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Two Dimensional Burn-up Calculation of TRIGA Core

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Abstract - TRIGLAV is a new computer program for burn-up calculation of mixed core of research reactors. The code is based on diffusion model in two dimensions and iterative procedure is applied for its solution. The material data used in the model are calculated with the transport program WIMS. In regard to fission density distribution and energy produced by the reactor the burn-up increment of fuel elements is determined. In this paper the calculation model of diffusion constants and burn-up calculation are described and some results of calculations for TRIGA MARK II reactor are presented.

1 Introduction

A new program TRIGLAV is developed for reactor calculation of mixed core of research reactors. The main goal of the program is the calculation of the fuel elements burn-up. The program is written in the modular form. It was originally developed for PC, VAX version is available as well. It consists of the main program and input data files. The program needs two different types of input data:

- the independent data where geometry of reactor, location of fuel elements, operation conditions and burn-up time step are specified and
- the material composition and burn-up data of each fuel elements.

The calculations are stepwise. In the first step, the homogenized effective group constants are calculated using multigroup transport code WIMS. The global calculation is performed in the second step using multigroup diffusion approximation. The results of calculations are flux and power distribution and multiplication factor of the system. On the basis of power distribution and total energy produced by the reactor, the burn-up increment for each fuel element is calculated in the last step.

Special attention is put on the calculation of the homogenized effective group constants. The transport calculations using program WIMS are performed for all core components. The average unit cell for fuel and non fuel elements is defined to simplify the geometry model of the unit cell. Some simplifications for calculations of operation parameters were also introduced in the model (fuel and moderator temperature, burnup). The temperature of the fuel is presented as a function of power and the temperature of water is taken as constant. The burn-up of the fuel elements is taken into account through repetition of the transport calculation in chosen time steps and at certain specific power. Xe correction is also taken into

account. Some results of the calculations for selected benchmarks are presented. The calculation of the fission density for simplified TRIGA geometry is made. The results show good agreement with reference results (maximal relative error is less than 2%).

2 Cross Section Calculation

The reactor core is in general a complex system including fuel elements, control rods, cooling material, construction elements, reflector etc. Because of its complexity, the calculations of reactor core are usually divided into two steps.

In the first step the effective macroscopic cross sections for all reactor components (such as fuel elements, control rods, irradiation channels, reflector) are calculated. Different transport approximations are used for calculations such as the method of collision probability, S_n method or P_{ij} method. Transport computer programs are usually used for solving complicated equations. One of them is the well known transport program WIMS [1].

The basic physical model for calculating effective macroscopic cross section in the program WIMS is the model of unit cell approximation. In this model fuel rod and surrounding water are treated explicitly in an infinite array of identical unit cells. The geometry of fuel unit cell is equal for all types of fuel elements (standard fuel elements, FLIP, LEU elements), the differences appear only in the material composition. For calculations of non-fuel components (graphite elements, control rods, irradiation channels etc.) the supercell approach is used. In this model non-fuel cell is surrounded by six fuel rods. The reflector cross sections can be calculated in unit cell model, where the homogenized core is proposed and reflector region is added at the outer boundary. It is recommended that the reflector region is as thick as possible and that zero flux boundary condition is used.

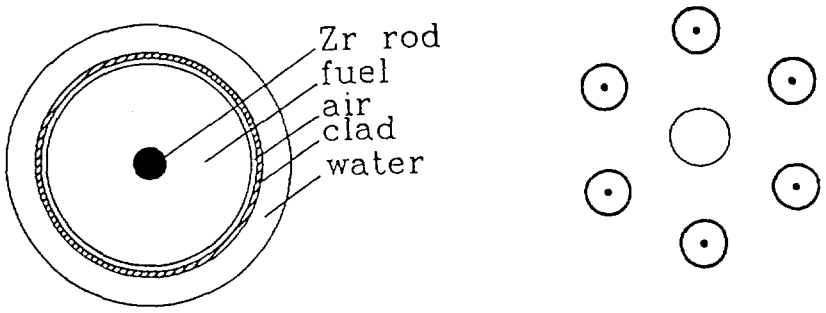

Figure 1: The fuel unit cell (a) and supercell with water cell in the center (b).

The program WIMS includes also the cross section library for all neutron reactions of reactor materials except for the moderator ZrH and for the burnable absorber Er

which are not characteristic materials used in power reactors. For research reactor calculations diffusion constants for both materials were processed [2] and added to the original library of the program WIMS. The cross sections are treated in 69 energy group structure between 0 to 10 MeV.

The burn-up calculations of unit cells are also possible with the program WIMS. These are made with the repetition of the transport calculations in prescribed burn-up time steps and at specific power of element. The power 10 kW per fuel element is proposed and the time step of burn-up is determined in regard to input burn-up data of fuel element.

The final results of the transport calculations are homogenized group constants for all elements in the core. They are in general functions of few variables:

$$\Sigma = \Sigma(T_f, T_m, P, \tau) \tag{1}$$

where

T_f is the temperature of the fuel,

T_m is the temperature of the cooling water,

P is the power of the reactor and

τ is the burnup of the reactor core.

The production of energy in the reactor changes the temperature of the fuel and the coolant. The neutron behavior of the reactor core is changed, as is the cross section of the materials.

Increasing temperature of fuel causes Doppler broadening of the U-238 resonances and also changes the neutron spectrum. Cooling water density decreases with increasing temperature (the effect is small and its influence on the neutron behavior is neglected, changing the water density by 1% changes the multiplication factor by $6 \cdot 10^{-4}$ [3]).

The temperatures of fuel and coolant are functions of the reactor power. The relation between them is given by termohydraulic equations. Various computer programs exist for calculation of these relation. Relation between the temperature of fuel and the reactor power obtained from experimental measurements is proposed in our program [4]. The temperature of cooling water is taken as a constant.

The equilibrium concentration of some absorption elements in the reactor core is also the function of the reactor power. Typical representatives are Xe and Sm. Equilibrium concentration of Xe builds up after a few hours of reactor operating. Small concentration of Xe already changes the thermal cross sections and spectrum, so special attention is paid to calculation of its concentration.

The temperature of reactor materials, equilibrium concentrations of absorber materials and burnup of reactor core are in general functions of reactor power. While the power is determined with the neutron flux into the reactor, we can conclude that the cross sections are the function of the neutron flux.

The cross section calculation model used in program TRIGLAV is presented in the Figure 2. At start of the calculations the power distributions $P_R^0(r)$ is proposed. At this power the fuel temperature is then calculated. Regarding the input data of

unit cells (geometry, composition, burn-up, water temperature) and the calculated fuel temperature calculation of homogenized multigroup diffusion constants with the program WIMS is performed. The results of cell calculations are multigroup diffusion constants for each unit cell in the reactor. Because of the extension of following calculations the multigroup cross sections are rather condensed into few group cross sections (usually in two to four groups, typical group boundaries in four group structure are 10 keV, 1 eV and 0.1 eV). The group collapsing involves averaging of the homogenized cross sections over the spectrum only, which appears in the WIMS output together with the cross sections. The computer code for collapsing WIMS homogenized cross sections to arbitrary few group structure is the program XSWOUT [8], a part of program TRIGLAV.

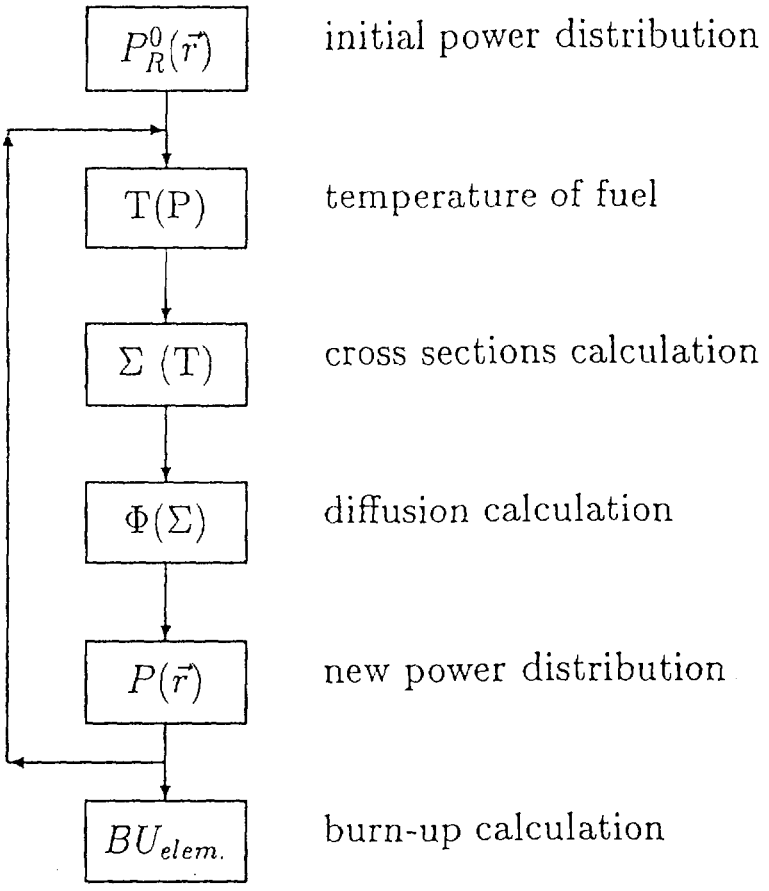

Figure 2: Flow chart of the cross sections calculations in the program TRIGLAV.

3 Diffusion calculations

In the second step diffusion calculation of whole reactor core is performed. Physical model of core calculation in the program TRIGLAV is based on two dimensional (r, ϑ) TRIGA geometry in four group diffusion approximation. The equations are solved numerically in the process of inner and outer iterations. The results of diffusion calculations are flux and power distribution, and multiplication factor of the

Working equations and numerical calculation procedure are given in [6], [7].

4 Burnup calculations

After the power distribution calculation $P_{elem.}$ has been performed, the burnup increment of fuel elements, $\Delta BU_{elem.}$, is calculated by equation:

$$\Delta BU_{elem.} = P_{elem.} \Delta t, \tag{2}$$

$$P_{elem.} = \alpha_p \frac{c}{\nu} \int_{V_{elem.}} \nu^g \Sigma_f^g \Phi(\vartheta_i, r_j) dV, \tag{3}$$

where α_p is the power normalization factor, $c = 3.2 \cdot 10^{-11} J$ is the fission energy and $\nu = 2.45$ is the average number of fission neutrons produced per fission of U^{235}. Δt is the reactor operating time at reactor power P. The burn-up unit is MWd (MW day), while in practical work the unit of burn-up of U^{235} nucleus in % is employed. The relation between both units is not linear because of plutonium production (the relation is presented on Figure 3).

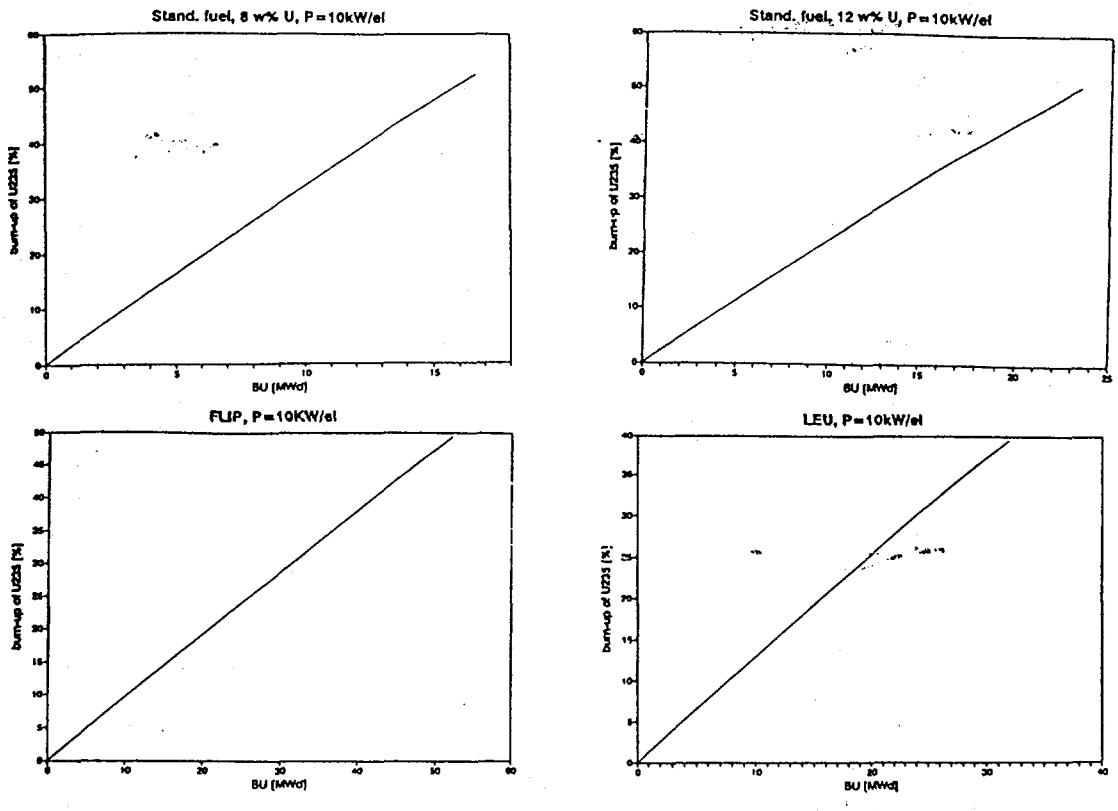

Figure 3: Relation between burn-up units calculated in MWd and in % of burn-up of U^{235} nucleus, for typical fuel elements in research reactor.

The calculated increment of the burn-up for each fuel element is summed to the initial burn-up of the element. In this way core burn-up evidence is managed.

5 The Reliability of Calculations

The accurate calculation of fission distribution is important for burn-up calculations. Some tests of diffusion calculations in two dimensional geometry have been made, including test described bellow. The test represents the hypothetical core configuration of TRIGA reactor in (r, ϑ) geometry (Figure 5).

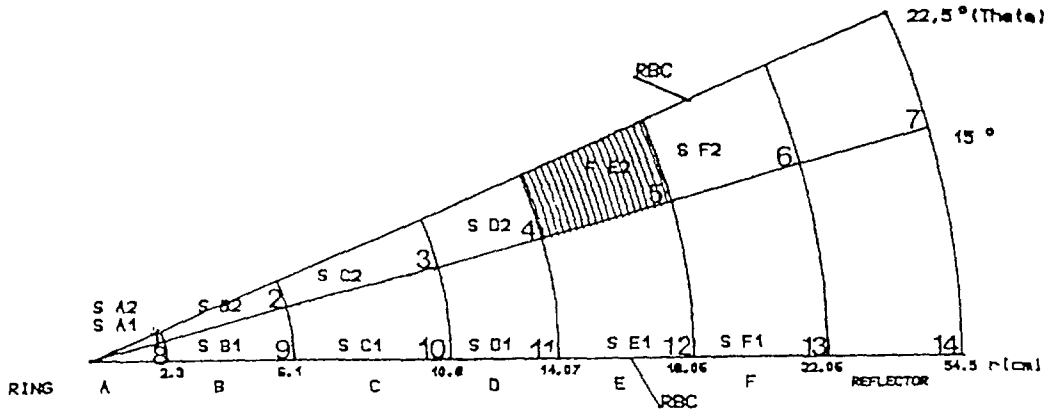

Figure 4: Geometry and material composition for 2D test: S-standard fuel element, F-FLIP fuel element, RBC- reflective boundary condition.

The core contains material corresponding to standard TRIGA fuel element with 8.5 w% of 20 % enriched uranium, while in E ring of the core every third fuel element is a FLIP fuel element (the case simulates isolated FLIP fuel element in a uniform matrix of standard elements). The reference solution represents the calculations of diffusion program FINELM [9]. Two group diffusion constants are taken from the program *FINELM* library. The mesh of 100 x 160 points $(rx\vartheta)$ is used in numerical calculations. The prescribed convergence criterion are 10^{-7} for inner calculations, 10^{-3} for fission density and 10^{-5} for multiplication factor. The results of multiplication calculations are presented in Table 1.

Table 1: Calculation of multiplication factor

program	outer iterations	$k_{eff.}$	difference [pcm]
FINELM	14	1.16034	/
TRIGA2D	25	1.16027	7

While the accuracy of FINELM's result is $\pm$ 10 pcm, the difference 7 pcm in calculation shows good agreement of the result.

Another important integral parameter of the reactor is the power distribution. The results of power calculations are presented in Table 2. They are normalized in the way that the sum of element's fission density over all elements equals 1. From the results it is obvious that the maximal differences in calculations appear on the boundary of two different materials (such as standard fuel - FLIP fuel or fuel -

reflector boundary), where there is a strong gradient of flux. The error is still inside the error interval acceptable for core design calculations (less than 2 %).

Table 2: Comparison of normalized fission density distribution for elements indicated on the Figure 4

element number	type	reference value FINELM	calculated value TRIGA2D	relative error of F_{el} [%]
1	standard	0.005	0.005	0.0
2	standard	0.031	0.031	0.0
3	standard	0.056	0.056	0.0
4	standard	0.067	0.067	0.0
5	FLIP	0.136	0.134	1.5
6	standard	0.074	0.075	-1.4
7	graphite	0.000	0.000	0.0
8	standard	0.010	0.010	0.0
9	standard	0.061	0.061	0.0
10	standard	0.113	0.112	0.0
11	standard	0.143	0.142	0.9
12	standard	0.145	0.144	0.7
13	standard	0.160	0.161	-0.6
14	graphite	0.000	0.000	0.0

6 Conclusion

The new computer program TRIGLAV for research reactor calculations, developed at "J. Stefan"Institute has been presented. The advantage of the program is that it is a 2D diffusion program and it can be applied to mixed reactor core configuration calculations. Basic physical models for cross sections and burn up calculations are described. Some results of diffusion calculations of hypothetical TRIGA core configuration are given.

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