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LEMBAR ABSTRAK

Susyadi, Hendro Tjahjono, Sukmanto Dibyo, Jupiter S Pane., *Study On The Characteristic Of Steam Generation In Helical Steam Generator Of Small Modular Reactor*. Jurnal Teknologi Reaktor Nuklir TRI DASA MEGA, 17 (2), 59.

Small modular reactor (SMR) is very suitable to be deployed in Indonesia especially for locations having low electrical grid capacity, so further investigation on the characteritic of this reactor is needed. In general SMR has a compact and integrated-to-vessel steam generator design. This design implies different approach in producing steam as compared to conventional nuclear power plant having inverted u-tube steam generator. For that reason, this research is intended to investigate the steam characteristic and how it is generated in the helical SG which is widely used in SMR. The method used is through numerical calculation of the SG model using RELAP5 code. In the model, the feed-water which has low pressure and temperature is flown into helical tubes while high pressure and temperatur fluid, which represents reactor primary system coolant, stays in outer side of the tube. Calculation result shows that the steam produced by helical steam generator is superheated, i.e. about 25 K above saturation temperature. This provides comparative advantage to SMR on the design and operational aspects compared to conventional reactors because the superheated steam it produces can reduce turbine losses and at the same time increase thermodynamic efficiency.

Keywords: helical steam generator, SMR, PWR, superheated steam, RELAP5

Sudjatmi K.A, Endiah Puji Hastuti, Surip Widodo, Reinaldy Nazar**.,** *Analysis Of Natural Convection In TRIGA Reactor Core Plate Types Fueled Using COOLOD-N2.,* Jurnal Teknologi Reaktor Nuklir TRI DASA MEGA, 17 (2), 67.

Any pretensions to stop the production of TRIGA fuel elements by TRIGA reactor fuel elements manufacturer should be anticipated by the operating agency of TRIGA reactor to replace the cylinder type fuel element with plate type fuel element that available on the market. In this study, the calculation of U_3Si_2Al fuel with uranium enrichment of 19.75 % and a load level of 2.96 gU/cm^3 was performed. Analyses were performed using the validated COOLOD - N2 program. TRIGA conversion core configurations of fuel plate type are composed of 16 fuel elements, 4 control elements and 1 irradiation facilities which are located in the middle of core. The calculation results showed that if the cooling temperature was 37°C, and the ratio of radial power peaking factor ≤ 1.92, then the maximum power that can be operated on free convection mode of operation was 600 kW. The thermalhydraulic characteristic obtained such as coolant temperature at the outlet side, cladding and meat were 82.39°C, 108.88°C and 109.02°C respectively, while the ΔT_{ONB} (Temperature Onset of Nucleate Boiling) was 7.18°C and OFIR (Onset of flow instability ratio) value was 1.03. The results are expected to be used as a reference for determining the power level of the TRIGA reactor core plate types fueled.

Keywords: TRIGA Convertion, COOLOD-N2, Thermalhydraulics characteristic, natural convection, plate type fuel element

Muh. Darwis Isnaini, Surip Widodo, Muhammad Subekti., *The Thermal-Hydraulics Analysis On Radia And Axial Power*

Fluctuation For AP1000 Reactor. Jurnal Teknologi Reaktor Nuklir TRI DASA MEGA, 17 (2), 79.

The reduction of fissile material during reactor operation affects reactivity reduction. Therefore, in order to keep the reactor operating at fixed power, it must be compensated by slowly withdrawing the control-rod up. However, it will change the shape of the horizontal/axial power distribution and safety margin as well. The research carries out the calculations of the core thermal-hydraulics to determine the effect of the fluctuations of the power distribution on the thermal-hydraulic AP1000's parameters and study their impacts on the safety margin. The calculation is done using the COBRA-EN code and the result shows that the maximum heat flux at the Beginning of Cycle (BOC) is 1624.02 kW/m^2. This heat flux will then decrease by 22.75% at the Middle of Cycle (MOC) and by 0.29% at the End of Cycle (EOC). The peak fuel centerline temperature at the BOC, MOC and EOC, are 1608.15°C, 1232.15°C, and 1301.75°C, respectively. These temperature differences are caused by the heat flux effects on sub-cooled boiling regions in the cladding surface. Moreover, the value of MDNBRs at the MOC and EOC are 3.23 and 3.00, which are higher than the MDNBR at the BOC of 2.49. It could be concluded that the operating cycle of the AP1000 reactor should be operated in the MOC and the EOC, which will be more safely than be operated in the BOC.

Keywords: Core thermal-hydraulics, AP1000, fluctuation of power distribution, COBRA-EN

Andi Sofrany Ekariansyah., *Analysis On The Core Condition Of AP1000 Advanced Power Reactor During Small Break LOCA.* Jurnal Teknologi Reaktor Nuklir TRI DASA MEGA, 17 (2), 87.

Accident due to the loss of coolant from the reactor boundary is an anticipated design basis event in the design of power reactor adopting the Generation II up to IV technology. Small break LOCA leads to more significant impact on safety compared to the large break LOCA as shown in the Three-Mile Island (TMI). The focus of this paper is the small break LOCA analysis on the Generation III$^+$ advanced power reactor of AP1000 by simulating three different initiating events, which are inadvertent opening of Automatic Depressurization System (ADS), double-ended break on one of Direct Vessel Injection (DVI) pipe, and 10 inch diameter split break on one of cold leg pipe. Methodology used is by simulating the events on the AP1000 model developed using RELAP5/SCDAP/Mod3.4. The impact analyzed is the core condition during the small break LOCA consisting of the mixture level occurrence and the fuel cladding temperature transient. The results show that the mixture level for all small break LOCA events are above the active core height, which indicates no core uncovery event. The development of the mixture level affect the fuel cladding temperature transient, which shows a decreasingly trend after the break, and the effectifeness of core cooling. Those results are identical with the results of other code of NOTRUMP. The resulted core cooling is also due to the function of coolant injection from passive safety feature, which is unique in the AP1000 design. In overall, the result of analysis shows that the AP1000 model developed by the RELAP5 can be used for analysis of design basis accident considered in the AP1000 advanced power reactor.

Keywords: analysis, mixture level, fuel cladding temperature, small break LOCA, RELAP5.

Amir Hamzah, Iman Kuntoro., *Radiation Shielding Conseptual Design Of RRI-50 Reactor.* Jurnal Teknologi Reaktor Nuklir TRI DASA MEGA, 17 (2), 99.

One of the parameters that must be met in the design of nuclear reactors is radiation shielding design to ensure the security and safety of workers and the surrounding community.This study has been conducted to design radiation shielding of RRI-50 with high density U9Mo-Al fuel elements that consist of 21 pieces of plate type fuel elements with dimension as same as RSG-GAS fuel elements but the active length is 70 cm.Core configurations consist of 16 fuel elements and 4 control elements and 5 irradiation positions to form a matrix of 5 x 5.The objective of this research is to design radiation shielding and determine the distribution of dose rates in the working area and the environment of RRI-50 reactor.The early stages of this research is to calculate source strength and inventory of radioactive materials within the reactor core with one operation pattern cycleof 50 MW for 22 days using ORGEN2.1 program.Based on core source strength and models that are created using the VisEd software, the analysis

parameter of the shielding was determined iteratively using MCNPX program.In the final stage, an analysis of the dose rate distributions in the whole space inside and outside the reactor building was conducted also using MCNPX program.The results show that the height of the water surface is 1000 cm and the combination of heavy concrete thickness of 90 cm and ordinary concrete thickness of 60 cm can be used as an biological shield.This design can reduce the dose rate to 0.05 μSv/h in the Operations Room and Experiments Room as well as outside the reactor building to 4.2 μSv/h and 0.03 μSv/h during reactor operation.The results also suggest that the installation of additional radiation shield of 280 cm thickness within 300 cm in front of the open radial neutron beam tube can reduce gamma and neutron dose rate to 3.3 μSv/h and $3,1x10^{-11}$ *μSv/h.The results of this study indicate that the design of the radiation shield can reduce radiation to meet the safety limits set by BAPETEN for personnel and surrounding communities.*

Keywords :Radiation shielding, dose rates, radiation safety, RRI-50