Nigerian Research Reactor-1 (NIRR-1) Core Life-time Analysis using Winfrith Improved Multigroup Scheme-Argonne National Lab (WIMS-ANL) and REactor BUrnup System-Argonne National Lab (REBUS-ANL) Computer Code

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Abstract

The core of the Nigeria Research Reactor-1 has been converted from Highly Enriched Uranium (HEU) fuel to Low Enriched Uranium (LEU); nevertheless, a core lifetime analysis of the converted HEU and current LEU fuel is required. The core lifetime analysis of the NIRR-1 converted HEU and current LEU cores was carried out using the Winfrith Improved Multigroup Scheme-Argonne National Laboratory (WIMS-ANL) and Reactor BUrnup System-Argonne National Laboratory computer codes in this work (REBUS-ANL). One group cross section was generated using the WIMS-ANL. For core lifetime analysis, these group cross sections were employed in the REBUS-ANL programme. A simple reactivity rundown was performed to estimate the core lifetime for HEU and LEU cores at two power levels using the REBUS-ANL code (full and half power). The lifetime of the LEU core was predicted to be approximately 11.91 percent longer than that of the HEU core at full power and 11.81 percent at half power based on the results obtained. This is in good agreement with the value 11 percent reported in the literature, indicating that the selected LEU core can be operated for a longer period of time than the HEU core.

1. Introduction

To prevent HEU from being used to manufacture nuclear weapons, a concerted effort was launched in 1978 to convert research reactors using HEU fuel to LEU. The conversion is intended to make test and research reactor fuel as unappealing as possible to groups/non-state actors interested in exploiting such highly enriched cores for non-peaceful purposes. The program's objective is to develop the technology required to convert research reactors from HEU to low enriched uranium (LEU) fuel. Since then, a worldwide effort has been made to restrict, and finally eradicate, the civil use of HEU (Travelli, 1993).Through use of High Enriched Uranium (HEU) to fuel research reactors instantly creates a slew of inescapable and obvious proliferation risks associated with material diversion and/or theft. Thus according sources, the US Department of Energy (DOE) established the Reduced Enrichment for Research and Test Reactors (RERTR) program in 1978 in response to rising concerns about proliferation and the ease with which state and nonstate actors may get nuclear weapons (Ibikunle et al., 2018, Feiveson et al., 2014 and Simon et al., 2021). A miniature neutron source reactor (MNSR) is one of numerous research reactors worldwide that use HEU as fuel. The Nigerian Research Reactor-1 (NIRR-1) is classified as a HEU because it is 90% enriched in fissile U-235 (Azande et al., 2010). It was created primarily for neutron activation analysis (NAA) and minimal radioisotope generation (Jonah et al., 2013). The NIRR-1 core was originally designed to run on HEU-90.2% fuel, but following a recent successful conversion, it now runs on LEU-13% fuel, light water as a moderator and coolant, and metallic beryllium as a reflector. The Beryllium shims, on the other hand, function as neutron reflectors for the reactor's longer operation. At the start of each operational cycle, the maximum allowable excess reactivity is 4 mk (Fig. 3.1, 3.2 and 3.3, Table 3.1) (Yahaya et al., 2017 and Simon et al., 2021). Previously, a feasibility study was conducted to estimate the enrichment necessary to convert a commercial Miniature Neutron Source Reactor (MNSR) (NIRR-1 in particular) from HEU (90.2%) to LEU (20%) fuel. Two LEU cores with uranium oxide fuel pins of varying diameters were investigated. According to the study, the findings obtained are equivalent to HEU core and indicate that it would be viable to employ any of the LEU choices for the conversion of NIRR-1 in particular from HEU to LEU (Jonah et al., 2009).

Omar et al., (2010) recently conducted a study of 235U burn-up for the HEU-fueled Syrian MNSR. They report strong agreement between WIMSD4-CITATION-based model predictions and the experimentally obtained measurements, with an optimum relative difference of no more than 5%.

Albarhoum (2010) used the WIMSD4 and CITATION codes to conduct an optimization study of uranium loading in MNSRs. Based on clad thickness and fuel meat radius adjustments, at least 31 fuel pins are saved while keeping safety criteria. The comparable core operation longevity, on the other hand, is predicted to drop by around 37%. The use of U–9Mo as an MNSR fuel has also been researched. According to reports, the preliminary excess reactivity of the 19.75% enriched monolithic U–9Mo core of MNSR is 3.595 mk, which is roughly 10% less than the comparable figure for the HEU core. As a result, the predicted operating period of the U–9Mo-fueled MNSR is reduced.

Khattab (2005) calculated the fuel efficiency and radionuclide inventory in the Syrian miniature neutron source reactor at various operational power levels using the WIMSD-4 algorithm (10, 20 and 30 kW). The WIMSD-4 code was used to determine

the changes in the fuel one group cross section and infinite multiplication factor for the MNSR along with the burn-up time. The Findings from this study indicate that after 10 years of reactor operation, the quantities of uranium-235 consumed and plutonium-239 created in the MNSR core are 7.708 g and 0.049 g, respectively. The Author also indicates that the proportion of uranium burn-up was 0.769%.

Earlier, Yahaya et al., (2017) used the programs WIMS and CITATION to determine the burn-up for the Nigerian MNSR HEU fuel and the core life time expectation after 10 years of operational period. The burn-up findings confirmed that the excess reactivity of NIRR-1 maintains a linear declining pattern with 216 Effective Full Power Days (EFPD) operations. The reactivity worth of top beryllium shim data plates was determined to be 19.072 mk. The results of the depletion study for the NIRR-1 core reveal that (7.9947 0.0008) g of U-235 was consumed during the course of 12 years of operation. The yield of Pu-239 build-up was determined to be (0.0347 0.0043) g.

Abrefah et al., (2013) evaluated the fuel burn up of the Ghana MNSR utilizing the ORIGEN2 and REBUS3 algorithms. They performed a fuel depletion study and found that the findings were in excellent accordance with publications for the HEU core for comparable computations using other approaches. For reference, they estimated the burnup and accumulation of U-235 and P-239 using both the REBUS3 and ORIGEN2 code modules. The ORIGEN2 code had a U-235 burnup percentage composition of 2.90%, whereas the REBUS3 code had a 2.86%. The magnitude of Pu-239 in the core is greatest at the end of the irradiation period, around 1.45E-01 grammes, so plutonium production in GHARR-1 is low. The accumulation of fissile Pu-239 and Pu-241 is insufficient to compensate for the reduction in reactivity caused by U-235 depletion, and the concentration of Pu-239 in spent fuel is insufficient to raise concerns about nuclear bomb production.

Recently, Khattab and Dawahra (2011) used the GETERA algorithm to generate the fuel burnup and radioactive inventory in the Syrian miniature neutron source reactor (MNSR) after ten years of operation. The GETERA code was used to determine the modifications in the fuel one group cross sections and infinite multiplication factor for the MNSR versus the burn-up time. They discovered that after 10 years of reactor operation, the quantities of uranium-235 consumed and plutonium-239 yielded in the MNSR core were 7.481 g and 0.0458 g, respectively. They also reveal a 0.747% uranium burn-up percentage. Winfrith Improved Multigroup Scheme-Argonne National Laboratory (WIMS-ANL) and REactor BUrnup System-Argonne National Laboratory (REBUS-ANL) computer codes were used in this investigation. The WIMS-ANL was used to generate the one group cross section data, and the REBUS-ANL was used to analyze core operational life.

2. High Enriched Uranium (Heu) And Low Enriched Uranium (Leu)

There are two fundamental uranium isotopes that exist in significant amounts in nature. These are U-238 and U-235, which have natural isotopic fractions of 99.29% and 0.71%, respectively. The relative concentration of these two isotopes, i.e. the enrichment or U-235 weight fraction of a particular material stock, may be altered using a number of isotope separation techniques that use physical processes to separate the species. In reality, the enrichment level defines the primary properties of any 40 uranium composition, both for reactor usage and for weapon application (Glaser, 2005). Throughout the whole energy range from thermal to fast neutron energies, U-235 has a high chance of fission after neutron absorption. U-238, on the other hand, is only fissionable beyond threshold energy of roughly 1MeV. Below this threshold energy, neutron capture dominates this isotope's overall absorption cross section (Glaser, 2005 and Azande et al., 2010). Uranium enriched to at least 90% U-235 has been used to build nuclear bombs, and uranium with this level of enrichment is known as highly enriched uranium (HEU). Because it is often less well-guarded and is commonly placed in cities or on university campuses, HEU fuel intended for research reactors is considered to be especially susceptible to diversion for use in nuclear weapons. HEU manufacture began during World War II as part of the United States' Manhattan project. Although HEU was accessible at a preliminary phase and was used in the nuclear device that devastated Hiroshima on August 6, 1945, production capacity was severely limited at the time (Glaser 2005). The massive US enrichment facilities under construction were finished until after World War II, and these facilities were based on the gaseous diffusion technique, as opposed to the calutrons employed during the war. The yearly production rate of HEU and Plutonium in the United States peaked in the early 1960s, at 80 metric tons and 60 metric tons, respectively (Albright et al., 1997). An inventory of around 200–300 metric tons can be anticipated to be absorbed in 41 deployed nuclear bombs globally (Glaser, 2004). Albright et al., (1997) concurred that the majority of surplus weapongrade HEU

in the United States is being held in reserve for use in naval reactors. This stockpile is large enough to fuel the whole nuclear-powered fleet in the United States for several decades (Glaser *et al.*, 2002) and should be well over 100 metric tons. HEU fuel is still used in around 150 nuclear-powered submarines and military surface boats (Ma *et al.*, 2001). In addition, seven Russian nuclear icebreakers and cargo ships operated by the Murmansk shipping firm utilize around 500 kg of HEU yearly. The overall amount of HEU still present in the civilian nuclear fuel cycle, including fresh and irradiated but unshipped back material, has been estimated to be around 50 metric tons (Albright and Kramer, 2004). This material is mostly kept as fuel components at reactor sites or interim storage facilities, either wet or dry. The most current assessment on spent fuel from research reactors, conducted under the auspices of the IAEA and based on 210 out of about 550 reactors, identified 22,686 HEU and 40,184 LEU fuel components stored globally (Ritchie, 1998), with an additional 32,932 assemblies contained in reactor cores. The term Low Enriched Uranium (LEU) was coined by the United States Atomic Energy Commission in or around 1995. The International Atomic Energy Agency eventually followed the same standard, defining low enriched uranium as "enriched uranium having less than 20% of the isotope U- 235." (Glaser, 2006). LEU is classified as an indirect use 42 material by the IAEA, which is a radioactive substance that cannot be used for "the construction of nuclear explosive devices without transmutation or additional enrichment" (Glaser, 2005). From a technical standpoint, the decision of the LEU limit is somewhat arbitrary. Similarly, the sufficiency of the conversion objective for research reactors slightly below that level, generally 19.75%, is far from evident. Two criteria are critical in determining the optimal enrichment level for research reactor fuel and from a non-proliferation standpoint: the usage of fresh or irradiated fuel and the contemporaneous and unavoidable creation of plutonium in the fuel during irradiation (Skjoldebrand, 1980). Travelli (1978) recognized that the proliferation resistance of nuclear fuels used in research and test reactors can be significantly improved by reducing their uranium enrichment to less than 20%, but significantly higher than natural, to avoid excessive plutonium production. The International Nuclear Fuel Cycle Evaluation (INFCE) documents use similar arguments (IAEA, 1980).

3. Description And Design Of The Nirr-1

Operated and maintained by the Center for Energy Research and Training (CERT) of Ahmadu Bello University in Zaria, Nigerian Research Reactor1 (NIRR1) is a commercial version of the Miniature Neutron Source Reactor (MNSR), a tank-type reactor. Designed and manufactured by the China Atomic Energy Research Institute (CIAE) (Jonah et al., 2009 and Ibrahim et al., 2012). It has a fuel burn-up of about 1% and is designed to have a core life of 10 years until the next cycle operation when operating at maximum flow rate for 2.5 hours a day, 5 days a week (Table 3.1)(Boafo et al., 2012). It was created primarily for neutron activation analysis (NAA) and minimal radioisotope generation (Jonah *et al.*, 2005 and 2006). Due to the aging lifespan of the core, controlled beryllium shims have been placed to the top aluminum tray to compensate for the loss of reactivity caused by fuel burn up and the accumulation of fission products. (Boafo et al., 2012). The Nigeria Miniature Neutron Source Reactor (NIRR-1) is a tank-in-pool type reactor (Fig. 3.1, 3.2, and 3.3) that used to run on 90.2% HEU fuel but currently runs on 13% UO_2-LEU fuel, light water as a moderator and coolant, and metallic beryllium as a reflector. It is primarily intended to function as a neutron source. The core consists of a cylindrical fuel assembly with roughly 348 fuel components (SAR, 2012). The core structure of a quintessential MNSR consists of 347 fuel rods, 4 tie rods, and 3 dummy components spread evenly on a total of ten concentric rings, each with a number of fuel rods tend to range from 6 to 62, and a thick beryllium reflector (10 cm) encompassing the core radially (Fig. 3.1, 3.2, and 3.3) (Yang, 1992). This same NIRR-1 has a single central control rod made of cadmium and stainless steel. The rod serves as a shim, as well as a safety and regulatory control module (Qazi et al., 1996). As a result, the reactor is safe and small, allowing it to be installed in highly populated areas without endangering lives (Yang 1992). Natural convection removes heat created in the core. The reactor has ten irradiation locations. Five of them are located within the annular beryllium reflector, while the remaining five are located outside of the annular beryllium reflector. The inner and outer locations have maximal thermal neutron fluxes of 1.0×10^{12} and $5.0 \times 10^{11}\ n/cm^2 \bullet s$, respectively. The reactor achieved first criticality on February 3, 2004 with an initial core excess reactivity of 3.77mk and has been operating safely for Neutron Activation Analysis (NAA) and minimal radioisotope generation since then. NIRR-1's initial reactor core used HEU fuel, which was placed in a cylindrical fuel enclosure. Thus every fuel pin was made of a 90.2 % enriched uranium-235 alloy exruded in the shape of UAl4 as the fuel meat, and an aluminum alloy as the clad component (SAR, 2012). The fuel element is typically 248mm long, with an active length of 230mm. The diameter of the fuel meat section is 4.3mm, and each HEU fuel rod contains

2.88 g of U-235. The cadmium absorbent control rod is 266 mm in length and 3.9 mm in width, and it is encased in a 0.5 mm thick stainless steel clad (Qazi et al., 1996). The control rod arrangement measures 450 mm in entire length. Three of the five outside irradiation channels are 22 mm in diameter, while the remaining two are 34 mm in diameter. An annular beryllium reflector is mounted on a lower beryllium reflector plate to encircle the fuel cage. The annular beryllium and lower reflector are arranged to form a lower aperture, permitting convective movement of water through the core and the removal of emissions released in the core by natural convection (Agbo et al., 2015). The containment vessel is a tubular alloy container with a diameter of 0.6 m and a height of 5.6 m that is suspended in a reinforced concrete pool lined with stainless steel (SAR, 2012). The intrinsic safety of all MNSRs is related to the restricted core excess reactivity in the range 3.5-4.0 mk (Ibikunle et al., 2018). The current NIRR-1 core is powered by around 1.4kg LEU, and the clean cold core excess reactivity is 3.94 mk, as determined during on-site zero-power and criticality studies (Table 3.1) (CERT, 2019). The core of NIRR-1 is positioned 4.7 meters below water level, near the bottom of the light water containment vessel. The vessel contains $1.5m^3$ of water, which is then immersed in a water-filled pool 30 3 as illustrated in Fig. 3.1, 3.2, and 3.3. (SAR, 2012).

Table 3.1
Summary of Key Nominal Design Parameters of Converted HEU and Current LEU Core (SAR, 2012)

Design data	HEU	LEU
Fuel Type	$U-Alalloy$	UO_2
Fuel Meat Size Diameter (mm) Height(mm)	$4.3230.0$	$4.3230.0$
Cladding	Al	Zircaloy-4
Cladding Thickness(mm)	0.6	0.6
Fuel Enrichment (nominal)	92.0%	13%
Gap Size (mm)	not available	0.05
Mass of U^{235} Per Pin (g)	2.88	3.89
Number of Pins	347	348
Number of Dummy Rods	3	2
Fresh Core Excess Reactivity(mk)	3.77	4.0
Shutdown Margin(mk)	3	3.7
Control rod worth	7.0	7.7
Average Coolant Void Coefficient	-0.36060 ± 0.03255	-0.36580 ± 0.03259
Coolant Temperature Coefficient	-0.00657 ± 0.00011	-0.00352 ± 0.00012
Fuel Temperature Coefficient	-0.00029 ± 0.00010	-0.00092
Effective Delayed Neutron Fraction(s)	8.08×10^{-3}	$8.34 \times 10^{-3} \pm 0.06$
Neutron Lifetime(μs)	8.12×10^{-5}	$5.09 \times 10^{-5} \pm 0.7\%$

4. Winfrith Improved Multigroup Scheme-argonne National Lab (Wims-anl) Code

The WIMS code, created at Argonne National Laboratory, has been widely used across the world for power and research reactor lattice physics analysis (ANL). There are several WIMS versions in use today. The RERTR program chose the D4 version that was produced in 1980. (Deen et al., 1995). The WIMS code, created at Argonne National Laboratory, has been widely used across the world for power and research reactor lattice physics analysis (ANL). There are several WIMS versions in use today. The code computes the cell-averaged diffusion coefficients (D), absorption cross section $\left(\Sigma_a\right)$, fission cross sections $(\nu\Sigma_f)$, infinite multiplication factor $\left(k_{inf}\right)$, and effective multiplication factor $\left(k_{eff}\right)$. The code may be used to compute the isotopic compositions and concentrations (gm/cm) of key isotopes present in the reactor core, as well as to solve the burn-up equations and fission products for a given specific power (Yasin, 2010). It was chosen for its precise lattice mechanics and unfettered distribution authority. In 1992, the Oak Ridge National Laboratory's Radiation Safety Information Computational Center (RSICC) provided the code and its 69-group library type 166259 developed in Winfrith. Since then, three significant features have been introduced to the RERTR software. The capacity to create up to 20 broad-group burnup-dependent macroscopic or microscopic ISOTXS cross-sections for each unit cell composition, an ENDF/B-V based nuclear data library4 (later upgraded to ENDF/B-VI), and a SUPERCELL option were the first (Costescu, 1994).

4.1 Lattice calculation

Even with a high-performance computer, performing a direct core calculation with several tens of thousands of fuel pins in its heterogeneous geometry model form, using fine-groups of a prepared reactor constant library, is difficult. The Monte Carlo approach can handle such a core computation, but it is difficult to acquire enough precision for a local calculation or modest reactivity due to statistical flaws. As a result, the Monte Carlo approach is not used for nuclear design calculations that need a quick computation time. Instead, the nuclear design calculation is split into two steps: lattice calculation in a two-dimensional (2D) unlimited arrangement of fuel rods or assemblies, and core calculation in a three-dimensional (3D) full core. The lattice computation generates few-group homogenized cross sections that preserve the crucial energy dependence (neutron spectrum) of nuclear processes, lowering the core calculation cost in terms of time and memory. Effective microscopic cross sections are initially generated in lattice calculation programs using fine-group infinite dilution cross sections depending on input parameters such as material compositions, dimensions, temperatures, and so on. The collision probability method is used to solve the effective cross sections, and then multi-group neutron spectra are obtained in each divided region (neutron spectrum calculation) (Yoshiaki et al., 2010).

4.2 Group collapsing and homogenization

The effective cross sections are handled with as little deterioration in accuracy as feasible in the core calculations with a massive quantity of space-dependent data (cross section and neutron flux) by using the findings from the multi-group lattice calculation. There are two techniques of processing. The first is homogenization, which reduces space-dependent information, and the second is group-collapsing, which reduces energy-dependent information. The basic principle behind both techniques is to keep the neutron reaction rate constant. A homogenized neutron flux g homo is initially defined in the homogenization as an averaged flux weighted by the volume V_k of the area (k). Following that, a homogenized cross section $\Sigma_{x,g}^{homo}$ meeting Eq. (4.1) is derived to reflect the reaction rate in the homogenized entire region of volume V^{homo}. In Eq. (4.3), the fine-group neutron flux $\varphi_{g,k}$ is utilized to determine the homogenized cross section.

$$\varphi_g^{homo} \equiv \frac{\Sigma_k \varphi_{g,k} V_k}{\Sigma_k V_k} = \frac{\Sigma_k \varphi_{g,k} V_k}{V^{homo}}$$

4.1

$$\Sigma_{x,g}^{homo} \varphi_g^{homo} V^{homo} = \Sigma_k \Sigma_{x,g,k} \varphi_{g,k} V_k$$

4.2

$$\Sigma_{x,g}^{homo} \equiv \frac{\Sigma_k \Sigma_{x,g,k} \varphi_{g,k} V_k}{\varphi_g^{homo} V^{homo}} = \frac{\Sigma_k \Sigma_{x,g,k} \varphi_{g,k} V_k}{\Sigma_k \varphi_{g,k} V_k}$$

4.3

(Yoshiaki et al., 2010)

5. Reactor Burnup System-argonne National Lab (Rebus-anl) Code

The REBUS-PC code has developed from its beginnings as a design tool for fast reactors to meet the demands of thermal reactor design and analysis. REBUS (the REactor BUrnup System) has been a multi-purpose tool since its inception in the 1960s (Olson, 2002). REBUS was created to analyze the fuel cycles of research reactors (Hoffman *et al.*, 2003). The specification of the external fuel cycle may include reprocessing, and discharged fuel may be recycled back into the reactor. In non-equilibrium cases, the initial composition of the reactor core may be explicitly specified, or the core may be loaded from external feeds and discharged fuel recycled back into the reactor, as in equilibrium problems (Abrefah et al., 2014). REBUS-PC can handle both equilibrium and non-equilibrium issues by employing a variety of core geometries, including triangular and hexagonal meshes (Waltar et al., 2011). Finite difference or nodal diffusion-theory approaches can be used to find the neutronics solution. Other characteristics include: completely automated restart capability, no limits on the number of neutron energy groups, and a general external cycle with no limits on the number of external feeds, reprocessing facilities, and so on. For non-equilibrium issues, fuel management is entirely universal (Ha et al., 2012). As is typical for thermal reactor systems, microscopic cross sections are allowed to fluctuate as a function of the atom density of various reference isotopes in the issue (Odoi, 2014). The prior capacity of fitting neutron capture and fission processes to low-order polynomials as functions of burn-up is preserved. Furthermore, as a function of burn-up, the user may now pick cubic spline interpolation for (n, gamma), (n, fission), (n, alpha), (n, p), (n, d), and (n, t) reactions. The user may set the location of the control rods at each time node in the issue. Output modifications have been considerably updated and reorganized for usage on a PC (Deen *et al.*, 2004).

5.1 Core lifetime analysis

Premised on a working schedule of two hours per day, five days per week, the core life of the HEU MNSR system is estimated to be around 10 years (Hassan et al., 2008). When the core's excess reactivity falls below 2.3mk, the core's lifespan is over. In terms of realistic reactor operational schemes, the anticipated core life time is 46.88 years if the core is run at 30kW for two hours per day, four days per week, and forty-eight weeks per year (Abrefah et al., 2013). The MNSR's reactivity losses grow owing to fuel depletion and the buildup of Xenon-135. Whenever the excess reactivity is insufficient to keep the reactor running continuously for 2.5 hours at the rated neutron flux level of $1 \times 10^{12} \frac{n}{cm^2} \bullet sec$, top beryllium shims should be added to compensate for the loss of excess reactivity. They are divided into four groups: 10 plates with a thickness of roughly 0.15 cm, 20 plates with a thickness of approximately 0.3 cm, 8 plates with a thickness of 0.6 cm, and 8 plates with a thickness of 1.2 cm. The shim plate has a minimum thickness of 0.14 cm and a maximum thickness of 10.95 cm (Khattab and Khamis, 2004). Every year, the MNSR depletes roughly 1mk reactivity. The extra reactivity of a newly loaded core, along with the additional reactivity of the top beryllium shim reflector, is sufficient to keep the core operational for 10 years (CIAE, 1993).

6. Results And Discussion

6.1 One group cross section generation using Winfrith Improved Multigroup Scheme-Argonne National Lab (WIMS-ANL) code.

The WIMS-ANL code for lattice cell calculations is an extension of the Winfrith WIMS-D4 algorithm. WIMS-ANL's input deck was configured and tested. The input deck is divided into three sections: prelude data, main card data, and edit data. The prologue part describes the type of problem, the primary transport procedure, the number of wide energy groups to be utilized, and the number of few energy groups to be used. It also specifies the number of materials, the maximum number of cell sections, and

the amount of nuclides that must be included in the edit chain. The annular area in the current cell is specified in the main card section. Each area is given a number. Every substance is assigned a number, and their outside radii are inputted as well. The edit data section contains information on the number of thermal groupings. In this scenario, the few groups were divided into two (i.e. fast and thermal). The nuclides that would be included in the ISOTXS files were also defined. As input, the number of burnup stages and the utilization of secondary libraries were entered. The section also demonstrates how neutron few groups are divided into large groups (Abrefah et al., 2014). After entering all of the required information, the deck was run to generate the ISOTXS file for use in the REBUS-ANL code. The WIMS-ANL code was used to create one group cross section data for the current study, and these one group cross sections were employed in the diffusion theory based code REBUS-ANL for core lifetime analysis. The WIMS-ANL computer code was used to compute the one group cross section data for both the converted High Enriched Uranium (HEU) and new Low Enriched Uranium (LEU) fuel of NIRR-1, first without burnup and subsequently with burnup. Table 1a displays the absorption cross-section (Σ_a), capture cross-section (Σ_γ), fission cross-section multiplied by the number of fission neutrons $\left(\upsilon\Sigma_f\right)$, fission cross section (Σ_f), number of fission neutrons (υ), transport cross section (Σ_{tr}), and diffusion coefficient (D) of converted HEU fuel without burnup. Table 1b depicts the scattering cross-sections of the transformed HEU fuel without burnup utilizing four energy groups.

Table 1
a: One group cross section data generated using the WIMS-ANL computer code at the without burnup for the converted High Enriched Uranium (HEU)

Broad Group	Absorption cross section(Σ_a)	Capture cross section (Σ_γ)	Nu*Fission cross section ($\upsilon\Sigma_f$)	Fission cross section (Σ_f)	Nu cross section (υ)	Transport cross section (Σ_{tr})	Diffusion cross section (D)
1	8.74E-04	4.92E-04	1.04E-03	3.82E-04	2.72E + 00	1.56E-01	2.14E + 00
2	6.36E-04	1.74E-04	1.13E-03	4.62E-04	2.45E + 00	3.59E-01	9.29E-01
3	1.07E-02	4.88E-03	1.43E-02	5.86E-03	2.43E + 00	4.95E-01	6.74E-01
4	1.52E-02	5.33E-03	2.40E-02	9.84E-03	2.44E + 00	5.95E-01	5.60E-01

Table 1
b: Scattering cross-sections of the converted HEU fuel without burnup

Group	1	2	3	4	5	6	7
1	6.64E-02	0.00E + 00	0.00E + 00	0.00E + 00	0.00E + 00	0.00E + 00	0.00E + 00
2	8.80E-02	2.40E-01	0.00E + 00	0.00E + 00	0.00E + 00	0.00E + 00	0.00E + 00
3	5.31E-04	1.18E-01	3.39E-01	6.50E-05	0.00E + 00	0.00E + 00	0.00E + 00
4	2.24E-07	7.13E-05	1.23E-01	1.23E-01	4.24E-03	6.10E-09	2.94E-10

Table 2a shows the absorption cross-section (Σ_a), capture cross-section (Σ_γ), fission cross-section multiplied by the number of fission neutrons ($\upsilon\Sigma_f$), fission cross section (Σ_f), number of fission neutrons (υ), transport cross section (Σ_{tr}) and diffusion coefficient (D) of the converted HEU fuel using seven energy groups at the End of Cycle (EOC), or after 252 Effective Full Power Days (EFPDs). Table 2b depicts the scattering cross-sections of the converted HEU fuel after 252 Effective Full Power Days employing seven energy groups at the End of Cycle (EOC) (EFPDs). When the values of the one group cross section data in Table 1a were compared to the values in Table 2a, it was discovered that the fuel's fission and absorption cross-sections had decreased. This decrease might be related to the isotopic composition of the nuclear fuel as well as the accumulation of several key fission products generated by exposure to neutrons of varying flux levels, resulting in fuel irradiation in the system's

core over the first ten years of operation. This is consistent with the one-group cross section values for generic MNSR fuel rods published in SAR (2012).

Table 2
a: One group cross section data generated using the WIMS-ANL computer code at the with burnup after 252 EFPD for the converted High Enriched Uranium (HEU)

Group	Absorption cross section (Σ_a)	Capture cross section (Σ_γ)	Nu*Fission cross section ($\upsilon\Sigma_f$)	Fission cross section (Σ_f)	Nu cross section (υ)	Transport cross section (Σ_{tr})	Diffusion cross section (D)
1	8.70919E-04	4.92271E-04	1.02939E-03	3.78648E-04	2.71860E+00	1.55776E-01	2.13982E+00
2	6.30931E-04	1.73315E-04	1.12033E-03	4.57616E-04	2.44819E+00	3.58776E-01	9.29085E-01
3	1.07227E-02	4.91288E-03	1.41400E-02	5.80978E-03	2.43383E+00	4.94664E-01	6.73858E-01
4	1.51181E-02	5.37516E-03	2.37324E-02	9.74296E-03	2.43585E+00	5.95423E-01	5.59826E-01

Table 2
b: Scattering cross-sections of the converted HEU fuel with burnup after 252 EFPD

Group	1	2	3	4	5	6	7
1	6.63891E-02	1.76555E-23	0.00000E+00	0.00000E+00	0.00000E+00	0.00000E+00	0.00000E+00
2	8.79850E-02	2.39799E-01	7.39847E-25	0.00000E+00	0.00000E+00	0.00000E+00	0.00000E+00
3	5.31250E-04	1.18262E-01	3.38590E-01	6.49631E-05	0.00000E+00	0.00000E+00	0.00000E+00
4	2.24281E-07	7.13458E-05	1.22919E-01	1.22532E-01	4.23933E-03	6.11694E-09	2.94322E-10

Similarly, with the existing Low Enriched Uranium (LEU) fuel, one group cross section data was obtained using the WIMS-ANL computer code, first without and then with burnup. Table 3a displays the absorption cross-section (Σ_a), capture cross-section (Σ_γ), fission cross-section multiplied by the number of fission neutrons ($\upsilon\Sigma_f$), fission cross section (Σ_f), number of fission neutrons (υ), transport cross section (Σ_{tr}) and diffusion coefficient (D) of the current LEU fuel without burnup. Table 3b depicts the scattering cross-sections of the transformed LEU fuel without burnup utilizing four energy groups.

Table 3
a: One group cross section data generated using the WIMS-ANL computer code without burnup for the current Low Enriched Uranium (LEU)

Group	Absorption cross section (Σ_a)	Capture cross section (Σ_γ)	Nu*Fission cross section ($\upsilon\Sigma_f$)	Fission cross section (Σ_f)	Nu cross section (υ)	Transport cross section (Σ_{tr})	Diffusion cross section (D)
1	0.000661718	0.0003725	0.000595601	0.000289218	2.059351962	0.118109892	1.620225441
2	0.000481525	0.000131738	0.000648831	0.000349787	1.854930995	0.271804174	0.703359549
3	0.008101127	0.00369472	0.008162576	0.004436692	1.839788701	0.374771773	0.510295302
4	0.011508143	0.004035421	0.013762847	0.007450009	1.847359848	0.450483242	0.423984227

Table 3
b: Scattering cross-sections of the converted LEU fuel without burnup

Group	1	2	3	4
1	0.757	0.573	0.434	0.329
2	1.51	1.15	0.868	0.657
3	2.27	1.72	1.3	0.986
4	3.03	2.29	1.74	1.31

Similarly, Table 4a shows the absorption cross-section (Σ_a), capture cross-section (Σ_γ), fission cross-section multiplied by the number of fission neutrons ($\upsilon\Sigma_f$), fission cross section (Σ_f), number of fission neutrons (υ), transport cross section (Σ_{tr}) and diffusion coefficient (D) of the current LEU fuel with burnup after 282 Effective Full Power Days (EFPDs). Table 4b depicts the scattering cross-sections of the present LEU fuel employing four energy groups after 282 Effective Full Power Days of burnup (EFPDs). When the values of the one group cross section data in Table 3a were compared to the values in Table 4a, it was discovered that the fuel's fission and absorption cross-sections had decreased. This decrease might be related to the isotopic composition of the nuclear fuel as well as the accumulation of several key fission products generated by exposure to neutrons of varying flux levels, resulting in fuel irradiation in the system's core over the first ten years of operation. This is consistent with the one-group cross section values for generic MNSR fuel rods published in SAR (2012).

Table 4
a: One group cross section data generated using the WIMS-ANL computer code at the with burnup for the current Low Enriched Uranium (LEU)

Broad Group	Absorption cross section (Σ_a)	Capture cross section (Σ_γ)	Nu*Fission cross section ($\upsilon\Sigma_f$)	Fission cross section (Σ_f)	Nu cross section (υ)	Transport cross section (Σ_{tr})	Diffusion cross section (D)
1	0.000659386	0.000372706	0.000590071	0.00028668	2.058292001	0.117940298	1.62008916
2	0.000477688	0.000131219	0.0006422	0.000346469	1.853560618	0.271634581	0.703423904
3	0.008122251	0.003722241	0.00810782	0.004399995	1.842688451	0.374521925	0.510181735
4	0.011446287	0.004069673	0.013604069	0.007376606	1.844217822	0.450803501	0.42385249

Table 4
b: Scattering cross-sections of the converted LEU fuel without burnup

Group	1	2	3	4	5	6	7
1	0.757	0.573	0.434	0.329	0.249	0.188	0.143
2	1.51	1.15	0.868	0.657	0.498	0.377	0.285
3	2.27	1.72	1.3	0.986	0.746	0.565	0.428
4	3.03	2.29	1.74	1.31	0.995	0.753	0.57

6.2 HEU and LEU Core Lifetime Estimates Based on REBUS-ANL Model

The core life was calculated using the REBUS-ANL model. A thorough input deck was created and tested. The REBUS input deck allows you to specify the various areas. The reactor is shown in a Cartesian plane. The dimensions of the different radii of the reactor's components (x axis) are stated, as are the parameters of the height (y axis). For analysis purposes, the components are separated into subsections. The greater the number of nodes desired, the greater the number of axial sections (the beryllium reflector was divided into 5 sub sections). The deck defined mesh intervals. The mesh intervals were designed keeping the region's stability in mind. In places with a lot of variation, a finer mesh was employed to capture every feature.

Because the output would produce a large number of nuclides, we indicated which ones we wanted to track in the deck. The nuclides Xe, Pm, and I are critical fission products in the reactor. The code allows the user to choose whether to halt the run after the neutronic calculation or to continue with the burnup calculation. In reactors that use fuel management, the homogenized fuel can be assigned to a certain fuel management path. The input also includes the number of days the reactor was operational, the power at which the reactor was operational, the start of a specific stage of operation, and the conclusion of that specific stage of operation. The deck also sets the timing of the neutronic computation. In other words, it splits the number of hours that the reactor operates into sub-intervals based on the reactor's operating time. Using the REBUS-ANL code for HEU and LEU cores at two power levels (full and half power), a simple reactivity rundown was performed to determine the core lifespan of NIRR-1 (Abrefah et al., 2014). A graph of excess reactivity against EFPDs and EHPDs was generated for HEU and LEU cores (Fig. 6.1 and 6.2) to show the precise amount of days that the HEU and LEU cores of NIRR-1 would take to attain an excess reactivity of 2.3 mk.

As the fuel burns, the excess reactivity of the core steadily decreases from 4mk to 2.3mk during the course of the fuel's lifetime in the core (Fig. 6.1 and 6.2). The NIRR-1 system is intended for intermittent use. The core lifetime analyses were carried out using the simple reactivity rundown calculation. Furthermore, as illustrated in Figs. 6.1 and 6.2, the equivalent reactivity rundown for HEU and LEU cores shows that the excess reactivity reduces essentially linearly with time. The first dip in the excess reactivity curve is caused by 135Xe accumulation, and it thereafter follows a linear declining pattern with 252 EFPDs and 506 EHPDs for HEU and 282 EFPDs and 566 EHPDs for LEU core. The core lifetime was calculated using the REBUS-ANL model for the HEU core at power levels of 31 kW and 15.5 kW and the LEU core at power levels of 34 kW and 17 kW. The method used is based on a simple reactivity rundown calculation that assumes the top Be shim tray is filled to a maximum of 10.95 cm for the fresh core. The end of core life was achieved when the core's excess reactivity dropped to 2.3mk in both models. According to the results shown in Figs. 6.1 and 6.2, the NIRR-1 HEU core operated at 31 kW would have a lifetime of about 252 EFPDs and 15.5 kW would have a lifetime of about 506 EHPD based on reactivity, while the NIRR-1 LEU core operated at 34 kW would have a lifetime of about 282 EFPDs and 17 kW would have a lifetime of about 566 EHPDs. That is, the lifespan of the LEU core is anticipated to be roughly 11.91% longer than that of the HEU core for full power and 11.81% for half power, which is consistent with the IAEA-TECDOC (2018) estimate of 11%. This might be attributable to HEU having a greater thermal utilization factor than LEU, resulting in more thermal neutrons being generated in HEU with higher fuel burnup.

7. Conclusion

The NIRR-1 fuel depletion analysis for the converted HEU core and current LEU core at full and half power was investigated, which is consistent with coordinated core conversion research from HEU to LEU fuel. The WIMS-ANL code was used to create one group cross section data, which was then utilized in the REBUSANL code for depletion analysis. The REBUS-ANL model was used to calculate the core lifetimes of HEU and LEU. The approach employed was based on a simple reactivity rundown calculation. For the depletion, two constant power levels of 31 kW and 34 kW for full power and 15.5 kW and 17 kW for half power were employed. Based on a regular working schedule of $2\frac{1}{2}$ hour operations per day, 5 days per week, and 48 weeks per year, the estimated core life for the HEU-fueled MNSR system was calculated to be around 250 EFPDs at 31kw, which is comparable to 10 years (SAR, 2012).The core lifetime of the selected LEU fuel has been calculated along the same lines as the converted HEU fuel, using appropriate values of the total EFPDs and EHPDs spanning the whole rundown computation in each scenario. Both of these estimations of the HEU core lifetime are quite comparable to SAR, 2012 calculations. Similarly, the predicted core lifetime for the core operating at half power of 15.5 kW is 506EHPDs. According to the NIRR-1 practical reactor operational schemes, the predicted core lifetime of 506 EHPDs is 14.06 years if the HEU core is run at 15.5kw for 6 hours per day, 3 days per week, and 48 weeks per year. The LEU core lifespan was calculated using the REBUS-ANL model, which was also utilized for the HEU core. The predicted core lifetime for the LEU core at 34 kW is 282EFPDs, while the expected core lifetime for the LEU core at half power of 17 kW is 566 EHPDs. In terms of NIRR-1 practical reactor operational schemes, the anticipated core lifetime of 282 EFPDs is 11.28 years if the LEU core is run at 34kw for 2 1 2 hours per day, 5 days per week, and 48 weeks per year. Similarly, if the LEU core is run at half power for 6 hours per day, 3 days per week, and 48 weeks per year, the anticipated core lifespan of 566 EHPDs is 15.72 years. This implies that the lifetime of the LEU core is predicted to be

approximately 11.91 % onger than that of the HEU core, which is in good agreement with the value 11 % eported in the literature, indicating that the selected LEU core can be operated for a longer period of time than the converted HEU core.

Declarations

COMPETING INTEREST STATEMENT

The authors state that they have no known competing financial interests or personal ties that may seem to have influenced the work described in this study.

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Figures

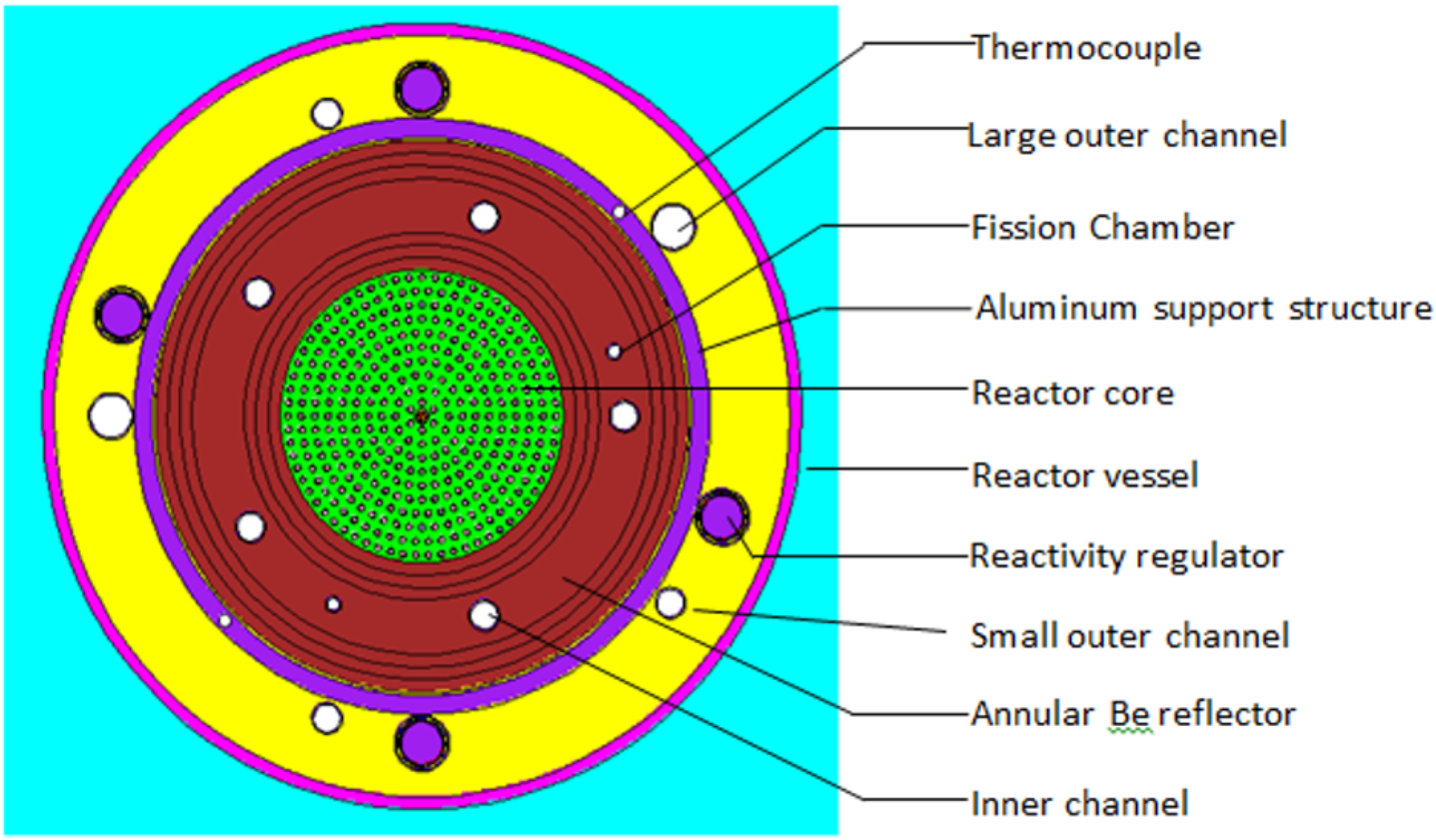

Figure 1

NIRR-1 radial core configuration showing fuel regions (SAR, 2012)

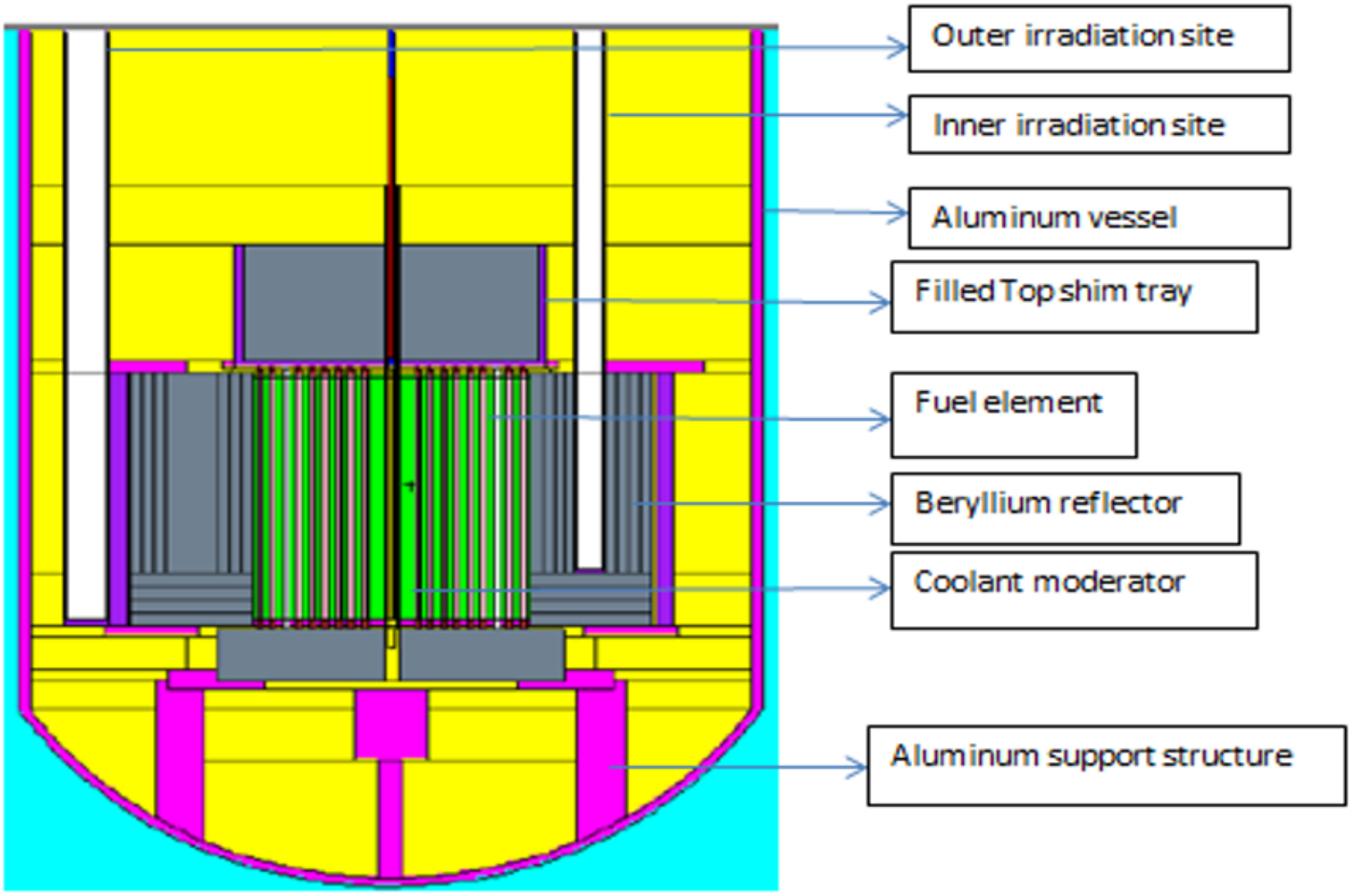

Figure 2

Axial cross section of NIRR-1 showing filled beryllium shim tray (SAR, 2012)

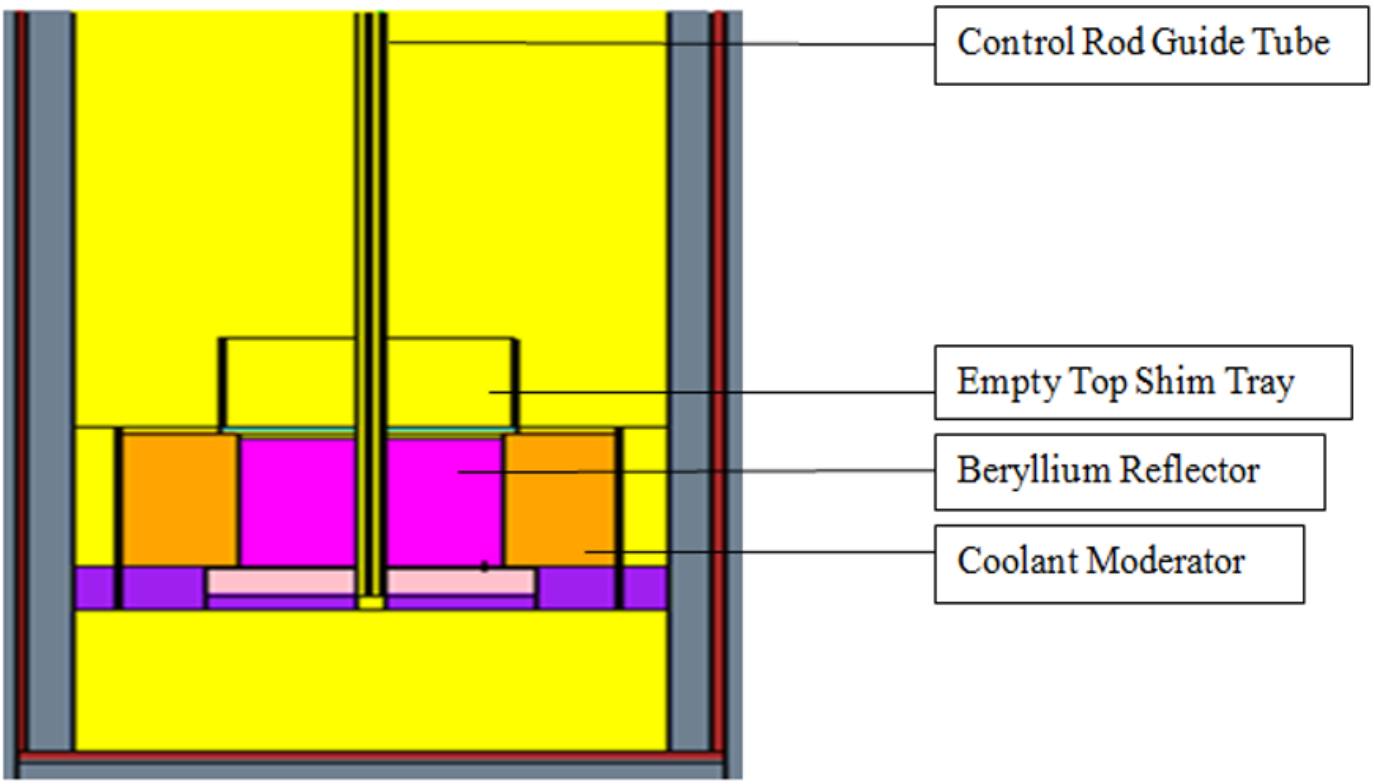

Figure 3

REBUS-ANL model of NIRR-1 showing empty beryllium shim tray (dimensions not to scale) (SAR, 2012)

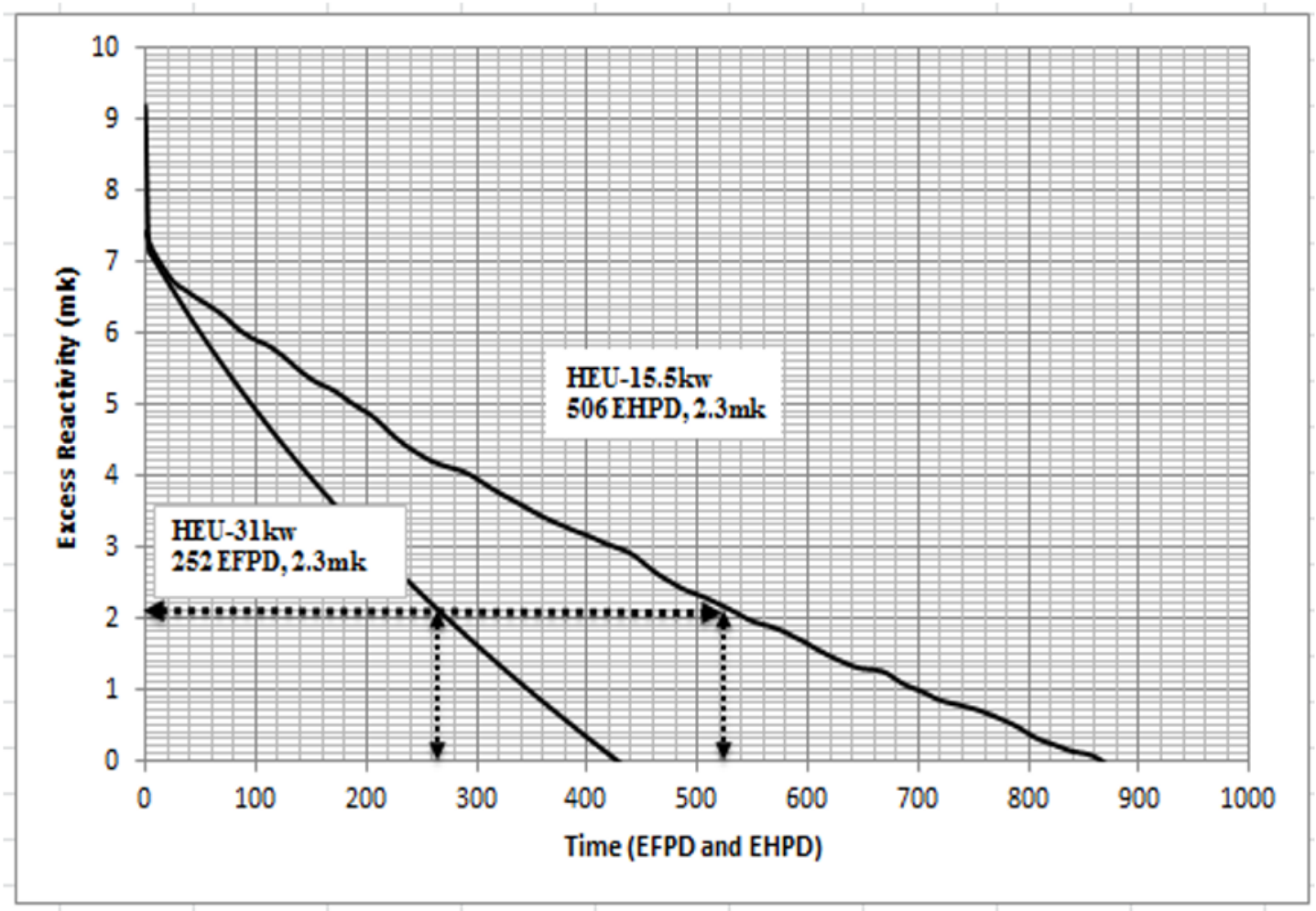

Figure 4

HEU Core Life Estimate Based on REBUS-ANL Model using Simple Reactivity Rundown Calculations at Two Power Levels of 15.5 kW and 31 kW

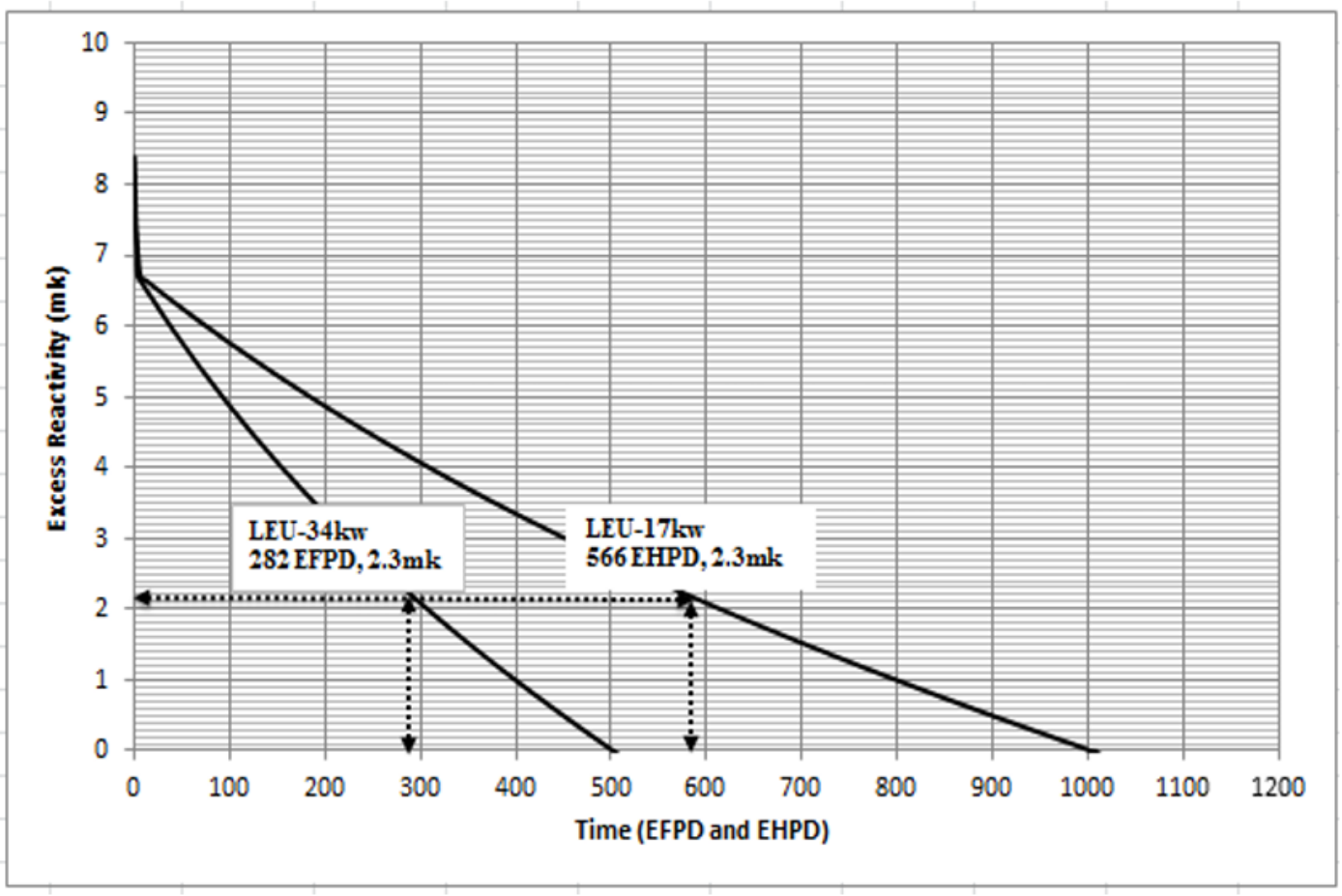

Figure 5

LEU Core Life Estimate Based on REBUS-ANL Model using Simple Reactivity Rundown Calculations at Two Power Levels of 17 kW and 34 kW