COMPUTER CODE PACKAGE *COMPACK-LHW* FOR M.T.R. RESEARCH REACTOR CORE CALCULATIONS

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ABSTRACT

This work investigates the capability of the developed computer code package, namely **COMPACK-LHW**, to perform accurately nuclear safety analysis in water cooled research reactor cores. This package, is provided with a preliminary version of a Graphical User Interface (GUI) to assist users in performing automatically all basic calculations steps of nuclear research reactors.

Based mainly on both **WIMS-D/4** and **MUDICO-2D/3D** modules, for the static part of the criticality neutronic analysis, and using an updated 69-group WIMS-D/4 cross sections library, an International Atomic Energy Agency two-dimensional 10 Mw Benchmark LEU (Low Enriched Uranium) and HEU (High Enriched Uranium) cores, has been selected as typical standard of a large class of M.T.R. research reactors to validate the whole package. The results obtained were compared to those given by selected laboratories.

1. INTRODUCTION

Reactor core calculations play a preponderant role in nuclear safety analysis, since it leads to determine parameters of great importance in reactor safety exploitation. The modern and actual nuclear reactor core calculations, applied by several specialized laboratories in the world, usually require practical and sophisticated code system to perform them. In this light, an attempt was made by the local research staff of the Algerian safety laboratory, under an IAEA Cooperation Research Project [1], to develop a computer code package namely ***COMPACK-LHW*** [2], specially tailored for ***M.T.R.*** research reactor applications. An extension of the developed package to Heavy Water moderated reactors capabilities has been included, and its related validation procedure is still in progress.

The package is based on both developed and adapted computer codes, an effort was done to automate the use of these codes by creating suitable relationships between them in appropriate manner as shown in figure 1. It includes, **WIMS-D/4**[3] for cell calculations, **MUDICO-2D/3D**[4] for diffusion and perturbation calculations, **RETRAC-PC** [5],[6] and **CRTA** [7], [8] for transient and accident analysis. Auxiliary programs for generating special libraries are also provided.

An additional effort was also devoted to provide the package with a preliminary version of a Graphical User Interface **(GUI)** which might assist users in performing automatically all basic calculations steps of nuclear research reactors.

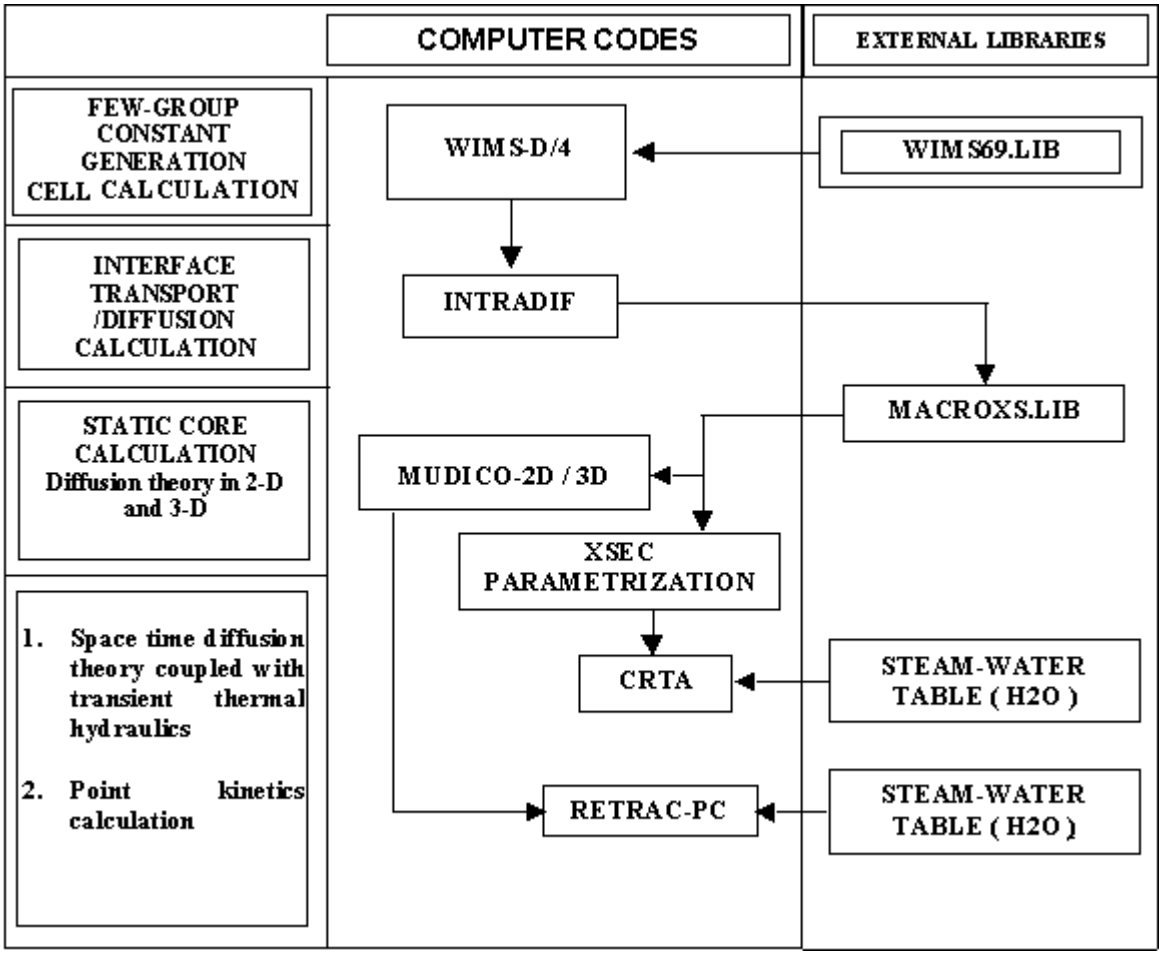


Figure 1: Flow chart of the computer code package *COMPACK-LHW*

2. DESCRIPTION OF THE VALIDATION PROCEDURE

2.1. INTRODUCTION

A safety related benchmark problem for an idealized light water, pool type reactor was specified in order to assess computational methods used in the developed code system namely COMPACK-LHW. The IAEA 10 Mw light water moderated research reactor, was defined for a series of benchmark calculations of specified reactors parameters with high (HEU-93%) and low (LEU- 20%) enrichment cores.

The core is an arrangement of 5x6 elements containing 21 standard MTR-type fuel elements of 23 plates each and 4 control fuel element with 17 plates. It 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. Detailed specifications for theses reactors HEU and LEU are given in ref. [9].

The main parameters which have been calculated for the light water reactor benchmark problem with HEU and LEU fuels are:

- Prompt neutron generation times;
- Delayed neutron fractions;
- Isothermal temperature and void reactivity coefficients;
- Radial and local power peaking factors;
- Power defect;

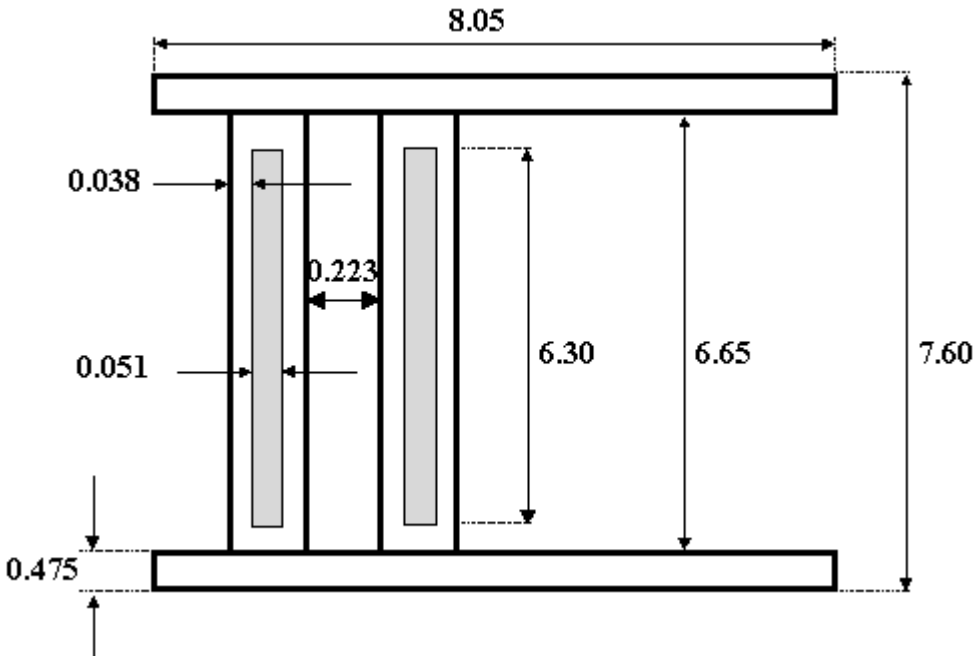
2.2. METHODS USED FOR STATIC CALCULATIONS:

The validation procedure for the static part of the criticality neutronic analysis, is set in two main steps and is based mainly on both ***WIMS-D/4*** provided with an updated 69-group cross sections library and ***MUDICO-2D*** modules.

In the first step, cell calculations were performed using the WIMS-D/4 code for different identified and homogenized unit cell type of fuel and non-fuel elements present in the core. The figure 2 shows a general view of the standard (SFE) and control fuel elements (CFE). The homogenized unit cell adopted for the whole static calculations is represented in figure 3. The model used to represent the standard and control fuel element are shown in figures 4 and 5 respectively.

A set of macroscopic cross sections were then generated in five energy groups (kindly see the group structure given on Table 1) for:

- Different initial U-235 burn-up of fuel elements (5%, 25% and 45%);
- Change of fuel temperature only (20^{o}C, 38^{o}C, 50^{o}C, 75^{o}C, 100^{o}C, 200^{o}C);
- Change of water temperature and density only: Values of k_{eff} were computed for both water temperature of 38^{o}C, 50^{o}C, 75^{o}C, 100^{o}C, and water densities of 0.993, 0.988, 0.975 and 0.958 g/cm^{3} respectively.



Standard Element : 23 fuel plates.
Control Element : 17 fuel plates + 4 Al plates.

Figure 2 : View of 10 MW-Benchmark MTR fuel element (SFE and CFE) (all dimensions are in cm).

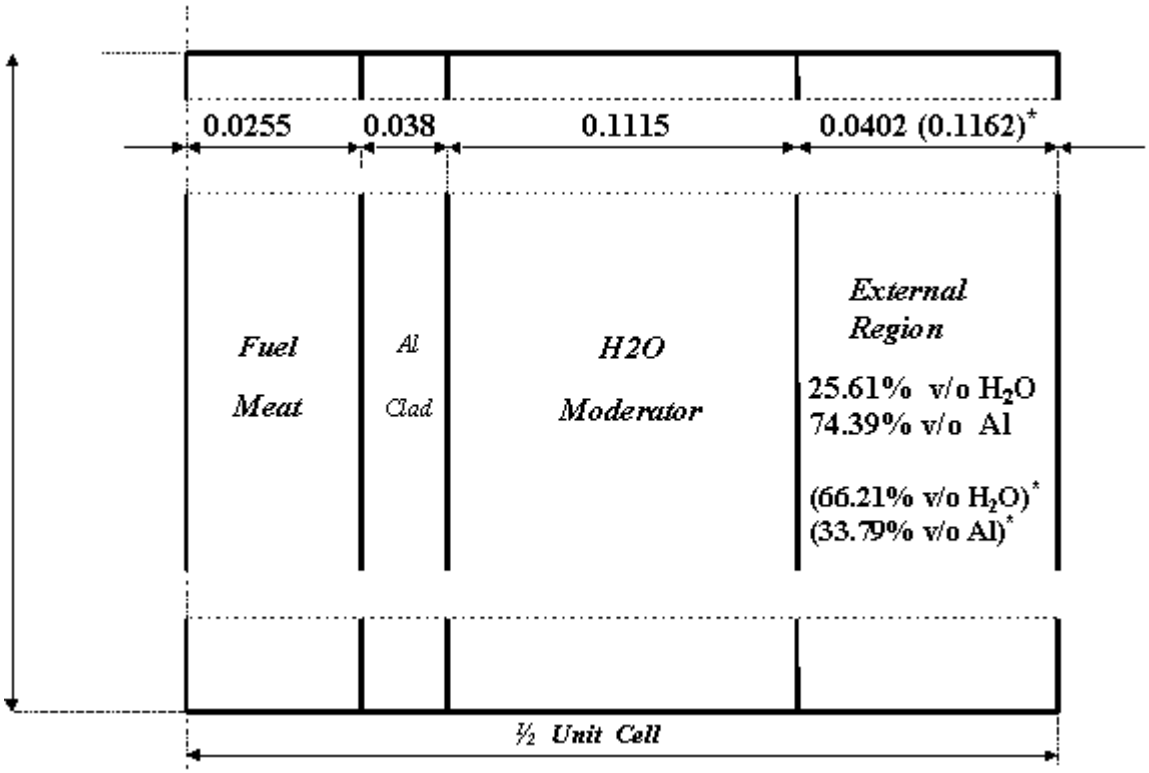


Figure 3: Geometry of unit cell used for Calculation of SFE and CFE cross sections. (dimensions are in cm). * (Values used for CFE)

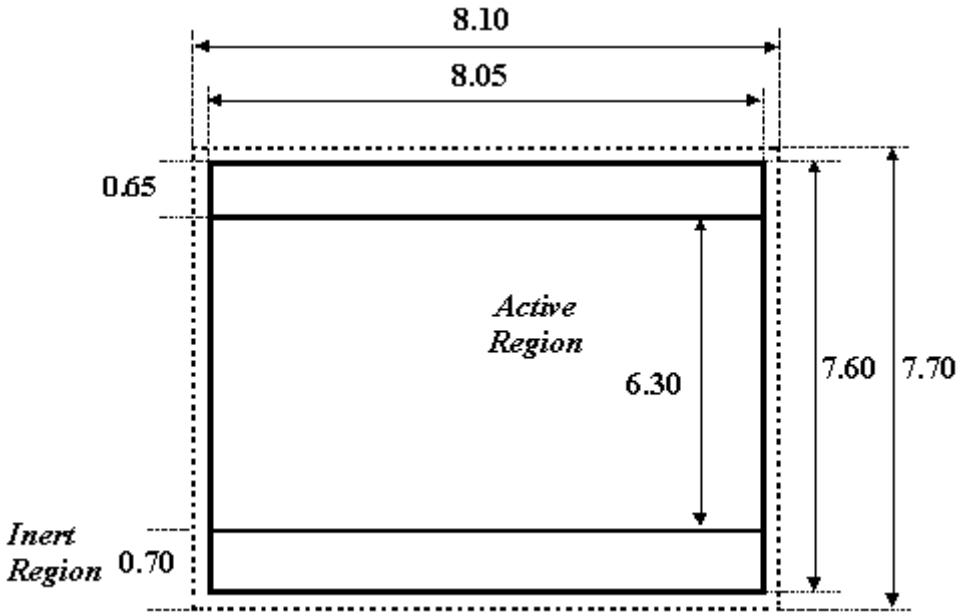


Figure 4: Model used for Standard Fuel Element SFE (dimensions are in cm).

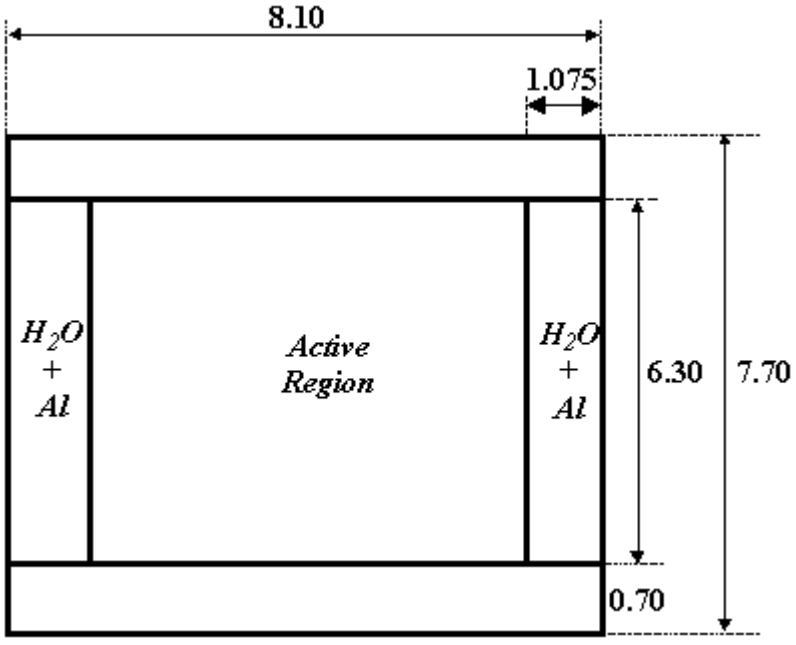


Figure 5: Model used for Control Fuel Element (CFE) without absorber (dimensions are in cm).

Table 1: Energy boundary for collapsed calculation

Group Energy	Energy range	WIMS - Groups
1	10.0 MeV – 0.821 MeV	1 – 5
2	821 KeV – 5.530 KeV	6 – 15
3	5.530 KeV – 0.625 eV	16 – 45
4	0.625 eV – 0.080 eV	46 – 55
5	0.080 eV – 0 eV	56 – 69

In the second step, global core calculations with the two dimensional multigroupe diffusion code MUDICO-2D were performed using symmetry in one-quarter of the reactor. The control rod channels were represented as H_2O + Al zones at both sides of the corresponding fuel elements. A total of 19 and 16 mesh intervals was used in the x and y direction respectively. The fluxes were normalized to a power of 2.5 Mw in the quarter core. The axial direction was represented with buckling of 1.70873E-3 corresponding to a chopped cosine axial flux distribution with a 8 cm reflector savings ($H_{extr.}$ = 76.0 cm).

2.3. RESULTS OF STATIC CALCULATIONS:

2.3.1. Basic kinetic Parameters

Perturbation calculations module of the MUDICO-2D code is used to evaluate basic kinetic parameters, namely effective delayed neutron fraction (β_{eff}) and prompt neutron generation time (Λ).The results obtained are compared to those given by ANL-USA and EIR-Switzerland laboratories in Table 2 .

Table 2: Basic kinetic 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

*LSN: Algerian Safety Analysis Laboratory

**ALN: Argonne National Laboratory

***EIR: Switzerland Laboratory

2.3.2. Isothermal reactivity feedback coefficients

- **Change of fuel temperature only:**

Values of k_{eff} were computed for fuel temperature of 38°C, 50°C, 75°C, 100°C and 200°C. The obtained values are resumed on Table 4 and compared to those given by other laboratories. The variations of reactivity to reference at 20°C are given by equation (1) and (2) and are presented on figure 6 and 7 for HEU and LEU respectively.

$$\rho = \frac{K_{eff} - 1}{K_{eff}} \quad (1)$$

and

$$\Delta\rho = \rho(T) - \rho(20°C) \quad (2)$$

The best polynomial fit of the reactivity variation $\Delta\rho$ (in pcm) for HEU and LEU case is given respectively by equation (3) and (4):

$$\Delta\rho \text{ (pcm)} = 0.1289 - 0.0148\ T_F \quad (3)$$

$$\Delta\rho \text{ (pcm)} = 35.992 - 2.117\ T_F \quad (4)$$

- **Change of water temperature only:**

Values of k_{eff} were computed for water temperature of 38°C, 50°C, 75°C and 100°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 figures 8 and 9 shows the variation of reactivity for HEU and LEU cores.

The best polynomial fit obtained by our laboratory for the variation of reactivity registered according to the change of water temperature leads to equations (5) and (6) below for both HEU an LEU cases, respectively :

$$\Delta\rho \text{ (pcm)} = 176.039 - 9.305\ T_w \quad (5)$$

$$\Delta\rho \text{ (pcm)} = 122.985 - 6.658\ T_w \quad (6)$$

- **Change of Water Density only and Core Void Coefficients:**

Values of k_{eff} are computed for water densities of 0.993, 0.988, 0.975 and 0.958 g/cm^3, which correspond to water temperature of 38°C, 50°C, 75°C and 100°C, and for densities of 0.9 and 0.8, which are void conditions. The reactivity changes are calculated to reference reactivity at 20°C, and presented in figures 10 and 11. The best polynomial fit which can represent this variation for the HEU and LEU cores are given below in equation (7) and (8) respectively:

$$\Delta\rho \text{ (pcm)} = 641.346 - 1.891\ T_w - 0.067\ T^2_w \quad (7)$$

$$\Delta\rho \text{ (pcm)} = 81.3 - 2.5\ T_w - 0.0755\ T^2_w \quad (8)$$

To obtain core void coefficients, the water concentration change was considered :

1. In both moderator and extra-region of the unit cell;
2. In moderator region only.

The reactivity changes are calculated to reference reactivity at 20°C, and presented in figures 12 and 13. The first case, which it seems to be a realistic one, is represented by equation (9) and (10) and the best polynomial fit obtained for this variation for the HEU and LEU cores are given below:

$$\Delta\rho \text{ (pcm)} = -3.81123 - 187.949\ \alpha_v - 2.8025\ \alpha_v^2 \quad (9)$$

$$\Delta\rho \text{ (pcm)} = 20.5323 - 222.624\ \alpha_v - 2.4118\ \alpha_v^2 \quad (10)$$

The global results obtained for the temperature coefficient of reactivity for both HEU and LEU cores over the temperature range of 38 – 50 °C are summarized in table 3. It's also shown in this table, the core void coefficient of reactivity due to change in water density from 0.958 – 0.90 g/cm^3 and from 0.998 – 0.958 g/cm^3. One can note, that except for fuel temperature coefficient for LEU case, the remaining results are slightly underestimated comparatively to those given by chosen Laboratories. However, theses coefficients can be assumed as conservative values when they are combined in the analyses of transients reactivity insertion.

Table 3 : Isothermal Reactivity Coefficients

Effect	Fuel	LSN	ANL	EIR
Temperature Range: 38 – 50 °C ($-\Delta\rho \times 10^{-5}$ /°C)				
Fuel Temperature only: α_{Tf}	HEU	0.023	0.055	0.02
	LEU	2.22	2.58	2.16
Water Temperature only: α_{Tw}	HEU	9.7	11.9	11.6
	LEU	7.1	8.1	8.2
Water Density only: α_{Dw}	HEU	7.8	10.4	10.4
	LEU	10.5	12.3	11.7
$\alpha_{Tw} + \alpha_{Dw}$	HEU	18.5	22.3	22.0
	LEU	17.6	20.4	19.9
Water Density Range: 0.958 – 0.90 g/cm^3 ($-\Delta\rho/\Delta\rho_w$)				
Voids or Water Density : α_v	HEU	0.228	0.296	0.300
	LEU	0.254	0.344	0.337
Water Density Range: 0.998 – 0.958 g/cm^3 ($-\Delta\rho/\Delta\rho_w$)				
Voids or Water Density : α_v	HEU	0.200	0.258	0.261
	LEU	0.231	0.305	0.299

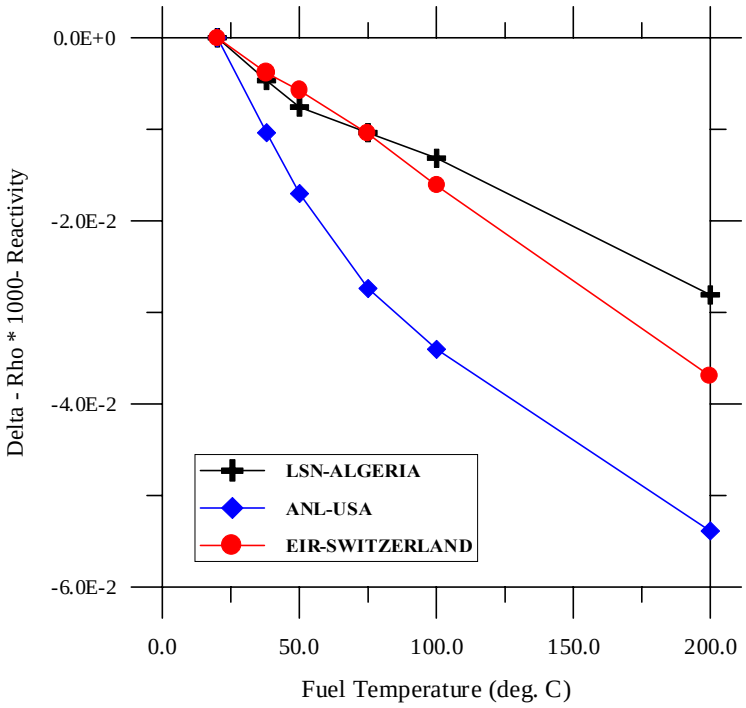


Figure 6 : Isothermal Reactivity feedback for fuel temperature effect only (HEU)

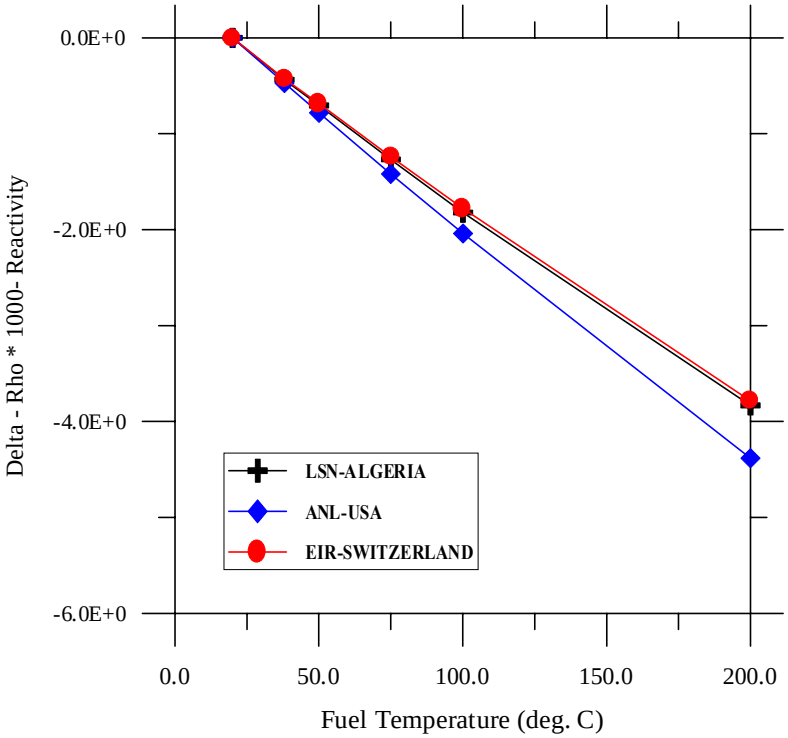


Figure 7 : Isothermal Reactivity feedback for fuel temperature effect only (LEU)

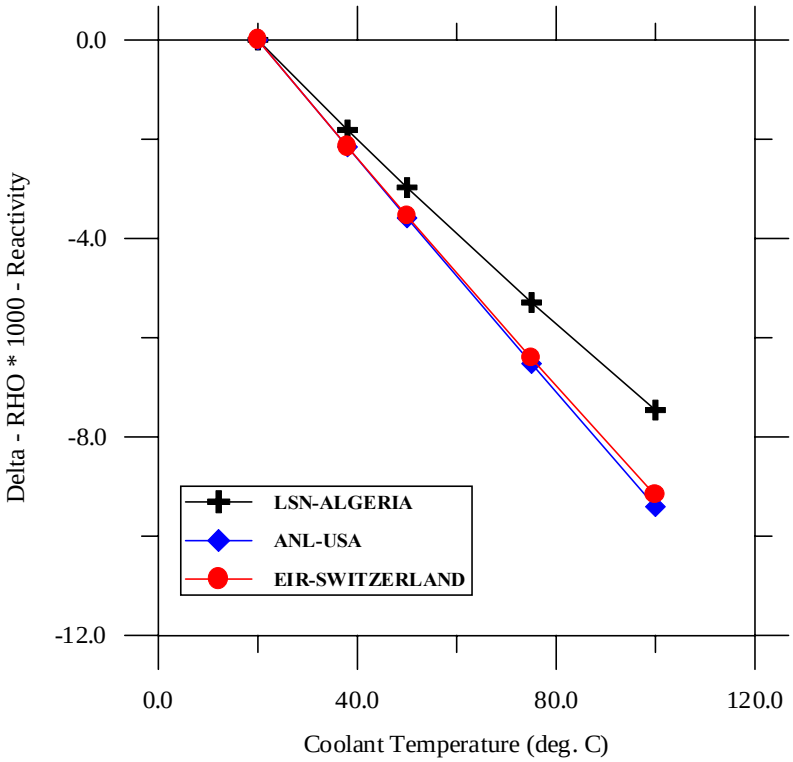


Figure 8 : Isothermal Reactivity feedback for water temperature effect only (HEU)

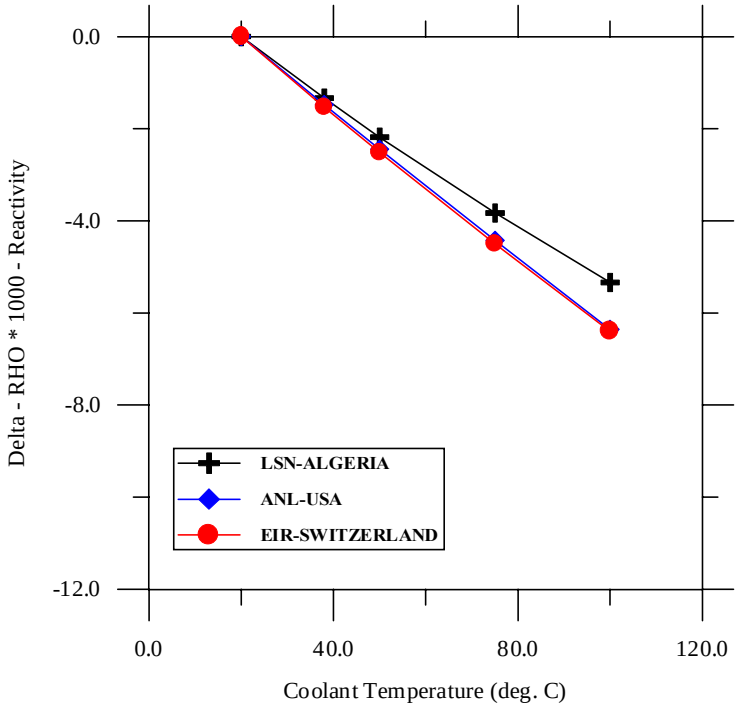


Figure 9 : Isothermal Reactivity feedback for water temperature effect only (LEU)

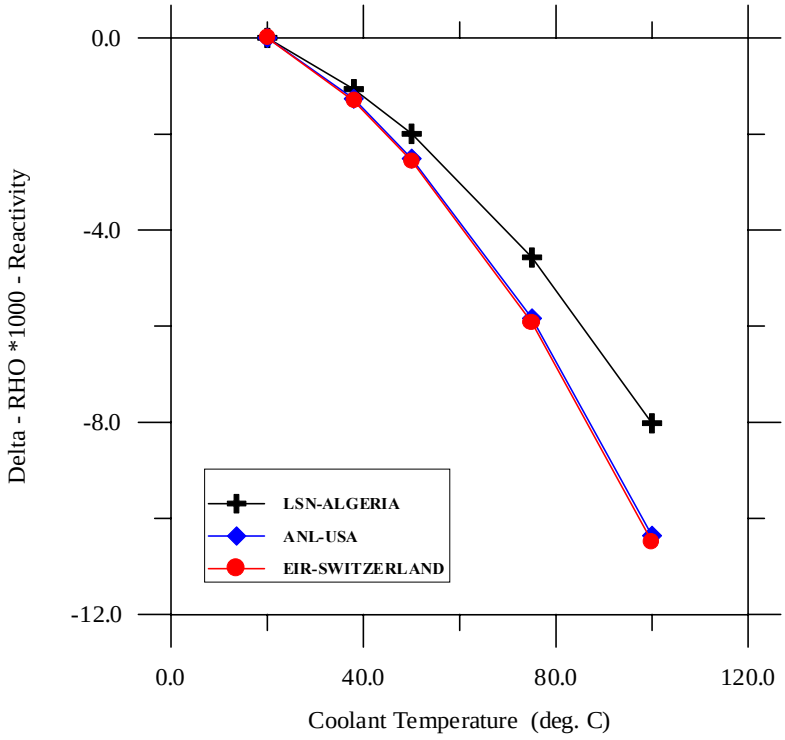


Figure 10 : Isothermal Reactivity feedback for water density effect only (HEU)

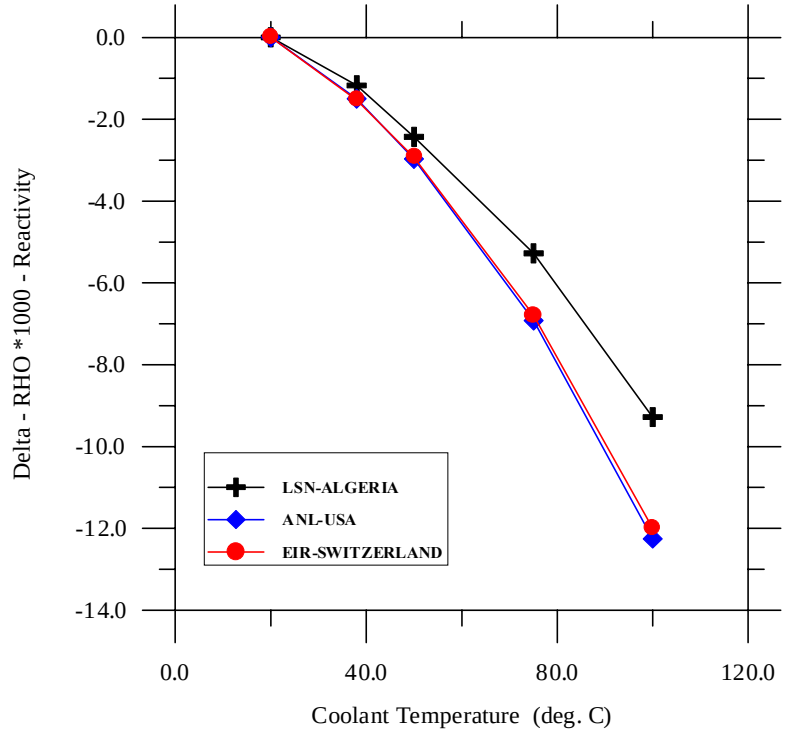


Figure 11 : Isothermal Reactivity feedback for water density effect only (LEU)

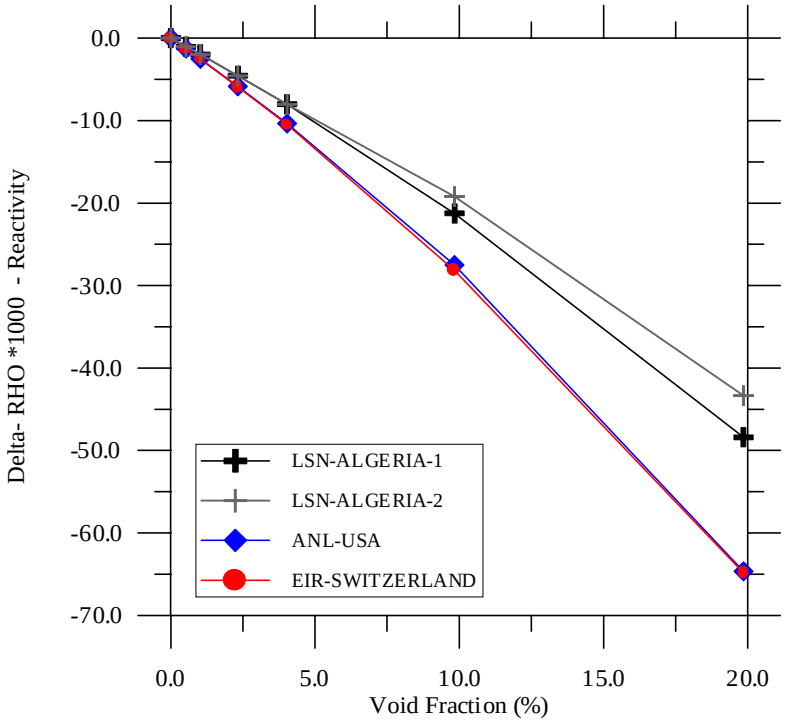


Figure 12 : Isothermal Reactivity feedback for water void effect only (HEU)

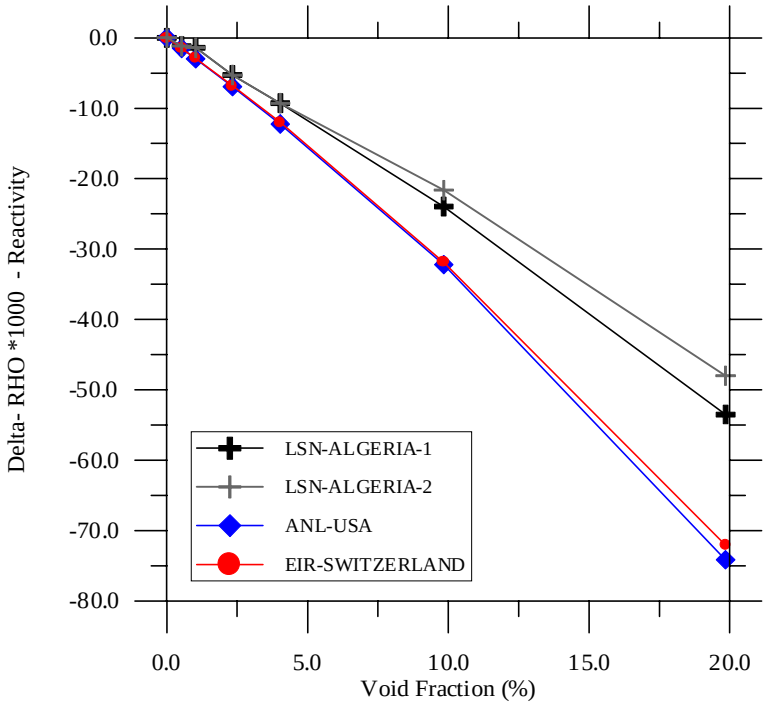


Figure 13 : Isothermal Reactivity feedback for water void effect only (LEU)

2.3.3. Power Defect of Reactivity

The power defect of reactivity which is an important parameter for reactor operation, is defined as the total of all reactivity effects induced by bringing the reactor from zero-power conditions to normal operating conditions. Thus, taking into account the different temperature conditions, the loss of reactivity between zero and full power is given by the equation (11) below:

$$\Delta\rho_{power} = \left(\alpha_{T_w} + \alpha_{D_w}\right)\overline{\Delta T_W} + \alpha_{T_f}\,\overline{\Delta T_f} \qquad (11)$$

α_{Tw}, α_{Dw} and α_{Tf} are the temperature coefficients of reactivity defined in table 3 and ΔT_w and ΔT_f are the mean temperature differences in the water and in the fuel from cold zero-power conditions to normal operating conditions.

For the HEU and LEU cases reported in Ref. [10] with an inlet temperature of 38°C and a flow rate of 1000 m^3/h, the mean temperature difference between zero power and full power would be about 4.5°C in the water and about 16.8°C in the fuel meat.

Table 4 shows the water, fuel and total reactivity differences between zero and full power computed using isothermal reactivity coefficient in Table 3 for the temperature range 38-50°C.

Table 4 : Power Defect of Reactivity ($\Delta\rho$ x 10^{-3})

$\overline{\Delta T_W} = 4.5\ ^{\circ}C$ $\qquad$ $\overline{\Delta T_f} = 16.8\ ^{\circ}C$

Effect	Fuel	LSN	ANL	EIR
Fuel Temperature	HEU	0.004	0.009	0.003
	LEU	0.373	0.433	0.363
Water Temperature + Density	HEU	0.832	1.004	0.990
	LEU	0.792	0.918	0.900
$\Delta\rho$ power	HEU	0.836	1.013	0.993
	LEU	1.165	1.351	1.263
β_{eff} (%)	HEU	0.776	0.761	0.778
	LEU	0.739	0.728	0.736
$\Delta\rho$ power, ¢	HEU	10.8	13.3	12.8
	LEU	15.8	18.6	17.2

2.3.4. Power Peaking Factors

The radial power peaking factor is defined as the ration of the average midplane power in a specified element to the average midplane power in the core. The local power peaking factor is defined as the ratio of the maximum midplane power to the average midplane power in the specified element that was substituted.

Two-dimensional diffusion calculations were performed by MUDICO-2D to evaluate radial and local power peaking factors of selected fuel elements in the HEU and LEU BOC cores as specified in Ref.[9]. In grid positions CFE-1 and SFE-1 at designed burn-up were replaced with elements having fresh control and standard fuel respectively. Final results are summarized in Table 5. Comparatively to the other laboratories, one can see that the values of peaking factors didn't change drastically for both cases (HEU and LEU), even when control fuel element is substituted by fresh one, case of which variation is expected to be higher then the present value. However, the variation registered to multiplication factor is very significant, especially for control fuel element substitution.

Table 5 : Power Peaking Factors

Core	Fresh Element	Element Substituted	LSN	ANL	EIR
		Radial x Local			
HEU Core	HEU in	None	1.51	1.46	1.47
	HEU Core	CFE - 1	1.55	1.81	1.32
		SFE - 1	1.52	1.59	1.64
LEU Core	LEU in	None	1.88	1.58	1.97
	LEU Core	CFE - 1	1.90	1.75	1.23
		SFE - 1	1.89	1.71	2.13

3. CONCLUSION

Overall, the results obtained using owning developed computer codes are in good agreement comparatively to those given by the contributed laboratories to TECDOC-643. The discrepancy found for few parameters suggest that more than some fine-tuning of the particular calculational methods and models used might be desirable.

Moreover, in order to comply with several specialized institutions in implementing appropriate global calculational procedure and to confirm the accuracy of neutronic calculations, we should complete further investigations by defining experimental program to compare the obtained results with measurements.

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