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LEMBAR ABSTRAK

Hendro Tjahjono**.,** *Investigation Of AP-1000 Containment Pressure And Temperature Transient Under Station Black Out Accident With Differents Pressure Set-Points*. Jurnal Teknologi Reaktor Nuklir TRI DASA MEGA, 17 (1), 1.

AP-1000 reactor applying external cooling concept to anticipate the increase in pressure due to Station Black Out (SBO). Disposal mechanism of decay heat conducted through the PRHR to IRWST and subsequently forwarded to the reactor containment. Containment is externally cooled through natural convection in the air gap and through evaporation cooling water poured on the outer surface of the containment wall when the pressure attaints 1.7 bars according the applied pressure set-point. With this mechanism, the pressure will increase until it reaches certain maximum value and then decrease when containment cooling already begun effective. The purpose of this study was to determine the effect of the set-point to the maximum pressure and temperature reached. The utilized method is to perform simulations using Matlab-07 model of analytical calculations based on a transient state that is capable of estimating the power of heat evacuated and the pressure in the containment. The simulation results show the pattern of pressure and temperature transient rises to a maximum and drops back to a value that is relatively constant. With the set-point variation ranging from 1.7 bars to 5 bars, the maximum pressure varies from 3.5 bars to 5 bars and the maximum temperature varies from 117 °C to 125 °C. It can be concluded that with increasing the set-point pressure of starting the external cooling with water, the maximum pressure and temperature increase.
Keywords: transient pressure, containment external coolingset-point, AP1000, SBO.

Anis Rohanda**.,** *Mass Changes Analysis of Fissile and Non Fissile Materials in The PWR 1000 MWe using ORIGEN-ARP 5.1.*, Jurnal Teknologi Reaktor Nuklir TRI DASA MEGA, 17 (1), 13.

Controlled fission reaction occurs in the reator core. Reactor components such as fuel, cladding and cooling water have an important role in the sustainability of the fission reaction. Fission reaction causes the formation of a number of fission product nuclides and activation products. Fission product nuclides are produced from thermal neutron capture reaction of fissile material while the activation products are originated from interaction of non-fissile materials such as cladding material and coolant by neutron and gamma. At each of reactor operation, the information of fuel material changes in the form of non-fissile or fissile material, is very usefull for the management of core fuel, such as for reactivity control, optimization and loading of fuel. Hence, it is need to perform a research in the fissile and non-fissile material changes in the reactor core. This can be done by observing the change of material mass in the reactor core. This objective of this research work is to determine the change in mass of material in the core, such as the mass of the nuclear fuel elements, cladding and cooling water after use in the core. From mass changes can be delivered to burn up calculation or fuel consumption level. The calculation were performed on the basis of the United States PWR 1000 MWe by using a fission inventory computer code of ORIGEN-ARP 5.1, a new version of ORIGEN with specific library for nuclear power plant. The analysis results show that the U-235 fissile material having a mass reduction up to 58 % or more than half from the initial U-235 mass for each operation cycle *period. Fertile material U-238 was reduced by about 2% only from the initial mass for each operating cycle period. For other cases, the non-fissile material case, mass changes reduced in various for each operation cycle, depend on activation cross-sections and decay and formation rate of parent nuclides.*

Keyword: fissile material, non fissile material, PWR, ORIGEN-ARP 5.1

Sukmanto Dibyo, Endiah P.H, Ign. Djoko Irianto., *Design Analysis Of Cooling System Process The Innovative Research Reactor 50 MW*. Jurnal Teknologi Reaktor Nuklir TRI DASA MEGA, 17 (1), 19.

Innovative Research Reactor RRI is a type of MTR (Material Testing Reactor), which is being prepared in the future as a design of new reactor. The power of RRI has been determined based on the core thermalhydraulic and neutronic calculation is 50 MWt. The reactor pressure is 8 kgf/cm ^{2} and coolant mass flow rate is 900 kg/s. The important challenge in the follow up of this reactor design is the design analysis of cooling system. The purpose of this study is to analyze the design of RRI reactor main coolant system at the power of 50 MWt (RRI-50) using ChemCAD 6.1.4. In this analysis the mass and energy balances at the primary and secondary cooling system are calculated as main coolant. Each of the cooling system consists of two lines operating in parallel and redundancy lines. Besides that, the thermal design of the component units have been analyzed using RELAP5, FrenchCreek and Analytical Methods. The analyses result obtained is a design of cooling system diagram which includes parameter of enthalpy, temperature, pressure and coolant mass flow rate of each line. Meanwhile, design result of main component unit are delay tank of 51.5 m^3 volume, 2 unit centrifugal pumps and 1 unit stand-by pump for the primary coolant pump each of 141 kW power and secondary coolant pump each of 206 kW power, 2 unit of shell-tube heat exchanger with overall thermal coefficient of 1377 W/m$^{2\cdot o}$C and 4 unit cooling tower that capable to release the heat to the air at approach temperature of 5,0 ^{o}C and range temperature of 9,0 ^{o}C. Design of reactor coolant system RRI-50 has decided the operating parameters of cooling system are temperature, pressure and mass flow rate by considering into the demands of the safety aspects of the reactor core therefore design of maximum coolant temperature to the reactor core is44,5 ^{o}C.

Keywords : RRI 50MW, design of cooling system, program ChemCAD 6.1.4.

Zuhair, Suwoto., *Analysis Of The Effect Of Water Ingress Accident On Doppler Reactivity Of RGTT200K Core*. Jurnal Teknologi Reaktor Nuklir TRI DASA MEGA, 17 (1), 31.

In HTR, the negative temperature reactivity coefficient guarantees fission reaction in the core remain under the control and decay heat will not melt the fuel which cause the release of radioactive substances into the environment. But the entry of water (water ingress) into the reactor core due to rupture of a steam generator tube heat exchanger, which is known as one of the design basis accidents, can introduce positive reactivity with other potential hazards such as graphite corrosion and damage of the reflector structure material. This paper will investigate the effect of water ingress accident on Doppler reactivity coefficient of RGTT200K core. The capability of the Doppler reactivity coefficient to compensate positive reactivity incurred during water ingress accident will be examined through a series of calculations with MCNPX code and ENDF/B-VII library for fuel temperature changes from 800K to 1800K. Three options of UO_2, ThO_2/UO_2 and PuO_2 fuel kernels with three lattice models of fuel pebble in the reactor core was applied for condition of water ingress with water density from 0 to 1000 kg/m^3. The results of the calculations show that Doppler reactivity coefficient is negative for the entire fuel options being considered even for a large possibility of water ingress. The effects of water ingress becomes stronger in lattice model with lower packing fraction because more volume available for water entering the reactor core. The effect of water ingress is also stronger in the uranium core compared to thorium and plutonium cores as a consequence of the Doppler phenomenon where the neutron absorption in resonance region of ^{238}U is greater than ^{232}Th and ^{240}Pu. It can be concluded overall that Doppler coefficient of RGTT200K core has capability to compensate the reactivity insertion introduced by water ingress accident. RGTT200K core with UO_2, ThO_2/UO_2 and PuO_2 fuels can maintain the inherently safety features in a passive way.

Keywords: water ingress, Doppler reactivity, RGTT200K

Sudarmono., *Investigation On Thermal-Flow Characteristics Of HTGR Core Using Thermix-Konvek Module And VSOP'94 Code*. Jurnal Teknologi Reaktor Nuklir TRI DASA MEGA, 17 (1), 41.

The failure of heat removal system of water-cooled reactor such as PWR in Three Mile Islands and Fukushima Daiichi BWR makes nuclear society starting to consider the use of high temperature gas-cooled reactor (HTGR). Reactor Physics and Technology Division – Center for Nuclear Reactor Safety and Technology (PTRKN) has tasks to perform research and development on the conceptual design of cogeneration gas cooled reactor with medium power level of 200 MWt. HTGR is one of nuclear energy generation system, which has high energy efficiency, and has high and clean inherent safety level.The geometry and structure of the HTGR200 core are designed to produce the output of helium gas coolant temperature as high as 950℃ to be used for hydrogen production and other industrial processes in co-generative way. The output of very high temperature helium gas will cause thermal stress on the fuel pebble that threats the integrity of fission product confinement. Therefore, it is necessary to perform thermal-flow evaluation to determine the temperature distribution in the graphite and fuel pebble in the HTGR core. The evaluation was carried out by Thermix-Konvek module code that has been already integrated into VSOP'94 code. The HTGR core geometry was done using BIRGIT module code for 2-D model (RZ model) with 5 channels of pebble flow in active core in the radial direction. The evaluation results showed that the highest and lowest temperatures in the reactor core are 999.3 ℃ and 886.5 ℃, while the highest temperature of TRISO UO_2 is 1510.20℃ in the position (z= 335.51 cm; r=0 cm). The analysis done based on reactor condition of 120 kg/s of coolant mass flow rate, 7 MPa of pressure and 200 MW_{th} of power. Compared to the temperature distribution resulted between VSOP'94 code and fuel temperature limitation as high as 1600^oC, there is enough safety margin from melting or disintegrating.

Keywords: Thermal-Flow, VSOP'94, Thermix-Konvek, HTGR, temperature

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