Investigating Possibilities of fuelling NIRR-1 with Low Enriched Uranium Silicide-Aluminum Dispersion Fuels

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Abstract- In the last quarter of 2018, low enriched uranium dioxide fuel with zirconium alloy cladding was used successfully to convert the core of NIRR-1 from HEU to LEU fuel and the removed core returned to the country of origin. The objective of this study was to investigate the possibility of fuelling the same system with alternate LEU fuel for future replacement of the current fuel, without any unacceptable compromise in reactor performance. Having more than one fuel options available for the same reactor system will present Nigeria an opportunity of making good economic decisions at the end of the cycle of the current LEU fuel. The performance of low enriched uranium silicide aluminum dispersion fuels in the core of NIRR-1 has been investigated and the results were identical with that of similar studies conducted elsewhere for generic MNSR system. Some of the calculated reactor parameters using this alternate LEU fuel were closely identical with that of the old HEU core. The computer software selected for these studies were the SCALE code system and the VENTURE PC. While the SCALE code system was employed to generate a properly averaged multigroup cross section library for the investigated LEU core models for NIRR-1 system, the VENTURE PC was utilized to give criticality information, few group fluxes and power density distributions within the core of the modelled system.

Keywords- Reactor, Reactivity, Fuel, Enriched, Silicide

1 INTRODUCTION

Several Low Enriched Uranium Fuels are available for research reactors core conversion studies from High Enriched Uranium (HEU) to Low Enriched Uranium (LEU) Fuel (Matos and Lell, 2005, IAEA, 1992, Nawaz et al, 2012). The majority of research conducted on this type of studies for the Nigeria Research Reactor (NIRR-1) focused on the use of Low Enriched Uranium dioxide (UO₂) fuel with Zirconium alloy cladding. In the last quarter of 2018, this fuel material was used successfully to convert the core of the system from HEU to LEU fuel, with the removed HEU core returned to the country of origin, China (Chakrov and Hanlon, 2018 and WNN, 2019). This HEU fuel material was an alloy of Uranium and Aluminum (i.e., uranium Aluminide fuel). Note that pure uranium metal is not suitable for use directly as reactor fuel due to its unacceptable high rate of growth and swelling under irradiation (Finlay and Ripley, 2020).

The objective of this study was to investigate the possibility of fuelling the NIRR-1 system with alternate LEU fuel material for future replacement of the current fuel without any unacceptable compromise in the neutron flux, safety and economy of the system. Having more than one fuel options available for the same reactor system will present the country an opportunity for making good economic decisions at the end of the cycle of the current LEU fuel. Just like the HEU Aluminide fuel material, ordinary aluminum will be use as the cladding material for the uranium silicide fuels selected for this study. Aluminum is a very good alloying and cladding material due to its low neutron absorption cross section, very good thermal conductivity, workability and very good corrosion resistant in water (Finlay and Ripley, 2020). The results from several research conducted with the two available uranium silicide fuel material, indicates that U:Si-Al matrix is less suitable for high power reactor core conversion than U:Si-Al for long term and high burnup reliability due to gas bubble morphology and the exothermic U:Si-Al reaction (USNRC, 1988). But the higher density of U:Si-Al has made it attractive for use as fuels in low burnup reactors (Finlay and Ripley, 2020).

Note that the Nigeria research reactor is a tank in pool type Miniature Neutron Source Reactor (MNSR) with a pin type fuel geometry, designed and supplied by China National Nuclear Cooperation (CNNC) and used for scientific research, neutron activation analysis, education and training (Chakrov and Hanlon, 2018). Detailed descriptions of the geometry specifications and design parameters of this reactor have been published in several literature materials that are available online (Salawu, 2014, Agbo et al, 2016, Yusuf, 2018, Salawu and Balogun, 2017, and WNN, 2019).

The computer software selected for this analysis was SCALE code system and VENTURE PC; both codes are manufactured in the United State of America for reactor safety analysis and design. While the SCALE code system was employed in this work to generate a properly averaged multigroup cross section library for the investigated LEU core models for NIRR-1 system, the VENTURE PC was utilized to perform neutronic analysis for the system. VENTURE is a finite difference computer code that uses multigroup diffusion theory approximation to solve the appropriate particle balance equation over space and energy domains of interest (White, 1999). It is used mainly to give criticality information, few group neutron fluxes and power density distributions within the core region of a reactor system. The methods and models utilized in these studies were very similar to those developed in our previous studies for the old HEU system as well as the current LEU configurations (Salawu, 2014).

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Table 1. Physical parameters of the investigated LEU fuels material for NIRR-1 with 347 active fuel pins in the core

| Fuel Type | Outer Dia.: Active fuel pin (mm) | Uranium density (g/cc) | Clad / Thickness (mm) | Mass (g) per pin | Mass (g) of u235 in the core | Mass (g) of u238 in the core |
|-----------|----------------------------------|------------------------|----------------------|-----------------|-----------------------------|-----------------------------|
| HEU Fuel 90.0% |                                   |                        |                      |                 |                             |                             |
| UAl=Al    | 4.3                              | 0.96                   | Al/0.6               | 2.89            | 0.32                        | 1002.83                     | 111.04                      |
| LEU (19.75%) |                                 |                        |                      |                 |                             |                             |
| USi=Al    | 4.74                             | 4.30                   | Al/0.38              | 3.45            | 14.01                       | 1197.15                     | 4861.47                     |
| USi-Al    | 4.3                              | 5.37                   | Al/0.6               | 3.54            | 14.39                       | 1228.38                     | 4993.33                     |

2 Methodology

The two uranium silicide aluminum dispersion fuels selected for these studies are the 19.75% enriched U:Si-Al and U:Si-Al with densities of 5.37 g of u/cc and 4.30 g of u/cc respectively. This research work was carried out in such a manner that material compositions of components in the core of the system were maintained as in the old HEU core except the fuel region that was replaced with the above uranium silicide aluminum dispersion fuels. There are no changes in the number of fuel pins, the dummy pins and the tie rods in the core of the system using any of the investigated LEU core models. In addition, the geometry and dimensions of the fuel pins in the old system were retained except for the U:Si-Al fuel pins in which the radius of the active fuel region was increased by 0.022 cm with a corresponding decrease in the cladding thickness (table 1.0). This modification becomes necessary because uranium density of U:Si-Al is a little bit lower than the expected minimum value for MNSR reactor core conversion (Matos and Lell, 2005).

Note that the outer diameter of the fuel pin (active fuel region + cladding) as well as the water volume fraction was maintained in these cores as it was in the old HEU core of NIRR-1 system. The region atom density and the homogenized atom density needed for these analyses were calculated using the methods provided in our previous studies (Salawu, 2014) and the 2D VENTURE model developed for the neutronic analysis of NIRR-1 system as provided in this reference material was adopted for these analyses without any major modifications except the fuel regions of the model that was replaced with 19.75% uranium silicide aluminum dispersion fuel. The material atom densities (region/homogenized) generated using these fuel materials as well as other components in the system were used within the SCALE code system to generate, for NIRR-1, a properly averaged microscopic broad group ANISN formatted cross section library in binary format. This library was later converted into a nuclide ordered cross section (ISOTXS) format and then into a group ordered cross section (GRUPXS) format within the VENTURE software before it was suitable for use in the VENTURE neutronic calculations.

With the appropriate material by mesh description and a set of two/four groups cross section libraries generated for the system, the VENTURE code computed effective neutron multiplication factor (k-effective) at different position of the control rod, the power distribution, and the few group flux profiles within the core of the investigated LEU core models for NIRR-1 system. With the control rod at the centre of the system, the k-effective value was recalculated at different thickness of the top Beryllium Shims to produce the Beryllium shim worth for NIRR-1 system. The x & y directed flux profile as well as the R & Z directed power density profile were produced by extracting the regular total point wise flux (RTFLUX) data and Point wise power density (PWDINT) data from the VENTURE “work” directory. These two data files (RTFLUX & PWDINT) were translated into a binary format for direct use within a MATLAB program to produce the 1D flux and power density profiles as well as their corresponding 2D & 3D surface plots for graphical visualization of the power and neutron flux distribution within the core region of the system.

The results of some of the calculations performed by VENTURE codes were processed by varieties of MATLAB programs for easy presentation and interpretation. The values of k-effective at different control rod withdrawal positions were used to plot a graph of the reactivity worth of the control rod for the system. The same is true for the reactivity worth of the top beryllium shim additions. Note that some of the above MATLAB codes were originally written by Professor John R White of UMass Lowell-USA for University of Massachusetts Lowell Research Reactor (UMLRR) and are modified in this work for post processing of the neutronic data generated for the Nigeria Research Reactor (NIRR-1).

3 Results and Discussion

Figure 1 shows a graph of reactivity against control rod withdrawal length for the investigated LEU cores for NIRR-1 system and that of the old HEU core. It is clearly shown in the figure that the control rod worth curve for these two uranium silicide aluminum dispersion fuels with aluminum cladding are very identical with that of the old HEU system. The calculated total control rod worth of about 8.0mk for the investigated uranium silicide fuels looks very good as compared to what was reported in the literature for the old HEU system and the present LEU core in the system (FSAR, 2004).

Figure 2 is a graph of reactivity against the thickness of the top Beryllium Shims (i.e., the shim worth curve) for NIRR-1 system with the investigated LEU silicide aluminum dispersion fuelled core. This shim worth curve looks very similar with that of the old HEU core as compared with the present Low enriched uranium dioxide fuel in the NIRR-1 core (Figure 2).
Just like we observed in our previous studies for both the old HEU and the current LEU fuel, there are three locations in the axial direction where power peaks in the NIRR-1 core when fuelled with any of the investigated LEU silicide fuels. Figures 3 and 4 show 1D power density profiles in NIRR-1 core when fuelled with U3Si2-Al with control rod positioned near the centre of the core.

In the z direction (i.e., from top to bottom), the maximum power peak occurred at a point near the location of the bottom beryllium (figure 3.0). But in the radial direction, the first power peak of about 4.6W/cc was observed at the centre of the system, near the first concentric circle while the second power peak of about 4.0 W/cc occur at a position close to the radial beryllium (figure 4.0) as expected because of the presence of a reflector in that location. The corresponding 2D and 3D surface plots of the power density distribution within the core region are shown in Figures 5 and 6 respectively. Similar behaviours were observed when low enriched U3Si2-Al fuel was used. These surface plots were very helpful in visualizing power distribution within the core of a reactor system. Identifying location with peak power density constitute useful information in reactor fuel management; as the hottest fuel pins can be identified and moved to the locations of low power density. This can enhance the core life time and the efficiency of the fuel pins (Salawu, 2014).

The results of the calculated peak power densities and the maximum neutron densities in the core are provided in Table 2. The values shown in this table are very identical with what we obtained in our previous studies for the old HEU core (i.e., 4.3 W/cc) and that of the 12.5% enriched UO2 core (i.e., 4.2 Watt/cc) in the system. The maximum power density using U3Si2-Al as fuel is very identical with that of U3Si2-Al core (Table 2).

The calculated excess reactivity for the system using these fuel materials is about 0.8%Δk/k (Table 2). These values agreed very well with the value obtained from a similar analysis for generic MNSR, performed by Matos and Lell in 2005. But note that we could not modelled the reactivity regulators in the core of NIRR-1 in this VENTURE model due to lack of sufficient information about the geometry dimension and material composition of these particular NIRR-1 components. The computed maximum neutron flux was on the other of ten to the power of twelve in each of the investigated cores (Table 2 and Figures 7 & 8). The maximum thermal neutron fluxes in the inner and outer irradiation channel sites were calculated to be 1.4e+12 and 0.35e+12 (Figures 8 and 9) respectively. These values are very good as compared to what was reported for the old HEU core as well as the current LEU fuel in the core of the system (FSAR, 2005 and Salawu, 2014). Figures 10 and 11 are the 2D and 3D surface plots providing pictorial representation of the thermal neutron flux distribution within the core region with uranium silicide aluminum dispersion fuels in the core. These figures are very useful in identifying location with high neutron flux for the placement of a sample for neutron irradiation in a research reactor system. Similar 2D and 3D surface plots for the fast neutron flux are provided in Figures 12 and 13 respectively. Note that the shapes of the 2D and 3D surface plots for the epithermal and resonance neutron fluxes distribution are very much similar with the shape of the fast neutron flux surface plots. The peak value of these fluxes occurs at the centre of the system which decreases continuously as you moved away from the centre of the reactor core.

Table 2: Calculated parameters for NIRR-1 system

| Fuel type     | Peak Power Density (Watts/cc) | Peak neutron Density (neutrons/cc) | Excess Reactivity (%Δk/k) |
|---------------|-------------------------------|----------------------------------|--------------------------|
| 19.75% U3Si2-Al | 4.62632                       | 6.59494E+12                     | 0.817                    |
| 19.75% U3Si-Al  | 4.61230                       | 6.62812E+12                     | 0.819                    |

Fig. 1: Reactivity versus control rod withdrawal length from the bottom of the active fuel

Fig. 2: NIRR-1 top Beryllium Shims Reactivity Worth
Fig. 3: NIRR-1 axially directed power density profile through peak with 19.75% $\text{U}_3\text{Si}_2$-$\text{Al}$ as fuel.

Fig. 4: NIRR-1 radially directed power density profile through peak with 19.75% $\text{U}_3\text{Si}_2$-$\text{Al}$ as fuel.

Fig. 5: 2D surface plot of the power density distribution within the core (control rod at the centre)

Fig. 6: 3D surface plot showing various location with maximum power within the core (control rod at the centre).

Fig. 7: The radially directed neutron flux distribution (from the centre towards the radial beryllium) in the NIRR-1 core for each energy group with control rod located at the centre of the core.

Fig. 8: Axially (top to bottom) directed neutron flux distribution at the location of inner irradiation channels.
CONCLUSION

The performances of low enriched uranium silicide aluminum dispersion fuels in the Nigeria Research Reactor core have been investigated and the results agreed reasonably well with that of similar studies conducted elsewhere for generic MNSR system. The calculated reactor parameters using this alternate LEU fuel were very similar with that of the old HEU core. The U$_3$Si$_2$Al fuel is considered more suitable for reactor core conversion but the higher density of U$_3$Si$_2$Al has made it more desirable for low burnup reactor system. The result from this study shows that uranium silicide aluminum dispersion fuels have the potentials for use as an alternate LEU fuel for NIRR-1 system. Therefore, these fuel materials can further be investigated with a more detail three-dimensional reactor physics software before accepting it as an alternate LEU fuel for NIRR-1 without any unacceptable penalty in the system performance.
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