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THE SUBCRITICAL FACILITY PUK

by

M. BUSTRAAN and A. TAS
(Reactor Centrum Nederland)

1964

Report prepared by the Reactor Centrum Nederland
Institute for the Development of Peaceful Applications of Nuclear Sciences

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Paper presented at the IAEA - Symposium on «Exponential and Critical Experiments»
Amsterdam, 2 - 6 September, 1963

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diameter 12 mm and inner diameter 10 mm, with a lattice spacing of 15.1 mm, giving a H_2O/UO_2 volume ratio of 1.07. The effective multiplication factor will be lower than 0.95.

It will be possible to split the core into two halves separated by a vertical watergap, the size of which may be varied from 2 mm to 38 mm.

The facility will be placed against the outer reflector of the Argonaut type reactor, L. F. R., at the reactor center at Petten. To obtain a cylindrical symmetric neutron flux the core will be rotated around its vertical axis.

The facility will be used

— to measure intracell flux distributions and lattice parameters,
— to measure the effects on power distribution of watergaps in a lattice and to study methods for preventing flux peaking,
— to supply general experimental data for checking calculational techniques.

Techniques for measurements of intracell flux distributions in tightly packed lattices and of power distributions near watergaps are described.

Attention is given to the calculation of spatial dependent thermal neutron spectra by a modified THERMØS program and to the calculation of power distributions near watergaps.

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THE SUBCRITICAL FACILITY PUK

1. Introduction

In addition to the critical facility KRITO [1], the subcritical facility PUK has been designed in order to perform measurements which are not, or not easily possible in KRITO.
The purpose for building these facilities is the development of a ship propulsion reactor of the PWR type, which is a joint undertaking of Euratom and RCN, known as the NERO project.

The subcritical facility has been designed and is under construction now. The first measurements will be done at the end of 1963.

The subcritical facility will principally be used to measure

- intracell thermal neutron flux distributions and clean lattice parameters,
- the effects on power distribution of watergaps in a lattice and to study methods to prevent power peaking, e.g. with nuclear poisons in the fuel,
- flux distributions near the boundary of regions with different ^{235}U contents in the fuel,
- the effect on power distribution of slight deviations of the fuel rod sizes and positions,
- and to supply experimental data for checking calculational techniques.

The subcritical facility is not suitable for determination of clean lattice bucklings.

2. Description and characteristics of the PUK facility

2.1. Short description of the facility

The subcritical facility PUK is situated on the shielding truck (fig. 1) of the Low Flux Reactor of Argonaut type, L.F.R., which will serve as neutron source.

The entire facility (fig. 2) is mounted on an aluminum frame, which can be lifted from the shielding truck. On this frame the core vessel, the graphite reflector of 30 cm thickness, the filling pump and the driving mechanism for rotating the fuel core are situated. The frame itself is the storage tank into which the water from the PUK core vessel will be dumped after each measurement.

Inside the core vessel a rotating table is mounted on which the fuel core can be rotated in the vessel. The fuel core construction as a whole can be lifted out of the core vessel. Up to now there are two core constructions.

2.2. General characteristics of the fuel core

The fuel core of the subcritical facility PUK consists of aluminum canned rods. The fuel is UO_2 of which the ^{235}U content in U is 3.8%, which is also used in the KRITO core. Some measurements, for instance of power peaking near a watergap and of flux distributions near the boundary of regions of two enrichments, will be performed with fuel of another enrichment in a part of the core.

/canned in aluminum tubes

The fuel rods/ with outer and inner diameter of 12 and 10 mm respectively, will be placed in a regular hexagonal lattice with a lattice spacing of 15.1 mm, giving a H_2O/UO_2 volume ratio of 1.07. This lattice is the same as in the hexagonal fuel assemblies of KRITO and NERO. The average H_2O/UO_2 volume ratio in NERO and KRITO is 1.42, the difference being caused by the watergaps around the hexagonal fuel assemblies. However, it is possible to get the KRITO core configuration with its average volume ratio by removing rods and inserting plates in the PUK fuel core.

2.3. Core construction and measuring facilities

There will be two different core constructions:

- One construction consists of a fuel core, which will especially serve to carry out the clean lattice studies.
- The other construction has the same lattice, but the bottom and the top grid are divided into two halves. The width of the vertical gap between the half fuel cores can be varied from 2 mm to 38 mm. Plates of variable thickness can be inserted into the gap. This construction allows the study of the influence of watergap size on power distribution in surrounding fuel rods.

In the following these two constructions will be called respectively the undivided fuel core and the divided fuel core.

In the case of the undivided core seven of the central fuel rods form a removable cluster in which the activation foils for flux density measurements can be inserted. The core, consisting of two halves, has two such measuring clusters, both consisting of 5 fuel rods. The clusters are fabricated with the utmost precision, in order to be able to perform very accurate measurements.

In some clusters well defined deviations in fuel rod positions are made in order to measure their effect on flux density peaking.

2.4. Core size and reactivity effects of perturbations

The choice of the core size is determined by the multiplication factor and the neutron spectrum.
For safety reasons the multiplication factor should be lower than 0.95. This sets an upper limit for the core size. On the other hand the core dimensions should be large enough to obtain at the centre a thermal neutron spectrum, which is a good representation of that in a larger, critical system.

In a fuel core the neutron spectrum is dependent on the magnitude of the leakage and thus on the core size. For increasing core size this dependence, which is initially strong, becomes gradually weaker. In the centre of the PUK core there will be no great influence of leakage on the thermal spectrum, but possibly there is an effect on the fast and epithermal spectrum, due to the fact that the transport mean free path for fast neutrons is of the order of the dimensions of the core, while it is much smaller for thermal neutrons. A deviation in the epithermal and fast neutron spectrum has no direct effect on the thermal spectrum in the case of isotropic scattering in the centre of mass system, because then the thermal source neutrons have a velocity distribution independent of the velocity distribution of the epithermal neutrons.

Therefore a deviation in the epithermal spectrum and a not complete isotropic scattering probably will have no measurable effect on the thermal neutron spectrum in the centre of the core.

We may therefore conclude that micro distributions of thermal flux density and power density in a subcritical core give a reliable picture of the same distributions for a critical core. However, the ratio between fast and thermal fission density and the ratio between epithermal and thermal fission density is possibly somewhat different for critical and subcritical cores.

The core with the clean lattice is a cylinder with equal height and diameter in order to get a small fast leakage, thus a high effective multiplication factor with a quantity of fuel which is as small as possible. Calculations show that for $k_{eff} = 0.95$ and for a ^{235}U content in U equal to 3.8%, the height of the core will be about 48 cm and the diameter 50 cm (fig. 7).

In KRITO reactivity effects have been measured for the following perturbations in the lattice:

- removing cluster of 7 rods	: + 240 pcm
- removing 1 row of rods to make a watergap of 18.1 mm	: + 760 pcm
- pulling out 2 rows of rods to make a gap of 33.3 mm	: + 390 pcm
- removing two clusters of 5 rods reduced the effect of the second measurement with	: - 180 pcm
and of the third measurement with	: - 420 pcm
- in one assembly a regular lattice of hexagons, giving a H_2O/UO_2 volume ratio of 1.92 (fig. 7)	: +1000 pcm

These measurements have been performed in a KRITO lattice, perturbed by fuel assemblies, and with a loading of 3.1% enriched UO_2. It provides, however, indications for safe handlings with the undermoderated fuel core of the PUK facility.

2.5. Neutron source

The position of the source is determined by the requirement that the thermal spectrum on the points of measurement is influenced as little as possible by the neutrons from the source. This requirement must be fulfilled for both the undivided and the divided core. Due to the effect of streaming of thermal neutrons from the source along the rods and along the watergap a source position below the oore was rejected.

A position of the source has been choosen such that the source neutrons are incident in radial direction from one side. For measurements of lattice parameters it is desirable that no radial flux gradient will occur in the central part, where the measuring cluster is located. For this reason the cylindrical core will be rotated around its vertical axis.

In the case of a core with a vertical watergap the orientation of the gap is perpendicular the stream of source neutrons. Halfway during the measurements the divided core will be turned 180° around its vertical axis.

When activating foils in a rotating core, we must take care to keep the ratation time small with regard to the half life of the product nuclides in the target material. Also we have to take an integer number of revolutions during activation.

In the case of radially incident neutrons it is not well possible to measure clean lattice bucklings. But this source positioning makes it possible to perform measurements of power distributions near vertical watergaps. The choice of this source position seems justified, since there are many publications about studies of bucklings in clean lattices and only a few about the influence on power peaking of irregularities in the clean lattice.

Preliminary measurements in the shielding facility of the LFR, a source of principally thermal neutrons, indicated that the maximum available thermal flux density in the PUK core is at least 10^8 cm^{-2} s^{-1}, which is sufficient for our measurements.

3. Measurements

3.1. Foilcasing and drilling of holes in UO_2 pellets

The clusters, which can be pulled out of the core, enable us to install foilholders between the fuel rods in the moderator.

Though it is possible to place foils between the fuel pellets by opening the fuel rod at the top, a foilcasing of choosable length is constructed which can easily be mounted between two pieces of a fuel rod (fig. 4b, 5). The advantage of this construction is that the quantity of active fuel, which must be uncanned is small. In some foilcasings a tube of cadmium of 0.45 mm thickness is placed. The sleeves over the foilcasing and fuel rod jam strongly, a force of about 12 kg is needed for dismantling. The caps are glued with araldite. The foilcasing is vacuum tight and it can resist a pressure of 50 ato. Opening of the foilcasing is easily possible with a special type tin-opener.

For measuring intracell fluxes in the fuel pellet, needles of a special alloy will be used, which are oriented parallel to the axis of the pellet, in order to avoid streaming effects. For making holes in the UO_2 pellet the following technique is used. In the green UO_2 pellet, pressed at 2800 kg cm^{-2} during 6 seconds, holes are drilled with high speed drills. A mould with changeable lids, in which are holes according to a wished diameter and pattern (fig. 4a), is used. After this the pellet is sintered. The sintered pellets show no cracks, even not in the case of six different sized holes. The inner walls of the holes are very smooth. During sintering crimp is 21%. The diameter of the holes is predictable within 2%.

In a green pellet of 20 mm height a hole of 0.80 mm diameter was drilled. After sintering, a needle of 0.60 mm fitted excellent. This proves roundness and straightness of the hole is satisfactory.

3.2. Quantities to be measured and the applied technique

The measurement of the disadvantage factor will be performed with a foil between the fuel pellets and a foil in its surrounding hexagonal moderator area.

The intracell flux density distribution will be measured by means of needles of 0.6 mm diameter and 5 mm length, inserted in the holes drilled in the UO_2 pellets placed in the foilcasing. The needles are made of a Dy-Al alloy with an absorption cross section equal to that of the fuel pellets in order to minimize the flux

disturbances. The needles in the fuel pellet are placed on equivalent positions (on radii which make an angle of 60° with each other). The radial distances are choosen such that each needle represents a ring with equal surface (fig. 4a). The needles in the moderator are positioned according the same pattern in a foilholder, made either of a thin plate of aluminum or of a thin plate of polyaethylene, which is tightened around the foil-casing.

The possibility of direct measurement of the intracell power density distribution by means of inserting fuel needles in the holes is investigated.

The same technique, but with another positioning of the needles in the fuel pellet (see also fig. 4a), will be used to measure:

- the flux/power density distributions across watergaps with and without nuclear poison in the fuel pellet,
- the influence on flux/power density distribution of a well defined deviation in the position and the diameter of the fuel rod,
- the flux/power distribution on the boundary of two regions with different enriched fuel

Further it is our intention to determine the ratio of thermal, epithermal and fast fission, by means of very flat foils of UO_2 3.8% enriched and of depleted uranium. The foils and the faces of the flat and smooth UO_2 pellets, which contact the foil, will be vacuum coated, in order to avoid the exchange of fission products. The activity of the fission products will be measured for both kind of foils after bare activation and of the 3.8% UO_2 foils after activation under cadmium.

The activations under cadmium will be performed in two manners:

- the first method is activation in a cadmium covered foi-casing (fig. 4b),
- the second method is activation of wires of 0.20 mm diameter in cadmium pipes of 0.20/1.00 mm. The last method will be used to measure space dependent cadmium ratios in fuel and in moderator.

3.3. Corrections for thermal neutron flux disturbances

Infinite cylindrical cell calculations have been performed in order to get an impression of the thermal neutron flux disturbance in the detection needle and its surrounding. In the performed calculations it is assumed that the energy averaged cross section of the wire is independent of its position. Also the influence of the presence of the wire on the local neutron spectrum has been neglected. This seems reasonable since the amount of absorbing detection material is small.

The thermal neutron flux depression calculated, is due to:

- the depression due to the thermal neutrons absorbed in the wire, which in the case of absence of the wire would return and contribute to the thermal neutron flux density on the location of the wire,
- the replacement of moderator by the Dy-Al wire. This replacement results in a thermal neutron flux depression in the wire, and its surrounding.

Not calculated is the flux depression due to absorption of epithermal neutrons, which would contribute to the thermal flux density if the wire were absent.

These calculations, concerning the influence on the thermal neutron flux density of an infinite long wire in an infinite long cylindrical unit cell of the PUK lattice, are made by the Carlson S2 method. The cross sections of this wire are the same as those of the fuel. The results of the calculations are tabulated in table I. The runs f 1 and 2 give the effect of inserting the wire in the centre of a fuel region and the runs m 1 to 8 the effect on thermal flux density at the location of the inserted Dy-Al detection wire is expressed in percents with respect to the average flux density in the wire. In column 1 the relative flux density disturbance is given. To calculate this relative disturbance the results of the calculations with and without the wire in the centre are normalized on the boundary of the cell.

To interpret the results for the actual needles of 5 mm length, it is assumed that the thermal flux density distribution is undisturbed to a certain distance from the position of the needle. In column 2 the disturbance due to inserting such a needle has been estimated by normalizing the flux densities, calculated with and without wire, on 1 mm from the outer edge of the wire. We see that the thermal flux density, measured with the mentioned needles, tends to rise in the fuel and to lower in the moderator. The depression of flux density in needles of 1.0 mm and 0.6 mm diameter is respectively about 2.7% and 1.6%, and in a smaller cell respectively about 4.1% and 2.5%. For the needle of 1.0 mm diameter situated in the fuel the increase in flux density is about 0.4%.

Thus it is worthwile to use needles of 0.6 mm or thinner in order to reduce the thermal neutron flux density disturbances.

Outer radii of cylindrical regions in cm						1 inf. wire normalized on cell boundary	2 estimated for finite wire norm. on 1 mm from wire boundary
						disturbance	
run	Dy-Al wire	gap	UO_2	Al	H_2O		
f 1	0	0	0.500	0.600	0.793	+ 0.4%	+ 0.3%
f 2	0.045	0.050	0.500	0.600	0.793		
	Dy-Al wire	H_2O	Al	UO_2			
m 1	0	0.518	0.615	0.793		- 3.5%	- 2.7%
m 2	0.050	0.518	0.615	0.793			
m 3	0	0.518	0.615	0.793		- 1.9%	- 1.6%
m 4	0.030	0.518	0.615	0.793			
m 5	0	0.300	0.447	0.670		- 4.9%	- 4.1%
m 6	0.050	0.300	0.447	0.670			
m 7	0	0.300	0.447	0.670		- 2.8%	- 2.5%
m 8	0.030	0.300	0.447	0.670			

Table I. Thermal neutron flux density disturbances due to a detection wire in cylindrical cells. (The thermal neutron flux density disturbances at the location of the inserted Dy-Al detection wire are given in % with respect to the average flux density in the wire.)

4. Calculation methods for the comparison of measurements

4.1. Power distributions near watergaps

The measured power distribution perpendicular to a watergap will be compared with the results of the following one-dimensional calculation methods:

- the one-dimensional multigroup diffusion theory program "Wanda". As the thickness of the metal plates and the width of the watergap is in most cases small compared to the mean free path, this method is rather doubtful.
- the same "Wanda" program in which fictive constants are introduced in such a way that the thermal flux density is calculated according to the P3 method and the fast flux density according to the diffusion method.
- the Carson Sn method. For calculations according this method the actual lattice with the single watergap is replaced by a series of unit cells, each consisting of a number of slabs. In this model the distance between two adjacent watergaps will be chosen so large that there is no mutual influence. It has been measured in KRITO that the power peaking occurs mainly in a 3 cm region, containing two rows of fuelrods, near the watergap. The fact that the epithermal neutron flux density distribution is not flat is accounted for by multiplying the source term in each region by the average epithermal flux density, obtained by the "Wanda" program.

In the above mentioned methods energy and cell averaged cross sections are calculated with the aid of the Sofocate and the Sn program. Later on the program described in the next paragraph will be used too.

4.2. Space and energy-dependent thermal neutron flux

4.2.1. General

The knowledge of the space and energy dependent thermal flux density is not only important for calculation of thermal energy and cell averaged cross sections, for calculation of the power distribution in the fuel pellet and thermal flux distribution in a unit cell, but also important for the correlation of measurements in the cell with foils of 1/v, non 1/v absorbing and fissionable material.

A program comparable with the THERMØS program of Honeck has been written for the digital computer ELECTROLOGICA X-1, with a memory capacity of 8000 words. A new two-dimensional (2D) geometrical model to calculate the transport kernel has been evaluated (see 4.2.2.).

The well known integral equation [2], [3] is discretized and solved by the so-called normalized overrelaxation iteration process. The program for our computer is divided into three parts. The first part calculates the scattering matrix and source terms, the second part calculates the transport kernels and the last part performs the iteration. The three

parts are independent so they can be replaced by other versions. For instance the two-dimensional geometrical model by a one-dimensional one, the scattering matrix program by experimental determined scattering cross sections. With the available version a problem of up to 5 energy groups and 10 space points can be calculated. This limit is due to the ALGOL version of the iteration and scattering kernel program. When the machine language version will be ready for use a problem of 10 to 20 energy groups and 20 to 10 space points can be calculated.

4.2.2. The two-dimensional geometrical model for calculation of the transportkernels

For the calculation of the transportkernel the smallest symmetric element of a lattice is used (fig. 6). The neutron tracks in it are just the same as in the lattice if the boundaries reflect neutrons perfectly. The symmetric element is subdivided in N regions. In each region the neutron density is concentrated in one or more source points. If in the calculation of the transportkernel all the regions are taken to be cylindrical shells a one-dimensional calculation is performed.
A two-dimensional calculation is performed if the distribution of the regions is taken to be two-dimensional.

It is assumed that the neutrons from the source points start isotropically in a discrete number (e.g. 36) of directions in the horizontal plane. In each direction the horizontal tracks in each region between two reflections against the boundaries of the smallest symmetric element are calculated with a set of analytical formulae. Suppose the calculated length of the horizontal track in respectively the first and second region passed by the neutron is respectively x_1 and x_2. If the total cross section in the first region is Σ_1 and in the second region Σ_2, the collison probability in region 1 and 2 is respectively:

$$p_1 = 1 - \Delta\phi . 2 \int_0^{\pi/2} \sin\theta \ \exp\left(-x_1 \Sigma_1 / \sin\theta\right) d\theta$$

$$p_2 = 1 - \Delta\phi . 2 \int_0^{\pi/2} \sin\theta \ \exp\left\{-(x_1 \Sigma_1 + x_2 \Sigma_2) / \sin\theta\right\} d\theta - p_1$$

ϕ : angle with horizontal axis

θ : angle with vertical axis

The integrals are calculated by linear interpolation in a 500 points table for $\sum_i x_i \Sigma_i$ from 0 to 7. The running time of the 2D transportkernel program would be less than 5 min on a IBM 7090 computer, in the case of 10 energy groups and 15 space points.

In the program the angles of the triangular symmetric elements are variable, except the right angle. So the program can be used for several regular lattices, e.g. hexagonal, rectangular and many others, see fig. 6. Calculations and measurements will be compared for a regular hexagonal lattice and also for hexagonal cells in which the central rod has been removed (fig. 6d).

References

[1] SM-42/58 Critical Facility KRITO Petten

[2] Honeck, H.C., Nuclear Science and Engineering: 8, (1960)

[3] Takahashi, H., Nuclear Science and Engineering: 5, (1959)

Acknowledgement

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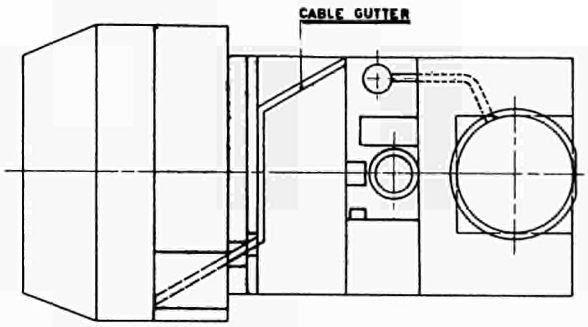


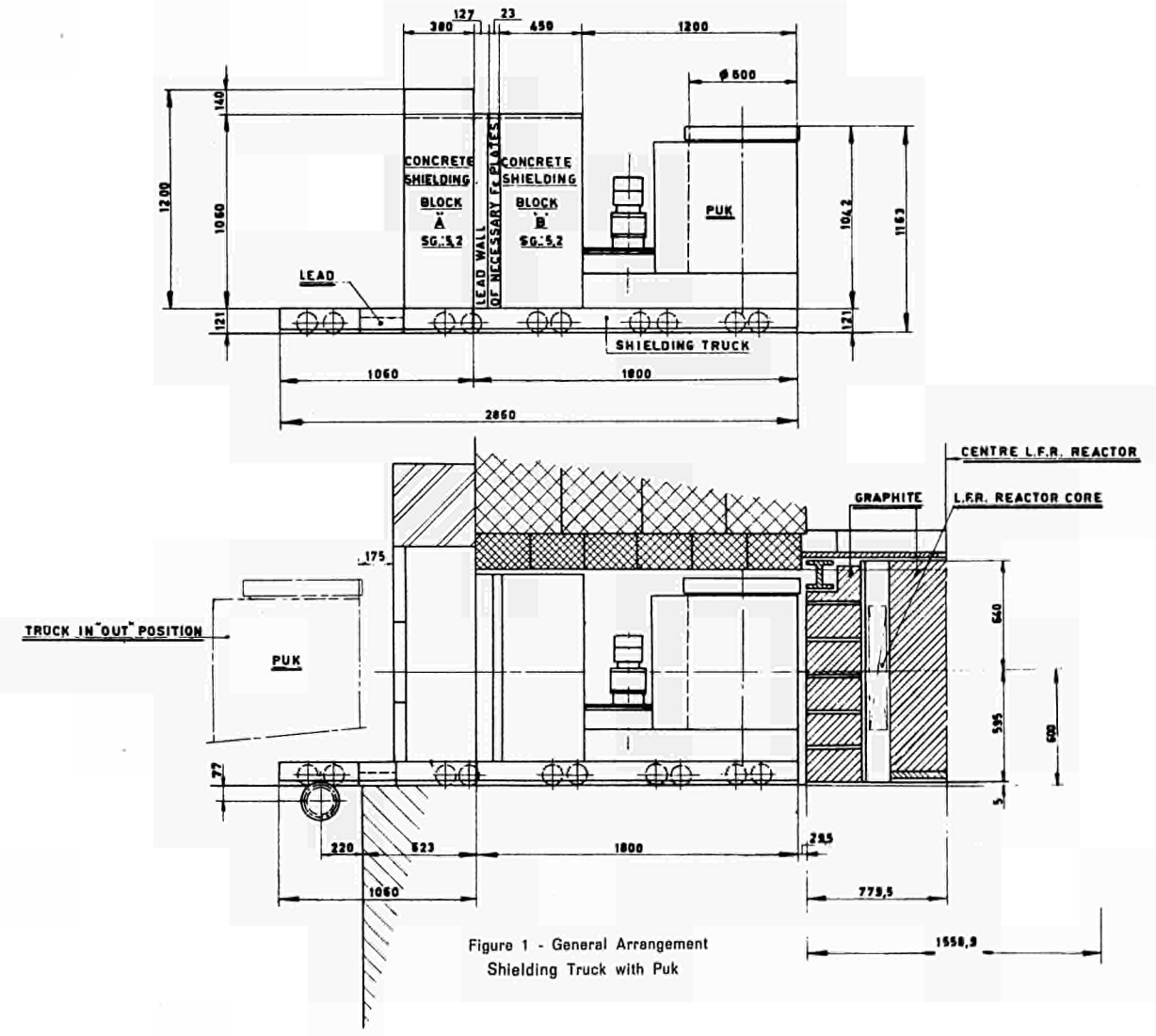
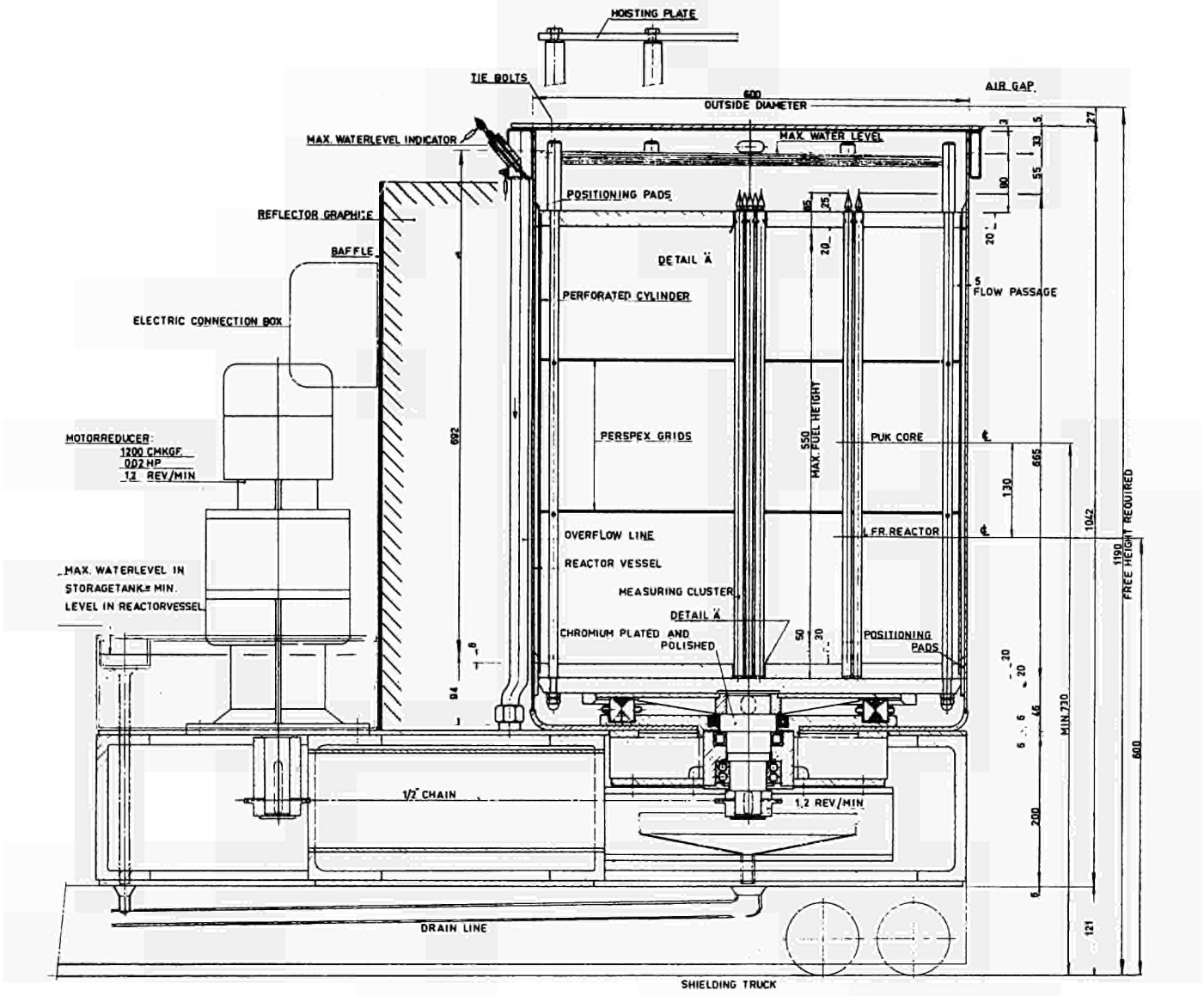
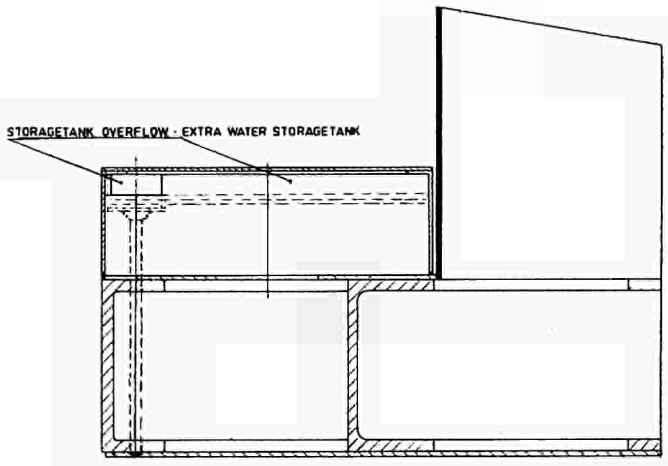
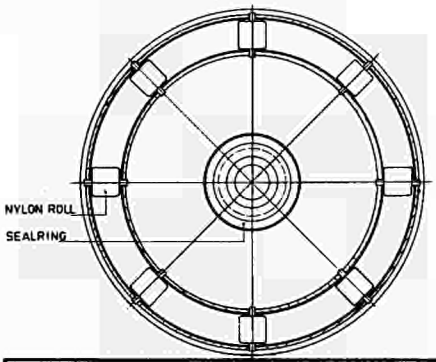


Figure 1 - General Arrangement
Shielding Truck with Puk

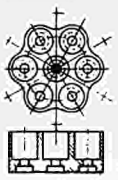




TOP VIEW ROLLER BEARING



DETAIL A



GENERAL ARRANGEMENT PUK	RCN
PUK FIG. 2	Dat 4-12-'62
REACTOR CENTRUM NEDERLAND	
TEK Nº 037-402-Te004	WIJZ. 0 DAT. 4-12-'62

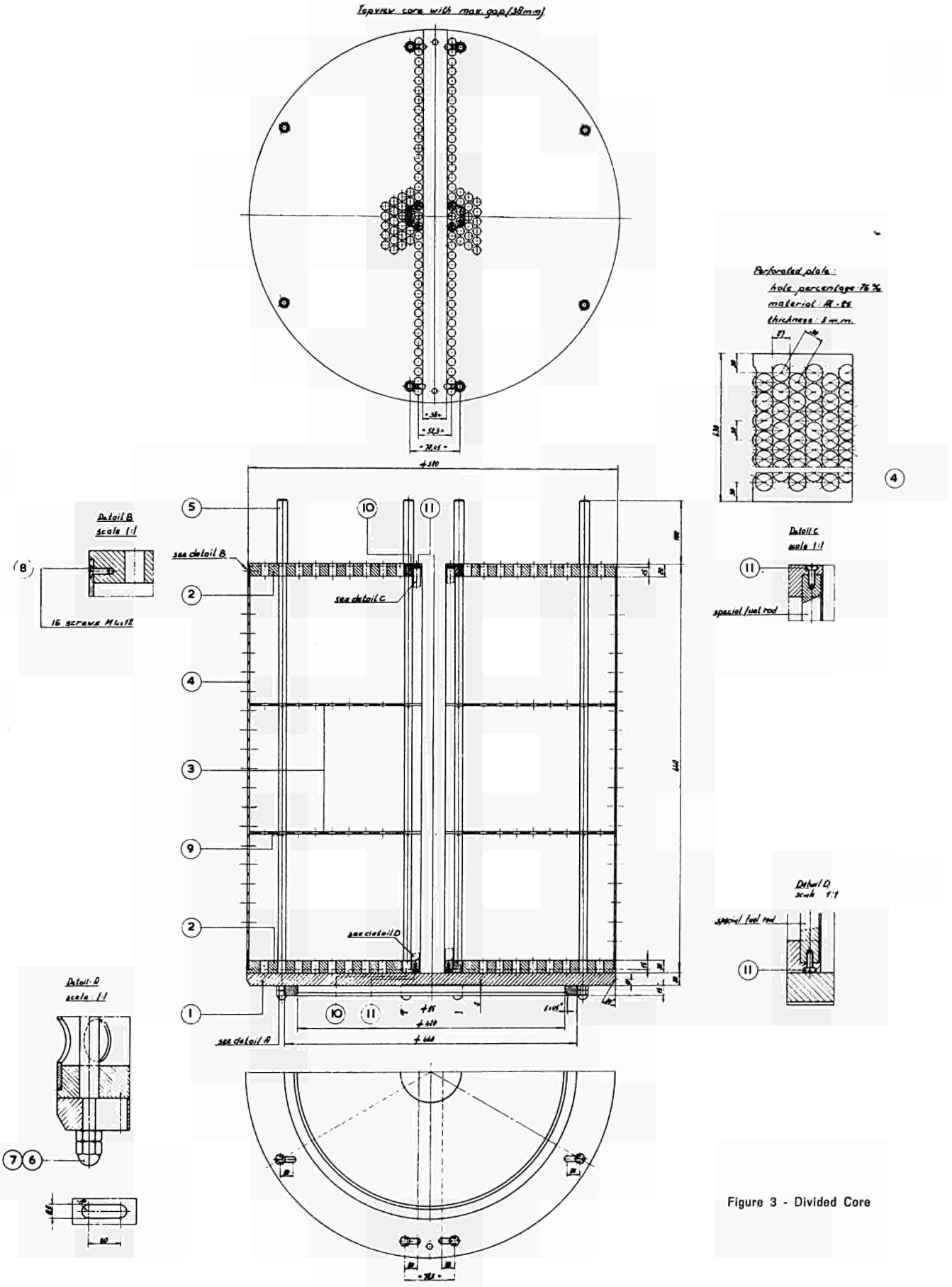
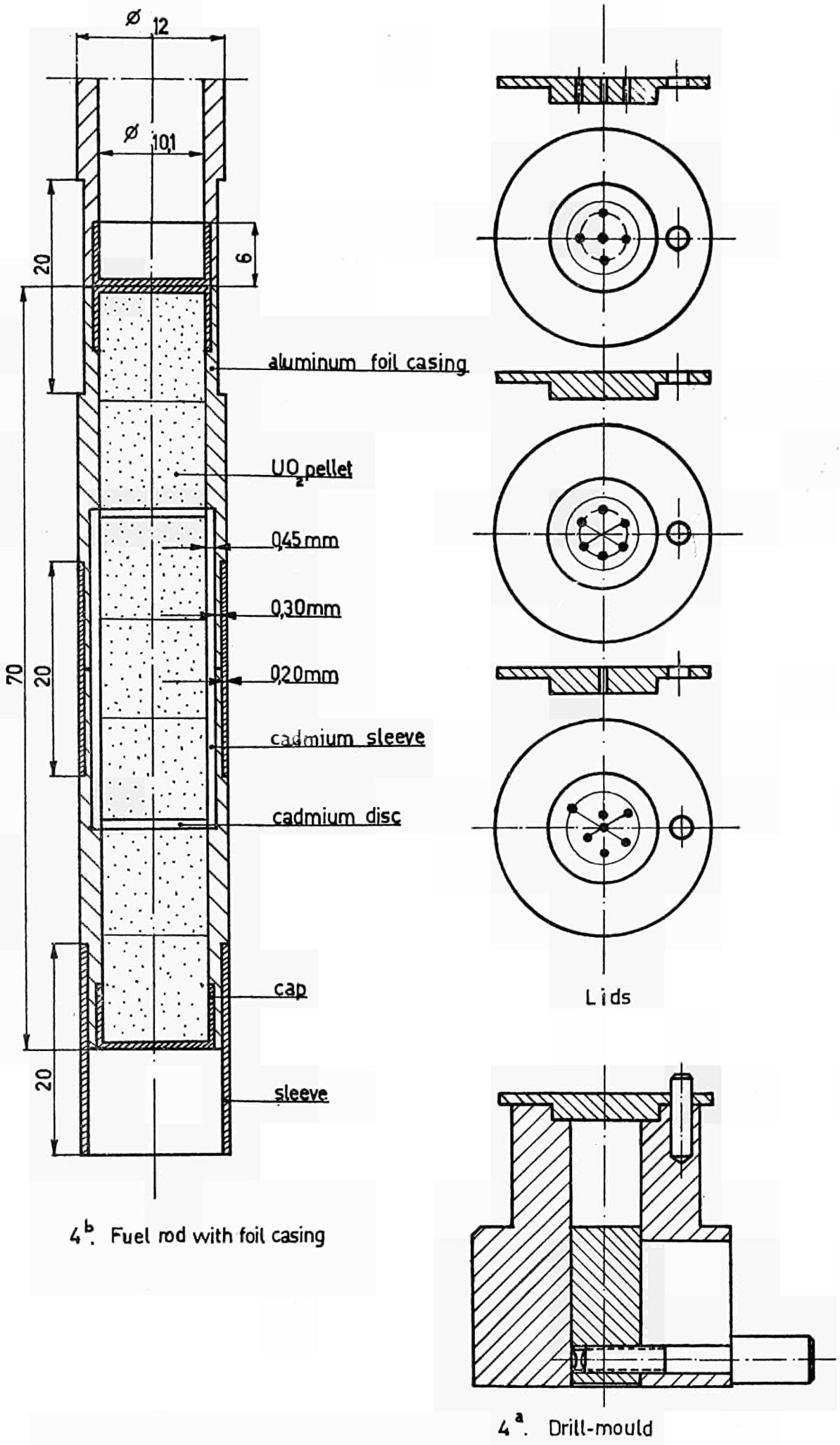


Figure 3 - Divided Core



4[b]. Fuel rod with foil casing

4[a]. Drill-mould

Figure 4. Divided fuel rod and drill-mould for UO_2 pellets

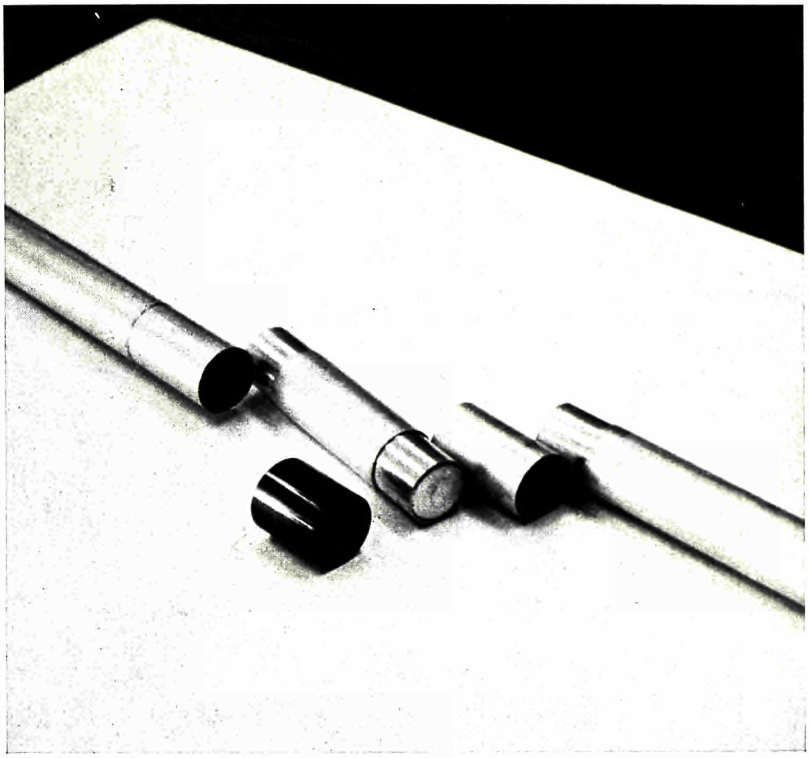


Figure 5. - Foil casing in fuel rod and
UO_2 pellet with different sized detection holes.

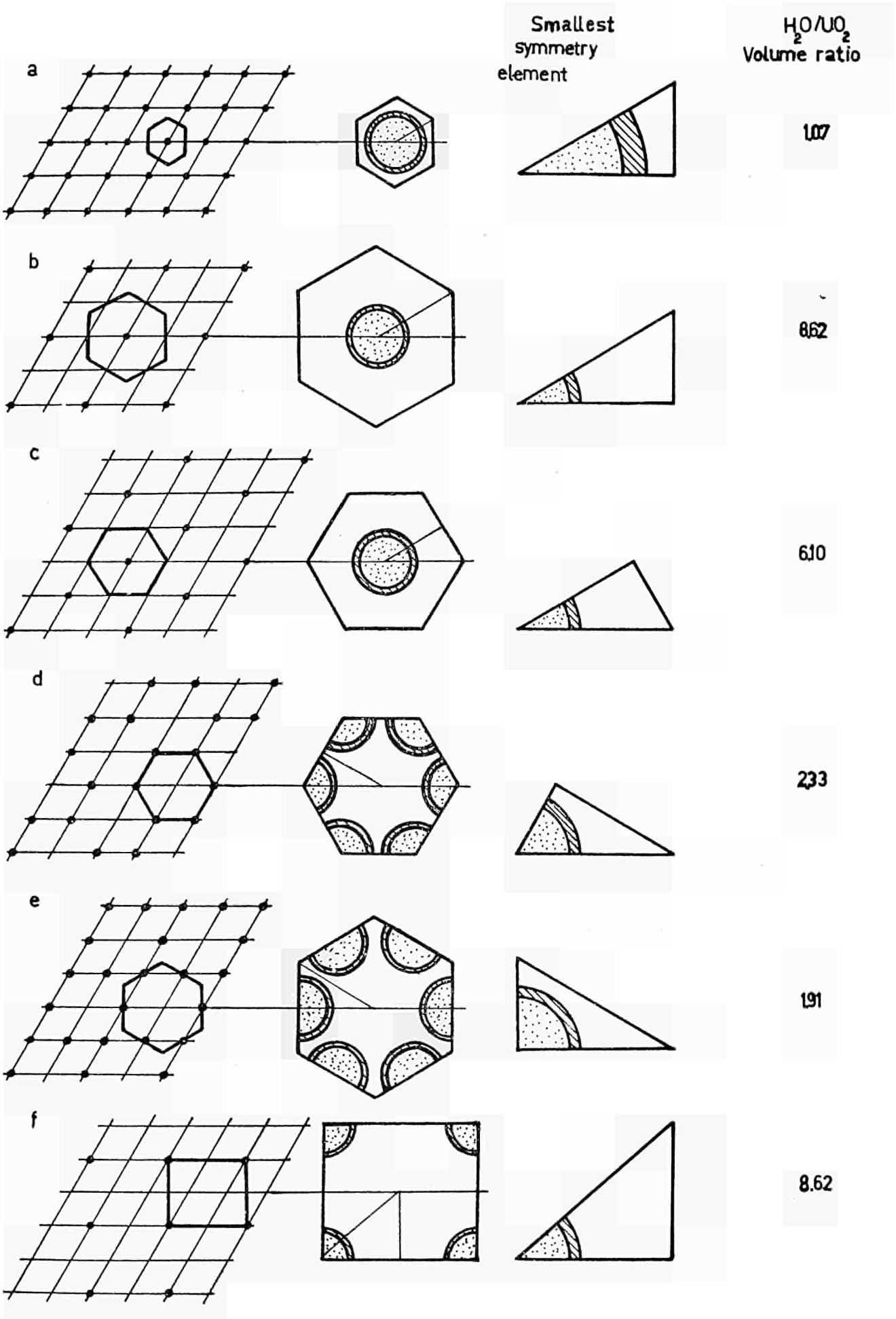
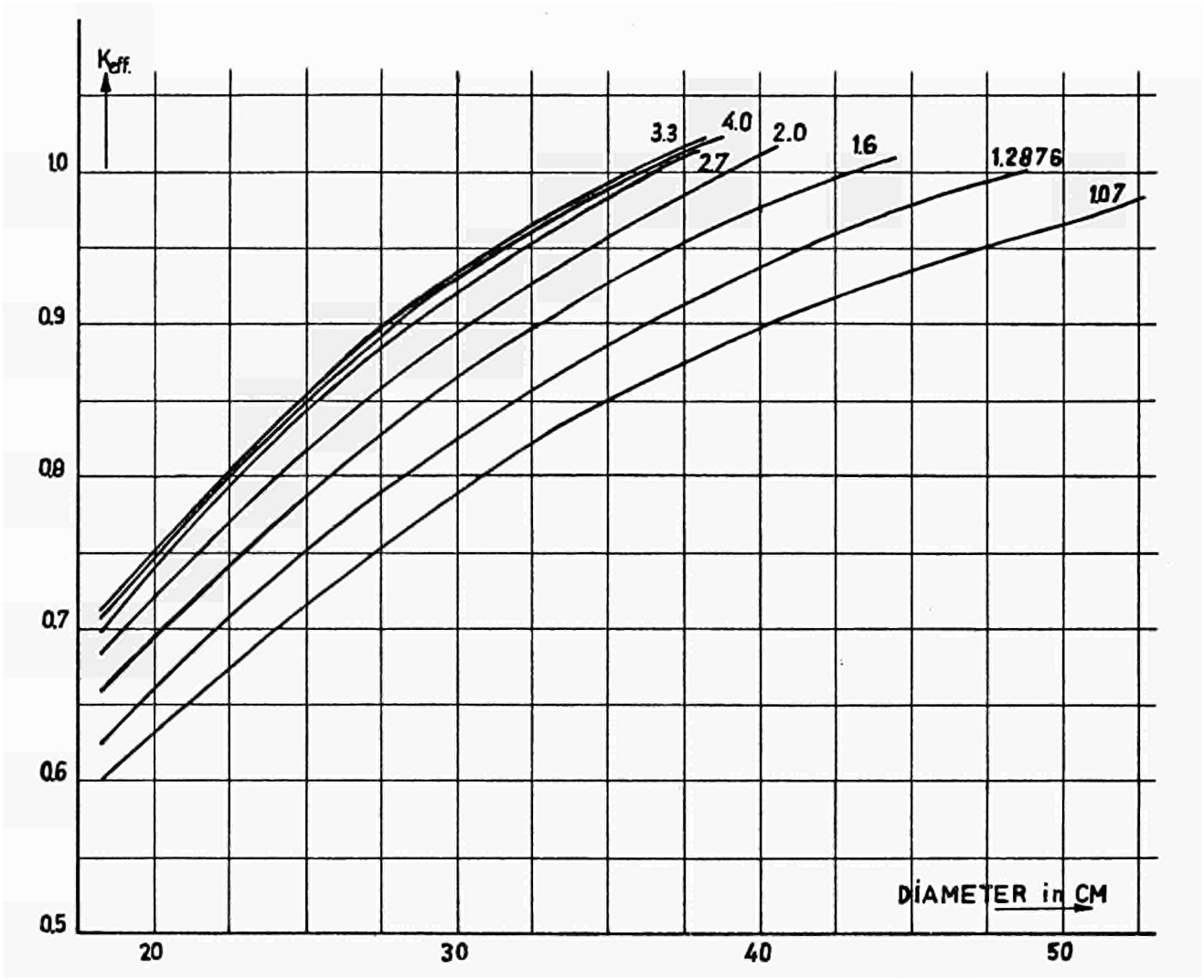


Figure 6. Cross_section through some lattices and the smallest symmetry element



Core height 50 cm.
^{235}U content in U 3,8 at.%.

Figure 7. K_{eff} versus core diameter with H_2O/UO_2 volume ratio as parameter