Available online at www.elixirpublishers.com (Elixir International Journal)

Nuclear and Radiation Physics

Elixir Nuclear & Radiation Phys. 53 (2012) 11988-11991

ISSN: 2229-712X

Fuel Depletion Studies of Ghana Research Reactor-1 using ORIGEN2: Part 2 - Analyses of Photon source density

R.G. Abrefah[1,2,*], S.A. Birikorang[1,2], B.J.B. Nyarko[1,2], J.J. Fletcher[2] and E.H.K. Akaho[2]

[1]Ghana Atomic Energy Commission, National Nuclear Research Institute, P.O. Box LG80, Legon- Accra, Ghana.

[2]University of Ghana, School of Nuclear and Allied Sciences, P.O. Box AE1, Atomic Energy, Accra-Ghana.

ARTICLE INFO

Article history:

Received: 4 October 2012;

Received in revised form:

7 December 2012;

Accepted: 7 December 2012;

Keywords

Burnup,

Photon density fission products,

actinides,

Activation products.

ABSTRACT

In connection with conversion of Ghana Research Reactor-1 (GHARR-1) from the use of Highly Enriched Uranium (HEU) fuel to the use of Low Enriched Uranium (LEU) fuel with power upgrading which started in 2006, it is imperative to know how the spent fuel is stored. After the conversion process, the current HEU core will need a Spent Nuclear Fuel (SNF) storage cask to store the HEU fuel. Studies into suitable spent fuel storage cask are ongoing to help solve this challenge. The photon source densities of fission products, actinides and activation products relevant for assessing radiological consequence as well as designing and fabricating Spent Nuclear Fuel storage cask for GHARR-1 using local shielding material has been estimated for its 90.2% HEU fuel. Results showed that the bounding radiological dose due to hypothetical accidents can be evaluated at the end of irradiation (at discharge). Results have also proven key to the selection of shielding materials for the SNF cask.

Introduction

The isotopic inventory of a nuclear reactor fuel continuously evolves from commissioning till end of cycle due to fuel depletion (Feltus, 1995). Burnup calculations are essential for the prediction of long term changes in the isotopic composition of the HEU core of the Ghana Research Reactor-1(GHARR-1) due to neutron irradiation during its operation. The changes that occur within space and time have an influence on the operating life stability and control of the reactor. It is important that there is enough excess reactivity to operate the reactor from the time of fresh fuel loading till shutdown. Burnup calculations in thermal reactors use transport theory to represent fuel rods, rod clusters, control rods and burnable poisons by many time-dependent variables which must be considered in the analyses and also by geometric complexities which introduces time dependent spatial variations (Crowther, 1973). In this work, ORIGEN2 code package, a zero-dimensional isotope decay and transmutation code was used to calculate the burnup. The code solves the transmutation equations using libraries of radioactive decay data and 1-group cross section data. The base code libraries allow the tracking of over 100 actinides and nearly 900 fission product nuclides. The results of the burnup variations of the isotopic inventories of these fission products are presented. The results would in particular be helpful in performing decay heat removal calculations for the design of a spent fuel storage cask for the HEU fuel of GHARR-1. The purpose of this work is to obtain the photon source densities that will be essential in the design of a nuclear spent fuel cask for GHARR-1 using local serpentine material. It is also essential in assessing the radiological consequence of GHARR-1.

Description of Reactor

The Ghana Research Reactor-1 (GHARR-1) is a commercial version of the Miniature Neutron Source Reactor (MNSR) and belongs to the class of pool-in-tank type reactors (Gao et al, 1993) (Nyarko et al, 2009). It is under-moderated with an H/U atom ratio of 197. The thermal power rate of the facility is 30 kW with a corresponding peak thermal neutron flux of 1.0×10E12 n/cm2 s (Nyarko et al, 2009). For fresh core, its cold clean excess reactivity is about 4 mk. Cooling is achieved by natural convection using light water. Presently, the GHARR-1 core consists of an HEU fuel assembly (U-Al alloyed) and 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 -7 mk, providing a core shutdown margin of 3 mk 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 (Nyarko et al, 2009). 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, respectively, on the side and underneath the fuel cage by a thick annulus and slab of beryllium alloy material. Due to its inherent safety features, stability of flux and moderate cost, the MNSR has recently found enormous application in various fields of science (Kennedy et al,2000) particularly in trace elements in matrices of biological and environmental samples (Su-De, 1984) and soil fertility studies and geochemical mapping (Umar,2003).

Theory

In determining the time dependence of nuclide concentrations, ORIGEN2 is primarily concerned with developing solutions for the following equation

$$\frac{dN_i}{dt} = \text{Formation rate} - \text{Destruction rate} - \text{Decay rate} \quad (1)$$

ORIGEN (Bell, 1973) considers radioactive disintegration and neutron absorption (capture and fission) as the processes appearing on the right-hand side of Eqn. (1). The time rate of

Tele:

E-mail addresses: adibola2001@ yahoo.com

change of the concentration for a particular nuclide, Ni, in terms of these phenomena can be written as

$$\frac{dN_i}{dt} = \sum_j \gamma_{ji}\sigma_{f,j}N_j\phi + \sigma_{c,i-1}N_{i-1}\phi + \lambda_i' N_i' - \sigma_{f,i}N_i\phi - \sigma_{c,i}N_i\phi - \lambda_i N_i \quad (2)$$

where (i = 1, ... I), and

$\sum_j^i \gamma_{ji}\sigma_{f,j}N_j\phi$ = 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}

$\lambda_i' N_i'$ = 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

$\sigma_{c,i}N_i\phi$ = is the destruction rate of N_i due to all forms of neutron absorption other than fission ((n,γ), (n,α), (n,p), (n,2n),(n,3n));

$\lambda_i N_i$ = radioactive decay rate of N_i

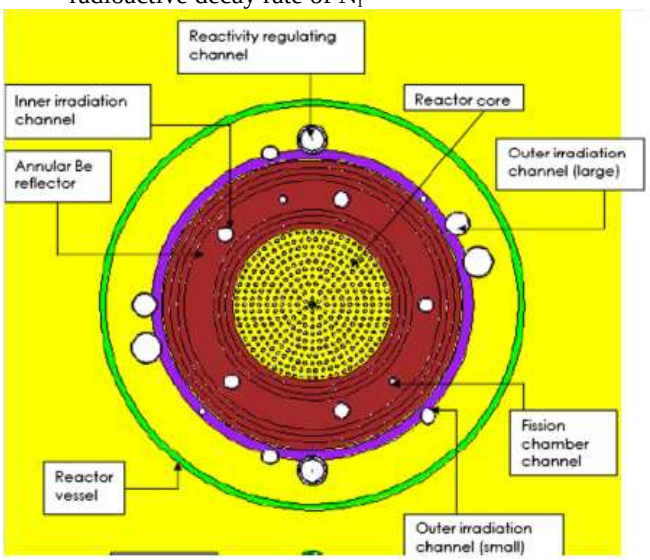

Fig 1: GHARR-1 core configuration showing fuel region.

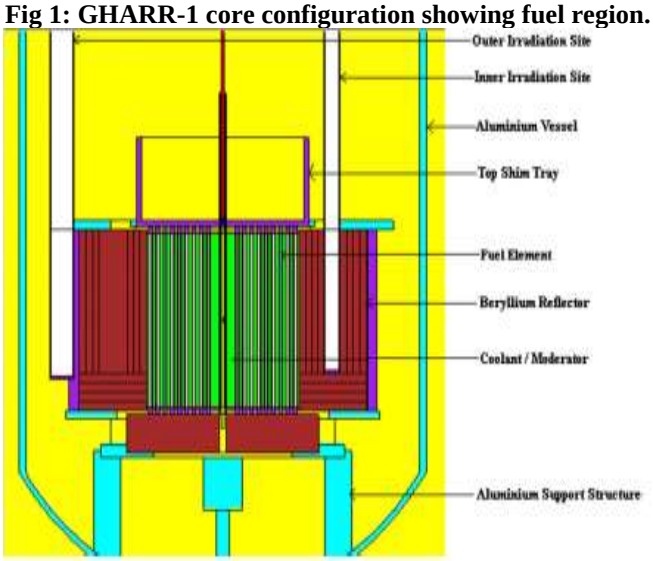

Fig 2: Vertical cross section of GHARR-1 reactor

Equation (2) is written for a homogeneous medium containing a space-energy-averaged neutron flux, φ, with flux-weighted average cross sections, σ_f and σ_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 ORIGEN2 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 Eqn. (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 ORIGEN2 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, Δtk, Δtk+1, ..., Δtn. 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.

Methodology

Pre-calculated libraries of 1-group cross section data are available for use in ORIGEN2 for several reactor systems. A standard library for an oxide-fueled LWR, the so-called bwrus library, represents the closest potential match to the basic neutronics characteristics of the core. It is preferable to replace cross section data for some or all of the nuclides in the library with data that are more appropriate for the particular system under analysis. For the present analysis, 1- group cross section data were calculated for the fuel rods using the WIMS-ANL code. Replacement cross section data were calculated for only a dozen also selected actinides and fission products. The calculated 1-group capture and fission cross sections at the mid-core life.

ORIGEN-2 code tracks radioactive nuclide inventories for 3 material groups, i.e., activation products, actinides and daughters, and fission products. The radioactive nuclide activities (in curies) are produced both during the irradiation period and cooling period afterward. A large amount of nuclide inventory data is available for analysis. For a bounding hypothetical accident radiological dose evaluation, the maximum value of the radionuclide activities over the whole core life history (including both irradiation period and cooling period) must be extracted if needed.

Results and Discussion

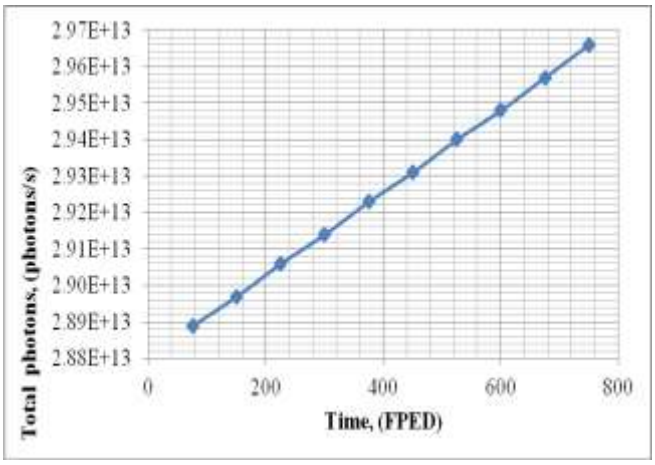

Fig 3: Photon density in activation products during irradiation period

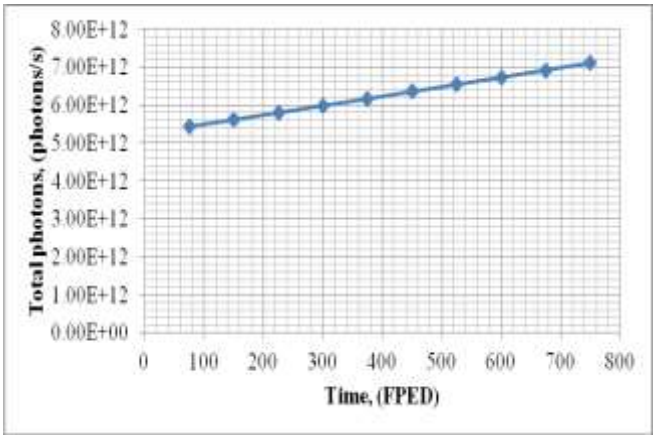

Fig 4: Photon density in actinides during irradiation period

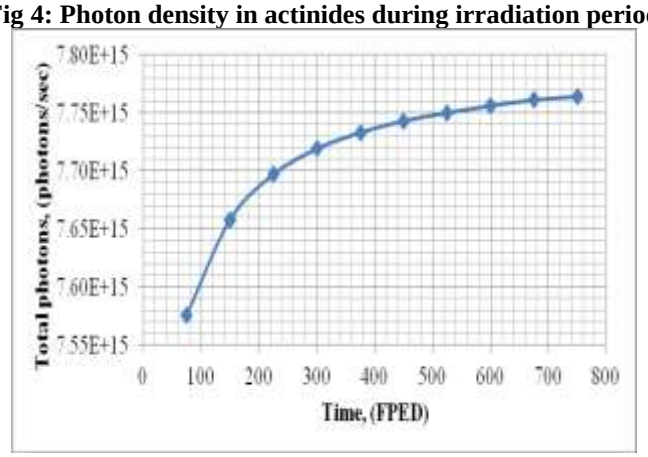

Fig Error! No text of specified style in document.**: Photon density in fission products during irradiation period**

Nuclide inventories as well as photon density generally increase with burn-up. The photon density in the fission products are the highest followed by the actinides and activation products in that order. Considering the beta and gamma particles as $\overline{E}_\beta = 0.4MeV$ and $\overline{E}_\gamma = 0.7MeV$ respectively where the average energy of the gammas is about twice that of the betas. This suggests that the energy release from beta emission release slightly exceeds that from the gamma emission from the fission products. Fission products are also high alpha emitters. As the fission products are born out of fission, they strive to achieve a more stable radionuclide. In an attempt to achieve stability, they give off photon emissions until they reach this stable state. For example, Caesium-137 is one such radionuclide. It has a half-life of 30 years, and decays by pure beta decay to a metastable state of barium-137 (Ba-137m). Barium-137m has a half-life of minutes and is responsible for all of the gamma ray emission. The ground state of barium-137 is stable.

For a bounding hypothetical accident radiological dose evaluation, the maximum values of the photon spectra over the whole core life history (including both irradiation period and cooling period) must be extracted. The bounding maximum photon spectral values are always found at the end of the irradiation period for activation products, actinides and fission products. The history of buildup of photon source during irradiation period is shown in figures 3, 4 and 5. For all the materials, the total photon source reaches its maximum at the end of irradiation period.

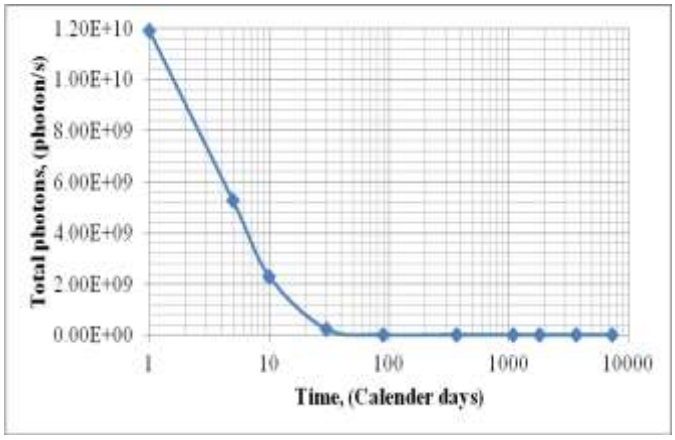

Fig 6: A log graph of GHARR-1 peak power pin photon source for actinides during cooling period for 8,000 calendar days

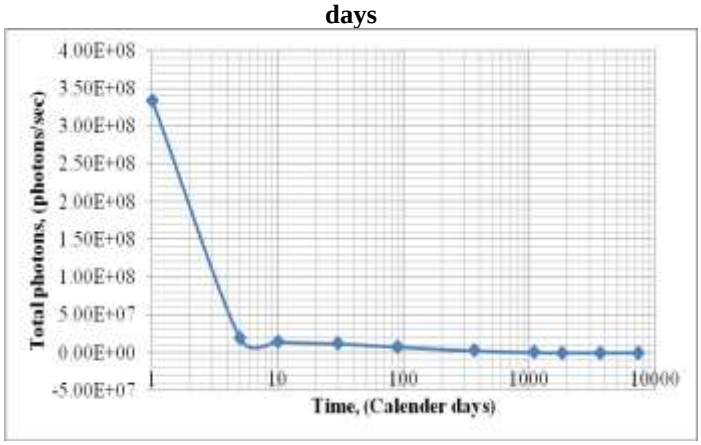

Fig 7: A log graph of GHARR-1 peak power pin photon source for activation products during cooling period for 8,000 calendar days

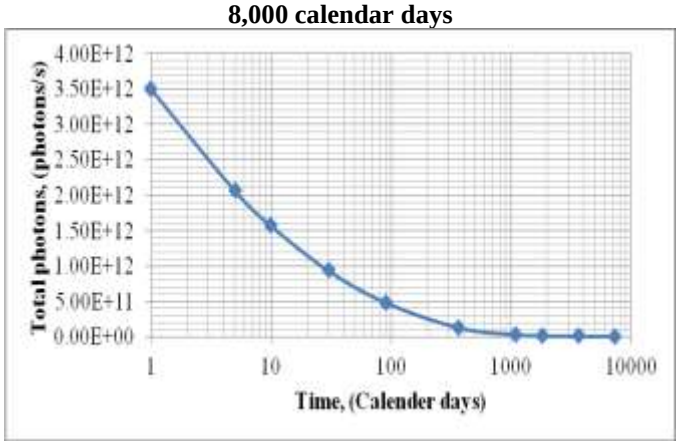

Fig 8: A log graph of GHARR-1 peak power pin photon source for fission products during cooling period for 8,000 calendar days.

The photon energies are higher for fission products followed by actinides and activation products respectively during spent fuel cooling. The total photon source decays quickly during the initial cooling period within ~ 30 days. After that the decay rate slows down progressively. Since almost 99% of the photon energy are produced by the fission products, the bounding radiological dose due to hypothetical accidents can be evaluated at the end of irradiation (discharge time) when the photon source is at its maximum in the entire core life history. This assumption is the most conservative approach to obtain the maximum radiological doses in any hypothetical accident scenario. Figures 6, 7 and 8 shows the GHARR-1 peak power pin photon source for actinides, activation products and fission

products during cooling period for 8,000 calendar days. The total photon values will also help decide on the type on shielding material that will be used in the fabrication of the spent fuel storage cask whilst taking into account cost and durability.

Conclusion

The photon source densities of fission products, actinides and activation products relevant for assessing radiological consequence as well as designing and fabricating spent nuclear fuel storage cask for GHARR-1 using local shielding material has been estimated for its 90.2% HEU fuel. ORIGEN2 code package, a zero-dimensional isotope decay and transmutation code was used to perform the core depletion analysis. The results showed that the bounding radiological dose due to hypothetical accidents can be evaluated at the end of irradiation (at discharge). It was also observed that the bounding maximum photon spectral values are always found at the end of the irradiation period for activation products, actinides and fission products. The results will be used in the analysis of the decay heat removal from the spent fuel storage cask, for which criticality safety and shielding analysis would be performed.

Reference

1. Bell, M.J.; ORIGEN B—The ORNL Isotope Generation and Depletion Code, ORNL-462 (CCC-217), Union Carbide Corporation (Nuclear Division), Oak Ridge National Laboratory, May,1973.
2. Crowther, R.L.; Lattice Burnup Calculation for Thermal Reactors Burnup Physics, *Proceedings of an IAEA Panel*, Viena-Austria, 1973.
3. Feltus, M.A.; Nuclear Core Design:NucE 403, Lecture Notes, Pennsylvania State University, U.S.A., 1995.
4. Gao, J.; "General description of the Ghana MNSR", MNSR Training, China Institute of Atomic Energy, 1993.
5. Kennedy , G.; St.Pierre, J.; Wang, K.; Zang, Y.;Preston, J,; Grant, C.;Vutchkov, M.; "Activation constant for SLOWPLOKE and MNS reactors calculated from the neutron spectrum and k0 and Q0 values", *Journal Radioanalytical Chemistry*, p. 167, 2000.
6. Nyarko, B.J.B.; "Neutron activation analysis", University of Ghana, School of Nuclear and Allied Sciences, Department of Nuclear Engineering, Lecture, pp. 1–3., 2009
7. Umar, I.; "The Potentials of MNSR in the Socioeconomic Development of Nigeria", *ICENSIAEA Small Research Reactor Workshop*, University of West Indies, Jamaica, 2003.