CALCULATION OF ACTIVITY INVENTORY IN THE GRAPHITE REFLECTOR OF DALAT NUCLEAR RESEARCH REACTOR USING MCNPX AND ORIGEN2.1 CODE

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Abstract: The objective of the study is to estimate activity inventory in the graphite reflector of the Dalat Nuclear Research Reactor (DNRR). The calculation of neutron energy spectrum and activation cross-section was done using MCNPX code. ORIGEN2.1 point depletion code was used to calculate the neutron induced activity of materials at different time points by modelling the irradiation history and radioactive decay. The calculated results will be used to formulate a preliminary decommissioning plan and to evaluate of the amount of radioactive waste from dismantling actions.

Keywords: *DNRR, MCNPX, ORIGEN2.1, activity inventory, decommissioning.*

INTRODUCTION

The DNRR with a nominal power of 500 kW was reconstructed and upgraded from the USA-made 250 kW pool-typed TRIGA Mark II reactor, with the highest thermal neutron flux of 2.10×10^{13} (n/cm^2.s), light water cooled, moderated and shielded. After the reconstructed and upgraded activities, since March 1984, the reactor has been officially put into operation for the purposes of radioisotope production, neutron activation analysis, fundamental and applied research, and training [1]. These are based on 3D Monte Carlo MCNPX model estimating neutron flux distribution and using these fluxes in a point depletion code ORIGEN2.1 to calculate actual activation of materials taking into consideration the detailed irradiation history. This report presents the calculated results of activities in the graphite reflector of the Dalat Reactor (DR). These results are important basis in preparation of the Decontamination and Dismantling plan (D&D) for the DR in future.

The graphite reflector surrounding the core is a structure retained from the former TRIGA reactor [2]. It consists primarily of a ring-shaped block of graphite having an approximate inside diameter of 45.7 cm, a radial thickness of 30.5 cm and a height of 55.9 cm. From the design stage of the DNRR, the selection of materials for the purpose of the activated radiation level is low or generates mainly short-lived isotopes was considered and realized. However, for the impurities, although with very small amounts but they have long-life activation products such as ^{55}Fe, ^{60}Co, ^{152}Eu, ^{154}Eu etc., which are the main factors contributing to the radiation dose while conducting dismantling activities of the reactor.

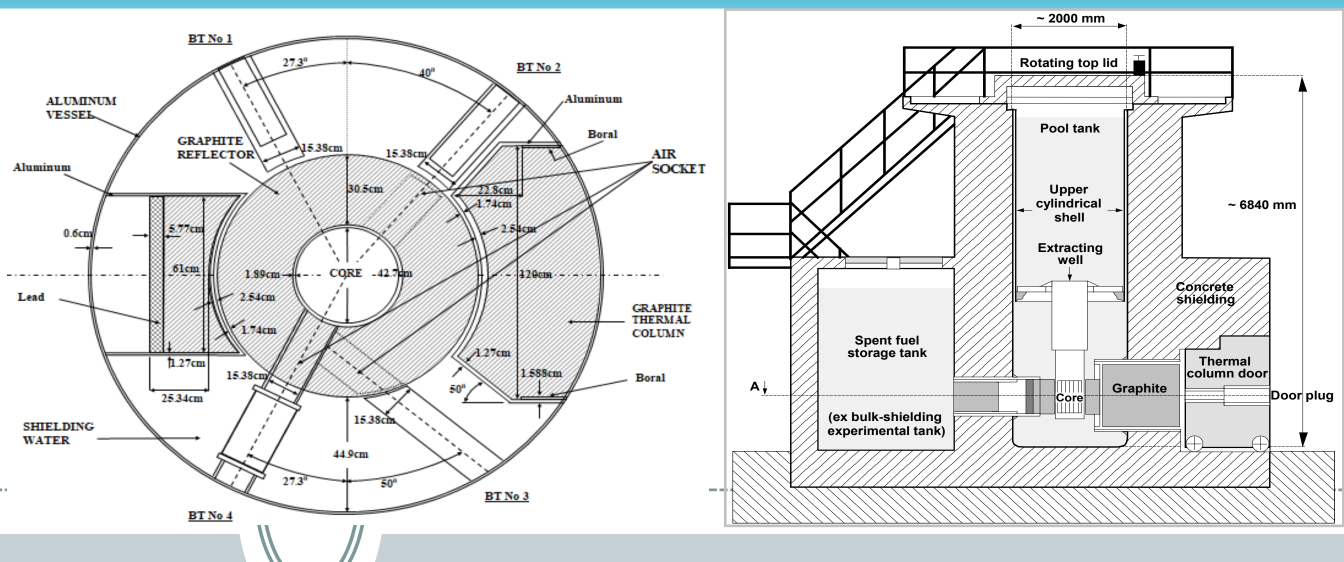

Figure 1. Vertical (left) and horizontal (right) section views of DNRR

RESULTS AND DISCUSSION

1. Method and model of calculation

The calculation of neutron energy spectrum and activation cross-section was done using MCNPX code [3]. The code treats an arbitrary three-dimensional configuration of materials in geometric cells bounded by first- and second-degree surfaces and fourth-degree elliptical tori. Point wise cross-section data are used. For neutrons, all reactions given in a particular cross-section evaluation (such as ENDF/B-VII.1) are accounted for.

ORIGEN2.1 uses a matrix exponential method to solve a large system of coupled, linear, first-order ordinary differential equations with constant coefficients. ORIGEN2.1 has been variably dimensioned to allow the user to tailor the size of the executable module to the problem size and the available computer space.

As the original libraries of OREGEN2.1 isn't good suitable for research reactors, the activation cross-sections of material compositions in the reactor were determined by MCNPX code. The flowchart of calculation procedure is shown in Fig. 2.

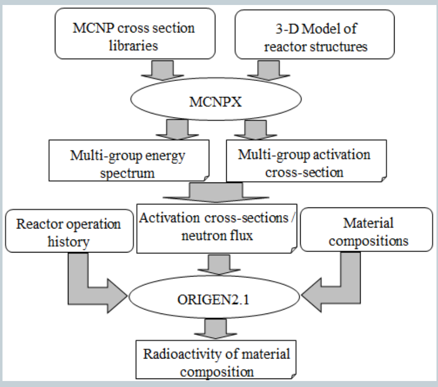

Figure 2. Flowchart of calculation procedure

The three dimensional calculated model for the entire of DR structures as the core, aluminum, graphite… was basically described as their real geometry.

The cross sections which are used in MCNPX calculation were selected from the END/B-VII.1 library. The neutron spectral densities and the activation cross-sections in nominal power condition and at different locations in the structural components of reactor were calculated by MCNPX code (number of histories, active cycles and standard deviation are 200000 particles, 2005 cycles and 0.0025, respectively). The 3D calculation models for the DNRR in MCNPX are shown in Fig. 3.

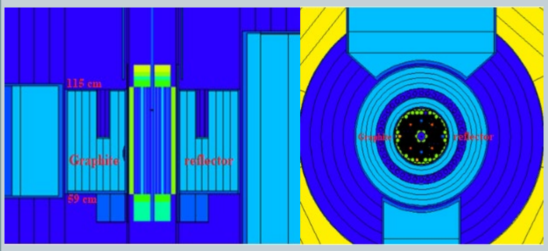

Figure 3. The vertical (left) and horizontal (right) cross-section models of the DNRR using MCNPX code

2. Results and Discussion

The calculated results for thermal neutron flux distributions at nominal power of 500kW is shown in Table 1. *(*) neutron flux=3.74×10^{12} n/cm^2.s ± 9.35×10^{9} and (**) rotary specimen rack.*

Table 1. The calculated neutron flux distribution in the graphite reflector

z axis (cm)	Distance from the reactor core (cm)				
	24	32	39	47	51
115	3.74E+12 *	**	2.56E+12	1.06E+11	9.03E+11
101	5.89E+12	4.36E+12	3.20E+12	1.73E+12	1.29E+12
73	6.19E+12	4.37E+12	4.24E+12	2.10E+12	1.54E+12
59	4.65E+12	3.86E+12	2.80E+12	1.57E+12	1.16E+12

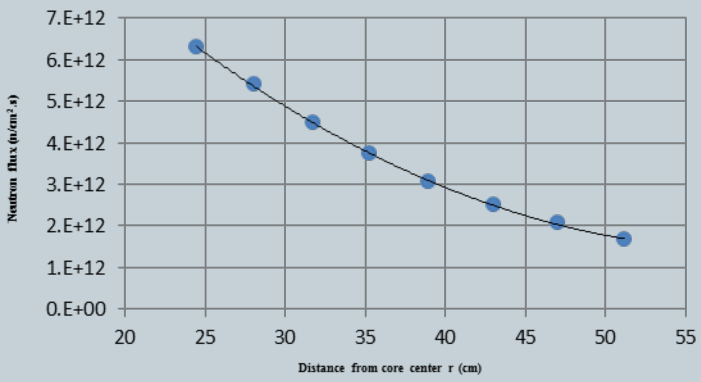

Figure 4. Neutron flux distribution in the graphite reflector.

In the reflector, there is a quantity of 770 kg graphite in the surrounding of the reactor core, therefore the specific activities are fairly high (Graphite: thousands of Bq/g – see in Table 2). The values of total activities of graphite in the reflector are 6.12E+04 MBq, 3.71E+04 MBq, and 2.38E+04 MBq in corresponding to decay time of 1 year, 5 years and 10 years.

Table 2. Calculated results of activity of graphite in the reactor reflector

Decay time (year)	Specific activity > 1 Bq/g		
	Mass (kg)	Specific activity (Bq/g)	Total activity (MBq)
1	770	7.95E+04	6.12E+04
3		5.95E+04	4.58E+04
5		4.82E+04	3.71E+04
10		3.09E+04	2.38E+04

Table 3 and Fig. 5 illustrate the calculated radioactivity of some long-life isotopes in the graphite reflector of the reactor. The main nuclides of the activated materials are ^{14}C, ^{55}Fe ^{60}Co, ^{152}Eu and ^{154}Eu after permanent shutdown reactor. The values in table 2 can be used for preparing and establishing the preliminary decommissioning plan for the reactor. However, before implementing the decommissioning activities, it will be necessary to establish a detailed plan based on both calculated and measured data after shutdown of the reactor.

Table 3. Calculated results of activity of long-life nuclides in graphite reflector

Nuclide	Activity (MBq) vs. decay time (year)			
	1	3	5	10
C^{14}	1502	1501	1501	1500
Co^{60}	32858	25256	19413	10057
Fe^{55}	7400	4342	2548	672
Eu^{152}	25472	23007	20778	16102
Eu^{154}	5161	4393	3739	2499

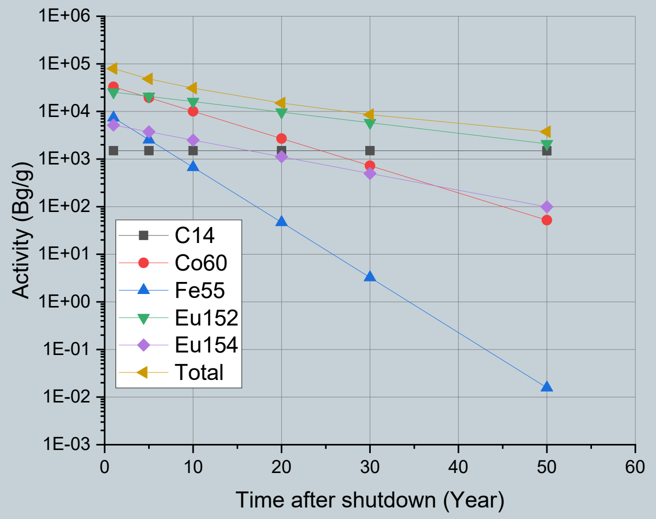

Figure 5. Activity of long life nuclides in graphite reflector

This report describes the methodology and results in activity inventory calculations for DNRR decommissioning planning. Total inventory of graphite reflector was around 6.12E+04 MBq (1 year) with main contributing nuclides being ^{14}C, ^{55}Fe, ^{60}Co, ^{152}Eu and ^{154}Eu based on results of impurity analysis contained in the graphite sample [5].

CONCLUSION

Neutron flux distribution in the different locations of the graphite reflector as well as activation cross sections of interested compositions were determined by using MCNPX code.

The values of total activities of graphite in the reflector are 6.12E+04 MBq, 3.71E+04 MBq, and 2.38E+04 MBq in corresponding to decay time of 1 year, 5 years and 10 years.

The activity of long-lived activation products such as ^{14}C, ^{55}Fe, ^{60}Co, ^{152}Eu and ^{154}Eu provide data for developing the decommissioning plan and the dismantling strategy of the operational DNRR.

Detailed studies in future should be systematic calculation and measurement of the activity inventory in all structural components of the reactor.

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