2015 International Nuclear Atlantic Conference - INAC 2015
São Paulo, SP, Brazil, October 4-9, 2015
ASSOCIAÇÃO BRASILEIRA DE ENERGIA NUCLEAR - ABEN
ISBN: **978-85-99141-06-9**

PARAMETERIZED REPRESENTATION OF MACROSCOPIC CROSS SECTION IN THE PWR FUEL ELEMENT CONSIDERING BURN-UP CYCLES

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ABSTRACT

Nuclear reactor core analysis involves neutronic modeling and the calculations require problem dependent nuclear data generated with few neutron energy groups, as for instance the neutron cross sections. The methods used to obtain these problem-dependent cross sections, in the reactor calculations, generally uses nuclear computer codes that require a large processing time and computational memory, making the process computationally very expensive. Presently, analysis of the macroscopic cross section, as a function of nuclear parameters, has shown a very distinct behavior that cannot be represented by simply using linear interpolation. Indeed, a polynomial representation is more adequate for the data parameterization. To provide the cross sections of rapidly and without the dependence of complex systems calculations, this work developed a set of parameterized cross sections, based on the Tchebychev polynomials, by fitting the cross sections as a function of nuclear parameters, which include fuel temperature, moderator temperature and density, soluble boron concentration, uranium enrichment, and the burn-up. In this study is evaluated the problem-dependent about fission, scattering, total, nu-fission, capture, transport and absorption cross sections for a typical PWR fuel element reactor, considering burn-up cycle. The analysis was carried out with the SCALE 6.1 code package. The results of comparison with direct calculations with the SCALE code system and also the test using project parameters, such as the temperature coefficient of reactivity and fast fission factor, show excellent agreements. The differences between the cross-section parameterization methodology and the direct calculations based on the SCALE code system are less than 0.03 percent.

1. INTRODUCTION

The nuclear reactor design is based on control of several variables. These variables include the nature of the fuel and the moderator, core compositions and geometry, and removal of the heat which is generated mainly by fission and partly by radioactive decay. An essential part of the reactor design is the core specification, since this determines the neutron behavior in the system and hence the criticality conditions [1]. Neutronic calculations are based on either transport or diffusion theory, which can be implemented by deterministic or stochastic method (Monte Carlo). In general the calculations require problem dependent nuclear data generated with few neutron energy groups, as for instance the neutron cross sections, which depend on the fuel element material composition as well as the thermal hydraulic parameters.
These nuclear parameters are called state variables, which include fuel temperature, moderator temperature, moderator density, boron concentration and the fuel burn-up [2]. During the reactor operation the fuel composition will change as fissile isotopes are consumed and fissions products are produced, resulting in a different behavior of the

absorption cross section. These changes, in both space and time, which occur in the composition of the fuel, can be determinate by the fuel burn-up calculation.

Presently, analysis and studies of the macroscopic cross section, as a function of nuclear parameters, have shown very distinct behavior that cannot be represented by simply using linear interpolation. Indeed, a polynomial representation is more adequate for the data parameterization. Nevertheless, existing methods do not indicate explicitly the type of polynomial fit that best represents the problem-dependent cross section. The methods used to obtain these problem-dependent cross sections, in the reactor calculations, generally uses nuclear computer codes that require a large processing time and computational memory, making the process computationally very expensive. Therefore, new methods have been studied for the purpose of seeking alternative procedures to provide the cross sections of rapidly and safely without the dependence of complex systems calculations. Over the years, methods of few-groups cross section parameterization using mathematical processes such as stepwise regression [3], quasi-regression [2] and sparse grids methods [4], were elaborated and suggested.

It was developed [5] a study of the cross section parameterization using Tchebychev polynomials based on problem-dependent cross sections calculated with the SCALE code system for the zero cycle condition, i.e., the fresh PWR reactor. This methodology was used in this work to performing the analysis of homogenized macroscopic cross sections behavior in a PWR fuel elements as a function of nuclear parameters, which include fuel temperature, moderator temperature and density, soluble boron concentration, uranium enrichment, and the burn-up.

The purpose of this present work is complementing the preview analysis [5] with a study of macroscopic cross section behavior considering a burn-up cycle and the fission product influence in fuel composition. In this study is evaluated the problem-dependent about fission, scattering, total, nu-fission, capture, transport and absorption cross sections.

The results of the methodology presented in this paper are a set of parameterized cross sections based on the Tchebychev polynomials by fitting the cross sections as a function of these nuclear parameters. Tchebychev polynomials were chosen since they present some advantages when compared to other polynomials. Their parameterized function is the result of a method of minimizing error from a process that leads to better accuracy in the calculation of coefficients and in the estimate of polynomial fit at specific points. The cross sections parameterization can serve as an alternative for use in reactor calculations with few energy groups without the need to perform all the steps usually required for this type of calculation [6]. Although the method developed in this work is suited for a particular type of fuel element of a typical PWR reactor it can be extended to any other PWR fuel element.

2. METODOLOGY

Typical PWR reactor fuel element (FE) with 12 integrated burnable absorber fuel rods was analyzed with the SCALE 6.1 code package, more specifically, the Monte Carlo code KENO-VI, the TRITON depletion sequence (T6-DEPL), and NEWT transport code (T-NEWT). The geometry and configuration of the FE was modeled with KENO-VI. The T6-DEPL sequence was used to perform the burn-up-dependent nuclide concentration determining the isotopic

composition for the first cycle of burn-up. The problem-dependent macroscopic cross sections were generated by T-NEWT. The equations presented are functions of the following selected parameters: $^{235}_{92}U$ enrichment, moderator temperature, boron concentration, fuel temperature and moderator density, for a PWR nuclear reactor [7]. Problem-dependent about fission, total, scattering, capture, transport and absorption cross sections were obtained as a function of the selected parameters [3].

T-NEWT calculations were performed using the SCALE 238-energy-group ENDF/B-VII library, named V7-238, which was developed based on the evaluated nuclear data libraries, ENDF/B-VII.0 [8]. The results with the 238- energy groups were collapsed to two groups of energy. The ENDF/B-VII cross section data yield significantly more accurate calculated eigenvalues. Tests utilizing projects parameters were carried out at the end of the process to verify the efficiency of the method. The calculations done with the SCALE 6.1 code served to benchmark the results obtained with the method based on the cross-section parameterization [9]. The 16x16 PWR FE modeled with the KENO code is displayed in Fig. 1.

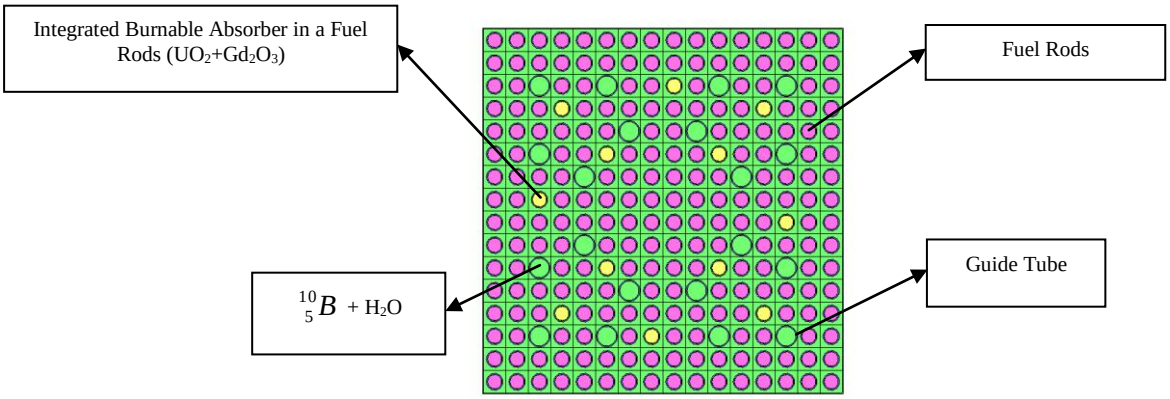

Figure 1: Fuel Element – Top View (KENO 3D) [5].

The first isotopic composition used in fresh reactor (zero cycle condition) was defined by four mixtures: UO_2 ($^{235}_{92}U$ enrichment 3 $^{w}/_{o}$), $^{10}_{5}B + H_2O$, $UO_2 + Gd_2O_3$ and Zircaloy. This composition present above was used as INPUT on the T6-DEPL three-dimensional (3-D), which was used to perform depletion calculations. It has the capability of simulating the depletion of multiple mixtures in a fuel assembly model, with allows a detailed representation of the local flux distribution for a specific fuel rod in the assembly to calculate fuel compositions.

In the first cycle fuel isotopic composition considered was generated by T6-DEPL, using 36 MW/MTU of burn-up. This composition consisted of 33 nuclides, 17 actinides and 16 fission products, which were important to fuel reactivity (i.e., nuclides with large neutron fission cross sections and nuclides with large neutron absorption cross sections). The 33 nuclides are the ones commonly considered in fuel compositions for burn-up credit criticality safety analyses that base validation of calculated nuclide concentrations on comparisons to available RCA (Radiochemical Assay) data [10].

A careful study was performed to determine the set of independent parameters to be used in the parameterization. The homogenized macroscopic cross section performed by T-NEWT was generated as a function of only one variable, keeping the other variables fixed with reference values. The reference values for the five-independent parameters are listed in Table 1.

Table 1: Reference Values

Nuclear Parameters	Reference Values
$^{235}_{92}U$ Enrichment	*3 $^{w}/_{0}$*
Boron Concentration	*500 ppm*
Fuel Temperature	*973 K*
Moderator Temperature	*573 K*
Moderator Density	*0.727084 g/cm^3*

3. CROSS SECTION PARAMETERIZATION

The cross section parameterization was carried out based on the Tchebychev polynomials since they have some inherent advantages [11] that are suited to our problem. These advantages of Tchebychev Polynomials are:

- The first order roots of polynomials of can be used in polynomial interpolation;
- They have the property of minimizing the error. The error is almost uniform for the entire range of analysis. Other polynomials also have these properties, but the Tchebychev polynomials are much more efficient;
- The shape of Tchebychev polynomials is preferable since it leads to a general improved accuracy;
- They have the property of accelerating convergence on a recurring process.

In this section we present the determination of the Tchebychev coefficients after an analysis of the correlation coefficient value, which means that when the value is minimum, the polynomial reaches the maximum fitting of the curve. In the process of searching for the polynomial description we wish to approximate the set of data points as closely as possible with a specified function (Tchebyshev polynomial), which is as smooth as possible. The smoothness requirement is met by limiting the number of coefficients allowed in the fitting function, for example, by restricting the polynomial degree in the case of Tchebyshev polynomial. Given a particular number of coefficients in the function in question, the fitting routines, like E01AEF from the Numerical Alghoritmic Group (NAG) library, determine the value of the coefficients such that the distance of the function from the data points is as small as possible. The necessary balance is decided by the user by comparing a selection of such fits having different numbers of coefficients. If the number of coefficients is too low, the approximation to the data will be poor. If the number is too high, the fit will be too close to the data, essentially following the random errors and tending to have unwanted fluctuations between the data points. A measure of the above distance between the set of data points and fitting function $f(y)$ is needed. The distance from a single data point (x, y) to the function

can simply be taken as $\epsilon_r = y_r - f(x_r)$, and is called the residual of the point. After the determination of the coefficients, the Tchebychev polynomials associated to each type of cross sections were written, for the fast and thermal energy groups. Throughout this paper, $f(y)$ means the cross section written as a function of a nuclear parameter as listed in Table 1 for the fast and thermal energy group. A set of polynomials functions for each type of cross section is constructed and the polynomials are written as follow:

$$f(y) = a_0 T_0(y) + a_1 T_1(y) + a_2 T_2(y) + \cdots + a_n T_n(y)\,, \tag{1}$$

where a_i is the Tchebychev coefficient, $T_i(y)$ is the polynomial of the first kind with Tchebychev argument y, with $T_0(y) = 1$; $T_1(y) = y$; $T_2(y) = 2y^2 - 1$; $T_3(y) = 4y^3 - 3$; $T_4(y) = 8y^4 - 8y^2 + 1$; etc.

The argument y varies in the range of –1 to +1 according to the expression:

$$y = \frac{2Y - Y_{min} - Y_{max}}{Y_{max} - Y_{min}}, \tag{2}$$

where Y is the value of the variable under study (nuclear parameter), Y_{max} and Y_{min} define the maximum and minimum values of among the analysis interval. For a point chosen within the analysis interval of any nuclear parameter considered, the values of the cross section were obtained with the polynomial functions. Then, the cross sections obtained directly with the SCALE system were compared with the results obtained by polynomial functions. In general the agreement between the polynomial interpolation and the SCALE calculations were very good.

The cross section representation for each parameter listed in Table 1 is presented. The results are indicated in Tabular form with the parameterized polynomials functions for the analyzed cross section as a function of state variables. In these tables $f(y)$ means cross section, y means argument and Y means nuclear parameter like fuel temperature, moderator temperature, moderator density, enrichment and boron concentration, where $T_F = \sqrt{t_f}$.

3.1. Cross Section as a Function of Fuel Temperature

After the determination of the Tchebychev coefficients the parameterized polynomials were written as a function of the fuel temperature considering a variation of only one variable and keeping the remaining variables fixed with the reference values. The values of fuel temperature (in Kelvin) used in this study to generate the cross sections were in the interval 573 K to 1473 K.

The behavior of the macroscopic cross sections represented by polynomial expansions is described in Table 2, whereas Groups 1 and 2 as the fast and thermal energy group, respectively. It can be observed that an increase in the fuel temperature leads to an increase of the resonant captures due to the Doppler Effect and consequently leading to an increase in the resonance absorption. The fission cross section for the fast group is not very sensitive to the variation in fuel temperature and is practically constant, whereas a rise in fuel temperature

leads to a decrease in resonance escape probability of moderated neutrons and then fission events in the reactor decrease with thermal neutrons.

Table 2: Cross-section as a function of the fuel temperature.

Cross Section	Group	Polynomials
Fission	1	0.00199
	2	$0.03945-0.00001y$
Total	1	$0.53646+0.00034y-0.00001y^2+0.00002y^3-0.00001y^4$
	2	$1.25352-0.00015y+0.00001y^3-0.00001y^4$
Scattering	1	$0.52708+0.00011y$
	2	$1.14846-0.00031y+0.00002y^3$
Absorption	1	$0.00942+0.00022y+0.00001y^3$
	2	$0.10506+0.00014y-0.00002y^2$
Capture	1	$0.00742+0.00023y+0.00001y^3$
	2	$0.06560+0.00016y-0.00002y^2$
Nu-Fission	1	0.00531
	2	$0.10343+0.00001y$
Transport	1	$0.21794-0.00005y$
	2	$0.75393+0.00005y-0.00003y^2+0.00001y^3$
Scattering 1-1	–	$0.50814+0.00029y-0.00001y^3-0.00001y^4$
Scattering 1-2	–	$0.01893-0.00017y-0.00001y^3$
Scattering 2-2	–	$1.14846-0.00031y+0.00002y^3$

3.2. Cross Section as a Function of the Moderator Temperature and its Respective Moderator Density

The parameterized cross sections were obtained considering the variation of the moderator temperature and its associated density, in the interval 300 K to 613 K and 1.003439 g/cm3 to 0.617629 g/cm3, respectively. The behavior of the macroscopic cross sections represented by the parameterized Tchebyshev polynomial expansions is described in Table 3.

An increase in the moderator temperature leads to a decrease in the moderator density. Since the volume of the moderator in the reactor core is essentially constant, the thermal expansion will lead to a decrease in the total number of moderator nuclei. The principal effect is the loss of moderation that causes an increase in resonance absorption so that there is an increase in the absorption rate and consequently leading to a decrease in the number of fissions in the nuclear fuel.

Table 3: Cross section as function of the moderator density and temperature.

Cross Section	Group	Polynomials
Fission	1	$0.00204-0.00004y-0.00001y^3-0.00002y^4$
	2	$0.04169-0.00265y+0.00053y^2-0.00062y^3-0.00094y^4$
Total	1	$0.61554-0.07736y-0.01118y^2-0.01645y^3-0.02778y^4$
	2	$1.54379-0.30950y-0.02232y^2-0.05041y^3-0.08690y^4$
Scattering	1	$0.60588-0.07711y-0.01116y^2-0.01634y^3-0.02761y^4$
	2	$1.42624-0.29492y-0.02302y^2-0.04829y^3-0.08315y^4$
Absorption	1	$0.00970-0.00026y-0.00001y^2-0.00010y^3-0.00016y^4$
	2	$0.11754-0.01458y+0.00069y^2-0.00212y^3-0.00374y^4$
Capture	1	$0.00765-0.00022y-0.00001y^2-0.00008y^3-0.00013y^4$
	2	$0.07585-0.01192y+0.00016y^2-0.00149y^3-0.00280y^4$
Nu-Fission	1	$0.00545-0.00012y-0.00005y^3-0.00008y^4$
	2	$0.10880-0.00633y-0.00142y^2-0.00166y^3-0.00246y^4$
Transport	1	$0.23678-0.01842y-0.00268y^2-0.00389y^3-0.00657y^4$
	2	$0.90827-0.15989y-0.01716y^2-0.02702y^3-0.04766y^4$
Scattering 1-1	–	$0.58216-0.07242y-0.01048y^2-0.01537y^3-0.02597y^4$
Scattering 1-2	–	$0.02372-0.00468y-0.00068y^2-0.00097y^3-0.00164y^4$
Scattering 2-2	–	$1.42624-0.29492y-0.02302y^2-0.04829y^3-0.08315y^4$

3.3. Graphical Comparison of Cross Section Behavior as a Function of Fuel and Moderator Temperature

The comparative analysis of macroscopic cross section behavior with increasing fuel and moderator temperature was plotted in Figs. 2 to 7, where is possible to verify the physical phenomenon from the polynomial description. In this graphics Σ^F represents the macroscopic cross section related to the variation of the fuel temperature, represented by a solid line, and Σ^M represents the macroscopic cross section related to the variation of the moderator temperature, represented by a dotted line.

The Figures. 2 and 3 show the variation of the macroscopic scattering cross section, considering changes in fuel and moderator temperature, for fast and thermal energy group, respectively.

An increase in moderator temperature reduces densities and macroscopic absorption and scattering cross section, corresponding in an increase in the mean-free-path and therefore leakage. The leakage of neutrons increases with the increasing moderator temperature. When the moderator density decreases and more fast neutron escape, the number of neutron with are moderated to thermal energy also decreases. Thus a decrease in moderator density leads to a decrease in thermal neutron absorption cross section in the moderator, and to a decrease in the moderation of the fast neutrons.

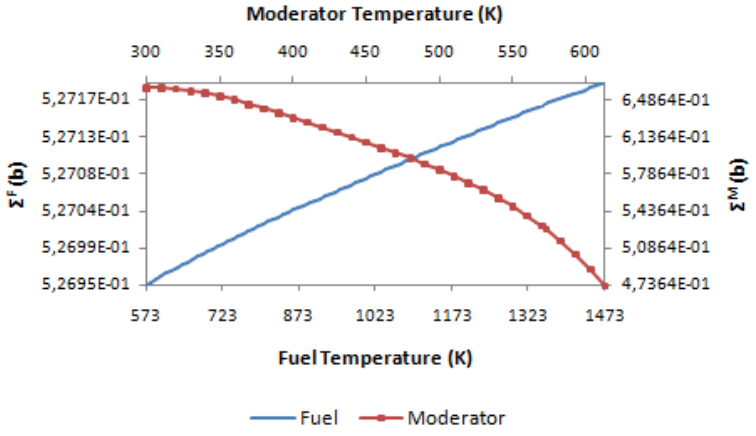

Figure 2: Fast scattering cross sections as function of fuel and moderator temperature.

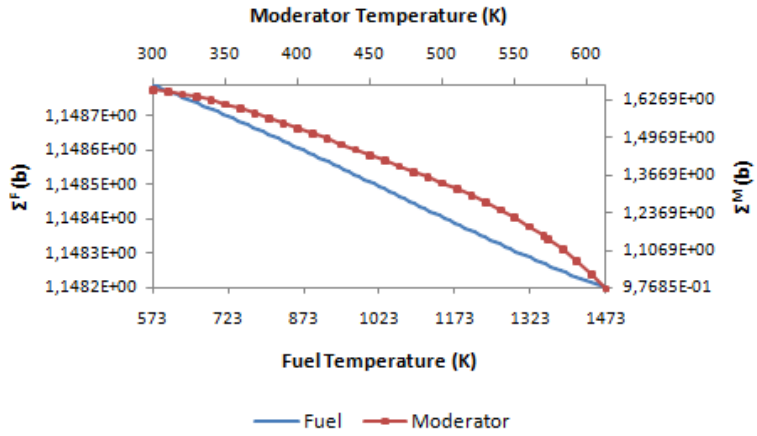

Figure 3: Thermal scattering cross sections as function of fuel and moderator temperature.

The Figs. 4 and 5 show the variation of macroscopic fission cross section behavior for fast and thermal group, respectively.

It is observed by the polynomial parameterized description, in Figs. 4 and 5, that as the moderator temperature increases, a decrease of fission cross section for both fast and thermal group occurs. The result is expected due to a reduction on the moderate neutrons. The analog fact is observed when fuel temperature increases.

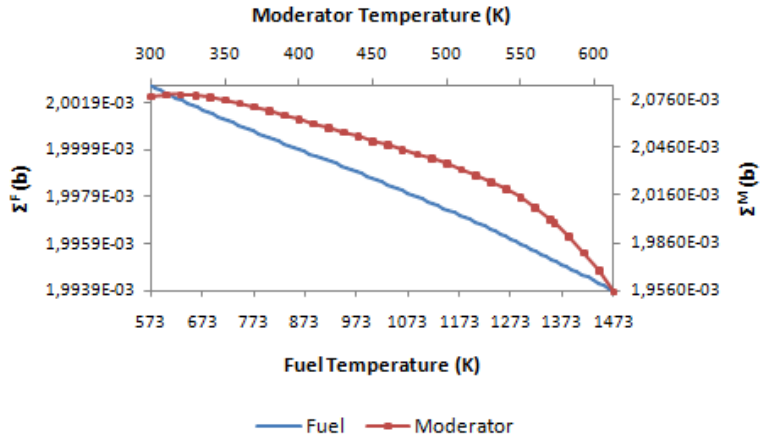

Figure 4: Fast fission cross sections as function of fuel and moderator temperature.

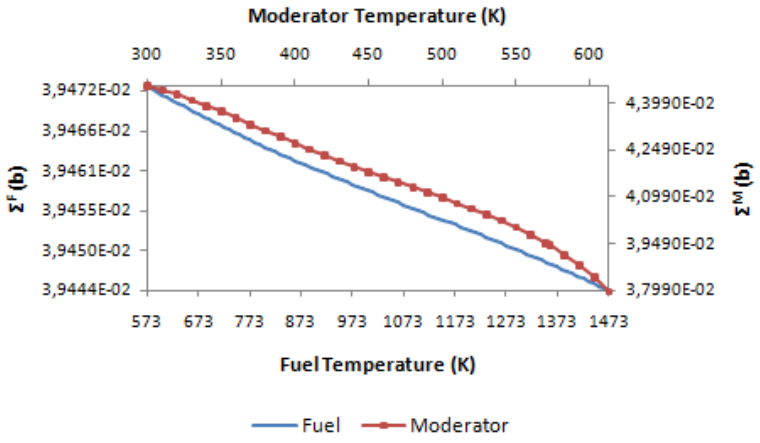

Figure 5: Thermal fission cross sections as function of fuel and moderator temperature.

The Figures. 6 and 7 show the variation of macroscopic capture cross section, considering changes in fuel and moderator temperature, for fast and thermal energy group, respectively.

An increase in the fuel temperature will lead to Doppler-broadening of resonances with a corresponding decrease in self-shielding. This width of the resonances broaden is called the Doppler Effect and arises primarily from capture resonances in fertile material in the fuel, $^{238}_{92}U$ is the principal contributor early in the core life, although $^{240}_{94}Pu$ becomes important later in core life. As a consequence of the Doppler Effect, resonance peaks are broadened, and neutron absorption in the resonance region is increased by an increase in temperature. The increase in the width of the resonance peaks has the effect of increasing neutron absorption in the interior of the fuel rods, thereby contributing to the decrease in the resonance escape probability with increasing temperature. The Doppler-broadening effect leads to increase parasitic capture on neutrons as fuel temperature increases. This important mechanism adds considerably to the safety of the reactor.

The decrease of capture cross section, for fast and thermal group, with the increase in the moderator temperature is expected due to the decrease in the moderator density.

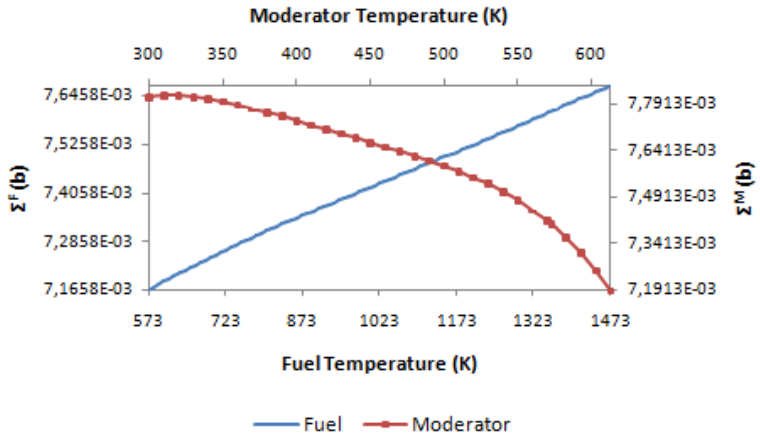

Figure 6: Fast capture cross sections as function of fuel and moderator temperature.

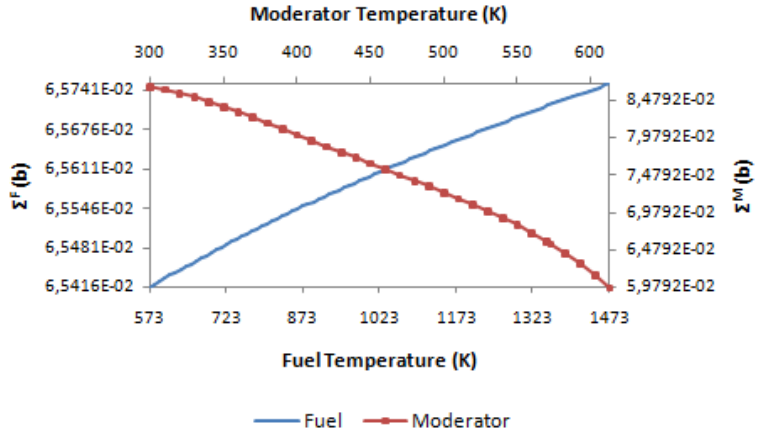

Figure 7: Thermal capture cross sections as function of fuel and moderator temperature.

4. NUMERICAL VALIDATION

The efficiency of the method developed based on the cross section parameterization is tested with results obtained with direct calculations using the SCALE code system. Comparison test results are presented for the analysis of project parameters such as the temperature coefficient of reactivity $\rho(T)$ and fast fission factor $\epsilon(T)$, calculated using results from the methods developed and results from the SCALE code system.

4.1. Comparison of Results Between SCALE and Developed Method Based on Cross Section Parameterization

As an example, the fuel temperature value of 1248 K was chosen to compare the results of SCALE calculations and results based on the cross-section parameterization. The remaining

variables were kept with the reference values. Two energy-groups cross section, called fission, total, scattering, transport, nu-fission, absorption and capture cross sections were calculated using the cross-section parameterization method and using the SCALE code. For this analysis an enrichment of 3 $^{w}/_{o}$, a moderator temperature of 573 K, a boron concentration of 500 ppm has been used. The results of the comparisons are shown in Table 4. It can be seen that the differences between the cross section parameterization methodology and the direct calculations based on the SCALE code system are less than 0.03 percent.

The results of Table 5 suggest that the methodology presented here can be used as an alternative to the SCALE calculations.

Table 4: Comparison of results between SCALE and developed method.

Cross Section	Group	Interpolation	SCALE	Deviation
Fission	1	*0.00199*	*0.00199*	*0.00403%*
	2	*0.03945*	*0.03945*	*0.00018%*
Total	1	*0.53663*	*0.53663*	*0.00072%*
	2	*1.25344*	*1.25344*	*0.00067%*
Scattering	1	*0.52713*	*0.52713*	*0.00006%*
	2	*1.14830*	*1.14831*	*0.00027%*
Absorption	1	*0.00954*	*0.00954*	*0.02785%*
	2	*0.10513*	*0.10513*	*0.00024%*
Capture	1	*0.00754*	*0.00754*	*0.03628%*
	2	*0.06568*	*0.06568*	*0.00028%*
Nu-Fission	1	*0.00531*	*0.00531*	*0.00360%*
	2	*0.10344*	*0.10344*	*0.00023%*
Transport	1	*0.21791*	*0.21791*	*0.00052%*
	2	*0.75395*	*0.75395*	*0.00025%*
Scattering 1-1	–	*0.50829*	*0.50829*	*0.00050%*
Scattering 1-2	–	*0.01884*	*0.01884*	*0.01148%*
Scattering 2-2	–	*1.14830*	*1.14831*	*0.00027%*

4.2. Comparison of Temperature Coefficients of Reactivity and Fast Fission Factor Calculated Using the Cross Section Parameterization Method and Results from the SCALE Code System

Reactor temperature affects the overall thermal-neutron energy spectrum and therefore nuclear cross sections. Temperature variations in a reactor core will affect core multiplication, both because of the resulting changes in core components that change macroscopic cross sections, and because of a change in the thermal motion of core nuclei which changes microscopic cross sections. The two dominant temperature effects in most reactors are the change in resonance absorption (Doppler effect) due to fuel temperature changes, and the change in the neutron energy spectrum due to changing moderator or coolant density (due to temperature, pressure, or void fraction changes).

From the standpoint of reactor safety and control there is no better parameter of importance than the variation of reactivity $\rho(T)$ with core temperature. This variation is usually expressed in terms of a temperature coefficient of reactivity $\alpha(T)$ defined as the derivative of

the core reactivity with respect to temperature [12]. The temperature coefficient of reactivity was used to test the accuracy of the cross-section parameterized methodology.

Table 5 displays the temperature coefficient of reactivity obtained through the variation of reactivity calculated from the k_{inf} values for temperatures $T_2 = 600\ K$ and $T_1 = 590\ K$. In this work $\rho(T)$ is expressed in $\Delta k/k$. We also compared the k_{inf} calculated with the SCALE code system and by cross-sections parameterization methodology.

Table 5: Comparison between temperature coefficients reactivity from Tchebychev Polynomials and SCALE system.

Results	$k_{inf}(T_2)$	$k_{inf}(T_1)$	$\rho(T_2)$	$\rho(T_1)$	α_T
SCALE	*0.85578*	*0.85597*	*-0.16851*	*-0.16825*	*-0.00003*
Tchebyshev Polynomials	*0.85559*	*0.85586*	*-0.16878*	*-0.16840*	*-0.00004*
Deviation (%)	*0.02237*	*0.01317*	*0.15512*	*0.09149*	–

The negative temperature coefficient of reactivity indicates an increase in temperature that causes a decrease in $\rho(T)$, hence a decrease in reactor power and temperature which tends to stabilize the reactor power level, becoming a desirable feature since it is an important factor in the reactor stability and operational safety. A reactor with negative temperature coefficient is therefore inherently self-regulating and safe. The percentage difference between the two calculations is very small which indicates that the cross-section parameterization methodology reproduces very well the direct calculations using the SCALE code system.

Another important parameter is the fission fast factor $\epsilon(T)$, which is the relation between the total number of fission neutrons, from both fast and thermal fission, and the number of fission neutrons from thermal fissions. This factor takes account of the fact that, although most fissions will be induced in fissile material by thermal neutrons, some fissions will be induced in both fissile and fissionable materials by fast neutrons [12]. The fast fission factor ϵ is usually quite close to unity in a thermal reactor with typical values ranging between $\epsilon = 1.03$ and $\epsilon = 1.15$.

Table 6 show the fast fission factor calculated using results from the methods developed and results from the SCALE code system, for temperatures $T_2 = 600\ K$ and $T_1 = 590\ K$.

Table 6: Comparison between fission fast factor from Tchebychev polynomials and SCALE system.

Results	$\epsilon(T_2)$	$\epsilon(T_1)$
SCALE	*1.0515168*	*1.0515165*
Tchebyshev Polynomials	*1.0515114*	*1.0515141*
Deviation (%)	*0.0005*	*0.0002*

The results presented in table 6 shows that the values of fast fission factor is within the range accepted and presents deviations smaller than 0.0006 percent, approving the use of parameterized equations in the cross-section calculation.

5. CONCLUSIONS

The work presented in this paper describes a methodology for the parameterization of problem-dependent cross section as a function of reactor parameters as the fuel temperature, moderator temperature and density, $^{235}_{92}$U enrichment, boron concentration, and burnup for a PWR nuclear reactor. The method is offered as an alternative to cross section determination using few energy groups in PWR fuel element calculations. Another important advantage of the cross section parameterization methodology is that once the cross-section data have been parameterized they can be stored and easily used in very simple computer devices for fast calculations. The results of comparison with direct calculations with the SCALE code system and also the test using project parameters show excellent agreements. Although the methodology has been derived for a particular reactor configuration the method can be easily extended for other reactor configurations.

The results presented in this work come from the first burn-up cycle, making it necessary for the next work, a more comprehensive analysis, taking into account the development of the cross section behavior from the zero cycle condition to the stability of the reactor. It would also be interesting to analyze the influence of fission products in the reactor criticality factor.

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