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LEMBAR ABSTRAK

Iman Kuntoro, Surian Pinem, Tagor Malem Sembiring, *Validation Of PWR-Fuel Code For Static Parameters In The LWR Core Benchmark*, Jurnal Teknologi Reaktor Nuklir TRI DASA MEGA, 20 (3), 111.

The PWR-FUEL code is a multi dimensional, multi group diffusion code with nodal and finite difference methods. The code will be used to calculate the fuel management of PWR reactor core. The result depends on the accuracy of the codes in producing the core effective multiplication factor and power density distribution. The objective of this research is to validate the PWR-FUEL code for those cases. The validation are carried out by benchmarking cores of IAEA-2D, KOERBERG-2D and BIBLIS-2D. The all three cases have different characteristics, thus it will result in a good accuracy benchmarking. The calculation results of effective multiplication factor have a maximum difference of 0.014 %, which is greater than the reference values. For the power peaking factor, the maximum deviation is 1.75 % as compared to the reference values. Those results show that the accuracy of PWR-FUEL in calculating the static parameter of PWR reactor benchmarks are very satisfactory.

Keywords: Validation, PWR-FUEL code, static parameter.

Reinaldy Nazar, Sudjatmi KA, K. Kamajaya, *The Thermohydraulic Analysis Of The Bandung Research Reactor Core With Plate Type Fuel Elements Using The CFD Code,* Jurnal Teknologi Reaktor Nuklir TRI DASA MEGA, 20 (3), 123.

Due to TRIGA fuel elements are no longer produced by General Atomic, it is necessary to find a solution so that the Bandung TRIGA 2000 reactor can still operate. One solution is to replace the type of fuel elements. Study on using the MTR fuel plate type RSG-GAS in the Bandung TRIGA 2000 has been done. Based on the results of the study, using CFD computer program, it is was found that TRIGA reactor with fuel plate type elements cannot be operated by up to 2000 kW power with natural convection cooling mode. Therefore, the reactor must be cooled by forced convection. The analysis by using forced convection showed that for 50 kg/s cooling flow rate and the temperature was varied such as 35 ^oC, 35.5 ^oC and 36 ^oC, the surface temperature of fuel elements ware between 110.37 ^oC - 111.27 ^oC and the cooling water temperature in the corresponding position is between 61.03 ^oC - 61.95 ^oC. The surface temperatures of fuel elements are approaching to the saturation temperature and have started to nucleate boiling, so the use of cooling flow rate entering core less than 50 kg/s should be avoided. The surface temperature of fuel elements was decrease under saturation temperature if cooling flow rate is greater than 65 kg/s, the the surface temperature of fuel elements achieved 96.65 ^oC and coolant temperature in the corresponding position was 54.38 ^oC.

Keywords: Bandung research reactor, plate type fuel element, thermohydraulic, CFD code

Susyadi, *Simulation Of Feed Water Temperature Decrease Accident In NuScale Reactor,* Jurnal Teknologi Reaktor Nuklir TRI DASA MEGA, 20 (3), 133.

Study on thermal hydraulic behavior of the NuScale reactor during secondary system malfunction that causes a feed water temperature decrease has been conducted using RELAP5 code. This study is necessary to investigate the performance of safety system and design in dealing with an accident. The method *used involves simulation of reactor transient through numerical modeling and calculation in RELAP5 code covering primary and secondary system, including the decay heat removal system (DHRS). The investigation focuses on the flow and heat transfer characteristics that occurs during the transient. The calculation result shows that at the beginning, core power increases up to trip set point of 200 MW which is driven by positive feedback reactivity of coolant overcooling and automatic control rod bank adjustment. Meanwhile, the core exit coolant temperature increases up to 600 K. and primary system circulation flow rate speeds up to 556 kg/s. After that, the reactor trips and power drops sharply, then followed by a closing of steam line and feed water isolation valves and an opening of DHRS valves. The simulation shows that, the DHRS are capable to transfer decay heat to the reactor pool and as a result the primary system temperature and pressure decreases. The reactor could stay in safe shutdown state afterward.*

Keywords: NuScale, RELAP5, feed water, decay heat, simulation

V. Indriati Sri Wardhani, Budi Santoso, *An Analysis Of Pump Power Calculation Of Coverted Bandung TRIGA Reactor With Pipe Routing Through Delay Tank,* Jurnal Teknologi Reaktor Nuklir TRI DASA MEGA, 20 (3), 143.

The Bandung TRIGA 2000 Reactor has been widely used for conducting training, researches and isotop production since 1965. This reactor have to be decommissioned due no further fuelproduced by original vendor. Therefore, conversion of cylinder fuel into plate is needed. PT INUKI has been able to produce its own plate type fuel so that by changing the reactor core which was originally cylindrical into a square shape or converting the fuel element from cylindrical to plate type operation of the Bandung TRIGA 2000 reactor can be maintained for a long time. On this conversion, the reactor's cooling system will change, which initially by natural convection to forced convection, while the direction of the cooling flow changes as well, which initially from bottom to top becomes from top to bottom. If the directional cooling flow of the plate TRIGA reactor system is made from top to bottom, without changing of piping, the result is a high exposure of Nitrogen-16 radiation on the surface of the reactor tank, therefore a delay tank is needed. By the new pipe routing system, it is necessary to reanalyze on determining the pump power requirements. The pump should be able to supply this energy. In other words, the total head produced by the pump must be equal to the total head required by the system. If the total system head data and coolant flow rate, and considering the efficiency of the pump and the motor drive pump have been analysed, so the pump power requirements can be calculated. The calculation result shows that the amount of pump power required to drain the cooling fluid in the primary system is 35 kW or 47 HP.

Keywords: conversion, cooling, plate type, pipe routing, pumps

Tagor Malem Sembiring, Pungky Ayu Artiani, *Subcritycality Analysis Of HTR-10 Spent Fuel Cask Model For The 10 MW HTR Indonesian Experimental Power Reactor,* Jurnal Teknologi Reaktor Nuklir TRI DASA MEGA, 20 (3), 151.

The 10 MW HTR Indonesian Experimental Power Reactor (RDE reactor) is designed identical with the HTR-10 in China, conceptually. However, the review results showed that the spent fuel cask model which is used between two reactors is fully different, such as size and capacity. The proposed cask model in RDE reactor can hold 15 times more fuel pebbles than HTR-10 has. This research activities deal with the subcriticality analysis for the spent fuel cask of RDE reactor if using the HTR-10 cask model. The subcriticality condition is designed to meet the limit of safety value. The objective of this research is to determine the subcriticality value in the normal and accident events for the spent fuel cask when it is in the reactor building and the spent fuel cask room. All calculations were carried out by MCNP6.1 code. The selected external events are the water ingress (reactor room), water flood and the combination event of water flood and earthquake. The calculation results showed that the maximum value of k_{eff} (3σ) are 0.47510 and 0.19214 for the cask in the reactor building and in the spent fuel cask room, respectively. This value is far from the limit value of 0.95. The calculation results showed that the spent fuel cask are in the safe condition eventhough in the worst combination events, the cask is flooded and earthquake. The HTR-10 spent fuel cask can be proposed as an alternative for the RDE reactor to get an efficient reactor building.

Keywords: spent pebble fuel element, HTGR, subcriticality, MCNP6.1, RDE reactor

Andi Sofrany Ekariansyah, Surip Widodo, Hendro Tjahjono, Susyadi, Puradwi I. Wahyono, Anwar Budianto, *Preliminary*

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Analysis Of Core Temperature Distribution Of Experimental Power Reactor Using RELAP5, Jurnal Teknologi Reaktor Nuklir TRI DASA MEGA, 20 (3), 159.

High Temperature Gas Cooled Reactor (HTGR) is a high temperature reactor type having nuclear fuels formed by small particles containing uranium in the core. One of HTGR designs is Pebble Bed Reactor (PBR), which utilizes helium gas flowing between pebble fuels in the core. The PBR is also the similar reactor being developed by Indonesia National Nuclear Energy Agency (BATAN) under the name of the Reaktor Daya Eksperimental (RDE) or Experimental Power Reactor (EPR) started in 2015. One important step of the EPR program is the completion of the detail design document of EPR, which should be submitted to the regulatory body at the end of 2018. The purpose of this research is to present preliminary results in the core temperature distribution in the EPR using the RELAP5/SCDAP/Mod3.4 to be complemented in the detail design document. Methodology of the calculation is by modelling the core section of the EPR design according to the determined procedures. The EPR core section consisting of the pebble bed, outlet channels, and hot gas plenum have been modelled to be simulated with 10 MWt. It shows that the core temperature distribution under assumed model of 4 core zones is below the limiting pebble temperature of 1,620 °C with the highest pebble temperature of 1,477.0 °C. The results are still preliminary and requires further researches by considering other factors such as more representative radial and axial power distribution, decrease of core mass flow, and heat loss to the reactor pressure vessel.

Keywords: pebble bed, core temperature, EPR, RELAP5

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