Investigative study of the radiation damage on fuel clad of miniature neutron source reactor using computational tools

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Abstract. Core conversion requires some evaluation of the reactor safety. Changes to the reactivity worth, shutdown margin, power density and material properties are crucial to the proper functioning of the reactor. The focus of this article is to study the neutron flux distribution in the reactor core and radiation damage on candidate clads. The Ghana Research Reactor-1 (GHARR-1) operates at maximum power of 30 kW in order to attain a flux of 1.0×10^{12} n·cm^{-2}·s for the high enriched uranium core. Using the GHARR-1 core geometry, considering 348 fuel pins, the multiplication factor (K_{eff}) is calculated at enrichments of 10%, 12.5%, 16%, 20%, 30% and 90.2%. The spectrum of neutron flux generated in the 26 group is also calculated at the specified enrichments. The ion/particle interactions with the targets (clad) were studied in the Stopping and Range of Ion in Matter code to establish the best clad material based on recorded defects and vacancies generated. From the calculations and simulations, the best choice from the candidate clads based on the assessment is SiC. The calculation of the fuel campaign length gives 7.5 years. The defects sustained by the prospective clad showed low susceptibility to swelling and other forms of deformation.

1. Introduction

The GHARR-1 is a commercial version of the miniature neutron source reactor which belongs to the class of tank-in-pool reactors [1]. The reactor is designed based on the Canadian SLOWPOKE reactor and is one of the smallest test reactors presently available in the world. They are currently found in operation in China, Ghana, Iran, Nigeria, Pakistan and Syria. The GHARR-1 is used for neutron activation analysis, radioisotope production, training of scientist and engineers in nuclear science [2].

1.1. High enriched uranium reactor core

The original description of the GHARR-1 core upon construction consists of ($U\text{-}Al_4$) fuel elements with enrichment of 90.2%. The fuel assembly is arranged in ten concentric rings around a central control rod guide. 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 U-235 content of about 1 kg. The small size of the core facilitates neutron leakage and escape in both axial and radial directions. To minimize such loses and hence conserve neutron economy, the core is heavily reflected on the side and underneath the fuel cage by a thick annulus reflector made of beryllium materials [3].

1.2. Core Conversion

Recently, the core was changed to a low enriched uranium fuel due to fears of proliferation by the direction of the International Atomic Energy Agency (IAEA). This has compelled the changes to the control rod and fuel element compositions, top and bottom grid plates [4]. This is shown in table 1 below.

Table 1. Core description of the Low Enriched Uranium (LEU) and High Enriched Uranium (HEU) composition.

No.	Descriptions	HEU	LEU
1.	Fuel composition	U-Al_4	UO_2
2.	Core Loading (g)	998.116	1358
3.	U-238 (wt.%)	8.3	87.05
4.	U-235 (wt.%)	90.2	12.5
5.	U-236 (wt.%)	1	0.2
6.	U-234 (wt.%)	0.5	0.25
7.	Cladding material	Al-303-1	Zirc-4
8.	Number of Fuel Rods	344	348
9.	Grid plates	LT-21	Zirc-4
10.	No. of Dummy Rods	6	2
11.	Materials for Dummy Rods	Al-303-1	Zirc-4

The reactor core is immersed in a large pool of water constructed from concrete. The core with the reactor vessel is lowered in this pool.

2. Methodology

The neutron flux and K_{eff} values are estimated using the new description of the core. Then the fuel campaign length was evaluated at various enrichment. The SRIM (Stopping and Range of Ion in Matter) & TRIM (Transport of Ion in Matter) code is then used to simulate the particle interactions with the 4 prospective clads.

2.1. Neutronics calculations

In calculating the flux, all cross-sections of the elements present in the reactor core were considered. The absorption and transport formulae are represented below as

$$\sigma_{\mathrm{a}} = \sigma_{\mathrm{f}} + \sigma_{\mathrm{c}}, \quad (1)$$

$$\sigma_{\mathrm{tr(i)}} = \sigma_{\mathrm{e}}(1-\mu_{\mathrm{e}}) + \sigma_{\mathrm{a}} + \sigma_{\mathrm{in}}. \quad (2)$$

Where σ_{e}, σ_{in}, σ_{a}, σ_{f}, σ_{c}, σ_{tr} represents elastic, inelastic, absorption, fission, capture and transport microscopic cross-sections respectively in barns (cm^{-1}). The multiplication factor (K_{eff}) is determined using formulae including the following below:

$$\overline{\nu_{\mathrm{f}} \cdot \Sigma_{\mathrm{f}}} = \sum_{i=1}^{I} \nu_{\mathrm{f}} \cdot \Sigma_{\mathrm{f}}^{i} \cdot \delta^{i}, \quad (3)$$

$$\overline{\Sigma_{\mathrm{a}}} = \sum_{i=1}^{I} \Sigma_{\mathrm{a}}^{i} \cdot \delta^{i}, \qquad \delta^{i} = \frac{\varphi_i}{\Sigma_{i=1}^{I} \cdot \varphi_i}, \quad (4)$$

$$\overline{D \cdot B^2} = \sum_{i=1}^{I} D^i \cdot B_i^2 \cdot \delta^{i}, \quad (5)$$

$$K_{\text{eff}} = \frac{\overline{\Sigma_{\text{f}} \cdot \nu_{\text{f}}}}{\overline{\Sigma_{\text{a}}} + \overline{D \cdot B^2}}. \tag{6}$$

Here B_i^2 defines the buckling for the i^{th} group of neutrons per surface area, D^i refers to diffusion coefficient for the i^{th} group of neutrons and ν_{f}^i also indicating the average number of neutrons per fission for the i^{th} group of neutrons, δ^i represents the partial neutron flux of the i^{th} group of neutrons.

It is essential to calculate the spectrum of neutrons. This is accomplished by solving the system of multi-group diffusion equations using the iterative algorithm [5, 6].

$$-D^i \cdot B_i^2 \cdot \varphi^i - \Sigma_{\text{a}}^i \cdot \varphi^i - \sum_{k=i+1}^{I} \Sigma_{\text{R}}^{i\to k} \cdot \varphi^i + \sum_{k=1}^{i-1} \Sigma_{\text{R}}^{k\to i} \cdot \varphi^k + \varepsilon^i \cdot \sum_{k=1}^{I} \nu_{\text{f}}^k \cdot \Sigma_{\text{f}}^k \cdot \varphi^k = 0. \tag{7}$$

For a 26 group: $I = 26$, k refers to the number of neutron groups; $\Sigma_{\text{R}}^{i\to k}, \Sigma_{\text{R}}^{k\to i}$ are the macroscopic cross-sections of the transition of neutrons from the i^{th} group to the k^{th} group and vice versa. Here φ^i, φ^k denotes neutron flux in the i^{th} and k^{th} groups respectively.

Resonance shielding is applied to σ_{f}, σ_{c} of the epithermal and thermal group [7, 8]:

$$\sigma_{\text{n}} = \sigma_{\text{n}}(293.6) \cdot \pi^{1/2} \cdot \left(\frac{293.6}{T_{\text{n.g.}}} \right)^{1/2} \cdot \frac{g_{\text{n}}}{2}, \tag{8}$$

where $\sigma_{\text{n}}(293.6)$ stands for the microscopic cross-section of the n-th process with $T = 293.6k$, g_{n} represents the adjustment factors of resonance self-shielding and $T_{\text{n.g.}}$ denotes neutron gas temperature as seen below [9]:

$$T_{\text{n.g}} = T_0 \cdot \left[1 + 1.4 \frac{\Sigma_{\text{a}}(T_0)}{\Sigma_{\text{s}} \cdot \varepsilon} \right]. \tag{9}$$

3. Results & Discussions

The results of the neutron flux spectrum (φ_i/φ), reactivity (ρ) and K_{eff} as carried-out at enrichments of 12.5%, 10%, 16%, 20%, 30% and 90.2% is shown below. The fuel campaign length was estimated considering the reflector saving using 50 effective day period. This is seen in figures 1-3.

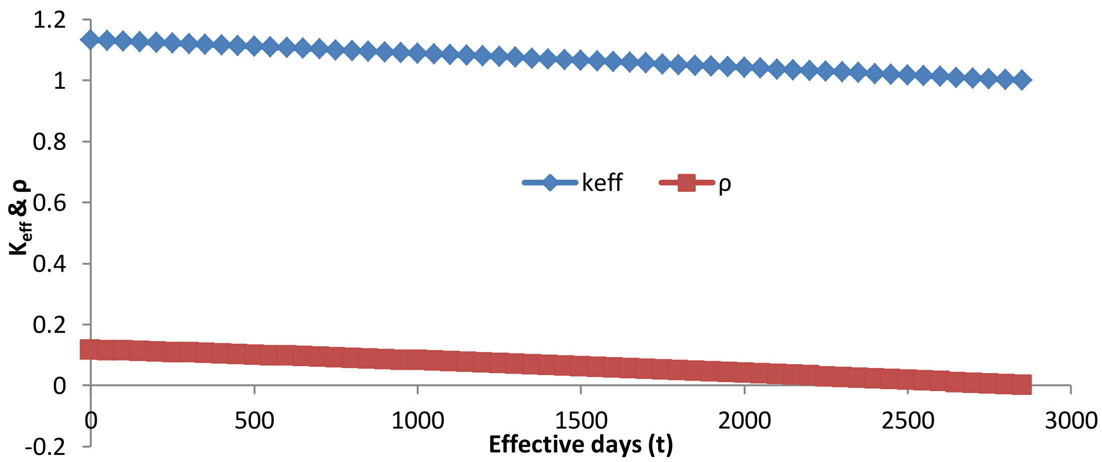

Figure 1. A graph of K_{eff} and ρ distribution at an enrichment of 12.5% LEU core.

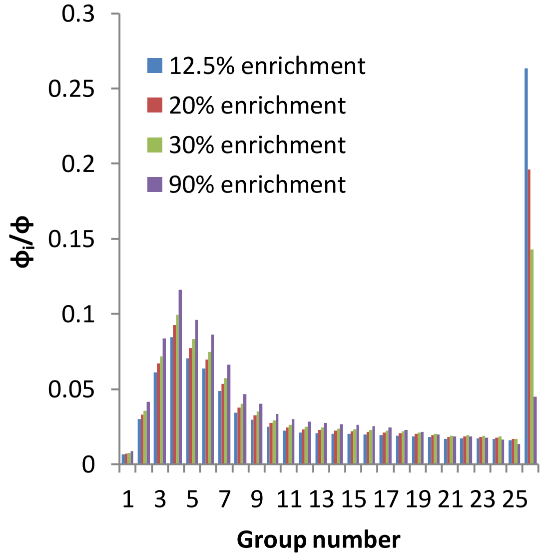

Figure 2. Neutron flux spectrum at enrichment of 12.5%, 20%, 30% and 90%.

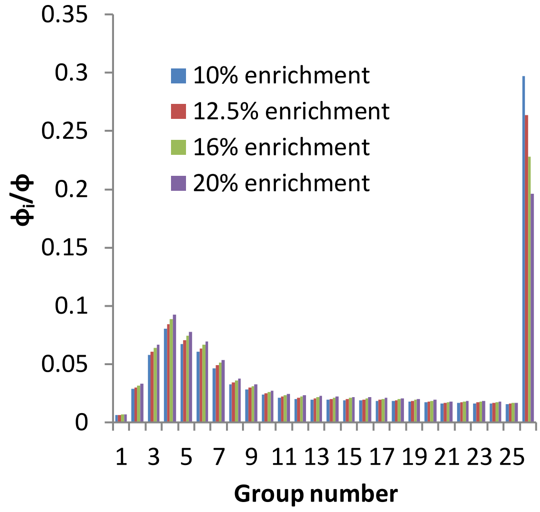

Figure 3. Neutron flux spectrum at enrichment of 10%, 12.5%, 16% and 20%.

3.1. Radiation damage assessments

The radiation damage assessment in the TRIM code is run using cobalt-60 for gamma interaction and then helium as the alpha source on the clad. Table 2 below describes the candidate materials used.

Table 2. Prospective clad materials with detailed compositions (wt.%).

Materials	Zr	Cr	Fe	Al	Sn	Ni	Mn	Mo	Si	C
Zircaloy-4	98.36	0.1	0.15		1.49					
FeCrAl		20	75	5						
310SS		25.22	55.55			19.51	1.9	0.122		
SiC									70.08	29.92

3.2. SRIM simulation (Gamma interactions)

The simulation of gamma interaction with each prospective clad is carried out. The results are presented below in figures 4-6.

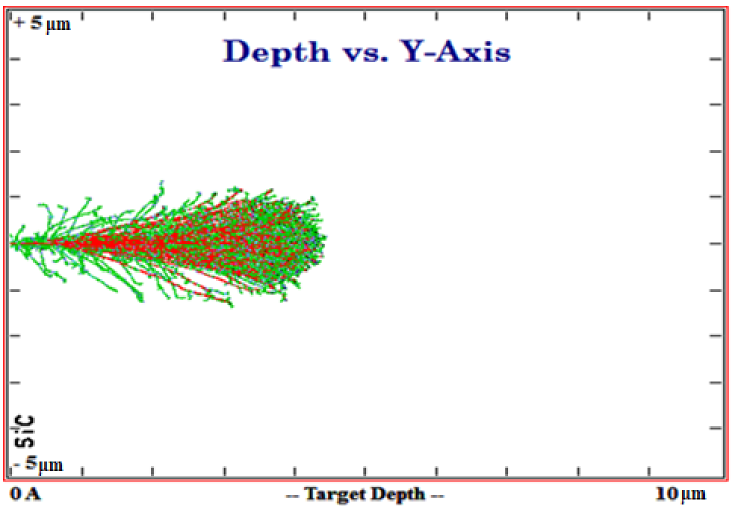

Figure 4. Depth in Y- Axis for λ-Interaction of SiC.

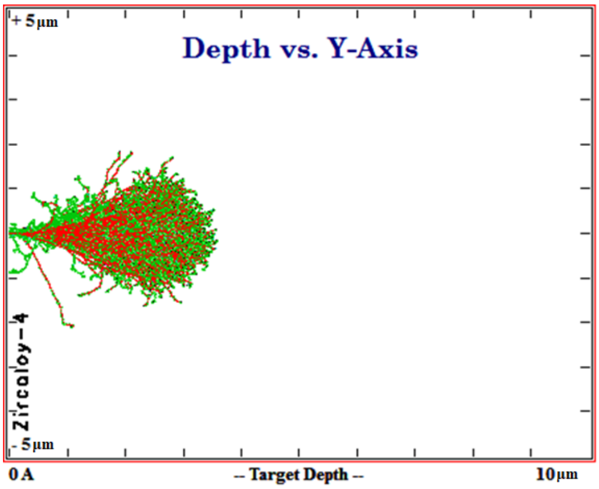

Figure 5. Depth in Y- Axis for λ-Interaction of Zircaloy-4.

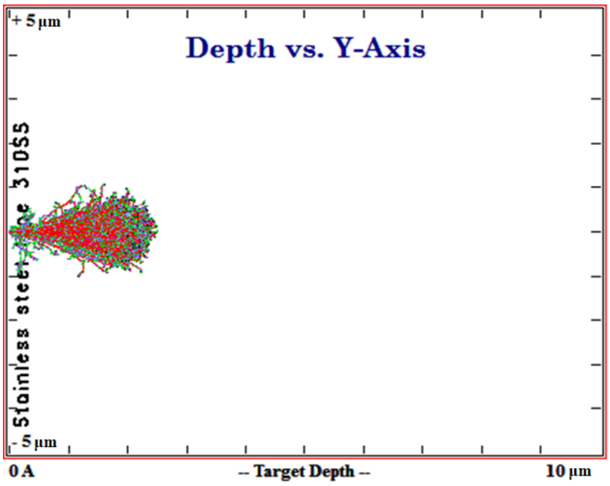

Figure 6. Depth in Y- Axis for λ-Interaction of Stainless steel 310SS.

3.3. SRIM simulation (Photon interactions)

A full cascade of the photon interactions with the clad after 99999 completed damage cascade for each candidate clad was carried-out. Table 3 shows gamma interactions with the clad.

Table 3. Vacancies / Ion & sputtering yield of prospective clads.

Clad Materials	Vacancies/Ion	Sputtering yield
Zircaloy-4	31289.4	0.293
SiC	17690	0.187
FeCrAl	30714	0.381
Stainless steel 310SS	34700.2	0.365

With regards to α- interactions, the damage to the candidate clads is minimal as the simulation records averagely 24.52 vacancies with just 4 backscattering ensuing in zirloy-4. The assessment of the displacement per atom (dpa) on the clad using the Norgett-Robinson Torrens model gave a dpa of about 0.00015 on zircaloy-4 clad. The displacement of lattice atoms will invariably affect the lattice structure and lead to the creation of vacancies either at the interstitial or lattice sites.

4. Conclusion

The best choice from the candidate clads based on the assessment is SiC, since it sustained the least vacancies created and sputter of atoms. It also registered the least recoil of 0.42. The material integrity within the simulation time for both interactions (α, λ) makes SiC the best. However, given the purpose of the reactor and the normal operation time including the economic prospects, zircaloy-4 clad will be proposed over SiC because SiC is refractory and maybe more appropriate for large reactors requiring refractory temperatures. From the neutronics calculations of the multiplication factor K_{eff} at the various enrichment, an LEU core at 12.5% is safe and best for the Miniature Neutron Source Reactor (MNSR). The calculation of the fuel campaign length gives 7.5 years, which is adequate enough for the reactor given the small core geometry.

Acknowledgments

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References

[1] *International Atomic Energy Agency Information circular* 1995 Project and supply agreement INCIRC/468
[2] Balogun G I 2003 *Ann. Nucl. Energy* **30** 81–92
[3] Odoi H C 2008 Reactor core conversion studies of Ghana Research Reactor - 1 and Proposal for Addition of Safety Rod *PhD thesis* (Ghana: University of Ghana)
[4] Odoi H C, Gbadago J K, Abrefah R G, Birikorang S A, Sogbadji R B M, Ampomah-Amoako E and Morman J 2014 *35th International Meeting on Reduced Enrichment for Research & Test Reactors* (Vienna: IAEA Vienna International Center) p S1–P3
[5] Golovatskiy A V, Nesterov V N and Shamanin I V 2010 *Izv. VUZ Fiz. (Izvestiya. Physics)* **53** 10–4 [in Russia]
[6] Abagyan L P, Bazazyants N O, Bondarenko I I and Nikolayev M N 1964 *Gruppovye konstanty dlja raschjota jadernyh reaktorov (Group Constants for Design of Nuclear Power Reactors)* (Moscow: Atomizdat) [in Russia]
[7] Bartolomey G G, Bat G A, Baybakov V D and Altukhov M S 1982 *Osnovy teorii i metody rascheta yadernykh energeticheskikh reaktorov (Fundamentals of the Theory and Methods for Design of Nuclear Power Reactor)* (Moscow: Energoizdat) [in Russia]
[8] Ganev I H 1992 *Fizika i raschet reaktora (Physics and the Design of Reactor)* (Moscow: Energoatomizdat) [in Russia]
[9] Beliavskii S V, Nesterov V N, Laas R A, Godovikh A V and Bulakh O I 2020 *Nucl. Eng. Des.* **360** 110524