MUDICO-2D- TWO-DIMENSIONAL MULTIGROUP DIFFUSION CODE FOR STATIC STUDIES AND KINETIC PARAMETERS CALCULATION IN LIGHT WATER RESEARCH REACTORS

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ABSTRACT

MUDICO-2D is intended for two-dimensional neutronic and thermalhydraulic coupling core calculations for research reactors applications. This code uses the fine mesh finite difference box scheme method and inner/outer iterative scheme to solve a set of multigroup two-dimensional, time independent neutron diffusion equations. A common first perturbation formulae's are applied to evaluate kinetic parameters which are prompt neutron lifetime and effective delayed neutron fractions. The thermalhydraulic equations are solved analytically on a representative fuel cell (cylindrical or plate geometry). The model used to evaluate coolant thermalhydraulic parameters ignores coolant boiling and assumes the coolant to go only once through the reactor core in the axial direction.
Validation results are reported for the International Atomic Energy Agency two-dimensional benchmark core research reactor.
Keywords : neutron diffusion equation, multigroup theory, perturbation theory, kinetic parameters, computer codes.

1. INTRODUCTION

In 1989, when the first Algerian Pool type Nuclear research reactor went critical, the safety laboratory has initiated a research program to develop practical computer codes specially tailored to research reactors. In the area of core calculations, most of our effort has been devoted to the development of computer codes employing multigroup diffusion techniques.. Furthermore, in response to the need shown to know the behavior of such reactor subject to any perturbation and in order to safely control it, an attempt was made to develop two-dimensional neutronic and thermalhydraulic coupling core calculations code for research reactors applications, namely MUDICO-2D [1].

Before incurring to dynamic studies of a given reactor, one must know it's kinetic parameters. The present work was therefore devoted to find practical methods of calculation after definitions of the parameters in question have been established. The parameters of interest, are reactivity, neutron generation life time and delayed neutron fraction. One way to determine these kinetic parameters is to use the well known technique of perturbation theory to express the corresponding change in multiplication factor in terms of the fluxes characterizing the unperturbed core.
The whole interest of the perturbation theory lies in the fact that if the perturbation is small enough, it's not necessary to perform a complete, new calculation for perturbed system or for each perturbation of interest. Instead, the adjoint flux is used as the weighting function to obtain the response to small perturbations. As an example of the application of the perturbation theory, small changes in reactor geometry induced by thermal expansion or composition which is a result of burn up and in neutron spectrum due to Doppler effect, occur during reactor operation, all these effects can well be found by means of perturbation theory.

The present article is organized as follow: in section 2. of this paper, the neutronic model is presented, along with a review of the multigoup diffusion equations solution method in two dimensional geometry. Section 3 consists of a presentation of the perturbation theory and how well expressed the kinetic parameters in the computational program. Section 4 present the thermalhydraulic model and the neutronic-thermalhydraulic coupling relations. The last section concludes with a comparison of the performance of the developed code MUDICO-2D with that of several existing code.

2.NEUTRONIC MODEL

2.1. MATHEMATICAL FORMULATION

Based on the concept of neutron balance, a multigroup diffusion equation is derived. The time - independent multigroup neutron diffusion equations are :

$$-\nabla D_g \cdot \nabla \phi_g + \Sigma_{Rg}\phi_g = \frac{1}{k_{eff}}\chi_g \sum_{g'=1}^{G} \nu_{g'}\Sigma_{fg'}\phi_{g'} + \sum_{g'=1}^{G}\Sigma_{sg'g}\cdot\phi_{g'} \qquad g' \neq g \qquad (2.1)$$

(g=1, 2,..., G)

where

$$\Sigma_{Rg} = \Sigma_{ag} + \delta D_g B^2 + \sum_{g'=1}^{G}\Sigma_{sgg'} \qquad (2.2)$$

$\delta = 1$ for x-y geometry
$\delta = 0$ for r-z geometry

G : Total number of energy group.
ϕ_g : Neutron flux in group g .
D_g: Diffusion coefficient in group g. [cm]
Σ_{Rg}: Macroscopic removal cross section in group g. [1/cm]
Σ_{ag}: Macroscopic absorption cross section in group g. [1/cm]
$\nu\,\Sigma_{fg}$: Macroscopic fission cross section times the average number of secondaries, in group g. [1/cm]
$\Sigma_{sg'g}$: Macroscopic scattering cross section from group g' to group g. [1/cm]
χ_g : Total fission spectrum fraction in group g.
K_{eff} : Effective multiplication factor.
B^2 : Geometrical buckling correction.

These equations are to be solved subject to the following "Neuman Dirichlet" conditions at the reactor boundary:

$$\frac{\partial\phi}{\partial r} + \gamma(r)\phi = \sigma(r) \qquad (2.3)$$

γ, σ are real constant.
r is the coordinate of the outer boundary.
If $\gamma(r) = 0$ the above conditions become "Neuman" boundary conditions,
if $\gamma(r) \rightarrow \infty$ and σ/γ = constant, it represent "Dirichlet boundary conditions,
if $\gamma(r) \neq 0$ mixed boundary conditions are applied, in this case with $\sigma = 0$,
$1/\gamma$ represents the extrapolation distance.

2.2. DERIVATION OF FINITE DIFFERENCE EQUATION:

The core may be modeled in either x-y or r-z geometry. In the x-y geometry, upon one quarter of the core can be modeled, by taking the following conditions : zero current (reflexive flux symmetry) is assumed at the top and at the left region boundary, where as zero flux condition is imposed at the right and at the bottom region boundary.
In the r-z geometry one half of the core is modeled and the regions issued represent cylindrical (toroidal) volumes.

Reduced into more general form the diffusion equation (2.1) becomes:

$$-\nabla D\,\nabla\phi + \Sigma\ \phi = S \tag{2.4}$$

The MUDICO program resolve this equation in two dimensional geometry's, X-Y and R-Z using the finite difference method. Up scatterings are taken into account. The core is divided in a lattice of mesh. Each mesh of index (i,j) is defined by its dimensions : $\Delta x_i, \Delta y_j$. The nuclear parameters D(r,z), $\sum(r,z)$ and S(r,z) are taken constant in each mesh and have the values D(i,j), $\sum(i,j)$ and S(i,j) in a mesh (i,j). Equation (2.4) is then, descritized in space as described in [2]. This procedure transforms this equation into the space linearized form which can be written as:

$$a^L_{i,j}\bar{\phi}_{i-1,j} + a^R_{i,j}\bar{\phi}_{i+1,j} + a^B_{i,j}\bar{\phi}_{i,j-1} + a^T_{i,j}\bar{\phi}_{1,j+1} + a^C_{i,j}\bar{\phi}_{i,j} = S_{i,j} \tag{2.5}$$

Where :

$$a^L_{i,j} = \frac{-2\Delta y_j}{\dfrac{\Delta x_{i-1}}{D_{i-1,j}} + \dfrac{\Delta x_i}{D_{i,j}}} \tag{2.6}$$

$$a^R_{i,j} = \frac{-2\Delta y_j}{\dfrac{\Delta x_{i+1}}{D_{i+1,j}} + \dfrac{\Delta x_i}{D_{i,j}}} \tag{2.7}$$

$$a^B_{i,j} = \frac{-2\Delta x_i}{\dfrac{\Delta x_{j-1}}{D_{i,j-1}} + \dfrac{\Delta y_i}{D_{i,j}}} \tag{2.8}$$

$$a^T_{i,j} = \frac{-2\Delta x_i}{\dfrac{\Delta x_{j+1}}{D_{i,j+1}} + \dfrac{\Delta y_i}{D_{i,j}}} \tag{2.9}$$

$$a^C_{i,j} = \Sigma_{i,j}\Delta x_i \Delta y_j - \left(a^L_{i,j} + a^R_{i,j} + a^B_{i,j} + a^T_{i,j}\right) \tag{2.10}$$

Equations (2.5) for all the grids may be expressed in the matrix form:

$$\mathrm{A}\Phi = \mathrm{S} \tag{2.11}$$

where Φ and S represent the elements, ϕ_{ij} and S_{ij} respectively for each energy group. For a mr x mz mesh system, they are a mr x mz vectors. A represent the elements a_{ij}.

For the surfaces of the mesh box which are on the outer boundary, the boundary condition given by equation (2.3) is applied to evaluate the current across that surface.

2.3. METHOD OF SOLUTION

The diffusion equation are solved in the finite difference approximation using the standard inner / outer scheme[3] The computation is accelerated by the over relaxation factor ω which is usually greater then one. Instead using the flux resulted from the straight forward method, we apply the following procedure for a given iteration n to evaluate the extrapolated flux ϕ_g^{ext} [4]:

$$\phi_g^{ext} = \omega \phi_g^{n} + (1 - \omega)\phi_g^{n-1} \tag{2.12}$$

In our program an atempt was made to calculate the optimal value of the relaxation parameter according to Chebyshev acceleration[5].

$$\omega^{(0)} = 1 \tag{2.13}$$
$$\omega^{1/2} = 1/(1-\rho^2_{Jacobi}/2) \tag{2.14}$$
$$\omega^{n+1/2} = 1/(1- \rho^2_{Jacobi}\, \omega^{(n)}/4); \quad n = 1/2, 1, ...\infty. \tag{2.15}$$

Where $1 \le \omega \le 2$.

Iterations are stopped when both the K_{eff} and the point wise flux convergence between the two succesive iterations have been reached.

- Convergence criterion for inner iterations:

$$\left|\frac{\phi_g^n - \phi_g^{n-1}}{\phi_g^{n-1}}\right| \lhd \varepsilon_1 \tag{2.16}$$

- Convergence criterion for outer iterations:

$$\left|\frac{k_{eff}^n - k_{eff}^{n-1}}{k_{eff}^{n-1}}\right| \lhd \varepsilon_2 \tag{2.17}$$

2.4. THERMALHYDRAULIC FEEDBACK PARAMETERS:

For a global core flux distribution calculations the program includes a rather simple Thermalhydraulics module to calculate the Temperature distributions. To take into account the fuel and moderator temperature effects in order to compensate feedback effects ($\frac{\partial \Sigma}{\partial T}$), a correction to the cross section is made.

The principle of the method is to reevaluate the cross section du to the changes in core parameters (i.e. fuel and moderator temperature) by assuming a linear model dependancy. These coefficients are given on input of the program for each type of cross section.

Such variations are described by the following equations:

$$\Sigma_a^g(T_F, T_M) = \Sigma_a^g(T_{F0}, T_{M0}) + A_a^F(T_F - T_{F0}) + A_a^M(T_M - T_{M0}) \tag{2.18}$$
$$\Sigma_s^g(T_F, T_M) = \Sigma_s^g(T_{F0}, T_{M0}) + A_s^F(T_F - T_{F0}) + A_s^M(T_M - T_{M0}) \tag{2.19}$$
$$\Sigma_f^g(T_F) = \Sigma_f^g(T_{F0}, T_{M0}) + A_f^F(T_F - T_{F0}) \tag{2.20}$$
$$D_g(T_M) = D_g(T_{M0}) + A(T_M - T_{M0}) \tag{2.21}$$

T_F, T_M are average fuel and moderator temperature [°C], T_{F0}, T_{M0} are reference temperatures [°C]. and A^F, A^M are fuel and moderator temperature coefficients respectively in [cm^{-1}/°C].

2.4. POWER PEAKING FACTORS

One of the basic parameters of the mudico-2D core calculations are the power peaking factors. In these 2D calculations without control absorber, the radical power peaking factor is defined as the ratio of the average power in the specific element to the average power in all fuel elements of the whole core.
The local power peaking factor is defined as the ratio of the maximum power in the specified element to the average power in this element.

3. PERTURBATION THEORY:

The kinetics parameters are calculated using first order perturbation theory as used in the EXTERMINATOR code[5] The details of this method are given in reference [6]. We resume only the results as follow:

- **Prompt neutron life time:**

$$l=\frac{\sum_K 1/v_K \sum_I \sum_J \Phi^*(I,J,K)\ \Phi(I,J,K)\ \Delta V(I,J)}{E} \qquad (3.1)$$

where:

$$E=\frac{1}{K_{eff}}\left[\sum_I \sum_J \sum_K \chi(K)\Phi^*(I,J,K) * \sum_K \nu\Sigma_f(I,J,K)\Phi(I,J,K)\right] \qquad (3.2)$$

$\Phi^*(I,J,K)$ is the adjoint flux of energy group K at mesh point (I,J), $\Phi(I,J,K)$ is the direct flux of energy group K at mesh point (I,J), $\chi(K)$ is the delayed neutron fraction in energy group K, v_K is the neutron mean velocity in the energy group K, $\Delta V(I,J)$ is volume of the mesh (I,J) and K_{eff} is the effective multiplication factor.

- **Prompt neutron generation time:**

$$\Lambda=\frac{l}{K_{eff}} \qquad (3.3)$$

- **Effective delayed neutron fraction:**

$$\beta_{i,eff}=\frac{\beta_i \sum_K \chi_{id}(K)\ \ H(K)}{E \quad K_{eff}} \qquad i=1, ..., I$$

$$\beta_{eff}=\sum_i \beta_{i,eff} \qquad (3.4)$$

where:

$$H(K)=\left[\sum_I \sum_J \Delta V(I,J)\Phi^*(I,J,K) * \sum_K \nu\Sigma_f(I,J,K)\Phi(I,J,K)\right] \qquad (3.5)$$

$\chi_{id}(K)$ is the delayed neutron fraction of group i in the energy group K.

4. THERMAL-HYDRAULIC MODEL

4.1. THE HEAT CONDUCTION MODEL

The heat conduction model considers both cylindrical and slab geometry's. A cell consisting of a succession of slabs, for rectangular element, or annulus, for cylindrical element, representing fuel, clad and moderator model the fuel element. This model preserves all material volume and fuel element high. Heat conduction in the solid part of the cell (fuel and clad) is neglected on axial direction, it is considered only on the cell thickness represented by the dimension (r) in both geometry's. The axial transfer of the heat is assured by the coolant. Then, the heat conduction equation for the fuel zone is given by :

$$\nabla_r K_b\left(\nabla_r T(r)\right) + Q(r) = 0 \qquad (4.1)$$

and for the clad by :

$$\nabla_r K_c\left(\nabla_r T(r)\right) = 0 \qquad (4.2)$$

T(r) is the temperature [°C] on the abscise r [m], K_b and K_c are the heat conduction coefficients of fuel and clad respectively [w/m/°C] and Q(r) is the specific power [w/m^3] on the abscise r.

The boundary condition on the cell axis is :

$$\frac{\partial T(r)}{\partial r} = 0 \qquad \text{on } r = 0 \qquad (4.3)$$

The axial transport of the heat by the coolant is expressed by the heat conservation conditions on the interface cladding coolant generating the heat balance equations [7]:

$$AH_w\left(T_c - T_w\right) = Sh_z Q \qquad (4.4.a)$$

$$GC_w\left(T_a - T_e\right) = Sh_z Q \qquad (4.4.b)$$

where A is the heat transfer aera [m^2], S the section of the cooling channel [m^2], V_w is the coolant volume belonging the section h_z [m^2], G is the flow rate [Kg/S], C_w is the specific heat [J/Kg/°C] of the coolant and T_e and T_s are the inlet and outlet temperatures [°C] of coolant respectively, related to T_w by :

$$T_w = \frac{1}{2}\left(T_s + T_e\right) \qquad (4.5)$$

To take into account the axial transport of the heat by the coolant represented by the equations (4.4), the cell is divided in Z axial intervals of height h_z. The heat conduction equations (4.1) and (4.2) are, then, resolved analytically for each region of height h_z.to determine the temperature distribution in the solid zone thickness (fuel and cladding) of the cell.

4.2- PHYSICAL PROPRIETIES OF THE COOLANT:

The heat transfer coefficient, H_w, is calculated from the DITTUS-BOELTER relation.[7] We assume a single-phase flow (no boiling occurs), which can be laminar or turbulent. The water coolant parameters such as density R_w, specific heat C_w, conductivity K_w and kinematics viscosity μ are interpolated from steam-water tables.[8]
Two effects are taken into account for the evaluation of the pressure variation in a cooling channel relating the inlet pressure and the outlet pressure :the gravity pressure looses and the friction effect.

4.3. NEUTRONICS-THERMOHYDRAULIC COUPLING:

Two links exist between the thermohydraulic and neutronic models:

- Temperature coefficients of the group constants: As described in section 2.4, these coefficients which represent the influence of the temperature distribution on the nuclear constants and so on the neutronic calculation, define the thermohydraulics feed-backs.

- Energy generated by the fuel E : related by a geometric coefficient to the specific heat Q used by the equation (4.1) of the thermohydraulic model, can be calculated by the flux given by the neutronic calculation, as :

$$E(i,j) = Q_f \sum_{g=1}^{NG} \Sigma_{fg}(i,j)\Phi_g(i,j) \qquad (4.6)$$

where Q_f is the energy released by fission.

5.VALIDATIONS

Validation results are reported for the International Atomic Energy Agency two-dimensional 10Mw Benchmark HEU and LEU core research reactor [9].

The MUDICO-2D diffusion code is used to perform steady state core calculation to calculate intrinsic configuration core parameter such as effective multiplication factor, power and neutron flux distributions, feedback coefficients (Doppler, fuel temperature, moderator temperature and density coefficients) are then obtained. Perturbation calculations module of the code is used to evaluate kinetic parameters, like effective delayed neutron fraction and prompt neutron life time.

The input nuclear data are obtained by performing cell calculations using the WIMS-D/4 [10] code for different identified and homogenized unit cell type of fuel and non-fuel elements present in the two cores. A set of macroscopic cross sections are then generated in five energy groups for different initial U-235 burn-up of fuel elements (5%, 25% and 45%), fuel and moderator temperatures (20 ^{o}C, 38 ^{o}C, 50 ^{o}C,75 ^{o}C,100 ^{o}C and 200^{o}C) and change of coolant density (5 , 10 and 20%) [11].

5.1. REACTOR CORE DESCRIPTION:

IAEA Tecdoc-233 [9] defined generic 10 Mw light water moderated research reactor for a series of benchmark calculations of specified reactors parameters with high and low enrichment cores. The core is an arrangement of 5x6 elements (see figure 2) containing 21 standard MTR-type fuel elements of 23 plates each and 4 control fuel element with 17 plates. The core is reflected by graphite on two faces and is surrounded by water. One flux trap is located in the center of the core in order to compute more realistic radial and local power peaking factors in the surrounding fuel elements.

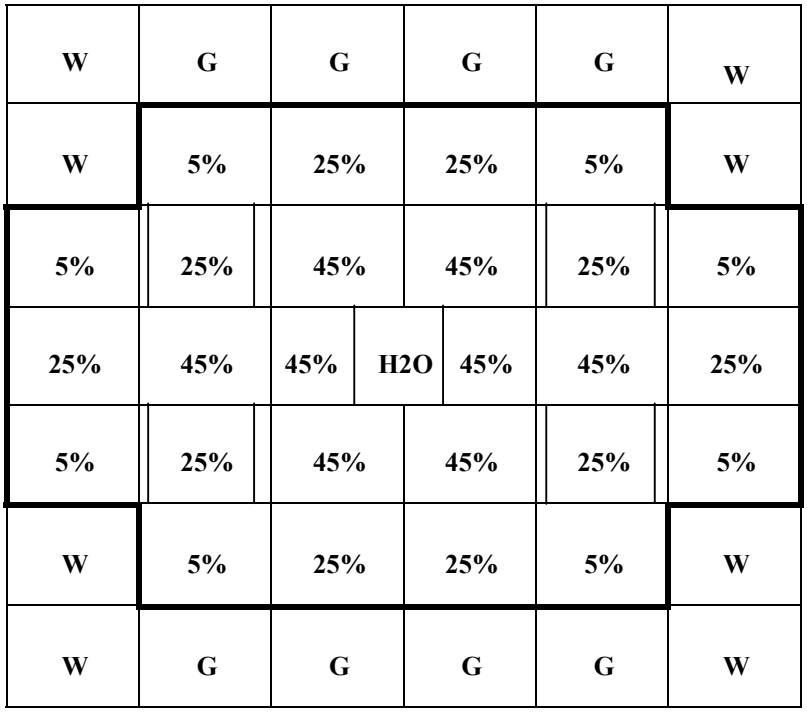

Figure 2: BOL Core Showing % U-235 Burn-up in each Fuel Element and Fuel Element Identification
W – Water, G – Graphite, SFE (CFE) - Standard (Control) Fuel Element

5.2. RESULTS :

5.2.1. EFFECTIVE MULTIPLICATION FACTOR, PROMPT NEUTRON GENERATION TIME AND DELAYED NEUTRON FRACTION:

MUDICO-2D calculations, using the cross section library issued from WIMS-D4 cell calculations and interface collapsing, are compared to those given by ANL-USA and EIR-Switzerland laboratories. The results are resumed in Table 1, 2 and 3 below.

Table 1: Effective multiplication factor

Fuel Element	**K-effectif**		
	LSN	ANL	EIR
HEU (93%)	1.03402	1.02839	1.02443
LEU (20%)	1.02449	1.01796	1.02462

Table 2: Delayed neutron Dependant Parameters

Fuel Element	**β x 10^{-2}**			**Λ, μ sec**		
	LSN	ANL	EIR	LSN	ANL	EIR
HEU (93%)	0.776	0.761	0.778	54.1	56.0	58.8
LEU (20%)	0.739	0.728	0.736	40.5	43.7	44.8

Table 3: Delayed neutron Group Parameters for HEU and LEU cores

	HEU (93 %) BOL core		LEU (20 %) BOL core	
Delayed Neutron Group	β_i		β_i	
	LSN	ANL	LSN	ANL
1	2.95245 E-04	2.9648E-04	2.81186 E-04	2.7926 E-04
2	1.65604 E-03	1.5822 E-03	1.57719 E-03	1.5178 E-03
3	1.44995 E-03	1.4352 E-03	1.38091 E-03	1.3731 E-03
4	3.16507 E-03	3.1144 E-03	3.01436 E-03	2.9627 E-03
5	9.95084 E-04	9.7969 E-04	9.47701 E-04	9.4536 E-04
6	2.01690 E-04	1.9914 E-4	1.92086 E-04	1.9716 E-4

5.2.2. ISOTHERMAL REACTIVITY FEEDBACK COEFFICIENTS:

5.2.2.1. CHANGE OF FUEL TEMPERATURE ONLY:

Values of k_{eff} were computed for fuel temperature of 38 ^oC, 50 ^oC, 75 ^oC, 100 ^oC and 200 ^oC. The obtained values are resumed on Table 4 and compared to those present by other laboratories.
The reactivity variation is calculated to reference reactivity at 20 ^oC by:

Table 4: Reactivity coefficient for change of fuel temperature only

	k_{eff}					
Fuel Temperature (^oC)	**HEU**			**LEU**		
	LSN	ANL	EIR	LSN	ANL	EIR
20	1.034028	1.0284	1.026934	1.02495	1.01797	1.026385
38	1.034023	1.028389	1.02693	1.02449	1.01748	1.025936
50	1.03402	1.028382	1.026928	1.02421	1.01716	1.025664
75	1.034017	1.028371	1.026923	1.02362	1.0165	1.025082
100	1.034014	1.028364	1.026917	1.02304	1.01586	1.024513
200	1.033998	1.028343	1.026895	1.02094	1.01345	1.022412

$$\rho = \frac{k_{eff} - 1}{k_{eff}}$$

$$\Delta\rho = \rho(T) - \rho(20\ ^oC)$$

The best polynomial fit of the reactivity variation $\Delta\rho$ (in pcm) for HEU and LEU case is given respectively by equations below:

$$\Delta\rho\ (pcm) = 0.1289 - 0.0148\ T_F$$

$$\Delta\rho\ (pcm) = 35.992 - 2.117\ T_F$$

5.2.2.2 CHANGE OF WATER TEMPERATURE ONLY:

Values of k_{eff} were computed for water temperature of 38 ^{o}C, 50 ^{o}C, 75 ^{o}C and 100 ^{o}C. Cross sections library for these different temperatures are obtained from WIMS-D4 cell calculation, and is used by MUDICO-2D to calculate k_{eff} corresponding to these temperatures. The results are compared to other laboratories as shown in Table 5.

Table 5: Reactivity coefficient for change of water temperature only

	k-effectif					
Water Temperature (^{o}C)	**HEU**			**LEU**		
	LSN	ANL	EIR	LSN	ANL	EIR
20	1.034028	1.02878	1.026918	1.02495	1.01823	1.026385
38	1.032093	1.02651	1.024648	1.02355	1.0167	1.024774
50	1.03086	1.02501	1.023185	1.02266	1.0157	1.023746
75	1.028405	1.02193	1.02019	1.02094	1.01366	1.021659
100	1.026114	1.01893	1.017331	1.01937	1.01168	1.019688

The best polynomial fit of results obtained by our laboratory leads to the equations below:

$$\textbf{HEU :} \quad \Delta\rho \text{ (pcm)} = 176.039 - 9.305\, T_w$$
$$\textbf{LEU :} \quad \Delta\rho \text{ (pcm)} = 122.985 - 6.658\, T_w$$

5.2.2.3. CHANGE OF WATER DENSITY ONLY:

Values of k_{eff} are computed for water densities of 0.993, 0.988, 0.975 and 0.958 g/cm^3, witch correspond to water temperature of 38 ^{o}C, 50 ^{o}C, 75 ^{o}C, 100 ^{o}C. These calculations are done for two cases:

- Change of water concentration in both moderator and extra-region of the unit cell;
- Change of water concentration in moderator region only.

The results are given on Table 6.

Table 6: Reactivity coefficient for change of water density only

	$\underline{k}_{eff}$					
Water Temperature (^{o}C)	**HEU**			**LEU**		
	LSN	ANL	EIR	LSN	ANL	EIR
20	1.03363	1.02794	1.026918	1.02344	1.0174	1.015604
38	1.03249	1.02659	1.025535	1.02221	1.01585	1.014023
50	1.0315	1.02528	1.024222	1.02197	1.01433	1.012586
75	1.02877	1.0218	1.020703	1.01794	1.01028	1.008642
100	1.02513	1.01709	1.015974	1.01381	1.00487	1.003375

The best polynomial fit which can represent adequately this variation is:

$$\textbf{HEU :} \quad \Delta\rho \text{ (pcm)} = 641.3 - 1.9\, T_w - 0.067\, T_w^2$$
$$\textbf{LEU :} \quad \Delta\rho \text{ (pcm)} = 81.3 - 2.5\, T_w - 0.076\, T_w^2$$
$$\textbf{HEU :} \quad \Delta\rho \text{ (pcm)} = -3.81123 - 187.949\, \alpha_v - 2.8025\, \alpha_v^2$$
$$\textbf{LEU :} \quad \Delta\rho \text{ (pcm)} = 20.5323 - 222.624\, \alpha_v - 2.4118\, \alpha_v^2$$

6. CONCLUSION

The results obtained for the International Atomic Energy Agency two-dimensional 10Mw Benchmark HEU and LEU core research reactor shows good agreement compared to work performed by other laboratories. The MUDICO-2D neutronic model has shown a good capability to calculate kinetics parameters for given core configuration. Further test cases should be performed to assess the capacity of this code in neutronics-thermahydraulic calculations. Furthermore, a new version of the code with 3-D capabilities .is under development.

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