, Sidik Permana, Asril Pramutadi, Dhita Ariyanti

Nuclear Chemical Engineering, Polytechnic Institute of Nuclear Technology, National Research and Innovation Agency, BRIN.

Department of Doctoral Nuclear Engineering, Faculty of Mathematics and Natural Sciences, Bandung Institute of Technology, Ganesha 10 Street, Bandung, 40132, Indonesia

Abstract

Heavy metal waste was one of big threat because of its negative impact on human health and environment. Heavy metals such as iron, Fe, was found in industrial wastewater, so it was necessary to reduce thus level or concentration. The use of activated carbon was one of the effective steps to carry out the sorption of Fe ions. Therefore, modification of activated carbon was carried out in this study to improve the quality of Fe2+ ion sorption on activated carbon. The methods of this research was devided on three steps. First stages was preparation of activated carbon form of sieving using a 70 mesh sieve and contacting the activated carbon with SLS surfactant for 24 hours. The second steps was exposured activated carbon modified using various dose gamma irradiation, then drying. The third steps was found out optimum contact time. Optimum contact time varied on 15, 30, 45, and 60 minutes. And the last stages was identification optimum dose irradiation. In this research, irradiation dose gamma used was 0- 10- 30- and 50 kGy. The results show that activated carbon that has been irradiated and contacted with SLS had much better sorption capacity than activated carbon it self. Capacity sorption of time contact 15 minutes was 0,8455 mg/g- 30 minutes was 0,9915 mg/g- 45 minutes was 0,9135 mg/g- and 60 minutes was 0,7305 mg/g. So, the optimum contact time was 30 minutes (0,9915 mg/g). Capacity sorption of dose gamma irradiation on 0 kGy was 0,66095 mg/g- 10 kGy was 0,86365 mg/g- 30 kGy was 0,800911 mg/g- and 50 kGy was 0,85125 mg/g. The conclusion of this research is modified activated carbon used gamma irradiation exposure and surfactant had better capacity sorption, by optimum time contact is 30 minutes (0,9915 mg/g) and 10 kGy (0,86365 mg/g) dose irradiation gamma. This research is expected to contribute to improving water quality.

Keywords: gamma irradiation, surfactant, adsorption, Fe2+ wastewater

Topic: Material and Process for Innovative Energy Systems

[ABS-6]

Methyl Ester Sulfonate (MES) Surfactant and Gamma Irradiation Modified Activated Carbon (g-MES/AC) as Cu2+ Wastewater Adsorbent

Sherina Massayu Putri (1), Alvino Andreas Lumban Tobing (2), Angelica Isabella Christian (3), Sidik Permana (4), Asril Pramutadi (5), Deni Swantomo (6), Dhita Ariyanti (7)

Nuclear Chemical Engineering, Politechnic Institute of Nuclear Technology, National Research and Innovation Agency, BRIN

Department of Doctoral Nuclear Engineering, Faculty of Mathematics and Natural Sciences, Bandung Institute of Technology, Ganesha 10 Street, Bandung, 40132, Indonesia

Abstract

Activated carbon is currently popular in advanced materials research and still continuosly improvement. One of activated carbon applications is on quality decrease problem solving of water, specially on presence of heavy metal waste, Cu2+ that caused by industrial activities. Therefore, the combination of activated carbon with surfactant Methyl Ester Sulfonate (MES) and gamma irradiation method is intended to adsorb Cu2+ cationic. In this research, activated carbon was treated by addition of surfactant Methyl Ester Sulfonate (MES) using various dose gamma irradiation 0- 10- 30- and 50 kGy. The first stages carried out preparation of activated carbon form of sieving using a 70 mesh sieve and contacting the activated carbon with MES surfactant for 24 hours. The second stages was exposured activated carbon modified using various dose gamma irradiation, then drying. The third stages was found out optimum contact time. Optimum contact time varied on 15, 30, 45, and 60 minutes. And the last stages was identification optimum dose irradiation. The results showed that Methyl Ester Sulfonate (MES) surfactant and gamma irradiation modified activated carbon adsorbs Cu2+ wastewater effectively. The adsorption capacity was 1,037 mg/gr on 15 minutes- 0,7075 mg/gr on 30 minutes- 0,762 mg/gr on 45 minutes- and the lower was 0,4725 mg/gr on 60 minutes. Optimum contact time of surfactant and gamma irradiation modified activated carbon to adsorbed Cu2+ was 15 minutes. The adsorption capacity was 0,15 mg/gram on 0 kGy- 0,803 mg/gram on 10 kGy- 0,098 mg/gram on 30 kGy- and 0,01 mg/gram on 50 kGy. So, the optimum dose irradiation of surfactant and gamma irradiation modified activated carbon to adsorbed Cu2+ was 10 kGy. From the result, it can be concluded that optimum condition of Methyl Ester Sulfonate (MES) surfactant and gamma irradiation modified activated carbon to adsorbed Cu2+ was 15 minutes and 10 kGy dose gamma irradiation. Hopefully, in the future, the use of irradiated activated carbon is expected to make clean water management more effective and efficient.

Keywords: gamma irradiation, surfactant, adsorption, Cu2+ wastewater
Topic: Material and Process for Innovative Energy Systems

[ABS-16]

ANALYSIS THE EFFECT OF GAMMA IRRADIATION DOSE VARIATION CO-60 ON THE RESULTS OF BIODIESEL RANDEMEN WITH A CONTENT OF 100% CRUDE PALM OIL (CPO) THROUGH THE USE OF COOKING OIL

Renaldy Bernardo Saragih, Wilada Nafi Royani, Dhita Ariyanti

Department of Nuclear Chemical Engineering, Polytechnic Institute of Nuclear Technology, National Research and Innovation Agency, Babarsari Street PO BOX 6101 YKBB Yogyakarta 55281, Indonesia

Directorate of Laboratory Management, Research Facilities, and Science and Technology Park, National Research and Innovation Agency, Yogyakarta 55281, Indonesia

Abstract

Biodiesel is an alternative fuel produced from renewable vegetable or animal sources. This fuel has the ability to replace or mix with conventional diesel, with the aim of reducing exhaust emissions that harm the environment. This study aims to analyze the level of influence of cobalt-60 gamma irradiation on the results of randemen in the manufacture of biodiesel based on cooking oil or Crude Palm Oil. Biodiesel is synthesized from cooking oil through two main processes. First, through esterification using methanol as a solvent in a mole ratio of 1:6, involving an acid catalyst at a concentration of 0.05% by weight of oil. Second, through transesterification with methanol in a mole ratio of 1: 6 with a base catalyst, with a catalyst concentration used of 1% by weight of oil. Characterization of biodiesel from synthesis was carried out using FTIR spectrophotometer analysis technique, which identified the presence of ether and methyl functional groups as characteristic. The results of the analysis showed the presence of C-O bonds at wavenumbers 1020/cm-1040/cm and C-H bonds at wavenumbers 2980.88 / cm. Furthermore, the test was conducted in accordance with the Indonesian national standard (SNI) 7182-2015. The test results showed the results of biodiesel randemen at irradiation doses of 35 Kgy has the largest volume of 108 mL with a density of 831 kg/m3, kinematic viscosity of 5,780 mm2 / s at 40 &#913-C, as well as the flame color reddish blue and slightly smoky, through gamma irradiation, biodiesel production can be increased in terms of production either with or without a mixture of other additional compounds that offer potential as an alternative fuel that is environmentally friendly and performs well.

Keywords: biodiesel, cooking oil, fuel, irradiation, gamma

Topic: Material and Process for Innovative Energy Systems

[ABS-21]
Comparison of Fission and Non-Fission Methods in Mo-99 Radioisotope Production

Dhiya Salma Salsabila (a), Anis Rohanda (b)

a)Master Program in Nuclear Science and Engineering Department, Faculty of Mathematics and Natural Science, Bandung Institute of Technology, Jalan Ganesha 10 Bandung 40132, Indonesia.
b)Research Center for Nuclear Reactor Technology, Research Organization for Nuclear Energy, National Research and Innovation Agency, Building No. 80, BJ Habibie Integrated Science Area, South Tangerang 15310, Indonesia

Abstract

The utilization of the G.A. Siwabessy Multipurpose Reactor (RSG-GAS) includes the production of radioisotopes. Among the radioisotopes produced by RSG-GAS is Mo-99. The Mo-99 radioisotope is used in the production of Technetium-99m (99mTc), which is used in diagnostic imaging in the medical field. There are two ways to produce the Mo-99 radioisotope. First, with a fission scheme using Low Enriched Uranium (LEU) in the form of UO2. Second, with a non-fission scheme by irradiating MoO3 samples. The aim of this study is to analyze the production results of radioisotopes using the fission scheme compared to the non-fission scheme using the ORIGEN 2.1 program. In this study, an MoO3 sample weighing 8.753 grams is compared to UO2 weighing 8.753 grams with the composition: U-235 1.508 grams, U-238 6.206 grams, and O-16 1.039 grams. Then, variations in the irradiation position in the reactor were made, namely at the Central Irradiation Position (CIP) at locations E7 and D6, as well as the Irradiation Position (IP) at point G7 with reactor power of 5, 15, and 30 MWt. The calculation results show that at CIP-D6 with a power of 30 MWt, the production result of the non-fission scheme is 7.05 Ci. Meanwhile, the production result of the fission scheme under the same conditions is 797.10 Ci. This is due to the better ability of UO2 to capture neutrons needed in nuclear reactions, as well as having more uranium isotopes that easily undergo neutron fission. Therefore, for Mo-99 production, it is recommended to use the fission scheme.

Keywords: RSG-GAS- Irradiation- Radioisotope- Mo-99

Topic: Material and Process for Innovative Energy Systems

[ABS-23]

Analysis of Fuel Salt Relocation from the Reactor Vessel to the Drain Tank in Molten Salt Reactor (MSR) with Moving Particle Semi-Implicit (MPS) Method

Aziz Satrio, Muhamad Daffa Fawwaz, Yulia Mifftah Huljanah, Asril Pramutadi Andi Mustari

Bandung Institute of Technology

Abstract

The Molten Salt Reactor (MSR) is a fourth-generation reactor whose technology has been proven by MSRE^s operations at ORNL. This reactor has a highly inherent safety system. Then, it has ability to extinguish multiple reactors and is carried out passively. Fuel salt in MSR can flow to drain tank by opening melt valve when temperature exceeds the limit. Understanding relocation of fuel salt is essential to improve reactor safety and performance. In this study, we present a particle-based approach using the Moving Particle Semi-Implicit (MPS) method, which represents fluid as particles that carry physical properties. We analysed the dynamics of several types of fuel salt at MSR. The fuel salt variations tested were LiF-NaF-KF, LiF-BeF2, KCl-MgCl2, and NaNO3-NaNO2-KNO3. In addition, temperature variations of 700K, 850K, and 950K were carried out on LiF-NaF-KF. The simulation uses the 2D geometry of MSRE reactor design. This simulation results show that NaNO3-NaNO2-KNO3 has movement with the greatest speed compared to other fuel salt. At 950K for LiF-NaF-KF has the greatest velocity compared to other variations. These results can be taken into account in selection of right saline fuel mixture and the understanding of its flow characteristics for safe and effective MSR operation, which is crucial for improving the safety of nuclear reactors.

Keywords: Drain Tank, Fuel Salt, MPS method, MSR

Topic: Material and Process for Innovative Energy Systems

[ABS-48]
Geometrical and Design Analysis of Betavoltaic Nuclear Battery using Monte Carlo Simulation

Khairul Basar, Rheza Pahlevi

Faculty of Mathematics and Natural Sciences, Institut Teknologi Bandung

Abstract

Betavoltaic nuclear batteries are one of the energy devices that have future use prospects. The operational efficiency of betavoltaic nuclear batteries is still quite small, which is caused by the geometrical design, or the semiconductor materials used. In this study, Monte Carlo simulation was used to determine the deposition energy of beta particles in semiconductor materials in betavoltaic nuclear battery systems. The performance of betavoltaic nuclear batteries was determined using the current-voltage characteristic curve on the semiconductor and beta particle parameters obtained from Monte Carlo simulations.

Keywords: betavoltaic, nuclear battery, Monte Carlo simulation

Topic: Material and Process for Innovative Energy Systems

[ABS-49]
Study Aging component reactor of Al6061T as lining reactor TRIGA 2000 in demin water contaminated By NaCl evaluation by outpile corrosion testing

Djoko Hadi Prajitno, Irawan Sugoro , Wahyu Ramdhani and Dani Gustaman Syarif

National Research and innovation Agency

Abstract

In order to increase safety, nuclear component of the reactor have got good performance on normal operational conditions. One of nuclear component of the reactor is lining tank reactor. In this study, the aging component reactor of Aluminum alloy Al6061 as lining tank reactor Triga 2000 in demin water has been evaluation by using electrochemical methods. The electrochemical testing were performed in demin water with and without the addition NaCl as contaminant at ambient temperature. The electrochemical technique used in this study were open circuit potential (OCP), Tafel polarization and electrochemical impedance spectroscopy (EIS). The results of open circuit potential tests showed that the corrosion potential (Ecorr) values of Al6061 T alloy are slightly different between in demin water with and without the addition NaCl. The corrosion potential Al6061 T alloy are higher in demin water without NaCl compare with NaCl addition. The are show that the interaction between demin water and the aluminum alloy have no significant effect on to its corrosion behavior. However, Tafel polarisation of aluminum alloy Al6061 T alloy in demin water with the absence of NaCl showed the lowest corrosion than aluminum alloy at condition in demin water with NaCl. Eis study shows that aluminum alloy Al6061 T alloy in demin water with the absence of NaCl showed the higher impedance than aluminum alloy in demin water with NaCl. The higher impedance indicated that the passive film of Al2O3 formed on the surface of aluminum alloy. In terms of corrosion rate values, aluminum alloy Al6061T showed the best corrosion resistance in demin water without NaCl compare Al 6061 T in demin water with NaCl.

Keywords: Please Just Try to SubmiAl6061 T ,corrosion, demin water, NaClt This Sample Abstract

Topic: Material and Process for Innovative Energy Systems

[ABS-59]
Microstructure Evaluation and Rietveld Analysis of Fe-Cr-Al based ODS Alloy using Arc Melt Casting Method

*Muhammad Angga Saputra(a)- Andriansyah(b)- Diene Noor Haerani(b)- Rohmad Salam(c)- Muhammad Fakhrudin(d)- Syahfandi Ahda(b)- Bernardus Bandriyana(b)- Tetty Kemala(a)- Nanda Shabrina(b)**

(a) Department of Chemistry, Faculty of Mathematics and Natural Sciences (FMIPA), IPB University

(b) Research Centre for Rector Nuclear Technology (PRTRN), Research and Organization for Nuclear Energy (ORTN), National Research and Innovation Agency of Indonesia (BRIN)

(c) Directorat for nuclear facility management (DPFK), National Research and Innovation Agency of Indonesia (BRIN)

(d) Research Centre for Advanced Material (PRMM), Research Organization for Nanotechnology and Material (ORNM), National Research and Innovation Agency of Indonesia (BRIN)

*nanda.shabrina@brin.go.id

Abstract

Fe-Cr-Al-based ODS (Oxide Dispersion Strengthened) alloy emerging as a promising candidate for fuel cladding applications. The ODS steel was fabricated by Arc Melt Casting as alternative method beside the common sintering method. The effect of yttria dispersion (0.5% and 3%) in Fe-Cr-Al matrix was investigated. Crystal structure, microstructure, element analysis and hardness were carried out using XRD (X-Ray Diffraction), Optical Microscopy, SEM-EDS (Scanning Electron Microscopy) and Vickers Hardness Test. Results shows by increasing 3% Y2O3, grain size area decrease 50% following by hardness number increase 23%. Crystal structure identification was analysis by Rietveld analysis using PAN-Analytical Highscore Software. In 0.5% Y2O3 addition, Alumunium-Iron found to be dominant beside Chromium Iron and Chromium. However, the analysis from EDS showed agglomeration of Al and Y2O3.

Keywords: ODS- Fe-Cr-Al- Y2O3- alloy- Arc melt casting

Topic: Material and Process for Innovative Energy Systems

[ABS-1]

Evaluation of nuclear reaction cross sections for optimization of production of the emerging diagnostic radionuclide 123I via proton and deuteron-particle induced transmutations

Lutfi Aditya Hasnowo(a,b)*

a)Polytechnic Institute of Nuclear Technology, National Research and Innovation Agency, Yogyakarta, 55281, Indonesia
b)School of Nuclear Science and Engineering, Tomsk Polytechnic University, Tomsk, 634050, Russia

Abstract

Proton and deuteron-particle induced reactions on 122,123,124Te targets were evaluated for the production of 123I. The literature data were compared with nuclear model calculations using the codes TALYS-2.0 and EMPIRE-3.2.3 (Malta). The statistically evaluated excitation functions were generated- therefrom the integral yields of the 123I were calculated. The amounts of the radioactive impurities were assessed.

Keywords: 122Te(d,n)123I, 123Te(d,2n)123I, 123Te(p,n)123I, 124Te(p,2n)123, Nuclear model calculations, Nuclear data evaluation, Thick target yields

Topic: Nuclear Data

[ABS-38]
Exploring of of Macro-Micro Minerals in Different Rice Species at Pandenglang Utilizing Neutron Activation Analysis Method

Sari Hasnah Dewi, Th. Rina Mulyaningsih, Saeful Yusuf, Alfian, Istanto, Siti Suprapti, Firda Amalia, Fatmawati and Ferly Hermana

Organization Research for Nuclear Energy - National Research and Innovation Agency

Abstract

Exploring of macro-micro minerals in different rice species produced in Pandeglang had been done using neutron activation analysis. Neutron activation analysis is a method using neutron particle which produced in nuclear reactor then be shoot into sample atom and then the sample is active and scattered Gamma Ray. Furthermore gamma ray detected by gamma detector. Kind of sample of rice have been studied were planted in city around Pandeglang city that are white rice Sobang, IR 12, IR 64, Cimanuk, Ciherang, Red Rice Huma, Red Rice, Black Sticky Rice Cimanuk, Sticky Rice Serang and Sticky Rice Karang Tanjung. Analysis showed samples contained macro mineral more than 100 mg/Kg, such as K, Ca, Mg, Na, and Cl. Contain of micro minerals in sample were in between 0.2 < x >30 mg/Kg : Zn, Cr, Co Br, Rb and Mn. Surprisingly all of those kind of rice have contained other mineral that not include in macro or micro minerals category such as Al, Cs and La in massive amount more than 5 mg/Kg. These minerals are literally not needed in human consumption and metabolism, it must be take into account into serious attention because of its toxicity. Evaluation of these elements is compared to sufficiency value of daily requirement RDA (Recommended Daily Acceptable). Among the concerns is the high content of many kind of metals in colored rice , especially Black Sticky Rice Cimanuk meanwhile for white rice species relatively contain low concentration of either micro-macro mineral and toxic element. In this study was discussed potential hazards for human while its deficiencies or excessive intake

Keywords: rice, macro-micro minerals, NAA

Topic: Nuclear Data

[ABS-74]
Development Module of Natural Nuclear Radiation Analysis for High School Level

Intan Fauziyyah, Sidik Permana, Zulfahmi, Ismail Humolungo, Adi rachmansyah, Dwi Irwanto, Nurhasan

Faculty of Mathematics and Natural Sciences, Institut Teknologi Bandung, Jalan Ganesha 10, Bandung, 40132, Indonesia

Abstract

Improving scientific literacy among high school students in Indonesia remains a challenge, prompting initiatives to enhance this skill. This research aims to develop a nuclear radiation practicum module for high school students and analyze environmental radiation dose rates. The study involves comparing radiation measurements using various tools and evaluating the impact of environmental background conditions on these measurements. It utilizes Cs-137 (Cesium 137) sources at different distances and with different shielding materials. The ADDIE approach-Analysis, Design, Development, Implementation, and Evaluation-guides the module development process.The study is limited to the development stage, focusing on creating models for the measurement process, mapping, and comparing tools and practicum materials. Radiation dose rate measurements were conducted at the Bandung Institute of Technology at 10 points with instrument heights of 0 cm and 100 cm from the ground, using GMC 500+ and Pocket Geiger tools. Results show that the GMC 500+ tool measured an average radiation dose rate of 0.1365 \pm 0.0160 \mu Sv/hour at 0 cm height and 0.1207\pm0.0163 \muSv/hour at 100 cm height. The Pocket Geiger tool recorded an average dose rate of 0.052\pm0.0154 \mu Sv/hour at 0 cm and 0.043\pm0.0176 \mu Sv/hour at 100 cm height. Measurements on different background surfaces (granite floors, paving blocks, soil, concrete, grass, and asphalt). Measurements using the Cs-137 source determine the effect of distance on the radiation source and shielding material, with distance variations from 0 cm to 100 cm and the use of shielding material stored between the source and the detectorThe final product is a validated nuclear radiation practicum module, which received positive feedback from material and media experts, peers, and high school physics teachers, indicating its readiness for implementation.

Keywords: Geiger Muller- Pocket Geiger- Environmental Radiation- Cesium 137, Practical Module

Topic: Nuclear Education and Social Aspects

[ABS-4]
Intrinsic Nuclear Security Attribute Analysis for Physical Protection System Development of Reaktor Daya Eksperimental

Dany Mulyana

Center of Nuclear Security Science and Policy Institute, Texas Engineering Experiment Station, Texas A&M University, College Station, TX, USA.

Abstract

A study on material theft analysis of fresh and spent fuels of Reaktor Daya Eksperimental (RDE) has been completed. This study covers an integrated analysis of nuclear safety, security, and safeguards. This study assumed that each pebble fuel contains about 7,223 tri-structural isotropic (TRISO) fuel kernel microspheres depleted up to 90 GWd/MTU. Both the TRISO and the graphite matrix of the fuel pebble was designed to contain fission products during and after irradiation. A set of Monte Carlo n-Particle (MCNP) neutronic simulations on the RDE^s fresh and spent fuel pebbles were completed to investigate the potential radiation exposure to the personnel, that can be a security barrier for RDE system. Taking only gamma ray into considerations, this study found that the fresh fuel pebble containing 5 g of UO2 enriched to 17 wt% of U-235 exposes radiation with a dose rate of only 0.38 microrem/h at a distance of 1 m away from the pebble. At the same distance, a 90 GWd/MTU spent fuel pebble cooled for one year resulted dose rate of about 1.0 rem/h. This study also found that a lead sphere shielding (density of 11.34 g/cm3) with a thickness of approx. 12.5 cm must cover the spent fuel pebble to reduce dose rate to 0.328 microrem/h.

Keywords: Nuclear Security, Physical Protection System, Pebble Bed Reactor, Reaktor Daya Eksperimental

Topic: Nuclear Nonproliferation Issues

[ABS-19]
Burnup Performance of modified CANDLE shuffling in axial direction on gas cooled fast reactors with UN-Th fuels

Jean Pierre Ndayiragije, Zaki Su'ud, Abdul Waris, Dwi Irwanto

Institut Teknologi Bandung

Abstract

The modified CANDLE (Constant Axial shape of Neutron flux, nuclide densities, and power shape During Life of Energy production) strategy shuffling in an axial direction has been successfully applied to gas cooled fast reactor. This paper investigated the utilization of natural uranium, enriched nitride and Thorium (238U 15N and 232Th) as fuel on 3000MWt reactor power and refuelling every 10 years of burnup cooled by helium gas. The reactor core is partitioned into ten equal-volume regions in the axial direction. Initially, a fuel input (238U 15N and 232Th) is placed in region 1. After ten years of burnup, a fuel in region 1 has been moved to region 2, then to region 3, and so on until the fuel in region 9 was moved to region 10. A fuel from region 10 has been got out. The neutronic computations were performed in two different ways utilizing the SRAC 2006 code and JENDL4.0 as a nuclear data library. Firstly, PIJ has been used for fuel cell calculations and secondly CITATION for reactor core calculation. The results show that the effective multiplication factor is greater than one, this indicates that the reactor is capable to operate through a burn-up period by employing 238U 15N and 232Th as fuel cycle input and the burnup level at the end life is about 30.29% HM.

Keywords: Thorium, Modified CANDLE, SRAC, Effective multiplication factor, burnup

Topic: Nuclear Nonproliferation Issues

[ABS-33]
Development of environmental and radiation monitoring system based on Private LoRaWAN Network

I Putu Susila (a), Gina Kusuma (a), Dian Fitri Atmoko (a), Arya Adhyaksa Waskita (b), Gede Arna Jude Saskara (c)*

a) Research Center for Nuclear Beam Analysis Technology, Research Organization for Nuclear Energy, National Research and Innovation Agency
KST BJ. Habibie, Tangerang Selatan 15310, Indonesia
*) i.putu.susila@brin.go.id
b) Research Center for Data and Information Sciences, Research Organization for Electronics and Informatics, National Research and Innovation Agency
KST Samaun Samadikun, Jalan Sangkuriang, Bandung 40135, Indonesia
c) Department of Information System, Faculty of Engineering and Vocational, Universitas Pendidikan Ganesha
Jalan Udayana 11, Singaraja 81116, Indonesia

Abstract

A scalable private LoRaWAN-based monitoring system has been developed for indoor and outdoor radiation monitoring around nuclear facility. The system consists of open hardware-based radiation detection devices, indoor and outdoor LoRaWAN gateways, an open-source LoRaWAN network server and database server for storing measurement data. In addition, a commercial air quality monitoring device has been integrated to the proposed system to ensure its scalability. The aims of the system are to evaluate and validate the feasibility of low-cost open hardware based radiation measurement devices, development of private LoRaWAN based infrastructure for the monitoring data pipeline, and provide testbed of wireless-based security for radiation detection system. The system has been deployed and tested to measure the coverage of LoRaWAN network, and performance measurement of real-time radiation and environmental data collection for indoor and outdoor measurement. The result demonstrates that the maximum coverage distance between gateway and end-device is ~600 m depend on terrain, and the continuous transmission of real-time measurement data i.e. radiation dose rate, environmental and air quality data can be done within a few seconds interval. In the future, security assessment of the proposed system need to be performed to ensure the integrity and reliability of the system when deployed as a measurement system for protecting environment and people from the risk of radiation exposure.

Keywords: Radiation safety and security- Radiation monitoring- Environmental monitoring- LoRaWAN

Topic: Nuclear Nonproliferation Issues

[ABS-75]

Assessment of Material Attractiveness in Light Water-Based Reactors: Evaluating Plutonium Isotope Composition and Burnup Impact on Proliferation Risk

Fungky Iqlima Nasyidiah(a), Sidik Permana(a,b), Dwi Irwanto(a,b), Akfiny Hasdi Aimon(c), Cici Wulandari(a,b)*

a)Department of Nuclear Science and Engineering, Faculty of Mathematics and Natural Sciences, Institut Teknologi Bandung, Indonesia
b)Nuclear Physics & Biophysics Research Division, Department of Physics, Faculty of Mathematics and Natural Sciences, Institut Teknologi Bandung, Indonesia
(c) Physics and Technology of Advanced Materials Research Division, Department of Physics, Faculty of Mathematics and Natural Sciences, Institut Teknologi Bandung, Indonesia
*psidik@itb.ac.id

Abstract

The assessment of nuclear safety and the potential risks associated with proliferation is a critical area of research in the context of nuclear energy utilization. This study focuses on evaluating the Material Attractiveness (ATTR) in several light water-based reactor designs, specifically NuScale, ESBWR, BWRX-300, and PWR, both during reactor operation and post-operation. The analysis employed Origen 2.2 for modeling radioactive decay and MCNP4C for assessing material attractiveness. The findings reveal an increase in the production of plutonium isotopes, with the exception of Pu-239, which decreases as a result of the fission. Initially, the Pu-240 composition across all four reactors is classified as Super-grade plutonium. However, with increased burnup, the Pu-240 composition transitions to a reactor-grade plutonium level. At the initial of irradiation, the ATTR values were determined to be 0.19 for the ESBWR, 0.20 for the PWR, 0.16 for the BWRX-300, and 0.21 for NuScale, categorizing them within the weapon-grade range. By the end of reactor operation, these values had significantly decreased, with ESBWR at 0.0125, PWR at 0.0148, BWRX-300 at 0.0111, and NuScale at 0.0133, placing them in the un-usable grade category. This marked reduction in ATTR values, correlated with increased burnup, indicates an effective decrease from weapon-grade to un-usable grade by the end of the reactor^s operational period. This trend is primarily influenced by the increased production of isotopes Pu-238, Pu-240, Pu-242, and the corresponding decrease in Pu-239, the primary fissile material.

Keywords: Non-proliferation- ATTR- Bare Critical Mass- Decay Heat- Spontaneous Fission Neutron- Neutron Prompt Life

Topic: Nuclear Nonproliferation Issues

[ABS-9]

Analysis of Radionuclide Content in Soil Samples Using the X-Ray Fluorescence Spectroscopy (XRF) Method

Ismail Humolungo (a), Sidik permana (a), Adi R. A Abdullah (a), Zulfahmi (a), Haryo Seno (b), Alan Maulana (b)*

a) Doctoral Program in Nuclear Engineering Department, Bandung Institute of Technology, Jl. Ganesha No. 10, Bandung, Indonesia.
b) Indonesias National Research and Inovation Agency. Jl. Tamansari No. 71, Bandung, Indonesia.

*ihumolungo41@gmail.com

Abstract

The high average value of natural gamma radiation dose rate in Mamuju Regency indicates that the uniqueness is related to the high concentration of radioactive elements of Uranium, Thorium, and Potassium deposited in the rock. In this study, XRF tests were carried out with EDXRF equipment on 5 soil samples from Mamuju Regency. Measurements were made on powder and pellet specimens. The results showed that the average concentration of radioactive elements uranium (U) was 712.17 ppm and 3581.45 for Thorium (Th), while for K was 9127.75 ppm. The thorium element appears to be more dominant with the Th/U ratio reaching 3.99 which is higher than the average Th/U ratio in the Earth^s crust of 3.70. A positive correlation between U and Th was obtained with a correlation coefficient value of 0.60, and a positive correlation was also shown by the relationship between U and Th, indicating the enrichment of U along with the enrichment of Th. Analysis of XRF data and dose rate data shows that there are two locations that show a very close relationship between the dose rate and the concentration of radioactive elements in the sample, while in other samples the influence of land use into gardens or highway strips affects the measured dose rate.

Keywords: XRF, Natural Radiation, Mamuju, Dose Rate

Topic: Radiation Physics

[ABS-13]
Comparative Study of Natural Background Radiation Between Mamuju Regency, West Sulawesi, and West Java

Zulfahmi (a), Sidik Permana(a,b), Adi R.A.Abdullah(a),Imam Ghazali Y(b), Ismail Humolongo(a),, Haryo Seno(c)

a)Doctoral Program in Nuclear Engineering, Bandung Institute of Technology, Jl. Ganesha No. 10, Bandung, Indonesia.
b)Master Program in Physic, Bandung Institute of Technology, Jl. Ganesha No. 10, Bandung, Indonesia.
c)Indonesia's National Research and Inovation Agency, Jl. Tamansari No. 71, Bandung, Indonesia.

Abstract

Some areas have unique radiation levels, caused by the high density of buildings or deposits of radioactive minerals. Mamuju district is the area with High Natural Background Radiation (HNBR) despite the low density of the population, while West Java possessing low-level radiation is the area with the largest population in Indonesia. Measurements of dose levels in these two areas were carried out for two weeks covering nine villages in Mamuju and nine districts in West Java. The method used is random sampling to collect data from one meter on the surface, while the tool used was the Ludlum model 19 series 8 Analog Survey meter. The results of radiation measurements were 2.208 mSv/year from 180 data in the Mamuju district and 0.3086 mSv/year in 103 data in West Java. The big difference between the two is due to the presence of radioactive element-carrying minerals in the Mamuju district, while in West Java areas there are lack of radioactive elements mineralization.

Keywords: radiation, scintillator, dose rate, Mamuju, west Java, HNBR

Topic: Radiation Physics

[ABS-39]
Characterization of neutron spectrum on the TRIGA 2000 reactor core using a passive single-moderator neutron spectrometer

Rasito Tursinah, Sidik Permana, Zaki Su'ud, Alan Maulana, Iso Suwarso, Prasetyo Basuki, Teguh Subekti, Tri C. Laksono, Putu Sukmabuana

Nuclear Science and Engineering Department, Bandung Institute of Technology, Jl. Ganeca 10, Bandung, Indonesia
Research Organization for Nuclear Energy, National Research and Innovation Agency, B.J. Habibie Science and Technology Park, Setu, Tangerang Selatan, Banten, Indonesia
Directorate of Nuclear Energy Facilities Management, National Research and Innovation Agency, Jl. Tamansari 71, Bandung, Indonesia

Abstract

The neutron spectrum in the reactor core has a shape with a combination of thermal, epithermal and fast neutrons. Neutron spectrum measurements in the reactor core generally use the multi-foil technique. However, measurements using this technique are quite difficult, expensive to manufacture, but provide less accurate results. The SCNS-reactor has been developed which is a single-moderator neutron spectrometer based on a passive neutron detector for measuring the neutron spectrum in the reactor core. Neutron spectrum measurements have been carried out on the surface of the TRIGA 2000 reactor core using a SCNS-reactor operated at 100 W for 30 minutes. The results of these neutron spectrum measurements were compared with simulations using MCNPX with ENDF7 nuclear data for neutron interactions with fuel and reactor core components, as well as S(&#945-,&#946-) for neutron interactions with the water moderator.

Keywords: Neutron spectrum- reactor core- TRIGA 2000- SCNS-reactor

Topic: Radiation Physics

[ABS-46]
Utilization of Nuclear-Based Analytical Techniques for Characterization and Preservation Studies of Cultural Heritage in Indonesia: Current Progress, Challenges, and Future Outlook

Moh Mualliful Ilmi

Research Centre for Archaeometry, National Research and Innovation Agency of Indonesia (BRIN)

Abstract

Indonesia boasts a rich tapestry of cultural heritage artifacts spanning from the prehistoric to the modern colonial and post-colonial eras, offering invaluable insights into ancient artistic techniques, materials, and historical periods. Our research group, in collaboration with fellow scholars, has embarked on pioneering efforts to unravel the material complexities of Indonesian cultural artifacts. Leveraging a diverse array of nuclear-based analytical tools such as X-ray fluorescence (XRF), synchrotron X-ray diffraction (SR-XRD), and X-ray absorption near-edge structure spectroscopy (XANES), alongside complementary laboratory techniques including Raman spectroscopy and microscopic observations using optical and scanning electron microscopy with energy dispersive X-ray spectroscopy (SEM-EDS), we have delved into the physicochemical properties of these artifacts. This integrated approach has yielded significant findings on optical characteristics, elemental compositions, mineralogical structures, electronic properties, microstructures, and elemental distributions at microscopic scales across artifact surfaces. Beyond material analysis, our research addresses the degradation processes affecting various cultural heritage items, from rock art and pottery to bones and other materials. Moving forward, our aim is to expand these investigations to comprehensively analyze deterioration mechanisms and devise effective preservation strategies for these irreplaceable cultural treasures. While our efforts have been fruitful, challenges remain, particularly concerning the limitations of portable analytical instruments that hinder on-site non-destructive analysis and micro-sampling. Future directions include the exploration of advanced imaging techniques such as hyperspectral imaging and reconstruction methodologies, which are pivotal for advancing our understanding and preservation efforts.

Keywords: Nuclear techniques, Physicochemical, Multianalytical, Characterization, Preservation,

Topic: Radiation Physics

[ABS-60]
Low-Cost Sensor Deployment on a Public Minibus in Fukushima Prefecture

Rakotovao Lovanantenaina Omega (a) , Yo Ishigaki (b) ,Sidik Permana (a),Yoshinori Matsumoto (c),Kayoko Yamamoto (d),Katsumi Shozugawa (e) and Mayumi Hori (f)*

(a) Graduate Program in Nuclear Science and Engineering, Faculty of Mathematics and Natural Sciences, Bandung Institute of Technology, Bandung City 40132, Indonesia
(b) Research Center for Realizing Sustainable Societies, The University of Electro-Communications, Tokyo 182-8585, Japan
(c) Department of Applied Physics and Physico-Informatics, Faculty of Science and Technology, Keio University, Yokohama 223-8522, Japan
(d) Graduate School of Information Science and Technology Department of Informatics, The University of Electro-Communications, Tokyo 182-8585, Japan
(e) Graduate School of Arts and Sciences, The University of Tokyo, Tokyo 153-8902, Japan
(f) College of Arts and Sciences, The University of Tokyo, Tokyo 153-8902, Japan

Abstract

This study examines the implementation of an affordable radiation monitoring system on a public minibus in Fukushima Prefecture. The purpose is to monitor and analyze the levels of radiation in the surrounding area. The study utilizes the Pocket Geiger (POKEGA) sensor, which is combined with a microcontroller and telecommunication system. This integration allows for the gathering and viewing of data in real-time. A monitoring system was implemented on a minibus route between Okuma and Tomioka, which were significantly impacted by the Fukushima Daiichi nuclear accident. The study utilized the Quantum Geographic Information System (QGIS) to calculate the average radiation dose rates over a 1 x 1 km grid. It found a notable decrease in radiation levels from 2022 to 2023. The ecological half-life, determined using land use classification by the Advanced Land Observation Satellite, demonstrated fast radiation decay across different types of land, highlighting the influence of environmental and decontamination parameters. This method not only improves the involvement of the community in monitoring radiation, but also offers information about the success of decontamination efforts and the changing levels of radiation in Fukushima Prefecture.

*Abstract adapted from the research article: Low-Cost Sensor Deployment on a Public Minibus in Fukushima Prefecture- by Rakotovao Lovanantenaina Omega ,Yo Ishigaki, Sidik Permana, Yoshinori Matsumoto, Kayoko Yamamoto, Katsumi Shozugawa, and Mayumi Hori, published in MDPI Sensors

Keywords: public bus, POKEGA, IoT, ambient dose, half-life

Topic: Radiation Physics

[ABS-61]
Radiation Risks and Assessment Challenges in Industrial and Small-Scale Tin Mining: A Study from Bangka

Rakotovao Lovanantenaina Omega (a), Zulfahami (a), Imam Ghazali Yasmint (b), Adi R.A. Abdullah (a), Ismail Humolungo (a), Sidik Permana (a)

(a) Graduate Program in Nuclear Science and Engineering, Faculty of Mathematics and Natural Sciences,
Bandung Institute of Technology, Bandung City 40132, Indonesia

(b) Joint Doctoral Program for Sustainability Research, Graduate School of Informatics and Engineering, The University of Electro-Communications Tokyo, Tokyo 182-8585, Japan

Abstract

The research conducted in Bangka, Indonesia examines the environmental and health consequences of Naturally Occurring Radioactive Materials (NORM) and Technologically Enhanced Naturally Occurring Radioactive Materials (TENORM) in the area where tin mining processing takes place. The study carried out in March 2023 specifically targets three main locations, which are Mentok, Pangkalpinang, and Sungaliat. Mentok faced enhanced surveillance as a result of its substantial tin mining operations. The quantitative results indicate that the level of ambient radiation exposure from industrial mining in Bangka is considerably greater than that from small-scale mining, with an average disparity of around 2.02 microsieverts per hour. The average annual exposure of workers in some areas was determined to be higher than the global average occupational exposure level of 5 millisievert per year, namely, in tailing. Public places, such as offices are recorded as low exposure. These findings suggest a risk of health hazards caused by radiation exposure, require rigorous safety measures and regulatory supervision for the concerned area. The study offers insights into the environmental radiation levels in the tin mining regions of Bangka, highlighting the necessity of implementing protective measures to ensure the safety of workers and the general public against potential radiation exposure.

Keywords: NORM, TENORM, radiation exposure, Bangka

Topic: Radiation Physics

[ABS-62]

Correction Factors Evaluation in Calculation of Saturation Activity for TRIGA 2000 In-Core Neutron Flux Measurements

Fahma Roswita(1,3), Haryo Seno(2), Dikdik Sidik Purnama(2), Nina Widiawati(1), Zaki Su^ud(3), and Nuri Trianti(1)*

1) Research Center for Nuclear Reactor Technology, Research Organization for Nuclear Energy, National Research and Innovation Agency
*fahma.roswita@brin.go.id
2) Research Center for Nuclear Beam Analysis Technology, Organization for Nuclear Energy, National Research and Innovation Agency
3) Nuclear Science and Engineering Study Program, Faculty of Mathematics and Natural Sciences, Bandung Institute of Technology

Abstract

Neutron flux measurement for research reactor in-core is usually using the neutron activation method. The sample in this study used an Au-197 sample. The neutron flux can be obtained by calculating the saturation activity received by the irradiated sample. In the calculation of saturation activity, there are correction factors that can affect the results of the conversion of saturation activity into neutron flux. The calculation of correction factors is carried out analytically by entering variables such as counting efficiency and calculation of uncertainty variables. From this calculation, it can be seen how much influence each component and variable of the correction factor for the saturation activity calculation on the results of neutron flux measurements in the TRIGA 2000 reactor core.

Keywords: neutron flux, saturation activity, correction factors, calculation, reactor

Topic: Radiation Physics

[ABS-67]

Early Warning System for Radiological Emergency Preparedness of TRIGA 2000 Reactor Site

Santiko Tri Sulaksono (a,b), Zahra Fauhan Salsabila (c), Danang Supriyanto (d), Haryo Seno (e), Dani Muliawan (d), Gallant Tsany Abdillah (d), Satrio Aris Setiawan (d)*

a) Department of Doctoral Nuclear Engineering, Faculty of Mathematics and Natural Sciences, Bandung Institute of Technology, Ganesha 10 Street, Bandung, 40132, Indonesia
b) Research Center for Nuclear Reactor Technology, Research Organization for Nuclear Energy, National Research and Innovation Agency (BRIN), Tamansari 71 Street, Bandung, 40132, Indonesia
*sant012[at]brin.go.id
c) Department of Electrical Engineering, Faculty of Science and Technology, Universitas Muhammadiyah Bandung, Soekarno-Hatta 752 Street, Bandung, 40614, Indonesia
d) National Research and Innovation Agency (BRIN), Tamansari 71 Street, Bandung, 40132, Indonesia
e) Research Center for Nuclear Beam Analysis Technology, Research Organization for Nuclear Energy, National Research and Innovation Agency (BRIN), Tamansari 71 Street, Bandung, 40132, Indonesia

Abstract

The National Research and Innovation Agency (BRIN) has a nuclear facility, one of which is the TRIGA 2000 research reactor. Radiation protection along with the emergency preparedness and response programs are essentials to be carried out in the nuclear reactor and its surrounding area. These programs require several types of devices to detect and measure radiation. The aim of this study is to obtain the effectiveness of the early warning system that has been developed for radiological emergency preparedness in the TRIGA 2000 reactor site. A Geiger Muller (GM) counter based on the ATMega 328 microcontroller was created for dose rate measurement and an ESP8266 module to send the results to the server. Javascript programming was created to send notifications to smartphones when the measured dose rate exceeds a certain dose rate. Comparison of the dose rate from the GM counter measurement and the data stored on the server showed a small error. This early warning system can also provide notifications to smartphones when the measured dose rate exceeds the threshold. Based on the test results, it can be concluded that the early warning system performed functions effectively to support the radiation protection program in the TRIGA 2000 reactor site.

Keywords: radiation protection, early warning system, radiological emergency, Geiger Muller counter
Topic: Radiation Physics

[ABS-68]
Data Acquisition of Gamma Radiation Measurement Using Long Range (LoRa) Radio Communication

Santiko Tri Sulaksono (a,b), Zalfa Raihan Salsabila (c), Danang Supriyanto (d), Haryo Seno (e), Dani Muliawan (d), Gallant Tsany Abdillah (d), Satrio Aris Setiawan (d)*

a) Department of Doctoral Nuclear Engineering, Faculty of Mathematics and Natural Sciences, Bandung Institute of Technology, Ganesha 10 Street, Bandung, 40132, Indonesia
b) Research Center for Nuclear Reactor Technology, Research Organization for Nuclear Energy, National Research and Innovation Agency (BRIN), Tamansari 71 Street, Bandung, 40132, Indonesia
*sant012[at]brin.go.id
c) Department of Electrical Engineering, Faculty of Science and Technology, Universitas Muhammadiyah Bandung, Soekarno-Hatta 752 Street, Bandung, 40614, Indonesia
d) National Research and Innovation Agency (BRIN), Tamansari 71 Street, Bandung, 40132, Indonesia
e) Research Center for Nuclear Beam Analysis Technology, Research Organization for Nuclear Energy, National Research and Innovation Agency (BRIN), Tamansari 71 Street, Bandung, 40132, Indonesia

Abstract

Measurement and monitoring of gamma radiation in nuclear installation areas or other areas are commonly carried out for research or radiation protection programs. Measurements in large areas require real-time data acquisition and long-distance data transmission using Long Range (LoRa) radio communication. This study aims to obtain the effectiveness of gamma radiation measurement data transmission using LoRa. Gamma radiation is measured using a Geiger Muller (GM) counter based on the ATMega 328 microcontroller and data transmission using the LoRa RF96 915MHz module. The distances, obstacles, and the length of data delivery time are the parameters observed in this study. The comparison between the GM counter measurement data, as a transmission module, and the data on the receiver module shows no significant differences. Therefore, it can be concluded that the gamma radiation measurement data acquisition system using LoRa functions effectively and can be implemented for radiation monitoring and long-distance measurements.

Keywords: gamma radiation monitoring, radiation protection, Long Range radio communication, LoRa

Topic: Radiation Physics

[ABS-70]

A Comparative Analysis of Organic and Inorganic Scintillation Detectors for Monitoring Environmental Radiation Using Geant4 Simulation

Imam Ghazali Yasmint(a,b), Sidik Permana(b,c), Zulfahmic*

a)Joint Doctoral Program for Sustainability Research, The University of Electro-communications, Tokyo, Japan
b)Department of Physics, Faculty of Mathematics and Natural Sciences, Bandung Institute of Technology Ganesha No. 10 Bandung, Indonesia
c)Department of Nuclear Science and Engineering, Faculty of Mathematics and Natural Sciences, Bandung Institute of Technology Ganesha No. 10 Bandung, Indonesia

Abstract

The number of nuclear physics technologies worldwide is steadily growing. Not just in the domain of nuclear reactors, but also in various other domains such as nuclear physics for medical, agricultural, industrial use, and more. As nuclear utilizes continue to grow, the flow of radioactive material will become more extensive. This provides additional difficulties in terms of monitoring. The purpose of this monitoring is to guarantee that no highly radioactive materials are being freely distributed throughout the community. Radiation detectors are essential for carrying out monitoring. A scintillation detector is currently being developed as a kind of radiation detector due to its superior efficiency compared to other detector designs. Scintillation detectors utilizing organic materials, such as plastic, are specifically designed to monitor radiation levels in the environment. This study aims to conduct a comparative analysis of organic and inorganic scintillation detectors using the Geant4 simulation code. The outcomes of this comparison will serve as a benchmark for evaluating the sensitivity of the organic scintillation detector.

Keywords: Efficiency, Geant4, Nuclear Radiation, Radiation Monitoring, Scintillation Detectors

Topic: Radiation Physics

[ABS-71]
Assessment of Transfer Factors for Natural Radionuclides and Radiocesium from Soil-to-Plant and Plant-to-Cow^s Milk on a Cattle Farm in Lembang

Imam Ghazali Yasmint(a,b), Abdul Waris(b,c,d), Sidik Permana(b,c,d), Eko Pudjadi(e), Ismail Humolungo(c), Deni Karsa Sondana(d)*

a)Joint Doctoral Program for Sustainability Research, The University of Electro-communications, Tokyo, Japan
b) Department of Physics, Faculty of Mathematics and Natural Sciences, Bandung Institute of Technology Ganesha No. 10 Bandung, Indonesia, 40132
c) Department of Nuclear Science and Engineering, Faculty of Mathematics and Natural Sciences, Bandung Institute of Technology, Ganesha No. 10 Bandung, Indonesia, 40132
d) Department of Physics Teaching, Faculty of Mathematics and Natural Sciences, Bandung Institute of Technology Ganesha No. 10 Bandung, Indonesia, 40132
e) Environmental Laboratory, Subs-section of Environmental Safety, Radioecology Section, Center for Technology of Safety and Radiation Metrology, National Nuclear Energy Agency of Indonesia, Lebak Bulus Raya No.49 Jakarta, Indonesia, 12440

Abstract

Besides nuclear reactors, humans are exposed to radiation from natural sources. Natural radioactivity in the environment originates from Uranium and Thorium Radionuclides exposure, typically found in the soil. The detection of radiocesium in the soil signifies the existence of residual byproducts resulting from nuclear reactor disasters and nuclear weapons experiments. Radionuclides present in the soil may enter a process known as soil-to-plant transfer, where they are transmitted to plants such as grass and vegetables. Cows and other animals commonly consume grass as their primary food source. The existence of radionuclides in the grass might indirectly impact human health when people consume meat and milk from these animals (via the transfer of radionuclides from plants to cow^s milk). Hence, it is necessary to conduct studies on soil, grass, and cow^s milk to determine the amount of natural radiation that enters the human body and to understand the transfer of radionuclides. Lembang, a sub-district in West Java Province, Indonesia, has emerged as the focal point for agricultural and veterinary education and research in the region. This study involved the collection of soil, grass, and cow^s milk samples from a cattle farm located in Lembang. Subsequently, the radioactive radiation was quantified with an ORTEC gamma spectrometer equipped with an HPGe detector. The radionuclides detected in this investigation were 226Ra, 232Th, 40K, and 137Cs. The activity concentration in soil is lower than the global average. The transfer factor obtained for soil-to-plant and plant-to-cow^s milk is consistent with the findings of prior research, which have demonstrated that 40K exhibits the highest transfer factor compared to other

radionuclides. The absence of 137Cs in milk samples enabled the determination of its transmission mechanism solely from soil to plant.

Keywords: Activity Concentration, Gamma Spectroscopy, Natural Radionuclides, Plant-to-Cow^s Milk, Radiocesium, Soil-to-Plant, Transfer Factor

Topic: Radiation Physics

[ABS-72]
Evaluation of a PVT-Based Scintillation Detector as a Cost-Effective Early Detection Solution in Nuclear Reactor Safety Systems

Imam Ghazali Yasmint(a,b), Sidik Permana(b,c), Rasito Tursinah(c,d), Amila Amatullah(c), Ismail Humolungo(c) and Marisa Variastuti(c)*

a)Joint Doctoral Program for Sustainability Research, The University of Electro-communications, Tokyo, Japan
b)Department of Physics, Faculty of Mathematics and Natural Sciences, Bandung Institute of Technology Ganesha No. 10 Bandung, Indonesia
c)Department of Nuclear Science and Engineering, Faculty of Mathematics and Natural Sciences, Bandung Institute of Technology Ganesha No. 10 Bandung, Indonesia
d)Nuclear Energy Research Organization, National Research and Innovation Agency (BRIN), Indonesia

Abstract

The safety measures for nuclear reactors are constantly advancing in line with the progress in utilizing nuclear reactors as power sources. Furthermore, the capabilities of early detection systems in nuclear reactors are always advancing to provide the most efficient detectors. Gamma radiation rise serves as an indicator in the safety system of nuclear reactors, specifically in relation to the gamma radiation detector. An outcome of the advancement of this detector is the creation of detectors that possess exceptional capabilities but come with an expensive cost. Poly Vinyl Toluene (PVT) plastic can function as a gamma detector and is cost-effective. This study aims to investigate the possibility of utilizing PVT-based scintillation detectors as a cost-effective solution for incorporating early detection capabilities into nuclear reactor safety systems. The work involved the characterization of PVT-based scintillation detectors, followed by a comparison with commercially available detectors. The primary focus of this research is the efficiency of the detector parameter.

Keywords: Efficiency, Gamma Radiation, PVT, Scintillation Detector

Topic: Radiation Physics

[ABS-24]

Study Of Transfer Factors For Radionuclide And Heavy Metal In Bananas (Musa Paradisiaca) At Tarahan-Lampung As Supporting Data For National Food Security

Rusbani Kurniawan, Radhia Pradana, Agus Nur Rachman, Ilma Dwi Winarni, Riyaz Yusuf, Nicha Miranda Sari, Ahmad Anggraria Jaya Agung, Wahyudi, Oumar Bobbo Modibo, Ilsa Rosianna, Eka Nugraha Djatnika

National Research And Innovation Agency - BRIN

Abstract

The Tarahan-Lampung Coal Steam Power Plant (CFPPs) is one of the largest CFPPs on the island of Sumatra. CFPPs may produce hazardous trace elements (HTEs) and contribute to air pollution. Since bananas are a major commodity in Lampung, it^s crucial to assess the presence of radionuclides and heavy metals in bananas intended for consumption. To accomplish this, a study was conducted to analyze the activity of radionuclides (226Ra, 228Th, 40K, 137Cs) and the heavy metal content (Fe, As, Pb, Hg, Zn) in soil and banana samples from the industrial area around CFPPs-Tarahan. The analysis involved using a High Purity Germanium Detector (HPGe) gamma spectrometer to determine radionuclide activity concentration and X-Ray Fluorescence (XRF) to measure heavy metal concentrations. The highest activity concentration for 226Ra, 228Th, 40K, and 137Cs in soil were 54, 64, 714, and 0.4 Bq/kg, respectively. The concentrations of heavy metals Fe, As, Pb, Hg, and Zn were also determined, producing the following values: 39100, 14, 27, 3, and 221 &#956-g/g. For bananas, the highest activity values of 226Ra and 40K were 15 and 414 Bq/kg, respectively, with no activity of 228Th and 137Cs found. The highest concentrations of heavy metals Fe, Pb, and Zn in dried and wet banana samples were 68, 1.9, 11, and 46.9, 1.4, 7.3 &#956-g/g, respectively. The average value of the radionuclide transfer factor from soil to plants for each radionuclide 226Ra and 40K was 0.3 and 1.2, respectively, while transfer factor values for 228Th and 137Cs were not detected.

Keywords: CFPPs, Gamma spectrometer, X-ray fluorescence, Erica Tools, Transfer factor, radionuclides

Topic: Radioactive Wastes

[ABS-35]
Distribution of Radon-222 Radioisotope in Soil at Siwabessy Science and Technology Area Lebak Bulus Jakarta

Satrio Gilang Ismaya (a), Rasi Prasetio (b), Agus Budhie Wijatna (a)*

(a) Department of Nuclear Engineering and Engineering Physics, Faculty of Engineering, Universitas Gadjah Mada, Indonesia 55281
*satrio.gilang.i@mail.ugm.ac.id
(b) Research Center for Radiation Process Technology, National Research and Innovation Center, Indonesia 15324

Abstract

Two temporary storage sites for nuclear mineral waste at KST Siwabessy Lebak Bulus, Jakarta have the potential to increase radon gas, a natural radioactive gas resulting from uranium decay. Radon-222 (^{222}Rn) is an alpha emitter contributing 1.15 mSv/year per capita to natural radioactivity in the world. Research at KST Siwabessy found that the temporary storage of nuclear mineral waste influences the distribution of ^{222}Rn. The soil radon concentration ranges from 680 to 18,410 Bq/m^3, with anomalies at points 14 and 19 (volleyball court and waste pool). These two points have a much higher value compared to the rest of the measurement points at KST Siwabessy, but are still within typical levels of 2,000 to 50,000 Bq/m^3. The temporary nuclear waste storage site can cause high radon concentrations, but the direction is random and limited to a distance of 75 meter.

Keywords: radon-distribution- concentration- soil- gas

Topic: Radioactive Wastes

[ABS-18]

ASSESSMENT OF GAMMA HEATING IN THE GRAPHITE-MODERATED MOLTEN SALT REACTOR TMSR-500 USING OPENMC SIMULATIONS

Nurul K Fitriyani, Rida S.N. Mahmudah, Azizul Khakim*

*Physics departemant, faculty of mathematics and natural science (FMIPA) Yogyakarta State University, Sleman 55581, Indonesia
Research Center for Nuclear Reactor Technology, Research Organization for Nuclear Energy, National Research and Innovation Agency (BRIN), Building No. 80 BJ Habibie Science and Technology Area, Setu, South Tangerang 15314, Indonesia.

Abstract

Ensuring the safety and efficiency of the TMSR-500 reactor, a Generation IV Molten Salt Reactor (MSR), is crucial for its planned development in Indonesia. This study evaluates gamma heating exposed to the graphite moderator, contributing to the main heat source in the graphite. The TMSR-500, an advancement of the Molten Salt Reactor Experiment (MSRE), utilizes molten salt fuel, is graphite-moderated, and operates at high temperatures. The effective moderation by graphite reduces neutron energy from fast to thermal levels, ensuring optimal fission sustainability and impacting gamma heating. OpenMC simulations were employed using the Python programming language and Monte Carlo methods to assess gamma heating with ENDF/B-VII.1 nuclear data and a particle count of one million. The simulations were conducted on The National Research and Innovation Agency^s computer facilities. The maximum gamma flux was observed at energies greater than 1 eV, with a measured value of 1.8096 x 10^(14) photons/cm^2 sec. The gamma heating in graphite was found to be 3.4917 W and the corresponding volumetric gamma heating was 9.1 x 10^(-3) W/cm^3. The results indicate that gamma heating in the TMSR-500 has a percentage of 2.0138 of the reactor^s total power, within the level of common knowledge.

Keywords: TMSR-500, gamma flux, gamma heating, graphite moderator, OpenMC

Topic: Theoretical and Computational Nuclear Physics

[ABS-20]
Validation of TRIAC Code for Tangential Stress of TRISO Coated-Particle Based on The Experimental Result at Tecdoc. IAEA 1674 Chapter 9t

Muhammad Ilham Bayquni (a), Fajar Al Afghani (a,b), Rohmat Sigit Eko (a), Fitri Miftasani (c), Nini Widiawati (c), Topan Setiadipura (c), Arya adhyaksa waskita (d), Anik Purwaningsih (c), Helmi Fauzi (a), Imam Abdurrosyid (a), Putra Octavianto (a)*

a) Research Center for Nuclear Material and Radioactive Waste, National Research and Innovation Agency, Tangerang Selatan, 15314, Indonesia
*faja020@brin.go.id
b) Department of Physics, Faculty of Mathematics and Natural Science, Universitas Indonesia, Depok, 16424, Indonesia
*
c) Nuclear Reactor Technology Research Center, National Research and Innovation Agency, Tangerang Selatan, 15314, Indonesia
d) Data and Information Science Research Center, National Research and Innovation Agency, Tangerang Selatan, 15314, Indonesia

Abstract

TRISO-coated particle fuel is extensively utilized in high-temperature gas-cooled reactors and other advanced reactors. The performance of these coated fuel particles is crucial for reactor safety. It^s essential to assess and determine the failure probability of coated fuel particles using appropriate fuel performance models and methods under both normal and accident conditions. To enhance the design process of coated particle fuel, a new TRISO fuel performance code-named TRIAC-BATAN was developed. This code is designed to calculate internal gas pressure, mechanical stress, and failure probability of coated fuel particles. This paper introduces TRIAC and benchmarks it against IAEA CRP-6 (tecdoc IAEA 1674 chapter 9) benchmark cases for analyzing coated particle failure, especially for tangential stress maximal. TRIAC-BATAN^s results align well with benchmark values, demonstrating its accuracy and applicability, particularly for tangential stress maximal. This work establishes a reliable foundation for the application of TRIAC-BATAN.

Keywords: validation, TRIAC-BATAN, tangential stress, computational

Topic: Theoretical and Computational Nuclear Physics

[ABS-25]
Thermalhydrolic Analysis on Hot Channels of TRIGA Kartini Reactor with Computational Fluid Dynamics (CFD) Using OpenFOAM

Aisah Rizqi Amaliyah (a), Ratna Dewi Syarifah (a), Nuri Trianti (b), Tri Nugroho Hadi Susanto (c)*

a) Department of Physics, Faculty of Mathematics and Natural Sciences, University of Jember, Jember, Indonesia
*rdsyarifah.fmipa@unej.ac.id
b) Research Centre for Nuclear Reactor Technology, Research Organization for Nuclear Energy, National Research and Innovation Agency, Bandung, Indonesia
c) Kartini Reactor Installation, Directorate of Nuclear Facility Management, National Research and Innovation Agency, Yogyakarta, Indonesia

Abstract

To address the growing energy demands in Indonesia, nuclear power plants, including the development of nuclear reactors, are being explored. One such reactor is the TRIGA Kartini reactor in Yogyakarta, primarily utilized for research and educational purposes. This study analyzes the thermal-hydraulic behavior of hot channels within the TRIGA Kartini reactor, operating at 100 kW, with a focus on temperature distribution and fluid flow-critical factors for ensuring reactor safety and efficiency. The research employs a geometric model, power-to-temperature conversion calculations, and numerical simulations using Computational Fluid Dynamics (CFD) in OpenFOAM. Results from these simulations are compared with experimental data from IFE, which recorded a temperature of 305K in the upper section of the reactor. The simulations are executed using two solvers, Fluid and XiFluid, under four flow conditions: laminar, k-epsilon, k-omega, and k-omegaSST. The findings reveal that the laminar flow condition produces the largest error, exceeding 1% for both solvers, followed by k-omegaSST with errors above 0.6%. For the Fluid solver, the k-omega model shows a 0.30% error, while the k-epsilon model yields the lowest error at 0.23%. Conversely, for the XiFluid solver, the k-epsilon model results in a 0.62% error, with k-omega closely behind at 0.29%. The k-epsilon model using the Fluid solver provides the closest alignment with the experimental data, making it the most accurate among the tested conditions.

Keywords: Thermal-hydraulic- TRIGA Kartini reactor- OpenFOAM- CFD

Topic: Theoretical and Computational Nuclear Physics

[ABS-28]

Neutronic Analysis of Small Long-Life Modular Boiling Water Reactor (BWR) with Thorium Nitride Fuel Using OpenMC

M. Khanifuddin Zuhri (a), Ratna Dewi Syarifah (a), Nuri Trianti (b)*

a) Department of Physics, Faculty of Mathematics and Natural Science, Universitas Jember, Jember, Indonesia
b) Research Center for Nuclear Reactor Technology, Research Organization for Nuclear Energy, National Research and Innovation Agency, Bandung, Indonesia
*rdsyarifah.fmipa@unej.ac.id

Abstract

Nuclear power plants (NPPs) are an alternative to fossil fuel-based power plants, such as coal, which harm the environment and human health. NPPs are built to meet electricity needs and have advantages like being environmentally friendly and economical. One reactor type in NPPs is the boiling water reactor (BWR), currently being developed into a small modular reactor (SMR). The operational duration of an SMR-type BWR can be determined by analyzing the effective multiplication factor (keff) values generated during the combustion process. This research used a homogeneous fuel design with thorium nitride, 8% enriched U-233, and 6% Pa-231. This research aims to determine the most optimal operational outcome for a small long-life BWR without refueling for 30 years. The research was conducted through simulations using the OpenMC program to obtain the most optimal keff value. The most optimal keff value at each parameter optimization stage was then used in the subsequent stages. The final results show that the BWR achieved the most optimal keff value using parameters axial steam percentage, fuel compound density, ENDF-8 library, fuel pin arrangement without assembly, nitrogen-14 nuclide, thermal scattering data, an input power of 100 MWth, and a 66% fuel fraction. These parameters allow the BWR to operate for 30 years with a maximum excess reactivity of 1.96% &#8710-k/k.

Keywords: Neutronics, Boiling water reactor, Thorium nitride, OpenMC

Topic: Theoretical and Computational Nuclear Physics

[ABS-29]

Neutronics Analysis of Heterogeneous Core of Small Modular Reactor Type GFR Thorium Nitride Fueled using OpenMC

Arroofi Candra Kartiko (a), Ratna Dewi Syarifah(a), Nina Widiawati(b), Artoto Arkundato(a), and Fajri Prasetya(c)*

a) Department of Physics, Faculty of Mathematics and Natural Sciences, Jember University, Jember, Indonesia.
b) Research Center for Nuclear Reactor Technology, BRIN, Indonesia
c) Department of Nuclear Science and Engineering, Bandung Institute of Technology, Bandung, Indonesia

Abstract

The use of mixed fuel in the form of thorium-uranium nitride (ThN-UN) offers the potential to improve thermal efficiency and optimize criticality, especially in fast reactors like Gas-cooled Fast Reactors (GFRs). To maintain its criticality, Th-232 absorbs neutrons and transforms into a new fissile material U-233. The inclusion of fertile material Th-232 necessitates a more precise reactor geometry design to ensure the reactor remains critical state until the end of the burn-up period. This study aims to compare variations in the percentage of U-233 enrichment in heterogeneous core geometries with five fuel variations (F1:F2:F3:F4:F5) using the OpenMC program at a power of 100 MWth. Benchmarking is performed by measuring the effective multiplication factor (k-eff) over 5 years of burn-up using reference data from previous studies. If the error value is less than 2%, further calculations will be conducted for both homogeneous and heterogeneous core configurations. The homogeneous calculations indicate that the U-233 enrichment percentage of 8.5% yields the most optimal results. This homogeneous data then is used for calculations in the heterogeneous core with 5 fuel variations. The heterogeneous core configuration is designed using five types different fuel percentages cases, each with variations in ring geometry. The comparison results show that the case 5 heterogeneous core geometry design, with a percentage distribution of 7%:7.5%:8.5%:9%:10,5%, achieves good optimization in terms of k-eff value, excess reactivity, neutron flux, and extended burn-up over a 10-years period.

Keywords: GFR- SMR- OpenMC- Thorium Nitride

Topic: Theoretical and Computational Nuclear Physics

[ABS-41]
Phenomenon Study of Heat Transfer from Clad Surface to Coolant in Pressurized Water Reactor (PWR)

Duwi Hariyanto(a), Sidik Permana(b)(c), Alan Maulana(d)

(a) Industrial Technology Faculty, Institut Teknologi Sumatera, Jl. Terusan Ryacudu, Lampung, Indonesia
(b)Nuclear Physics and Biophysics Research Division, Physics Department, Faculty of Mathematics and Natural Sciences, Institut Teknologi Bandung, Jl. Ganesha 10, Bandung, Indonesia
(c)Nuclear Science and Engineering Department, Faculty of Mathematics and Natural Sciences, Institut Teknologi Bandung, Jl. Ganesha 10, Bandung, Indonesia
(d)Research Center for Nuclear Reactor Technology, National Research and Innovation Agency, Indonesia

Abstract

The heat transfer study from the clad surface to the coolant has become a concern in the thermal analysis of nuclear reactors. The study was proposed to model the phenomenon of heat transfer from the clad surface to the coolant in a pressurized water reactor (PWR). Finite element-based modelling was carried out using the computational fluid dynamics (CFD) module in the Fluent software. In the modelling, heat generation was considered only in the fuel pellets, and the coolant flow rate was varied from bottom to top with the help of a pump. The modelling results showed that the lower the coolant flow rate, the higher the temperature on the clad surface. The coolant flow rate of 5,0 m/s can keep the coolant in the liquid phase.

Keywords: pressurized water reactor, thermal analysis, clad, heat transfer, computational fluid dynamics, Fluent

Topic: Theoretical and Computational Nuclear Physics

[ABS-50]

Investigative Study of Fuel Salt Drainage from Molten Salt Reactor (MSR) to Drain Tank Using Moving Particle Semi-Implicit (MPS) Method

Dhiya Salma Salsabila (a), Fajri Prasetya (a), Fabian Yoga Prastha (a), Tsania Eksa Angelina (a), Asril Pramutadi (b), Sidik Permana (b)*

a) Master Program in Nuclear Science and Engineering Department, Faculty of Mathematics and Natural Science, Bandung Institute of Technology
Jalan Ganesha 10 Bandung 40132, Indonesia
*prasfajri@gmail.com
b) Nuclear Science and Engineering Department, Faculty of Mathematics and Natural Science, Bandung Institute of Technology
Jalan Ganesha 10 Bandung 40132, Indonesia

Abstract

This study investigates the dynamic behavior of fuel salt drainage within a molten salt reactor system, focusing on pressure distribution, velocity profiles, and volume clearance rates over time. Detailed analyses at discrete time points reveal distinct pressure patterns influenced by the density and viscosity of three different fuel salt compositions (A, B, C). Initial high pressures at 0.2 seconds are attributed to collisions as the fuel salt first interacts with pipe corners. Subsequent time intervals show varying pressure ranges, with fuel salt A consistently exhibiting elevated pressures in regions 2 and 3, indicative of its high viscosity and density characteristics. Velocity profiles highlight that fuel salt C has higher flow velocities, attributed to its lower viscosity, which significantly influences fluid dynamics. Additionally, volume clearance rates indicate faster reactor clearance for fuel salt C at 800K, owing to its lower density and consequent lower kinematic viscosity. These findings underscore the complex relationship between fuel salt composition, viscosity, and reactor system dynamics, which is crucial for optimizing reactor design and operational efficiency in molten salt reactor technology.

Keywords: MSR, MPS, Fuel Salt, Pressure, Velocity

Topic: Theoretical and Computational Nuclear Physics

[ABS-56]
Simulation on Neutron Imaging Facility Based on 30 MeV Cyclotron

Dian Adi Prastowo, Fahrurrozi Akbar, Andeka Tris Susanto, Rissa Damayanti, Anas Fahmi Imron, Haryo Seno, Rasito, Ridwan, Mujamilah, Abu Khalid Rivai

Research Center for Nuclear Beam Analysis Technology
National Research and Innovation Agency of Indonesia

Abstract

A new research facility based on 30-MeV cyclotron is now being developed at the National Research and Innovation Agency (BRIN) of Indonesia. One of the instruments at this facility will be neutron imaging (NI). This study aims to design an optimal neutron delivery system for NI using the Particle and Heavy Ion Transport code System (PHITS). The cyclotron accelerates 30 uA protons up to 30 MeV and then subsequently bombarded these protons onto a beryllium target. The neutrons then are moderated by high-density polyethylene. The beamtube is 1-meter long and is constructed based on the refined version of the existing neutron imaging facility at BRIN^s G.A. Siwabessy Research Reactor. Through simulations, we investigated the influence of different reflector and filter materials on the neutron flux. At the start of the beam tube, the overall neutron flux is 3.74 x 10^6 neutrons/(cm^2.s), of which 42.7% are thermal. Our findings suggest that an aluminium reflector coupled with a 5-cm thick graphite filter is the most effective configuration for minimizing neutron flux loss. This configuration delivers 1.5 x 10^4 thermal neutrons/(cm^2.s), which is 69.02% of overall neutrons exiting the beam tube. This number is comparable to other thermal neutron radiography facilities worldwide.

Keywords: Neutron Radiography, Cyclotron, Beamline, PHITS

Topic: Theoretical and Computational Nuclear Physics

[ABS-64]
NEUTRONICS ANALYSIS OF THE FUJI-U3 REACTOR WITH VARIOUS MIXTURES OF THORIUM, URANIUM AND PLUTONIUM FUEL

Faris Rasyad Mutashim (a,b), Cici Wulandari (a,b), Sidik Permana (a,b), Dwi Irwanto (a,b)*

a) Nuclear Physics & Biophysics Research Division, Department of Physics, Faculty of Mathematics and Natural Sciences, Institut Teknologi, Bandung, Indonesia
b) Department of Nuclear Science and Engineering, Faculty of Mathematics and Natural Sciences, Institut Teknologi, Bandung, Indonesia
*psidik[at]itb.ac.id

Abstract

Indonesia experiences limited fossil energy sources which creates a scarcity of electrical energy. The use of new renewable energy has been considered to overcome this problem. Nuclear power plants are one of the appropriate solutions for utilizing new, renewable energy using nuclear power. National Energy Council and the Ministry of Energy and Mineral Resources are targeting that by 2032 nuclear power plants will be commercialized in Indonesia. Of the various types of nuclear reactors, the Molten Salt Reactor is the most suitable type to be developed in Indonesia, which is rich in thorium and uranium. In the development of MSR, the Fuji-U3 reactor scheme was created which has 3 types of reactor cores with varying amounts of different fuel. In this research, MSR Fuji-U3 will be simulated using the SRAC program utilizing the JENDL-4.0 nuclear data library. The results obtained will produce various types of reactor neutronic data. This data is then analyzed in such a way as to obtain a final conclusion regarding the Fuji-U3 reactor. The Keff of the reactor has a value that decreases during operation and greatly influences the amount of fissile substances in the fuel. The CR value is the opposite of Keff. The highest burn-up was obtained by a mixture of thorium-uranium 233 fuel. The smallest peak power ratio was obtained by a mixture of natural uranium fuel and weapons grade plutonium.

Keywords: nuclear power plant, FUJI-U3, neutronic analysist, K-eff

Topic: Theoretical and Computational Nuclear Physics

[ABS-65]
Study of The Effect of Gap Width on APWR Passive Containment Cooling System

Santiko Tri Sulaksono (a,b), Zaki Suud (a), Efrizon Umar (b)*

a) Department of Doctoral Nuclear Engineering, Faculty of Mathematics and Natural Sciences, Bandung Institute of Technology, Ganesha 10 Street, Bandung, 40132, Indonesia
*sant012[at]brin.go.id
b) Research Center for Nuclear Reactor Technology, Research Organization for Nuclear Energy, National Research and Innovation Agency (BRIN), Tamansari 71 Street, Bandung, 40132, Indonesia

Abstract

Advance Pressurized Water Reactor is equipped with a safety feature called Passive Containment Cooling System (PCCS). This system uses natural air circulation that cools the outer surface of the container to remove decay heat released inside the containment vessel. The air in the gap between the outer wall of the containment and the baffle absorbs heat and moves up towards the chimney. Cold air will enter through the inlet so that a natural circulation cycle occurs. To obtain the optimal gap width, it is necessary to analyze the heat transfer on the surface of the containment. In this study, a CFD analysis has been carried out with a 1:40 model dimension of the original containment with a gap width of 1 cm, 2 cm and 3 cm. Based on the results of the analysis, the optimum steady-state temperature of the outer surface of the containment occurs at a gap width of 2 cm.

Keywords: passive containment cooling system, natural circulation, CFD

Topic: Theoretical and Computational Nuclear Physics