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Abstract Collection

Endiah Puji Hastuti, Sudjatmi K. Alfa, Sudarmono., *Analysis on The Performance of The Bandung Conversion Fuel-Plate Triga Reactor in Steady State with Constant Coolant Flow Rate.* Tri Dasa Mega, 22 (2), 41.

Bandung TRIGA 2000 Reactor, a General Atomic (GA)-made research reactor used for training, research andiIsotope production, has been upgraded to operate at power of 2000 kW using TRIGA fuel rod type. Recently, the TRIGA reactor fuel element producers are going to discontinue the production of TRIGA fuel element. To overcome the unavailability of TRIGA fuel element, BATAN planned to modify TRIGA 2000 fuel type from rod-type to U_3Si_2-Al plate-type fuel with 19.75% enrichment, similar to the domestically fabricated one used in RSG-GAS. The carried out design emphasized on the determination of operation condition limits for setting the reactor protection system in accordance to the reactor safety calculation results. The conceptual design of the innovative fuel plate TRIGA reactor cooling system is expected to remove heat generated by fuels with nominal power of 1 MW up to 2 MW. The design is developed through modelling and safety analysis using COOLOD-N2 validated code. The safety margin is set to its flow instability at transient condition of the fuel plate, which is ≥ 2.38; departure from nucleate boiling ratio ≥1.50; and no onset of nucleate boiling, $\Delta T_{ONB} \geq 0^oC$. The primary coolant flow rate accommodating the existing Bandung TRIGA reactor capability is as high as 50 kg/s. The analysis results show that at power of 1 MW, the reactor can safely operate, while at power of 2 MW the safety margin is exceeded. In other words, the plate TRIGA reactor that employs forced convection mode operates safely at 1 MW with excess power 120% of its nominal power.

Keywords: 1 MW, thermalhydraulic design, steady state condition, triga plate, constant flowrate.

Sukmanto Dibyo, Ign. Djoko Irianto, V. I. Sri Wardani, Marliyadi Pancoko., *Analysis on Flow Pressure in the Pneumatic Braking System of FHS-RDE using Fluent 6.3 Software.*, Tri Dasa Mega, 22 (2), 49.

The High Temperature Gas Cooled Reactor (HTGR) is considered as one of the nuclear reactors of generation-IV type in the future. The fuel handling system is one of the important processes in HTGR as well as in the design of RDE (Reaktor Daya Eksperimental). In the Fuel Handling System (FHS), the fuel pebble is transferred pneumatically along the pipe using carrier gas into the core of the reactor. Therefore, the pneumatic is an important system in operation stability of FHS. During the developing process of FHS-RDE, a branch pipe as a braking pipe system is provided on top of the pneumatic system to reduce the speed of the fuel discharged from the pneumatic pipe. The pneumatic pipe has an inner diameter of 65 mm and 20 m in length, whereas a branch pipe diameter for the braking system is 30 mm. The pneumatic system pressure is greater than the reactor cooling system pressure of 3.0 MPa. This work was performed to investigate the pressure drop and flow pattern of the braking system of FHS by various carrier gas inlet pressure. The analysis was carried out by Fluent 6.3 Software. Based on the design parameter of FHS used in the analysis, the results show that the performance of the braking system is not significant to reduce the pressure in the top region of the pneumatic pipe. To obtaining a significant reduction in pressure, and evaluation on the design of the branch pipe as well as the radius of curvature of the bend at the top pipe is suggested

Keywords: Pneumatic pipe, pressure, braking system, fuel handling of RDE, carrier gas.

R. Andika Putra Dwijayanto, Dedy Prasetyo Hermawan., *Investigation on Inherent Safety of One Fluid-Molten Salt Reactor (OF-MSR) with Various Starting Fuel.* Tri Dasa Mega, 22 (2), 54.

Molten salt reactor (MSR) is often associated with thorium fuel cycle, thanks to its excellent neutron economy and online reprocessing capability. However, since ^{233}U, the fissile used in pure thorium fuel cycle, is not commercially available, the MSR must be started with other fissile nuclides. Different fissile yields different inherent safety characteristics, and thus must be assessed accordingly. This paper investigates the inherent safety aspects of one fluid MSR (OF-MSR) using various fissile fuel, namely low-enriched uranium (LEU), reactor grade plutonium (RGPu), and reactor grade plutonium + minor actinides (PuMA). The calculation was performed using MCNPX2.6.0 *programme with ENDF/B-VII library. Parameters assessed are temperature coefficient of reactivity (TCR) and void coefficient of reactivity (VCR). The result shows that TCR for LEU, RGPu, and PuMA are -3.13 pcm, -2.02 pcm and -1.79 pcm, respectively. Meanwhile, the VCR is negative only for LEU, whilst RGPu and PuMA suffers from positive void reactivity. Therefore, for the OF-MSR design used in this study, LEU is the only safe option as OF-MSR starting fuel.*

Keywords : MSR, temperature coefficient of reactivity, void coefficient of reactivity, low enriched uranium, reactor grade plutonium, minor actinides.

Pande Made Udiyani and M. Budi Setiawan., *Source Term Assessment for 100 MWe Pressurized Water Reactor*., Tri Dasa Mega, 22 (2), 61.

One of the barriers on the implementation of nuclear energy in Indonesia is public perception towards the safety of nuclear power plants (NPP). Therefore, it is necessary to perform a study about the radiation impact of normal and abnormal operations of an NPP. In accordance to the program of Ministry of Research and Technology period 2020-2024, concerning the plan to build a small modular reactor (SMR)-type NPP, a radiation safety study has been performed for the 100 MWe Pressurized Water Reactor (PWR-100MWe). Source term release of radioactive substances into the environment from PWR-100MWe is a starting point in the study of the radiological consequences of reactor operation. Therefore, this paper will examine the PWR-100MWe source term under normal and abnormal operating conditions, according to the design and the design basis accident (DBA). The initial trigger of the DBA is Lost of Coolant Accident (LOCA) such as Small LOCA and Large LOCA. Due to the limitations of available SMR data, the study of PWR-100MWe source term refers to the assumption of the release fraction of fission products per subsystem in a larger 1000MWe PWR. It is expected from this assumption that pessimistic source term will be obtained. The study begins with calculation of PWR-100MWe core inventory using ORIGEN2 code based on PWR-100MWe reactor parameters. Through the mechanistic source term model and PWR-1000MWe release parameters, source terms will be obtained for normal operation and abnormal conditions i.e. DBA. Normal source term is used to calculate the consequences of normal operation, which will be used for environmental monitoring and environmental safety analysis of the site. Whereas accident source term is the basis for calculating the radiological consequences of accidents used for SAR documents and nuclear preparedness.

Keywords : SMR, PWR-100MWe, normal operation, source term, accident.

Sudjatmi K. Alfa, Endiah Puji Hastuti, Prasetyo Basuki, Santiko T. Sulaksono, Rian Fitriana., *Bandung TRIGA 2000 Reactor power analysis as a function of the number of fuel elements and the power peaking factor.* Tri Dasa Mega, 22 (2), 68.

The reactivity value of the Bandung TRIGA 2000 reactor core has decreased over time, so the power generated by the reactor is also getting smaller, despite the control rod position is fully withdrawn. Therefore, it is necessary to reshuffle and refuel the fuel element to increase the excess reactivity by considering the safety parameters, such as axial and radial power peaking factors, DNBR, dT_{sat}, and temperature on the cladding and in the center of the fuel element. The analyzed reactor safety parameters are the number of fuel elements, which varied at 105, 110, and 115 elements, as well as power peaking factor, which varied at 1.55, 1.65, 1.75, 1.85, and 1.95. The calculations were done using MCNP and COOLOD-N2 programs. If DNBR ≈ 1.3 is determined as the safety limit for the operation of the Bandung TRIGA 2000 reactor, at PPF 1.95 (105, 110, and 115 fuel elements), it can be considered to operate the reactor at the power of 600-700 kW. However, at PPF of 1.75 (105, 110, and 115 fuel elements), the reactor can be operated at the power of 700-800 kW, and at PPF of 1.55 (105, 110, and 115 fuel elements), the reactor can be considered for operation at the power of 800-900 kW. The results of these calculations can be used for consideration in determining the operating limits of the Bandung TRIGA 2000 reactor

Keywords: TRIGA, fuel element, power peaking factor, DNBR, boiling

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