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Estimation of the dose rate of nuclear fuel of Ghana Research Reactor-1 (GHARR-1) using ORIGEN-S and MCNP 6

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ABSTRACT

Ghana is in the process of converting its fuel from Highly Enriched Uranium (HEU) to Lowly Enriched Uranium (LEU). Radioactive fission and activation products generated in the irradiated nuclear fuel are hazardous to personnel, environment and the public. Investigation into the levels of radiation dose will help in a safe and efficient core conversion process. Two computer codes used were ORIGEN-S; for computing changes in the isotopic concentrations during neutron irradiation and radioactive decay as well as to determine the source term and MCNP6; which used the source term estimated by ORIGEN-S code to calculate the dose rate. The criticality of the core at different heights above the bottom of the core was also obtained. Most of the radionuclides present after the core depletion contributed to the source term of $1.767 \times 10^{13} \pm 0.0008$ photons/sec which was observed after thirty days of the cooling period. The dose rates ranged between $3.51 \times 10^{-25} \pm 0.0003$ mGy/h and $4.27 \times 10^{4} \pm 0.0006$ mGy/h at different positions above the reactor core, the control room (wall, door and window) and the rabbit room. The criticality (k_{eff}) also decreased from 0.99442 ± 0.00006 to 0.01238 ± 0.00002 as the core moved from the bottom of the reactor vessel to the top.

1. Introduction

Although many Research Reactors (RR) have been shut down and there is a fall in the number of new RRs being brought into operation, it does not make RRs unnecessary as the new ones are innovative, multipurpose reactors designed to produce high neutron fluxes that will meet the nuclear research and development needs of the countries in which they are being built (Essel, 2016).

Research Reactors of less than 100 kW operates with a lifetime core and as such no spent fuel arises until they are permanently shut down. The program to convert research reactors from the use of highly enriched uranium (HEU) to low enriched uranium (LEU) started under the U.S. Department of Energy (DOE) with the RERTR (Reduced Enrichment for Research and Test Reactors) in 1978. The current core of Ghana Research Reactor-1 (GHARR-1) reactor which is less than 100 kW is in the process of being converted from Highly Enriched Uranium to Low Enriched Uranium (Abrefah et al., 2012).

Several numerical and computational methods or approaches based on nuclear reactor physics, nuclear engineering, depletion codes and mathematical theories exist for performing core depletion analysis and particle transport analysis (Abrefah et al., 2013).

With improved mathematical and numerical computational methods, direct solution of the transport equation for all time is computationally prohibitive so that quasi-static and iterative approximations must be made.

On the other hand, the use of depletion codes is very important in the calculation of dose rate in various systems. Several depletion codes exist that can be used to deplete reactor core in order to achieve radionuclides inventory and source term. ORIGEN-S is a readily available tool which uses matrix exponential methods to solve a large system of differential equations.

At the back end of the fuel cycle, the spent nuclear fuel is a huge challenge reactor operator's encounter. This must be safely removed, transported, stored and managed well pending its reprocessing or disposal. Spent fuel management is one of the most vital and common problems for countries with reactors. It also invokes the concern of the public at large (IAEA, 1993).

The composition of nuclear fuel are fission products, actinides and activation elements transmute by beta and gamma decay into other daughter radionuclides which could be very harmful to the body. The transmutation also results in the release of neutrons and photons. These particles can be harmful when certain limits are exceeded. It is

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imperative to protect the personnel, public and environment from their effect.

The determination of the dose rate emanating from the fuel as well as the amount of neutrons present is key to the core conversion activity. As the core is moved out of the reactor vessel, the photons and actinides as well as the other fission product that are released should be As Low As Reasonably Achievable (ALARA).

This work seeks to determine the dose rate so as to help in putting in place proper measures during unloading of the core. The type of shielding to be adopted as well as the correct Personal Protective Apparels (PPAs) to be used will be obtained.

2. Ghana Research Reactor-1 (GHARR-1)

Ghana Research Reactor-1 (GHARR-1) is a commercial Miniature Neutron Source Reactor (MNSR) similar to the Canadian SLOWPOKE in design (Akaho et al., 1995).

It is a 30 kW tank-in-pool reactor, producing a peak or maximum thermal neutron flux in the core and its inner irradiation channels of 1×10^{12} n cm^{-2} s^{-1}. The reactor which is designed to be compact and safe has made valuable contribution in the nuclear power industry. It is used mainly for Research and Development in reactor and nuclear engineering, neutron activation analysis, production of short-lived radioisotopes, human resource development for Ghana's nuclear program and for education and training.

Presently, GHARR-1's core consists of a fuel assembly HEU (UAl_4 alloyed) fuel elements arranged in ten concentric rings about a central control rod guide tube, which houses the reactor's only control rod. The control rod's reactivity worth is about 7mk, providing a core shutdown margin of 3mk of reactivity. The small core has a low critical mass. However, its relatively large negative temperature coefficient of reactivity is capable of boosting its inherent safety properties. The small size of the core facilitates neutron leakage and escape in both axial and radial directions. To minimize such loses and thereby conserve neutron economy, the core is heavily reflected on the side and underneath the fuel cage (the cage consists of the cylindrical frame that hold the 344 fuel pins in place, the fuel pins themselves and the tie rods; it is referred to as the cage because it resembles a bird cage. The cage is made of stainless steel and has a diameter of 23 cm ad a height of 23 cm) by a thick annulus and slab of beryllium alloy material (Akaho et al., 1995). Figs. 1 and 2 depicts cross sectional view of GHARR-1.

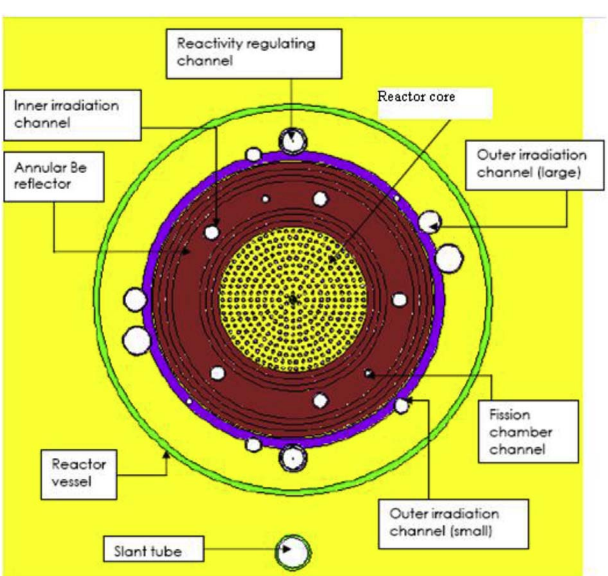

Fig. 1. GHARR-1 core configuration showing region.

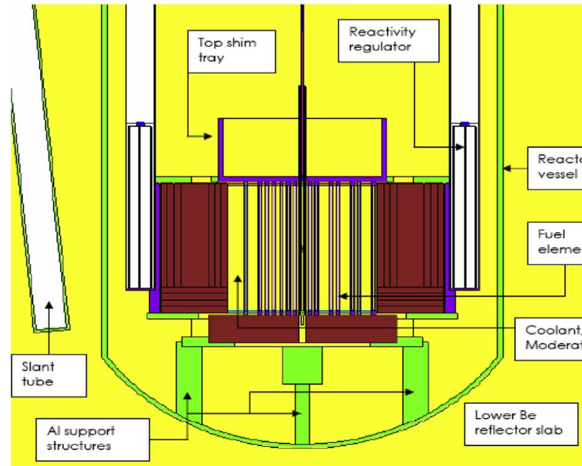

Fig. 2. MCNP plot of vertical cross-section of GHARR-1 reactor (in full power mode).

3. Theory

In the course of this study two main codes were employed, ORIGEN-S and MCNP6. In determining the time dependence of nuclide concentrations, ORIGEN-S, solves for the formation and disappearance of a nuclide by radioactive disintegration and neutron transmutation. Mathematically,

$$\frac{dN_i}{dt} = Formation\ Rate - Destruction\ Rate - Decay\ Rate \tag{1}$$

ORIGEN-S considers radioactive disintegration and neutron absorption (capture and fission) as the processes appearing on the right-hand side of Eq. (1). The time rate of change of the concentration for a particular nuclide, N_i, in terms of these phenomena can be written as

$$\frac{\mathrm{dN_i}}{\mathrm{dt}} = \sum_j \gamma_{j,i}\sigma_{f,j}N_j\phi + \sigma_{c,i-1}N_{i-1}\phi + \lambda'_iN'_i - \sigma_{f,i}N_i\phi - \sigma_{c,i}N_i\phi - \lambda_iN_i \tag{2}$$

wherei = 1, …I.$\sum_j\gamma_{j,i}\sigma_{f,j}N_j\phi$ = yield rate of N_i due to the fission of all nuclides N_j; $\sigma_{c,i-1}N_{i-1}\phi$ = rate of transmutation into N_i due to radiative neutron capture by nuclide N_{i-1}; $\lambda'_iN'_i$ = rate of formation of N_i due to the radioactive decay of nuclides $N_i\sigma_{f,i}N_i\phi$ = destruction rate of N_i due to fission.$\sigma_{c,i}N_i\phi$ = the destruction rate of N_i due to all forms of neutron absorption other than fission ((n,g), (n,a), (n,p), (n,2n),(n,3n)); λ_iN_i = radioactive decay rate of N_i

Equation (2) is written for a homogeneous medium containing a space-energy-averaged neutron flux, 4, with flux-weighted average cross sections, σ_f andσ_c, representing the reaction probabilities.

The flux is a function of space, energy and time, is dependent upon the nuclide concentrations. The mathematical treatment in ORIGEN-S assumes that the space-energy-averaged flux can be considered constant over a sufficiently small time interval, Δt.

Similarly, it is assumed that a single set of flux weighted neutron cross sections can be used over the same time step. For a given time step, these assumptions are necessary if Eq. (2) is to be treated as a first-order, linear differential equation. The time-dependent changes in the flux and weighted cross sections are simulated in ORIGEN by providing a capability of updating the values for the space-energy-averaged flux and, therefore, for the weighted cross sections for each successive time step, Δt_k, Δt_{k-1}……Δt_n. These values are derived from lattice cell analyses using physics transport methods to update cross sections that represent the lattice geometry, conditions, and the nuclide concentrations.

Two basic ways are used in Monte Carlo methods for solving the transport equation: mathematical method for numerical integration and computer simulation for physical processes. Mathematical approach is