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VALIDATION OF THE DETERMINISTIC DIFFUSION METHOD FOR THE NEUTRONIC CALCULATIONS OF THERMAL RESEARCH REACTORS OF TRIGA-TYPE USING THE WIMSD-IAEA-69 NUCLEAR DATA LIBRARY

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The objective of this paper is to assess the suitability and the accuracy of the deterministic diffusion method for the neutronic calculations of the TRIGA type research reactors in proposed condensed energy spectra of five and seven groups with one and three thermal groups respectively, using the calculational line: WIMSD-IAEA-69 nuclear data library/ WIMSD-5B lattice and cell calculations code/ CITVAP v3.1 core calculations code. Firstly, The assessment goes through analyzing the integral parameters - k_{eff}, ρ^{238}, δ^{235}, δ^{238}, and C^* - of the TRX and BAPL benchmark lattices and comparison with experimental and previous reference results using other ENDLs at the full energy spectra, which show good agreement with the references at both spectra. Secondly, evaluation of the 3D nuclear characteristics of three different cores of the TRR-1/M1 TRIGA Mark-III Thai research reactor, using the CITVAP v3.1 code and macroscopic cross-section libraries generated using the WIMSD-5B code at the proposed energy spectra separately.

The results include the excess reactivities and the worth of control rods, which were compared with previous Monte Carlo results and experimental values, that show good agreement with the references at both energy spectra, albeit better accuracies are shown with the five groups spectrum. The results also includes neutron flux distributions which are settled for future comparisons with other calculational techniques, even, they are comparable to reactors and fuels of the same type. The study reflects the adequacy of using the pre-stated calculational line at the condensed energy spectra for evaluation of the neutronic parameters of the TRIGA type reactors, and future comparisons of the un-benchmarked results could assure this result for wider range of neutronics or safety-related parameters.

Keywords: *Neutronic Calculations, TRIGA, TRR-1/M1, WIMSD-5B, TRX and BAPL Benchmark Lattices.*

INTRODUCTION

Nuclear research reactors are neutrons factories, in which the main purposes are to generate and utilize neutron flux of sufficient intensity for research, testing and other applications [20]. The TRIGA-type research reactors are the most widely used non-power nuclear reactors in the world, due to the inherited safety features, operational flexibility,

wide capabilities, and the ability to utilize LEU fuels efficiently, with 66 installed units all over the world [7]. The TRIGA reactors are designed, manufactured and supplied by the U.S. General Atomics Co., and are typically used for research, isotopes production and reactor materials testing. The inherent safety, which is the most unique feature of the TRIGA-type reactors is gained through the utilization of the UZrH fuels, the homogeneous Uranium-Zirconium-Hydride (UZrH) solid fuel-moderator elements which have a large ($\sim 10^{-4}$ $\Delta k/k$ per °C) and prompt negative temperature coefficient of reactivity, also the fuel metallurgical stability and its capability to withstand high operating temperatures - usually up to 750 °C with a safety limit of 1150 °C. These safety features have enabled the TRIGA-type reactors to be routinely and safely operated at steady state or pulsed operational mode [4, 7, 30].

In order to ensure the safe operation and the proper utilization of research reactors, neutronic calculations are usually performed to evaluate the neutronics and safety-related parameters of the reactor [25, 26]. And so, the neutronic calculations are performed in the design, licensing, commissioning or the operational phases with major applications of: reactor physics calculations, criticality problems, safety analysis, core conversion feasibility studies, and fuel management studies. Typically, as a first step in the neutronic calculations, standardized benchmark problems for well defined reactor conditions are chosen and studied [28], with the purpose of obtaining a qualitative and quantitative measures of the reliability and accuracy of the calculational and modeling methods, numerical codes, and cross-section data sets, respectively. Benchmarking is achieved by comparison with: other numerical codes results, experimental data, or standard analytical solutions. Once the benchmarked calculational route is established, the neutronic calculations are performed with more assurance and reliability of the results. Typically, the neutronic calculations of the TRIGA type research reactors are performed using the continuous energy Monte Carlo method or the deterministic diffusion method using seven or more neutron energy groups with three or more thermal groups for accurate prediction of the neutronic parameters [8, 14, 15,1 8, 23, 24, 30]. And currently, the applicability of using five energy groups with one thermal group is under assessment.

Currently, the widely-used deterministic diffusion calculational route: WIMSD-IAEA-69 working nuclear data library [1]; WIMSD-5B lattice and cell transport calculations code [9,17]; CITVAP v3.1 multi-group diffusion core calculations code [6,31], is to be benchmarked for evaluating a qualitative and quantitative measures of its reliability and accuracy for calculating neutronics and safety related parameters of the MTR research reactors utilizing the TRIGA-type UZrH fuels, in five and seven energy groups spectra with one and three thermal groups respectively. Benchmarking will be achieved through: evaluation of the integral parameters of experimental benchmark lattices, the TRX and BAPL standard benchmark lattices [1,11,12]; and evaluation of the in-core nuclear characteristics of a TRIGA reactor, the TRR-1/M1 TRIGA Mark-III research reactor [21,25].

PROBLEM SPECIFICATION

1- Description of the TRX and BAPL Benchmark Lattices

Two types of thermal benchmark lattices of the BAPL - Bettis Atomic Power Laboratory, Westinghouse, USA - were analyzed: the H_2O-moderated uranium metal lattices TRX-1 and TRX-2 [10][11], and the H_2O-moderated uranium oxide critical lattices BAPL-UO_2-1, BAPL-UO_2-2 and BAPL-UO_2-3 [12]. The TRX-1 and TRX-2 benchmark lattices are light water moderated uranium metal critical lattices, with 1.3 wt%

enriched uranium metal rods with diameters of 0.9830 cm arranged in triangular arrays, lattice pitch of 1.8060 cm and 2.1740 cm respectively, and experimental buckling of 57.00 m^{-2} and 54.69 m^{-2} respectively [1, 3, 10, 11]. The BAPL-1, 2 and 3 benchmark lattices are light water moderated uranium oxide critical lattices, with 1.311 wt% enriched uranium oxide rods with diameters of 0.9728 cm arranged in triangular arrays, lattice pitch of 1.5578 cm, 1.6523 cm, and 1.8057 cm respectively, and experimental buckling of 32.59 m^{-2}, 35.47 m^{-2}, and 34.22 m^{-2} respectively [1, 3, 12].

2- Description of the Thai Research Reactor TRR-1/M1

The Thai Research Reactor 1/Modification 1 (TRR-1/M1) is a TRIGA Mark III research reactor, which is being operated by Thailand Institute of Nuclear Technology (TINT) [16, 21, 25]. The reactor is a pool-typed reactor with a movable core that is light water cooled, moderated by light water and the zirconium-hydride ZrH, and reflected by water and graphite. The reactor fuel is the LEU TRIGA Uranium-Zirconium-Hydride ($UZrH_{1.6}$) fuel-moderator fuel rods, which are stacked circularly in hexagonal rings that are labeled; B, C, D, E, F, and G ring, outwardly from the center thimble (CT). The reactor is multipurpose with a steady state operation up to 2 MW_{th} and pulsed mode operation at around 23 MW-sec, and a pulsed width at a half of maximum of 10 msec. The reactor is typically used for production of radioisotope, neutron beam experiments and reactor physics experiments. The reactor has a max in-core thermal and fast flux of 3.1E13 and 1.8E13 respectively. Since the initial criticality of the TRR-1/M1 on November 7, 1977, the whole core was composed of fuel rods of 8.5% w/o uranium load and 20% w/o U-235 enrichment (SFE-Type). Since 1980, and with an objective of improving the lifetime of the fuel element, the burned standard fuel elements are gradually core-by-core replaced by fresh 20% w/o uranium load and 20% w/o U-235 enrichment fuel elements (LEU-Type) with a neutron poison of 0.47 wt % erbium, resulting in mixed core operation until now [8, 16, 25]. Three models of the Thai research reactor TRR-1/M1 are analyzed, which contain different arrangements of the reactor components:

(1) Core #1; consists of 96 fuel rods (SFE), 16 irradiation position (IP), 3 neutron detectors (ND), 1 central thimble (CT), 4 fuel-follower control rods (CR), and 1 air-follower transient rod (TR) [16].
(2) Core #2; consists of 95 fuel rods (SFE), 5 fuel rods (LEU), 12 irradiation position (IP), 3 neutron detectors (ND), 1 central thimble (CT), 4 fuel-follower control rods (CR), and 1 air-follower transient rod (TR) [16].
(3) Core #3; consists of 67 fuel rods (SFE), 38 fuel rods (LEU), 7 irradiation position (IP), 3 neutron detectors (ND), 1 central thimble (CT), 5 fuel-follower control rods (CR) [8].

Gallmeier et al. [1998] and N. Kriangchaiporn [2007] stated detailed specifications of the TRR-1/M1 reactor and different arrangements of the reactor components within the core grid. Figure (1) shows the different core grid positions for the fuel rods, CRs, IPs, … etc, for the 1st core. Figure (2) shows a descriptive position of the fully inserted and fully withdrawn control rods which indicate the relative positions for the 38.1 cm length B_4C absorber part as compared to the 38.1 cm length fuel-follower part of SFE fuel-type, also the fuel rods are shown with the 38.1 cm active length that is reflected from top and bottom by 8.7 cm of graphite. The 1st and the 3rd core are modeled as fresh cores with no burned fuel elements. The 2nd core has an equilibrium burnup distribution, which is the 1st core EOC burnup distribution upon completion of a burnup cycle of 61.23 MWD and replacement of 5 out of the 11 SFE fuel rods of the C-ring with 5 fresh LEU fuel rods [16].

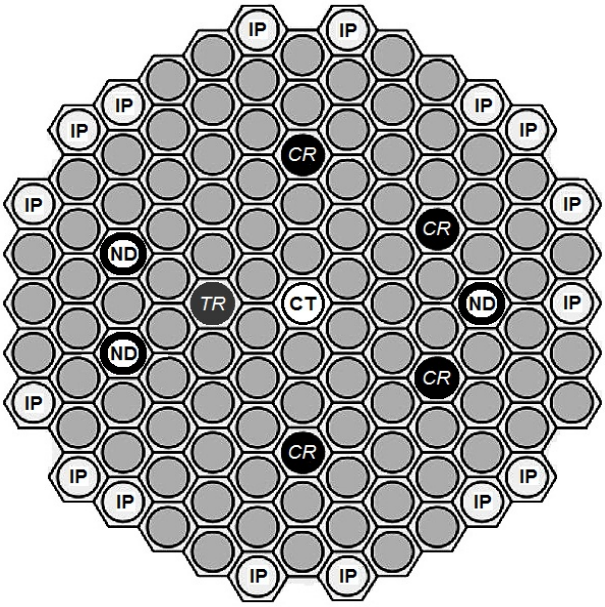

Figure 1. Midplane cross sectional cut through the 1st core, indicating: control rods (CR), transient rod (TR), irradiation positions (IP), neutron detectors (ND) and the remaining fuel rods positions.

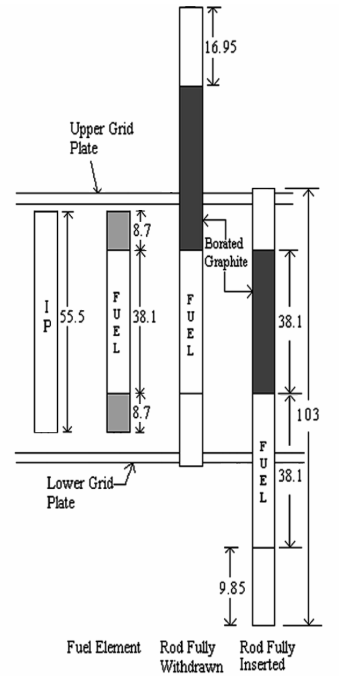

Figure 2. The fuel rods and the control rods different positions: fully inserted and fully withdrawn.

CALCULATIONAL PROCEDURE

1- Specification of the Calculational Route

The utilized numerical tools: WIMSD-IAEA-69 nuclear data library [1]; WIMSD-5B lattice and cell calculations code [9, 17]; CITVAP v3.1 core calculations code [6, 31], which are available with the MTR_PC v3.0 computer package of the Argentinean INVAP Co [32], were used in the condensed energy spectra of five and seven energy groups with

one and three thermal groups, respectively, as shown in table 1, for evaluation of the lattice integral parameters, and the cell and core neutronic parameters.

Table 1. The utilized five and seven energy spectra, with the upper and lower energy limits indicated.

Energy Group		5 Energy Groups	
		E_U	E_L
Epithermal	E_1	10 MeV	0.821 MeV
	E_2	0.821 MeV	5.530 KeV
	E_3	5.530 KeV	2.1 eV
	E_4	2.1 eV	0.625 eV
Thermal	E_5	0.625 eV	0.0 eV
Energy Group		7 Energy Groups	
		E_U	E_L
Epithermal	E_1	10 MeV	0.500 MeV
	E_2	0.500 MeV	9.118 KeV
	E_3	9.118 KeV	1.123 eV
	E_4	1.123 eV	0.625 eV
Thermal	E_5	0.625 eV	0.140 eV
	E_6	0.140 eV	0.050 eV
	E_7	0.050 eV	0.000 eV

1.1- The WIMSD-IAEA-69 Nuclear Data Library

The WIMSD-IAEA-69 library is the WIMS Library Update Project 69-group nuclear data library [1], which was basically originated with the aim to introduce the most recent evaluated nuclear data files into research and power reactors neutronic calculations. The library has advantages over the older ones, and specifically it has been chosen as it is updated through the use of new and revised evaluated nuclear data files, it is well tested and bench-marked, and the library contains increased number of materials, resonance absorbers, … etc.

1.2- The WIMSD-5B Transport Cell and Lattice Calculations Code

The Winfrith Improved Multigroup Scheme version D, WIMSD-5B is a general deter-ministic code for lattice physics calculations that is widely used in the neutronic calculations of thermal reactors [2]. The WIMS code calculates the flux as a function of energy and space in the cell by solving the energy and space dependent Boltzmann neutron transport equation numerically with energy and spatial discretization - the broad energy groups and finite differ-rence approximations respectively - for a variety of geometries; homogeneous, slab, rod clus-ters and finite cylinders [2, 19]. The code solves the neutron transport equation by a variety of methods, and currently, the discrete ordinate method DSN was selected for all WIMS calculations. The currently used version WIMSD-5B is an updated version of the code that was released from Winfrith in 1998 for distribution by the OECD/NEA Data Bank [17, 33].

1.3- The CITVAP v3.1 Diffusion Core Calculations Code

CITVAP v3.1 [31] is an enlarged and improved version of the original and well known multigroup diffusion-depletion code CITATION-II [6], that is developed by the Argentinean INVAP's Co. Nuclear Eng. Division. The code solves the multigroup diffusion equation - the diffusion theory approximation of the neutron transport theory -

using the finite difference representation explicitly in space and time. CITVAP preserves all the calculational capabilities of CITATION - with modifications for implementations on PCs: burnup depen-dent calculations, fuel management calculations, criticality calculations and criticality search.

2- The TRX and BAPL Benchmark Lattices

The integral parameters of the TRX and BAPL benchmark lattices [5]; TRX-1 and TRX-2, BAPL-1, BAPL-2, and BAPL-3 thermal assemblies - were calculated with the lattice calculations code WIMSD-5B [17] and the WIMS nuclear data library WIMSD-IAEA-69 [1] at five and seven energy groups condensation spectra as shown in table 1. The main transport calculations are performed using the discrete ordinate method DSN of the WIMS code. The reaction rates and the macroscopic cross sections of the U-235 and U-238 isotopes were edited into two groups; an epi-thermal and thermal groups with the energy ranges; 10MeV > $E_{epi\text{-}thermal}$ > 0.625 eV and 0.625 eV > $E_{thermal}$ > 0.0 eV respectively. The specification of the calculated integral parameters is shown in equations: 1, 2, 3, and 4 [22]

k_{eff} = the finite medium effective multiplication factor;

ρ^{238} = ratio of epithermal to thermal U-238 capture reaction rate;

$$\rho^{238} = (\Sigma_c)^{238}_{epith} \Big/ (\Sigma_c)^{238}_{th} = (\Sigma_a - \Sigma_f)^{238}_{epith} \Big/ (\Sigma_a - \Sigma_f)^{238}_{th} \qquad \text{(Eq. 1)}$$

δ^{235} = Ratio of epithermal to thermal U-235 fission reaction rate;

$$\delta^{235} = (\Sigma_f)^{235}_{epith} \Big/ (\Sigma_f)^{235}_{th} \qquad \text{(Eq. 2)}$$

δ^{238} = Ratio of U-238 fission to U-235 fission reaction rate;

$$\delta^{238} = (\Sigma^t_f)^{238} \Big/ (\Sigma^t_f)^{235} \qquad \text{(Eq. 3)}$$

C^* = Ratio of U-238 capture to U-235 fission reaction rate;

$$C^* = (\Sigma^t_c)^{238} \Big/ (\Sigma^t_f)^{235} = (\Sigma^t_a - \Sigma^t_f)^{238} \Big/ (\Sigma^t_f)^{235} \qquad \text{(Eq. 4)}$$

3- The Neutronic Calculations of the TRR-1/M1 Research Reactor

Application of the deterministic diffusion method in the neutronic calculations of thermal research reactors is divided into two steps; the preliminary cell calculations step and the core calculations step. Currently, the cell calculation step is performed using the transport cell calculations code WIMSD-5B along with the WIMSD-IAEA-69 nuclear data library, which are used to generate the condensed and homogenized cell constants for all the materials in the core in the proposed condensed energy spectra. Afterwards, macroscopic cross sections libraries are prepared, which mainly contain: the macroscopic absorption cross sections Σ_{ag}, production cross sections $\nu\Sigma_{fg}$, scattering matrix, and the diffusion coefficients. The tabulated homogenized and condensed cell constants in the macroscopic cross sections libraries were used in the core calculations step for zones definition. The geometry specification, zones definition, and calculation routine constitute the CITVAP v3.1 inputs, which is used for evaluation of the reactor neutronic and safety related parameters.

3.1- Application of the WIMSD-5B Lattice Physics Code

The cell constants were generated using three models that sufficiently and accurately represent all the materials in the core under different spectra. The first model is used to evaluate the cell constants of the SFE and LEU fuel elements, which is a hexagonal pin-cell model representing the circular fuel rod, associated clad, and the hexagonal water region around each rod. The core is divided in the radial plane into equal hexagons that fit together with no space in-between as shown in figure 1. The cell constants of the fuel rod and the associated clad and water within each hexagon were weighted and averaged together using the weighting flux to give the cell constants of the whole hexagon to be defined later in CITVAP as a distinct zone.

The second model is used to evaluate the cell constants of the non-fuel material in the core; the irradiation positions, the control rods, … etc. The second model incorporates the MULTICELL option available in the WIMS code, in which separate pin-cells were used together for the SFE and LEU - for generation of spectra, and cells for the CRs, and IPs - water filled or voided. Each cell is composed of a cylindrical annulus of the fuel type SFE or LEU, B_4C absorber, water or air according to the cell type, followed by a ring of st. st. cladding, and followed by a hexagonal water zone. The third cell type is used to generate the cell constants of the graphite plugs, aluminum grid and support plates, and the water reflector. The slab model used represents half of the active core height, the top graphite plugs, the aluminum plates, and the axial water reflector. As the neutron spectrum is considered unchanged between the radial and axial water reflectors and also for the aluminum grid and support plates at different locations, the same cell constants were used for each case separately. The macroscopic cross sections libraries were managed by the HXS v4.1 program - available with the MTR_PC v3.0 package - that handled the homogenized and condensed cell constants for each zone in the reactor in the CITATION format that were used in the core calculation step afterwards.

3.2- Application of the CITVAP v3.1 Core Calculations Code

The multigroup diffusion-depletion code CITVAP v3.1 is used in the core calculation step using the five and seven energy groups macroscopic cross sections libraries. The calculations are done using the 3D XYZ geometry representation for the three cores, and using sufficient amount of water reflector axially and radially. The presentation of different zones in the XY plane in the CITVAP is done by transferring separately each hexagonal containing a fuel rod, control rod, … etc, into an equivalent squared shape zone as shown in figure 4 for core #2 and core #3, and conserving the volume ratios. The cell constants of the material within each hexagon were homogenized together while each hexagon was presented separately with no homogenization between different hexagons. Axially, separate zones representing graphite plugs, aluminum plates, and the water reflector were used. The number of mesh points for each zone containing rod was standardized for all the calculations to 10x10 along the XY directions to neutralize all parameters in the calculations. Each fuel element is divided axially into three zones of a central fueled part of and a top and bottom zones for the graphite plugs. The fuel follower control rods are divided axially into four zones, from top to bottom; void zone, boron carbide zone, SFE zone, and water zone. The irradiation channels are composed axially of one void zone.

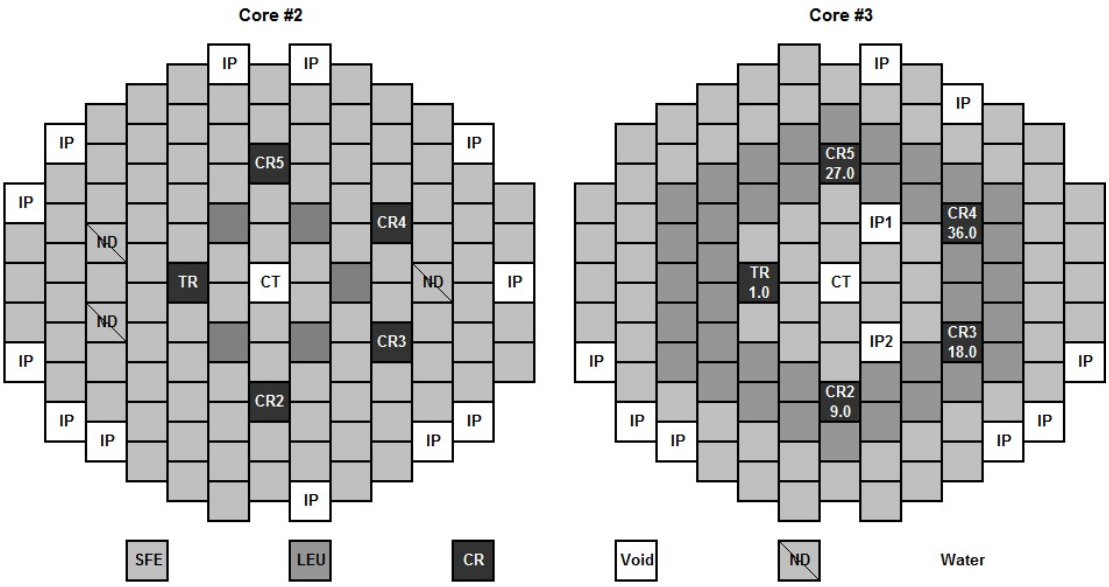

Figure 3. The CITVAP core models for cores #2 and #3 of the TRR-1/M1 reactor, illustrating various core zones and also the CRs withdrawal length of core #3.

RESULTS AND DISCUSSIONS

1- The Integral Parameters of TRX and BAPL Benchmark Lattices

The integral parameters of the TRX and BAPL benchmark lattice [5], were evaluated using the DSN main calculation routine of the WIMSD-5B code and the WIMSD-IAEA-69 nuclear data library at five and seven energy groups spectra as shown in table 1, and the results are presented in table 2. The table also presents the reference results: the experimental results [5], the continuous energy Monte Carlo MCNP results using the CENDL-3.0 evaluated nuclear data library [13], and the WIMSD-5B results using a 69-group cross-section library generated from the JEFF-3.1.1 ENDL [29]. The deviations of the current results from the experimental values are expressed in the percent differences as compared to the experimental values for all of the integral parameters of the five assemblies, and are given in parenthesis each integral parameter.

Generally, the currently calculated effective multiplication factors using the five and seven energy groups show good agreement with the experimental results, albeit better accuracies are encountered with the seven energy groups results. The maximum errors encountered are 0.91% for the TRX-2 and 0.52% for the TRX-1 which correspond to 898 pcm and 527 pcm for the five and seven energy groups, respectively. For the remaining integral parameters, the five and seven energy groups calculations show nearly the same trend and fairly good agreements with the experimental results. Generally, no major differences were encountered, and the sources of the differences between the current and the reference results are back to the differences in the basic nuclear data sets used, the condensed energy spectrums calculations as compared to full spectrum calculations of the references, and the different solution technique for the different numerical codes used.

Table 2. The Integral Parameters of the TRX and BAPL benchmark lattices.

Lattice	Integral Parameter	Experimental [5]	CENDL-3.0 [13]	JEFF-3.1.1 [29]	WLUP-69 [a] (5 groups)	WLUP-69 [a] (7 groups)
TRX-1	k_{eff}	1.0000 (0.30)[b]	0.9975 (-0.25)[c]	0.9885 (-1.15)[c]	1.0056 (0.56)[c]	0.9948 (-0.52)[c]
UMe	ρ^{28}	1.3200 (1.60)	1.3608 (+3.09)	1.3285 (+0.64)	1.3422 (1.68)	1.3436 (1.79)
	δ^{25}	0.0987 (1.00)	0.0980 (-0.71)	0.0975 (-1.21)	0.0970 (-1.75)	0.0969 (-1.80)
	δ^{28}	0.0946 (4.30)	0.0962 (+1.69)	0.0915 (-3.27)	0.0990 (4.69)	0.1009 (6.70)
	C^*	0.7970 (1.00)	0.7922 (-0.60)	0.7926 (-0.55)	0.7900 (-0.88)	0.8002 (0.40)
TRX-2	k_{eff}	1.0000 (0.10)	0.9982 (-0.18)	0.9895 (-1.05)	1.0091 (0.91)	0.9984 (-0.16)
UMe	ρ^{28}	0.8370 (1.90)	0.8530 (+1.91)	0.8273 (-1.16)	0.8407 (0.44)	0.8420 (0.60)
	δ^{25}	0.0614 (1.30)	0.0620 (+0.98)	0.0603 (-1.79)	0.0595 (-3.08)	0.0595 (-3.05)
	δ^{28}	0.0693 (5.10)	0.0681 (-1.73)	0.0685 (+1.15)	0.0705 (1.73)	0.0715 (3.14)
	C^*	0.6470 (0.93)	0.6387 (-1.28)	0.6388 (-1.27)	0.6374 (-1.48)	0.6454 (-0.25)
BAPL-1	k_{eff}	1.0000 (0.10)	1.0023 (+0.23)	0.9970 (-0.30)	1.0063 (0.63)	0.9986 (-0.14)
UO_2	ρ^{28}	1.3900 (0.72)	1.3923 (+0.16)	1.3851 (-0.35)	1.4037 (0.99)	1.4002 (0.74)
	δ^{25}	0.0840 (2.40)	0.08199 (-2.39)	0.0815 (-2.98)	0.0821 (-2.31)	0.0818 (-2.59)
	δ^{28}	0.0780 (5.10)	0.07362 (-5.61)	0.0753 (-3.46)	0.0780 (-0.03)	0.0790 (1.34)
	C^*	-	0.7972	0.7919	0.8047	0.8130
BAPL-2	k_{eff}	1.0000 (0.10)	1.0021 (+0.21)	0.9967 (-0.33)	1.0064 (0.64)	0.9990 (-0.10)
UO_2	ρ^{28}	1.1200 (0.89)	1.1602 (+3.59)	1.1187 (+0.12)	1.1679 (4.28)	1.1647 (3.99)
	δ^{25}	0.0680 (1.50)	0.0669 (-1.61)	0.0667 (-1.91)	0.0670 (-1.54)	0.0667 (-1.84)
	δ^{28}	0.0700 (5.70)	0.0633 (-9.57)	0.0650 (-7.14)	0.0671 (-4.19)	0.0678 (-3.15)
	C^*	-	0.7274	0.7223	0.7334	0.7407
BAPL-3	k_{eff}	1.0000 (0.10)	1.0021 (+0.21)	0.9975 (-0.25)	1.0072 (0.72)	1.0003 (0.03)
UO_2	ρ^{28}	0.9060 (1.10)	0.9130 (+0.77)	0.8996 (-0.71)	0.9180 (1.32)	0.9154 (1.04)
	δ^{25}	0.0520 (1.90)	0.0515 (-0.96)	0.0512 (-1.54)	0.0515 (-0.97)	0.0513 (-1.29)
	δ^{28}	0.0570 (5.30)	0.0518 (-9.12)	0.0535 (-6.14)	0.0549 (-3.64)	0.0553 (-2.91)
	C^*	-	0.6511	0.6468	0.6559	0.6622

[a] WLUP-69 is the WIMSD-IAEA 69 group nuclear data library.
[b] Values within brackets are experimental errors.
[c] Values within brackets are percent differences from experimental values
= [(calculated value - experimental value) / experimental value] x 100.

2- The Neutronic Study of the TRR-1/M1 TRIGA Mark-III Research Reactor

The in-core nuclear characteristics of the TRR-1/M1 TRIGA Mark-III Thai research reac-tor are studied at proposed spectra of five and seven groups, as in table 1. Comparisons were set for three cores of the reactor: Core #1, #2, and #3 as described in section 2.2 and figures 1 and 3. The results are mainly reactivity calculations and neutron flux eigen-value problems.

2.1- The Reactivity Calculations of the TRR-1/M1 Research Reactor

The effective multiplication factor k_{eff} is by far the most important parameter in the reactor analysis, and it depends on the material composition and its distribution in the core, and the geometry of the core, also excess reactivity of the core which is a measure of the departure
from criticality, which is given in two equivalent units, pcm and \$, by;

$$\rho\ (\text{pcm}) = \frac{k_{eff} - 1}{k_{eff}}\ , \text{and}\ \rho\ (\$) = \frac{k_{eff} - 1}{\beta k_{eff}}$$

The delayed neutron fraction (β) equals 0.007 for LEU TRIGA fuels.

The reactivities and effective multiplication factors were calculated using the diffusion depletion code CITVAP v3.1 in 3D geometry, based on macroscopic cross-sections libraries generated using the WIMSD-5B lattice calculation code, and the WIMSD-IAEA-69 nuclear data library. The effective multiplication factors were calculated for the 1st, 2nd, and 3rd using the five and seven energy groups libraries, also, the total control rods worths, which were evaluated by the difference in reactivities for the totally inserted and the totally withdrawn control rods, were calculated for the 1st and 3rd cores . The current reactivities results using the deterministic diffusion method are compared to the previous Monte Carlo MVP and MCNP results for the three cores, moreover, comparison with the experimental values for the 1st and 2nd cores. The reference results of the 1st and 2nd cores are the BOC excess reactivities evaluated experimentally and using the Monte Carlo MVP code, also the total control rods worth for the 1st core as shown in tables 3 and 4 [16]. The reference results of the 3rd core are the effective multiplication factors for the three control rods positions - fully inserted, working position, and fully withdrawn - also the relative reductions in reactivities upon control rods insertion for the current and the reference Monte Carlo calculations as shown in table 5 [8]. The deviations between the current and the reference Monte Carlo results for the 3rd core are presented in terms of standard deviations as well as percent deviations.

The calculated fresh cores excess reactivities for the 1st core show good agreement with the references, as shown in table 3, with max. deviations in reactivities for the fully with-drawn control rods core of -4% and -7% which corresponds to -210 pcm and -371 pcm for the results of the five and seven energy groups, respectively. The calculated total control rods worth equals the difference in reactivities between the fully inserted and the fully withdrawn positions. The current control rods worth results deviated more than that of the reactivity results, even acceptable, with max deviations of -841 pcm and -1127 pcm for the results of the five and seven energy groups, respectively.

Table 3. The 1st core criticality calculations results.

Fully Withdrawn (Excess Reactivity)	ρ (\$) Experimental	ρ (\$) MVP	ρ (\$) + R WIMS/CITATION	K_{eff} WIMS/CITATION
5 Groups	**7.43**	**8.04 ±0.10**	**7.13 (-4.0%) (-210 pcm)**	**1.0525**
7 Groups			**6.90 (-7.1%) (-371 pcm)**	**1.0508**
Total Control Rods Worth	ρ (\$) Experimental	ρ (\$) MVP	ρ (\$) + R WIMS/CITATION	K_{eff} WIMS/CITATION
5 Groups	**15.01**	**16.38±0.50**	**16.21 (-8.0%) (-841 pcm)**	-
7 Groups			**16.76 (-11.7%) (-1127 pcm)**	-

The calculated excess reactivities for the 1st core upon completion of the 61.23 MWD burnup cycle, and fuel rods replacements - 2nd core BOC, show good agreement with the references, as shown in table 4, with max. deviations of 4.7% and 8.9% which corresponds to 224 and 430 pcm for the results of the five and seven energy groups respectively.

Table 4. The 2nd core criticality calculations results.

Fully Withdrawn	ρ (\$) Experimental	ρ (\$) MVP	ρ (\$) + R WIMS/CITATION	K_{eff} WIMS/CITATION
5 Groups	**6.87**	**7.67 ±0.10**	**6.55 (-4.7%) (-224 pcm)**	**1.0481**
7 Groups			**6.26 (-8.9%) (-430 pcm)**	**1.0458**

The calculated effective multiplication factors for the 3rd core for the three control rods positions using the five and seven energy groups libraries show good agreements with the reference Monte Carlo MCNP results, albeit better accuracies are encountered with the five energy groups library as shown in table 5. Using the five energy groups library instead of the seven energy groups library has reduced the deviations from 3.17, 2.40 and 2.78 to 0.38, 0.10 and 0.71 units of standard deviations for the fully inserted, working position and fully with-drawn control rods positions respectively, with the same trend in units of percent deviations. The differences in the reactivities between the current diffusion results using the five energy groups library and the reference Monte Carlo results are -214, -67, and -465 pcm for the fully inserted, working position and fully withdrawn control rods positions respectively.

Table 5. The 3rd core reactivity results.

Core State	k_{eff}	ρ (pcm)	$(k_{MC}^{a} - k_{D}^{a}) / \sigma^{b}$	$(k_{MC} - k_{D}) / k_{MC}$	$\Delta\rho_{D}^{c}$ (pcm)	$\Delta\rho_{D}^{c} / \Delta\rho_{MC}^{c}$ $(\Delta\rho_{D} - \Delta\rho_{MC})$
	5 Energy Groups Condensation Structure					
Fully Inserted	0.97495	-2569.4	0.38	0.21%	12053.60	0.9796 (-250)
Working Position	1.03102	3008.7	0.10	0.07%	6475.57	0.9421 (-398)
Fully Withdrawn	1.10478	9484.2	0.71	0.51%	-	-
	7 Energy Groups Condensation Structure					
Fully Inserted	0.96015	-4150.4	3.17	1.72%	12258.90	0.9963 (-45)
Working Position	1.01533	1509.9	2.40	1.59%	6598.65	0.9600 (-275)
Fully Withdrawn	1.08824	8108.5	2.78	2.00%	-	-

[a] k_{MC} and k_{D} are the effective multiplication factors calculated by the Monte Carlo (reference) and the diffusion method (current), respectively.
[b] σ is the standard deviation of the reference Monte Carlo calculations.
[c] $\Delta\rho_{MC}$ and $\Delta\rho_{D}$ are the change in reactivity of the core upon control rods insertion from the fully withdrawn case for the Monte Carlo and the diffusion method respectively.

Generally, the criticality calculations results - the reactivities and the worth of the control rods - show good agreement with the references specially for the five energy groups calc-ulations, as shown in tables 3, 4, and 5. Even the seven energy groups results deviated from the references more than that of the five energy groups calculations, the feasibility of using such calculational routine in the neutronic calculations of the TRIGA type reactors has been previously assessed, and currently the feasibility of using a five energy groups library in a detailed 3D-diffusion calculations is under assessment.

2.2- The Neutron Flux Distribution of the TRR-1/M1 Research Reactor

The results of the neutron flux problem were evaluated suing the five and seven energy groups libraries in order to study the effect of this parameter on the neutron flux distributions. The results of the neutron flux problem are normalized to 2 MW power using an epithermal range of (107–0.625) eV and thermal range of (0.625-0) eV. The thermal and epithermal flux have been generated for the 3rd core at the central thimble (CT) along the Z-axis as shown in figure 4. Table 6 lists the maximum thermal and epithermal fluxes in the selected irradiation positions, , also the average thermal and epithermal flux for the 38.1 cm located in the core active part - the flux is averaged axially and radially within each irradiation position. The location of this maxima in the flux away from the top of the active distance of the core 38.1 cm - top of the fuel rods active part - is 19.55 for all calculated fluxes.

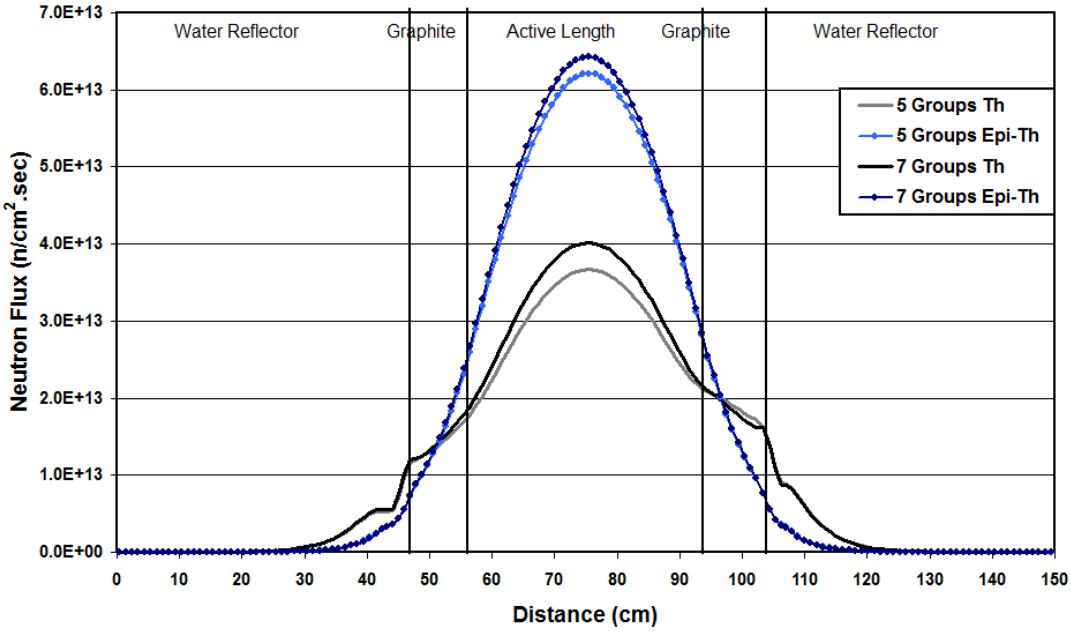

Figure 4. The axial distribution of the thermal and epithermal flux in the central thimble for the five and seven energy groups calculations.

Table 6. The maximum and average thermal and epithermal neutron fluxes for the five and seven energy groups calculations, taking the 7 groups results as a reference.

Group	5 Groups		7Groups	
Position	Thermal + R	Epi-Thermal + R	Thermal	Epi-Thermal
CT Avg.	4.915E+13 (-2.9%)	2.910E+13 (-6.4%)	5.062E+13	3.110E+13
CT Max.	3.665E+13 (-8.7%)	6.212E+13 (-3.4%)	4.015E+13	6.428E+13
IP1 Avg.	4.728E+13 (-2.7%)	2.241E+13 (-6.4%)	4.858E+13	2.396E+13
IP2 Avg.	4.737E+13 (-2.6%)	2.246E+13 (-6.3%)	4.862E+13	2.399E+13
Outer IPs Avg.	1.072E+13 (-0.2%)	1.243E+13 (+0.8%)	1.075E+13	1.233E+13

The calculations using the five energy groups library have underestimated the average thermal and epithermal fluxes in the central thimble by 6.4% and 2.9% as compared to the seven energy groups calculations respectively, also the maxima of the thermal and epithermal fluxes has been underestimated by 8.7% and 3.4% respectively.

Another limiting factor that is important for the thermalhydraulic calculations is the hottest rod, which is the rod that generates the maximum power in the reactor, also which impose the minimum thermalhydraulic requirements. Both calculations yielded the same hottest rod, of the B-ring with power density for the fuel element zone of 34.2 (-0.9%) and 34.5 watt/cc for the five and seven energy groups calculations respectively.

CONCLUSIONS

The aforementioned results and discussions have cleared a conclusion that the deterministic diffusion method using a five energy groups condensation spectra with one thermal group is valid in the neutronic calculations of thermal research reactors of TRIGA type, using the calculational route; WIMSD-IAEA-69/WIMSD-5B/CITVAP v3.1. And this conclusion has been settled through evaluation of lattice, cell, and core neutronics parameters, of different models and comparison with experimental and reference results for the validation and benchmarking process. Moreover, neutron flux distributions of the TRR-1/M1 reactor were left for future comparisons, which could assure that conclusion.

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