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LEMBAR ABSTRAK

Sri Sudadiyo., *Preliminary Design Of RDE Feedwater Pump Impeller.* Jurnal Teknologi Reaktor Nuklir TRI DASA MEGA, 20 (1), 1.

Nowadays, pumps are being widely used in the thermal power generation including nuclear power plants. Reaktor Daya Eksperimental (RDE) is a proposed nuclear reactor concept for the type of nuclear power plant in Indonesia. This RDE has thermal power 10 MW_{th}, and uses a feedwater pump within its steam cycle. The performance of feedwater pump depends on size and geometry of impeller model, such as the number of blades and the blade angle. The purpose of this study is to perform a preliminary design on an impeller of feedwater pump for RDE and to simulate its performance characteristics. The Fortran code is used as an aid in data calculation in order to rapidly compute the blade shape of feedwater pump impeller, particularly for a RDE case. The calculations analyses is solved by utilizing empirical correlations, which are related to size and geometry of a pump impeller model, while performance characteristics analysis is done based on velocity triangle diagram. The effect of leakage, pass through the impeller due to the required clearances between the feedwater pump impeller and the volute channel, is also considered. Comparison between the feedwater pump of HTR-10 and of RDE shows similarity in the trend line of curve shape. These characteristics curves will be very useful for the values prediction of performance of a RDE feedwater pump. Preliminary design of feedwater pump provides the size and geometry of impeller blade model with 5-blades, inlet angle 14.5 degrees, exit angle 25 degrees, inside diameter 81.3 mm, exit diameter 275.2 mm, thickness 4.7 mm, and height 14.1 mm. In addition, the optimal values of performance characteristics were obtained when flow capacity was 4.8 kg/s, fluid head was 29.1 m, shaft mechanical power was 2.64 kW, and efficiency was 52 % at rotational speed 1750 rpm.

Keywords: Blade, impeller, pump, RDE

Muhammad Mu'Alim, Yohannes Sardjono**.,** *Modeling The Radiation Shielding Of Boron Neutron Capture Therapy Based On 2.4 MeV D-D Neutron Generator Facility.,* Jurnal Teknologi Reaktor Nuklir TRI DASA MEGA, 20 (1), 13.

Radiation shield at Boron Neutron Capture Therapy (BNCT) facility based on D-D Neutron Generator 2.4 MeV has been modified with pre-designed beam shaping assembly (BSA). Modeling includes the material and thickness used in the radiation shield. This radiation shield is expected to protect workers from radiation doses rate that is not exceed 20 mSv·year-1 of dose limit values. The selected materials are barite, paraffin, polyethylene and lead. Calculations were performed using the MCNPX program with tally F4 to determine the dose rate coming out of the radiation shield not exceeding the radiation dose rate of 10 μSv·hr-1. Design 3 was chosen as the recommended model of the four models that have been made. The 3rd shield design uses a 100 cm thickness of barite concrete as primamary layer to surrounding 100 cm x 100 cm x 166.4 cm room, and a 40 cm borated polyethylene surrounding the barite concrete material. Then 10 cm barite concrete and 10 cm of borated polyethylene are added to reduce the primary radiation straight from the BSA after leaving the main layer. The largest dose rate was 4.58 μSv·h-1 on cell 227 and average radiation dose rate 0.65 μSv·hr-1. The dose rates are lower than the lethal dose

that is allowed by BAPETEN for radiation worker lethal dose.

Keywords: Radiation shield, tally, radiation dose rate, BSA, BNCT

Andi Sofrany Ekariansyah, Endiah Puji Hastuti, Sudarmono., *RELAP5 Simulation For Severe Accident Analysis Of RSG-GAS Reactor.* Jurnal Teknologi Reaktor Nuklir TRI DASA MEGA, 20 (1), 23.

The research reactor in the world is to be known safer than power reactor due to its simpler design related to the core and operational chararacteristics. Nevertheless, potential hazards of research reactor to the public and the environment can not be ignored due to several special features. Therefore the level of safety must be clearly demonstrated in the safety analysis report (SAR) using safety analysis, which is performed with various approaches and methods supported by computational tools. The purpose of this research is to simulate several accidents in the Indonesia RSG-GAS reactor, which may lead to the fuel damage, to complement the severe accident analysis results that already described in the SAR. The simulation were performed using the thermal hydraulic code of RELAP5/SCDAP/Mod3.4 which has the capability to model the plate-type of RSG-GAS fuel elements. Three events were simulated, which are loss of primary and secondary flow without reactor trip, blockage of core subchannels without reactor trip during full power, and loss of primary and secondary flow followed by reactor trip and blockage of core subchannel. The first event will harm the fuel plate cladding as showed by its melting temperature of 590 °C. The blockage of one or more subchannels in the one fuel element results in different consequences to the fuel plates, in which at least two blocked subchannels will damage one fuel plate, even more the blockage of one fuel element. The combination of loss of primary and secondary flow followed by reactor trip and blockage of one fuel element has provided an increase of fuel plate temperature below its melting point meaning that the established natural circulation and the relative low reactor power is sufficient to cool the fuel element.

Keywords: loss of flow, blockage, fuel plate, RSG-GAS, RELAP5

Jati Susilo, Tagor Malem Sembiring, M. Imron, Geni Rina Sunaryo., *Verification to The RSG-GAS Fuel Discharge Burn-Up Using SRAC2006 Module of COREBN/HIST.*, Jurnal Teknologi Reaktor Nuklir TRI DASA MEGA, 20 (1), 35.

For 30 years operation, some of the modifications to the RSG GAS core has been done, that are changes included the type of fuel from U3O8-Al to U3Si2-Al with the same density 2.96 gU/cc, the loading pattern of standard fuel elements / fuel control elements from 6/1 & 6/2 to 5/1 pattern, and in core fuel management calculation tool has been change from IAFUEL to BATAN-FUEL. To obtain an extension of the operating license for the next 10 years, the RSG-GAS Periodic Safety Assessment Document is need to prepared. According to the Regulatory Body Chairman Regulation No.2 2015, RSG-GAS safety assessment should be done independently. As part of this assessment the fuel discharge burn-up must be estimated. In this research, to ensure that the misposition of fuel element in the core has not occurred, the investigation to the document operating report related the fuel placement has been done. Therefore, by using 78th to 93rd operation data, verify of the fuel discharge burn-up of the RSG-GAS has been performed by using SRAC2006 module of COREBN/HIST. In addition, the results of these calculations are also made comparative with the operating report data that is calculated by using BATAN-FUEL. Maximum fuel discharge burn-up (57.73% of U-235) was verified still under permissible value determined by the regulatory body (<60% of U-235). Maximum differences value between two computer codes was about 2.12 % of U-235 (3.80%) that is fuel at the B-7 position. Fuel discharge burn-up of RSG-GAS showed almost the same value for each the operation cycle, range of 1.52% of U-235. So it can be concluded that the RSG-GAS core operation over the last ten years was in good fuel management performance, in accordance with the design. BATAN-FUEL has been comformed well enough with COREBN/HIST.

Keywords: Discharge Burn-Up, RSG-GAS, COREBN/HIST, BATAN-Fuel

Mike Susmikanti, Roziq Himawan, Jos Budi Sulistyo., *The Analysis Of Optimal Crack Ratio For PWR Pressure Vessel Cladding Genetic Algorithm.* Jurnal Teknologi Reaktor Nuklir TRI DASA MEGA, 20 (1), 47

Several aspects of material failure have been investigated, especially for materials used in Reactor Pressure Vessel (RPV) cladding. One aspect that needs to be analyzed is the crack ratio. The crack ratio is a parameter that compares the depth of the gap to its width. The optimal value of the crack ratio reflects the material's resistance to the fracture. Fracture resistance of the material to fracture mechanics is indicated by the value of Stress Intensity Factor (SIF). This value can be obtained from a J-integral calculation that expresses the energy release rate. The detection of the crack ratio is conducted through the calculation of J-integral value. The Genetic Algorithm (GA) is one way to determine the optimal value for a problem. The purpose of this study is to analyze the possibility of fracture caused by crack. It was conducted by optimizing the crack ratio of AISI 308L and AISI 309L stainless steels using GA. Those materials are used for RPV cladding. The minimum crack ratio and J-Integral values were obtained for AISI 308L and AISI 309L. The SIF value was derived from the J-Integral calculation. The SIF value was then compared with the fracture toughness of those material. With the optimal crack ratio, it can be predicted that the material boundaries are protected from damaged events. It can be a reference material for the durability of a mechanical fracture event.

Keywords: Fracture mechanics, RPV cladding, J-Integral, Stress Intensity Factor, Genetic Algorithm

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