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LEMBAR ABSTRAK

Anhar R. Antariksawan, Surip Widodo, Hendro Tjahjono., *Parametric Study Of LOCA In TRIGA-2000 Using RELAP5/SCDAP Code.* Jurnal Teknologi Reaktor Nuklir TRI DASA MEGA, 19 (2), 59.

A postulated loss of coolant accident (LOCA) shall be analyzed to assure the safety of a research reactor. The analysis of such accident could be performed using best estimate thermal-hydraulic codes, such as RELAP5. This study focuses on analysis of LOCA in TRIGA-2000 due to pipe and beam tube break. The objective is to understand the effect of break size and the actuating time of the emergency core cooling system (ECCS) on the accident consequences and to assess the safety of the reactor. The analysis is performed using RELAP/SCDAPSIM codes. Three different break size and actuating time were studied. The results confirmed that the larger break size, the faster coolant blow down. But, the siphon break holes could prevent the core from risk of dry out due to siphoning effect in case of pipe break. In case of beam tube rupture, the ECCS is able to delay the fuel temperature increased where the late actuation of the ECCS could delay longer. It could be concluded that the safety of the reactor is kept during LOCA throughout the duration time studied. However, to assure the integrity of the fuel for the long term, the cooling system after ECCS last should be considered.
Keywords: Safety analysis, LOCA, TRIGA, RELAP5

Jati Susilo, Tagor Malem Sembiring, Winter Dewayatna**.,** *Irradiation Characteristic Of Natural UO_2 Pin PHWR Target At PRTF Of RSG – GAS Core.,* Jurnal Teknologi Reaktor Nuklir TRI DASA MEGA, 19 (2), 71.

The RSG-GAS reactor has a facility for irradiation of the fuel pin of nuclear power reactor, namely Power Ramp Test Facility (PRTF). The in-house fabrication PWR fuel pin has prepared for irradiations in the PRTF facility, currently, while the various enrichments of uranium are analyzed using the analytical tool. In the next step, it is planned to perform an irradiation of PHWR fuel pin sample of natural UO_2 in the facility. Before irradiation in the core, it should be analyzed by using the analytical tool. The objectives of this paper are to optimize irradiation time based on the burn-up, the generated linear power and the neutron flux level at the target. The 3-dimension calculations have been carried out by using the CITATION code in the SRAC2006 code system. Since the coolant of the reactor is H_2O, the effect of moderators in the pressurized tube, H_2O and D_2O, were analyzed, as well as pellet radius and moderator densities. The calculation results show that the higher linear power as irradiation time longer is occurred preferably in the D_2O moderator than in H_2O. For the D_2O moderator, the higher pressure affects the lower density and longer irradiation time. The maximum irradiation time for natural UO_2 fuel pin with the pressurized D_2O moderator is about 9.5×10^4 h, with the linear power of 700 W/cm. During irradiation, neutronic parameters of the core such as excess reactivity and ppf show a very small change, still far below design value.
Keywords: PHWR, Neutron Flux, Thermal Power, PRTF, RSG-GAS

Hendro Tjahjono., *Investigation Of RDE Thermal Parameters Changes In Response To Long-Term Station Black Out.* Jurnal Teknologi Reaktor Nuklir TRI DASA MEGA, 19 (2), 83.

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Due to long-term station black out (SBO) of the RDE (Experimental Power Reactor), the residual heat from the core will be removed to a residual heat removal system (RHRS). The objective of this study is to know the transient characteristic of RDE thermal parameters in response to the loss of residual heat removing ability for long-term. To achieve this objective, an analysis model of reactor thermal parameters changes during SBO, using Matlab program to simulate heat transfer equations of conduction, convection and radiation has been performed. Using this program, the changes of RDE thermal parameters until 800 hours after reactor trip have been analyzed. It is concluded that, in long-term SBO condition, the reactor is still safe with the maximum core temperature of 1140℃, which is still far under the safety limit of 1600℃ as stated in the design criteria. More attentions are needed to be taken with the increasing of concrete temperature up to 600℃ when the water storage is empty. Therefore, the availability of water in the RHRS shall absolutely be maintained.

Keywords: Experimental power reactor, decay heat removal, passive, transient, Matlab

Muhammad Darwis Isnaini, Muhammad Subekti, Geni Rina Sunarya., *Prediction Of Fuel Temperature Of AP1000 Due To The Formation Of Crud And Oxide Layer*., Jurnal Teknologi Reaktor Nuklir TRI DASA MEGA, 19 (2), 93.

An analysis to predict the fuel temperature due to crud and oxide layer formed on the hot sub-channel cladding surface of AP1000 reactor has been performed. During reactor operation, the heat transfer and cooling process occur on the fuel cladding surface. During the heat exposure process, an oxide layer and crud are formed on the cladding surface. The decrease of heat transfer performance will increase the fuel and cladding temperatures. Therefore, the effect of fuel temperature increase during the heat exposure process has to be analyzed. The analysis was conducted for nominal power of 3400 MWt using COBRA-EN code, by varying the modular oxide thickness of 0, 20, 40, 60, 80, 100 and 120 μm, crud thickness of 0, 10 and 20 μm and black oxide thickness of 0, 10, 20, 30 and 40 μm. For full cycle hot sub-channel condition, the combination of crud thickness of 20 μm and modular oxide thickness of 115 μm give prediction of the peak fuel center line temperature and the peak cladding surface temperature of 1870.73℃ and 609.40℃, respectively. However, the oxide layer is predicted only formed on hot sub-channel during BOC (about 40% of full cycle). The results show that the prediction of the peak fuel center line temperature and the peak cladding surface temperature are 1713.18℃ and 451.87℃, respectively. Compared to the normal and fresh fuel conditions, the peak fuel center line temperature and the peak cladding surface temperature increase by 6.53% and 29.86%, respectively.

Keywords: Fuel temperature, crud, oxide layer, COBRA-EN, AP1000

I Putu Susila, Arif Yuniarto., *Design, Implementation And Verification Of Software Code For Radiation Dose Assessment Based On Simple Generic Environmental Model.* Jurnal Teknologi Reaktor Nuklir TRI DASA MEGA, 19 (2), 103

Radiation dose assessment to determine the potential of radiological impacts of various installations within nuclear facility complex is necessary to ensure evironmental and public safety. A simple generic model-based method for calculating radiation doses caused by the release of radioactive substances into the environment has been published by the International Atomic Energy Agency (IAEA) as the Safety Report Series No. 19 (SRS-19). In order to assist the application of the assessment method and a basis for the development of more complex assessment methods, an open-source based software code has been designed and implemented. The software comes with maps and is very easy to be used because assessment scenarios can be done through diagrams. Software verification was performed by comparing its result to SRS-19 and CROM software calculation results. Dose estimated by SRS-19 are higher compared to the result of developed software. However, these are still acceptable since dose estimation in SRS-19 is based on conservative approach. On the other hand, compared to CROM software, the same results for three scenarios and a non-significant difference of 2.25% in another scenario were obtained. These results indicate the correctness of our implementation and implies that the developed software is ready for use in real scenario. In the future, the addition of various features and development of new model need to be done to improve the capability of software that has been developed.

Keywords: Radiation dose assessment, software code, radioactive discharge, environment, IAEA SRS-19.

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