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Generating cross section library for Nigeria Research Reactor-1 (NIRR-1)

Salawu A.*[1], Balogun G. I.[2], Jonah S. A.[2] and Zakari Y. I.[3]

[1]*Department of Physics, University of Abuja, Gwagwalada, Abuja, Nigeria*
[2]*Center for Energy Research and Training Abu Zaria, Nigeria*
[3]*Department of Physics, Ahmadu Bello University, Zaria, Nigeria*

ABSTRACT

A number of cell models of the heterogeneous regions in a Miniature Neutron Source Reactor (MNSR) as well as a full core computational model have been developed and a cross section library has been generated for the Nigeria Research Reactor-1 (NIRR-1), a version of the MNSR, manufactured by China Institute of Atomic Energy. Because of the low density of U238 in the present HEU system, the resonance shielding effect does not produce any significant reduction in the absorption rate in the three regions of the energy dependent U238 cross section generated for NIRR-1 system. The same is true for the resonance plus spatial self shielding effect on the U238 absorption cross section except at a very low thermal energy of about 0.005eV. Various plots of multigroup neutron flux as a function of energy or lethargy has shown that the NIRR-1 behaves as expected of a typical thermal reactor system. The position of the reference NIRR-1 on the plot of k-infinity as a function of hydrogen to uranium ratio predicted an over moderated system for the reactor with a hydrogen to uranium ratio of about 180.

Keywords: Reactor, Uranium, Neutron, Code, Over-moderated, Cross section, k-infinity, Multigroup, Flux, Cell, Model, Heterogeneous, NIRR-1, MNSR

INTRODUCTION

All nuclear reactor systems are usually constructed and tested on paper before the actual manufacture and the real system are expected to behave as predicted from the computer calculation. A number of different computer codes such as MCNP, CITATION, VENTURE, SCALE etc are available to perform these types of reactor physics computer calculation. While some of the codes require the generation of a properly homogenized multigroup problem dependent cross section library to perform the calculations, others (like MCNP) are written to treat the heterogeneous geometry and the continuous energy cross sections in an explicit manner. Generating a problem-specific cross section library for use in a deterministic code requires the development of a number of models for the reactor system (See White, 1999). Different cell models of the heterogeneous regions in the MNSR are required to generate a properly averaged multigroup cross section library that is appropriate for a homogeneous system. Our goal in this study is to develop cell models for the heterogeneous regions in a Miniature Neutron Source Reactor (MNSR) as well as developing a full core computational model to generate a cross section library for the Nigeria Research Reactor-1 (NIRR-1) using the most recent version of the evaluated nuclear data library. The Nigeria Research Reactor-1 (NIRR-1) is a Miniature Neutron Source Reactor Manufactured by China Institute of Atomic Energy (Jonah *et al*, 2012). The major heterogeneous region in this reactor is the fuel cage consisting of 347 fuel pins, 4 tie rods and 3 dummy pins (See Ahmed *et al, 2006*). The fuel cell model and the control model developed in this work are used to generate cross section library for the active fuel region and the control region of the system.

These two libraries are then combined with the cross section data for the homogeneous region of the system to generate a cross section library for the entire reactor system using the full core computational model. This library can be used for the neutronic/depletion analysis of the present NIRR-1 core using the VENTURE and BURNER codes. VENTURE is a recent version of diffusion theory based deterministic code (See White and Tooker, 1999) that can be used like CITATION (See Balogun, 2003) to perform neutronic analysis for the Nigeria Research Reactor-1 (NIRR-1).

MATERIALS AND METHODS

The cross section libraries developed in this work were generated using the 2011 version of the SCALE code system called SCALE 6.1 and the selected nuclear data library is ENDF/B-VII.0, the current evaluated nuclear data library used around the world. Different modules of SCALE code system were used to generate the cross section libraries, perform the neutron flux calculations, as well as provide $k - infinity$ from the criticality calculation. These modules include AJAX, WORKER, CRAWDAD, BONAMI, CENTRM, PMC, PALEALE, XSDRNPM and WAX modules. All the input files associated with each of these SCALE modules run were originally prepared by Professor John R.White of the University of Massachusetts Lowell (MA-USA) during the successful core conversion studies of University of Massachusetts Lowell Research Reactor (UMLRR) and they were modified in this work to generate cross section libraries for a Miniature Neutron Source Reactor (i.e. NIRR-1).

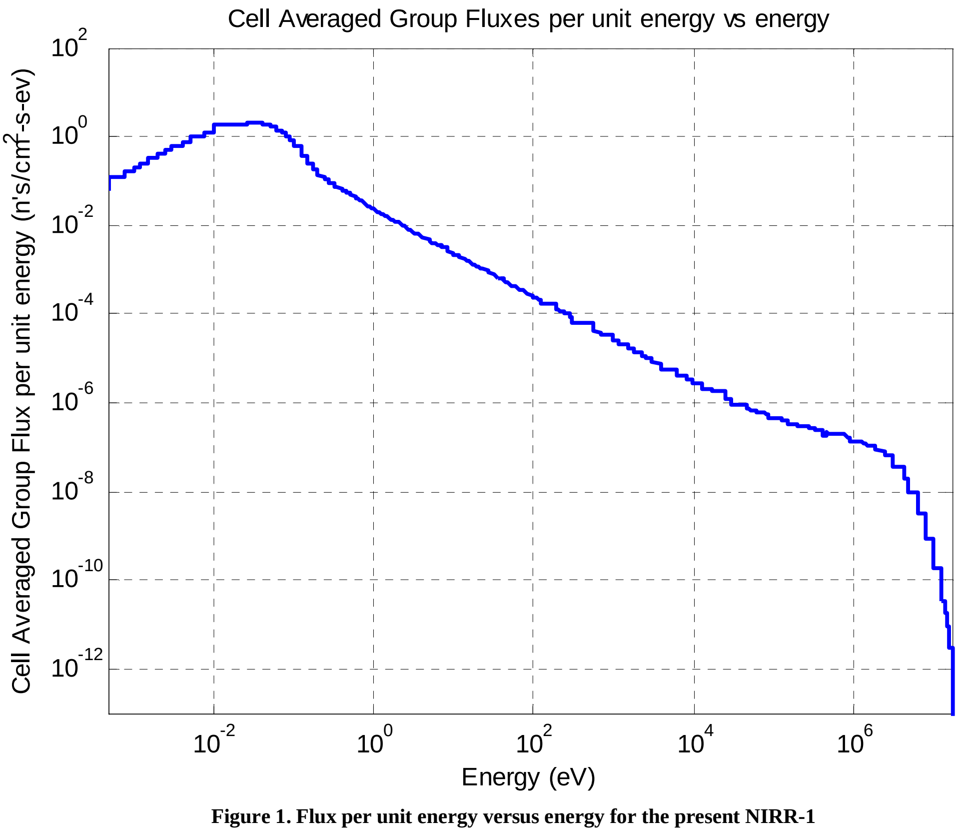

Figure 1. Flux per unit energy versus energy for the present NIRR-1

A total of 52 different nuclides were selected from the SCALE master library for various reactor physics calculation of the present NIRR-1 system. These nuclides represent the detail composition of all the materials present in the cold core of NIRR-1 as well as the fission product isotopes and the isotopes from the decay of various fission products in the current NIRR-1 system. This also include isotopes produced from the result of a neutron capture in the primary nuclides present in the core primary elements such as U236, Pu239 and many other isotopes produced from a variety of different reactions in the core of a typical nuclear research reactor system. The fuel cell model was used to perform a criticality calculation to generate a plot of k-infinity as a function of hydrogen to uranium ratio. The cell averaged multigroup neutron flux data and the 238 multigroup neutron energy boundary data were extracted from the SCALE output file and a plot of multigroup neutron flux (per unit energy or per unit lethargy) as a function of multigroup neutron energy was produced for graphical visualization.

RESULTS AND DISCUSSION

The result of the properly averaged multigroup neutron flux per unit energy as a function of energy is provided in figure 1 and the multigroup neutron flux per unit lethargy versus energy is provided in figure 2. These two composite flux spectra display the shape of the expected flux distribution in a typical thermal system. They display a profile that resemble a rough fission spectrum at high energy and a shape that looks Maxwellian at the low end of the spectrum.

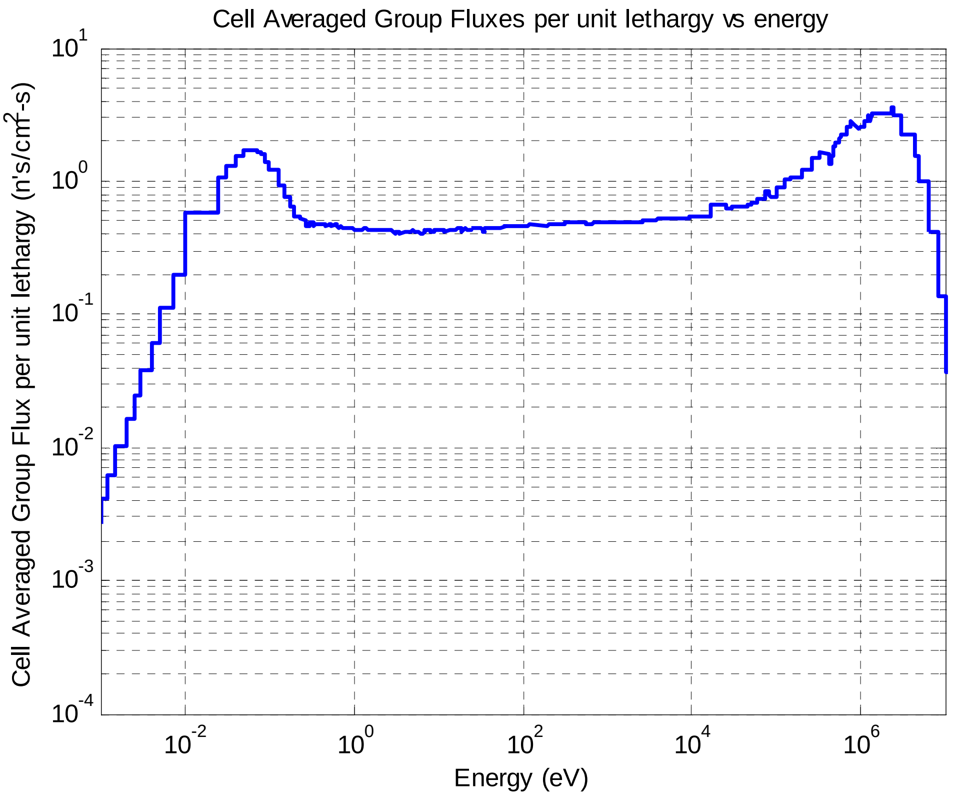

Figure 2: Flux per unit lethargy versus energy for the present NIRR-1

The flux in the intermediate energy region displays an approximately $\frac{1}{E}$ behaviour in the flux per unit energy spectrum (see figure 1) and this type of behaviour is caused by the slowing down mechanism in the intermediate energy region (DOE, 1993). In the case of the flux per unit lethargy, this region displays a nearly constant flux profile or a flux with a slightly positive slope (see figure 2).

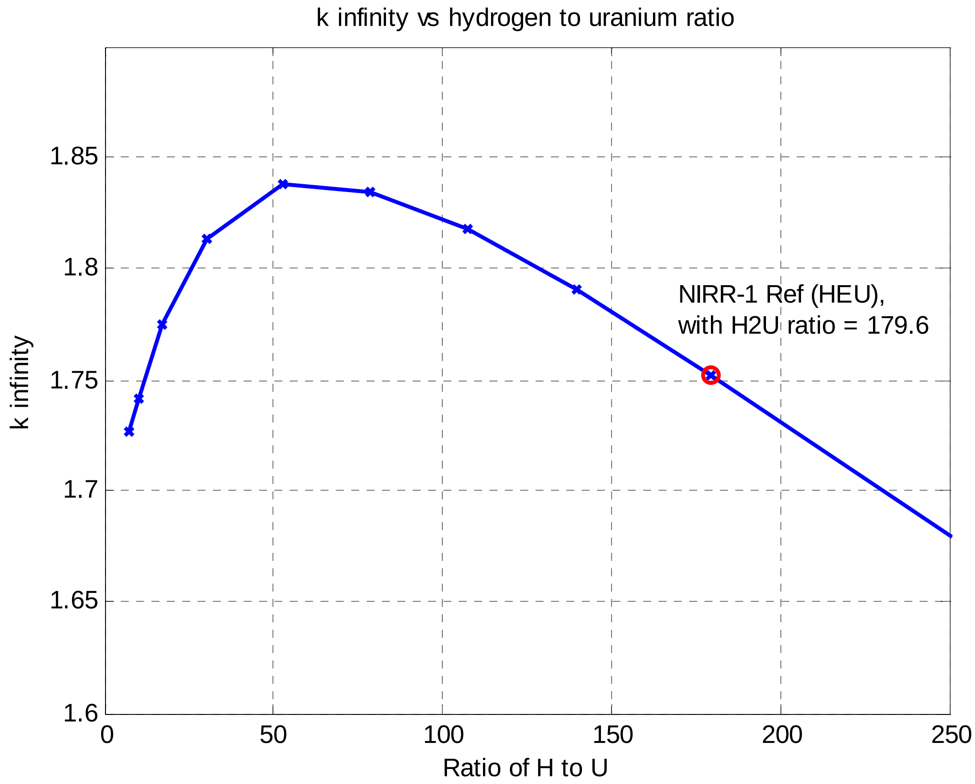

Figure 3: The changes in k-infinity versus H to U ratio for the present NIRR-1 system.

There are different kinds of temperature coefficient of reactivity (negative or positive) that are usually considered in the design of a nuclear reactor system because different materials are usually at different temperatures during reactor operation. The moderator and the fuel temperature coefficient of reactivity are considered the most important in reactor design (DOE, 1993). A negative moderator temperature coefficient is desirable in reactor design due to its self regulating effect but a negative fuel temperature coefficient is considered more important. This is because the fuel temperature immediately increases following an increase in reactor power level as compared to the moderator temperature. If there is any large positive reactivity insertion, the moderator temperature cannot turn the increase in power for several seconds whereas the fuel temperature coefficients start adding negative reactivity immediately (DOE, 1993). A decrease in power during reactor operation in a typical MNSR might be due to the combination of the effects of the negative reactivity from the build up of fission product poison (such as xenon and samarium) as well as the negative fuel temperature coefficient and not the effect of the moderator temperature coefficient. It has been reported that NIRR-1 is an under moderated system (See Ahmed *et al*, 2006 and 2008) without providing detail design information or graphical illustration to support such statement. By choosing high enriched uranium as fuel with hydrogen to U235 ratio of 197 and light water as moderator/coolant (See Ahmed *et al*, 2006) does not mean that the system will automatically become under-moderated. In reactor design, a HEU or LEU fuelled system, cooled and moderated by light water, is usually considered under or over moderated depending on the location of the reference system on the curve of neutron multiplication as a function of moderator to fuel ratio (Graves, 1979 and DOE, 1993). Figure 3 shows the result of the variation in k-infinity versus hydrogen to uranium ratio for the present NIRR-1 system. The position of the reference NIRR-1 as provided in the figure predicted an over moderated system for the core with hydrogen to uranium ratio of about 180 for NIRR-1 (see figure 3). Note that the hydrogen to U235 ratio in the system was calculated in this work to be about 196. The multiplication factor of an over moderated system decreases as hydrogen to uranium ratio increases. The result of the criticality calculation (using a 3D MCNP model) performed for an in-hospital MNSR with LEU fuel (at different level of enrichment below 16%) predicted that k effective decreases as H/U235 ratio increases (Ke, et al, 2009) and this is a prediction of an over-moderated system (See DOE, 1993).

A number of curves showing the energy behaviour of the U238 absorption cross section generated in this work for NIRR-1 system include the infinitely dilute multigroup cross section plots, the resonance shielded cross section and the resonance plus spatial shielded cross section plots (See figure 4). This figure illustrates the general behaviour of the U238 energy dependent neutron cross section in the NIRR-1 system. Just as expected of most isotopes and many reaction types within a typical thermal system like NIRR-1, these energy cross section plots consist of three distinct regions including the low energy region, the resonance energy region and the high energy region. The interface between the low energy and the resonance energy region is a little bit greater than 1eV. There is a very wild variation in the resonance energy region with a smooth energy changes in the high energy region. The low energy region displays an approximately 1/v behaviour as expected in the thermal energy region of the system. Because of the low density of U238 in the HEU core (with 90.2% U235 enrichment), the resonance shielding effect does not produce any significant reduction in the absorption rate in the three regions of the energy dependent U238 cross section generated for NIRR-1 system. The same is true for the resonance plus spatial self shielding effect on the U238 absorption cross section except at a very low thermal energy below about 0.005eV. At such low thermal energy, the probability for neutron absorption in U235 for fission process dominates the reaction rate as compared to the absorption process in U238.

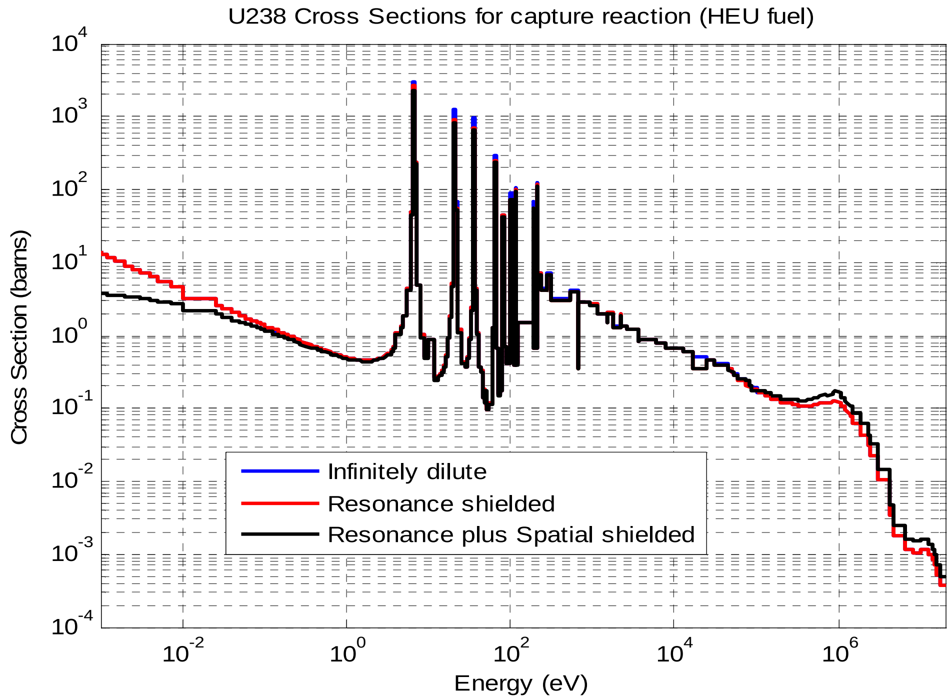

Figure 4: Resonance and, spatial shielded U238 cross section versus energy for NIRR-1 core.

CONCLUSION

A cross section library has been created for the Nigeria Research Reactor-1 (NIRR-1) using the most recent version of the evaluated nuclear data library and this library will be used to perform detail reactor design calculation for the Nigeria Research Reactor-1 (NIRR-1) using the most recent version of deterministic code, VENTURE-PC. Various plots of multigroup neutron flux has predicted that the NIRR-1 behaves as expected of a typical thermal reactor system. A plot of k-infinity as a function of hydrogen to uranium ratio predicted an over moderated system for the Nigeria Research Reactor with hydrogen to uranium ratio of about 180 for NIRR-1 system. Except at a very low energy of about 0.005eV, the resonance and spatial shielding effects does not produce any significant reduction in the absorption rate of the energy dependent U238 cross section library generated for the present HEU fuelled NIRR-1 system.

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