J. Tek. Reaktor. Nukl.
Vol. 17 No. 3 Oktober 2015, Hal. 175-177
(Masa berlaku Akreditasi s/d Mei 2018)

ISSN 1411–240X
Nomor: 632/AU3/P2MI-LIPI/03/2015

JURNAL TEKNOLOGI REAKTOR NUKLIR TRI DASA MEGA

Volume 17, Nomor 3, Oktober 2015

LEMBAR ABSTRAK

Azizul Khakim., *Thermal Hydraulic Safety Analysis Of Bulk Shielding Kartini Reactor.* Jurnal Teknologi Reaktor Nuklir TRI DASA MEGA, 17 (3), 115.

Bulk shielding is an integrated facility to Kartini reactor which is used for temporary spent fuels storage. The facility is one of the structures, systems and components (SSCs) important to safety. Among the safety functions of fuel handling and storage are to prevent any uncontrolable criticality accidents and to limit the fuel temperature increase. Safety analyses should, at least, cover neutronic and thermal hydraulic calculations of the bulk shielding. Thermal hydraulic analyses were intended to ensure that heat removal and the process of the spent fuels cooling takes place adequately and no heat accumulation that challenges the fuel integrity. Validated code, PARET/ANL was used for analysing the spent fuels cooling with natural convection mode. The calculations results concluded that natural convection cooling mode can adequately cools down the decay heat without significant increase in fuel temperatur.
Keywords: bulk shielding, spent fuels, natural convection, PARET

*Endiah Puji Hastuti, M. Subekti, Sukmanto Dibyo, M. Darwis Isnaini***.,** *Thermalhydraulic Design And Cooling System Optimization Of The High Power Inovative Research Reactor.,* Jurnal Teknologi Reaktor Nuklir TRI DASA MEGA, 17 (3), 127.

Implementation of reactor innovation has been implemented in a variety of new research reactors that currently are being built. At this time BATAN is designing a conceptual design of the high power research reactor whichhas entered the stage of design optimization. The conceptual design specifications of the innovative research reactor isa pool type reactor, water-cooled and reflected by D_2O. The reactor core has a 5 x 5 grid with 16 fuels and 4 control rods, which is inserted into a tube containing D_2O as an irradiation position. Reactor power of 50 MW is designed to generate thermal neutron flux of $5x10^{14}$ n/cm^{2}s. The compact core reactor is using U_9Mo-Al fuel with uranium loading of7-9 gU/cm^3. Thermal hydraulic design includes modeling, calculation and analysis of the adequacy of coolant created synergy with the physical design of reactor safety. This paper aims to deliver the results of thermal hydraulic calculation and system design analysis at steady state condition. The analysis was done using various calculation programs that have been validated, i.e. Caudvap, PARET-ANL, Fluent and ChemCad 6.4.1. The calculation results show that the heat generation can be transferred without causing a two phase flow boiling by applying pressurized reactor core design, while the main components of initial design system with an integrated heat dissipation has been done, to complete the conceptual design of the RRI-50 thermalhydraulics.
Keywords/Kata kunci:inovative research reactor, Caudvap, PARET-ANL, Fluent, ChemCad 6.4.1

Tagor Malem Sembiring, Surian Pinem, Peng Hong Liem., *Validation Of Full Core Geometry Model Of The Nodal3 Code In The PWR Transient Benchmark Problems.* Jurnal Teknologi Reaktor Nuklir TRI DASA MEGA, 17 (3), 141.

The coupled neutronic and thermal-hydraulic (T/H) code, NODAL3 code, has been validated

ISSN 1411–240X
Nomor : 632/AU3/P2MI-LIPI/0

in some PWR static benchmark and the NEACRP PWR transient benchmark cases. However, the NODAL3 code have not yet validated in the transient benchmark cases of a control rod assembly (CR) ejection at peripheral core using a full core geometry model, the C1 and C2 cases. By this research work, the accuracy of the NODAL3 code for one CR ejection or the unsymmetrical group of CRs ejection case can be validated. The calculations by the NODAL3 code have been carried out by the adiabatic method (AM) and the improved quasistatic method (IQS). All calculated transient parameters by the NODAL3 code were compared with the reference results by the PANTHER code. The maximum relative difference of 16% occurs in the calculated time of power maximum parameter by using the IQS method, while the relative difference of the AM method is 4% for C2 case. All calculation results by the NODAL3 code shows there is no systematic difference, it means the neutronic and T/H modules are adopted in the code are considered correct. Therefore, all calculation results by using the NODAL3 code are very good agreement with the reference results.
Keywords: nodal method, coupled neutronic and thermal-hydraulic code, PWR, transient case, control rod ejection.

Pande Made Udiyani, Sri Kuntjoro., *The Influence Of Atmospheric Conditions To Probabilistic Calculation Of Impact Of Radiology Accident On PWR-1000MWe.* Jurnal Teknologi Reaktor Nuklir TRI DASA MEGA, 17 (3), 149.

The calculation of the radiological impact of the fission products releases due to potential accidents that may occur in the PWR (Pressurized Water Reactor) is required in a probabilistic. The atmospheric conditions greatly contribute to the dispersion of radionuclides in the environment, so that in this study will be analyzed the influence of atmospheric conditions on probabilistic calculation of the reactor accidents consequences. The objective of this study is to conduct an analysis of the influence of atmospheric conditions based on meteorological input data models on the radiological consequences of PWR-1000MWe accidents. Simulations using PC-Cosyma code with probabilistic calculations mode, the meteorological data input executed cyclic and stratified, the meteorological input data are executed in the cyclic and stratified, and simulated in Muria Peninsula and Serang Coastal. Meteorological data were taken every hour for the duration of the year. The result showed that the cumulative frequency for the same input models for Serang coastal is higher than the Muria Peninsula. For the same site, cumulative frequency on cyclic input models is higher than stratified models. The cyclic models provide flexibility in determining the level of accuracy of calculations and do not require reference data compared to stratified models. The use of cyclic and stratified models involving large amounts of data and calculation repetition will improve the accuracy of statistical calculation values.
Keywords: accident impact,PWR 1000 MWe, probabilistic, atmospheric, PC-Cosyma

Hendro Tjahjono., *Effect Of Air Condition On AP-1000 Contaiment Cooling Performance In Station Black Out Accident.* Jurnal Teknologi Reaktor Nuklir TRI DASA MEGA, 17 (3), 159.

AP1000 reactor is a nuclear power plant generation III+ 1000 MWe which apply passive cooling concept to anticipate accidents triggered by the extinction of the entire supply of electrical power or Station Black Out (SBO). In the AP1000 reactor, decay heat disposal mechanism conducted passively through the PRHR-IRWST and subsequently forwarded to the reactor containment. Containment externally cooled through natural convection in the air gap and through evaporation cooling water poured on the outer surface of the containment wall. The mechanism of evaporation of water into the air outside is strongly influenced by the conditions of humidity and air temperature. The purpose of this study was to determine the extent of the influence of the air condition on cooling capabilities of the AP1000 containment. The method used is to perform simulations using Matlab-based analytical calculation model capable of estimating the power of heat transfered. The simulation results showed a decrease in power up to 5% for relative humidity rose from 10% to 95%, while the variation of air temperature of 10 °C to 40°C, the power will decrease up to 15%. It can be concluded that the effect of air temperature increase is much more significant in lowering the containment cooling ability compared with the increase of humidity.
Keywords: containment cooling, AP1000, air condition, SBO

ISSN 1411–240X
Nomor : 632/AU3/P2MI-LIPI/0

V. Indriati Sri Wardhani dan Henky P. Rahardjo., *Characterization Of Boundary Layer Thickness Of Nano Fluid ZrO_2 On Natural Convection Process,* Jurnal Teknologi Reaktor Nuklir TRI DASA MEGA, 17 (3), 167.

Cooling system is highly influenced by the process of convection heat transfer from the heat source to the cooling fluid. The cooling fluid usually used conventional fluid such as water. Cooling system performance can be improved by using fluids other than water such as nano fluid that is made from a mixture of water and nano-sized particles. Researchers at Batan Bandung have made nano fluid ZrO_2 from local materials, as well as experimental equipment for studying the thermohidraulic characteristics of nano fluid as the cooling fluid. In this study, thermohidraulic characteristics of nano fluid ZrO_2 are observed through experimentation. Nano fluid ZrO_2 is made from a mixture of water with ZrO_2 nano-sized particles of 10^{-7}-10^{-9} nm whose concentration is 1 g/ltr. This nano fluid is used as coolant in the cooling process of natural convection. The natural convection process depends on the temperature difference between heat source and the cooling fluid, which occur in the thermal boundary layer. Therefore it is necessary to study the thermal boundary layer thickness of nano fluid ZrO_2, which is also able to determine the local velocity. Experimentations are done with several variation of the heater power and then the temperature are measured at several horizontal points to see the distribution of the temperatures. The temperature distribution measurement results can be used to determine the boundary layer thickness and flow rate. It is obtained that thermal boundary layer thickness and velocity of nano fluid ZrO2 is not much different from the conventional fluid water.
Keywords: Boundary layer, nanofluid ZrO_2, natural convection.

J. Tek. Reaktor. Nukl.
Vol. 17 No. 3 Oktober 2015, Hal. 178-179
(Masa berlaku Akreditasi s/d Mei 2018)
ISSN 1411–240X
Nomor: 632/AU3/P2MI-LIPI/03/2015

JURNAL TEKNOLOGI REAKTOR NUKLIR TRI DASA MEGA

Volume 17, Nomor 3, Oktober 2015

INDEKS

ISSN 1411–240X
Nomor : 632/AU3/P2MI-LIPI/03/2015

K

L

U

J. Tek. Reaktor. Nukl.
Vol. 17 No. 3 Oktober 2015, Hal. 180
(Masa berlaku Akreditasi s/d Mei 2018)
ISSN 1411–240X
Nomor: 632/AU3/P2MI-LIPI/03/2015

JURNAL TEKNOLOGI REAKTOR NUKLIR TRI DASA MEGA

Volume 17, Nomor 3, Oktober 2015

INDEKS PENULIS

ISSN 1411–240X
Nomor: 632/AU3/P2MI-LIPI/03/2015

JURNAL TEKNOLOGI REAKTOR NUKLIR TRI DASA MEGA

Volume 17, Nomor 3, Oktober 2015

UCAPAN TERIMA KASIH

Redaksi mengucapkan terima kasih kepada Mitra Bestari :

1. Dr.-Ing. Ir. Sihana
2. Dr. Suharyana, M.Sc.
3. Dr. Yus Rusdian Akhmad, M.Eng.
4. Prof. Dr. Efrizon Umar

Atas partisipasinya dalam memberikan evaluasi dan saran perbaikan Volume 17 No. 1, 2 dan 3 tahun 2015 Jurnal Teknologi Reaktor Nuklir Tri Dasa Mega.

Dewan Redaksi