POST-CONVERSION ACTIVITIES: STANDARDIZATION OF METHODOLOGIES FOR OPTIMAL AND EFFECTIVE UTILIZATION OF MNSR FACILITIES FOR NAA,

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ABSTRACT

Over the years, studies under the aegis of the IAEA Coordinated Research Project entitled "Conversion of MNSR to LEU" and RERTR programme have been performed to convert all Miniature Neutron Source Reactor (MNSR) facilities to LEU. As the name suggests, the MNSR is a compact low-power research reactor designed mainly for use in neutron activation analysis and limited radioisotope production.

The prototype MNSR was built by the China Institute of Atomic Energy (CIAE), Beijing, China and was critical in 1984. Subsequently, the commercial versions of the reactor have been installed in China, Ghana, Iran, Nigeria, Pakistan and Syria. The nominal power of MNSR is approximately 30 kW and they have common operational, utilization and spent fuel management issues. The cores are fueled with HEU (>90% enrichment) consisting of a total ^{235}U loading of approximately 1 kilogram.

The IAEA in collaboration with M3 Programme of US DOE have supported the HEU core discharge and conversion of the prototype MNSR in China and are in the process of completing the processes for the MNSR facilities in Ghana and Nigeria with those in Iran, Pakistan and Syria are to follow. Therefore, in this work, uniform and standardized methodologies for reactor characterization and NAA protocols are proposed for optimal and effective utilization of MNSR facilities after conversion to LEU. The methodologies are based on the determination of neutron flux spectrum parameters by computational and experimental procedures. These data are indispensable for NAA protocols and preliminary data are presented for the MNSR facility in Nigeria (i.e. NIRR-1).

1. Introduction

The MNSR is a compact low-power research reactor designed mainly for use in neutron activation analysis and limited radioisotope production. The prototype was built by the China Institute of Atomic Energy (CIAE), Beijing, China and was critical in 1984. Subsequently, the commercial versions of the reactor have been installed in China, Ghana, Iran, Nigeria, Pakistan and Syria. The nominal power of MNSR is approximately 30 kW and they have common operational, utilization and spent fuel management issues. The cores are fueled with HEU (>90% enrichment) consisting of a total ^{235}U loading of approximately 1 kilogram. The Nigeria Research Reactor-1 (NIRR-1) is the last of the commercial MNSR facilities to be commissioned outside China in 2004 and it is first nuclear research reactor in Nigeria [1- 4]. Like all MNSR facilities, NIRR-1 is specifically designed for use in neutron activation analysis (NAA), therefore is a need for a careful and complete characterization of the neutron flux parameters in the irradiation channels in order to optimize its utilization for NAA via relative, absolute and the single comparator methods. Like the Canadian SLOWPOKE, NIRR-1 can operate on the same fuel loading for over ten years. It is known to exhibit stable neutron flux characteristics, which make them suitable for NAA via the k_0-standardized method [2].

The neutron spectrum parameters that characterize reactor facilities for NAA, especially via the k_0-standaradization method based on the *Hogdahl* convention for the so-called "*1/v*" nuclides include the non-ideality of the epithermal neutron flux shape factor, α, approximated by a $1/E^{1+\alpha}$ distribution; and the thermal-to-epithermal neutron flux ratio, *f*. The experimental methods for the determination of neutron spectrum parameters required in both the *Hogdahl* convention and the *Wescott* formalism have been enumerated in several published works [5,6]. However for reactors with stable neutron flux characteristics, the "Cd-ratio for multi-monitor" method has been recommended for neutron spectrum monitoring. Moreover, the method has been adjudged as the most accurate for α-monitoring [7].

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2. Methodology

In the experimental methodology Jonah et al., 2005 [2] have developed for the HEU core of NIRR-1. In their work, two sets of detector foils are prepared for each of the irradiation channels. One set was arranged in a stack inside the vial for the 'bare' irradiation. A second set arranged in a stack inside 1 mm thick Cd-cover and kept inside the vial for the 'Cd-covered' irradiation. Characteristic data of the detector foils are given in Table 1 except the data of F_{Cd} (^{198}Au) for which a value of 0.991 was used and G_e (^{95}Zr) for which a value of 0.983 was adopted. In the inner channel, B2, the bare irradiation was carried out for 30 minutes and the Cd-covered irradiation lasted for 1 hour. However in the outer channel, B4, the irradiation time was increased by a factor of 2 since the neutron flux in the outer channel is 50% of the value in the inner channel. All irradiations were performed at a thermal power level of 15.5 kW, which corresponds to a preset neutron flux value of 5.0 x10^{11} n/cm^2.s on the control console. On the basis of the activities of the monitor foils are then used to calculate the respective Cd ratios, which is used to determine the neutron flux parameters.
Similarly, in another previous work [8], we have developed a methodology based on the MCNP simulation to determine the neutron spectrum parameters f and a in the irradiation channels of NIRR-1 HEU core and the proposed LEU. The neutron flux distributions were simulated in 640 energy-group structure in the irradiation inner and outer channels respectively [9]. The standard MCNP volume-averaged track length estimator via tally card f4:N was used to calculate the neutron flux distributions in the irradiation channels on the basis of the 640 group energy structure.
Reaction rate due to irradiation in an energy–dependent neutron flux density, φ(E), for a monitor reaction having an energy dependent cross section σ(E) is given below as:

$$R = \int_0^\infty \varphi(E)\sigma(E)dE \qquad (1)$$

Considering that the energy bin is relatively small for the 640 energy group structure, the equation above, which describes a continuous representation can be modified to a discrete interval description given below:

$$R = \sum_0^\infty \varphi(E)\sigma(E) \qquad (2)$$

Where,

φ(E) = neutron flux density per unit energy interval,

σ(E) = energy-dependent activation cross section

The φ(E) data were obtained from the standard MCNP output for the respective experimental channel and were processed into point-wise data format for the calculation of the reaction rate. Similarly, the capture cross section data as a function of energy, σ(E) were retrieved for the ENDF-VI data libraries from the Nuclear Data Services of the IAEA, Vienna, Austria.

In the “Cd-ratio for multi-monitor” method for the determination of f (flux ratio of thermal to epithermal neutrons) and α (measure of the deviation of epithermal neutrons from the ideal *1/E* distribution), the parameters are defined as follows:

$$f = Q_{O,i}(\alpha).(F_{Cd,i}.R_{Cd,r} - 1).\frac{G_{e,i}}{G_{e,i}} \tag{3}$$

where, *i* is the monitor with well known Q_O value.

A set of N monitors having reaction rates with and without Cd cover are used to deduce the R_{Cd}. The R_{Cd} data are then used to calculate f and α based on the plot of

$$\log \frac{\overline{E}_{r,i}^{-\alpha}}{(F_{Cd}.R_{Cd,i} - 1)Q_{O,i}(\alpha)G_{e,i}/G_{th,i}} \text{ versus } \log E_{r,i} \tag{4}$$

Here, the slope of the graph is α.

Mathematically, by using an iterative procedure for a set of N monitors, eq. 4 can then be written as given below in eq. 5 and the α value is found as the root of the equation.

i.e.

$$\alpha + \frac{\sum_{i=1}^{N}\left[\left(\log \overline{E}_{r,i} - \frac{\sum_{i=1}^{N}\log \overline{E}_{r,i}}{N}\right)\left(\log \frac{\overline{E}_{r,i}^{-\alpha}}{(F_{Cd,i}.R_{Cd,i} - 1)Q_{O,i}(\alpha)G_{e,i}/G_{th,i}} - \frac{\sum_{i=1}^{N}\log \frac{\overline{E}_{r,i}^{-\alpha}}{(F_{Cd,i}.R_{Cd,i} - 1)Q_{O,i}(\alpha)G_{e,i}/G_{th,i}}}{N}\right)\right]}{\sum_{i=1}^{N}\left(\log \overline{E}_{r,i} - \frac{\sum_{i=1}^{N}\log \overline{E}_{r,i}}{N}\right)^2} = 0 \tag{5}$$

where,

$$Q_{O,i}(\alpha) = \frac{Q_{O,i} - 0.429}{(\overline{E}_{r,i})^{\alpha}} + \frac{0.429}{(2\alpha + 1)(0.55)^{\alpha}} \tag{6}$$

In this work the R_{Cd} data are calculated on the basis of a Cd-cut off energy of 0.55 eV using the equation below.

$$R_{CD} = \frac{\sum_{0}^{20MeV} \varphi(E)\sigma(E)}{\sum_{O.55eV}^{20MeV} \varphi(E)\sigma(E)} \quad (7)$$

3. Results and Discussion

Results of the neutron spectral distributions obtained by MNCP in the experimental channels of NIRR-1 HEU and LEU cores are displayed in Fig. 1. As can be seen the neutron energy spectra belonging to the two cores are comparable. This is because in the feasibility search the dimensions of the current HEU core and the proposed are retained except for changes in the materials of the fuel and dimensions of Control Rod and its guide tube. The changes are to maintain important safety criteria for the LEU core, which include that the shutdown margin must be greater than 2.5 mk and the safety reactivity factor must be greater than 1.5. As expected, the neutron flux distributions in the inner channels for HEU and LEU cores show pronounced fast neutron components compared with the outer channels. This is due to compact nature of the cores and the proximity of the inner channels to cores. Reaction rates of the monitor reactions have been calculated by the expression given in eq 3. Consequently, the R_{Cd} data deduced using eq. 4 are substituted in eqs. 4 and 5 to determine f and α values in the experimental channels for the two cores. A comparison of measured and calculated f and α values for the HEU core is given in Table 1. The measured data for the HEU core agree well with calculated data, which indicate the suitability of the computational method used in the present work. Furthermore, calculated data for the proposed LEU core presented in the Table are identical with the parameters of the current HEU core. The values of for the LEU core in inner and outer channels are slightly lower than for the HEU core indicating hardening of the neutron spectra for LEU core due to composition of the fuel. This indicates that the impact of LEU conversion of NIRR-1 on neutron spectrum parameters for NAA is minimal because the core configuration of the current HEU core is similar to the proposed LEU core except for the minor changes to the CR and guide tube.

Table 1 Measured and calculated neutron spectrum characteristics in inner and outer channels of NIRR-1 HEU and LEU cores

parameters		α		f	
		Expt.	MCNP	Expt.	MCNP
HEU	Inner	-0.052±0.002	-0.056±0.004	19.2±3.3	17.2±1.1
	Outer	0.029±0.005	0.021±0.005	48.3±3.3	46.7±2.9
LEU	Inner	-	-0.047±0.006	-	14.7±0.7
	Outer	-	0.028±0.004	-	43.7±2.7

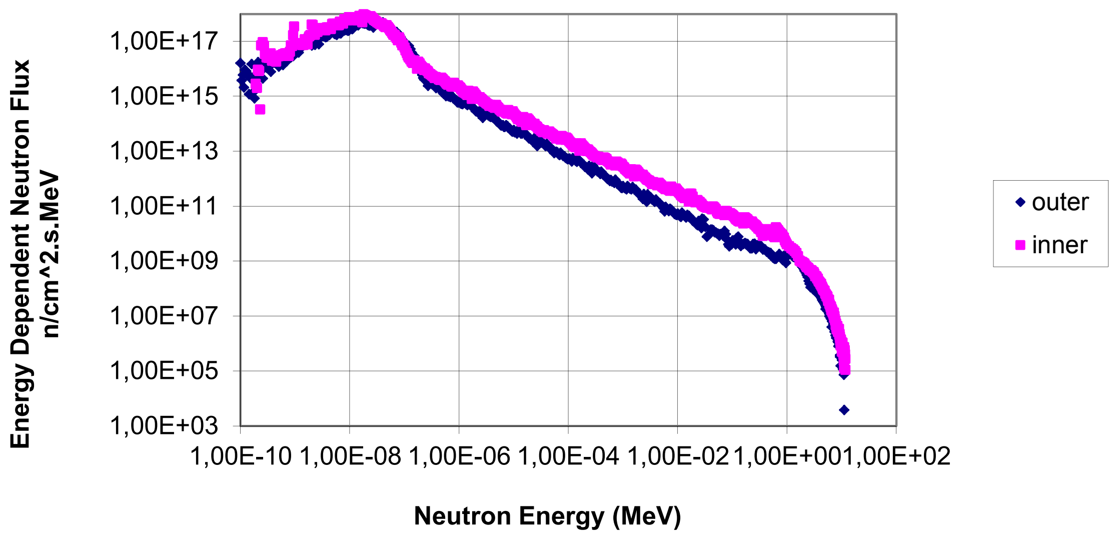

Fig. 1 Simulated neutron spectral distributions in inner and outer irradiation channels of NIRR-1 LEU core

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