Jurnal Teknologi Reaktor Nuklir

Tri Dasa Mega

Journal homepage: jurnal.batan.go.id/index.php/tridam

Abstract Collection

Arif Yuniarto, Moh Cecep Cepi Hikmat., *The Study of Atmospheric Dispersion Model on Accident Scenario of Research Reactor G. A. Siwabessy Using HotSpot Codes as A Nuclear Emergency Decision Support System.* Tri Dasa Mega, 21 (1), 1.

G.A. Siwabessy Multipurpose Reactor (RSG-GAS) is a research reactor with thermal power of 30 MW located in the Serpong Nuclear Area (KNS), South Tangerang, Banten, Indonesia. Nuclear emergency preparedness of RSG-GAS needs to be improved by developing a decision support system for emergency response. This system covers three important aspects: accident source terms estimation, radioactive materials dispersion model into the atmosphere and radiological impact visualization. In this paper, radioactive materials dispersion during design basis accident (DBA) is modeled using HotSpot, by utilizing site-specific meteorological data. Based on the modelling, maximum effective dose and thyroid equivalent dose of 1.030 mSv and 26 mSv for the first 7 days of exposure are reached at distance of 1 km from the release point. These values are below IAEA generic criteria related to risk reduction of stochastic effects. The results of radioactive dispersion modeling and radiation dose calculations are integrated with Google Earth Pro to visualize radiological impact caused by a~~n~~ nuclear accident. Digital maps of demographic and land use data are overlayed on Google Earth Pro for more accurate impact estimation to take optimal emergency responses.

Keywords: G.A. *Siwabessy research reactor, Nuclear emergency, Atmospheric dispersion model, Decision support system, HotSpot codes*

Ramadhan Valiant Gill S.B, Yohannes Sardjono., *Optimization On Of Collimator Aperture Geometry For BNCT Kartini Research Reactor Using MCNPX.* Tri Dasa Mega, 21 (1), 9.

Boron Neutron Capture Therapy (BNCT) is one of the promising cancer therapy modalities due to it selectivity which only kills the cancer cells and does not damage healthy cells aroundcancer. In principle, BNCT utilizes the high ionization properties of alpha (^{4}He) and lithium (^{7}Li) particles derived from the reaction between epithermal and boron-10 neutrons ($^{10}B + n \rightarrow {}^{7}Li + {}^{4}He$) in cells, where trace distance of alpha and lithium particles is equivalent with cell diameter. The neutron source used in BNCT can come from a reactor, as a condition for conducting BNCT therapy tests, there are five standard parameters that must be met for a neutron source to be used as a source, and the standards come from IAEA. This research is based on simulation using the MCNPX program which aims to optimize IAEA parameters that have been obtained in previous studies by changing the shape of the collimator geometry from cone shape to cylinder with variations diameter from 3, 5 and 10 cm and also the simulation divided into two schemes namely first moderator Al is placed in a position 9.5 cm behind the collimator and the second is the moderator Al is pressed into the base point of the aperture in the collimator. In this work, neutrons originated from Yogyakarta Kartini research reactor have the energy range in the continuous form. The results of the optimization on each scheme of the collimator are compared with the outputs that have been obtained in previous studies where the aperture of the collimator is in the cone shape. The most optimal output obtained from the results is a collimator with a diameter of 5 cm in the second scheme where the results of IAEA parameters that are producedΦ_{epi} (n/cm^2 s) = 2.18E+8, $\dot{D}_f/\Phi_{epi}$ (Gy-cm^2/n) = 6.69E-13, $\dot{D}_\gamma/\Phi_{epi}$ (Gy-cm^2/n) = 2.44E-13, $\frac{\Phi_{th}}{\Phi_{epi}}$ = 4.03E-01, and J/Φ_{total} = 6.31E-01. These results can still be used for BNCT experiments but need a long irradiation time and when compared to previous studies, the output of the collimator with the diameter of 5 cm is more optimal

Keywords: BNCT, Collimator, IAEA Parameters, MCNPX, Cylindrical shape

Entin Hartini, Hery Adrial, Santosa Pujiarta., *Reliability Analysis of Primary and Purification Pumps in RSG-GAS Using Monte Carlo Simulation Approach.* Tri Dasa Mega, 21 (1), 15.

Reliability and maintenance play an important role in ensuring successful operation of a system. Reliability analysis is often used to determine the probability whether or not a system is functioning. However, limited available data and information are causing uncertainties and inaccuracies on component parameters. The purpose of this study is to conduct component/system reliability analysis using Monte Carlo simulation-based *method. This method enables us to estimate the reliability of components/systems including parameter uncertainty and imprecision. It is also useful to predict and evaluate maintenance decisions related to reliability. Monte Carlo method employs random number generation based on the probability of the distribution of processed data, of which then validated with real available data to ensure the simulation condition is relatively similar to real-life condition. The data used in this research is failure data on RSG-GAS components/systems for core configuration number of 81 to 95, accumulated from year 2013 to 2018. The results show that reliability values of components JE01/AP01-02 on TTF 233.619 is 0.579 while for components KBE01/AP-01-02 in TTF 185.38 is 0.368. The component reliability value is 60%, which implies that maintenance may be performed after 225 days and 100 days for components JE01/AP01-02 and KBE01/AP01-02, respectively*

Keywords: Reliability, Monte Carlo, Component damage, RSG-GAS

Tulis Jojok Suryono, Sigit Santoso, Restu Maerani., *Modeling of Operator's Actions on a Nuclear Emergency Condition Using Multilevel Flow Modeling*. Tri Dasa Mega, 21 (1), 23.

In nulear emergency condition, after determining the initiating event and the type of the anomaly, operators should take counteractions to control the reactor to mitigate the accident and to bring back the plant to the safe condition.The actions should based on emergency operating procedures. In order to minimize the human error related to the actions, some necessary information is needed. Such kind of information is the consequence of the actions, which can be derived by modeling the counteractions. Multilevel flow modeling (MFM), a functional modeling,is chosen to model the counteraction with the consideration that it is based on cause-effect relations and consequence reasoning, it provides realization relationship which corresponds physical components with their functions, and it provides comprehensive diagnosis based on human perspective of the system objectives. The counteractions are represented by the control functions in the MFM. This paper discusses how to model the counteractionsand the consequences of the actions to the system components, which are necessary to enhance situation awareness and to reduce human errors.

Keywords: Operator actions, Emergency operating procedures, Multilevel flow modeling, Control function, safey, human error

Mike Susmikanti, Roziq Himawan, Entin Hartini, Rokhmadi., *Analysis of 3D Semi-Elliptical Crack on Reactor Pressure Vessel Wall with Load Stress and Crack Ratio.* Tri Dasa Mega, 21 (1), 33

Reactor Pressure Vessel (RPV) wall is an important component in the Nuclear Power Plant (NPP). During reactor operation, RPV is subjected to high temperature, pressure, and neutron exposure. This condition could lead to RPV structure failure. In order to assure the integrity of RPV during the reactor lifetime, it is mandatory to perform a structural integrity assessment of RPV by evaluating postulated crack in RPV. In the previous study, the crack has evaluated in 2-D. However, 3-D analysis of semi-elliptic crack shape in the surface of the thick plate for RPV wall using SA 508 Steel is yet to be analyzed. The objective of this study is to analyze and modeling the evaluation in variation crack ratio with some load stress in 3-D. The Stress Intensity Factor (SIF) and J-integral are used as crack parameter. The J-Integral were calculated using MSC MARC MENTAT based on Finite Element Method (FEM) for obtaining the SIF value. The inputs are a crack ratio, load stress, material property, and geometry. The modeling of SIF value and goodness of fit are using MINITAB. The fracture condition could be predicted in comparison to the SIF value and fracture toughness. For the load stress 70 MPa and 80 MPa, with a crack ratio 0.25, 0.33 and 0.5, the material on RPV wall will in fracture condition.

Keywords: Semi Elliptic Surface Crack, 3-Dimension, Reactor Pressure Vessel, Elastic-plastic fracture mechanics, J-integral

Restu Maerani, Tulis Jojok Suryono, Muhammad Subekti., *Requirement Analysis of Computer-based Instrumentation and Control System for Reaktor Daya Eksperimental*, Tri Dasa Mega, 21 (1), 39.

Developing and licensing of digital Instrumentation and control (I&C) system for nuclear power plant (NPP) are challenging especially for the new construction since digital technology are composite with a very high complexity of many integrated systems. National Nuclear Energy Agency of Indonesia (BATAN), who designing "Reaktor Daya Eksperimental" (RDE), should prepare the documents to meet the licensing requirements of national regulator in this case Nuclear Energy Regulatory Agency of Indonesia (BAPETEN). BAPETEN's chairman regulation No.6 year of 2012 is the first national requirement which state requirements related to design of computer-based system concerning on safety of power reactor that should be followed. Since BAPETEN only denotes requirements without state which code and standards to be used, therefore BATAN can add references from International Nuclear Energy Agency (IAEA) guidelines. In this paper, requirement document traceability is developed to determine which code and standards should be used to verify and validate the I&C computer-based system of RDE. The hierarchy of regulatory and utility requirements are developed to guide the design basis documentation. Developing requirements analysis of computer-based I&C system RDE are completed after determining the design requirements from the utility and regulatory requirements. This methodology will help the design

engineers to follow the utility requirements by concerning to the production, and follow the regulatory requirements concerning the safety aspect

Keywords: Computer-based system, I&C System, Requirements Analysis, Licensing, RDE

Jurnal Teknologi Reaktor Nuklir

Tri Dasa Mega

Journal homepage: jurnal.batan.go.id/index.php/tridam

Keywords Index

Jurnal Teknologi Reaktor Nuklir

Tri Dasa Mega

Journal homepage: jurnal.batan.go.id/index.php/tridam

Acknowledgment

The following Peer Reviewers:

- Dr. Pande Made Udiyani, M.Si.
- Drs. Amir Hamzah, M.Si.
- Prof. Drs. Surian Pinem, M.Si.
- Dra.Elisabeth Supriyatni, M.App.Sc.
- Dr. Ir. Andang Widi Harto, M.T.
- Dr. Mulya Juarsa, S.Si., M.E.Sc.
- Dr. Sigit Santoso
- Dipl. Ing. (FH) Andi Sofrany Ekariansyah
- Drs. Tukiran Surbakti
- Dr. I Putu Susila

who have been involved in the reviewing of the articles in this issue of Tri Dasa Mega Vol. 21 No. 1 February 2019 are greatly acknowledged.