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# USE OF HIGHLY ENRICHED URANIUM IN THE MATERIAL TESTING REACTOR BR2

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Summary - In the material testing reactor BR2, the use of highly enriched uranium is determined by the consideration of the fast, epithermal and thermal neutron flux effectively available for the experimental devices. The choice of the core configuration is defined by combining the localisation of the experimental devices and of fuel elements of various burnup, such as to satisfy the irradiation conditions of the experimental load, compatible with an economic use of the fuel elements and safe operation of the reactor.

Taking into account the present manufacturing technology for MTR fuels (37 Wt % uranium density in the fuel meat) the highly enriched uranium cannot be avoided; if higher concentration of uranium could be realised by some new manufacturing technology, the $^{235}U$ density of fuel elements at elimination should be kept at the required level and the enrichment could be reduced accordingly.

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Résumé - Dans le réacteur d'essai de matériaux BR2, l'utilisation d'uranium hautement enrichi est déterminée en fonction du flux de neutrons thermiques, épithermiques et rapides effectivement disponibles pour les dispositifs expérimentaux. Le choix de la configuration est défini en combinant la localisation des dispositifs expérimentaux et des éléments combustibles de différent épuisement pour satisfaire les conditions d'irradiation de la charge expérimentale, de façon compatible avec une utilisation économique des éléments combustibles, la sécurité des opérations au réacteur étant maintenue.

Tenant compte de la technologie actuelle de fabrication des plaques combustibles pour réacteurs d'essai de matériaux (MTR) 37 % en poids de densité d'uranium dans l'alliage combustible, l'utilisation d'uranium hautement enrichi ne peut pas être évité. Si une plus haute concentration d'uranium pourrait être réalisée par une nouvelle procédure de fabrication, l'enrichissement pourrait être réduit dans la mesure où la densité d'$^{235}U$ dans les éléments combustibles à l'élimination serait maintenue au niveau requis.

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Samenvatting - In de materialentestreactor BR2 wordt het gebruik van hoog-verrijkt uranium bepaald in functie van de thermische, epithermische en snelle neutronenfluxen die effectief beschikbaar zijn voor de proefopstellingen. De keuze van de configuratie wordt bepaald door combinatie van de plaats der proefopstellingen en de brandstofelementen met verschillende afbrandgraad ten einde aan de bestralingsvoorwaarden van de proevenlading te voldoen op een wijze die in overeenstemming is met een economisch gebruik van de brandstofelementen terwijl de veiligheid van de operaties aan de reactor verzekerd blijft.

Rekening houdend met de huidige fabricatietechnologie van brandstofplaten voor materialentestreactoren (MTR) kan het gebruik van hoog-verrijkt uranium niet vermeden worden daar het densiteitsgewicht van het uranium in de brandstoflegering 37 % bedraagt. Indien een hogere uraniumconcentratie zou kunnen verwezenlijkt worden door een nieuwe fabricatieprocedure, zou de verrijking kunnen verminderd worden in die mate dat de densiteit van het $^{235}U$ in de te elimineren brandstofelementen het vereiste peil zou behouden.

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## TABLE OF CONTENTS


1. INTRODUCTION 1

2. GENERAL DESCRIPTION OF THE REACTOR 2

3. PRESENT CHARACTERISTICS OF THE DRIVER ZONE 2

4. REQUIREMENTS OF THE IRRADIATION LOOPS 3

5. REQUIREMENTS OF THE DRIVER ZONE 4

5.1. Antireactivity of experimental devices 6
5.2. Criticality of core arrangements 6
5.3. Reactivity variations 8
5.4. Economical use of the fuel elements 9

6. CONCLUSION CONCERNING THE USE OF HIGHLY ENRICHED URANIUM AT THE MATERIAL TESTING REACTOR BR2 11

APPENDIX: PART OF NEUTRONS AVAILABLE PER FUEL CELL 13

1. DESCRIPTION OF THE METHOD 13

2. REQUIREMENTS FOR THE IRRADIATION LOOPS 17

3. APPLICATION OF THE METHOD FOR THE RACTOR BR2 19

LIST OF REFERENCES 21

LIST OF TABLES AND FIGURES 22

## 1. INTRODUCTION

Many material testing reactors are heteregeneous thermal flux engineering testing reactors, cooled and moderated by low pressurized light water, in a compact core of highly enriched uranium, positionned in and reflected by a beryllium matrix.

The main apparent limits of this type of material testing reactors are:

- the size of the experimental devices;
- the ultimate cooling capacity;
- the capability of the cadmium-beryllium shim-safety rods to regulate the chain reaction during the fuel and burnable poisons consumption, during the reactivity transients (Xe, Sm, experimental transients,..) and for the scram in unforeseen or unwanted circumstances.

Less evident limits appear in the point of view of uranium utilization, since this type of material testing reactor is often optimized with respect to the best concentration of $^{235}U$ giving the maximum reactivity invested, without consideration to:

1) the fast, epithermal and thermal **neutrons effectively available** for the experimental devices;

2) the design of the **core configuration**, which is the result of the adaptation and the development of the reactor characteristics to the needs of research programs and irradiation services, following the desiderata of the various experimenters, taking into account the security of the people and the installations. In particular, the modifications of the core configuration, the choice of the **fuel elements** (and of the burnable poisons) and the localisation of the experimental devices and of fuel elements of various burnups, are defined such as **to satisfy the experimental load with efficiency and flexibility.**

3) **the economical use** of the fuel elements, when a given experimental load must receive satisfying irradiation conditions, compatible with the safe operation of the reactor, i.e.

   a) number of fuel elements and quantity of highly enriched uranium loaded in the reactor core;
   b) operation period of a fuel load for as long as possible;
   c) high burnup of fuel elements at elimination.

We have elaborated a simple method to calculate the requirements of the driver zone which is able to satisfy the experimental devices, which are localized in a material testing reactor like BR2, where the predominant process is an elastic scattering of neutrons with a degraded energy spectrum. If there is a possibility to descretize a reactor load into cells containing a reflector beryllium plug or a fuel element cooled by water, filled with an experimental device, we obtain a dimensionless parameter $A_i$, which we have called «part of neutrons available per cell i» :

$A_i = 2\, r_o\, (\, \nu\bar{\Sigma}_f - \bar{\Sigma}_a)$ ; there, $\nu\bar{\Sigma}_f$ and $\bar{\Sigma}_a$ are the **average macroscopic** cross-section for fission and for thermal absorption (including leak out of the system and sinks in the cadmium screens) and $r_o$ is the equivalent radius of the cells i. Despite the appearing simplicity of the method utilized in this report, important and sufficiently accurate conclusions can be obtained (see appendix).

There exists a minimum value of the parameter A, which permits operation of the reactor to satisfy the experimental load and which corresponds to a minimum mass $^{235}U$ to be loaded per fuel channel of the reactor. Taking into account the present manufacturing technology for MTR fuels, the highly enriched uranium cannot be avoided; if higher concentration of uranium may be realised by some manufacturing technology, the value of the parameter $A_i$ of fuel elements at elimination must be conserved.

## 2. GENERAL DESCRIPTION OF THE REACTOR BR2 (fig. 1 to 4) [1]

BR2 is a heterogeneous thermal high flux engineering testing reactor, designed in 1957 for CEN/SCK by NDA (Nuclear Development Corporation of America - White Plains - New York - USA). It has been built on the site of the CEN/SCK laboratories in MOL. Belgium. Routine operation of the reactor started in January 1963 and to this day, continues to contribute greatly to the development of many large nuclear projects within the European Community and also to carry out irradiations for countries throughout the world.

The reactor is cooled and moderated by pressurized light water (12 bar) in a compact core of highly enriched uranium positioned in and reflected by a beryllium matrix. The maximum thermal flux reaches $10^{15}$ n/cm² s and the ultimate cooling capacity, initially foreseen for 50 MW, has been inceased in 1971, to 125 MW.

The beryllium matrix has 79 cylindrical holes in an hexagonal lattice of 9.644 cm pitch at the midplane, all channels have individual access through holes in the top cover of the reactor vessel. Individual guide tubes join the holes in the top cover with the holes in the matrix and thus create 79 individual reactor channels. There are 64 standard channels (Ø 84 mm), 10 reflector channels (Ø 50 mm) and 5 large channels (Ø 200 mm). All channels can receive fuel elements, control rods, plugs or experiments, which allows a great flexibility for a great variety of core arrangements. The five 200 mm channels and thirteen of the 84 mm standard channels have additionnal apertures through the bottom cover of the reactor vessel, which allows through loop experiments.

The standard fuel elements are assemblies of concentric plates or tubes of about 3 inches outer diameter, each having a 30 inches fuel length. The water gap between the plates is 3 mm wide, the plate thickness is 1.27 mm. A fuel plate consists of a cermet mixture of fully enriched (93 %) U $Al_3$, 60 mg/cm² of $^{235}U$ and burnable poisons ($B_4C$, $Sm_2O_3$), cladded with aluminium sheet. A standard fuel element (VIn type) contains 400 g $^{235}U$ within six concentric plates and allows for an experimental space having a diameter of 25.4 mm while a fuel element Vn type contains five concentric plates of the same composition and allows for an experimental space of 34 mm.

The chain reaction is controlled by a number of cadmium-beryllium shim-safety rods (Ø Cd = 50/60 mm) and by a regulating rod (Ø Cd = 34/39 mm), which occupy matrix lattice positions and have electromechanical drives. Each rod moves inside its own aluminium thimble, which allows them to be placed in various positions, following the core arrangement.

The nominal specific power is given by the maximum heat flux allowed at the hot spot, 600 W/cm² for nominal hydraulic conditions and for probable onset of nucleate boiling. This implies that the maximum power generated in a hot channel loaded with a standard fuel element is 3.60 to 4.25 MW according to the axial flux distribution.

## 3. PRESENT CHARACTERISTICS OF THE DRIVER ZONE

The optimization of the driver zone in function of the requirements of the experimental devices has reduced very much the consumption of fuel elements: the specific consumption is going down from 23 to 6 fresh standard fuel elements per $10^3$ MWd. At the beginning of the year 1971, the first «Cermet C» type fuel elements containing 330 g $^{235}U$ (50 mg $^{235}U/cm^2$) in the form of U $Al_3$ (+ 2.8 g boron ($B_4C$) + 1.3 g samarium ($Sm_2O_3$)) in standard fuel elements, were loaded routinely in place of 244 g $^{235}U$ (36 mg $^{235}U/cm^2$) in the form of U $Al_4$ for the alloy fuel elements previously used. Since the cycle 4/72 the loading of standard fuel elements «Cermet G» type containing 400 g $^{235}U$ (60 mg $^{235}U/cm^2$) in the form of U $Al_3$ (+ 3.8 g boron ($B_4C$) + 1.4 g samarium ($Sm_2O_3$)) has begun. The uranium density in the fuel meat of the present Cermet type elements is 37 Wt % compared to 22 Wt % for the alloy elements. Reactor operation is carried out on the basis of an

operating cycle; the present nominal cycle length is 4 weeks, and consists of 7 days shut-down for loading and normal maintenance work, followed by 21 days of operation, in such a way the fuel elements can be reloaded at least 3 times to reach their burnup at elimination. The general utilization of these new fuel elements has permitted longer operation periods (21 days compared to 14 days for the alloy fuel elements) and better irradiation conditions due to the motion of the shim rods and to the hardness of the neutron spectrum (fig. 6, 7). Furthermore, the costs of the fuel were diminished, by reducing the number of the fresh fuel elements (fig. 3) and by increasing their burnup at elimination [2, 3].

For present core configurations, we have determined that the mean $^{235}U$ enrichment must be higher than 73.7 % in standard fuel elements «Cermet G» type, supposing that the geometry of these fuel elements has been optimized (fig. 9) [4]. Let us note that this value is given at the end of an operation period (Xenon saturated) without experimental load in standard fuel elements and without excess of reactivity in the control rods; that indication cannot correspond to practical operation conditions combined with the requirements of an experimental load nor an operational power rating. Indeed, the criticality conditions are determined, for a given core configuration, by an average content of $^{235}U$ per fuel cell, balanced with absorption due to the experimental devices, control rods and burnable poisons required to allow a sufficiently long operation period.

## 4. REQUIREMENTS OF IRRADIATION LOOPS AT BR2

Since the early days of its operation, the exploitation of BR2 as an engineering testing reactor has been characterized by the high part of the experimental load taken up by irradiations conducted in the frame of fast reactor fuel research and development programs. The major part of these irradiations are concentrated on the study of fuel element behaviour under irradiation up to high burnup. In order to simulate more closely the fast reactor conditions, loops and instrumentated rigs with cadmium screen have been developped. Up to now, an appreciable number of fuel pins has been tested in these devices. Because of its high neutron flux and especially, its high fast neutron flux, BR2 is well adapted to this kind of irradiations in comparison to other thermal MTR's (fig. 10). The use of the MTR's for fast reactor programs is directed by a lot of studies which arrive in general at the following conclusions about specific advantages and needs for a true fast reactor environment [5].

Specific advantages are recognized in MTR's:

1) availability of high fast neutron flux
2) availability of space for in-pile instrumentation
3) cheapness with regard to fast reactor irradiations
4) possibility of analysis of the thermal, mechanical and chemical irradiation conditions
5) adequate loops allow irradiation up to and even beyond fuel pin failure, which is of particular importance with regard to power cycling, transient overpower and safety tests.

A true fast environment is necessary to have a representative test with regard to neutron spectrum reaching typical linear power values, with a good ratio between burnup in the fuel and radiation damage in the cladding. The present policy tends to irradiate simultaneously several screened loops and to increase the power density in the BR2 core, more particularly in the vicinity of the loops. The prediction of the irradiation characteristics for a device to be loaded into the BR2 reactor relies upon:

- the nuclear calculations performed by means of the computer codes (GAMS, DTF IV, TRIBU, DIAMANT ...);

- the measurements carried out in BR02, the nuclear mockup of BR2;

- the analysis of the data given by the instrumentation of the devices already under irradiation in BR2, and also the neutron flux measurements.

The irradiation characteristics consist of thermal and fast neutron flux, neutron spectrum inside the irradiation loops, gamma heating, reactivity effects, specific power rating of fuel experience,... Indeed, because of its flexibility, the reactor can accommodate itself to requirements for new devices which lie beyond the scope of the current irradiation possibilities; actions are also taken to correct for unexpected mismatchings in the performance of experimental devices with respect to each other, or to increase the reactivity in order to compensate the effect of the experimental loading. The various experimental data resulting from fast reactor fuel irradiations at BR2 justify a good consistency and the validity of the neutronic calculation codes and of the dosimetry work [6, 7, 8].

The reactor power level and the arrangement of the fresh and partially burnt fuel elements in the core are adjusted in order to attain compatibility between the specifications of different devices; if this cannot be achieved, the core configuration has to be modified. For example, the irradiation of the loop MOL 7C1 (Safety test of loss of coolant in a bundle of thirty fast reactor fuel pins, cooled by liquid sodium) has required the following conditions at BR2 [9] :

1) The irradiation loop MOL 7C1 was localised in the central hole H 1 and surrounded by a fuel element 200 mm diameter, ATR type, containing 6 fuel plates and the cadmium screen (Ø Cd = 137 mm).

2) The power generated in the ten fuel elements of the central ring (channels A and B) was designed 31.9 MW.

3) The total power of the reactor was provided and confirmed 100 MW with 12186 g $^{235}U$ in the core loading (34 standard fuel channels + 3 equivalent ones in chanel H 1) to give the nominal linear power wanted at the experimental loop (380 W/cm).

4) The reactivity of the bundle of shim-safety rods was measured before the start-up: 13.0 $, and an excess of reactivity: 5.48 $ to permit a sufficient operation time (this load had allowed an operation period of 2100 MWd) and a sufficient availability to follow the transient xenon-samarium poisoning effects, indeed the MOL 7C1 experiment needed four provided power reductions.

5) The measurements of the r .activity effects connected to the MOL 7C1 experiment in the channel give a gain 1.19 $ fc. the loop and the fresh fuel element Ø 200 instead of beryllium. The reactivity effect of the fuel pins contained in the loop is estimated by the difference of reactivity between the loop and the nuclear mock-up of MOL 7C1, which was measured to be a gain of 0.40 $. Let us note that the reactivity effect of two standard fuel elements (VIn G 400 g $^{235}U$ 20 % burnt) in the central ring in place of beryllium, gives a gain of reactivity, of about 5 $.

## 5. REQUIREMENTS OF THE DRIVER ZONE

The general description of the reactor BR2 has shown that a great variety of core arrangements can be carried out in order to satisfy the requirements of irradiation loops and rigs with efficiency and flexibility.

In the reactor BR2, the predominant process is an elastic scattering of neutrons with a degraded energy spectrum and the hexagonal lattice is large enough ( $\Sigma_t . r_o > 2$) that the influence of second vicinity cells can be neglected in the calculation of the adjoint of the thermal flux. We consider the dimensionless parameter $A_r = r_o ( \nu\bar{\Sigma}_f - \bar{\Sigma}_a)$, where $\nu\bar{\Sigma}_f$ and $\bar{\Sigma}_a$ are the average

macroscopic cross-section for fission production and for thermal absorption (including the leak out of the system and the sinks in the cadmium screens) and where $r_o$ is the equivalent radius of the cells; this parameter $A_i$, which we have called part of neutrons available per cell, is useful for calculation of the adjoint of the thermal neutron flux (fig. 11, 12, 13). In practice, the thermal and fast neutron flux can be calculated from a reference load by means of perturbation factors, which are determined experimentally and by computer codes (fig. 14). According to the first order perturbation theory, the total excess of reactivity is easily achieved by summing up the reactivity variations due to the fuel composition (fig. 15) and to the absorbing devices. Let us note, that the regional summation of local reactivity effects aids to ensure a well balanced load. The dimensionless parameter $A_i$, i.e. the part of neutrons available per cell, is determined by the following three conditions:

1) the size of the lattice cell and of the fuel element;

2) the content and the composition of the fuel element ($^{235}U$, burnable poison,...);

3) the microscopic distribution of the thermal neutron flux in the lattice cell.

The contribution of the part of neutrons available per cell due to $^{235}U$ in a fuel cell may be roughly approximated by the expression:

$$A = 2r_o\left(\nu\bar{\Sigma}_f - \bar{\Sigma}_a\right) = 1.874 \times \frac{2\, r_o\, Mu5}{\pi r_o^2\, L_f} = 7.5\ \frac{Mu5}{2\pi r_o\, L_f}\ .$$

where $2\pi r_o L_f$ is the external surface of the fuel cell; at BR2 this rough approximation gives $A \approx Mu5/324$ g ($r_o = 5.063$ cm and $L_f = 76.2$ cm), but according to the microscopic thermal neutron flux, a better more accurate calculation gives (fig. 11, 12)

$$A \approx -0.06 + Mu5/400 \text{ g in the standard fuel cell}$$

The requirements of the driver zone are determined

- mainly by the reactivity analysis of the fuel load prepared in such a way that it satisfy the requirements of the irradiation loops;
- secondarily by the adjustment of the local neutron flux and with due consideration of the cooling possibilities.

The reactivity analysis determines the limits and the requirements of the driver zone:

- the value of the antireactivity of the experimental devices, expressed as part of neutrons available per fuel cell;
- the reactivity of the core configuration which depends on the part of neutrons available per fuel cell and on the geometrical arrangement of the fuel elements and irradiation loops;
- the operation period determined by the extraction of the shim-safety rods and by the consumption of burnable poisons, because the load has to satisfy the local and overall criticality conditions at any time during operation, taking into account all possible transient;
- the limitations of the partially burned fuel elements for use in the chosen core configuration, the metallurgical properties in function of the admissible burnup and also the corrosion of the cladding material in operating and storage conditions, result in restricting the life-time of the fuel elements.

### 5.1. Antireactivity of experimental devices

The solution for a very simple arrangement of fuel cells in an hexagonal lattice is sufficient to estimate the requirements of the driver zone, expressed as part of neutrons available per fuel cell. For the example of a symmetrical group of seven cells (the central one being parametric and six cells containing fuel elements), surrounded by two rings of reflector cells; the system is presented at the appendix and has the solution:

1) if the central cell is filled with a fuel element $A_{min} = 0.25$

2) if the central cell is a reflector cell $A_{min} = 0.30$

3) if the central cell contains an absorbing device having a part of neutron available: $A_o < o$ $A_{min} = 0.30 + \frac{1}{6}\left(1 - \frac{1}{1-A_o}\right)$

This value of the parameter A must be understood as not including the leak of fast and epithermal neutrons, as if the reactor load would be composed of several regions, each of which being in near-criticality conditions, to satisfy with flexibility the irradiation conditions of many irradiation loops.

This is in good agreement with the perturbation theory, which gives the formulation for the reactivity effects of the variations of fuel composition and that of the experimental devices. In fact, the general particularities of $A_i$ per fuel cell, needed for clusters of fuel elements, may be accepted with the first order perturbation theory for any reactor core, if the reactive equilibrium of the load is not changed:

$$\frac{\delta k_{eff}}{k_{eff}} = \frac{\delta \int_{dvi} (\nu\Sigma f - \Sigma a).\ \phi^*.\ \phi'.dvi}{\sum_i \int_{dvi} \nu\Sigma f.\ \phi^*.\ \phi'.dv} \approx \frac{\delta A_i.\phi_i^2.F_i}{\sum_i 2r_o\ \nu\bar{\Sigma}f.\phi_i^2.F_i}$$

Here the perturbed thermal neutron flux $\phi_i$ and the utilization factor of highly enriched uranium will be taken into account; $F_i$ is called perturbation factor if the adjoint of the thermal neutron flux $\phi^*$ has been calculated in a core configuration containing identical fuel cells. This formula is useful for calculating how a given load is balanced in the same way as the reference load and for prediction of the operation period [9].

### 5.2. Criticality of core arrangements

The analysis of core configurations carried out at BR2 shows that the gain of reacticity by increasing the number of fuel cells is small and smaller than mentionned below for cermet fuel elements having a higher content of $^{235}U$; indeed $k_{eff} - 1 \div (Mu5)^{-1}$.
For comparison with experiments, we calculate in the test configuration 4 : $k_{eff}$ ($A_i = 0.55$) = 1.357, against $k_{eff}$ = 1.1037 measured in the reactor BR02 (nuclear mock-up of the reactor BR2): the difference of 25 % is interpreted as the effect of leaking out of fast and epithermal neutrons.

| Configuration | Number of fuel cells | $k_{eff}$ ($A_i$=0.50) | A min ($k_{eff}$=1) | Remarks |
|---|---|---|---|---|
| 0 | 6 | 1.2085 | 0.284 | two reflector rings give $k_{eff}$ = 1.180 |
| 1 | 7 | 1.2585 | 0.247 | two reflector rings give $k_{eff}$ = 1.202 |
| AB | 12 | 1.2478 | 0.252 | central ring A and B |
| 4 | 14 | 1.2925 | 0.221 | test configuration |
| 7B | 36 | 1.3621 | 0.176 | surrounding the channel H 1 |
| 8 | 40 | 1.3801 | 0.166 | surrounding the channel H 1 and H 3 |

This calculation is made for $A_i = 2\, r_o\, (\nu\bar{\Sigma}_f - \bar{\Sigma}_a) = 1.30 - 0.80 = 0.50$

The lowest value of part of neutrons available in the most compact core (configuration 4) realized at BR2 is calculated $A_i = 0.22$ at the end of an operating period, without experimental load and without excess of reactivity in the control rods.

The history of the core configuration at BR2 has begun with a test load (configuration 4), followed by another one for the utilization of the central channel H 1 (Ø 200 mm) by the configuration 6 and 7 and now for simultaneous utilization of two or three channels Ø 200 mm: H 1, H 3 and H 4 (configuration 8 and 9). The main characteristics of these configurations are (fig. 17) :

*Configuration 4* [10]

- compact core of 14 standard alloy fuel elements surrounding the channel DO;
- 7 shim-safety rods, the reactivity worth of which is very great : 25 $
- no flexibility for the adaptation of irradiation conditions;
- no high thermal neutron flux in reflector cell for radioelements production;
- in the central channel DO (Ø 84 mm) a unique irradiation loop could replace the shim-safety rod (7.25 $), if wanted.

*Configuration 7*

- dispersed core of ∿ 36 standard cermet fuel elements surrounding the channel H 1;
- 6 shim safety rods, with a reactivity worth of ∿ 13.7 $;
- good flexibility for the adaptation of irradiation conditions, but the coupling between 0° and 180° areas is too weak due to the presence of the reflector channels H 2 and H 3 and of shim-safety rods in channels A 90 and A 270;
- the central channel H 1 (Ø 200 mm) is used for an absorbing irradiation loop and the channels H 2 and H 5 for radioelements production;
- the configuration 8 used standard alloy fuel elements and two additional shim-safety rods in the channels D 60 and D 300.

*Configuration 8*

- very dispersed core of ∿ 40 standard cermet fuel elements surrounding the channels H 1 and H 3;

- 7 shim safety rods (6 equivalent shim-safety rods, because half worth in channels F 106 and G 180) with a low reactivity value of ∿ 11.6 $;

- very good flexibility for the adaptation of irradiation conditions, particularly for adaptation in channel H 3 [11], but the configuration is not symmetrical;

- the configuration 9 is the mirror configuration of the configuration 8, by reflexion on the axis L 0 - L 180, with utilization of the channel H 4 for an irradiation loop.

**5.3. Reactivity variations**

The most important reactivity variations in a high flux reactor are ($\beta_{eff} = .0072$):

1) consumption of highly enriched uranium
2) consumption of the main burnable poison $B_4C$ to compensate the consumption of uranium
3) consumption of the second burnable poison $Sm_2O_3$ to compensate the initial Xenon effect
4) Temperature effect
5) $^6Li$ and $^3He$ effect produced in the beryllium matrix
6) Xenon-Samarium effects which consist of :
   if the reactor is shut down since several days :

$$\Sigma_{\substack{Sm.PF \\ \tau \to \infty}} = \gamma_{Sm.}\,\Sigma_f\,(1 + \frac{\sigma_{Sm}\,\phi}{\lambda_{Pm}}) = 0.0113\;\Sigma_f\,(1 + \frac{\bar{\phi}}{1.66\;10^{14}}) \approx 0.034\;\Sigma_f$$

if the reactor has its full power rating

$$\Sigma_{Sm.eq} = \gamma_{Sm.}\,\Sigma_f = 0.0113\;\Sigma_f$$

$$\Sigma_{Xe.eq} = (\gamma_J + \gamma_{Xe})\;\Sigma_f / (1 + \lambda_{Xe}/\sigma_{Xe}\,\phi) = 0.064\;\Sigma_f$$

When the power of the reactor is increasing, the initial Samarium contained in the fresh fuel elements can compensate the difference between the Xenon effect and the Samarium fission product contained in the partially burnt fuel element; a maximum peak of the Xenon-Samarium transient after the first start of the reactor cycle may reach 1.50 $ plus a temperature effect of 0.25 $. A Xenon transient may occur during an intermediate shut down: $\delta\rho = \Sigma_{Xe.eq}/\nu\,\Sigma_f = 0.0263$ or 3.7 $.

The reactivity worth of the shim-safety rods must be at least 10 $, which are distributed e.g. as follows:

4.5 $ for safety action during an incident power excursion and/or in case of malfunctioning of another shim-safety rod in a coupled region of the reactor core.

3.7 $ taking into account for Xenon transient when the reactor has its full power rating.

1.75 $ for Xenon-Samarium and temperature transients at the initial start-up of the reactor cycle and after this, for uranium consumption.

( $\sim$ 1 $) for equilibrium adjustments in the regions of the reactor core.

The reactivity for **uranium consumption** must be 2 $ or $\frac{\delta f}{f} = (\delta\rho = 2\,\$) \times (\beta_{eff} = .0072) = 0.014$ where f is the utilization factor of uranium for thermal neutrons $f = \Sigma_f / \Sigma_a$. The gradient of the utilization factor of uranium f is lower for higher contents of uranium per cell, giving thus higher burnup available (fig. 11).

**5.4. Economical use of the fuel elements**

The economical use of the uranium content in the fuel elements is the result of a three-parameter analysis:

1) excess of reactivity given by the reactivity worth of the shim-safety rods in a core configuration and the reactivity balance between the fuel elements and the experimental devices.

2) maximum burnup of the partially burnt elements at elimination.

3) additional burnup of the fuel elements, modified by addition of burnable poisons for utilization on a longer operation period.

For a reactor core configuration composed of regions near criticality conditions containing a ring of six fuel cells and a shim-safety rod, we calculate that the average value of the parameter A of fuel cells in the reactor load must be at least:

| | |
|---|---|
| - just for criticality at the end of an operation period supposing that the coupling between the regions balances the leak of last and epithermal neutrons | 0.30 |
| - antireactivity Xenon, Samarium and temperature effect (5.5 $) | 0.06 |
| - antireactivity of experimental devices taken equal to the half of the reactivity worth of the shim-safety rods | 0.06 |
| - antireactivity by absorption in structure materials and matrix | 0.06 |
| **average parameter A in fuel cells** | **0.48** |

This value of the averaged part of neutrons available per fuel cell (A = 0.48) is given at the start-up of an operation period, for a fuel load composed of fresh and partially burnt fuel elements. These are needed for the adjustment of the hot spot and of the irradiation conditions of the experimental devices and for equilibrium of the reactivity balance between subassemblies in the reactor core. This value of the parameter A is a minimum because it assumes that the reactivity variation by consumption of highly enriched uranium equals the Xenon-Samarium transient (1.5 $) at the start-up of the operation period and that it is compensated.

- secondarily by the excess of reactivity contained in the shim-safety rods
- efficiently by the consumption of burnable poisons.

We consider that the maximum burnup of a partially burnt fuel element is reached, when this fuel element is no longer sufficiently reactive:

| | |
|---|---|
| - for the most compact core configuration (configuration 4) | $A_{min} = 0.221$ |
| - for the present configuration containing more fuel elements | $\delta A = -.055$ |
| - Xenon-Samarium peak and temperature effect (1.75 $) | $\delta A = 0.016$ |
| - absorption in structure materials and matrix | $\delta A = 0.06$ |
| - absorption in experimental device | $\delta A = 0.06$ |
| total | $A = 0.30$ |

The lower limit of part of neutron available per fuel cell is thus: A = 0.30 representing a **mean burnup at elimination** of (fig. 13) :

32 % for alloy fuel element VIn 244 g $^{235}U$
57 % for cermet fuel element VIn 400 g $^{235}U$

The **operation period of a fuel load** is determined by the possible variation of the averaged part of neutrons available in the fuel cells; this is given by

- the value of the Xenon-Samarium transient of reactivity (1.50 $) if samarium is added as a burnable poison;

- the composition of uranium and of burnable poisons;

- the excess of reactivity contained in the shim-safety rods and in the retractable absorbing devices, limited by the reactivity of the core configuration and by the antireactivity of the experimental load.

The **economical use of highly enriched uranium** depends on the presence of burnable poisons, like boron; these have beneficial effects

- increase of the operation time of the fuel load;

- higher burnup of the fuel elements at elimination, since the fresh fuel elements have a higher content of highly enriched uranium to recover an acceptable part of neutrons available per fuel cell despite the burnable poison;

- reduction of the reactivity worth of the absorbing devices, due to the increase of the total absorption in the reactor;

- harder neutron spectrum.

## 6. CONCLUSION CONCERNING THE USE OF HIGHLY ENRICHED URANIUM AT BR2

At the Belgian engineering testing reactor BR2, the use of highly enriched uranium in the driver zone is combined with a specially designed core configuration satisfying the irradiation conditions of irradiation loops with efficacity and flexibility.

In the reactor BR2, the predominant process is an elastic scattering of neutrons with a degraded energy spectrum and the hexagonal lattice is large enough ( $\Sigma_{th} r_e \geq 2$) so that we can only consider the dimentionless parameter $A_i = 2r_e (\nu\bar{\Sigma}_f - \bar{\Sigma}_a)$, where $\nu\bar{\Sigma}_f$ and $\bar{\Sigma}_a$ are the average macroscopic cross-section for fission production and for thermal absorption (including the leak out of the system and the sinks in the cadmium screens) and where $r_e$ is the equivalent radius of the cells; this parameter $A_i$, which we have called part of neutrons available per cell, is determined by three conditions:

1) the size of the lattice cell and of the fuel element;

2) the content and the composition of the fuel element ($^{235}U$, burnable poison ...);

3) the microscopic distribution of the thermal neutron flux in the lattice cell.

The reactivity of core configurations analysed shows that the size of the core configurations or the number of fuel elements has a lower influence on the reactivity of a load than the part of neutrons available per cell, which is defined by the characteristics of the fuel elements, of the shim rods and of the experimental devices.

The lower limit of the part of neutron available per fuel cell is $A = 0.30$ for fuel elements at elimination in large and in compact core configuration; at BR2 this limit represents in the standard fuel element of BR2 (Ø 84 mm, fuel length 76.2 cm, six concentric fuel plates) (see figure 13):

32 % burnup for alloy fuel elements, 244 g $^{235}U$ when fresh
57 % burnup for cermet fuel elements, 400 g $^{235}U$ when fresh.

The average part of neutrons available per fuel cell necessary for the start-up of an operation period and for coupling of the regions of the reactor core containing a shim-safety rod or an irradiation loop is $A = 0.46$; this presupposes at BR2 an average content of $^{23}U$ per fuel cell higher than 220 g $^{235}U$, plus the counterpart of burnable poisons and of the partially burnt fuel elements. Let us note that the presence of $^{238}U$ (by lower enrichment) increases slightly the term of absorption and modifies the microscopic distribution of the thermal neutron flux for the determination of the parameter $A_i$.

The upper limit of the uranium content in a fuel cell is given by the maximum density of uranium, which a fuel plate can contain. These limits are:

- metallurgical composition: $UAl_4$ in alloy fuel elements and $UAl_2$ in actual fuel elements;
- performance under irradiation: maximum density of fissions in fuel plates, limited by the presence of fission gazes, by cladding corrosion, by the maximum heat flux, ...;
- minimum water gap between the fuel plates limited by the pressure drop on the reactor core and by the possible coolant flow, avoiding the risks of boiling at the hot plate.

An economical use of highly enriched uranium is obtained by a good management of the driver zone complying with the requirements of the irradiation loops, by a sufficiently long operation period of the fuel load and by a high burnup of fuel element at elimination. Let us note that the presence of burnable poisons, like boron, in the fuel elements, is needed, not only to reach higher burnup at elimination and to increase the operation time of a fuel load, but also to furnish a large spectrum of partially burnt fuel elements, necessary to ensure the various irradiation conditions required by the experimental devices.

Taking into account the present manufacturing technology of the MTR fuels, the highly enriched uranium cannot be avoided because of the high value of the uranium content, fixed now by the maximum density of uranium, which a fuel plate must contain. Presently, a standard fuel element of BR2 (VIn type) contains 400 g $^{235}U$ (93 % enriched) within six concentric plates (thickness of cladding and fuel meat: 1.27 mm), i.e. 60 mg/cm$^2$ of $^{235}U$ in the form of $UAl_3$ (+ 3.8 g boron ($B_4C$) + 1.4 g samarium ($Sm_2O_3$)), representing 37 Wt % uranium density in the fuel meat; these fuel elements permit three reloadings and the burnup at elimination is reached at 57 % mean burnup. If higher concentration of uranium are developped in the manufacturing technology, the value of the parameter A of the fuel elements at elimination must be kept at the required level.

# APPENDIX A - PART OF NEUTRONS AVAILABLE PER CELL

## 1. DESCRIPTION OF THE METHOD

In many cases, a reactor may be discretized into cells containing either a fuel element cooled by water, or a reflector beryllium plug surrounded by similar cells, in such a way that the predominant process is an elastic scattering of degraded energy spectrum of neutrons, accompanied by

- the leak of neutrons out of the system by diffusion
- the absorption in fuel, poison or structure materials
- sources by slowing-down of the epithermal neutrons
- sinks by cadmium screens in the shim-safety rods and/or experimental devices.

The **transport theory** in large cells permits easy calculations of the averaged thermal neutron flux per cell

$$\int \Sigma_t \ \phi \, dv_i = \sum_j \int ( \xi \Sigma_s \ \phi_{epi} + \Sigma_s \phi ) \, dv_j \, P_{j \to i}$$

where $\Sigma_t$ [cm$^{-1}$] total macroscopic cross-section

$\Sigma_s$ [cm$^{-1}$] scattering macroscopic cross-section

$\Sigma_a = \Sigma_t - \Sigma_s$ absorption macroscopic cross-section and axial leak

$\Sigma_f$ [cm$^{-1}$] fission macroscopic cross-section

$\xi \Sigma_s$ [cm$^{-1}$] slowing down macroscopic cross-section

$\phi$ [n/cm$^2$.s] thermal neutron flux

$\phi_{epi}$ [n/cm$^2$.s] epithermal neutron flux

$P_{j \to i}$ collision probability in the cell $\int dv_i$ for a neutron born in the cell $\int dv_j$

If the equivalent radius of a cell is large enough: $\Sigma_t r_o > 2$ the non escape probability for a neutron born uniformly in an infinite cylinder is:

$$P_{i \to i} = 1 - 1/(2 \ \Sigma_t r_o) + 0.1084/( \Sigma_t r_o)^3 + \ldots$$

It is thus possible to consider

$$P_{i \to i} = \frac{1}{6}\,(1 - P_{i \to i}) \quad \text{for first vicinity cells in a hexagonal lattice}$$

$$P_{i \to i} = 0 \qquad \text{for second vicinity cells}$$

Let us denote by $K_{i \to i}$ the probability that a fission neutron starting within $\int dv_i$ becomes thermal in $\int dv_i$ ; if the slowing-down kernels are normalised $\sum_i K_{i \to i} = 1$, then the epithermal source reads :

$$\int \xi \Sigma s.\ \phi_{ep.}\, dv_i = k \sum_j \int \nu \Sigma f.\ \phi . dv_j . K_{j \to i}$$

where the factor k includes the leak of fast and epithermal neutrons.

The slowing-down kernels permit, without too much trouble, to make it unnecessary to evaluate the slowing-down process by determining the spectral distribution. The equation for the stationary state of the thermal neutron flux becomes:

$$\int \Sigma t\ \phi\, dv_i = \sum_j \int \Sigma s\ \phi\, dv_j\, P_{j \to i} + k \sum_j \int \nu \Sigma f\ \phi\, dv_j\, K_{j \to n}\, P_{n \to i}$$

The adjoint of the thermal flux has a more simple equation:

$$\int \Sigma t\ \phi^*\, dv_i = \sum_j \int \Sigma s\, \phi^*\, dv_j\, P_{j \to i} + k \sum_j \int \nu \Sigma f\ \phi^*\, dv_j\, P_{j \to n}\, K_{n \to i}$$

because the slowing-down process is smoother than the diffusion process.

The equation of the adjoint of the thermal flux becomes simple if most of the factors $P_{i \to i}$ are zero:

$$\int \Sigma t\ \phi^*\, dv_i \approx \sum_j \int (\Sigma s + k\ \nu \Sigma f)\, \phi^*\, dv_j\, P_{i \to i}$$

$$\int (\Sigma t - (\Sigma s + k\ \nu \Sigma f)\, P_{i \to i})\ \phi^*\, dv_i = \sum_{j \neq i} \int (\Sigma s + k\ \nu \Sigma f)\ \phi^*_i\, dv_j\, P_{j \to i}$$

$$(\alpha_i - A_i)\ \bar{\phi}^*_i = \frac{1}{6} \sum_{j \neq i} \alpha_j\ \phi^*_j$$

first vicinity

where $A_i = 2 r_o (k \nu \bar{\Sigma}_{fi} - \bar{\Sigma}_{ai} + \bar{\Sigma}_{si}) = 2 r_o (k \nu \bar{\Sigma}_{fi} - \bar{\Sigma}_{ai})$

$$\alpha_i = (\bar{\Sigma}_{si} + k \nu \bar{\Sigma}_{fi}) / \bar{\Sigma}_{ai} = 1 - A_i/(2 r_o \bar{\Sigma}_{ai}) \approx 1$$

since $P_{i \to i} = 1 - 1/(2 \Sigma_{ai} r_o)$ and $P_{i \to j} = \frac{1}{6} (1 - P_{i \to i})$

This elliptic system is written very simply: the adjoint of the thermal flux in the lattice cell i is related to the adjoint flux in the first vicinity cell j (hexagonal lattice). This relation is a function of a single significative dimensionless parameter $A_i$, which we have called **part of neutrons available per cell.**

For certain arrangement of a fuel load, which we call core configuration. it is possible to discuss the requirements of the irradiation loops and the requirements of the driver zone by calculating :

1) The map of the adjoint of the thermal neutron flux in function of the arrangement of the fuel elements; this map is called unperturbed flux when the coefficients $A_i$ of each fuel cell are all equal to each other.

2) The map of the perturbed thermal flux (first approximation).
Since $\bar{\Sigma}_a \bar{\phi} \approx \xi \bar{\Sigma}_s \bar{\phi}_{epi}$ and $(1 - A_i) \phi^*_i = \text{const.}$

we define the local perturbation factor in a fuel cell:

$$F_i = \frac{\phi_i}{\phi_{ic}} = \frac{\bar{\Sigma}_{ao}}{\bar{\Sigma}_a} \times \frac{1 - 2 r_o (\nu \Sigma_{fo} - \Sigma_{ao})}{1 - 2 r_o (\nu \Sigma_f - \Sigma_a)}$$

This perturbation factor permits very quick calculations for the fuel management of balanced loads, giving a sufficient information about the averaged thermal and fast fluxes and of the local reactivity of experimental devices and fuel elements.

3) The critical value of the coefficients $A_i$ or the excess of reactivity of a configuration, may be calculated by a simple approximation, having the following quadratic form:

$$\rho = \frac{k_{eff} - 1}{k_{eff}} = \frac{\frac{1}{6} \sum_i \sum_{(i)} \alpha_i \phi^*_j \phi^*_i - \sum_i (\alpha_i - A_i) \phi^{*2}_i}{\sum_i 2 r_o \nu \bar{\Sigma}_{fi} \phi^{*2}_i}$$

The term : $\left( \sum_i \phi_i^{*2} - \frac{1}{6} \sum_i \sum_{i \neq j} \phi^*_i \, \phi^*_j \right) / \sum_i 2 r_0 \nu \Sigma_{fi} \phi^{*2}_i$

means the leak of thermal neutrons out of the system, while the leak of fast and epithermal neutrons is not expressed here.

..) In practice, if a given load is balanced, the comparison of the excess of reactivity is easily achieved by summing up the local variation of the reactivity due to absorbing materials in fuel cells, according to the first approximation of the perturbation theory:

$$\delta\rho = \frac{\sum_i \int (\delta \nu \Sigma_{fi} - \delta \Sigma_{ai}) F_i \, \phi^2_{io} \, dv_i}{\sum_i \int \nu \Sigma_{fi} F_i \, \phi^2_{io} \, dv_i}$$

where $\phi_{io}$ is the unperturbed thermal flux
$F_i$ is the perturbation factor, calculated in § 2.
Let us note, that regional summation of local reactivity effects aids to ensure a balanced load.

The useful dimensionless parameter $A_i$, i.e. the part of neutrons available per cell, is determined by three conditions:

1) the size of the lattice cell and of the fuel element;

2) the composition of the fuel element ($^{235}U$, burnable poisons, ...);

3) the microscopic distribution of the thermal neutron flux in the lattice cell.

Remarks must be made about this simplified calculation for neutrons available in general fuel arrangements in the reactor core:

1) The leak of fast and epithermal neutrons is not represented, except by the undetermined factor k, as the slowing-down kernels have been normalized. Harder neutron spectrum as well as smaller fuel core imply a greater leak of fast epithermal neutrons.

2) The adjoint of the thermal neutron flux may represent the unperturbed thermal flux if the fuel cells are identical (homogeneous fuel load); this permits perturbation calculations from the reference load. It should be noted that for fuel elements, without burnable poison, the local perturbation factor depends only on the squared mean burnup $\bar{\beta}^2$ if the part of neutrons available for fresh fuel elements is $A = 0.50$.

3) The thermal neutron flux must be calculated or measured inside experimental devices and fuel cells by more accurate methods, which take into account the microscopic depression of the flux and the adequate boundary conditions (net current of thermal neutrons, power generated in the surrounding cells, ...).

4) The hardness of the neutron spectrum is a function of the absorbing load; therefore it is higher for fuel elements containing burnable poisons. It must be noted that many irradiation loops are interested rather in the fast neutron flux than in the thermal neutron flux, which may be eliminated by a cadmium or a silver screen.

## 2. REQUIREMENTS FOR IRRADIATION LOOPS

The requirements for irradiation loops consist of:

1) fast epithermal and thermal neutron fluxes effectively available for the experimental devices;

2) the reactive effect of the irradiation loops in the core configuration, which may not collide with safe operation conditions.

These requirements depend on the **size** of the irradiation loops, whether it occupies the center part of a fuel element, or a whole lattice cell, or even a larger channel representing a multiple of a standard lattice cell. The fast and epithermal neutron flux depends mainly on the **power** generated in the close vicinity, while the thermal neutron flux depends on the **microscopic flux distribution**, taking into account the experimental device itself and the surrounding fuel element or obturator plug.

Fortunately, the most absorbing devices: shim-safety rods, cadmium screens, ... may be expressed very simply by considering the black surface when the part of neutrons available per cell is determined : $A = -2\,\phi/r_o$, where $\phi$ is the outer black diameter(no inner slowing-down) and $r_o$ is the equivalent radius of the lattice cell.

It is also fortunate, that if an experimental device is completely surrounded by identical fuel elements and the influence of the second vicinity cells is neglected, it can be shown that the relative distribution of the local adjoint of the thermal neutron flux depends only on the characteristics of the experimental device, in such a way that it is possible to relate the requirements of the experimental device to the characteristics of the fuel cells by considering the single parameter A, called part of neutrons available per cell. If an experimental device is not completely surrounded by a fuel element ring, it becomes evident that the distribution of the adjoint of the thermal neutron flux is only slightly changed by various values of the parameter A of identical fuel elements.

The solution of **very simple arrangements** of fuel cells in the form of hexagonal lattice is sufficient to estimate the requirements of irradiation loops, expressed as part of neutrons available per cell: For the example of a symmetrical system of six identical fuel cells, surrounding a reflector cell loaded with an irradiation loop or a shim-safety rod, the system is expressed by the following equation:

$$(1 - A_i)\ \phi^*_i = \frac{1}{6} \sum_{\text{first vicinity}} \phi^*_j$$

The map of the adjoints of the thermal flux for this arrangement, reads:

$$\begin{bmatrix} A_o-1 & 1 & 0 & 0 \\ 1 & 6A-4 & 2 & 1 \\ 0 & 2 & -6 & 2 \\ 0 & 1 & 2 & -6 \end{bmatrix} \begin{bmatrix} \phi^*_o \\ \phi^*_u \\ \phi^*_2 \\ \phi^*_3 \end{bmatrix} = 0 \text{ which leads to } \begin{array}{l} \phi^*_o = 1/(1-A_o) \\ \phi^*_u = 1 \\ \phi^*_2 = 0.4375 \\ \phi^*_3 = 0.3125 \end{array}$$

The excess of reactivity may be calculated in first approximation. by the quadratic formula:

$$\rho = \frac{k_{eff} - 1}{k_{eff}} = \frac{\frac{1}{6} \sum \sum \phi^* \phi^* - \sum (1-A_i) \phi^{*2}}{\sum 2 r_o \nu \bar{\Sigma}_f \phi^{*2}} = \frac{1/(1-A_o) - 6A - 2.8125}{6 \times 2 r_o \nu \bar{\Sigma}_f}$$

if $k_{eff} = 1$ or $\rho = o$, the value of the critical parameter A of the fuel cells is:

$$A = 0.46875 - 1/6(1-A_o)$$

The reactivity effect of the irradiation loop ($A_o$ is negative) in terms of the additional part of neutrons available needed in the fuel cells and of the importance of the reactivity effects follows immediately. If a shim-safety rod takes the place of the irradiation loop, its withdrawal from the reactor leads to an increase of the part of neutrons available which may give sufficient flexibility for the management of the experimental loading.

*The contribution of the part of neutrons available per cell due to* $^{235}U$ *in a fuel cell may be* roughly approximated by the expression

$$A = 2 r_o ( \nu \bar{\Sigma}_f - \bar{\Sigma}_a) = 1.874 \times \frac{2 r_o Mu5}{r_o^2 L_f} = 7.5 \frac{Mu5}{2 r_o L_f}$$

where $2 \pi r_o L_f$, is the external surface of the fuel cell; at BR2 this rough approximation gives $A \approx Mu5/324$ g ($r_o = 5.063$ and $L_f = 76.2$ cm) but according to the microscopic thermal neutron flux, a better more accurate calculation gives (fig. 11, 12):

$A \approx - 0.06 + Mu5/400$ g in the standard fuel cell.

The **lower bound of the part of neutrons available** per cell for an assembly of seven identical fuel elements, without experimental device and without absorbing regulating rod, is obtained from the condition $\rho = o$:

$$6A + 1/(1-A_o) - 2.8125 = o \text{ or } A_{min} = 0.25 \text{ if } A_o = A$$

Six fuel elements will give $A_{min} = 0.30$ if $A_o = o$

This lower bound of the parameter A must be understood as not including the leak of fast and epithermal neutrons, as if the load of the reactor were a large assembly composed of several clusters of fuel elements, each of which being in near critical conditions, in order to satisfy with flexibility the irradiation conditions of many irradiation loops. General particularities of $A_i$ per fuel cell for clusters of fuel elements, may be accepted for any reactor core, if the reactivity is equilibrated. If the core assembly is small, one third of the neutrons produced may leak out as fast or epithermal neutrons, in such a way that the criticality is given for a calculated $k_{eff} = 1.333$. Assuming that $\bar{\Sigma}_f / \bar{\Sigma}_a = 0.65$ and $\nu = 2.43$, we get:

| | | |
|---|---|---|
| for seven fuel elements | $A_{min} = 0.27$ | if $A_o = A$ |
| for six fuel elements | $A_{min} = 0.34$ | if $A_o = o$ |

The ***upper bound of the part of neutrons available*** *per cell for a compact core of six identical* fuel elements, surrounding an irradiation loop containing a large cadmium screen ($A_o = - 4$) and without absorbing regulating rod, is given by:

$$6A + 1/(1-A_o) - 2.8125 = 0 \quad \Rightarrow \quad A = 0.435 \qquad \text{if } A_o = - 4$$

neglecting the leak of fast and epithermal neutrons; if the compact core is incomplete and contains less then 6 fuel elements' the upper bound of the parameter A must be higher. Such a case would be a core of a beam reactor.

If a **shim-safety rod** occupies the central hole of a symmetric assembly of six identical fuel elements, the reactivity worth of this shim-safety rod can be calculated as part of neutrons available in the fuel cells:

$$6\,\delta A = 1 - 1/(1 - A_o)$$

For example, a cadmium screen $\phi = 6.0$ cm in a cell of radius $r_o = 5.063$ cm and for fuel elements for which:

$$2\,r_o\ \bar{\nu\Sigma}_f = 1.32 \text{ (alloy standard fuel element at BR2)}$$

and an effective delayed neutrons fraction $\beta_{eff} = .0072$
we found:

$$\delta A = \frac{1}{6} \times .703 = 0.12 \quad \text{if } A_o = -2.37$$

$$\delta\rho = \frac{1}{\beta_{eff}} \times \frac{\delta A}{2\,r_o\,\nu\Sigma_f} = 12.3\ \$$$

## 3. APPLICATION OF THE METHOD FOR THE REACTOR BR2

The microscopic distribution of the thermal neutron flux in a standard fuel element may be calculated by the integral transport theory applied to an annular and concentric geometry and corrected for the epithermal neutron flux. Some results of this calculation are interesting (fig. 11) [12] :

1) The disadvantage factor of the fuel plates is $f_u = \phi_{U5} / \phi_{H2O}$

for an alloy fuel element A VIn 244 g $^{235}U$ $f_u = 0.910$
for a cermet fuel element G VIn 400 g $^{235}U$ $f_u = 0.857$

2) The absorption of neutrons takes mainly place in the fuel

$$\bar{\Sigma}_a = 0.0134 + 0.1142\,f_u \frac{Mu5}{400\,g} + \Sigma_a \text{ poisons} + \ldots [cm^{-1}]$$

3) The slowing-down effect takes mainly place in the water

$\bar{\xi\Sigma}_s = 0.509\ cm^{-1}$ at 0.5 eV in a standard fuel cell

$\bar{\xi\Sigma}_s = 0.220\ cm^{-1}$ at 0.5 eV in a reflector cell

4) The scattering is very important: it occurs mainly in the water, secondarily in the beryllium:

$\bar{\Sigma}_s = 1.80\ cm^{-1}$ and $P_u = 0.945$ in standard fuel cell

$\bar{\Sigma}_s = 0.944\ cm^{-1}$ and $P_u = 0.900$ in reflector cell

5) If the net current of neutrons is zero at the equivalent radius ($r_o = 5.063$ cm) of the fuel cell, a linear relation is found for the thermal neutron flux averaged in the mixture of water and fuel:

$$\overline{\xi\Sigma_s}\ \phi_{epi} = \overline{\Sigma}_a\ \overline{\phi}_{H_2O}$$

6) The thermal neutron flux in the central hole of a standard fuel element filled with an aluminium plug depends on the depression of the flux in the surrounding medium.

If the net current is zero at the boundary of the fuel cell, and after elimination of the absorbing term $\Sigma_{a.u5}$, the linear relations between thermal and epithermal flux in the three regions of a standard fuel cell become:

$$\phi_o = 0.750\ \phi_{Be} - 0.292\ \phi_{epi}$$

$$\phi_{H_2O} = 0.925\ \phi_{Be} - 0.670\ \phi_{epi}$$

The part of neutrons available per cell may be easily calculated for fuel cells and common experimental devices: $A_i = 2\ r_o\ (k\ \nu\overline{\Sigma}_{fi} - \overline{\Sigma}_{ai})$

For a particular fuel cell, the factor k is not known, because it depends on the fast and epithermal leaks out of the system; unless experimental determinations are available or difficult calculations are performed, it is assumed that $k = 1$.
For cadmium screens, it is assumed that only the diameter of the black surface is significative: $\overline{\Sigma}_a = - \ \emptyset/r_o^2$.

We thus obtain (fig. 12):

$A = Mu5/400\ g - 0.06$ for a standard fuel cell with alloy fuel element
$A = 1.32 - 0.77 = 0.55$ for a fuel cell containing a fuel element A VIn 244 g $^{235}U$
$A = 2.05 - 1.11 = 0.94$ for a fuel cell containing a fuel element A VIn 400 g $^{235}U$
$A = 2.05 - 1.44 = 0.61$ for a fuel cell containing a fuel element G VIn 400 g $^{235}U \pm$ poisons

$A = - 0.043$ for a reflector cell
$\delta A = - 0.20$ for a fafnir simulation rig $\emptyset_{Cd} = 2.53$ cm - 2 x $\lambda$ extrapol.
$\delta A = - 1.54$ for a regulating rod $\emptyset_{Cd} = 3.90$ cm
$\delta A = - 2.37$ for a shim-safety rod $\emptyset_{Cd} = 6.00$ cm

The perturbation factor may be calculated easily, according to the definition of the unperturbed thermal neutron flux given for a homogeneous composition of fuel cells: standard fuel element A 244 g $^{235}U$ filled with an aluminium plug. The macroscopic cross-sections for thermal neutrons have to be ponderated by the cosine axial flux distribution to take into account the burnup of the fuel elements (fig. 13 b). The calculated and measured values of the perturbation factor (arbitrary scale) are represented by figures 14 a and 14 b for standard fuel element VIn of type A (alloy) and G (cermet) in function of mean burnup. The reactive effect of the fuel element has been calculated and measured in function of mean burnup on fig. 15 [13, 14, 15].

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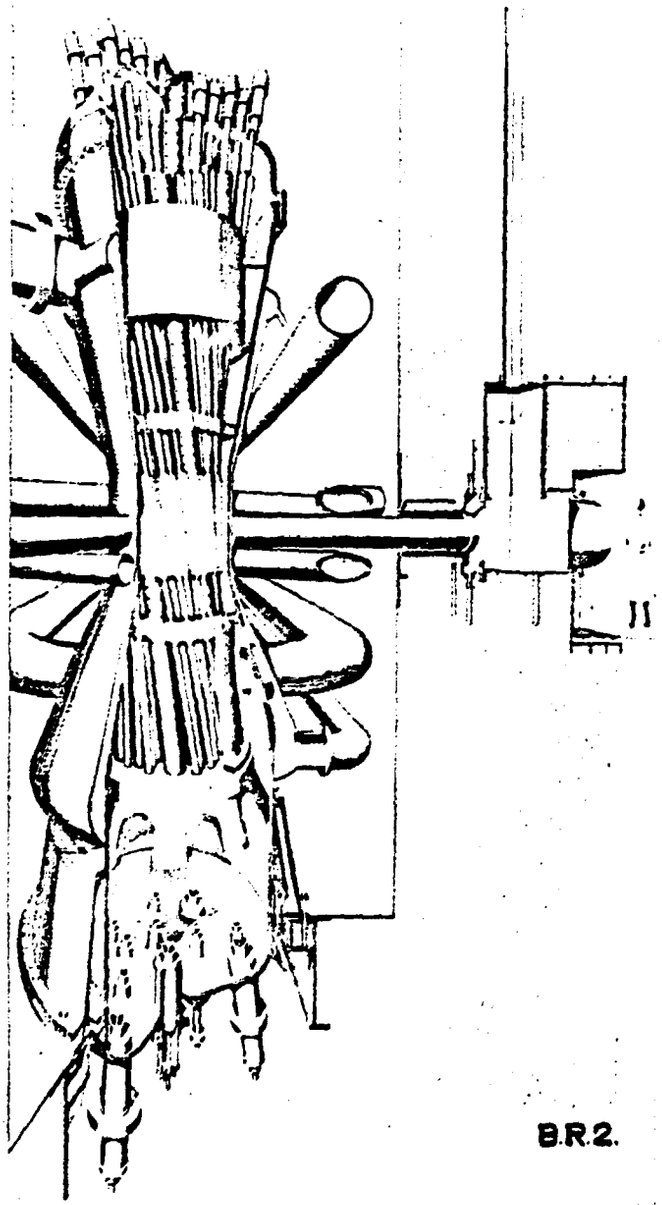
## LIST OF TABLES AND FIGURES

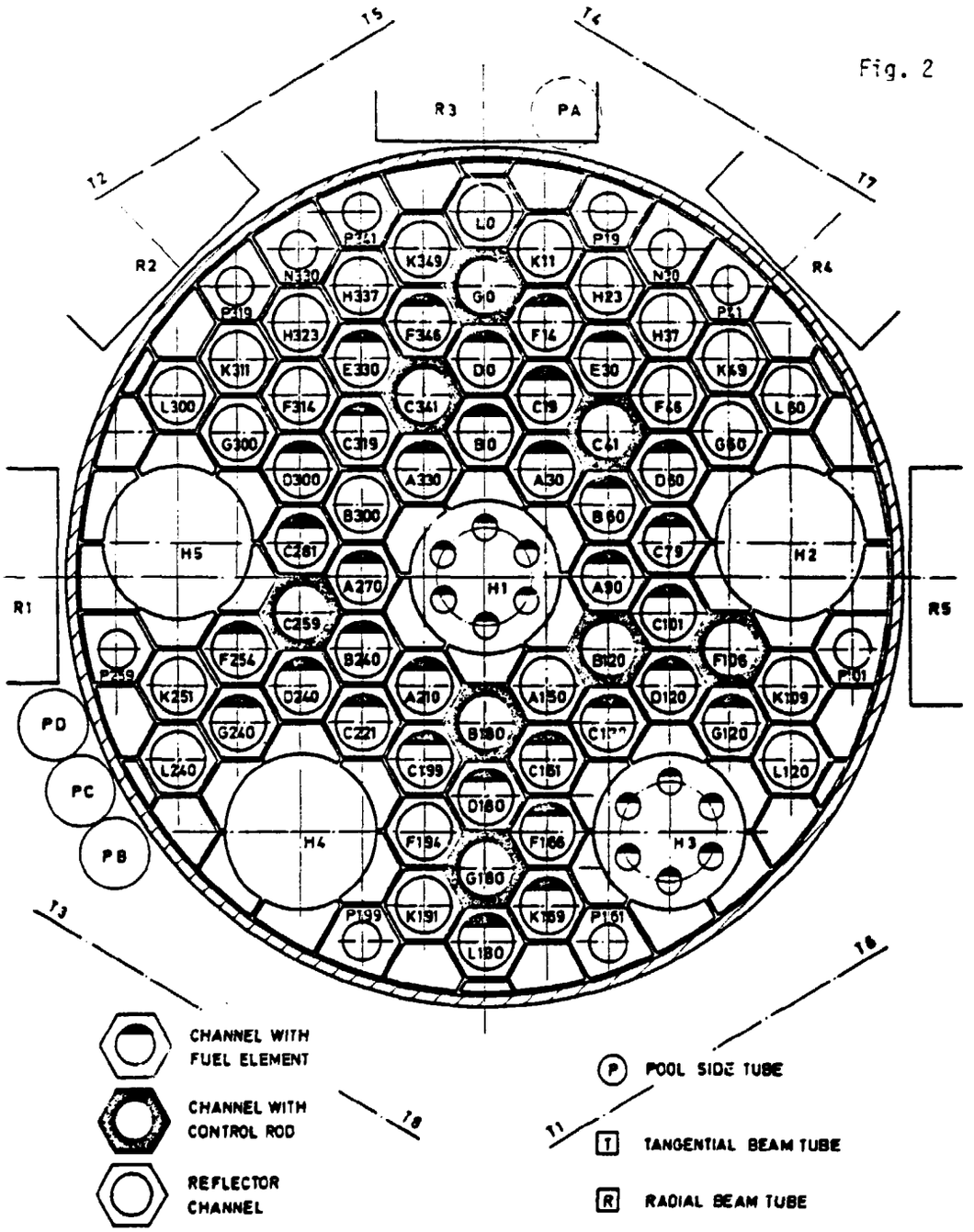
1. General view of the BR2 reactor
2. BR2 core - cross-section at mid plane
3. BR2 fuel element and beryllium filling plug
4. BR2 through H 3 channel with fuel element S XIIIs
5. Standard fuel element cross-section and dimensions
6. Neutron flux characteristics in partially burnt fuel elements at BR2
7. Nuclear heating and shim-safety rod position during a running period of BR2
8. Evolution of the parameters for operation of the reactor BR2
9. *Correlation between the enrichment at the end of an operation cycle and the antireactivity* of the experimental load
10. Neutron spectra in various reactors
11. Table of the microscopic radial distribution of the thermal neutron flux in standard fuel cell at BR2
12. Part of neutrons available per standard cell at BR2 as a function of $^{235}U$ content
13. Part of neutrons available per standard cell at BR2 as a function of the burnup of standard fuel element
14. Perturbation factor per fuel cell at BR2
    14 a. Theoretical
    14 b. Measured
15. Reactive effect of the fuel element at BR2
16. Radial distribution of the adjoint of the thermal flux in core configuration at BR2
17. Core configuration
    17 a. Core configuration 4
    17 b. Core configuration 7
    17 c. Core configuration 8

Fig. 1




General view of the BR2 reactor.

Fig. 2

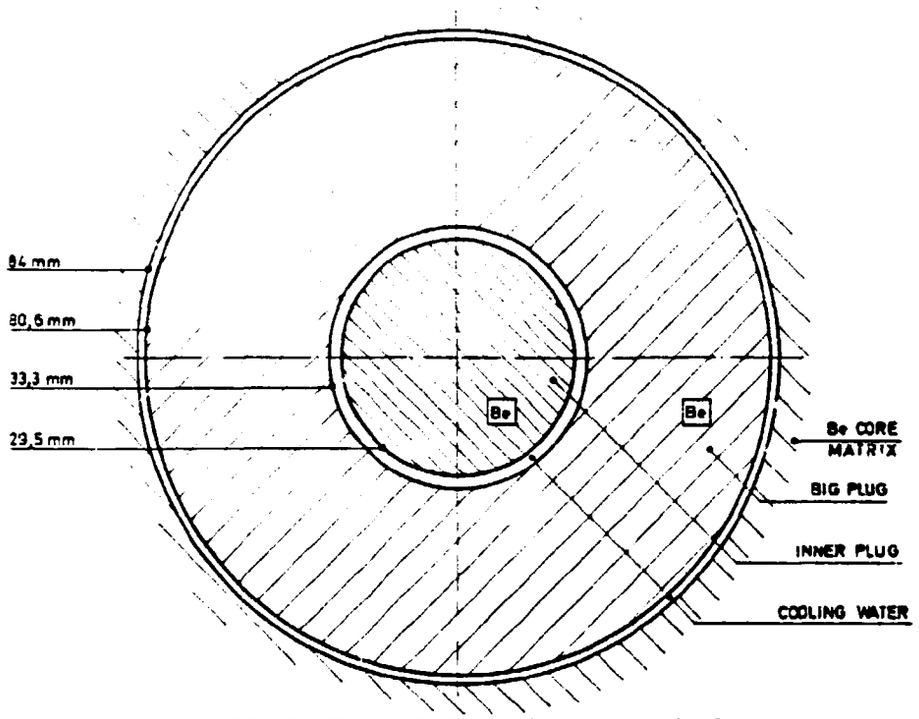



## BR2 CORE

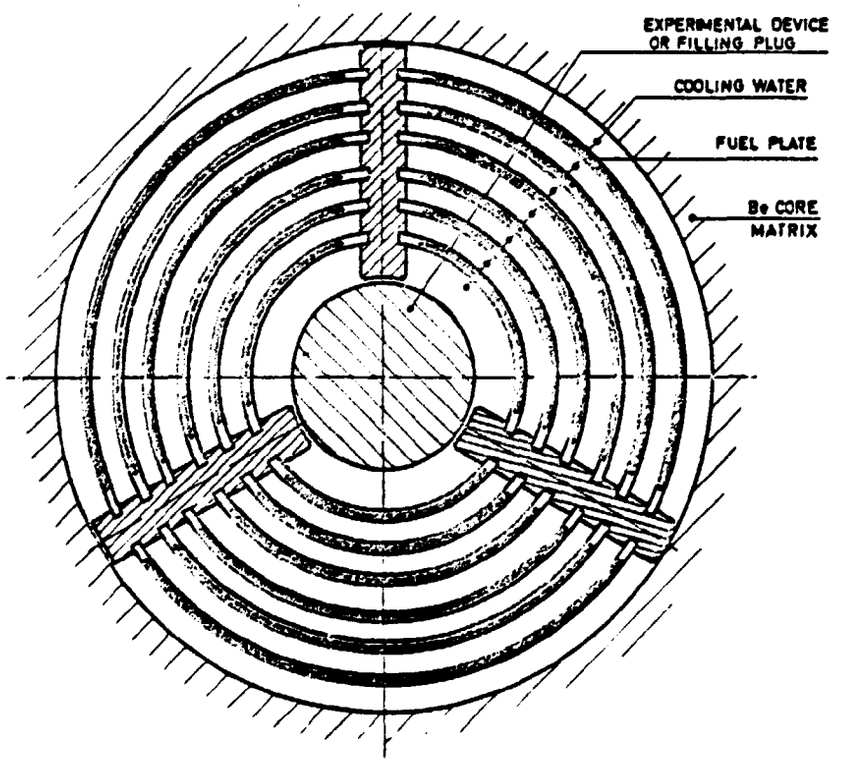
**(Cross section at mid plane)**

TYPICAL CONFIGURATION

Fig. 3




BR2 Be FILLING PLUG ( Cross section)




BR 2 FUEL ELEMENT ( Cross section )

Fig. 4

BR2 CROSS SECTION AT REACTOR MID-PLANE

THROUGH H3 CHANNEL WITH THE MFBS 7 LOOP

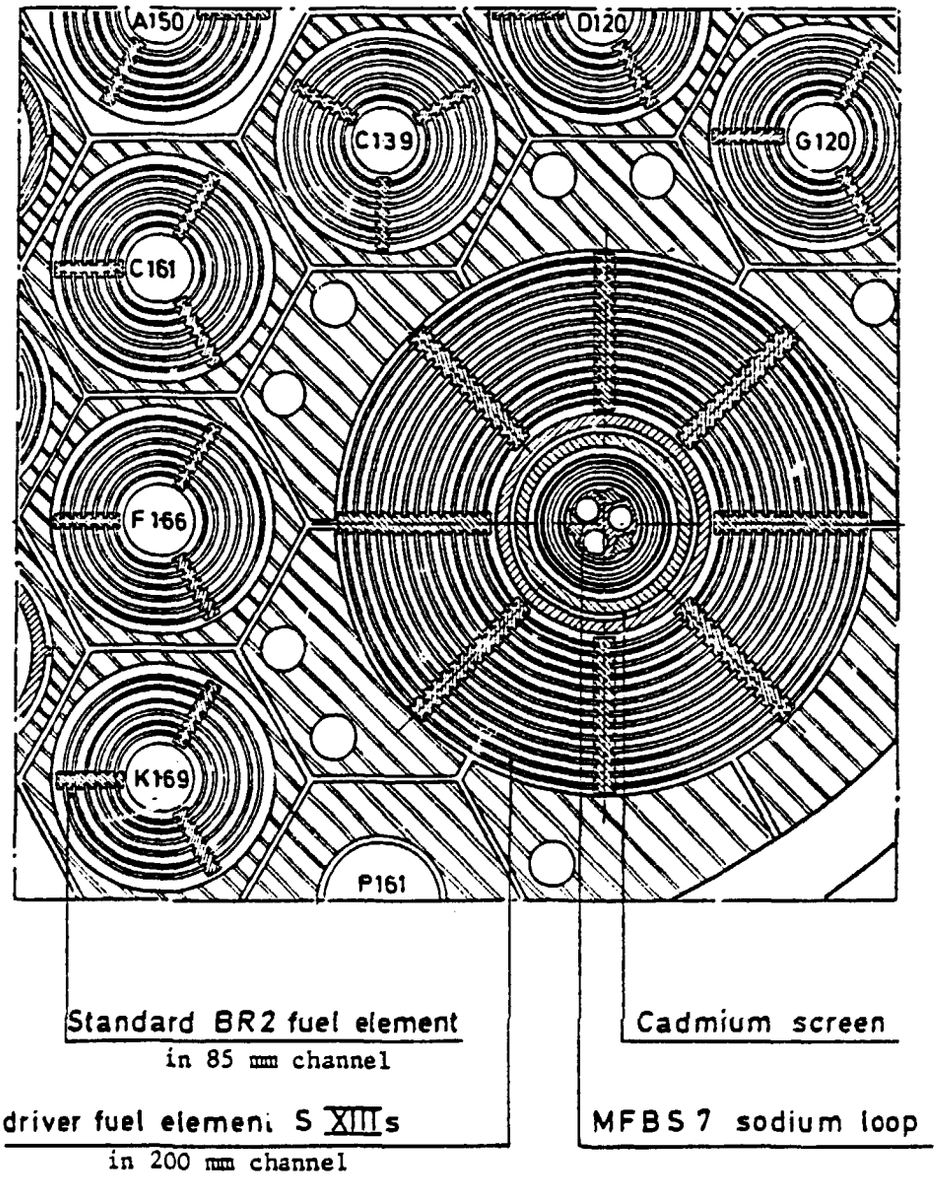
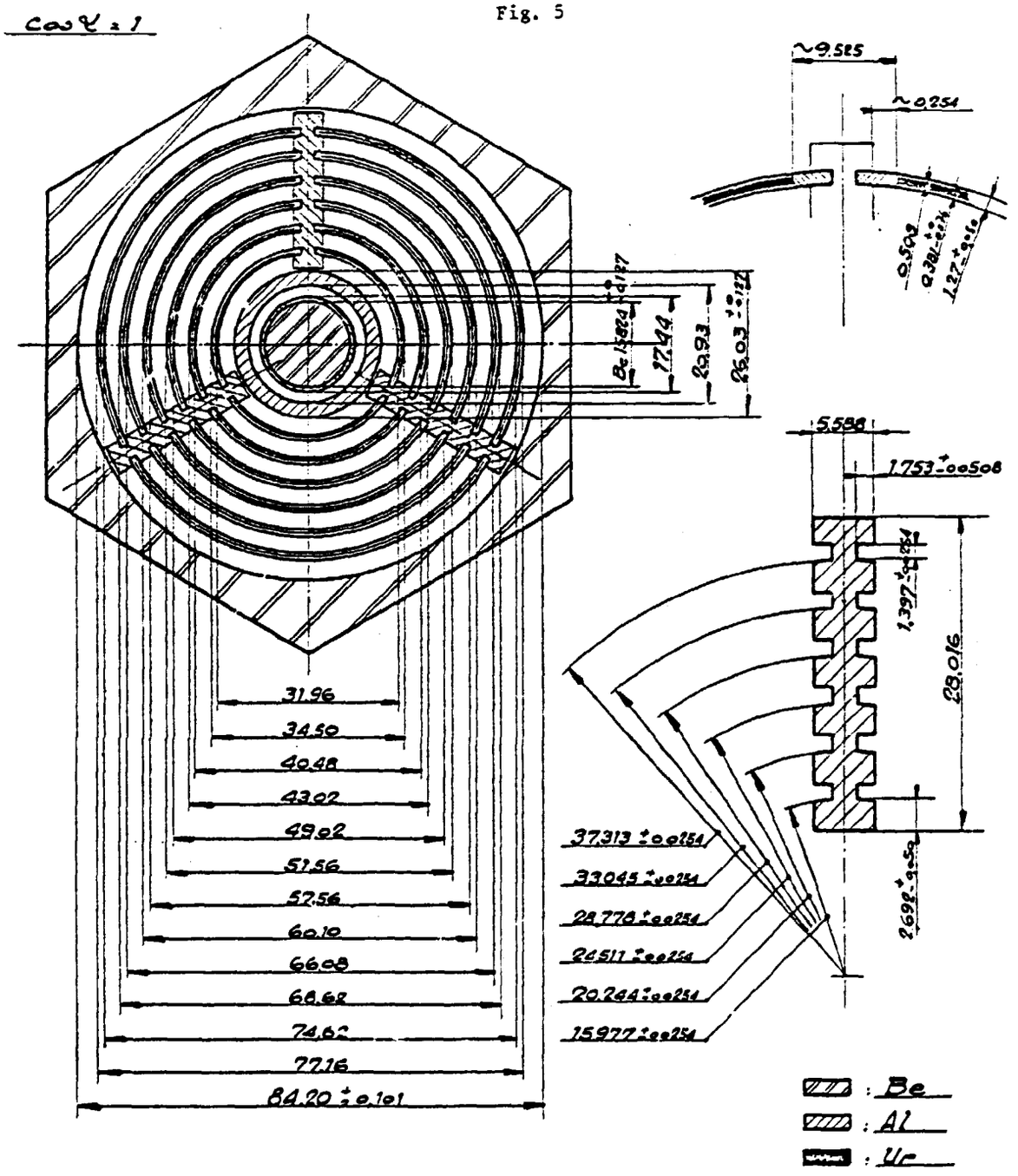


Fig. 5

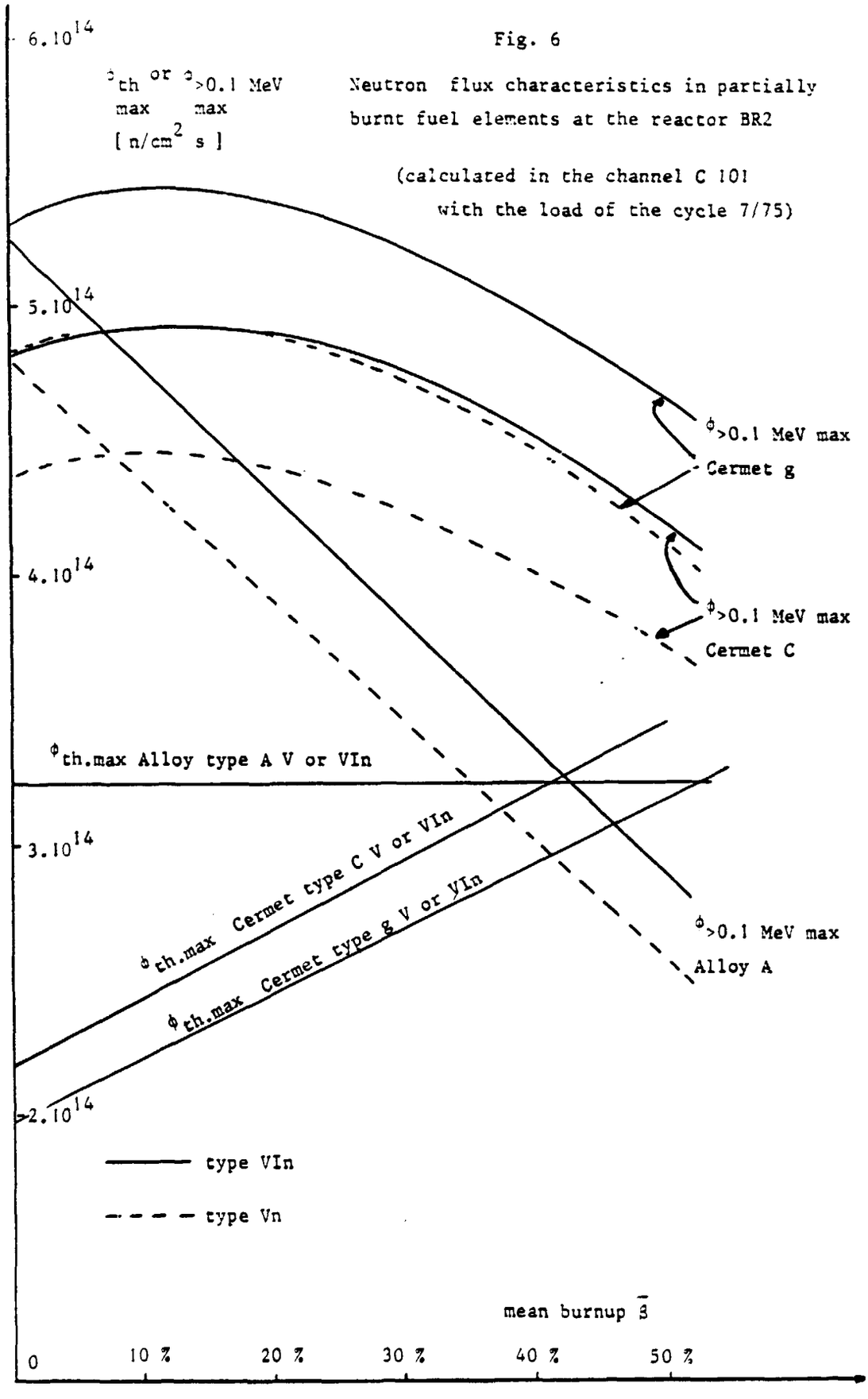



**Sylcor fuel element**
**Cross section**

Fig. 6

Neutron flux characteristics in partially burnt fuel elements at the reactor BR2

(calculated in the channel C 101 with the load of the cycle 7/75)



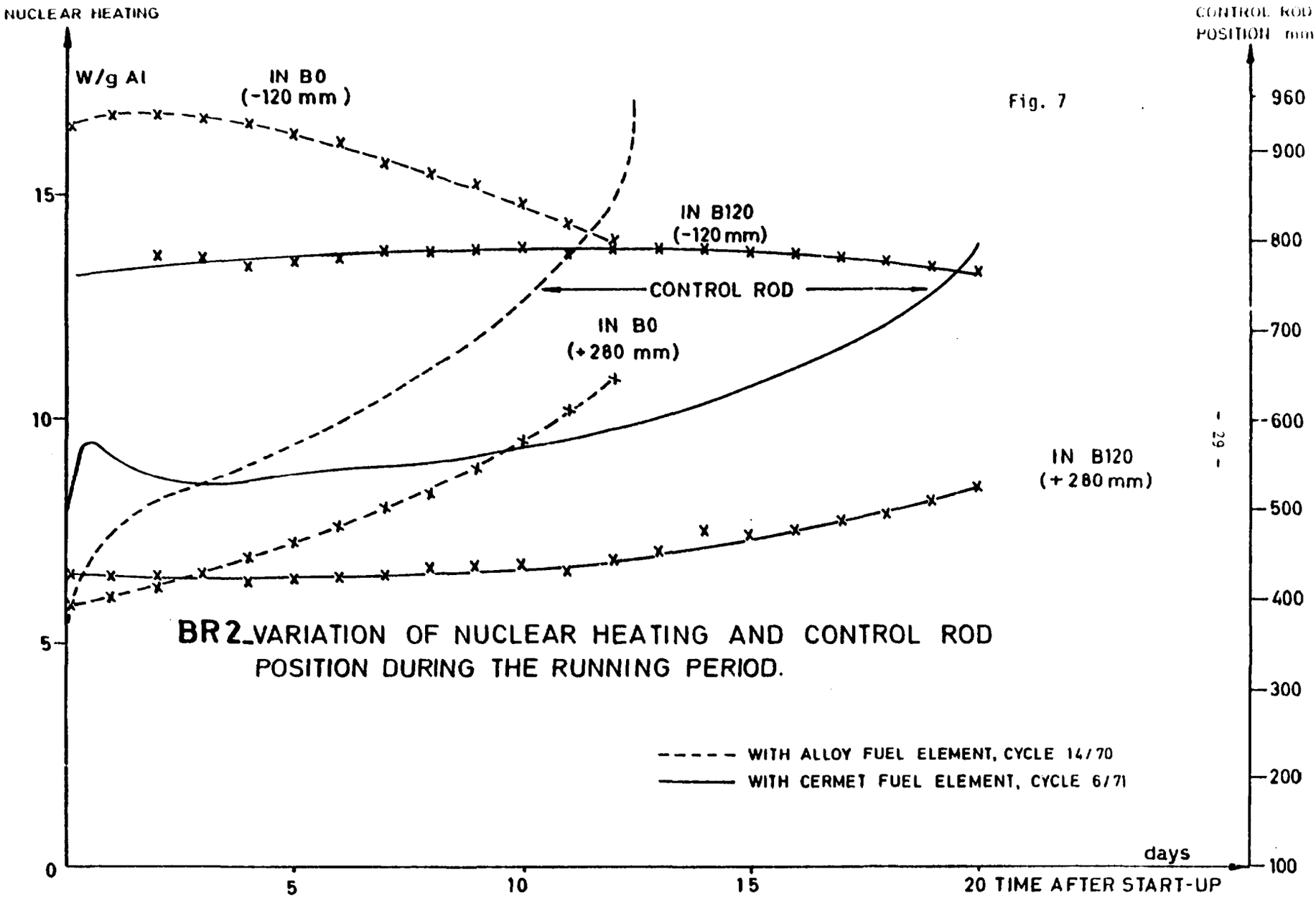



Fig. 7

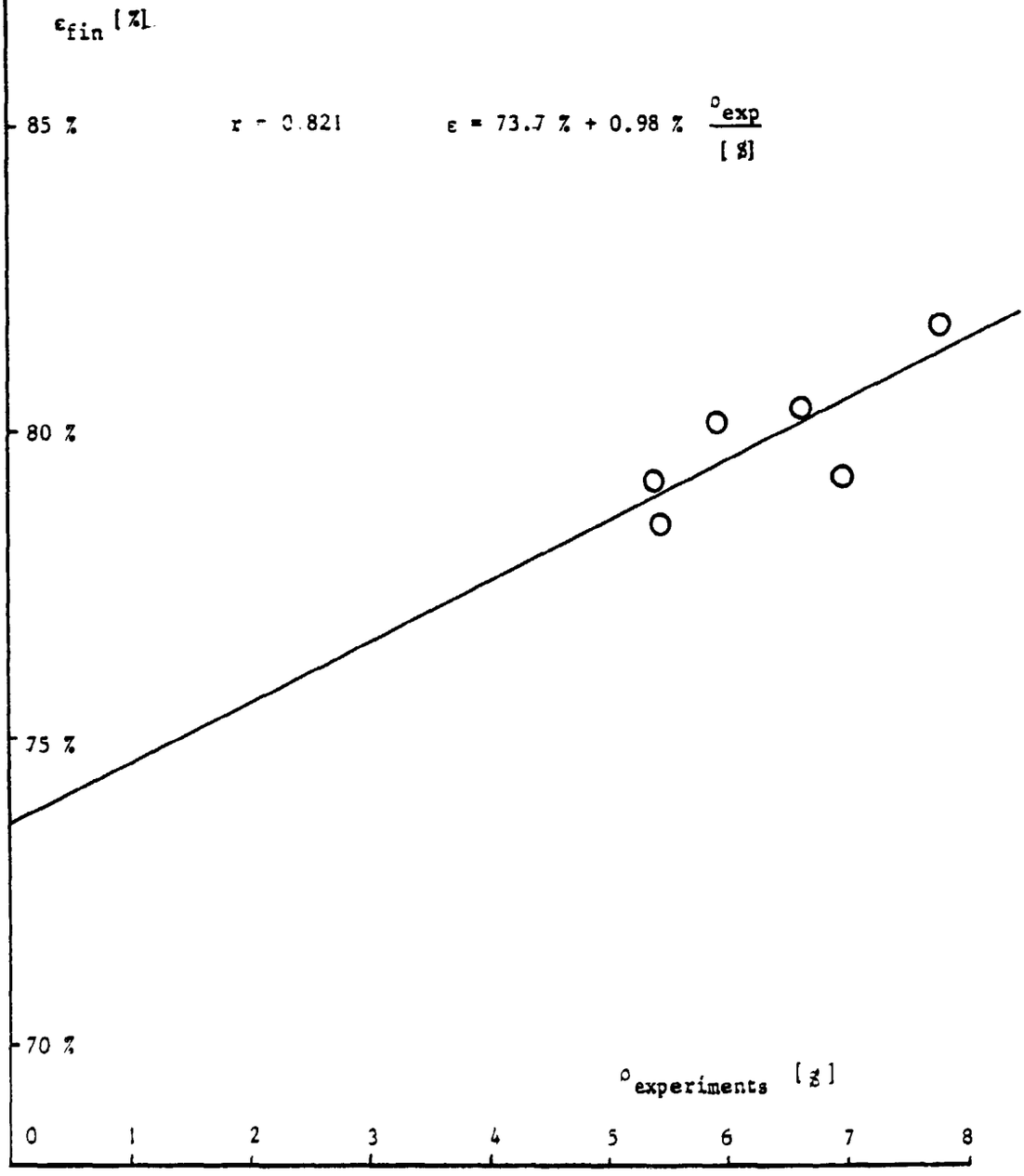
**BR2** VARIATION OF NUCLEAR HEATING AND CONTROL ROD POSITION DURING THE RUNNING PERIOD.

Fig. 8 Evolution of the parameters for operation of the reactor BR2

| Year | Configuration | | | Operation time | Nominal power | Energy produced | Uranium consumed | Number of fresh fuel elements used | | Specific consumption |
|---|---|---|---|---|---|---|---|---|---|---|
| | No. | fuel el. | control rods | (day) | (Mw) | (Mwd) | (kg) | alloy | cermet | fuel el/$10^3$ MWd |
| 1963 | 5 | 18 to 20 | 8 + 2 | 154.5/370 = .418 | 17 - 34 | 3.292 | 4.082 | 76 | | 23.1 |
| 1964 | 5 | 20 | 8 + 2 | 226.5/376 = .603 | 34 | 7.595 | 9.418 | 163 | | 21.5 |
| 1965 | 5 - 6 | 20 to 28 | 8 to 10 + 2 | 189.8/365 = .52 | 34 - 57 | 8.599 | 10.663 | 188 | | 22.3 |
| 1966 | 6 | 28 | 8 + 2 | 220.0/364 = .605 | 57 | 12.576 | 15.593 | 176 | | 14.0 |
| 1967 | 6 | 28 to 32 | 8 + 2 | 174.4/358.7 = .486 | 57 - 63 | 10.564 | 12.863 | 144 + 12 IIIs | | 14.3 |
| 1968 | 6 | 30 to 35 | 8 + 2 | 213.9/356.1 = .601 | 58.5 - 67 | 13.168 | 15.960 | 183 + 24 IIIs | | 14.3 |
| 1969 | 6 | 35 to 38 | 8 + 1 or 2 | 229.3/371.3 = .618 | 66.5 - 71 | 15.155 | 18.228 | 165 + 12 IIIs | | 11.3 |
| 1970 | 6 | 38 | 8 + 1 | 218 /366.1 = .595 | 67 - 70 | 15.057 | 18.119 | 177 + 24 IIIs | | 12.5 |
| 1971 | 6 - 7 | 38 | 6 to 8 + 1 | 203.9/369.4 = .552 | 70 | 14.478 | 17.415 | 58 + 18 IIIs | 114 + 6 IIIs | 12.3 |
| 1972 | 7 | 38 | 6 + 1 | 230.4/355.9 = .647 | 70 - 73.5 | 17.568 | 21.132 | 11 | 142 + 18 IIIs | 9.0 |
| 1973 | 7 | 38 | 6 + 1 or 2 | 248.7/378.3 = .657 | 70 - 73.5 | 20.643 | 24.830 | 12 | 118 + 30 IIIs | 6.7 |
| 1974 | 7 - 8 | 32 to 38 | 6 or 7 + 1 or 2 | 217.3/351.9 = .618 | 70 - 80 | 16.320 | 19.630 | 1 | 132 + 6 IIIs + 2 XIIIs | 9.0 |
| 1975 | 8 | 32 to 34 | 7 + 1 | 218.8/364.5 = .600 | 74.5 - 80.5 | 16.487 | 19.873 | | 86 + 6 IIIs + 2 XIIIs | 6.0 |
| 1976 | 8 | 31 to 37 | 7 + 1 | 235.8/370.2 = .637 | 74.5 - 79.5 | 17.373 | 20.896 | | 80 + 18 IIIs + 1 XIIIs | 5.3 |
| 1977 | 8 - 9 | 34 to 38 | 7 + 1 | 226.3/397.8 = .569 | 77 - 100 | 19.583 | 23.553 | | 115 + 1 VIs + 2 Xs | 6.5 |

Fig. 9

Correlation between the enrichment of the fuel at the end of an operation cycle and the antireactivity of the experimental load (Configuration 8M and 80)



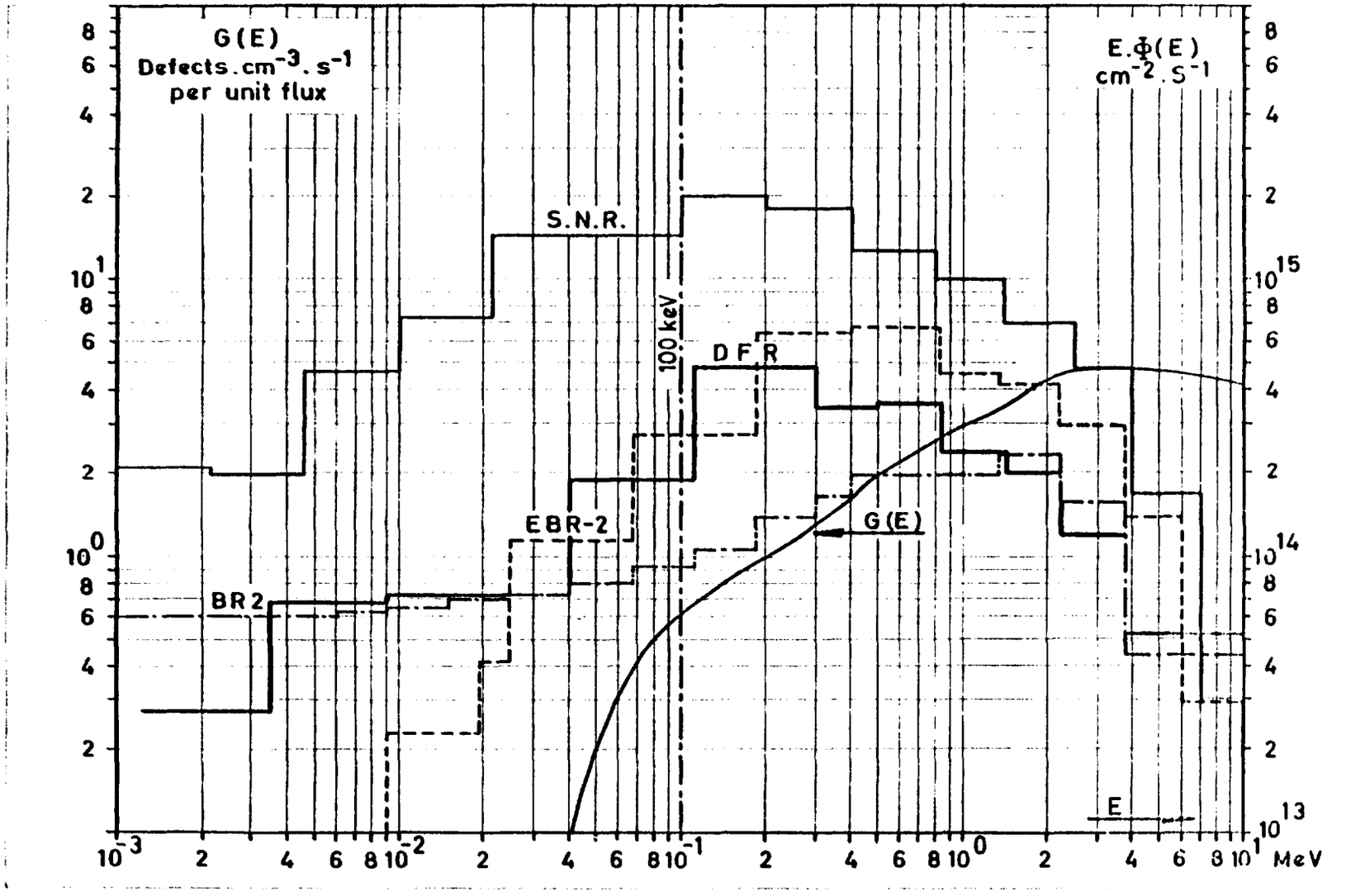



Fig. 10 Neutron spectra in various reactors and steel damage function.

Fig. 11

Table of the microscopic radial distribution of the thermal neutron flux in standard cell at BR2

| Fuel element $^{235}U$ content | A VIn 0 | A VIn 61 g | A VIn 122 g | A VIn 183 g | A VIn 244 g standard | A VIn 300 g | A VIn 400 g | A VIn 500 g | G VIn 400 g |
|---|---|---|---|---|---|---|---|---|---|
| $r\sqrt{\frac{T}{To}}$ (x) | .01268 | .02394 | .03000 | .03366 | .03626 | .3786 | .04021 | .04175 | .04218 |
| $\phi_{epi} = \frac{\phi_{th(r\to\infty)}^{0.5} \frac{2}{\sqrt{\pi}} r\sqrt{\frac{T}{To}}}{1-\frac{1}{2}x\, r\sqrt{\frac{T}{To}}}$ | .0144 | .0273 | .0344 | .0386 | .0417 | .0437 | .0469 | .0481 | .0486 |
| $J_-$ ($r_o = 5.063$) (xx) | .1360 | .1984 | .1320 | .1173 | .1071 | .1003 | .0912 | .0850 | .0833 |
| $\phi(r = o)$ | .6279 | .4345 | .3380 | .2790 | .2400 | .2136 | .1805 | .1596 | .1537 |
| $\phi(r_1 = 1.662)$ | .6494 | .4333 | .3261 | .2609 | .2182 | .1892 | .1532 | .1302 | .1238 |
| $\phi(r_2 = 2.089)$ | .6645 | .4399 | .3279 | .2597 | .2147 | .1844 | .1465 | .1220 | .1153 |
| $\phi(r_3 = 2.516)$ | .6800 | .4521 | .3377 | .2673 | .2207 | .1891 | .1495 | .1237 | .1166 |
| $\phi(r_4 = 2.943)$ | .6955 | .4706 | .3550 | .2836 | .2356 | .2028 | .1614 | .1338 | .1261 |
| $\phi(r_5 = 3.370$ | .7150 | .4955 | .3815 | .3102 | .2616 | .2282 | .1855 | .1561 | .1479 |
| $\phi(r_6 = 3.797)$ | .7371 | .5395 | .4212 | .3525 | .3053 | .2728 | .2311 | .2061 | .1936 |
| $\phi_u$ | .6987 | .4794 | .3669 | .2975 | .2505 | .2185 | .1780 | .1508 | .1432 |
| $\phi_u / \phi_{H_2O}$ | .992 | .967 | .953 | .936 | .910 | .906 | .881 | .868 | .857 |
| $f = \frac{\int \Sigma f\, \phi\, dv}{\int \Sigma a\, \phi\, dv}$ | 0 | .493 | .603 | 650 | .678 | .693 | .710 | .723 | .544 |
| $J_{Net} = \frac{1}{2\pi r_o} \int \Sigma a\, \phi\, dv$ | .0192 | .0365 | .0457 | .0515 | .0555 | .0582 | .0617 | .0641 | .06475 |
| $J_{Net}$ (if $r\sqrt{\frac{T}{To}} = o$) | .0155 | .0259 | .0306 | .335 | .353 | .0366 | .0382 | .0393 | .0396 |

(x) $r\sqrt{\frac{T}{To}}$ Westcott's spectrum index in an unperturbed medium.

(xx) $J_-$ Inward current of thermal neutrons;
supposition is taken, that the thermal neutron flux is unity for in beryllium moderator : $\phi_{th}^{0.5}(r\to\infty) = 1$

*Note that the Westcott's spectrum index is measured at BR2 with cobalt detectors* (100 % Co $\phi$ 3 x 1 mm) by means of the formula :

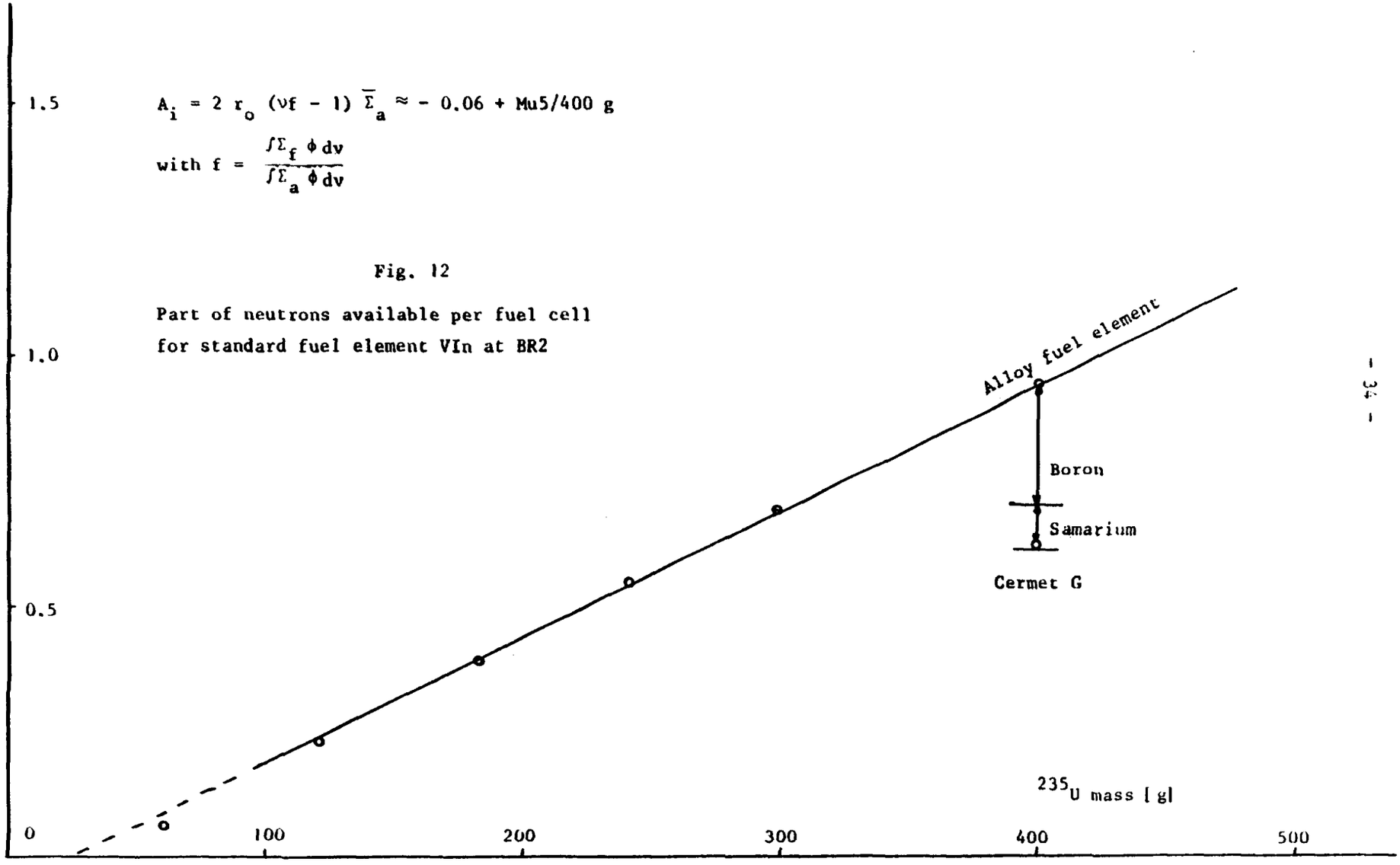
$$r\sqrt{\frac{T}{To}} = \frac{0.962}{1.269\, R_{cd} - 0.323}$$

$$A_i = 2\, r_o\, (\nu f - 1)\, \overline{\Sigma}_a \approx -0.06 + Mu5/400\ g$$

with $f = \dfrac{\int \Sigma_f\, \phi\, dv}{\int \Sigma_a\, \phi\, dv}$

Fig. 12

Part of neutrons available per fuel cell for standard fuel element VIn at BR2



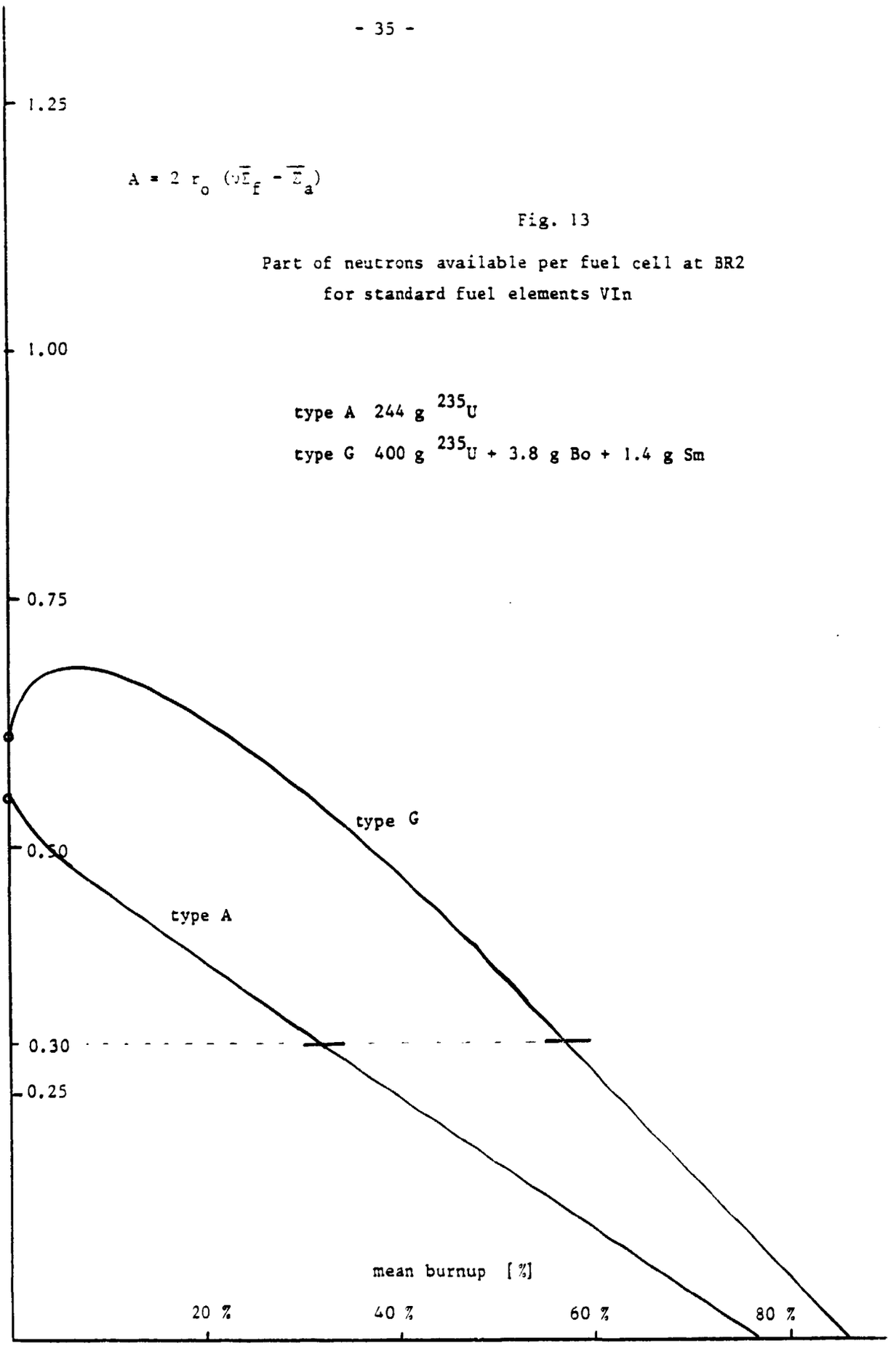
$$A = 2\, r_o\, (\nu\bar{\Sigma}_f - \bar{\Sigma}_a)$$

Fig. 13

Part of neutrons available per fuel cell at BR2
for standard fuel elements VIn

type A 244 g $^{235}U$

type G 400 g $^{235}U$ + 3.8 g Bo + 1.4 g Sm



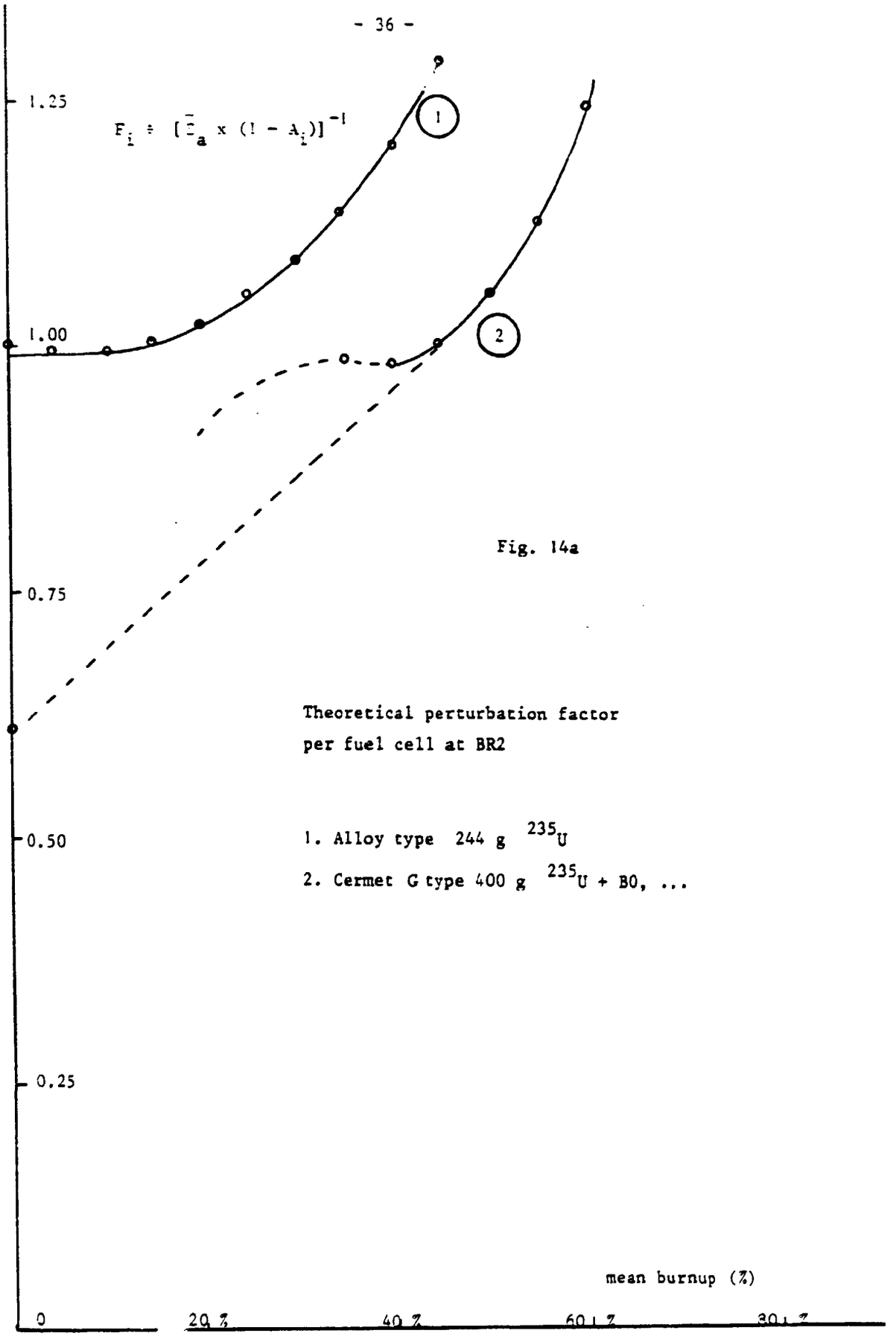



Fig. 14a

Theoretical perturbation factor
per fuel cell at BR2

1. Alloy type 244 g $^{235}U$
2. Cermet G type 400 g $^{235}U$ + B0, ...

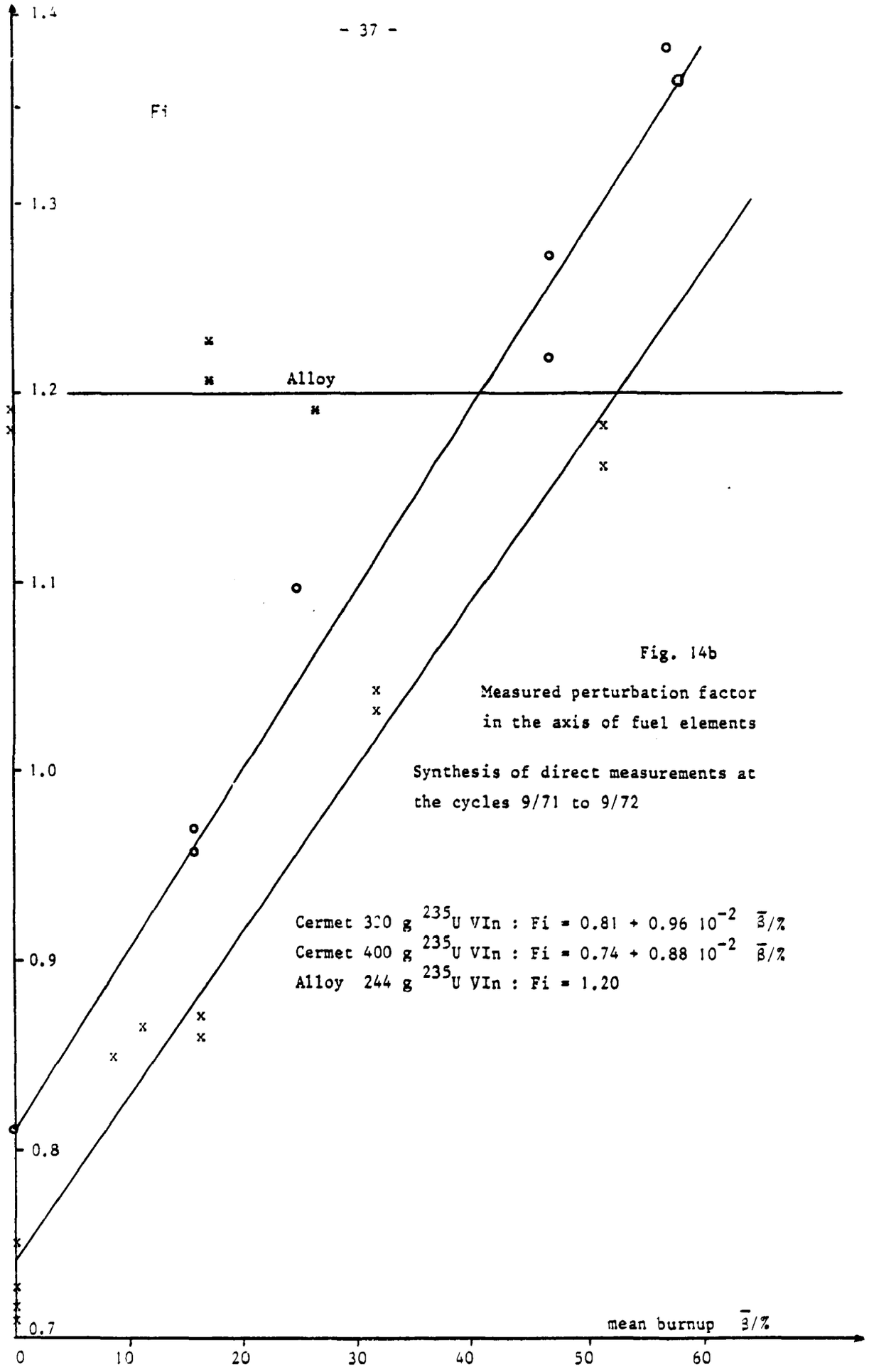



Fig. 14b

Measured perturbation factor in the axis of fuel elements

Synthesis of direct measurements at the cycles 9/71 to 9/72

Cermet 300 g $^{235}U$ VIn : $Fi = 0.81 + 0.96\ 10^{-2}\ \bar{\beta}/\%$

Cermet 400 g $^{235}U$ VIn : $Fi = 0.74 + 0.88\ 10^{-2}\ \bar{\beta}/\%$

Alloy 244 g $^{235}U$ VIn : $Fi = 1.20$

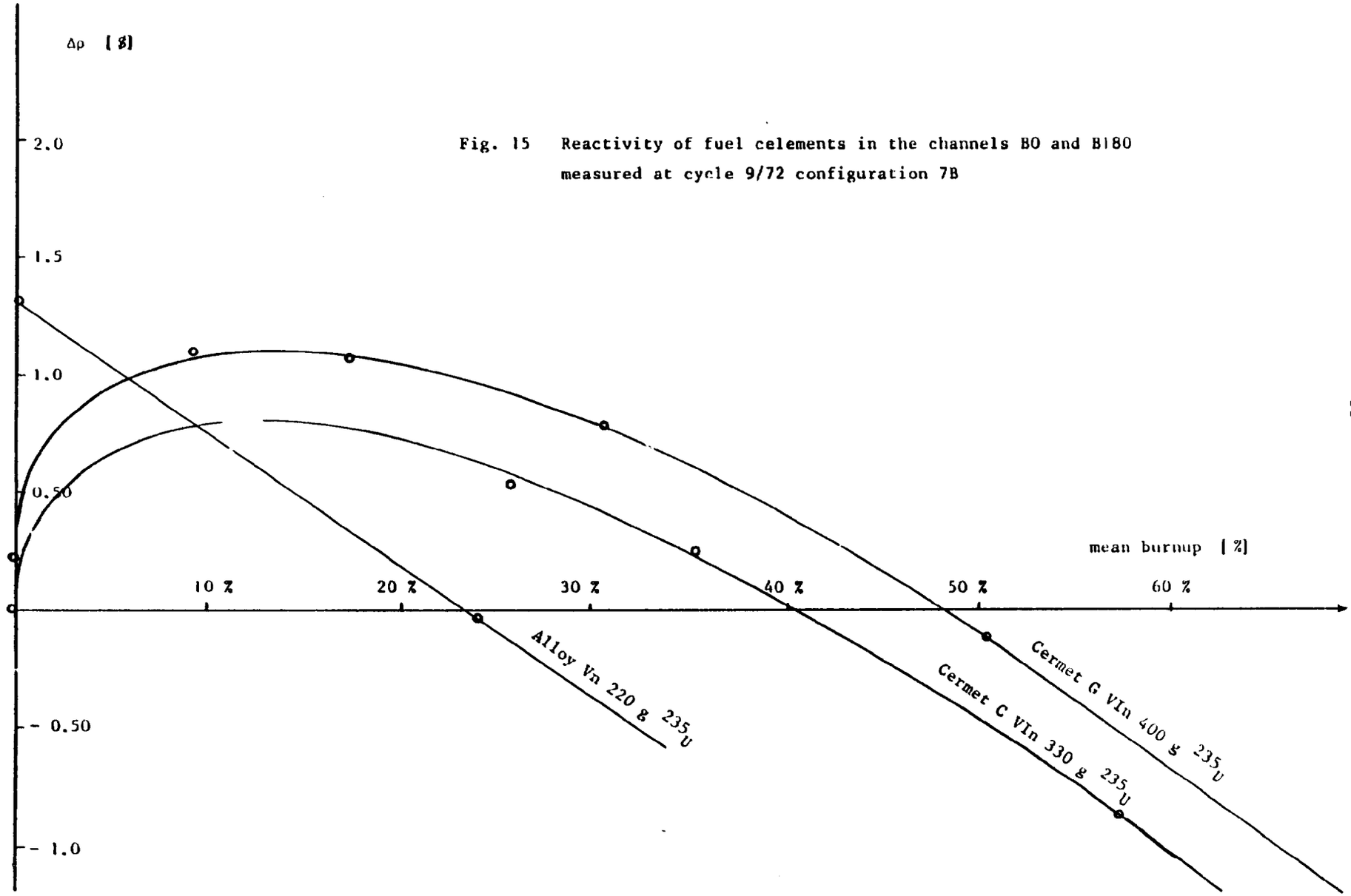



Fig. 15 Reactivity of fuel celements in the channels B0 and B180 measured at cycle 9/72 configuration 7B

Table 16

Radial distribution of the adjoint of the thermal neutron flux in core configurations at BR2, where $A = 2 r_0 (\nu \overline{\Sigma}_f - \overline{\Sigma}_a)$

$P_{ii} = 0.945$ & $A_i = 1.3 - 0.8 = 0.50$ for fuel cells
$P_{ii} = 0.900$ & $A_i =$ - 0.043 for reflector cells

| Configurations | 0 | AC | ACD | ACDE | ACDEF' | AB | ABC' | ABC'D | ABC'DE | ABC'DEF' |
|---|---|---|---|---|---|---|---|---|---|---|
| No. of fuel cells | 6 | 18 | 24 | 30 | 36 | 12 | 18 | 24 | 30 | 36 |
| $k_{eff}$ (A = 0.5) | 1.2085 | 1.2782 | 1.3217 | 1.3489 | 1.3723 | 1.2478 | 1.3037 | 1.3408 | 1.3641 | 1.3859 |
| $\delta$A 6 rods | 0.1246 | 0.1252 | 0.1062 | 0.0950 | 0.0748 | 0.0671 | 0.0630 | 0.0626 | 0.0652 | 0.0669 |
| $A_{min}$ (k = 1) | 0.2838 | 0.2293 | 0.2004 | 0.1837 | 0.1705 | 0.2519 | 0.2127 | 0.1891 | 0.1748 | 0.1621 |
| Channels | | | | | | | | | | |
| H 1 | 0.864 | 0.546 | 0.491 | 0.459 | 0.414 | 0.588 | 0.565 | 0.549 | 0.565 | 0.551 |
| H 1_ | 1 | 0.669 | 0.623 | 0.596 | 0.550 | 0.709 | 0.712 | 0.712 | 0.742 | 0.734 |
| A | 0.421x | 1.113 | 1.083 | 1.074 | 1.021 | 1.031 | 1.114 | 1.147 | 1.220 | 1.230 |
| F | 0.320 | 0.733x | 0.771x | 0.778x | 0.759x | 0.969 | 1.066 | 1.157 | 1.240 | 1.268 |
| C'' | 0.157 | 0.943 | 1.026 | 1.116 | 1.177 | 0.437x | 0.523x | 0.605x | 0.700x | 0.776x |
| C' | 0.157 | 0.943 | 1.026 | 1.116 | 1.115 | 0.437 | 0.820 | 0.936 | 1.075 | 1.135 |
| D | 0.109 | 0.473 | 0.865 | 0.912 | 0.989 | 0.318 | 0.402 | 0.760 | 0.854 | 0.966 |
| E | 0.066 | 0.387 | 0.420 | 0.783 | 0.874 | 0.185 | 0.263 | 0.307 | 0.611 | 0.729 |
| F' | 0.057 | 0.308 | 0.383 | 0.459 | 0.825 | 0.163 | 0.195 | 0.271 | 0.350 | 0.673 |
| F'' | 0.057 | 0.186 | 0.383 | 0.459 | 0.484 | 0.163 | 0.240 | 0.319 | 0.404 | 0.453 |
| g | 0.038 | 0.141 | 0.262 | 0.289 | 0.359 | 0.110 | 0.135 | 0.209 | 0.248 | 0.324 |

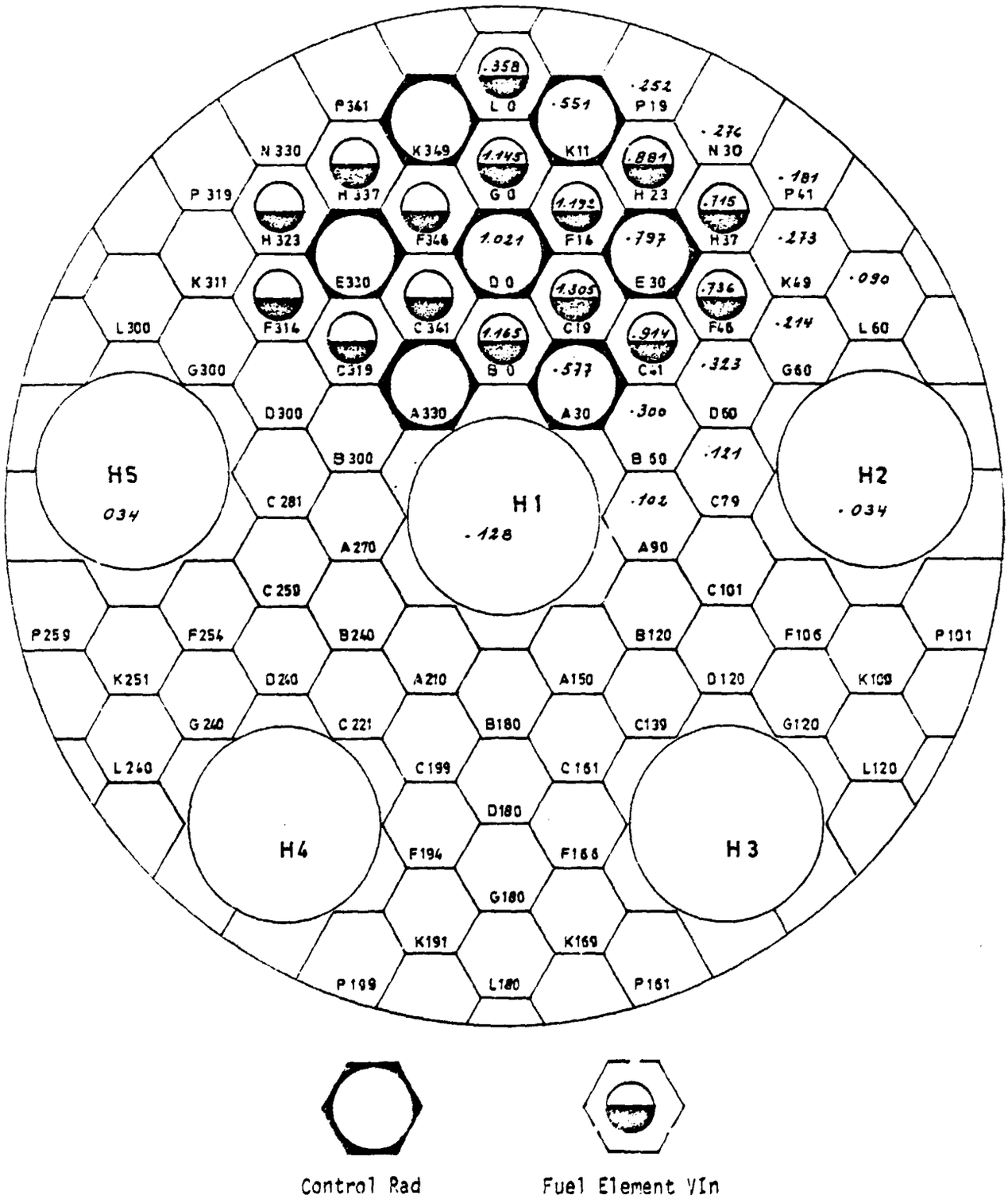
| *Configurations* | 0 | 1 | AB | 4 | 7B | 8 |
|---|---|---|---|---|---|---|
| No. of fuel cells | 6 | 7 | 12 | 14 | 36 | 40 |
| $k_{eff}$ (A = 0.5) | 1.2085 | 1.2585 | 1.2478 | 1.2925 | 1.3621 | 1.3801 |
| $\delta$A 6 rods | 0.1246 | 0.0764 | 0.0671 | 0.1282 | 0.0653 | 0.0548 |
| $A_{min}$ (k = 1) | 0.2838 | 0.2470 | 0.2519 | 0.2210 | 0.1761 | 0.1658 |
| $k_{eff}$ BR02 | | | | 1.1037 | | |

x contains a shim-safety rod
C' channels : C 41, C 101, C 161, C 221, C 281, C 341
C'' channels : C 19, C 79, C 139, C 199, C 259, C 319
F' channels : F 14, etc.
F'' channels : F 46, etc.

Fig. 17a

Core Configuration 4

Map of the unperturbed adjoint of the thermal neutron flux ($A_i$ = 0.50)



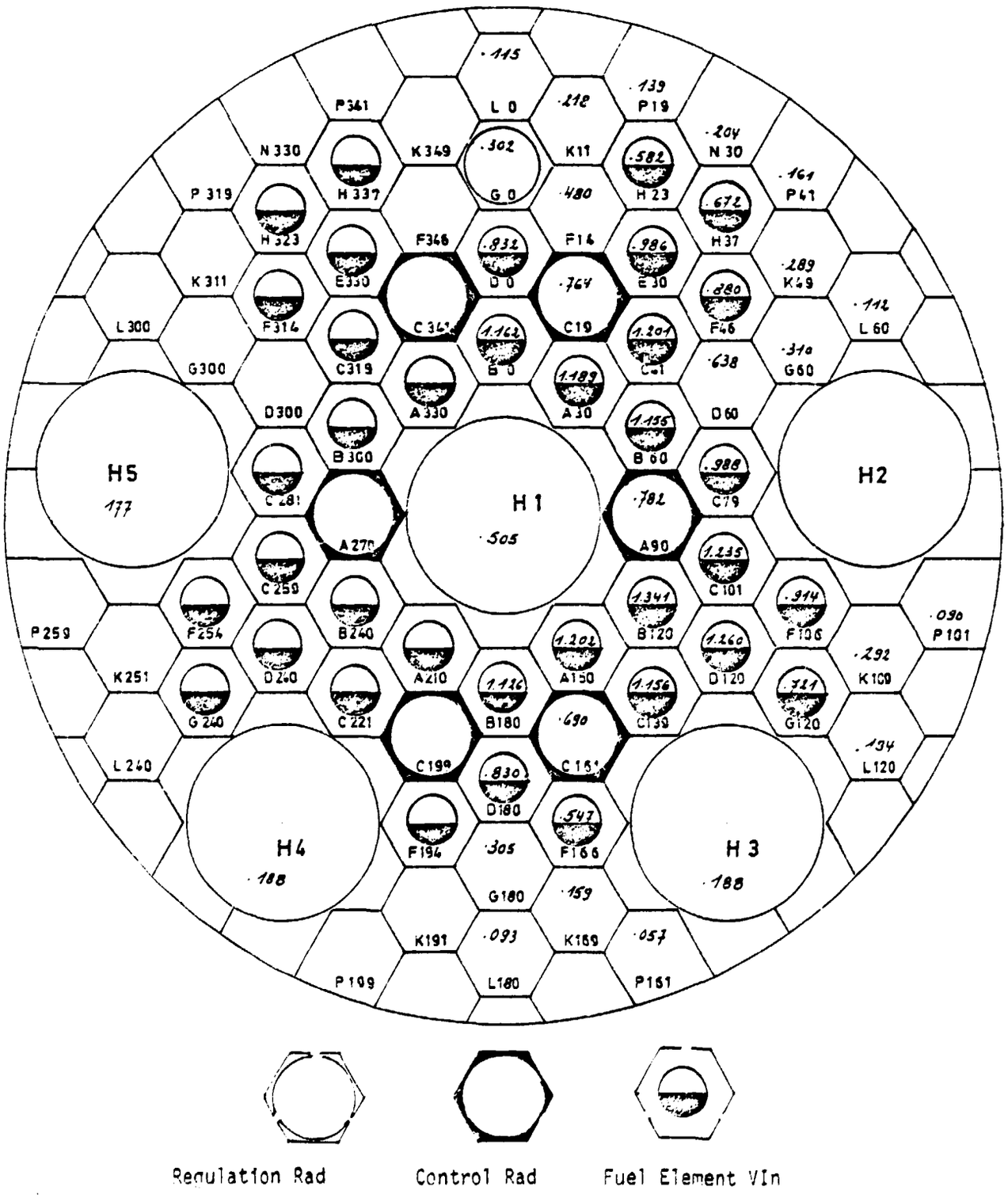

Control Rad

Fuel Element VIn

Fig. 17b

Core Configuration 78

Map of the unperturbed adjoint of the thermal neutron flux ($A_1$ = 0.50)




Regulation Rad  Control Rad  Fuel Element VIn

Fig. 17c

Core Configuration 8

Map of the unperturbed adjoint of the thermal neutron flux ($A_1$ = 0.50)

In channel H 3 : 3 equivalent standard fuel elements