Validation of simbat-PWR using standard code of cobra-en on reactor transient condition/ Muhammad Darwis Isnaini, Muhammad Subekti

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Abstrak

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 is 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 are calculated by MATPRO sub routine (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% higher than COBRA-EN and 7.70% lower 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 in the SIMBAT-PWR and the COBRA-EN calculation, are due to the difference of the gap between heat transfer coefficient and the cladding thermal conductivity.