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Research Reactor Calculations

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RESEARCH REACTOR CALCULATIONS

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Basic principles and physical models for practical research reactor calculations are presented using widely available computer codes. Core and fuel management calculations of plate-type and TRIGA reactors are considered.

1 Introduction

The utilisation of reactor calculations at research reactor centres started approximately 10 years ago with appearance of cheap small computers. It shows rapidly growing trend ever since. Several centres have already passed the basic level and perform reactor calculations which are of practical value: detailed burn-up determination, core optimisation for production of particular isotopes, power distribution analysis of mixed cores, etc. Reactor calculations have become standard part of the development programs at research reactors and are often initialised already together with the construction of a new reactor.

Reactor calculations are particularly important for ageing reactors, where typical problems such as safety analysis report upgrading, core conversion, fuel storing and shipment, normally involve practical reactor calculations.

The purpose of this presentation is to describe reactor calculations which can be performed by the staff of a small research reactor with the computer codes available through the IAEA and NEA CPL. Emphasis is on typical reactor physics codes which are frequently used for research reactor calculations: WIMS (1), TRIGAC, TRIGLAV (2) and BINODE (3). WIMS is a general lattice cell code and may be in principle used for any thermal type research reactor. TRIGAC and TRIGLAV are developed for TRIGA reactor fuel management calculations in 1-D and 2-D cylindrical geometry, respectively. BINODE is a general multigroup diffusion code based on nodal method for X-Y-Z geometry. In combination with WIMS it is used for MTR calculations.

The presentation is divided into two parts. General principles, the purpose and applications of research reactor calculations are presented in the first part. Physical description of two common research reactor types is presented in the second part. It is the basis for the physical models and the source of data for practical calculations which are also treated in the second part.

2 General Principles and Application of Research Reactor Calculations

2.1 Computer Codes for Research Reactor Calculations

Practically all standard reactor physics codes can be installed on modern PC computers without simplifications of the mathematical models. Numerical accuracy is normally not affected as well.

Classification with respect to their computer related requirements, which was normal during early stages of user oriented research reactor calculations is no longer relevant. It is more appropriate to qualify the computer codes with respect to the practical requirements from the aspect of the user:

- physical model (adequacy of geometry, group structure, etc. for particular application)
- method of solution (complexity of the physical model)
- flexibility (application for non-standard problems)
- performance (running time, numerical accuracy, user friendliness, presentation of results).

Some typical reactor physics codes are presented in Table 1. Their applicability for research reactor calculations is estimated with respect to the above criteria (4).

Table 1: Applicability of typical reactor physics codes for research reactor calculations

Code	Reactor	Complexity	Flexibility	Performance
Cross section processing codes				
NJOY		H	M	H
Unit-cell codes				
WIMS	thermal reactors	H	H	H
LEOPARD	thermal reactors	H	M	M
Diffusion calculations				
EXTERMINATOR	MTR	M	H	L
SIXTUS	hexag. TRIGA	H	M	H
BINODE	MTR	M	H	H
Integrated packages				
TRIGAC	TRIGA	L	L	H
TRIGLAV	TRIGA	L	L	M
2DB-LEOPARD	MTR	M	M	M
Monte-Carlo				
MCNP	TRIGA	H	H	L

L = low
M = medium
H = high

The performance of the codes can be estimated only in a relative sense, i.e. only the codes of the same type may be compared. In general, performance of the codes depends on the effectiveness of the mathematical methods. The largest differences in the effectiveness are observed with the diffusion codes: the codes based on modern nodal methods are approximately one hundred times faster in criticality calculations than the codes, based on the finite differences. However, the nodal computer codes are normally designed only for simple geometry (e.g. Cartesian). For complicated geometry like annular TRIGA with non-periodic lattice, only finite differences methods can be applied.

The complexity of the code depends mainly on the nature of the physical problem it is designed for: the codes based on the transport equation (e.g. WIMS) are far more complex than the codes for diffusion calculations.

The number of integrated packages designed for research reactor calculations is small. The advantage of the packages is their simple use and high computational performance. The drawback is their non-generality due to built-in assumptions on reactor geometry and composition.

Large computing capacity of modern PCs (e.g. Pentium generation) makes feasible also Monte-Carlo calculations of complicated systems, e.g. spent fuel storage, complete core geometry. However, the calculations are still not feasible for power operation and burn-up conditions and are for this reason limited to benchmark and safety studies.

2.2 Applications of Research Reactor Calculations

The extent and the scope of the reactor calculations depend on the particular needs and experience of the reactor staff. However, main needs for reactor calculations stem from operational problems which are very similar for all research reactors. A list of the most common practical applications of reactor calculations is provided in Table 2. The applications are divided into three levels according to the sophistication of the computer codes and experience required from the user: basic, advanced and design level.

Table 2: Practical applications of reactor calculations

Application	Complexity
Basic level	
power and flux distributions	L
fuel element burn-up	M
fuel element burn-up	M
Advanced level	
reactivity coefficients	M
spectrum calculations	M
core optimisation	H
inverse kinetics-reactivity measurements	H
pulse experiments (if applicable)	M
Design	
cross-section processing	H
benchmark experiments	H
reactor redesign and upgrading	H
Monte-Carlo calculations	H
transients	H

L = low
M = medium
H = high

The basic level applications may be performed using either integrated packages (e.g. TRIGAC) or simple combinations of basic computer codes: WIMS+BINODE, WIMS+EXTERMINATOR. Results of these calculations are normally used only for fuel management purposes and have no implication on reactor operation and safety. Relatively low accuracy and reliability of the calculations may be tolerated at this level.

Only after developing the basic level, applications of the reactor calculations may be brought to the advanced level where the reactor physics phenomena simulated by the calculations are much more complicated. Standard integrated packages may still be used for certain applications on this level (e.g. TRIGAC for core optimisation), but in general, case dependent combinations of the codes must be used or even new programs have to be developed (e.g. inverse kinetics). Results at this level may have influence on reactor operation and safety (e.g. reactivity coefficients if they are calculated to be used in transient analysis). High level of accuracy and reliability is required.

The design level calculations are normally not performed at small research reactors. The team and the experience required for the design level calculations are significantly larger than in case of first two levels. Organisation of work is different as well. The quality assurance (QA) program has to be implemented due to the importance of the results, introducing several new elements: formal organisation of work, documentation, quality control. Quality control is normally performed by cross-checking and repeating the calculations by two or more independent workers.

The applications in Table 2 are graded with respect to the complexity of the calculations they involve. The applications exceeding basic level are normally also relatively complex. They require use of several different computer codes, modifications of the codes, modifications and extensions on the effective cross-sections libraries and use of special input options. Only the basic level applications are treated in this presentation.

3 Physical Models

3.1 Physical Parameters of Research Reactors

The decision which computer codes are appropriate for particular application depends on the geometry, material data and operational parameters of the reactor. This information is needed also for defining the physical models and for preparing the input data of the calculations. First step in reactor calculation is therefore collecting material, geometry and operational data of the reactor. The task is normally not trivial if we try to collect 'as built' and not just typical or generic data of particular reactor. The set of data required for the calculation depends on the computer code and on the problem which is solved. The data for WIMS unit cell burn-up calculation are presented in this paragraph for TRIGA and MTR. The diffusion calculations require hardly no additional data except general reactor geometry and dimensions. They are provided as well.

All relevant geometry and material data should be in principle contained in the Final Safety Analysis Report of the reactor. In practice, only part of the information is found there. It is also not very reliable and accurate since the reactor description in SAR is often based on generic and not on 'as built' data. The most reliable source of practical data is the design documentation of the reactor (plans, blueprints, drawings, fabrication specifications). It contains normally detailed data on geometry but only general data on material specifications. The material data are normally

found in internal reports of the reactor manufacturer or in general literature. Such data are, however, also mainly generic and normally only approximately correspond to the particular case.

The exception are the data about the enrichment and weight of uranium which are part of the safeguard documentation and are for this reason in details provided together with the fuel elements. The rest of the material data (e.g. material density, metallurgical composition in case of alloys, impurities important for neutrons, concentration of burnable poisons,...) are normally not available for the particular reactor, especially if the reactor is old.

Developing a complete and consistent material and geometry database for reactor calculations of a particular reactor is therefore a tedious and time consuming task. Attempts were made by the IAEA to prepare such sets of data at least for the most common reactor types, usually within the framework of co-ordinated research programs involving research reactor calculations. A good compilation of data for research reactor calculations can be found in ref. (5). Here we present data for 2MW MTR and TRIGA-Mark II reactors, for which we discuss also the physical models later in the text.

The reactor core geometry, i.e. distances between the fuel elements, positions of the control rods etc., is practically defined by the fuel element support plates. Schematic drawing of 250 kW TRIGA top grid plate is presented in Fig.1. The dimensions of the core and reflector are presented in Table 3.

Typical 2 MW MTR geometry (5) is presented in Fig. 2 and in Table 4. Schematic drawing of TRIGA and MTR fuel elements is presented in Fig.3, corresponding material data and dimensions are given in Tables 5 and 6.

The data presented in Tables 3 - 6 are complete for preparing the geometry and material input specifications for reactor calculation. The only parameters which remain to be defined are temperature and specific power. They are particularly important in calculations of temperature reactivity effects and burn-up, respectively. If the calculations are performed for zero power conditions the definition is trivial since temperature is equal for all materials and regions considered in the calculation.

Table 3: TRIGA Mark II reactor geometry

number of rings			6
total number of grid positions			91
ring		radius [cm]	number of positions
A		0	1
B		4.05	6
C		7.98	12
D		11.95	18
E		15.92	24
F		19.89	30
reflector	outer diameter [cm]		109
	inner diameter [cm]		44.1
	height [cm]		55.9
	Al cladding thickness [cm]		0.62

For power operation conditions, the temperature of the fuel is different from water temperature, they both depend on the local power density, properties of the cooling channels, water inlet temperature, cooling mode, etc. In principle they are not related to the fuel specific power in a unique way. For example, fuel and water temperature is higher at the top of the fuel element than at the bottom even if the power density at these two points is equal.

This effect is however not very strong in research reactors and may be neglected, particularly in 1-D and 2-D calculations. We often assume that fuel and moderator temperature are functions of the fuel element power. The relation is normally derived from the empirical data and depends on the fuel and reactor type, mode of cooling (natural convection or forced cooling) and all other parameters determining the heat transfer conditions in a particular reactor. A typical correlation used in TRIGAC code for TRIGA Mark II reactor cooled by natural circulation is presented in Fig.9.

3.2 Physical Models for Unit-Cell Calculations

The global reactor calculation is normally performed in two steps. The homogenised effective group constants are calculated in the first step using multigroup transport approximation. The global reactor calculation is performed in the second step using multigroup diffusion approximation. The definition of the homogenised regions depends on the reactor geometry and on the flexibility of the geometry modelling in both steps. The lattice-cell codes which are normally applied in the first step (e.g.WIMS) are per definition designed for the geometry of a single unit cell or at most for the geometry of a cluster of unit cells. The homogenised regions in diffusion calculation must be for this reason defined such that they correspond to the unit cells. If the reactor core is periodical lattice of fuel elements, the unit cells are equal to the lattice cells with fuel elements in the centre. In this case the shapes of all unit cells are equal.

The core geometry of TRIGA reactor presented in Fig.1 is not periodical and the unit cells are not equal. In principle each unit cell should be modelled separately. However, for practical reasons and because the unit cells are normally cilindrised in the lattice-cell calculation, we define the average unit cell by dividing the core volume with the number of unit cells in the core (which is equal to the number of positions in the grid plate). For example, assuming data from Table 3, the radius of the average unit cell is equal to $22.05\text{cm}/\sqrt{91} = 2.31\text{cm}$. The fuel rod unit cell in cylindrical geometry consists of the following regions: zirconium rod, fuel (zirconium hydride + enriched uranium), gap, cladding (stainless-steel) and water. The influence of the neighbouring unit cells may be taken into account by surrounding the cell by a cluster of six unit cells. This is important in two cases:

- the central unit cell does not contain fuel rod
- the fuel rod in the central cell is significantly different from the rods in the cluster (e.g. in enrichment).

In first case the surrounding fuel represents source of neutrons for the inner cell which may be an empty or water filled position, an irradiation channel, a graphite dummy element, a control rod or any other kind of the unit cell not containing fuel. Second case is important for mixed cores where the fuel elements of one kind (enrichment) are mixed with the fuel elements of another kind (enrichment). By using the cluster geometry for calculating the homogenised cross-sections of the central unit cell we take into account effective leakage into or from the central cell due to the irregularity of the lattice in the vicinity. The influence of the leakage from the reactor is taken into account by performing the calculations using critical buckling in WIMS. The effect of

leakage on the spectrum and consequently on the homogenised cross-sections is significant for small reactors and must not be neglected. The leakage spectrum must be considered also in the burn-up calculation.

The calculation of the cluster in WIMS may be performed either in 2-D geometry (PIJ option) or in 1-D approximation. Experience shows that the difference between the results is small if only the central unit cell cross-sections are considered. On the other hand, the 1-D calculation is an order of magnitude faster.

The principle of 'extended' unit cell may be applied also for MTR homogenised cross-sections calculations. The appropriate most simple 1-D geometry of the unit cell is plate geometry as it is evident from Fig.3. The unit cell is equal to the fuel plate plus water on both sides. The unit cells are periodical in the fuel element. The fuel element may be divided into two regions: the unit cell region, containing active part of the fuel plates and the rest, containing structural parts and part of the fuel plates not containing uranium, as presented in Fig.4. The second region is often called 'the extra region'.

The effective homogenised cross-sections of the fuel element may be calculated in first approximation by assuming a single fuel plate unit cell surrounded by appropriate volume of extra region. The volume ratio of all materials must be equal to the volume ratio in the fuel element. The extra region in the unit cell is for this reason very thin. It is usually treated as homogeneous. An example of such 'extended' unit cell is presented in Fig.5 taken from ref.(5). If such model is used in WIMS we may separately derive homogenised cross sections for the fuel plate unit cell and for the extra region by running the code two times and using appropriate REGION commands. Of course, we may also get the cross sections homogenised over entire extended cell. In the diffusion calculation we may for this reason proceed in two ways:

- the fuel elements are treated as homogenised units (Fig. 2)
- each fuel element is divided into fuel region and two extra regions (Fig.6).

Second model is more complicated but feasible for modern nodal codes (e.g. BINODE) even in 3-D calculation.

The unit cell physical model in WIMS can be developed a step further. Instead of using only one fuel plate unit cell surrounded by thin extra region to represent the fuel element, it is convenient to use as many fuel plate unit cells as there are in the fuel element (e.g. 19). The volume of the extra region is in this case the same as in the fuel element. The model is feasible in WIMS, particularly because the problem is reduced by half due to the symmetry reasons.

The advantage of this model is that it may be used also for the control fuel elements. The control element is equal to the standard except that it contains 4 fuel plates less, two at each side, to provide gap for inserting the control blades. In the model, the outer two fuel plates are replaced by water and aluminium plate, respectively. Let us call this region 'the control rod region'. If the control rod cross-sections are calculated, the fuel plate at the corresponding position is replaced by the absorber material. By using the REGION card, the cross sections may be derived separately for the fuel plate region, the control rod region (containing control rod or not) and for the extra region or, together for the inner two regions or for all three of them. Which combination will be used depends on the geometrical model in the diffusion code. If the control rod region and extra region are treated explicitly, the definition of nodes becomes rather complicated (Fig. 7) but still feasible.

The reflector is very important in research reactors. The cross sections can be calculated using the same model of several unit cells described above in which the reflector region is added at

the outer boundary. In case of TRIGA a cluster of fuel rod cells is surrounded by a circular region of reflector material. In case of MTR, a group of fuel plate cells are in the centre and the reflector region outside. In both cases it is recommended to model the reflector region as thick as possible and to use zero flux boundary condition. The inner reflector aluminium cladding may be explicitly treated. The reflector may consist of various materials, particularly in MTR (e.g. water, graphite, beryllium).

Note that the (n,2n) reaction which is very important for beryllium in fast energy groups is not treated explicitly in the WIMS neutron balance equations. Implicitly it is taken into account by reducing the absorption cross section σ_a tabulated in the WIMS library

$$\sigma_a \longrightarrow \sigma_a - 2\sigma_{n,2n}$$

where $\sigma_{n,2n}$ is (n,2n) reaction cross-section. This approximation is acceptable as long as the contribution of beryllium in the homogenised region is small. In case of large homogeneous beryllium regions (e.g. beryllium elements or beryllium reflector) this approximation leads to negative macroscopic absorption cross-sections in fast energy groups which usually do not comply with the neutron balance equation in standard multigroup diffusion codes. Normally the negative values for σ_a are formally accepted by the diffusion codes, however the neutron production is not treated correctly and considerable errors may be expected in k-eff as well as in flux distributions.

The group structure is an important characteristic of the physical model. General idea is to use as many groups as it is acceptable from the aspect of the computer time. Modern computers do not impose so severe limitations on the number of groups as it used to be not long ago. The transport calculations in WIMS may be performed simply in 69 group structure. The diffusion calculations are traditionally performed in 4 to 6 groups. Typical group boundaries in four group structure are 10 keV, 1eV and 0.1eV.

If the transport calculation is performed in 69 group approximation, the cross-sections have to be collapsed to 4 groups for diffusion calculation. The group collapsing to arbitrary few-group structure is not automatically provided in WIMS (except to two groups). The group collapsing is not complicated calculation, it involves only averaging of the homogenised cross sections over the spectrum, which appears in the WIMS output together with the cross-sections:

$$\Sigma_G = \frac{\sum_{g \in G} \Sigma_g \Phi_g}{\sum_{g \in G} \Phi_g} \qquad G = 1,2,\ldots,NG$$

Σ and Φ are group cross-section and flux, respectively. G is index of the collapsed group, NG is number of the collapsed groups, e.g. NG=4. Summation is performed over all fine groups g within the broad group G. The formula may be used for all cross sections (including transport cross-section which is inversely proportional to the diffusion coefficient) except for the scattering matrix. The scattering cross section is averaged over both broad groups, the one from which the neutrons are scattered, G,and the one to which they are scattered, H:

$$\Sigma^{G\to H} = \frac{\sum_{h\in H}\Phi_h \sum_{g\in G}\Sigma^{g\to h}\Phi_g}{\sum_{g\in G}\Phi_g \sum_{h\in H}\Phi_h} \qquad G,H = 1,2,\ldots,NG$$

The format of the collapsed cross-sections depends on the diffusion code in which they are used. The group collapsing codes are for these reasons normally developed by the WIMS user. A code for collapsing WIMS homogenised cross-sections to arbitrary few group structure and format of BINODE is called XSWOUT.

The research reactors contain nuclides which are not typical for the power reactors, e.g. hydrogen bound in zirconium crystal, erbium and samarium as burnable poisons, cadmium as control rod material. The cross section libraries of the computer codes which were primarily designed for the power reactors usually do not contain specific isotopes for research reactors. Adding of new isotopes to the program libraries (e.g. of WIMS) is complicated task normally exceeding the capabilities of typical research rector staff. Efforts are made to supply the libraries of some important codes for the calculations of the most common reactor types and make them available to the users through international libraries. The WIMS library was updated at J.Stefan Institute for typical TRIGA reactor nuclides. It is available in NEA CPL.

Use of appropriate cross-sections is essential in calculations of particular effects. It is quite evident that the cadmium control rods can not be calculated if cadmium is not part of the library. However it is less evident that the calculations of TRIGA reactor are not correct if the scattering cross-sections are not available for hydrogen bound in zirconium hydride. Fig.8 shows comparison of fuel temperature reactivity coefficient calculated using hydrogen bound in water cross-sections (original WIMS library) and hydrogen bound in zirconium hydride. It may be observed that the fuel temperature effect can not be reproduced without appropriate hydrogen scattering cross-sections.

Another general feature of research reactor calculations is that U-238 is not very important due to relatively high enrichment (from 20% to over 90%). The resonance effects of U-238 in the neutron balance are not dominant. The contribution of Doppler effect in the fuel temperature reactivity effect is small. Moreover, the fuel temperature reactivity effect itself is small in MTR due to low fuel temperature. The resonance treatment (e.g. Dancoff and Bell factors) is not as important as for power reactors. The same is true for the quality of the resonance cross-sections of U-238. No special attention is normally devoted to resonance calculation modelling for research reactors.

Generally speaking, the cross-sections and the unit cell calculations are not the main source of errors in the research reactor calculation. The main source of errors in global reactor calculation is believed to be the leakage operator, $-\nabla D_g \nabla$. The contribution of leakage in neutron balance of research reactor is large: typical k-inf of TRIGA reactor fuel is 1.4 and of MTR 1.7. Even small errors in leakage significantly effect k-eff. The diffusion approximation of the leakage operator itself is questionable due to large gradients of neutron flux in small and highly heterogeneous systems. The discretisation of the ∇ operator by numerical methods introduces some error which is normally small and under control.

The most questionable, however, is the calculation and homogenisation of the diffusion coefficient. There are many methods for its calculation, some of them even available in WIMS, neither of them offering significantly better or more reliable results. Typical absolute error of k-eff due to leakage is estimated to be around 2%Δk/k in multigroup diffusion calculation of small reactor. It depends on the size of the core and on the geometry of the core/reflector boundary. It is not reasonable to invent sophisticated models for calculating, homogenising and collapsing D_g to reduce absolute error of leakage calculation because the diffusion approximation for small research reactors is rather weak, anyway. There are two choices: either to accept the accuracy of the diffusion approximation, which is in fact sufficient for all practical applications at research reactors except the design, or to perform the calculations using much more demanding multigroup transport or Monte-Carlo methods.

4 Conclusions

The computer codes which are most frequently applied in user oriented research reactor calculations are discussed. Their use for typical MTR and TRIGA reactor calculations is treated. Basic physical models for 2-D and 3-D diffusion calculations are presented. Emphasis is on the calculations of homogenised multigroup cross-sections using WIMS code.

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Table 4. Typical 2 MW MTR design parameters, ref. (5)

Reactor type	Pool-Type MTR
Steady-State Power Level	2 MW
Number of Standard Fuel Elements	19
Number of Control Fuel Elements	4
Irradiation Channels	1 at Core Center
Core Geometry	4 x 6 Arrangement
Grid Plate	6 x 9 Positions
U-235 Content/Core	3988 g
Active Core Volume	86 l
Average Volumetric Power Density	23.3 kW/l
Average Linear Power Density	0.08 kW/cm
Specific Power	502 kW/kg U-235
Moderator, Coolant	Water
Reflector	Water
Fuel Element	MTR-Type (76 x 80 x 600
mm)	
Number of Fuel Plates in:	
Standard FE	16 (19)
Control FE	12 (15) + 2 Al Plates
	2 Control Blades/Element
Plate Dimensions	Standard MTR Plate
Plate Thickness	1.27 mm
Meat Thickness	0.51 mm
Shape of Plate	Straight
Fuel Loading: Standard FE	140 - 180 g U-235
Control FE	105 - 135 g U-235
Number of Fuel Elements in the Core	24 +/- 1
Standard FE	20 +/- 1
Control FE	4

Table 5. Material composition and dimensions of different TRIGA fuel elements

Dimensions	Standard 8.5 %	Standard 12 %	FLIP	LEU
element length (cm)	72.1	72.1	73.3	75.4
fuel length (cm)	38.1	38.1	38.1	38.1
axial top reflector (cm)	8.7	8.7	8.7	6.5
axial bottom reflector (cm)	8.8	8.8	8.7	9.4
element diameter (cm)	3.75	3.75	3.75	3.75
fuel diameter (cm)	3.64	3.64	3.64	3.64
Zr - rod diameter (cm)	0.635	0.635	0.635	0.635
cladding thickness	0.05	0.05	0.05	0.05
Composition				
enrichment (%)	20	20	70	20
amount of U (%)	8.5	12	8.5	20
mass of U-ZrH (g)	2244.3	2360.0	2258.7	2462.0
mass of U in U-ZrH (g)	190.7	276.5	191.8	494.9
mass of U-235 (g)	37.9	55.0	134.2	97.8
H:Zr ratio	1.65	1.65	1.65	1.57
amount of Er (%)	-	-	1.53	0.44

Table 6. MTR fuel element design description

Type	MTR, Straight Type
Enrichment	93%
Lattice Pitch	77 x 81 mm
Fuel Element Dimensions	76 x 80 x 600 mm
Plate Thickness	1.27 mm (Inner Plates)
	1.50 mm (Out. Plates)
Water Channel Thickness	2.916 mm
Plates/Standard Fuel Element	19
Plates/Control Fuel Element	15 Fuelled and 2 Al
Plates	
Fuel Meat	UAl_x -Al (17.5 w% U)
Meat Dimensions	0.51 x 63 x 600 mm
Clad Thickness (Al)	0.38 mm (Inner Plates)
	0.495 mm (Outer
Plates)	
U-235 Density in Fuel Meat	0.4914 g/cm^3
U-235/Standard Fuel Element	180 g
U-235/Control Fuel Element	142 g
Coolant Flow Rate	300 m^3/h
Core Inlet Temperature	38°C
Volume Fractions	
Standard FE	
Fuel Meat	0.0979
Aluminium	0.2870
Water	0.6151
Control FE	
Fuel Meat	0.0773
Aluminium	0.2806
Water	0.6421

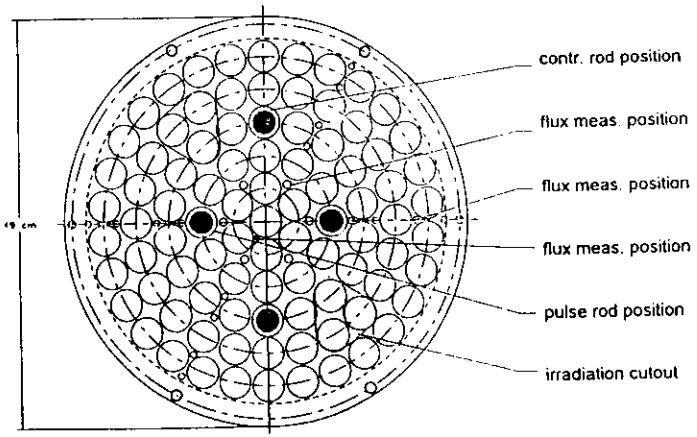

Fig. 1. 250 kW TRIGA Mark II upper grid plate

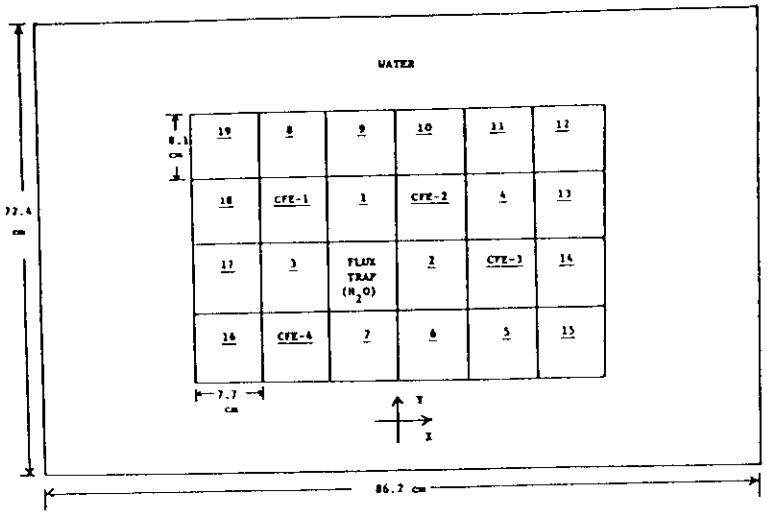

Fig. 2. Typical MTR X-Y geometry (5)

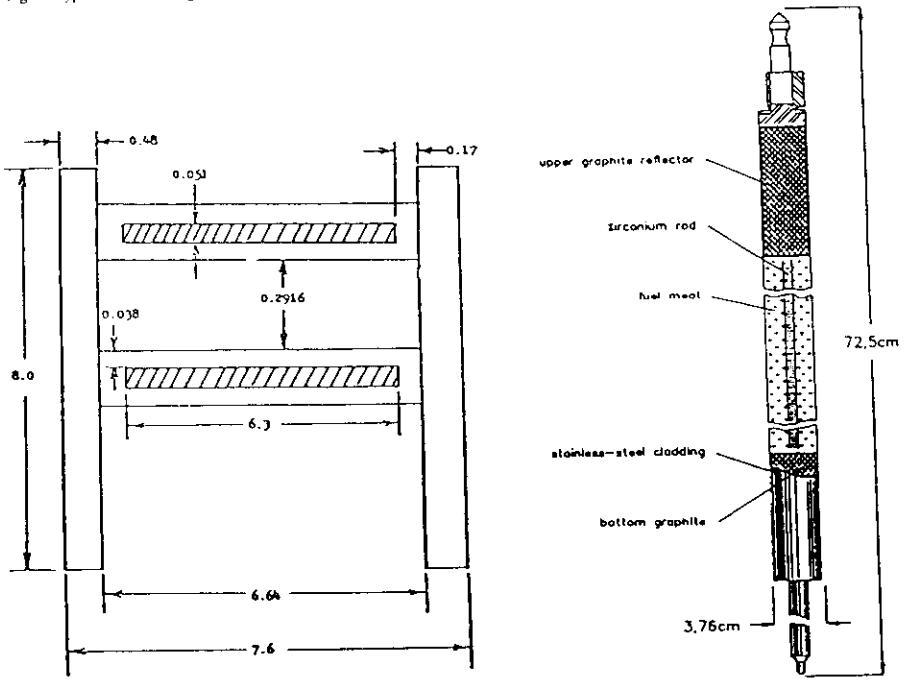

Fig. 3. Schematic drawing of TRIGA and MTR fuel elements

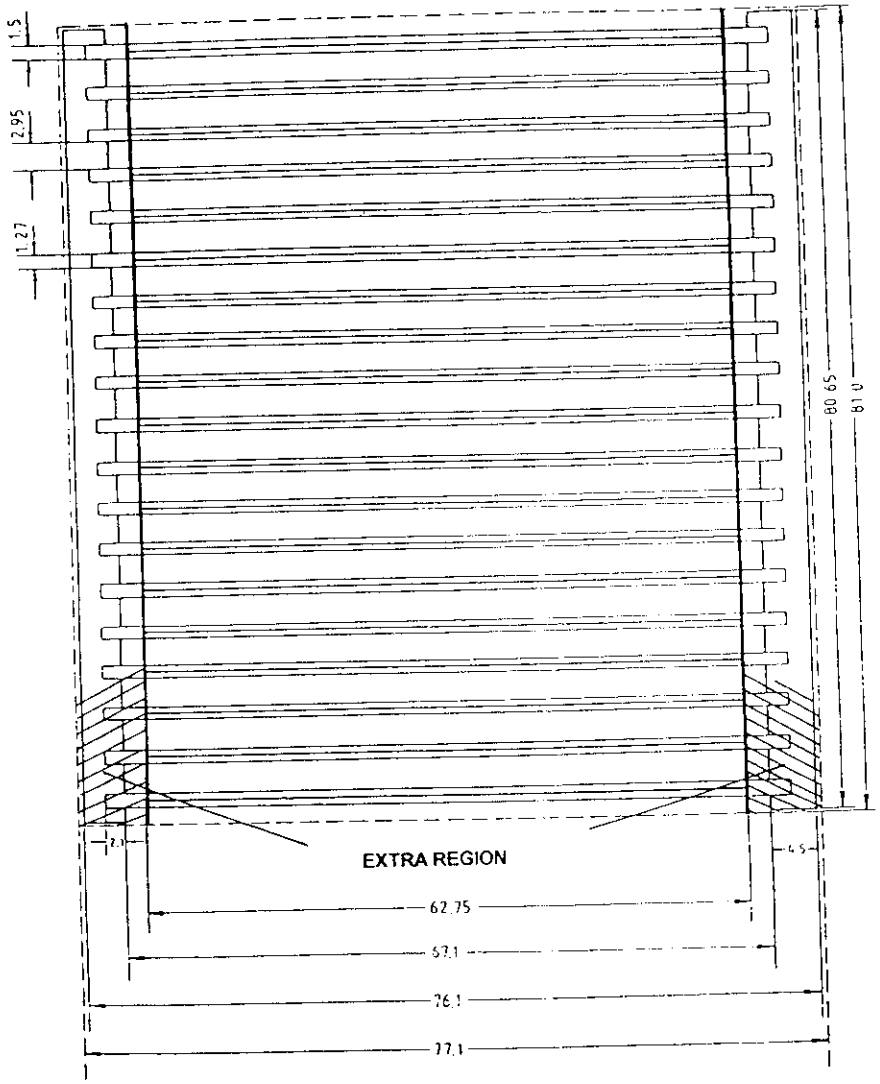

Fig.4. Definition of extra region

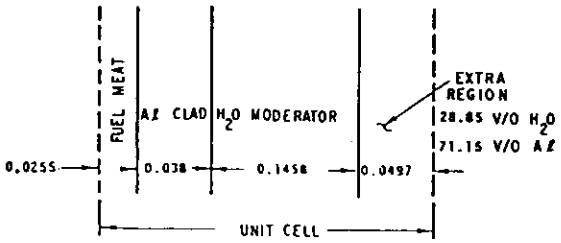

Fig. 5. Fuel plate unit cell with extra region (5)

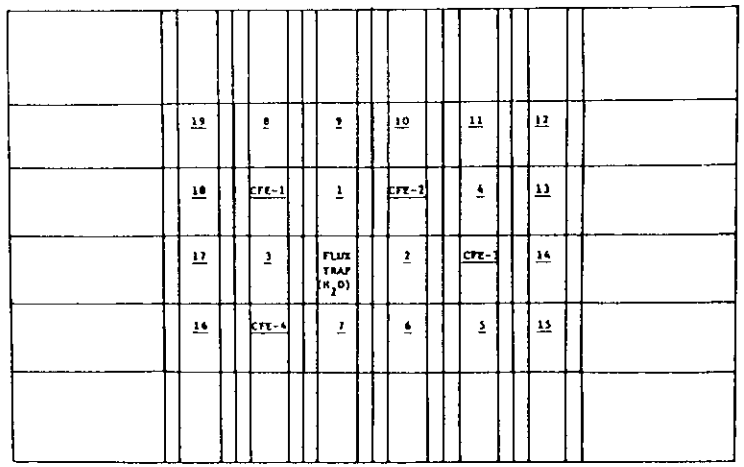

Fig.6. MTR core model separately treating fuel plate part and extra region of fuel elements

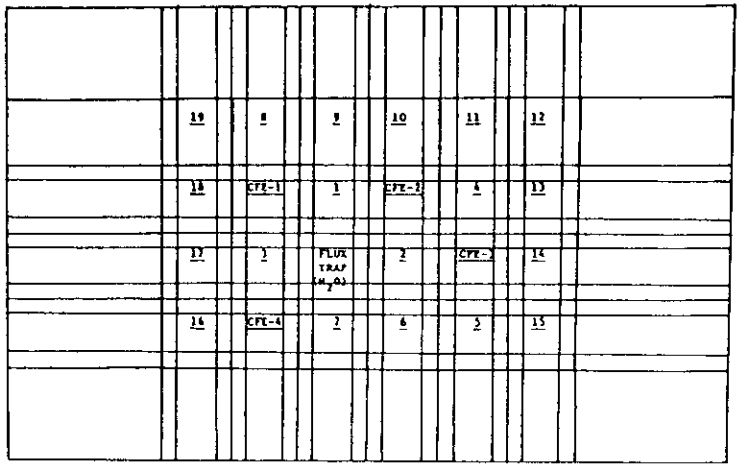

Fig. 7. MTR core model for control elements

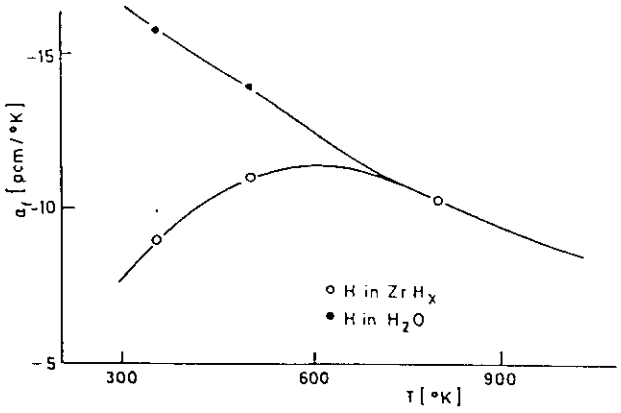

Fig. 8. TRIGA fuel temperature coefficient calculated with two sets of cross-sections for hydrogen in fuel: hydrogen bound in water (upper curve) and hydrogen in zirconium hydride