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Abstract Collection

Tulis Jojok Suryono, Sudarno, Sigit Santoso, *Information Processing in the Reactor Protection Systems of High Temperature Gas-cooled Reactors.*, Tri Dasa Mega, 22 (3), 81.

Reactor protection systems (RPS) transform process variable signals from the sensors into initiation and actuation signals to trip the reactor if the signal's value exceeds the predefined trip setpoints of the RPS. Information on the current value of the process variables signals and the trip setpoint should be displayed properly on the visual display unit (VDU) in order to maintain the situation awareness of the operators in main control rooms (MCR). In addition, it is also helpful for them to investigate the cause of an accident after the reactor trip and to mitigate the accident based on the appropriate emergency operating procedures. This paper investigates how the information is processed in the RPS of Experimental Power Reactor (EPR) based on high temperature reactor (HTR) technology, and how the information is displayed on the human machine interface (HMI) of the MCR of the EPR. It is conducted by classifying the RPS into three layers based on its components and their functions, followed by the investigation of the type and the information processing in each layer. The results show that the form of the information has been changed throughout the RPS, started from the sensors and until it is displayed on the VDU. The results of the investigation are necessary to understand the concept of RPS, especially for new operators, and to prepare the mitigation actions based on the process variable that cause the reactor trip.

Keyword: Experimental power reactor, Reactor protection system, Human machine interface, Information processing, Situation awareness

Wahid Luthfi, Surian Pinem., *Calculation of 2-Dimensional PWR MOX/UO2 Core Benchmark OECD NEA 6048 with SRAC Code.* Tri Dasa Mega, 22 (3), 89.

The mixed uranium-plutonium oxide fuel (MOX/UO2) is an interesting fuel for future power reactors. This is due to the large amount of plutonium that can be processed from spent fuel of nuclear plants or from plutonium weapons. MOX/UO2 fuel is very flexible to be applied in thermal reactors such as PWR and it is more economical than UO2 fuel. However, due to the different nature of neutron interactions of MOX in PWR, it will change the reactor core design parameters and also its safety characteristic. The purpose of this study is to determine the accuracy of SRAC2006 code system in generation of cross-sections and calculation of reactor core design parameters such as criticality, reactivity of control rods and radial power distribution. In this study, PWR MOX/UO2 Core Transient Benchmark is used to verify the code that models a MOX/UO2 fueled core. SRAC-CITATION result is different from DeCART by 0.339% from. SRAC-CITATION result of single rod worth in All Rods Out (ARO) conditions are quite good with a maximum difference of 6.34% compared to BARS code and 4.74% compared to PARCS code. In All Rods In (ARI) condition, SRAC-CITATION results compared to the PARCS code is quite good where the maximum difference is 9.72%, but compared to BARS code, it spikes up to 33.24% at maximum difference. In the other case, overall radial power density results are quite good compared to the reference. Its maximum deviation from DeCART code is 5.325% in ARO condition and 6.234% in ARI condition. Based on the results of these calculations, SRAC code system can be used to generate cross-section and to calculate some neutronic parameters. Hence, it can be used to evaluate the neutronic parameters of the MOX/UO2 PWR core design.

Keyword: MOX/UO2 fuel, Criticality, Power peaking factor, SRAC2006

Muzakkiy Putra M. Akhir, Rina Kamila., *Development of Automatic Data Processing for BATAN's HRPD and FCD/TD Using Python Code.* Tri Dasa Mega, 22 (3), 97.

High Resolution Powder Diffractometer (HRPD) and Four Circle Diffractometer/Texture Diffractometer (FCD/TD) are two BATAN-owned neutron diffractometers which have been fully operational since 1992. These are used to investigate structure and texture of crystalline materials, respectively. Before analyzing, the acquired raw neutron diffraction data should first be processed in a specific way to achieve the suitable data format required by the analysis software. This data processing step is a repetitive task

for every single experiment which is previously done manually and very time-consuming. The purpose of this development project was to optimize this step to be fully automatic and executable by a code. This work was performed by means of Python code utilizing the array manipulation in re-arranging and re-formatting the raw data. The resulted Python codes were named as hrpd.py and fcdtd.py. These have been successfully done and validated, making data processing step easier, simpler, and significantly faster with only 20 seconds or less required.

Keyword: HRPD, FCD/TD, Automatic Data Processing, Neutron Diffraction, Python

Prasetyo Haryo Sadewo, Puradwi Ismu Wahyono., *Safety Analysis of Neutron Interaction with Material Practicum Module for the Kartini Internet Reactor Laboratory.*, Tri Dasa Mega, 22 (3), 104.

Kartini Research Reactor, which is situated in Yogyakarta, is a 100 kW TRIGA (Training, Research, and Isotope Production by General Atomic)-type reactor mainly used for educational and training purposes. A system for remote learning on nuclear reactor physics named the Internet Rector Laboratory has been developed and is fully operational since 2019. To enrich its curriculum, a new practicum module has been developed, that can be immediately implemented and does not require any additional equipment or materials. To ensure safety in reactor kinetics and radiation protection, a safety analysis on the implementation of the practicum module has been conducted using MCNP and ORIGEN utilizing the current conditions of the reactor regarding its fuel burnup and control rod positions at a certain power level. Based on the results of the analysis, the practicum is safe to perform from a neutronic and radiation protection perspective. Given the long half-life and the large amount of radiation exposure that comes from activation products of iron, it is recommended that only cadmium, boron, graphite, and aluminum are allowed to be irradiated during the practicum.

Keyword: Internet Reactor Laboratory, Activation Product, Radiation Protection, Reactor Safety

Muhammad Darwis Isnaini, Iman Kuntoro, Muh. Subekti., *Transient Analysis of Simultaneous LOFA and RIA in RSG-GAS Reactor after 32 years Operation.* Tri Dasa Mega, 22 (3), 111

During the operation of the research reactor RSG-GAS, there are many design parameters should be verified based on postulated accidents. Several design basis accidents (DBA) such as loss of flow accident (LOFA) and reactivity-initiated accident (RIA) also have been conducted separately. This paper discusses about possibility of simultaneous accidents of LOFA and RIA. The accident analyses carry out calculation for transient condition during RIA, LOFA, and postulated accident of simultaneous LOFA-RIA. This study aims to conduct a safety analysis on simultaneous LOFA and RIA, and investigate the impact on safety margins. The calculations are conducted by using the PARET code. The maximum temperature of the center fuel meat at nominal power of 30 MW and steady state conditions is 126.10°C and MDNBR of 2.94. At transients condition, the maximum center fuel meat temperature for LOFA, RIA and simultaneous LOFA-RIA are consecutively 132.99°C, 135.67°C and 138.21°C, and the time of reactor trip are 3.2593s, 3.6494s and 2.7118s, respectively. While the MDNBR for LOFA, RIA and simultaneous LOFA-RIA are respectively at transient condition are 2.88, 2.58 and 2.63, respectively. It is shown that, simultaneous LOFA-RIA has the fastest trip time. In this case, the low flow trip occurs first in advance to over power trip. From these results, it can be concluded that the RSG-GAS has adequate safety margin against transient of simultaneous LOFA-RIA.

Keyword: RSG-GAS, Simultaneous, LOFA, RIA, PARET

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