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LEMBAR ABSTRAK

Susyadi., *Thermal-Hydraulic Analysis Of SMR With Naturally Circulating Primary System During Loss Of Feed Water Accident*., Tri Dasa Mega, 18 (3), 117.

Small Modular Reactors (SMRs) have several advantages over conventional large reactors. With integral and simplified design, application of natural laws for safety system, and lower capital cost this reactor is very suitable to be deployed in Indonesia. One of SMR designs being developed implements natural driving force for its primary cooling system. With such innovative approach, it is important to understand safety implication of the design for all operating circumstances. One of conditions need to be investigated is the loss of feed-water (LoFW) accident. In this study, thermal-hydraulic performance of the SMR with naturally circulating primary system during LoFW accident is analysed. The purpose is to investigate the characteristics of flow in primary system during the accident and to clarify whether the naturally circulating coolant is adequately capable to transfer the heat from core in order to maintain safe condition under considered scenario. The method used is by representing the reactor system into RELAP5 code generic models and performing numerical simulation. Calculation result shows that following the initiating event and reactor trip, primary system flow becomes significantly fluctuated and coolant temperature decreases gradually, while in secondary side steam quality descends into saturated. The primary flow slows down from ~711 kg/s to ~263 kg/s and starts to increase up again at t= ~46 seconds. At the slowest point, fuel centerline and coolant temperatures were ~565 K and ~554 k, showing that temperatures of the fuel and coolant are still below its design limit and saturation point, respectively. This fact reveals that throughout transient the two main thermal hydraulic parameters stay in acceptable values so it could be concluded that under LoFW accident the SMR with naturally circulating primary system is in safe condition.

Keywords: SMR, loss of feed water, natural circulation, reactor safety, RELAP5

Setiyanto, Tukiran S**.,** *Analysis Of Gamma Heating At TRIGA MARK Reactor Core Bandung Using Plate Type Fuel.,* Tri Dasa Mega, 18 (3), 127.

In Accordance with the discontinuation of TRIGA fuel element production by its producer, the operation of all TRIGA type reactor of at all over the word will be disturbed, as well as TRIGA reactor in Bandung. In order to support the continuous operation of Bandung TRIGA reactor, a study on utilization of fuel plate mode, as used at RSG-GAS reactor, to replace the cylindrical model has been done. Various assessments have been done, including core design calculation and its safety aspects. Based on the neutronic calculation, utilization of fuel plate shows that Bandung TRIGA reactor can be operated by 20 fuel elements only. Compared with the original core, the new reactor core configuration is smaller and it results in some empty space that can be used for in-core irradiation facilities. Due to the existing of in-core irradiation facilities, the gamma heating value became a new factor that should be evaluated for safety analysis. For this reason, the gamma heating for TRIGA Bandung reactor using fuel plate was calculated by Gamset computer code. The calculations based on linear attenuation equations, line sources and gamma propagation on space. Calculations

were also done for reflector positions (Lazy Susan irradiation facilities) and central irradiation position (CIP), especially for any material samples. The calculation results show that gamma heating for CIP is significantly important (0,87 W/g), but very low value for Lazy Susan position (lest then 0,11 W/g). Based on this results, it can be concluded that the utilization of CIP as irradiation facilities need to consider of gamma heating as data for safety analysis report.

Keywords: gamma heating, nuclear reactor, research reactor, reactor safety

Sigit Santoso., *Factors Influencing Human Reliability Of High Temperature Gas Cooled Reactor Operation.,* Tri Dasa Mega, 18 (3), 135.

Operator roles and intervene actions on the operation of gas cooled reactor would be different compared to their roles in other reactor types. Analysis of operator performance and the influencing factors can be conducted comprehensively in Human Reliability Analysis (HRA). Using HRA, the impact of human errors on the system and the ways to reduce human error impact and frequency can be idenfified. The paper discusses factors influencing reactor operator performance to response to the cooling accident of the high temperature gas cooled reactor (HTGR). Analysis and qualification of influencing factors, which are performance shaping factors (PSF), were conducted based on time reliability curve and Cognitive Reliability and Error Analysis Method (CREAM). Based on time reliability curve, results showed that time variable contributes to the improvement of operator performance (PSF<1), especially when the safety features of the system properly work as in the design. Based on CREAM, it can be identified that in addition to the time variable, human machine interface design and sufficiently training also contribute to the improvement of operator performance. This study found that total PSF equals to 0.25, in which the positive dominant factor is time variable whose PSF is0.01 and the negative dominant factors are procedure and working cycle whose PSF is 5. Those PSF values reflected the multiplier factors to the human error probability. The analysis of performance shaping factors should be developed on the other operation and accident scenarios of HTGRs prior to be further applied for a comprehensive assessment and analysis of human reliability and for the design of human machine interface system at control room.

Keywords: PSF, HTGR, human operator, control room, human reliability

V. Indriati Sri Wardhani, Henky P. Rahardjo., *Pengaruh Bentuk Routing Perpipaan Sistem Pendingin Primer Reaktor TRIGA Konversi Terhadap Penurunan Aktivitas N-16 di Permukaan Tangki Reaktor.,* Tri Dasa Mega, 18 (3), 145.

Program konversi reaktor TRIGA 2000 Bandung dari bahan bakar silinder menjadi bahan bakar pelat perlu perancangan sistem pendingin reaktor yang baru. Perancangan sistem pendingin reaktor yang baru tersebut diusahakan tidak banyak mengalami perubahan dari sistem pendingin reaktor yang telah ada, mengingat ruang dan tempatnya tidak mungkin diubah. Oleh karena itu perlu dilakukan analisis untuk memilih routing perpipaan sistem pendingin reaktor TRIGA pelat yang dapat memenuhi persyaratan pendinginan sistem yang sesuai dengan kondisi ruang dan tempat yang telah ada. Mengingat batasan ruang yang ada maka ada 4 (empat) kemungkinan bentuk routing yang bisa dirancang. Dari keempat kemungkinan routing tersebut kemudian dilakukan analisis waktu tempuh partikel N-16 yang memancarkan radiasi gamma (γ) dari teras ke permukaan tanki reaktor. Penelitian dilakukan dengan mengasumsikan rapat massa (ρ) fluida pendingin konstan (fluida inkompresibel), seluruh N-16 yang dihasilkan dalam teras reaktor terangkut ke permukaan tanki reaktor. Hasilnya menunjukkan bahwa routing alternatif 3 adalah yang paling optimum, karena waktu tempuhnya mendekati 5 (lima) kali waktu paruh N^{16} (36,7047 detik), sehingga aktivitasnya turun dari 100% menjadi 3% nya (A/A_0 = 0,0317) dan panjang pipanya masih cukup untuk dimasukkan ke dalam ruang sistem pendingin reaktor yang tersedia.

Kata kunci: routing, perpipaan, aktivitas N-16, waktu paruh, reaktor TRIGA pelat

Roziq Himawan, Mike Susmikanti., *Circumferential Inhomogenity Analysis In G.A. Siwabessy Reactor's Primary Cooling Pipe.,* Tri Dasa Mega, 18 (3), 155.

In the in-service inspection conducted to G.A. Siwabessy reactor's primary cooling system pipe, it was found the presence of inhomogenity inside of welding part. To verify whether the inhomogeneity could be tolerated or not, comparative data from welding pre-service inspection is needed. Unfortunately, this weld wasn't covered in pre-service inspection.

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Therefore, this inhomogenity was needed to be analyzed. The purpose of this study is to evaluate the stress intensity factor of the inhomoginity, whether it is within a limit value or not and to predict the crack growth. Analysis were performed based on fracture mechanics theory using parameter of stress intensity factor. Two models were used for calculation approach that are plane crack model and semi-elliptic crack model. Hence, in order to predict the length of inhomogenity in the future, crack growth calculations were performed. The results showed that stress intensity values from both two models are remain below fracture toughness value of pipe's material. Besides that, stress intensity factor from plane crack model is higher than those from semi-elliptic crack model. Under consideration that inhomogenity has an arc shape in actual, thus, stress intensity factor from this inhomogenity still low enough compare to the fracture toughness. Crack growth calculation's results showed that after 300^{th} cycle of loading, the length of inhomogenity reaches approximately 2 mm. Base on operation data of G.A. Siwabessy reactor, 300 cycle number is corresponds to 30 years operation. Based oh these results it caould be concluded that the presence of inhomogenity in the welding part does not affect the structure's integrity of piping system.

Keywords : Inhomogenity, fracture mechanics, fracture toughness, stress intensity factor, crack growth

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