ICANSE 2018

International Conference on Advances in Nuclear Science and Engineering

November 29-30, 2018 - East Hall - Institut Teknologi Bandung, Indonesia
Ladies and Gentlemen Welcome to ICANSE 2018, Welcome to Bandung, Indonesia. The Sixth International Conference on Advances in Nuclear Science and Engineering (ICANSE-2018) is conducted in Aula Timur Hall, ITB, Bandung, Indonesia, at 29-30th November 2018. This conference is organized by Bandung Institute of Technology (ITB, Indonesia). This conference was also supported by Indonesian Journal of Physics (IJP). This conference 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.

About 100 persons from Indonesia, Japan, China, and Turkey participate in this conference. About 70 presentations including 8 invited talks will be presented. The presentations are grouped into 13 areas of particular interest: (1) Innovative NPP and Reactor Physics (2) Advanced small reactors without on-site refueling, (3) Innovative transmutation systems, (4) Radioactive waste, (5) Nuclear nonproliferation (6) Innovative energy systems, (7) Material and process for innovative energy system, (8) Radiation physics, (9) Nuclear data, (10) Accelerator, (11) Theoretical and computational nuclear physics and particle physics, (12) Nuclear education and (13) Energy policy.

We would like to thank all participants of ICANSE 2018. We would like to thank all members of International Board, members of Organizing Committee, and my gratitude to all those who help the success of this conference.

Zaki Su’ud                Sidik Permana
ICANSE 2018 Chairman        ICANSE 2018 Co-Chairman
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[KEY-01]
NEUTRON BALANCE LIMITATION ON NUCLEAR ENERGY UTILIZATION

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ABSTRACT

Reactor physics and fuel burnup are discussed in order to obtain a simple global view of the nuclear reactor characteristics. It may provide some idea of free thinking and overall vision. At the beginning of this lecture, governing equations for nuclear reactors are presented. Since the set of these equations is so big and complicated, it is simplified by imposing some extreme conditions. Then the steady state of reactor is discussed and its adjoint is also discussed. η-value is derived form the adjoint function and several values are shown for different fuel types. Next the nuclear equilibrium equation is derived. Some features of future nuclear equilibrium state are obtained by solving this equation. The neutron economy is very severe for the breed-and-burn reactor. The neutron balance limitation is discussed.
[KEY-02] 

SELECTION CRITERIA FOR THE STRUCTURAL AND FIRST WALL MATERIALS OF FUSION REACTORS

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ABSTRACT

Structural materials for fusion reactors are subjected to thermal, mechanical, chemical and radiation loads. This includes ① higher operating temperatures, ② chemically aggressive coolants as energy carrier, such as molten salts, liquid lithium metal or eutectic lithium-lead, lithium-tin, and ③ furthermore magneto-hydrodynamic effects. In a fusion reactor, first wall around the fusion chamber must withstand to high energetic charged particle fluxes, Bremsstrahlung and γ-ray radiation, and most importantly to unconventionally high energetic intense neutron fluxes with a mean energy ~14 MeV. The latter are expected to lead to much higher material damage than observed by fission reactors, not only due to higher neutron kinetic energy, but also, and even more important due to detrimental threshold reactions for structural materials in MeV range. Any maintenance and repair work on the first wall of the fusion chamber will cause a long-term plant shutdown and will be very costly. Moreover, the structure should be compatible with lithium bearing coolants, such as natural lithium, Li17Pb83, Li25Sn75, Li2BeF4, NaF•LiF•BeF2, Li2BeF4 + UF4 and Li2BeF4 +ThF4.

In the present study, the radiation damage parameters (DPA, He-production, H-production) for a first wall load of 5 MW/m² for a multitude of structure materials, such as SS-306, ODS-steel, Vanadium, Niobium, Tantalum, Chromium, Molybdenum and Tungsten Alloys have been calculated by using a variety of lithium containing materials and coolants. The calculations are conducted with MCNP6 package in spherical geometry for a fusion chamber of R=3 m, corresponding to ~700 MWth fusion energy and a neutron flux of 2.22*10¹⁴ 14-MeV neutrons/(cm²·sec) at the first wall for MFE reactor. For the investigations of an IFE reactor, spectral shifting in the highly compressed (D,T) pellet (600 gr/cm³) have been considered. The study has been extended also to a fusion-fission (hybrid) reactor with ① thorium and ② minor actinides suspended as TRISO pellets in the coolant. In the course of calculations, tritium breeding ratio (TBR), blanket energy multiplication, fissile fuel breeding and transmutation of transuranic elements are also evaluated.
[KEY-03]

Efrizon Umar

National Nuclear Agency (BATAN)

Indonesia
RECENT PROGRESS IN ELECTROMAGNETIC PRODUCTION OF KAON ON THE NUCLEON AND NUCLEI

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ABSTRACT

The electromagnetic production of kaon can be performed by using real or virtual photons. By using real photons the process is called photoproduction, whereas by using virtual photons (or electron beam) the process is called electroproduction. The production on the nucleon (proton or neutron) can be used to study a number of phenomelogically related aspects, such as hadronic coupling constants, isospin symmetry, nucleon resonances, missing resonances, and baryon radii through the baryon form factors. These topics have attracted a lot of attention in the last decades. Thanks to the recent advancements in the detector and accelerator technologies a number of precise experiments have been performed, in which experimental data with unprecedented accuracies have been produced. However, the most important result from these phenomelogical studies of this process is the elementary operator or phenomelogical model. With this model, one can predict the probability of producing hypernucleus in terms of differential cross section. A simple hypernucleus is a nucleus in which one of the protons has been substituted by a hyperon. A more complex hypernucleus can consist of more than one hyperon. The hypernucleus is very interesting because its spectrum is obviously different from that of the conventional nuclei. The simplest hypernucleus is the hypertriton that consists of one proton, one neutron and one \( \Lambda \)- or \( \Sigma \)-hyperon. Hypertriton is important since it plays the role of deuteron in the conventional nuclear physics. In this talk I will present recent progress in both theoretical and experimental investigations of kaon production on the nucleon and nuclei.

Keywords: Kaon, \( \Lambda \)-hyperon, photoproduction, isobar model
Almost all nuclides with \((Z, A)\) are transmuted into heavier nuclide with \((Z\pm1, A+1)\) by capturing a neutron and subsequent disintegration by β or EC (electron capture) decay mode. It is theoretically possible to generate rare and expensive nuclide from abundant and inexpensive one using thermal neutrons in existing nuclear reactors. One prominent example of this transmutation process is “making gold from mercury” as what nuclear alchemy literally means. Element mercury (Hg) consists of 7 kinds stable isotopes (Hg-196, Hg-198, Hg-199, Hg-200, Hg-201, Hg-202, Hg-204). Among them, Hg-196 is a neutron deficient nuclide that has the least number of neutrons (116). Hence the thermal capture cross section of Hg-196 is no less than 3 kbarn. The compound nucleus of Hg-197 has relatively short half life of 2.7 days and turns into Au-197 by electron capture decay. This transmutation chain reminds us an idea of rapid production of gold from mercury, the natural abundance of Hg-196, however, is not more than 0.15%. Therefore the production of gold cannot be expected in massive and profitable scale. This study is promoted by pure scientific interest and foreseeable future innovation on emerging strategy for rare material production. As an initial stage of this alchemy study, the authors evaluate the reactivity impact of mercury loading into the multi-purpose research reactor RSG-GAS located in Serpong. It is assumed that one irradiation position (IP) around the central irradiation position (CIP) is occupied by mercury encapsulated in SUS tube lined with thin quartz glass. The length of mercury tube and its axial position in the active core are parametrized. Annular shape tube is also examined in addition to cylindrical shape to confirm the self-shielding effect caused by large cross section nuclides such as Hg-199 (2.15 kbarn) and Hg196 (3.08 kbarn). The production rate of Au-197 is calculated considering the realistic operation schedule of RSG-GAS in the year 2018 with half full power (15MW). The one operation cycle length of RSG-GAS is 5 days and this cycle is repeated 10 times until next maintenance and refueling. This 50 days operation is conducted 3 times a year, namely, annual operation days is expected as 150 days/year. Assuming a cylindrical mercury tube with 6.8cm diameter that contains about 30kg of natural mercury and 40g of Hg-196, the production rate of Au-197 is evaluated as about 500g/75EFPD that is equivalent to 500g/year.

**Keywords:** Hg-196, Au-197, transmutation, thermal neutron
[KEY-06]  
SPENT NUCLEAR FUEL MANAGEMENT: TOWARD ZERO RELEASE NUCLEAR WASTES  
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ABSTRACT  
Public acceptance is one of the important aspects regarding the nuclear energy utilization, especially for country that would like to “go nuclear”. There are three main issues about nuclear energy. They are nuclear reactor safety, nuclear proliferation, and spent nuclear fuel management. However, the safety of the present nuclear reactors are excellent. The nuclear proliferation is more political issue rather than scientific and technological issue. Therefore the main issue regarding the nuclear energy utilization is the spent nuclear fuel management. Hopefully, the better spent nuclear fuel management may increase the public acceptance on nuclear energy. In this conference several methods for the spent nuclear fuel management will be discussed.  

Keywords: reactor safety, nuclear proliferation, spent nuclear fuel management, public acceptance
[KEY-07]

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[KEY-08]

Geni Rina Sunaryo
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[ABS-01]  
THE ANALYSIS AIR-COOLED HEAT EXCHANGER REACTOR CAVITY COOLING SYSTEM (RCCS) ON EXPERIMENTAL POWER REACTOR (RDE) DESIGN

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ABSTRACT

The National Nuclear Energy Agency (BATAN) has designed the Experimental Power Reactor (RDE). The Cavity Cooling System Reactor (RCCS) is a new nuclear power plant, and it will be incorporated into High-Temperature Reactor (HTR), which is one of the proposed generation IV reactors. While HTR has been designed and operated, the RCCS design is unique to the recent generation of gas-cooled reactors as a passive decay heat removal system. But many aspects of the new water cooled design are still investigated on RCCS. The temperature, velocity and optimal diameter of the cavity cooling system which are used is a necessary test. The RCCS designs have not studied for RDE design. The fluid flow and heat transfer on the RCCS facility have not an investigation for air-cooled in cavity cooling systems. The objective of this study is to investigate the air-cooled heat exchanger of RCCS for RDE design 10 MW thermal. The variation temperature, flow velocity, a number, and diameter cooling pipe cavity was analyzed and simulated with Computation Fluid Dynamic (CFD) FLUENT 6.3 software. The flow velocity and delta temperature on process RCCS were obtained and investigated for RDE design.

Keywords: HTR, RDE, RCCS, Air-Cooled Heat Exchanger, CFD

Topic: Innovative nuclear energy systems
[ABS-02]  
DESIGN STUDY OF NEUTRONIC ANALYSIS OF SODIUM-COOLED FAST REACTOR (SFR) FOR VARIOUS OUTPUT POWER USING RADIAL FUEL SHUFFLING STRATEGY

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ABSTRACT

Design study of neutronic analysis of Sodium-Cooled Fast Reactor (SFR) based on the variations of output power using radial fuel shuffling strategy has been done. SFR is one of the generation IV reactors type that currently being researched for commercial applications. The reactor design uses natural uranium as a fuel and Sodium as a coolant. The research has been carried out by using the SRAC code and JENDL-3.2 as a library with two dimensional R- Z shuffling strategy of cylinder core for variations of 300, 350, 400, 450 dan 500 MWt power output. The global neutronic parameter such as multiplication factor (keff) and burn up analysis are observed. The reactor core is divided into 10 regions that have the same volume radially. At beginning, the reactor was fully filled with natural uranium fuel called fresh fuel and was prepared for the first core cycle. The burns up result in the first region is shuffling into the second region, the burn up result in the second region is shuffling to third region, and so on until the burn results in the tenth region. The burn results in the tenth region is removed from the reactor core, then the first region can be filled with fresh fuel and so on up to 100 years of reactor operation. The neutron calculation results indicate that the multiplication factors (keff and kinf) are in a critical condition occurring for 300 MWt of output power. The density of U-235 nuclides at 300 MWt has a greater value from the beginning to the end of the burn up period. Overall, the output power of 300 MWt has requirements and a greater chance of being operated for SFR reactors as designed in this study.

Keywords: SFR, multiplication factor, burn up, shuffling strategy

Topic: Advanced small reactors without on-site refueling
[ABS-03]  
ESTIMATION ON PRESSURE AND VELOCITY PARAMETER IN THE FUEL HANDLING PNEUMATIC DESIGN OF RDE

sukmanto dibyo, topan setiadipura, marliyadi pancoko 
BATAN

ABSTRACT

The RDE is small size of High Temperature Gas-Cooled Reactor type. Its being designed that have a thermal power of 10 MWt, which is intended to produce the electricity and heat utilization. The reactor uses helium gas as a coolant system at the pressure of 3.0 MPa. The spherical fuel pebbles are transported pneumatically in the pipes and recharged from the reactor core continuously without shut-down the reactor. During the transportation, the fuel pebble is lifted perpendicularly along the pipe into the reactor core. Therefore, fuel pebble transportation in the pneumatic system is an important effects to stability the fuel handling operation. This paper to estimate the fuel pebble velocity through a vertical pipe with respect to the pressure of pneumatic system in the design of RDE. The pneumatic vertical pipe has 22 m length, 65 mm inner diameter and fuel pebble of 60 mm diameters, weight of one fuel pebble is 200 gr and contains an uranium kernel coated in matrix graphite. The pneumatic system of carrier gas compressed as the the conveying medium must be able to transport the fuel pebble through the pneumatic pipe one by one continuously, so that it could be loaded into the reactor core safely. In the pneumatic conveying, the general equation for the velocity of an object movement, drag force and Bernoulli principle were used in the analysis. An initial velocity in range of 20 to 23 m/s was analyzed, meanwhile, the fuel pebble discharged from the pneumatic pipe was determined at low velocity in the range of 0.1 to 5.0 m/s. In this case a collisions at high velocity with the reactor core that causes pebble damage can be avoided. Result shows that the pneumatic transportation can meet the operational requirements of the fuel handling system when the carrier gas pressure in the range of 90.000 to 110.000 Pa is higher than the reactor pressure system. This analysis is expected may provide a part of component specification especially for the design of pneumatic transportation to achieve an operating stability in the fuel handling system.

Keywords: fuel pebble velocity, pneumatic pressure, fuel handling system, design of RDE

Topic: Advanced small reactors without on-site refueling
[ABS-04]

DESIGN STUDY OF HELIUM COOLED FAST REACTOR USING MODIFIED CANDLE BURNUP SCHEME IN RADIAL DIRECTION FOR SMALL REACTOR DESIGN

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ABSTRACT

Design study of helium cooled fast reactor using modified CANDLE burnup scheme in radial direction for small reactor has been performed. In this study reactor core used is a balanced cylinder 2D R-Z with size 6m3. In Modified CANDLE burnup scheme in radial direction, the active core is subdivided into ten regions with the same volume in the radial direction. When startup, each region contains fuel with different composition. The first region contains natural uranium (fresh fuel). The second region contains fuel from natural uranium burning for 10 years, the third region contains fuel from natural uranium burning for 20 years and so on. Finally, the tenth region contains fuel from natural uranium burning for 90 years. In this study, cell burnup calculation were performed using the 2002 SRAC code system with JENDL-4.0 library data and core calculation using FI-ITB CH1 program. The results of calculation show that 64% fuel volume fraction provides a better criticality and stability value with 1% excess reactivity.

Keywords: MCANDLE in radial direction; small reactor; fuel volume fraction

Topic: Innovative nuclear energy systems
[ABS-06]
GAMMA HEAT ANALYSIS IN VARIOUS POWER LEVELS OF RSG G.A. SIWABESSY SILICIDE CORE

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ABSTRACT

Reaktor Serba Guna G.A. Siwabessy (RSG-GAS) is the largest research reactor in Southeast Asia which acts as a national facility to irradiate material. Gamma heat is a very important factor for the safety analysis in every material irradiation activity at the irradiation facility. Gamma heat is the main research topic of several forms of safety investigations at world research reactors. Gamma heat information is useful to predict the temperature of the material to be irradiated and negligence in predicting gamma heat can cause overheating. Gamma heat values are very dependent on the characteristics of the reactor core. Changes in reactor power can affect the core characteristics. RSG-GAS is designed to have a 30 MWt of nominal power but it is currently operated at 15 MWt power level. In this study observed changes in gamma heat as a function of reactor power and material target on the Central Irradiation Position (CIP) silicide core of RSG-GAS by using a modified GAMSET program. Modifications are made by adjusting the material and power configuration in the core. The results of the analysis show that gamma heat will be increase in accordance with the increase in power level in various material targets such as graphite (C), aluminium (Al), iron (Fe) and zirconium (Zr). Gamma heat also tends to increase according to the increase of atomic number target. Verification has been carried out with the results of calculations in the 35 MW CEA Grenoble reactor with the smallest yield difference of 1.23% in the graphite target and 2.71% in the iron target.

Keywords: RSG-GAS, gamma heat, core power, GAMSET

Topic: Radiation physics
[ABS-07]
PREMINILARY STUDY OF IRON CORROSION IN HIGH TEMPERATURE STAGNANT LIQUID LEAD BY USING MOLECULAR DYNAMICS

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ABSTRACT

Liquid metal was used as a type of coolant in the 4th generation nuclear power plants, one of the examples is the LMFR (Liquid Metal Fast Reactor). Liquid Lead is one of the materials that could be used as the coolant. The liquid Lead was considered since it has many chemical and physical advantages compared with other materials. However, its corrosive nature is the main problem of liquid metal Lead within fuel cladding and steels.

This study was conducted to improve the understanding of this phenomena. In this research, the corrosion regarded as degradation phenomena of material as a result of the high dissolution of iron atoms interaction within the high-temperature liquid Lead. We calculated the diffusion coefficient of the iron pure system of Fe-56 within Pb-208 and mix system of Fe-56 within Pb-208 and Pb-206.

Keywords: Corrosion, Dissolution, Diffusion coefficient, Fe-56, Pb-206, Pb-208.
Topic: Material and process for innovative energy systems
[ABS-08]

STUDIED OF HCLL (HELIUM COOLED LITHIUM LEAD) BLANKET DESIGN BY USING MCNPX PROGRAM

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ABSTRACT

The fusion reactor using a tritium-deuterium reaction is going to be a prolific source of 14.1 MeV (80% of fusion energy) neutrons, which will be released from the plasma. This reactor features a major radius of 8.1 m and minor radius of 2.5 m. Thus, one part of the fusion reactor which is an important design is Blanket Module (BM). One of the two European blanket concepts is Helium Cooled Lithium Lead (HCLL). HCLL blanket uses Pb-Li as a neutron multiplier, Li as a breeder, and He as a cooler is in high pressure (8 MPa). The aim of this study is to investigate the feasibility of blanket reactors such as neutron analysis, shield efficiency, and tritium breeding capabilities to achieve TBR value more than one which will assure tritium self-sufficiency in steady state operations. The MCNP program with Monte Carlo is a tool for calculating this. The neutron flux produced by this simulation shows that the reactor has no leak. And the production of tritium from breeding blanket will be optimal when the utilizing of breeding materials is Pb-Li.

Keywords: blanket, HCLL, Lithium lead, MCNPX, TBR, tritium

Topic: Innovative nuclear energy systems
INVESTIGATION OF RDE THERMAL PARAMETERS DURING DLOFC IN THE ABSENCE OF WATER INSIDE THE REACTOR CAVITY COOLING SYSTEM

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ABSTRACT

Reaktor Daya Eksperimental (RDE) is a 10MWt small modular of High Temperature Gas-cooled Reactor (HTGR) with passive safety features. This reactor has a Reactor Cavity Cooling System (RCCS) to evacuate heat loss from reactor vessel in operation condition and also the decay heat in the case of accident condition. One of the most important accidents to be considered is the Depressurized Loss Of Forced Cooling (DLOFC) for a long time period. During this accident, the reactor decay heat is evacuated to the cavity cooling system which is supposed in condition of totally loss its cooling water. In the present work, sensitivity analyses on peak temperatures of both the core and reactor building (RB) after reactor shutdown was conducted for both situations. The analysis simulation was performed by solving numerically the two dimensional equation of heat conduction. The implicit approach was used in the discretization of the equation, Matlab application program is used to execute the simulation program. The simulation was performed in two periods such as 1500 hours calculation to reach the steady state condition in normal operation and 1000 hours of DLOFC period. The results show the increasing temperature of core after reactor shut-down until attain the peak temperature of 1092.5°C. According to reactor building, the peak temperature of internal concrete wall is 107.6°C and for external concrete wall is 39.9°C. In absence of RCCS cooling water panel, the decay heat is evacuated to reactor building in all directions. The peak heat evacuated through such as: upper wall is 3.34 kW, bottom wall is 4.36 kW, side wall is 28.08 kW, so the peak of total heat evacuated through the reactor building is 35.77 kW. Although the reactor core remains safe because the peak temperature is still under the limit of 1600°C, a precaution should be taken on the internal side wall temperature that exceeds the limit of 70°C.

Keywords: HTGR, DLOFC, RCCS without water, Matlab simulation

Topic: Innovative nuclear energy systems
[ABS-10]

STUDY OF MITIGATION NUCLEAR DISASTER, CASE STUDY: MUNTOK SUB DISTRICT, WEST BANGKA REGENCY, BANGKA BELITUNG

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ABSTRACT

The construction plan of an 825 ha Nuclear Power Plant in the Teluk Inggris, Muntok Sub District, West Bangka Regency in order to supply national electricity needs has caused a polemic especially for the communities in Bangka Belitung. The Fukushima Daichi and Chernobyl nuclear accidents have caused most people still phobic about the utilization of nuclear power in Indonesia. Furthermore, the lack of communities’ understanding about nuclear technology and mitigation strategies towards nuclear disasters also affect communities’ level of vulnerability. This paper aims to formulate mitigation strategies in creating a basic sense of awareness towards nuclear disasters for communities in Muntok Sub District. The results show that there are several phases of mitigation that must be conducted by the communities towards nuclear disaster. In the preparation phase, the communities must know the condition of the potential evacuation routes (streets) and evacuation destination. While in the preparedness phase, the communities should minimize the radiation exposure’s effect by considering time, distance and shielding when evacuate.

Keywords: mitigation, nuclear disaster, evacuation, communities

Topic: Nuclear education
[ABS-11]
OPTIMIZATION OF MODIFIED CANDLE BURN UP STRATEGY ON GAS COOLED FAST REACTOR (GFR)

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ABSTRACT

We conducted an optimization of modified CANDLE burn up strategy on Gas Cooled Fast Reactor (GFR). The reactor cores subdivided into 10 regions with equal volume in radial directions. The natural uranium is initially put in the first region. After one cycle of 10 years burn up, fuel shifted from first region into second region. The Fuel from second region shifted to third and so on until fuel in nineth region shifted to tenth region and finally fuel in tenth region was carried out from reactor core. Optimization evaluated by changing fuel region shuffling scheme. Two shuffling scheme was use in this study. First scheme is performed by positioning region 1,10, 9, 8, 7, 6, 5, 4, 3, 2 sequentially. Second scheme is performed by positioning region 1, 10, 3, 8, 5, 6, 7, 4, 9, 2 sequentially. Calculation has been done by using SRAC system code with JENDL-32 as library, with cylindrical two dimensional R-Z core models. Modified CANDLE method was used in order to make reactor can be operated using natural uranium. This natural uranium initially being burned by guessed power level of burn up. The height and diameter core are 350 cm and 240 cm respectively. The volume fraction for this design is 65% fuel, 10% cladding and 25% coolant; with output power 550 MWTh. The results show that the reactor has been demonstrated on excellent performance in both scheme, however second scheme have lower average power densities than first scheme.

Keywords: GFR, natural uranium, burn up, shuffling scheme, modified CANDLE

Topic: Nuclear nonproliferation issues
[ABS-12] SIMULATION OF DEEP TRENCHES BETAVOLTAIC DESIGN USING NI-63 IMPLEMENTING NUMERICAL MCDE

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ABSTRACT

The electrical performance of basic planar design for the surface area is limited to the deposition of source on flat surface. However, there are several ways to increase the utilization of beta source deposited on surface which one of our proposal using etching technique to form deep trenches over the p-n junction. Such deep trenches would be evaluated by dividing it into several regions, which represents the shape of the whole surface. Herein, we employ Ni-63 as the beta source of betavoltaic. In our result compared to the planar design, the deep trenches could give an overall improvement in electrical performance, which the conversion efficiency increased about 29.69%.

Keywords: Betavoltaic, Deep Trenches, Nuclear Battery, Planar

Topic: Innovative nuclear energy systems
[ABS-13]
STUDY ON THE CORE SHAPE OF INTRINSIC SAFE FAST REACTOR PREVENTING RE-CRITICALITY

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ABSTRACT

In thermal neutron reactors such as light water reactors, the reactor core is designed with nearly maximum reactivity system. Since the neutron is not slowing down in the Sodium-cooled fast reactor, there is no peak value of (VNa / Vf) that maximizes the effective multiplication factor. There is a possibility of re-criticality when fuel compaction occurred in severe accident including core melting. In JSFR(Japan Sodium-cooled Fast Reactor), which is being studied mainly by JAEA(Japan Atomic Energy Agency), introduction of FAIDUS(Fuel Assembly with Inner Duct Structure), which discharges fuel to the outside of the reactor core at the early stage of fuel melting, is considered as the countermeasure to avoid re-criticality. In this research, we aim to investigate core configuration of intrinsic safe Sodium-cooled fast reactor which prevent re-criticality only by change of the core shape. We devised a reactor core structure in which the lower part of the radial blanket is deleted by about half and it is set as the sodium-region. The fuel pool at the time of core melting spreads beyond the core diameter and flattened, aiming at the reduction of the criticality of the molten pool. We adopted short length radial blanket assembly. As a result, negative reactivity of about 10 $ was inserted, and the molten fuel pool became subcritical.

Keywords: fast breeder reactor, core geometry, short length blanket fuel assembly, re-criticality, core disruptive accident

Topic: Innovative nuclear energy systems
[ABS-14]

THE TEMPERATURE DEPENDENCE ANALYSIS OF CARBON MONOXIDE CONVERSION ON CUO BED OF HELIUM PURIFICATION SYSTEM OF RDE

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ABSTRACT

RDE is an experimental power reactor that designed base on HTGR technology. This reactor has an indirect cycle power conversion system with primary coolant is helium and it secondary is water. RDE uses pebble fuel and multipass refuelling scheme. Helium coolant was predicted to be contaminated with various gas impurities such as CO2, H2O, O2, CH4, N2, and H2 and CO. Those impurities must be maintained at certain limits by using Helium Purification System (HPS). One of HPS functions is removing of chemical and particulate contaminants from the primary coolant to maintain at a specified value. Carbon monoxide (CO) is difficult to remove from the helium due to its small molecular size. Firstly, in HPS CO must be converted to CO2 by using cupric (II) oxide (CuO) bed and then CO2 will be adsorbed by molecular sieve column. The CuO bed operation temperature has a significant contribution to CO oxidation process results. This paper discusses the influence of oxidation temperature on CuO bed against formation of CO2. Analyses were performed by using ChemCAD computer code. The inlet mass flow rate of helium to CuO bed is assumed at 10.5 kg/s, 30 bar of pressure and temperature has been varied to get the optimum result of CO2. The simulation has shown that CuO bed optimum temperature for producing CO2 is in the range of 250-375ºC. In the HPS, this temperature can be maintained by using heater that has been installed before CuO bed.

Keywords: temperature dependence, CO, oxidation, CuO bed, RDE

Topic: Innovative separation and fuel cycles/radioactive wastes
[ABS-15]
APPLICATION OF GRAPH METHOD FOR ANALYZING THE PHYSICAL PROTECTION SYSTEM EFFECTIVENESS

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ABSTRACT

Physical protection system (PPS) effectiveness of nuclear facility needs to be assessed regularly for providing adequate and optimum protection against sabotage or unlawful acts. The main challenge of this routine activity is to reduce the annual budget with still maintain the quality of assessment. The computer-based analytical tool offers alternative solutions by measure the likelihood adversary threat scenario against the essential parameter detect, delay and response to determine the most vulnerable path in the system. There are several available tools which has been developed such as EASI and SAPE, however for research purpose it is more suitable to have the tool that can be customized and enhanced further. The objective of this study is to demonstrate the research on computer-based analytical tool for assessing the physical protection system effectiveness as part of BATAN nuclear security program. This tool is use the combination of adversary path analysis method and the application of graph theory. The capabilities of this tool are able to model adversary line in multiple path for sabotage scenario, calculate the probability of interruption, and specify the critical detection point as PPS performance measurement and determine as well as rank the most vulnerable path in facility.

Keywords: Physical Protection System, Analytical Tool, Network, Adversary Path Analysis.

Topic: Nuclear nonproliferation issues
DESIGN CONCEPTUAL OF 800MWT LONG LIFE PRESSURIZED WATER REACTOR USING (TH,U)O2 FUEL WITH GD2O3 AND PA231 AS BURNABLE POISON

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ABSTRACT

Long life Pressurized Water Reactor (PWR) has been reviewed as innovative reactor design that can fulfil electricity demand. The aim of this study is to perceive optimum design of 800MWe long life PWR using (Th,U)O2 fuel with Gd2O3 and Pa-231 as burnable poison. An established computer code of SRAC 2006 with JENDL 4.0 as data nuclear library has been used for the analysis. A two-dimensional R-Z geometry and 40% fuel volume fraction were used for core geometry analysis. Different fraction of Uranium dioxide, Uranium-235, Gadolinium, and Protactinium-231 have been carried out. The result of this study is 800MWe PWR design using 60% UO2 fuel with enriched 11-13% U-235 that can operate for 10 years without refueling. The reactor can produce power density of 38.6 watts/cc in beginning of life (BOL) and 45.4 watts/cc in ending of life (EOL). This study is expected to be a reference for long life PWR using Thorium-Uranium fuel cycle.

Keywords: Pressurized water reactor, thorium-uranium fuel cycle, burnable poison.

Topic: Advanced small reactors without on-site refueling
[ABS-17]
COMPARISON OF MOX AND (U,Pu)N FUEL FOR MODULAR NUCLEAR POWER PLANT LONG LIFE GAS COOLED FAST REACTOR

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ABSTRACT

This paper will discussed about optimization and comparison of Mixed-oxide (MOX) and Uranium Plutonium Nitride (U,Pu)N fuel in modular NPPs long life Gas Cooled Fast Reactor which can be operated over 20 years. In future, the application of this modular NPPs will be the optimum option source of energy for generate electric power for big city or small islands with more secure. The design of this NPPs with cylindrical core, MOX and (U-Pu)N as a fuels, and 100 MWth Power. The calculation for neutronic result used SRAC code with library nuclear JENDL-4.0. To decrease the initial reactivity swing and make the reactor have long life characteristics, it is added burnable poison to the fuel. The reactors can achieve desirable result with optimum design

Keywords: GCFR; Modular; Long Life; SRAC

Topic: Innovative nuclear energy systems
[ABS-18]
FULL CORE OPTIMIZATION OF SMALL MODULAR GAS COOLED FAST REACTOR USING OPENMC CODE

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ABSTRACT

Indonesia has many small regions or islands that need small power plants. Gas-cooled Fast Reactors (GCFR) is one of the fourth generations of an advanced reactor with better safety and optimum electricity production that match Indonesia’s needs. The neutronic analysis study and optimization of small modular GCFR has been performed with OpenMC code that use Monte Carlo and Parallelization Method to speed up the calculation time. The design of the reactor is cylindrical 3D with radius and height are 100 cm and 120 cm with 50 cm of additional radius for the reflector. The Core using several types of fuel composed of natural Uranium mixed with Plutonium and variation in hexagonal assembly design to achieved optimum core design. The optimum design of reactor has flattened flux and power distribution.

Keywords: Small Modular; OpenMC; Parallelization; GCFR

Topic: Innovative nuclear energy systems
[ABS-19]  
ANALYSIS OF MODULAR GAS-COOLED FAST REACTOR WITH PLUTONIUM FUEL USING PARALLELIZATION MCNP6 CODE

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ABSTRACT

Modular Gas-cooled Fast Reactor (GFR) is one of six advanced reactor concepts set by the generation IV international forum. Modular GFR has the potential for use actinide recycling and closed fuel cycle as well as applying fast reactor, using helium gas as the main coolant, high working temperature and low void reactivity effect. The neutronics analysis of nuclear reactor means behavior study of subatomic particles that interact with matter. In this paper, the feasibility of plutonium fuel in modular Gas-cooled Fast Reactor (GFR) was investigated. The Monte Carlo method has advantages in full-scale and heterogeneous three-dimensional (3D) geometry modeling using Evaluated Nuclear Data File (ENDF/B-VIII.b5) nuclear data but requires a highly computation time. Since the progress of high performance computing, the reactor physicist community began proposing to use Monte Carlo method for nuclear reactor simulation through the parallelization of calculations. The GFR feasibility design study will carried out with plutonium fuel as fuel cycle inputs with a power range of 400-800 MWth. The most important neutronic parameters characterizing of GFR core are determined both for beginning of life (BOL) conditions and during burnup calculation. The neutronic parameters compared include keff, reactivity and neutron flux profiles. The results of fuel burnup and isotopic transmutation confirm the capability of the core to work in both open and closed cycles.

Keywords: Neutronic parameters, Modular Gas-cooled Fast Reactor (GFR), parallelization, MCNP6

Topic: Innovative nuclear energy systems
[ABS-20]
COMPARATIVES STUDIES OF A SAFETY ANALYSIS FOR MOLTEN SALT REACTORS

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ABSTRACT

The Generation IV nuclear power system has been developed by the Generation IV Forum (GIF) with its benefits such as highly economical, enhanced safety, low waste production, and proliferation resistant. MSR is one of Generation IV nuclear power system which uses molten salt as a fuel and a coolant, therefore its technology differs from the conventional solid-fuel reactors. In this study, the author focus on the safety characteristic analysis of the MSRs, in which a point kinetic equations and core heat transfer have been modeled by developing a mathematical model for MSRs. The founded model is applied to analyze the safety characteristics of the molten salt actinide recycler and transmuter system (MOSART) by simulating transient basic condition, and that is the unprotected loss of flow (ULOF). The result of reactor accident simulation is power distribution and temperature in the terrace reactor toward the time and showing that MOSART conceptual design is a reactor design which is stable inherently. This research purposes to give basic understanding about safety characteristic of the MSR.

Keywords: Safety analysis, ULOF, MSR

Topic: Theoretical and computational nuclear physics and particle physics
[ABS-21]

SIMULATION OF VMAT (VOLUMETRIC MODULATED ARC THERAPY) DELIVERY TECHNIQUES ON CYLINDER PHANTOM BASED ON DICOM DATA USING MONTE CARLO METHOD - EGSNRC

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ABSTRACT

This research aims to studied one of the radiotherapy techniques of VMAT and calculate the dose distribution on phantom cylinders using the Monte Carlo EGSnrc method. VMAT is one of radiotherapy technique where is all the fractions of the dose are given continuously when the gantry rotates around the patient with manages gantry rotation speed, MLC leaf position and the dose fraction given. The simulation process begin with head linac modeling using BEAMnrc, and using the Monte Carlo VMAT simulation uses DOSXYZnrc software. The parameters for the BEAMnrc and DOSXYZnrc simulations using information from DICOM data set rtpian AAA are read using the pycom code. The results of this study provide information about the characteristics of 6 MV photon files such as fluence, fluence energy, spectral distribution and fluence energy distribution, and dose distribution in phantom cylinders. The VMAT Monte Carlo simulation was carried out using 300 million particles which resulted in a VMAT dose distribution in the form of a dose profile curve and an isodosis curve, the result is shows that the curve of VMAT simulation using phantom cylinders produced the maximum dose distribution in the isocenter region.

Keywords: VMAT, Monte Carlo, EGSnrc, DICOM, AAA

Topic: Radiation physics
[ABS-22]

STUDY ON INSERVICE INSPECTION PROGRAM AND METHOD FOR FUEL HANDLING SYSTEM OF RDE

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BATAN

ABSTRACT

One of the main programs should be established on designing process of fuel handling system is establish in-service inspection program for maintaining the integrity of system, structure and component during service life time. The most important role of in-service inspection is the nondestructive examination techniques. The objective of this study is propose the preliminary program for examining the integrity of fuel handling system during fabrication, commissioning, and operation and determining the best method to confirm the defects. The proposed programs are describe as follow, defining the operation environment of fuel handling system, identifying the material characteristics and manufacturing process which has indication to promote the defects, and selecting the appropriate method of nondestructive examination and analyses technique for such kind of defects. The proposed in-service inspection program is expected giving significant additional value in fuel handling system design of RDE.

Keywords: Keywords: in-service inspection program, non-destructive examination, fuel handling system, reaktor daya experimental

Topic: Material and process for innovative energy systems
[ABS-23]
NEUTRONIC PERFORMANCES OF UN AND UO2 FUEL OF LEAD208-BISMUTH EUTECTIC- COOLED REACTOR WITH MODIFIED CANDLE BURN UP SCHEME

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ABSTRACT

The nuclear reactors contribute to producing around 16% of the world's electricity. Nuclear reactor design is called by “generation”. Most of the reactors which operate in the world are 2nd generation. The fourth generation nuclear reactor are currently developing. One type of fourth generation reactor is the Lead Bismuth Eutectic (LBE) -cooled reactor (LFR). UO2 (Uranium dioxide) is a reactor fuel for SVBR 75-100 (Russian). However, the disadvantage of UO2 is lower breeding ratio. In this study we were calculating the UN (Uranium Nitride) neutronic performances as the other LFR fuel candidate. The neutronic calculation was performed by SRAC (PIJ-CITATION) module. The neutronic performances of these fuels will be compared to get the better fuel for LFR.

Keywords: Generation IV, Lead208-Bi eutectic, Modified CANDLE, UN, UO2, neutronic

Topic: Material and process for innovative energy systems
NEUTRONIC SURVEY OF LEAD208-BISMUTH EUTECTIC-COOLED REACTOR WITH MODIFIED CANDLE BURN UP SCHEME

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ABSTRACT

Lead208-Bismuth Eutectic (LBE)- cooled reactor is one of the generation IV reactor. The fourth generation has several objectives: economic competitiveness, inherent safety, minimize radioactive waste, and non-proliferation. Modified CANDLE burnup scheme is one of solution to fulfill these objectives. The scheme does not need fuel enrichment and reprocessing Pu from LWR. Lead208-Bi eutectic was employed as a coolant of the reactor due to its low neutron absorption cross section. However, neutronic survey of the design it is needed. In this study, we were calculating the neutronic parametric using SRAC (PIJ-CITATION) module. The neutronics survey will be used to get the better design of LBE- cooled reactor in the future.

Keywords: Generation IV, Lead208-Bi eutectic, Modified CANDLE, neutronic survey,

Topic: Advanced small reactors without on-site refueling
[ABS-25]

IMPACT OF COOLANT (FLI AND FLIBE) VARIATIONS AT MOLTEN SALT REACTOR (MSR) FUJI-U3 AND MOLTEN SALT FAST REACTOR TYPE (MSFR)

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ABSTRACT

Molten Salt Reactor (MSR) and Molten Salt Fast Reactor (MSFR) is the fourth generation reactor design. Fuji-U3 is one of MSR developed by Japan. In this research neutronic, analysis on MSR Fuji-U3 and MFSR is done with its respective fuel adapted to specification standards in reactor design. The variation is to compare coolant (FLi and FLiBe) at FUJI-U3 and MSFR. The SRAC 2006 simulation tool with nuklida JENDL 4.0 is used for this research. Different simulation is run with different of coolant for MSR and MSFR until the condition for reactor reactivity is met. Moreover, the analysis based on conversion ratio, effective multiplication factor also investigated.

Keywords: k-eff, conversion ratio, reactivity and molten salt reactor.

Topic: Nuclear nonproliferation issues
[ABS-26]

TRANSMUTATION OF NUCLEAR WASTE PLUTONIUM AND MINOR ACTINIDES USING ACCELERATOR DRIVEN SUBCRITICAL SYSTEM WITH FUEL CONSISTS OF U-ThO2(NO3)2, PUMA(NO3)4 AND HNO3

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ABSTRACT

Nuclear waste should be manage to avoid the endanger environmental conditions. There are several methods of processing nuclear waste including geological repositories, reprocessing it to be used as fuel again or doing transmutation. In this study, the nuclear waste is transmuted by using an Accelerator Driven Subcritical System (ADS). ADS is a system consisting of a subcritical reactor core and a particle accelerator. Accelerated particles will pound the heavy nucleus, the reaction between the particles and the heavy nucleus is called a spallation reaction which produces spallation products. Nuclear waste which was processed in this study is Plutonium and Minor Actinides. The fuel consists of U-ThO2(NO3)2, PuMA(NO3)4 dan HNO3. Fresh fuel density depends on the concentration of this three solutions. In this study, concentration of the fuels solution were performed to make the reactor reach subcritical conditions in RGPuMA, WGPuMA and SGPuMA. Then also will be seen the changes of plutonium and minor actinide density in RGPuMA, WGPuMA and SGPuMA at 100 MWth and 200 MWth.

Keywords: ADS, Plutonium and Minor Actinides, Spallation Reaction, Transmutation

Topic: Innovative transmutation systems
[ABS-27]

ASSESSMENT OF DIFFUSION GAMMA AND NEUTRON DOSES BY COMPARING THE SIMULATION DATA USING MCNPX AND THE MEASUREMENT DATA DOSE USING DETECTOR INSTRUMENT, FOR THE PURPOSE OF OPTIMIZING THE RADIATION PROTECTION IN TRIGA 2000 BATAN.

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ABSTRACT

The use of all nuclear technology involving the radiation source practice need to be protected to avoid the risk that can affect the Worker, publics and Environment against the exposure. Defined by the International Atomic Energy Agency (IAEA), Radiation protection, also known as radiological protection is the protection of people from harmful effects of exposure to ionizing radiation, and the means for achieving this. One of the practices of nuclear techniques which the study will focus is the TRIGA (Training, Research, Isotopes, General Atomics) that can be defined as class of nuclear research reactor designed and manufactured by General Atomics. BATAN is one of Indonesia organization which use the TRIGA reactor. The study will focus on the assessment of the diffusion Gamma and neutron doses, and the optimization of the Radiation protection policy in the reactor TRIGA 2000 Bandung. There will be a comparison between the simulation data dose using Monte Carlo MCNPX with the measurement data dose using detector Instruments. A preventive dose reduction techniques, worker and publics protection will be elaborate to optimize the Radiation protection program.

Keywords: Doses, TRIGA 2000, Radiation protection, Simulation, Measurement, MCNPX

Topic: Radiation physics
[ABS-28]

STUDY OF CUT OFF ENERGY FOR ELECTRON GAMMA SHOWER (EGS) MONTE CARLO CODE IN GAMMA KNIFE PERFEXIONTM SIMULATION SYSTEM

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ABSTRACT

Gamma Knife Perfexion TM is radiosurgery procedure in tumor or cancer treatment. Gamma Knife uses 192 photon beam from cobalt-60 radioactive sources. For simulation of the treatment procedure, Monte Carlo (MC) as transport particle is the gold standard to analyze radiation characteristic in the matter. The code in Monte Carlo for photon/gamma beam is Electron Gamma Shower National Research Council (EGSnrc). The application will simulate particle from generating until loss of energy in certain energy cut off. The electron has rest mass in 521 KeV. In this research, the study of cut off energy above electron need to be observed to face in optimum calculation time in MC code. Gamma Knife collimator systems build in BEAMnrc code. Collimator 8mm is used to simulation due to the smallest condition. Specific material and geometry have been provided by the research team in Elekta Instrument AB Sweden. Source cobalt-60 generated at first component module with point source geometry. The scoring plane is on the last collimator channel. In case to answer the problem in this research, cut off energy for all material which is applied is 521 KeV and 700 KeV for group 1 and 2. In the group 1 scoring plane, numbers of particles are 1.189E-04 and 9.000E-07 for photon and electron respectively. For the second group, numbers of particles are 1.130E-04 and 9.000E-07 for photon and electron respectively. The number of particles is quantitative analysis. Qualitative analysis in the scoring plane is particle energy. Each cohort has similar energy such as 1.131 and 1.133 MeV. Calculating time for each history particles is 0.00116 and 0.00011. Lower cut off group has the longer time than Group 2 because of low energy tracking. Two groups have similar results but need different calculation team. For the number of particles, the differences are 5%. On the other hand, calculating time is 10 times faster than group 1. By this result, applying 700 KeV for the cut of electron energy is better than 521 KeV.

Keywords: Gamma Knife, EGSnrc, Cut off, calculation time

Topic: Radiation physics
[ABS-29]
STUDY DESIGN OF HTGR WITH THORIUM BASED FUEL AND ZRC TRISO-COATED PARTICLE (TRIZO)

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ABSTRACT

In this study will be carried out a study of the neutronic and thermal hydraulic aspects of HTGR reactors with power and fuel variations. HTGR or High-Temperature Gas-cooled Reactor is a helium-cooled high-temperature reactor and graphite moderator. The use of helium as a substitute for CO2 as a cooler combined with graphite moderator offers increased neutron and thermal efficiency. The combination of helium as a coolant and graphite as a moderator allows producing high temperatures and hence is called High-Temperature Gas-cooled Reactor (HTGR). The reactor is designed with a power of 30 MWT-300 MWt and Thorium based fuels. The fuel used is TRISO CFP or Triple-isotropic Coated Fuel Particle or coated fuel particles that have the function to counteract chemical attacks that cause corrosion and barrier from fission product release. In this coated fuel particle, zirconium carbide is used as a substitute for silicon carbide. This variation is expected to overcome the weakness of SiC layer at high temperatures. The reactor core design used in the calculation is 3D Triangular with multigroup diffusion calculation and burn-up analysis using the 2006 version of the SRAC (Standard Reactor Analysis Code) code developed by the Japan Atomic Energy Research Institute (JAERI) with the JENDL (Japanese Evaluated Nuclear) nuclear database Data Library) -4.0

Keywords: HTGR, Neutronic, Thorium, SRAC, ZrC

Topic: Nuclear data
[ABS-30]
CHARACTERIZATION OF SOURCE OF GAMMA KNIFE PERFEXIONTM BASED ON MONTE CARLO METHOD

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ABSTRACT

Characterization of source of Gamma Knife PerfexionTM is done by varying the energy selection of 60Co. Energy 60Co is used as input to Monte Carlo based simulations using software EGSnrc. In this research five energy variations of 60Co were chosen. The results obtained are the spectral distribution curve of the relationship between energy and the relative intensity of the particles. Rogers spectrum distribution is used as validity comparator data of research results. After commissioning simulation results data with comparative data, significant results were obtained with a difference of <2.5%. So it can be concluded that the characteristics of the source produced and the characteristics of the comparative data are the same.

Keywords: characterization of source, Gamma Knife PerfexionTM, Monte Carlo simulation

Topic: Radiation physics
Study startup Helium Cooled Fast Reactor long live with natural uranium has been done. The reactor core is designed with a power of 800 MWth with fuel fractions 50, 55, 60 and 65%. Calculation using SRAC system code and JENDL-40 library, with cylindrical cell two dimensional R-Z core models. Reactor design optimization is evaluated to utilize natural uranium as reactor fuel. Optimization evaluated by burning natural uranium for 40 years and put each of its burnup result per 2 year, with scheme modified CANDLE (Constan Axial shape of Neutron flux, nuclide densities and power shape During Life of Energy producing reactor). In Modified CANDLE burn-up scheme strategy, the active core is subdivided into ten regions (region-1 until region-10) with the same volume in the axial direction and 2 region in the radial direction. The fresh fuel such as natural Uranium is initially put in region-1 and another region filled mixed spent fuel and natural Uranium, after one cycle of 10 years of burn-up it is shifted to region-2, region-2 to region-3 until region 10 and then region-1 is filled by fresh natural Uranium fuels. Excess reactivity at beginning operation more than 1%. Reactor with fuel fraction 50;55% obtained optimal design. The ratio of peak to average power density is 1,57 in radial and 2,57 in axial direction with power density 300 W/cc.

**Keywords:** modified Candle, Spent Fuel, Power Flatening

**Topic:** Innovative nuclear energy systems
[ABS-32]
FAST NUCLEAR REACTOR BURN-UP ANALYSIS USING FOURTH ORDER RUNGE-KUTTA METHODS

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ABSTRACT

Burn-up calculations involve the processing of nuclear data as microscopic cross section. Macroscopic cross section calculation was done by multiplying the microscopic cross section with the nuclide density and all materials involved in the reactors itself. The calculation of diffusion equation was done in two dimensional cylindrical coordinate system, involving the macroscopic cross section to obtain the distribution of neutron flux, neutron source density, power density and k-eff. The calculation gives k-eff = 1.0017918649. Thus, it can be concluded that the reactor is in a super critical state. Subsequently, we use maximum flux to be calculated in the burn-up equations to obtain new nuclide density over 10 years. The process of burn-up calculation involves a system of 28 differential equation, generated from nuclear transmutation chain. It takes a long time and is hard to be solved analytically. In this thesis the system of differential equations was numerically solved by The Fourth Order Runge-Kutta Methods to obtain the curves of nuclide density versus time.

Keywords: Burn-up, One Speed Diffusion, Fourth Order Runge-Kutta Methods, Transmutation Chain

Topic: Theoretical and computational nuclear physics and particle physics
[ABS-33]

ANALYSIS OF OXIDATION PERFORMANCE IN HELIUM PURIFICATION SYSTEM OF THE INDONESIA EXPERIMENTAL POWER REACTOR

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ABSTRACT

RDE is an Indonesia experimental power reactor which is designed based on the high-temperature gas cooled reactor (HTGR). RDE is designed to generate the thermal power as high as 10 MWT. As an HTGR type reactor, the primary coolant of RDE is a helium gas. Helium gas in the primary coolant system has to be purified in the helium purification system (HPS) to assure the thermal characteristic related to the safety or reactor. HPS serves to separate and to remove the primary coolant gas impurities such as H2, H2O, CH4, CO, CO2, N2, and O2 in order to be under a certain concentration. H2 and CO gases are substances that are difficult to separate and to remove from the primary coolant. Oxidation of gaseous H2 and CO to be H2O and CO2 gas is an alternative process to separate and to remove H2 and CO from the primary coolant. The objective of the research is to analyze the process of oxidation of H2 and CO using CuO oxidizer in the HPS. Analyses were performed by taking into account the reactants characteristics, reaction rate, primary coolant flow rate, the concentration of H2 and CO, and the volume of column oxidizer. The analysis results showed that the optimal of the oxidation reaction is determined by the reaction of the H2 gas with CuO, the faster of reaction will be achieved at the higher temperature. The optimal reaction of oxidation will occur on the temperature 300 0C. The other result showed that in the volume of oxidizer column 1200 liters, the mass of CuO 7572 kg, and the lifetime of oxidizing 10,000 days will give the optimal reaction oxidation of H2 and CO in the HPS.

Keywords: RDE, HPS, purification, oxidation, performance

Topic: Innovative nuclear energy systems
A COMPARATIVE STUDY OF TH-U233 AND TH-PU MOLTEN-SALT REACTOR (MSR) WITH 2-REGION CORE

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ABSTRACT

Sustainable closed fuel cycle and inherent safety are one of the key aspects of Generation IV reactor design. Hence Molten salt reactor (MSR) is chosen as one of the candidates of Generation IV reactor concepts. FUJI-U2 or MSR-2R is a 2-region core MSR design that was proposed as a possible simplification of FUJI-U3 (3-region core). Taking into consideration that uranium-233 doesn’t exist naturally, FUJI-U2 utilize both plutonium and uranium-233 as a fissile material. To understand the characteristics further, a comparative study of 2-region core MSR with Th-U233 and Th-Pu as fuel has been carried out. Nuclear analysis code SRAC2006 with the nuclear data library of JENDL4.0 was used, and CITATION was employed to validate the core. The design parameters set for each region are fuel composition, core radius, and fuel fraction. The simulation burn up calculation is set to 2000 days in total, with 21 step each for 100 days cycle. U-233 fueled FUJI-U2 core 1 and core 2 radius are 5.10 and 3.95 cm respectively, with the fuel fraction of 31.5 vol% and 18.9 vol%. As for plutonium fueled FUJI-U2, each core radius are 4.75 and 4.40 cm with the fuel fraction of 27.3 vol% and 23.4 vol%. Temporary results shows that Kinf of uranium-233 fueled FUJI-U2 for core 1 and core 2 are 1.032075 and 1.122150 . While K-infinity of plutonium fueled FUJI-U2 core 1 and core 1 are 1.018639 and 1.059092, respectively. Burn up calculation, conversion ratio, fuel elemental and isotopic composition, power density, and neutron spectrum of each core will also be discussed.

Keywords: k-infinity; molten salt reactors ; 2 region core

Topic: Nuclear data
[ABS-35]
CORE ANALYSIS OF LONG LIFE PWR USING THORIUM CYCLE

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ABSTRACT

A conceptual study of long life thorium cycle fuel based PWR has been conducted. It aimed to evaluate how the variation of several parameters affects the neutronic behavior of reactor core. The simulation is performed using PIJ of SRAC Code System based on library SRACLlib-JDL40. The parameters acted as independent variables are fuel fraction and burnable poison concentration. These parameters are adjusted in such a way that the reactor with thermal power of 100 – 500 Mwt could be operated up to 10 – 15 years without refueling. The result show that the excess reactivity could be minimized by adding burnable poisons to fuel and increasing the fuel volume fraction.

Keywords: PWR; thorium cycle; fuel fraction; burnable poisons; excess reactivity; reactivity swing

Topic: Advanced small reactors without on-site refueling

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ABSTRACT

The present work discusses two different models of thorium irradiation in the TRIGA 2000 reactor core to compare the neutronic characteristics of irradiation of thorium dioxide (ThO2) and thorium zirconium hydride (ThZrH1.6). The Monte Carlo N-Particle (MCNP) Transport code, based on the Monte Carlo method, is used to design three dimensional models for TRIGA reactor core at typical operating power 2 MW. These models are used to determine the effective multiplication factor (k_eff) and neutron flux distribution. The ORIGEN code was used to calculate the 233U isotope production. It is found that decreasing of k_eff from the effect of hydride irradiation rod is a little lower than the oxide irradiation rod.

Keywords: irradiation, ThO2, ThZrH, TRIGA reactor, MCNPX, ORIGEN

Topic: Innovative nuclear energy systems
[ABS-37]
NEUTRON CONTAMINATION FF AND FFF ELEKTA AGILITY 10 MV PHOTON BEAM

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ABSTRACT

High energy Linear Accelerator (Linac) photon mode (E≥10 MV) can produce neutron as
unexpected contaminant. Neutrons are produced from the photoneutron interactions
between high-energy photons and the Linac head components. Although the number of
neutrons produced are very small compared to photons or electrons but in neutron
radiation protection field, it is necessary to be evaluated because it has high
radiobiological effect (RBE) value that can be more destructive compared to photons or
electrons. The neutron spectrum produced by the Linac head was evaluated using a
Monte Carlo simulation (MC). The geometry and composition of the material from the
Linac head are modeled based on information from the manufacturer. MCNP version 6
software is used to model the Linac head and the photon transport process to produce
neutrons. The simulated field size is 10x10 cm2 in Source to Surface Distance (SSD) = 100
cm. Evaluation of neutron contamination was carried out for the Linac with the
Flattening Filter (FF) and without FF or the Free Flattening Filter (FFF). The scoring plane
(the calculated neutron calculation area) is placed 1 cm above the phantom. The
neutron type produced by the head Linac Elekta Agility 10 MV photon mode is mostly
thermal and fast neutron. Although there are differences in the neutron intensity of FF
and FFF, but the type of neutron produced by the two modes have the same energy.
Based on the neutron photo reaction energy threshold it can be concluded that the
neutrons produced from the head Linac Elekta Agility are the result of interactions
between photons with isotopes 65Cu, 96Mo and 184W due to the photoneutron
interaction energy for the three isotopes it is less than 10 MV. Head Linac 10 MV photon
mode can produce neutrons as a contaminant beam. Neutrons are produced by the
photoneutrons interaction between high-energy photons and components in the Linac
head. There are differences in intensity of neutrons produced between FF and FFF
modes, but not for the quantitative analysis for neutron spectrum.

Keywords: Neutron, Elekta Agility, MCNP, Flattening Filter

Topic: Radiation physics
[ABS-38]

EVALUATION OF THE CHARACTERISTIC OPERATION OF THE PRIMARY COOLING PUMP OF BANDUNG TRIGA 2000 RESEARCH REACTOR

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National Nuclear Energy Agency of Indonesia

ABSTRACT

The Bandung TRIGA 2000 research reactor has been operating for 18 years. Currently, there are plans to convert the reactor core from cylindrical fuel element to plate type fuel element. The conversion process is expected not to change too many existing systems, including the primary cooling systems. For this reason, it is necessary to evaluate the pump capability of the primary cooling system. Based on experimental studies, then merging with the existing installation system head, it has been obtained the first pump (PR001) and the second pump (PR002) mass flow rate of 45 and 45.5 kg/s, respectively. The pump capability decreased by around 5% compared to the results of its commissioning. If the two pumps are combined in parallel then the combined mass flow rate can increase to 62 kg/s. These results need to be considered in designing a new core with primary cooling systems that are slightly modified because they have to add the delay tank.

Keywords: primary pump, reactor conversion, pump capability, mass flow rate, delay tank

Topic: Innovative nuclear energy systems
[ABS-39]  
STUDY ON THE EFFECTS OF ENRICHMENT AND FRACTION OF COATED FUEL PARTICLES ON FISSION UTILIZATION OF 100 MWT PRISMATIC-TYPE OF HIGH TEMPERATURE GAS REACTOR

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ABSTRACT

High Temperature Gas-cooled Reactor (HTGR) is one of Generation-IV reactor technology that has graphite moderate, Helium gas cooled, and Coated Fuel Particle (CFP) layered by Tristructural-Isotropic (TRISO). The HTGR has outlet temperature of around 1000°C that can be utilized for many co-generation processes other than to generate electricity. Due to characteristics of the CFP and TRISO, utilization of fissile material during reactor operation become important. The aim of this study is to analyze the effects of enrichment and fraction of Coated Fuel Particle (CFP) on the fissile utilization of 100 MWT HTGR for three different fuels; \( \text{UO}_2 \), (Th-U)\( \text{O}_2 \) and (U-Pu)\( \text{O}_2 \). Fissile enrichment are analyzed from 1-20% while fraction of CFP from 10-60%. Using SRAC2006 code and JENDL-4.0 as nuclear data library, calculation and analysis are performed to find the optimal values of several important neutronic parameters.

Keywords: CFP, enrichment, HTGR, TRISO

Topic: Innovative nuclear energy systems
[ABS-40]

NEUTRONIC ANALYSIS OF TOKAMAK FUSION REACTOR WITH LIF AS A TRITIUM BREEDER

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ABSTRACT

The recently developed fusion reactor is the Tokamak reactor, ITER, which was built in France as a world project. Many studies have been conducted, one of which is a blanket design simulation by MCNP (Monte Carlo N-Particle) for neutron calculations. The reactor is assumed to be in static operation, whether in steady state or initial transient conditions. The neutron calculation results will obtain neutron flux distribution, neutron dose and power distribution when the reactor is in steady state, which is then used as the initial input of the neutron calculation temporarily. Meanwhile, Tokamak has difficulty optimizing tritium breeding. At present, the TBR (Tritium Breeding Ratio) only reaches less than one, which requires at least one to maintain the condition of the self-sufficiency reactor. This research idea is to match lithium with molten salt, FLiBe or FLiNaK, leading to TBR equal to 1.15. To optimize TBR, variations are needed, namely: first, variations in blanket size which affect the composition of the material in it; second, variations in material are used as breeding tritium, neutron multipliers to improve the welfare of fusion-fusion fuels; third, variations in coolant that can transfer heat to the reactor correctly.

Keywords: Flibe, Flinak, MCNP, Molten Salt, Steady-state, TBR

Topic: Innovative nuclear energy systems
[ABS-41]

ACTINIDE MINOR ADDITION ON URANIUM PLUTONIUM NITRIDE FUEL FOR MODULAR GAS COOLED FAST REACTOR

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ABSTRACT

This study has made a calculation of Actinide Minor Addition on Uranium Plutonium Nitride Fuel for Modular Gas Cooled Fast Reactor. The purpose of this study was to determine the characteristics of minor actinides when incorporated into plutonium nitride uranium fuel. Neutronics calculation was design by using SRAC Code version 2006 (Standard Reactor Analysis Code) with the data nuclides from JENDL-4.0 under the Linux Operating System with nuclear data library JENDL4.0. Neutronic calculations were done through some steps of parameter survey to obtain the ultimate optimization results. The initial calculation was calculation of fuel cell (PIJ calculation) by using hexagonal cell and then followed by calculation of core reactor (CITATION calculation) by using program code SRAC2006. The addition of minor actinide has been calculated in this research. The minor actinides which were added were americium (Am-241 and Am-243) and neptunium 237. They were put into the uranium plutonium nitride material to decrease the k-eff value. The addition of minor actinide into the reactor aimed to reduce the number of minor actinides in the world. Minor actinide is a nuclear waste fuel or often called spent nuclear fuel (SNF) which has high toxicity. Neptunium 237 is a minor actinide with the highest percentage of SNF.

Keywords: Minor Actinide, Uranium Plutonium Nitride, Gas Cooled Fast Reactor

Topic: Advanced small reactors without on-site refueling
[ABS-42]
THE EFFECT OF DOPANT CONCENTRATION ON THE PENETRATION DEPTH OF BETA PARTICLES ON 147PM-GAN NUCLEAR BATTERY

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ABSTRACT

Beta particles emitted from the 147Pm source which pound the GaN p-n junction can be simulated with the monte carlo method. The penetration depth is one of the important parameters that can be calculated. The depletion region created in the p-n junction is determined by how far the beta particles can enter the material. The electrons emitted will be scattered until they reach the minimum energy. In this study the concentration of dopant and the thickness of the layer in the GaN p-n junction were varied. The monte carlo method in this simulation is completed using CASINO v.2.48. The output of this simulation has a fairly wide application including the design of betavoltaic nuclear batteries.

Keywords: Penetration Depth; Nuclear battery; Monte carlo

Topic: Theoretical and computational nuclear physics and particle physics
[ABS-43]
DEVELOPMENT OF FISSION PRODUCT RELEASE ANALYSIS IN TRIAC-BATAN

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ABSTRACT

A triso analysis code, called TRIAC-BATAN, is being developed to support the goal to acquire the design and safety analysis capability of a pebble bed reactor at BATAN. Previous development is limited to the triso fuel failure analysis which able to provide the triso fuel failure ratio for certain geometrical, physical, and reactor normal and accident condition. This paper presents the development and initial benchmark of fission product release analysis in TRIAC-BATAN. The fission product release and transport in this code considers the important phenomena for fission product behavior in pebble bed reactor, including the recoil and release of fission products from the fuel kernel, transport through the coating layers, transport through the surrounding fuel matrix. The above fission product transports are performed for an intact and failed fuel in which the number of failed fuel versus time is achieved from the previously available modul of TRIAC-BATAN. The transport is performed using a diffusion-based model. Initial benchmark results of the code for the IAEA CRP-6 Benchmark case 3 is presented in this paper.

Keywords: fission product release; triac-batan; pebble bed reactor

Topic: Theoretical and computational nuclear physics and particle physics
[ABS-44]
STUDY ON TRISO FUEL FAILURE RATIO OF REAKTOR DAYA EKSPERIMENTAL

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ABSTRACT

Safety goal of Reactor Daya Eksperimental, an Indonesia pilot plant of pebble bed reactor being developed by BATAN, is to minimized radioactive release below the allowable limit. The first and most important barrier of the radioactive release in the pebble bed reactor type is the tri-isotropic (TRISO) coated particle. This paper presents a triso fuel failure analysis of the RDE design in normal operation and in depressurized loss of forced cooling (DLOFC) accident which is assumed as one of the severest hypothetical accident. A newly developed triso fuel failure analysis code, called TRIAC-BATAN, is utilized in this calculation. TRIAC-BATAN employed a vessel-model to define the triso fuel in which a triso is failed if the pressure on its Silicon Carbide layer is larger than ductile strength of that layer. Fuel temperature history along the normal operation and in accident condition are the main input for the fuel failure analysis beside the geometrical and physical specification of the fuel. These fuel temperature histories are previously calculated from the neutronic and thermal hydraulic module of PEBBED code. Uncertainty of the geometrical data of the triso fuel also considered in the fuel failure analysis. Calculation results shows that in the normal operation no triso fuel failure expected to occurred, while in the DLOFC accident the ratio increase to ~10-13 which is still a negligible ratio which shows a good confinement capability of the RDE triso fuel.

Keywords: triso fuel failure anlaysis, triac-batan, reactor daya eksperimental

Topic: Innovative nuclear energy systems
[ABS-45]
INVESTIGATION OF ZIRCONIUM CARBIDE CERAMIC COATING ON SURROGATE MICROKERNELS DEPOSITED BY PLASMA-PULSED LASER DEPOSITION: A PRELIMINARY STUDY

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ABSTRACT

High Temperature Gas-cooled Reactor (HTGR) is one of advanced nuclear reactors with high safety system feature. Indonesia has a plan to build HTGR as an Experimental Power Reactor (RDE-Reaktor Daya Eksperimental). One of the safety system feature is its fuel system. There are two types of fuel of HTGR i.e. pebble bed type and prismatic type which all of the types are containing TRISO (Tri-Structural Isotropic) which consist of Inner Pyrolitic Carbon, SiC (Silicon Carbide) dan Outer Pyrolitic Carbon. One of the issues of the system is silver (Ag) and palladium (Pd) as the fission product could interact with SiC layer and cause corrosion. One of the candidate to resolve the issue is to replace SiC with ZrC (Zirconium Carbide). In this preliminary study, Zirconium Carbide ceramic has been deposited on a surrogate (uranium-free) fuel microkernels using Plasma-Pulsed Laser Deposition (PLD) at Center For Science and Technology of Advanced Materials laboratory – National Nuclear Energy Agency of Indonesia (BATAN). ZrC was deposited with the chamber pressure around 235 mTorr, the substrate temperature of 850°C, the number of laser shots of 90,000 and the oxygen background gas with 40 sccm (standard cubic centimeters per minute). Afterward, the samples were analyzed using Microscope Optic (OM), Scanning Electron Microscope – Energy Dispersive X-ray Spectroscope (SEM-EDS) and XRD (X-Ray Diffractometer). The results showed that most of the Zirconium Carbide deposition was react with oxygen at high temperature on the surface of the sample and deposited on the surface of surrogate microkernels.

Keywords: HTGR, ZrC, Ceramic, PLD, Surrogate

Topic: Material and process for innovative energy systems
[ABS-46]
THERMODYNAMICAL AND STRUCTURAL PROPERTIES OF SiC CERAMIC MATERIAL IN LIQUID LEAD METALS

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ABSTRACT

It has been investigated the thermodynamical and structural properties of SiC ceramic material in high temperature molten liquid lead. Th research was done for the purpose of finding the characteristic SiC ceramic material when immersed and interact with liquid lead. Knowing its characteristics have benefit for application of the material in nuclear reactor design. The research is done using the molecular dynamics method to predict some of properties of the ceramic materials as diffusion coefficient, CNA common neighbour analysis, RDF curve, etc.

Keywords: Nuclear materials, SiC, ceramics, liquid lead, molecular dynamics

Topic: Material and process for innovative energy systems
CANDLE reactor is one of an innovative fast reactor design that employed with specific burn-up strategy. An initial core should be prepared which is contracted a special reactor for the first several years. After initial core have burned, the new fuel for the remaining cores is generated which can be sustained to use only by natural uranium. The fuel breeding using U10Zr have been investigating to obtain better CANDLE burn-up performance and it proposed the ease of designing a long life reactor. In this study, we also used lead bismuth cooled as the survey parameter to develop the coolant effect for neutronic performance with thermal power output is 2000 MW. Reactor calculation have been performed with Monte Carlo method, MCNP code, to determined detail analysis calculation. This reactor design demonstrated the sustainability of uranium fuel to produce plutonium as breeding fuel concept. The result of the calculation showed that spent fuel burnup is about 400 MWd/tHM during 80 years operation, it means, about 40% of natural uranium burns up without enrichment. This value is competitive which presently planned by fast reactor systems.

Keywords: CANDLE reactor, lead bismuth cooled, fast reactor, burn-up performance

Topic: Innovative nuclear energy systems
[ABS-48]
AGENT-BASED MODEL IN RECTANGULAR GRID FOR
CONDENSABLE GRANULAR GAS SIMULATION

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ABSTRACT

Particles of granular gas are represented as point in two-dimensional grid. Each point can move randomly with tuneable probability. By setting constant value of left, down, right, up motion probability, several phases as solid, liquid, gas, and granular pile can be obtained. In this work the probability is considered also as function of neighbour points in order to accomodate particle-particle attraction and repulsion forces, and also surface tension to perform a cluster of particles. Temperature works in averaging the probability to all direction since gas state can be achieved only in higher temperature, when other forces are negligible.

Keywords: agent-based model, granular gas, simulation, toy model, nuclear physics

Topic: Theoretical and computational nuclear physics and particle physics
[ABS-49]

ACCIDENT ANALYSIS DUE TO FUEL TEMPERATURE INCREASE IN RDE REACTOR CORE

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ABSTRACT

The 10 MW Indonesia’s Experimental Power Reactor (RDE) is a High Temperature Gas Cooled Reactor (HTGR)-type and planned to be operational in the Puspiptek Serpong area in 2022/2023. The reactor applies helium gas coolants, graphite moderator, 17% 235U enrichment fuels and has 8 control fuel rods. To design a nuclear reactor, there are many safety aspects which should be taken into account and one of them is neutronic safety analysis dealing with an accident of reactivity core change due to increases of fuel and moderator temperatures. To begin with the safety analysis, the RDE core should be modeled utilizing the combination of nuclear data libraries and a computer code. Prior to estimate the reactivity core change in the RDE core, the calculation of neutron effective multiplication factors during the accident was accomplished using the MCNPX computer code. In this paper, the neutronic analysis of RDE core has been focused on how the fuel temperature increase implies the reactivity core change in the RDE reactor. By combining the three major world nuclear data libraries and the MCNPX code, the all calculated results showed that all core reactivity changes in the RDE core due to fuel temperature increases starting from 26.85 °C (300 K) to 2,726.85 °C (3,000 K) are all negatives except those in the reflector zone which principally accumulates generated neutrons always coming from the central of the core during reactor operation. Therefore, those does not affect at all the safety of the reactor core during the accident. Indeed, the RDE reactor core is totally safe in the event of fuel-temperature-increase during its reactor operation and hence the RDE reactor is in a steady, safe operation.

Keywords: Accident analysis, fuel temperature increase, RDE, MCNPX

Topic: Innovative nuclear energy systems
[ABS-50]

VARYING VISCOSITY POTENTIAL IN LONGITUDINAL MICROTUBULES DYNAMICS

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ABSTRACT

Longitudinal Microtubules (MTs) dynamics are investigated by taking several potential functions into a standard Hamiltonian model, that described damping effect to the growing of MTs. The analysis of length contraction in MTs is useful to describe the interaction MTs with surrounding liquids. The model proposed might be a representation of physical phenomenon in the case of broken MTs because of infected cell. A hybrid analytical-numerical methods are conducted to solve the dynamical equations. The evolution of energy profiles under different potentials of viscosity regime will be analyzed. The obtaining results is how effective the model to describe the phenomenon.

Keywords: Microtubules, Hamiltonian Model, Numerical Methods.

Topic: Theoretical and computational nuclear physics and particle physics
[ABS-51]
SIMULATION OF PHASE CHANGE INTERFACE IN STEFAN PROBLEMS WITH FINITE VOLUME PARTICLE (FVP) METHOD

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ABSTRACT

Heat transfer involving phase change is one of the importance in many fields in engineering and industry. In this study, a particle-based method using Finite Volume Particle (FVP) Method was developed to simulate heat transfer and phase change interface of Stefan problems. As the method implies, the calculation domain was represented by particles of size 1 mm. The equilibrium phase change model was used to simulate heat transfer between particles. Two cases were simulated, i.e. solidification of infinite cylinder water in freezing temperature, and solidification in infinite corner. The simulation results were then compared to the analytical solution and other simulation methods, and showed relatively good agreement. This particle method even provide a more straightforward phase change interface than that of the conventional mesh methods.

Keywords: stefan problems, finite volume particle method, phase change

Topic: Theoretical and computational nuclear physics and particle physics
[ABS-52]

SAFETY ASSESSMENT OF PRESSURE VESSEL COMPONENTS FOR REAKTOR DAYA EKSPERIMENTAL

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ABSTRACT

A small modular nuclear reactor with passive safety features is one of the best solutions to support the Indonesia sustainable energy development. The 10 MW Indonesia’s Reaktor Daya Eksperimental (RDE) which is to be constructed in Serpong Nuclear Zone, Puspiptek and was firstly developed in 2014 is expected to be operational in 2022/2023. The RDE is dedicated as a basic reference and milestone for Indonesia to become a Nuclear Power Technology Provider in the near future. This research paper presents structural stresses analyses on design of reactor pressure vessel components for Indonesia’s RDE in ensuring the safety design targets of RDE that are no dangerous radioactive release to the workers, populations, and environments even in the severest accident. Results are presented from the structural stress assessment under following conditions such as thermal loading, operating pressure, and mechanical loads. In previous work, all calculations of geometry models and dimensions of pressure vessel components in the design computation had been performed by using computer code of RPV_RDE.exe. These components were designed in accordance with the ASME code specification for SA516-70 carbon steel. In the present work, finite element analysis is used to assess the occurred structural stresses. This work recommends that the pressure vessel components of RDE are safe in the design calculations by analysis approach as described in model simulation.

Keywords: Reactor pressure vessel, reaktor daya eksperimental, structural components, stresses analyses

Topic: Innovative nuclear energy systems
[ABS-53]
THE EFFECT OF TEMPERATURE VARIATIONS ON WOOD’S METAL PLATE MELTING PROCESS BY USING MPS

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ABSTRACT

Two-dimensional simulation of Wood’s Metal Plate (WMP) Melting Process has been conducted by varying the molten temperature. This simulation was performed by using Moving Particle Semi-Implicit (MPS) method which can calculate the Navier-Stokes and heat transfer equation without divining mesh system. As a substitute, particles are selected to represent every part of simulation such as fluid, wall and dummy type. The experiment of WMP melting process has been conducted by Sudha (2018). The molten Wood’s Metal was originally set at 573 K. In this simulation, the hole formation in WMP will be investigated as a consequence of molten temperature variation. Furthermore, temperature distribution at top side of WMP and solid-liquid phase change will be analyzed for initial temperature of molten are 523 K, 473 K and 423 K.

Keywords: Heat; Hole; Melting; Particle; Temperature

Topic: Innovative nuclear energy systems
IMPLEMENTATION AND OPTIMIZATION OF ULOF ACCIDENT ANALYSIS FOR GCFR PROGRAM IN CLUSTER COMPUTER

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ABSTRACT

Accident analysis code is important to investigate inherent safety performance of next generation Nuclear Power Reactors. are standard methodology to investigate the performance of Nuclear Power Plants during their operations. In this study Two dimensional accident analysis code of ULOF accident for gas cooled fast reactors has been developed and optimized for cluster computers. The analysis includes transient space time kinetics and transient thermal hydraulic which are tightly coupled during implementation. The space time kinetics is implemented using adiabatic approach and together with transient thermal hyraulic equations are discreetized and implemented on cluster computers using fortran language and MPI approach. Some part of the program is a tightly coupled problem while some other part in nature is loosely coupled problem. Therefore we need optimization process for overall simulations in the cluster computer. From our simulations results the optimization of space time kinetic and transient thermal hydraulic timing scheme will significantly influence the acceleration performance during implementation in the cluster computer up to 40 cores.

Keywords: Accident analysis, ULOF, space time kinetic, transient thermal hydraulic

Topic: Innovative nuclear energy systems
THIN FILM DEPOSITION SIMULATION USING LAMMPS COUPLED WITH THIN FILM YSZ EXPERIMENT USING PULSED LASER DEPOSITION (PLD)

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ABSTRACT

Barium Strontium Titanate (BST) thin films have been experimentally deposited using a Pulse Laser Deposition (PLD) method. PLD is a physical deposition method where a laser beam is used to evaporate a target (BST) to be deposited on a substrate in a vacuum chamber. Here, the BST vapour expanded in a plasma plume to the surface of the substrate due to Coulomb repulsion and recoil from the target surface. The substrate was heated at the temperature of 700°C, 750°C, and 800°C, the oxygen pressure of the chamber was set at 200 mTorr. The results of the BST films were characterized using Atomic Force Microscope (AFM) and illustrated in 2D and 3D. In order to model this deposition process and possibly be able to predict the morphology and structure of deposited films using computational methods, a molecular dynamic Large-scale Atomic/Molecular Massively Parallel Simulator (LAMMPS) code was used. The BST vapour was simulated as single crystal Ag nanoclusters moving to the substrate. The cluster diameters were set to 3 nm or 5 nm, cluster velocities set to 200 m/s, 400 m/s, or 800 m/s, and the deposition rate adjusted to ensure relaxation times between impactions of 5 ps to 20 ps. The results of the simulation were illustrated using Visual Molecular Dynamics (VMD) software and directly shown as images in .png format.

Keywords: Pulse Laser Deposition, Atomic Force Microscope, Visual Molecular Dynamics, LAMMPS.

Topic: Material and process for innovative energy systems
[ABS-56]
CORE PHYSICS CHARATERISTICS OF TRISO FUEL FOR INDONESIAN HTGR DESIGN

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ABSTRACT

Indonesia plans to build an experimental 10 MWt power reactor (Reaktor Daya Eksperimental, RDE) based on inherently safe high temperature gas-cooled reactor (HTGR) technology in the Center of Development of Science and Technology Area in Serpong, South Tangerang. The RDE was initiated at the end of 2014, and the Basic Engineering Design was completed in 2017. Currently the development program is focused on the Detail Engineering Design. The type of fuel aimed in the design is pebble type that contains fuel kernels made of uranium oxide coated in four layers of three isotropic materials. Those layers are a porous buffer layer made of carbon, followed by a dense inner layer of pyrolytic carbon (PyC), followed by a ceramic layer of Silicon Crabide (SiC) to retain fission products, and followed by a dense outer layer of PyC. One of the issues of the fuel system is the problem of corrosion caused by the interaction of fission products such as silver (Ag) and palladium (Pd) with SiC layer. One of the candidate to resolve the issue is to replace SiC with ZrC (Zirconium Carbide) which is known to be more resistant to corrosion and high temperatures. In this preliminary study, we examine the effects of this material replacement on the core physics characteristics of the RDE at different operating temperatures. For the purpose we use SRAC Code sytem to calculate the multiplication factors of the spherical fuel design.

Keywords: HTGR, RDE, SiC, ZrC, SRAC Code, core physics

Topic: Innovative nuclear energy systems
[ABS-57]
COMPARATIVE ANALYSIS ON CORE PERFORMANCE OF MOLTEN SALT FAST REACTOR (MSFR) FOR DIFFERENT FUEL LOADING TYPES

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ABSTRACT

Nuclear energy as an alternative energy source can be used to fulfill the energy demand of Indonesia and can be fitted with an energy mix policy for medium and long-term energy policy. Nuclear energy has a better energy density, which has some stockpiles in the earth crust including in Indonesia. In the present study, Molten Salt Fast Reactor (MSFR) as a power plant for alternative energy was introduced as a fast spectrum neutron reactor with having some start-up mechanisms. An established computer code of SRAC (Standard thermal Reactor Analysis Code System) was used for the analysis. Some different fuel loading are used and compared such as is ThF4 + U233F4, ThF4 + PuF3, and ThF4 + (TRU)F3. Some basic parameters are evaluated including the effective multiplication factor, conversion ratio, and different temperature effect and volume of molten.

Keywords: doppler coefficient, effective multiplication factor, MSFR, thorium, uranium

Topic: Innovative nuclear energy systems
[ABS-58]  ANALYSIS ON NEUTRONIC ASPECT AND NUCLIDE COMPOSITION IN MOLTEN SALT REACTOR (MSR) OF FUJI-U3 REACTOR TYPE

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ABSTRACT

Generation fourth reactor is becoming an interesting topic to be studied and evaluated especially some advanced reactor type and some innovative reactor system. Molten salt reactor of Fuji-U3 was designed which has a higher safety level. The fuel was designed in a form of molten that the fuel is in liquid phase form, which is drastically different with conventional reactor which uses a solid fuel form. In the present study, an analysis on neutronic aspect of MSR of Fuji U3 type by adopting computation code of SRAC code with some nuclear data library of JENDL-3.2, JENDL-3.3, dan JENDL-4.0. A typical fuel composition of 7LiF – BeF2 – ThF4 – UF4 (71.76 %mol – 16 %mol – 12 %mol – 0.24 %mol has been used for the analysis. Some parameters have been evaluated including multiplication factor for reactor operation condition analysis, fuel conversion ratio for evaluating fuel sustainability aspect as well as nuclide density composition of fuel. Some different fuel temperature degrees have been used to evaluate this effect to the reactor performance especially for evaluating Doppler effect as well as the volume fraction effect to the reactor condition.

Keywords: Fuji U3, MSR, Nuclear data, Doppler coefficient, effective multiplication factor, thorium

Topic: Innovative nuclear energy systems
[ABS-59]

ANALYSIS ON NATURAL BACKGROUND RADIATION AND DOSE RATE IN BANDUNG CITY AND ITS SURROUNDING AREA

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ABSTRACT

Natural background radiation is the largest contributor to the radiation dose received by humans in one year. Natural background radiation arises because of the presence of natural radionuclides in the environment. Natural radionuclides can be found in water, soil, air, material around volcanoes, etc. Radiation that reaches a certain threshold can be harmful to human health. Bandung is one of the largest cities in Indonesia surrounded by many mountains. The city is full of various human activities. In addition, there is an active research reactor in the city. Therefore radiation monitoring in the Bandung city and its surrounding areas is needed to determine the natural background radiation levels. So it can guarantee the security and comfort of the area. Monitoring of natural background radiation levels has been carried out using the GMC-320 Plus survey meter, which can measure beta and gamma radiation. Measurements are carried out in several locations in Bandung and its surroundings. After getting a natural background radiation levels, conversion to dose rate was carried out. Based on the results, the natural background radiation levels and the dose rate in the environment in Bandung and its surrounding area are still below the threshold and at the safe level.

Keywords: Dose Rate, GMC-320 Plus Survey Meter, Natural Background Radiation

Topic: Radiation physics
[ABS-60]

A PERFORMANCE OF AUSTENITIC 15CR25NI STEEL AFTER 850°C-30 MINUTES WATER QUENenchING TREATMENTS

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ABSTRACT

An austenitic stainless steel namely 15Cr25Ni has been synthesized by casting techniques at temperature more than 1250 °C in an inductive furnace. As-cast composition (%wt.) is 58%Fe, 15%Cr, 25%Ni, 0.34%C and less than 0.1% of impurities that comprised of titanium, phosphor, copper, niobium and sulphur elements in steel. The steel might be dedicated for structural components and must resist to mechanical loads, high pressure-temperature and corrosion. Temperature treatments in water cooling media are required to increase the strengths. X-ray diffraction shows that as-cast has fcc crystal structure with lattice parameter of 3.632 Å. Similarly, water quenched sample has lattice parameter of 3.587 Å, lowered of that as-cast. Microstructure reveals as-cast grains look big and describe un-deformed structures; with average grain size of 6 mikrometer. Water quenched sample shows coarse grain. Sensitization and viscosity (nu) of medium have important role in formation of grain boundaries. Due to rate of decreasing temperature is heavily influenced by heat diffusion from high to low temperature space. Moreover, high temperature corrosion tests of 15Cr25Ni steel at 850 °C for more than 15 hours showed; there are no damage and drift phases in the corrosion products. The steel has good corrosion resistance at temperatures 850 °C.

Keywords: 15Cr25Ni austenitic steel, water-quenching, X-ray diffraction, electron microscope, hardness, oxidation

Topic: Material and process for innovative energy systems
[ABS-61]
THERMAL-HYDRAULICS ANALYSIS OF SIMPLE APPARATUS FOR MSR LOOP

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ABSTRACT

Molten Salt Reactor (MSR) is a unique Generation IV nuclear reactor system, since it is the only reactor system that uses a liquid fuel. As a consequence, fuel and moderator are mixed together. Therefore the thermal-hydraulics characteristics of MSR become interesting to be evaluated. A simple apparatus for a proxy study of MSR primary loop has been developed. This simple loop was produced from ABS material by using 3D printer. This apparatus can simulate the flow characteristics of fluid with temperature up to $150 \, ^\circ C$. Figure 1 shows the schematic diagram and the picture of the real simple loop of MSR. Several thermal-hydraulics characteristics of the loop will be presented at the conference.

Figures 1 The schematic diagram and the real simple loop of MSR.

Keywords: Generation IV, MSR, thermal-hydraulics analysis, simple loop
[ABS-62]
PRELIMINARY DESIGN STUDY OF HIGH TEMPERATURE GAS COOLED FAST REACTORS

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ABSTRACT

Energy conversion efficiency play important role in achieving economical design of NPP and directly influences thermal pollution to the environment. In recent decades natural gas fueled power plants becomes very competitive with advanced high temperature gas turbine system. In this study the feasibility to develop high temperature modular gas cooled fast reactors have been investigated. The study focus on neutronic and thermal hydraulic design to get some design with enhanced safety performance and the selection of special materials to support high temperature operation up to 1000°C. High Temperature Ceramic materials are intensively used for structural materials. The quasistatic safety analysis is used to estimate the performance of the designs during some accident conditions. The analysis have been performed using FI-ITBCHI code system.
[ABS-63]
EVOLUTION OF MODIFIED CANDLE REACTOR DESIGNS AND ANALYSIS

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ABSTRACT

Modified CANDLE designs is a modification of CANDLE design with introduction of discreet regions. have evolved from axial direction shuffling with some variations related to the position of initial stage fuel near the most active regions, to the radial direction shuffling, and then to the combined axial-radial shuffling. The axial-radial combined shuffling in general give excellent criticality condition so that the design can accommodate the core with discharge burnup in the order of about 20% HM burnup level. Such burnup level is expected can be handled using next generation fast reactor advanced materials. The design of Modified CANDLE have been successfully implemented using various fast reactors type including Pb-Bi/Pb cooled fast reactors, Pb-208 Cooled fast reactors, and helium cooled GCFR with Nitride, Carbide or Metallic fuels. Some designs with superior criticality have been successfully used to design small type Modified CANDLE up to about 200-250 MWt power levels. In term of Simulation models the Modified CANDLE have been analyzed with various code systems including combinations of SRAC with FI-ITB CH1-code systems, combination of SLAROM with FI-ITB CH1-Code systems, or SRAC(+CITATION) code systems, and recently it is simulated using MCNP code system. All simulation systems give comparable results. However MCNP based simulations need very long CPU time and therefore only worth to be implemented in cluster computer system.