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LEMBAR ABSTRAK

Susyadi, Hendro Tjahyono, Sukmanto Dibyo, Jupiter S. Pane., *Investigasi Karakteristik Termohidrolika Teras Reaktor Daya Kecil Berpendingin Sirkulasi Alam Menggunakan RELAP5.* Jurnal Teknologi Reaktor Nuklir TRI DASA MEGA, 18 (1), 1.

Small modular reactor (SMR)is very prospective to be deployed in Indonesia. Its low output power, compact design and capability to be constructed modularly provide better deployment flexibility compared to a large conventional reactor. There are various designs of SMR, one of them implements natural circulation for its primary cooling system or in other words the reactor uses no primary pumps. Besides, the dimension of fuel element is shorter than the one used by large reactor. These two aspects may produce different heat transfer behavior which could lead to a safety implication. For that reason, this research investigates thermal hydraulic characteristics of the core of SMR with naturally circulating coolant, especially on the fuel and coolant temperatures and mass flow rate. The purpose is to identify the thermal safety margin difference of the reactor compared with conventional PWR. The investigation was performed using RELAP5 in which the core was partially represented by means of generic models of the program and continued with steady state calculations. The result shows that during nominal power operation, the reactor has better of 2K degree for boiling temperature margin than the large conventional PWR. In addition, the excellence of SMR safety margin was shown by the increase of primary coolant flow rate following the increase of power which means that the reactor has a distinctive inherent safety.

Keywords: small modular reaktor, PWR, natural circulation, RELAP5, thermal-hydraulic

Sumijanto, Sriyono**.,** *Optimasi Laju Alir Massa Dalam Purifikasi Pendingin RGTT200K Untuk Proses Konversi Karbonmonoksida.,* Jurnal Teknologi Reaktor Nuklir TRI DASA MEGA, 18 (1), 11.

Carbonmonoxide is a species that is difficult to be separated from the reactor coolant helium because it has a relatively small molecular size. So it needs a process of conversion from carbon monoxide to carbondioxide. The rate of conversion of carbonmonoxide in the purification system is influenced by several parameters including concentration, temperature and mass flow rate. In this research, optimization of the mass flow rate in coolant purification of RGTT200K for carbonmonoxide conversion process was done. Optimization is carried out by using software Super Pro Designer. The rate of reduction of reactant species, the growth rate between the species and the species products in the conversion reactions equilibrium were analyzed to derive the mass flow rate optimization of purification for carbonmonoxide conversion process. The purpose of this study is to find the mass flow rate of purification for the preparation of the basic design of the RGTT200K coolant helium purification system. The analysis showed that the helium mass flow rate of 0.6 kg/second resulted a notoptimal conversion process.Theoptimal conversion processwas reached at a mass flow rate of 1,2 kg/second. A flow rate of 3.6 kg/second–12 kg/secondresulted an ineffective process. For supporting thebasic design of the RGTT200K helium purification system, the mass flow rate for carbonmonoxide conversion process is suggested to be1.2 kg/second .

Keywords:Carbonmonoxide,conversion, purification, mass flow rate, RGTT200K.

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Reinaldy Nazar., *Karakteristik Perpindahan Panas Konveksi Alamiah Aliran Nanofluida AL_2O_3-Air Di Dalam Pipa Anulus Vertikal.* Jurnal Teknologi Reaktor Nuklir TRI DASA MEGA, 18 (1), 21.

Results of several researches have shown that nanofluids have better thermal characteristics than conventional fluid (water). In this regard, ideas for using nanofluids as an alternative heat transfer fluid in the reactor coolant system have been well developed. Meanwhile the natural convection in a vertical annulus pipe is one of the important mechanisms of heat transfer and is found at the TRIGA research reactor, the new generation nuclear power plants and other energy conversion devices. On the other hand the heat transfer characteristics of nanofluids in a vertical annulus pipe has not been known. Therefore, it is important to do research continuously to analyze the heat transfer nanofluids in a vertical annulus pipe. This study has carried out numerical analysis by using computer code of CFD (computational of fluids dynamic) on natural convection heat transfer characteristics of nanofluids flow of Al_2O_3-water 2% volume in the vertical annulus pipe. The results showed an increase in heat transfer performance (Nusselt numbers - N_U) by 20.5% - 35%. In natural convection mode with Rayleigh numbers $2.471e^{+09} \leq R_a \leq 1.955e^{+13}$ obtained empirical correlations for water is $N_U = 1.065\left(R_a \frac{D_H}{x}\right)^{0.179}$ and empirical correlations for Al_2O_3-water nanofluids is $N_U = 14.869\left(R_a \frac{D_H}{x}\right)^{0.115}$.

Keywords: Al_2O_3-water nanofluids, the natural convection, the vertical annulus pipe

Tukiran Surbakti, Tagor Malem Sembiring., *Neutronics Analysis On Mini Test Fuel In The RSG-GAS Core.* Jurnal Teknologi Reaktor Nuklir TRI DASA MEGA, 18 (1), 29.

Research on UMo fuel for research reactor has been developed. The fuel of research reactor is uranium molybdenum low enrichment with high density. For supporting the development of fuel fabrication, an neutronic analysis of mini fuel plates in the RSG-GAS core was performed. The aim of analysis is to determine the numbers of fuel cycles in the core to know the maximum fuel burn-up. The mini fuel plates of U7Mo-Al and U6Zr-Al with densities of 7.0 gU/cc and 5.2 gU/cc, respectively, will be irradiated in the RSG-GAS core. The size of both fuels, namely 630x70.75x1.30 mm were inserted to the 3 plates of dummy fuel. Before the fuel will be irradiated in the core, a calculation for safety analysis from neutronics and thermal-hydraulics aspects were required. However, in this paper, it will be discussed safety analysis of the U7Mo-Al and U6Zr-Al mini fuels from neutronic point of view. The calculation was done using WIMSD-5B and Batan-3DIFF codes. The result showed that both of the mini fuels could be irradiated in the RSG-GAS core with burn up less than 70 % within 12 cycles of operation without over limiting the safety margin. If it is compared, the power density of U7Mo-Al mini fuel is bigger than U6Zr-Al fuel.

Key words: mini fuel, neutronics analysis, reactor core, safety analysis

Muhammad Darwis Isnaini, Muhammad Subekti., *Validation Of SIMBAT-PWR Using Standard Code Of COBRA-EN On Reactor Transient Condition.* Jurnal Teknologi Reaktor Nuklir TRI DASA MEGA, 18 (1), 41

The validation of Pressurized Water Reactor typed Nuclear Power Plant simulator developed by BATAN (SIMBAT-PWR) using standard code of COBRA-EN on reactor transient condition has been done. The development of SIMBAT-PWR has accomplished several neutronics and thermal-hydraulic calculation modules. Therefore, the validation of the simulator is needed, especially in transient reactor operation condition. The research purpose is for characterizing the thermal-hydraulic parameters of PWR1000 core, which be able to be applied or as a comparison in developing the SIMBAT-PWR. The validation involves the calculation of the thermal-hydraulic parameters using COBRA-EN code. Furthermore, the calculation schemes are based on COBRA-EN with fixed material properties and dynamic properties that calculated by MATPRO subroutine (COBRA-EN+MATPRO) for reactor condition of startup, power rise and power fluctuation from nominal to over power. The comparison of the temperature distribution at nominal 100% power shows that the fuel centerline temperature calculated by SIMBAT-PWR has 8.76% higherresult than COBRA-EN result and 7.70% lower result than COBRA-EN+MATPRO. In general, SIMBAT-PWR calculation results on fuel temperature distribution are mostly between COBRA-EN and COBRA-EN+MATPRO results.The deviations of the fuel centerline, fuel surface, inner and outer cladding as well as coolant bulk

temperature inthe SIMBAT-PWR and the COBRA-EN calculation, are due to the value difference of the gap heat transfer coefficient and the claddingthermal conductivity.

Keywords: transient, thermal-hydraulics, PWR, simulator, COBRA-EN, MATPRO.

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