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LEMBAR ABSTRAK

Dion Bagus Nugraha B., Andang Widi Harto, Sihana., *Moderator To Fuel Ratio And Uranium Fraction Analysis Of Square Lattice Molten Salt Transatomic Power,* Tri Dasa Mega, 19 (1), 1.

Molten Salt Reactor Transatomic Power (MSR TAP) is a further development of the nuclear reactor Generation IV Reactor Molten Salt Reactor (MSR). MSR TAP generates clean electric power. It has a passive safety, resistance to proliferation, and low cost. MSR TAP can consume the rest of the nuclear fuel/spent nuclear fuel (SNF) of a commercial Light Water Reactor (LWR) fuel or use the main fuel, a salt solution UF_4 - LiF - BeF_2. MSR TAP uses Zirconium Hydride material for the moderator. This research has a purpose to determine the optimal size of uranium mole fraction on fuel and moderator radius from core design in order to produce optimum enrichment with the value 1 < k_{eff} <1.0065 using MCNP5 program. On the other hand, this research also aims to look for the optimum enrichment, which have inherent safety characteristics with α_{Void} < 0. Variations were made including the changes in the geometry of the moderator radius with a variation of 0.5 cm, 1 cm, 1.5 cm, 2 cm, 2.5 cm, 3 cm, 3.5 cm, 4 cm, and 4.5 cm; and the changes in the fuel uranium molar UF4 - LiF - BeF2 with molar variation of 15%, 20%, 25% and 30%. The geometry of Transatomic Power (MSR TAP) of companies Transatomic Power Corporation was used. The results show that the optimum variation is the salt solution UF_4 - LiF - BeF_2 with 25 % uranium mole fraction, 2.6 % enrichment and moderator radius of 1.5 cm. The optimum variation gives the k_{eff} value of 1.00124 ± 0.00078. The optimum value of reactivity void coefficient is -0.0684. It indicates an inherently safe design.

Keywords: Molten Salt Reactor Transatomic Power, MCNP5, Uranium Fuel Mole Fraction, Optimum Variation, Moderator, Inherent Safety.

Entin Hartini**.**, *Implementation Of Missing Values Handling Method For Evaluating The System/Component Maintenance Historical Data,* Tri Dasa Mega, 19 (1), 11.

Missing values are problems in data evaluation. Missing values analysis can resolve the problem of incomplete data that is not stored properly. The missing data can reduce the precision of calculation, since the amount of information is incomplete. The purpose of this study is to implement missing values handling method for systems/components maintenance historical data evaluation in RSG GAS. Statistical methods, such as listwise deletion and mean substitution, and machine learning (KNNI), were used to determine the missing data that correspond to the systems/components maintenance historical data. Mean substitution and KNNI methods were chosen since those methods do not require the formation of predictive models for each item which is experiencing missing data. Implementation of missing data analysis on systems/components maintenance data using KNNI method results in the smallest RMSE value. The result shows that KNNI method is the best method to handle missing value compared with listwise deletion or mean substitution.
Keywords: missing value, data evaluation, algorithm, implementation.

Andi Sofrany Ekariansyah, Surip Widodo., *Preliminary Study On RELAP5 Simulation Of DVI Line Break Accident In The ATLAS Facility Using Best Estimate Plus Uncertainty Method* Tri Dasa Mega, 19 (1), 19.

The Best Estimate plus Uncertainty (BEPU) is a methodology, which was introduced in the deterministic safety analysis to evaluate limitations of codes in simulating realistic plant behavior by providing quantified uncertainty bands of calculation results. It has been already widely accepted in licensing nuclear power plant by regulatory bodies of United States (USNRC), Argentina, and Canada. The uncertainty evaluation in the BEPU method is performed by different approaches such as GRS, IRSN, ENUSA, AEAT, and UNIPI. Due to the complexity of other approaches, the purpose of this study is to present some key aspects of the BEPU process using the GRS methodology by selecting the ATLAS test facility to simulate 50% break of DVI line since any safety analysis performed so far was using deterministic best estimate approach only. As comparison of the best estimate simulation performed by RELAP5/SCDAP/Mod3.4, experimental data related to the event was used. After 100 simulations, the uncertainty bands of peak heater of clad temperature and primary pressure transient obtained were only in a close agreement with the experimental data in the earlier period and less than 250 seconds during the transient condition. Therefore the overall accuracy of the best estimate simulation plays a key role on the final results of the uncertainty analysis because the propagation of any discrepancy in the best estimate with the experimental data will occur throughout the simulation. After that, selecting the important parameters to be randomly generated needs to be performed carefully by studying the important phenomena related to the event analyzed and associated plant model.
Keywords: best estimate plus uncertainty, DVI line break, ATLAS facility, RELAP5, simulation

Sukmanto Dibyo, Ign. Djoko Irianto., *Design Analysis On Operating Parameter Of Outlet Temperature And Void Fraction in RDE Steam Generator*, Tri Dasa Mega, 19 (1), 33.

HTGR is one of the next generation reactor types. HTGR is currently considered as one of the leading reactors for the future nuclear power plant. The steam generator is one of the main components in HTGR as well as in RDE. In the steam generator, the heat is transferred by high temperature helium gas in the shell side to water in the tube side to generate the superheated steam. The purpose of this work is to design the operating parameter of outlet temperature and void fraction of steam based on feed water mass flow rate and inlet temperature variations in RDE steam generator. In this work, the ChemCAD program was used. Both inlet and outlet temperature of helium gas have been set up as boundary conditions. The result shows that using the mass flow rate of 4.3 kg/s - 4.8 kg/s and water inlet temperature of 110 ^{o}C - 160 ^{o}C, the superheated steam outlet temperature (void fraction = 1.0) is obtained in the range of 275.5 ^{o}C – 600 ^{o}C.This analysis is beneficial to assess 10 MW RDE design especially in the steam generator system operating parameters.
Keywords: outlet temperature, void fraction, superheated steam, RDE steam generator

Sri Kuntjoro., *Criticality Analysis Of Uranium Storage Facility With Formation Racks,* Tri Dasa Mega, 19 (1), 41.

Uranium materials are needed for the uranium fuel production of research reactors and radioisotope. Before the uranium material is used, it is stored in the storage facility. One of the prerequisites for uranium material storage facilities is that it must be in the sub-critical condition. The purpose of this study is to analyze the criticality condition of uranium material storage facility located in PT. Inuki (Persero) and to ensure that the criticality condition is always in sub-critical state. Criticality analysis was performed using MCNP-5 program to determine the level of criticality of the three uranium material storage facilities at initial conditions and conditions after adding the storage racks. For analysing storage facilities 1 and 2, three scenarios of container on the storage rack formations were considered. Meanwhile, for analysing the storage facility 3, one scenario was considered. The results confirm that all strorages at initial condition and after adding storage racks formation were still in sub-critical condition (k-eff<1). These results are then used as the basis for the uranium materials management. It is also used as a basis for issuing an operational license by the nuclear energy regulatory body (BAPETEN).
Keywords: criticality, uranium storage facility, k-eff

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