FLUX MEASUREMENTS IN THE LEU REFUELED SLOWPOKE-2 REACTOR OF ÉCOLE POLYTECHNIQUE DE MONTRÉAL

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The HEU (93%) SLOWPOKE-2 reactor of École Polytechnique de Montréal was refueled in September 1997 with (20%) LEU fuel after twenty years of operation.Upon completion of refueling the thermal and epithermal neutron fluxes were measured at four radii in the new core by irradiating gold and copper wire at 20W for 10 minutes. These flux profiles were normalized to the values measured in an irradiation site in the beryllium reflector. We have also used the DRAGON/DONJON chain of codes and a detailed SLOWPOKE reactor model to calculate the flux distribution in the core and in the irradiation site. The calculated flux profiles along the four vertical lines are in agreement with the experimental values. This work also enabled us to validate our codes, normally used for power reactors, for a low power research reactor.

INTRODUCTION

SLOWPOKE is an acronym for Safe Low Power Critical Experiment. It is a 20 kW pool-type research reactor, it produces a thermal neutron flux in the irradiation sites of 10^{12} n.cm^2.s^{-1} which is used mainly for neutron activation analysis.

After twenty years of operating with highly enriched uranium (HEU) fuel, the SLOWPOKE reactor of École Polytechnique de Montrèal was refueled in September 1997 with new low enriched uranium (LEU) fuel. It is the second LEU core in Canada. The overall SLOWPOKE reactor geometry has been retained, and only the fuel cage and fuel elements have been altered for the LEU installation. Table 1 shows the main differences between the two cores.

Core	**HEU**	**LEU**
Fuel material	U/AL	UO2
Enrichment (% U235)	93	19.76
No. of fuel elements	296	198
Fuel length (mm)	220	227
Total mass of U^{235} (kg)	0.818	1.118
Fuel density (g/cm^3)	3.45	10.6
Fuel radius (mm)	2.108	2.083
Fuel sheath material	Al	Zr-4
Inner sheath radius (mm)	2.108	1.121
Outer sheath radius (mm)	2.616	2.629

Table 1: Physical Properties of SLOWPOKE-2 Core

Figure 1 shows a 2-D cut of the reactor inside the pool.

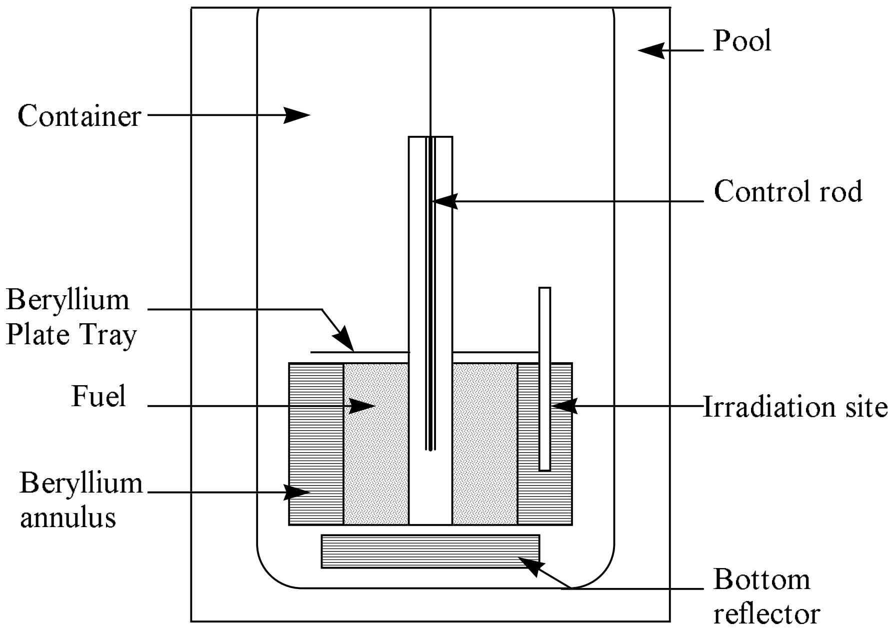


Figure 1: Representation of The LEU SLOWPOKE Reactor

Core excess reactivity after the 198 elements and without Beryllium plates was about 3.5 mk.

The objective of this work was to measured thermal and epithermal fluxes in the core normalized to values in irradiation site, and also to use our codes and reactor model to calculate these fluxes.

FLUX MEASUREMENT

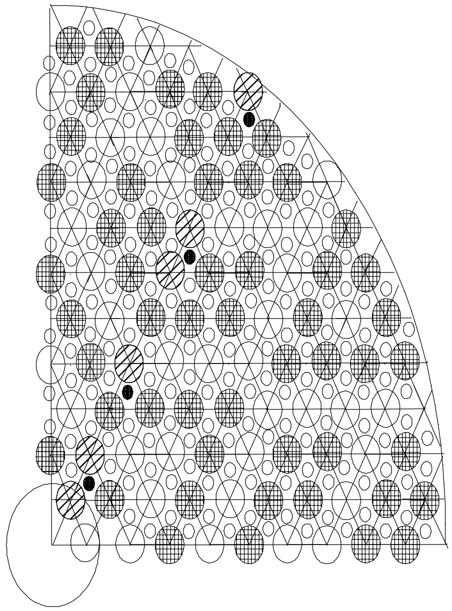
Immediately after final fuel loading, the fuel cage was removed from the pool to insert 40 cm Copper and Aluminium-Gold braided wires into four water holes (Figure 2) in the fuel cage situated at radii 17 mm(A), 39 mm(B), 71 mm(C) and 104 mm(D), at the same time two 5 cm Copper and Aluminium-Gold wires were inserted in an irradiation site in the annular beryllium reflector. After the cage was replaced in the pool, the reactor was operated at 20 W for 10 minutes.

Thermal and epithermal fluxes were measured by irradiating the copper and Aluminium-Gold wires which became radioactive by (n,γ) reactions.

$$^{63}\mathrm{Cu} + \mathrm{n} \rightarrow {}^{64}\mathrm{Cu} + \gamma \qquad T(^{64}\mathrm{Cu}) = 12.7 \text{ h}$$
$$^{197}\mathrm{Au} + \mathrm{n} \rightarrow {}^{198}\mathrm{Au} + \gamma \qquad T(^{198}\mathrm{Au}) = 2.7 \text{ d}$$

^{64}Cu emits a positron (β^+) which produces two 511 keV annihilation photon, while ^{198}Au disintegrates by emitting a particle beta (β^-) followed by a photon of energy 411. keV.

The wires were cut into 1cm pieces and, the activity of each piece was measured with a Germanium detector. The gamma-ray spectra were stored on disk and the specific activities were calculated using the program EPPA (École Polythechnique Activation Analysis)[1] .



Copper and Aluminium Gold wires

Fuel element

Water hole

Hole used in calculation

Figure 2: Positioning of Copper and Aluminium-Gold wires

Specific activation is related to the peak area by the formula :

$$A_{sp} = \frac{\mathrm{A}e^{\lambda \mathrm{t}_d}}{\varepsilon(1 - e^{-\lambda \mathrm{t}_c})W} \tag{1}$$

where

A	the peak area
W	the sample weight
ε	the detector efficiency
λ	the decay constant
t_d	the decay time
t_c	the counting time

Cu is activated mainly by the thermal neutrons while Au is activated significantly by epithermal neutrons: its main resonance is at 5.65 eV with a resonance integral of 1550 b.

The activity is the sum of the activities due to thermal and resonance neutrons[2] :

$$A = N\Phi_{th}\,\sigma_{th}S + N\Phi_r\,I_0 S = NS(\Phi_{th}\,\sigma_{th} + \Phi_r\,I_0) \tag{2}$$

where

N : number of atoms of the stable nuclide

ϕ_r : resonance neutron flux

I_0 : resonance integral

$S = (1 - e^{-\lambda t_i})$: saturation factor

For each position in the core there is one equation for Cu and one for Au. To obtain the values of thermal and epithermal flux we solved the two equations using the following data:

σ_{th}(Cu) = 4.5 b ; σ_{th}(Au) = 98.8 b ; I_o(Cu) = 4.97 b; I_o(Au) = 1550 b[3] and the relation $\Phi_{ep} = 0.055\Phi_{th}$ in the irradiation site for the new LEU core. The epithermal to thermal ratio in the irradiation site was measured during normal reactor operation at 10 kW power using the bare monitor method of reference[3] .

Results for thermal and epithermal fluxes for all position are plotted in Figure 3, all values are relative to those in the irradiation site.

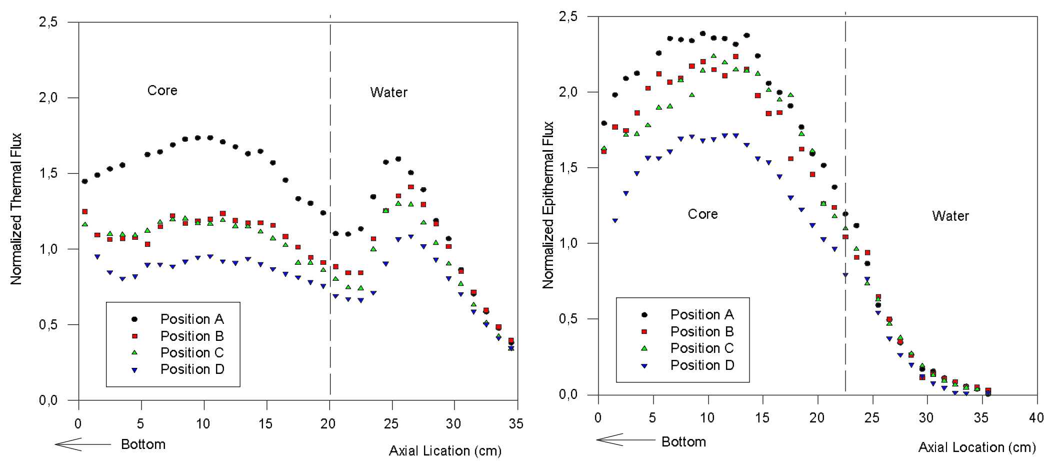


Figure 3: Measured Flux Distribution

FLUX CALCULATIONS

The transport calculations were performed with the DRAGON lattice code [5] . The diffusion code DONJON[6] was then used to compute fluxes and multiplication factors. The microscopic cross section library ENDF/BV (89 energy groups) was used to perform transport calculations. When the transport equation is solved and the thermal regions are homogenized, macroscopic properties are condensed to six energy groups, and then stored in the COMPO files. This file is directly accessed in the DONJON computation.

group	**energy inteval**
1	10^7 - 8.2085 10^5
2	8.2085 10^5 - 5.5308 10^3
3	5.5308 10^3 - 4.0
4	4.0 - 6.25 10^{-1}
5	6.25 10^{-1} - 10^{-1}
6	10^{-1} - 2.0 10^{-4}

Table 2: energy group

Transport calculation

A SLOWPOKE model[7] is used to compute neutron transport. The geometry of the LEU reactor is composed of 8 different circular regions (Figure 4)

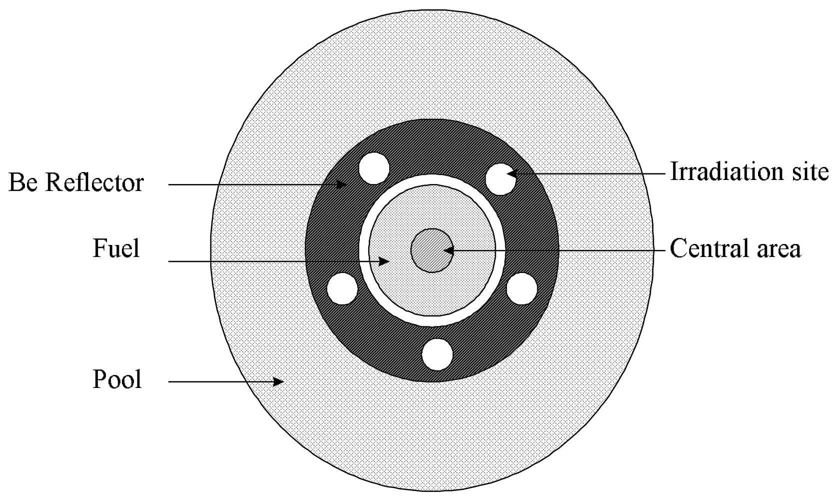


Figure 4: 2-D Transport Model

To take account of the leakage effects, a B_1 homogeneous model [8] is used in DRAGON. After having fixed an initial value of the buckling B^2 and a first estimate of the cross section, flux and the eigenvalue are calculated by solving the transport equation. Then a new value of B^2 is calculated along with a new group-dependent leakage coefficient d_g. The total cross section is increased by an amount of d_gB^2 :

$$\Sigma_{t,g}^{corr} = \Sigma_{t,g} + d_g B^2 \tag{3}$$

A new B^2 value is then calculated. The iteration end when the value of k_{eff} becomes equal to 1. The value of the B^2 is then used to calculate the diffusion coefficients. The nuclear properties are homogenized and condensed with 6 energy groups, the intervals of energy of the groups are presented in Table 1

Diffusion calculation

The COMPO file and the 3D diffusion model (Figure 5) , are used in the DONJON code to solve the multigroup diffusion equations :

$$A_{gg}\overrightarrow{\phi_g} = \sum_{\substack{h=1 \\ h\neq g}}^{G} A_{gh}\overrightarrow{\phi_h} + \frac{1}{k_{eff}}\sum_{h=1}^{G} B_{gh}\overrightarrow{\phi_h} \tag{4}$$

A_{gg} : matrices composed of the diffusion terms
A_{gh} : scattering matrices
B_{gh} : matrices containing the neutron production terms

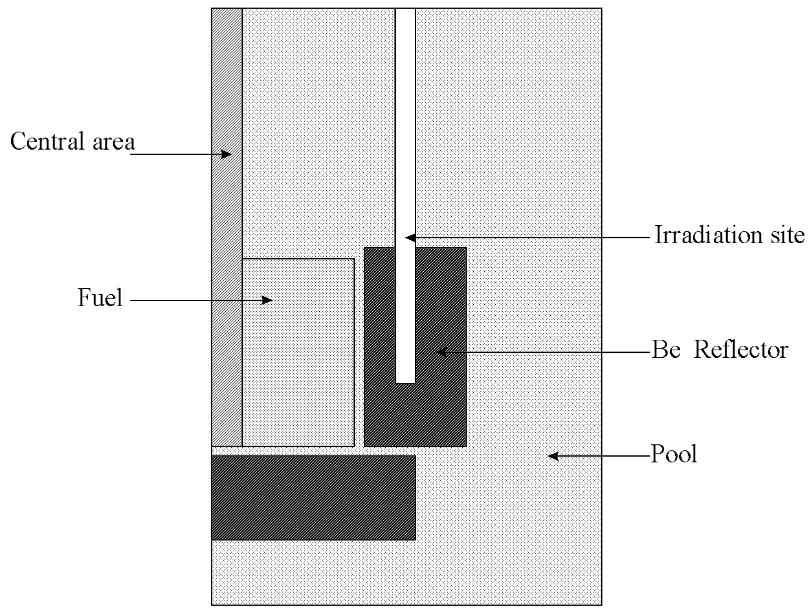


Figure 5: 3-D Diffusion Model

The value of the coefficient k_{eff}, as well as fluxes at the various points of the reactor are calculated. Using the INIRES model in the DONJON code we recovered the flux values in the holes of interest. These values are normalized to the computed values in the irradiation site. Flux values for groups 6 (thermal) and 3 (epithermal) are plotted in Figure 6.

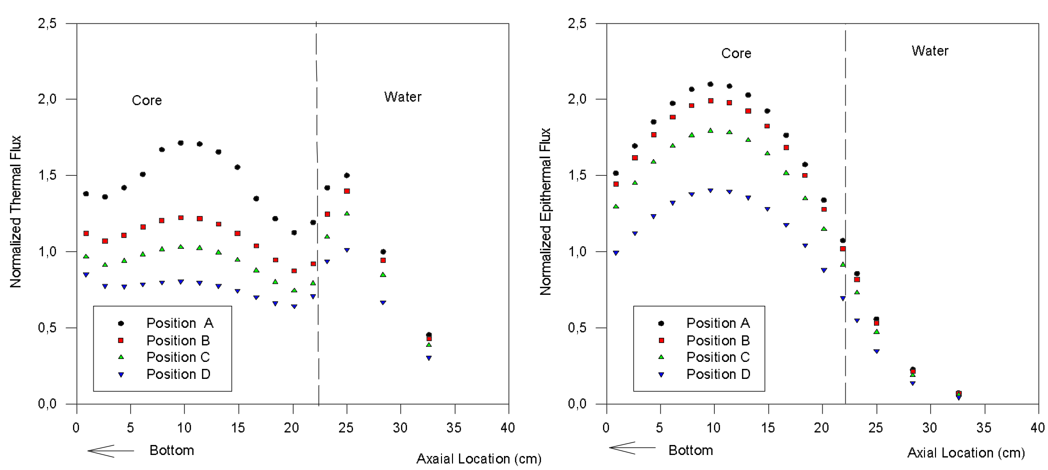


Figure 6: Calculated Flux Distribution

DISCUSSION

Measured and calculated thermal fluxes reach a maximum at the center of the reactor core, and a second maximum in the top reactor water; indeed, at the core exit the neutrons are thermalized and then absorbed by the layer of water. After reaching a maximum at the center of the core, the epithermal flux decreases as one moves away from the center either radially or axially.. Figure 7 shows that normalized calculated and measured thermal and epithermal fluxes have the same axial profile. The slight difference in the reactor core may be due to the fact that calculations and measurements are not made at the same hole (Figure 2).

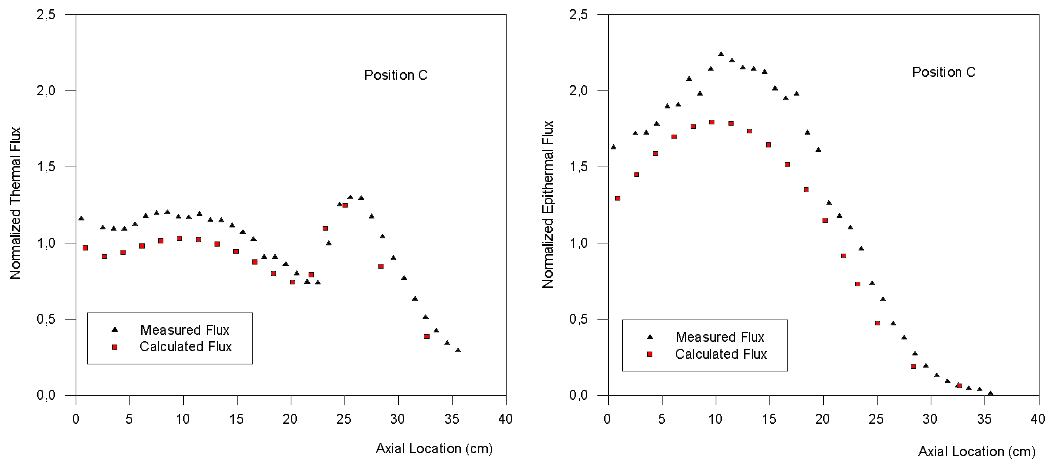


Figure 7: Measured and Calculated Flux Distribution

We have calculated the radial variation of the thermal and epithermal fluxes along a radial line containing the holes of interest Figure 2. The values were normalized to those calculated at the irradiation site. As can be seen in Figure 8, the calculations and measurements show the same trends.

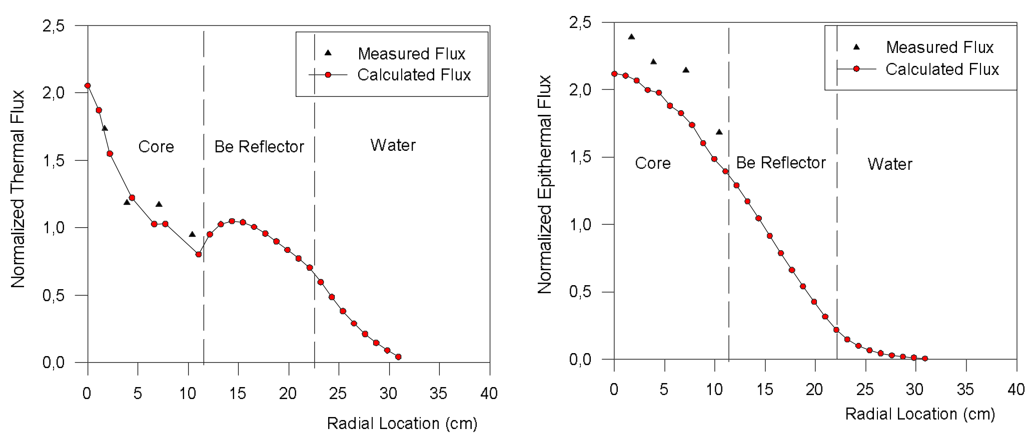


Figure 8: Measured and Calculated Radial Flux Distribution

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