EUROPEAN ATOMIC ENERGY COMMUNITY - EURATOM

ORGEL - PROGRAM

REACTOR PHYSICS

JUNE 1962

THIS REPORT DESCRIBES
THE WORKS CARRIED OUT
IN THE FRAME OF THE
ISPRA REACTOR PHYSICS DEPARTMENT

INDEX

I. ORGEL PHYSICS CALCULATION METHODS

The reactor physics calculations are divided into the following parts :

1) Unirradiated lattice calculations

The calculations of the lattice parameters are performed by the CAROLINE I code (ref. 1).

This code is written in Fortran language for IBM 7090.

The method used to calculate the unirradiated lattice parameters has been established at Euratom as the basis of the French correlation on heavy water cooled and moderated lattices and tested on the results from recent Canadian and Euratom's critical experiments on natural uranium-organic cooled-heavy water moderated lattices.

Among the input data of Caroline I it is necessary to calculate some "spectral constants" to take into account the change of the spectrum due to the strong difference of temperature between the heavy water and the coolant.

The calculation of these "spectral constants" is performed by TERMIDOR (ref. 2), a code written in Fortran language for IBM 7090. The assomption is made that the hardening of the spectrum due to the absorption in the fuel element and the heating effect due to the scattering interaction of the neutrons with the hot coolant, are independent from one another; on the basis of this assomption the evaluation of the latter effect is performed by superposing in the many thermal group model two pure Maxwellian terms at the physical temperatures of the moderator and coolant.

2) Xenon poisoning calculation

The calculation of the Xenon poisoning corresponding to equilibrium state is performed by the perturbation method (ref. 3).

An average factor G is defined which allows to reduce the calculation on the reactor to a calculation on a point. The curves of G as a function of the maximum flux of the reactor are plotted by assuming the following parameters :

$$\varrho_o = \frac{R_o}{R_1} \; ; \; \varrho_1 = \frac{R_1}{R_e} \; ; \quad h = \frac{H}{H_e}$$

where :

R_o = radius of flattened zone of the pile

R_1 = external radius of the pile

R_e = extrapolated radius

H = height of the pile

H_e = extrapolated height of the pile

For the calculation of the Xenon poisoning the values of R_o, R_1, R_e, H, H_e, are established from a one-group calculation of the core.

3) Effects of non-uniform structure materials on the reactor

The effects on the lattice balance of structure materials as joints between fuel elements in the channel, structures of the control system and so on, are evaluated by PDQ and CURE codes.

4) Long term reactivity calculation and fuel cycle

The long term reactivity effects are calculated by a method established at Euratom (ref. 4).

The point reactivity curve is determined by using mainly the AECL proposed data for the fissions products, the Westcott factors for the spectrum effects.

The neutron temperature is calculate d by means of a simplified model (ref. 5) which evaluates separately by simple methods the heating effects of the coolant and the hardening due to the selective absorption of the fuel. A typical curve showing the dependence of the neutrons temperature on the coolant temperature is plotted in fig. 1.

Due to the fact that in our correlation for unirradiated lattices the p factor has no physical meaning, one cannot use this value to calculate the initial conversion factor. Therefore, we evaluate this factor by using the values suggested by Hellestrand to calculate the effective resonance integral.

The method established for point reactivity calculation is being tested on the experimental Canadian result s for rods irradiated in NRX and oscillated in GLEEP.

The variation of the reactivity balance following the fuel cycle assumed in the design is calculated by the RLT2 code (ref. 6).

The method utilized in this code is based on the perturbation theory. The steady state conditions are supposed.

The following types of fuel cycles have been considered up to now in the code :

Homogenized fuel model
Single rod fuelling
Two slug bidirectional fuelling
Continuous bidirectional fuelling.

This program will be compared in the next months with the more elaborated "fuel move" code (ref. 7) to test the validity of the approximations done.

Critical size and control rod calculations

Two group calculations to determine the critical size are performed by WANDA code.

After having fixed the total reactivity to be controled, it was possible to determinate the number of control rods by means of the supercell method (infinite assembly, one energy group, homogenized core).

A more accurate calculation of the efficiency of the control rods was successively made by the equivalent poison method, using the programmes WANDA 4 (1 dimension, 2 groups, homogenized core) and PDQ for 2 dimensions (14). With the latter a certain fine structure can be considered. The extrapolation length for the thermal group at the rod surface has been found by the KUSHNERIUK method, while the one for fast neutrons has been taken as infinite.

II. ORGEL DYNAMICS

The ORGEL dynamic behaviour is being studied by the Theoretical Dynamics Group in two ways :

1) a detailed core dynamics study by means of a relatively complex model,
2) a plant dynamics study with the core only as a simplified point reactor.

To 1) : The mathematical set-up of the core dynamics is given in the internal reports (10) and (11). The machine calculations will be performed during June 1962.

The reactor kinetics are described by the usual equations, where the main variable is the total reactor power supposed to be rigorously proportional to the neutron density. The number of delayed neutron groups is reduced to two by a least square method (12); influence of photoneutrons from D_2O is included.

The reactivity feedback is represented by several temporally constant temperature coefficients times the temperature excesses of the corresponding materials over their steady state initial values. The neutron temperature is a linear combination of the D_2O and the coolant temperature.

The materials involved are the fuel (UC), the can (SAP), and the coolant.

Heat losses through pressure tube are described by an overall heat transfer coefficient, whilst the moderator temperature is kept constant.

Since all temperatures and also their transient excesses are space dependent they must be averaged over the volume before introducing them in the kinetic equation by the squared neutron flux. (one group method).

Nevertheless the spatial temperature distribution is considered in the heat transfer equations. For each material the thermal balance is established. The axial derivitives in the coolant's balance are replaced by finite difference quotients which force to introduce some inlet and outlet temperatures. In order to keep a sufficient fine structure, the channel length is thus subdivided into several zones, formerly 5, now 3. The zone onlet (or outlet) temperatures are proper averages of the mean temperatures of adjacent zones.

On account of the pyrolytic disintegration danger a uniform core outlet temperature for all channels is desirable. Thus the channel flow rates must be adjusted proportionally to the power developed. Provided the same geometry for all channels, the radial coolant speed distribution is proportional to the neutron flux distribution. Though a centrally flattened flux is assumed, the heat transfer coefficients then vary considerably in radial core direction.

The can and fuel temperature excesses are for this reason better described when introducing 2 (formerly 3) concentric shells of the core. Thus 2x3=6 subvolumes exist, the heat transfer behaviour of which is described in parallel.

Special care has been devoted to a good description of the temperature-dependent thermal contact resistance between fuel and can, since this value is responsible for inherent reactor stability, the fuel temperature coefficient being practically the only negative one.

Normally, the system is not closed without describing back-reactions on the inlet temperature propagated through the primary circuit. But we simulate some kind of regulator which allows a control rod motor to drive as long as the coolant outlet temperature deviates from a reference temperature (400°C). The characteristics of the regulator should be found out from the simulation itself. Then the original regulator can be coupled with the simulator. In this way each manual action on the reactivity is more or less rapidly counter-balanced by the regulator action and no power change follows.

It has been shown that mass flow changes are not desirable in order to change the reactor power. Thus the only remaining control method is to change the inlet temperature. If the secondary flow rate is decreased, an increase in the thermal resistance through heat exchanger results. Since the heat exchanger inlet temperature is constant its outlet temperature grows up which is just the desired effect for the core. Some advanced signal from the secondary plant can be foreseen for the shim rods, if necessary.

As long as the time behaviour of the core inlet temperature is not known from plant simulation, arbitrary variations of it are introduced in order to study the power and temperature development inside the core. The complete system consists of 22 differential plus 42 algebraic equations.

To 2) : The plant dynamics considers as main components : (point) core, heat exchanger, turbine with intermediate superheating, condenser, pumps, preheating of supply water; and in the primary circuit : moderator cooling, organic pressure control. All necessary control loops are simulated in such a way that a minimum number of true inputs exists. The heat exchanger type is a BENSON boiler without steam drum and shifting limits between economizer, evaporator, and superheater. A primary bypass loop is probably not necessary. Perturbations like power changes as well as accidental cases like ruptures etc. shall be studied.

This work is done under contract by the German firm SIEMENS-SCHUCKERTWERKE AG. and will pass into the Ispra analogue machine in July and August 1962.

After having exercised these two problems, which each one requires the whole analogue facility of two PACE computers, it is intended to couple the systems. Then 3 computers will be available, but the description of both parts must be simplified. The purpose is to examine such accidents like primary pump failures (with flow rate changes) which influence markedly

the temperature field inside the core.

From the theoretical standpoint there is a severe drawback of the model insofar as the core heat transfer can be described spatio-temporally (repartition into subvolumes), but not the neutronics. The usual kinetic equations are the temporal part of separated solutions of the partial differential equation for neutron diffusion. Thus no spatial distortions of the neutron flux or the power distribution can be taken in account.

Of course, such distortions exist as well already for the steady initial state due to rod insertions and the temperature field as for the transients due to shim rod movements and new temperature excesses.

As a basic study it is thus intended to abandon the known kinetic equations and to treat a time spatio-temporal dynamics starting from the neutron diffusion equation itself. The complication is considerable because the problem is an initial value problem for the time but a boundary value problem in space. The number of space coordinates to be considered as well of neutron energy groups in the diffusion equation is a question of digital machine capacity. The calculation must be started from zero power where the neutron flux is not yet distorted by temperature reasons.

Studies on self stability of ESSOR have been also made at Ispra while the design was still in evolution, and the stability appeared to be sufficient, due to temperature coefficients alone.

Now, a detailed mathematical model has been worked out by Groupement Atomique Alsacienne Atlantique and Interatom, taking into account the different control devices and test- loops regulation, with special attention to the interference of loop regulations on the kinetics of the reactor, and its control.

The model was initially represented by a set of 95 equations. Due to the limited capacity of the computers available at Ispra (two PACE 231 R, i.e. 200 amplifiers), the problem has been simplified to 80 equations. This ultimate problem is now undergoing preliminary tests on computer, and results should be available in June '62.

III. AQUILON II. CRITICAL EXPERIMENTS

A set of experiments was carried out during the months of June and July 1961, on the Aquilon II heavy water, natural uranium research reactor at Saclay, France (ref. 8).

The object of these experiments was to determine the material buckling of natural uranium oxide lattices, moderated by heavy water and cooled by organic liquid of ORGEL type.

The measurements were performed using the method of progressive substitution developped by R. Naudet and wo-worker at Saclay.

The ORGEL elements were made up of a cluster of 19 rods of clad natural uranium oxide, submerged in organic liquid and held in an hexagonal tube (fig. 2).

The characteristics of the three ORGEL element types used for the measurements are the following :

Type	AO 120	AO 121	AO 161
Fuel rod diameter (mm)	12	12	16.2
Uranium oxide density (g/cm^3)	10.15	10.15	9.85
Can inside diameter (mm)	13	13	17
Can thickness (mm)	1	1	1
Can material	Mg	Mg	Al
Magnesium density (g/cm^3)	1.74	1.74	
Aluminium tube and can density(g/cm^3)	2.70	2.70	2.70
Volume organic liquid cm^3/cm	12.33	12.33	17.43
Equivalent inside tube diameter(mm)	76.50	76.50	95.38
Tube thickness (mm)	1.2	1.2	1.0
Center-to-center distance between rods (mm)	15	16	20

The organic liquid usedwas monoisoprophyldiphenyl having the empirical formula $C_{15}H_{16}$ with a density at 20°C of 0.973 g/cm^3 (4.78 10^{22} hydrogen atoms per cm^3).

A chemical analysis carried out on an aluminium sample piece taken from a tube showed the presence of the following impurities (% by weight) :

Cu	$<$	0.01
Mg	$<$	0.01
Mn	$<$	0.01
Si	=	0.18
Fe	=	0.22
Zn	$<$	0.01
Ti	=	0.006

The buckling results B^2 and the corresponding uncertainties δB^2, shown in table 1, relate to a heavy water purity of 99.8% and to a measurement temperature of 20°C.

Table I

Element	AO 120 (Mg)			AO 121 (Mg)			AO 161 (Al)		
Pitch(cm)	19	21	24	19	21	24	19	21	24
$B^2(m^{-2})$	5.97	3.59	3.19	4.12	3.83	3.43	2.91	3.39	3.71
$\delta B^2(m^{-2}) \pm$	0.125	0.15	0.13	0.12	0.14	0.12	0.14	0.15	0.12

The comparison between the experimental buckling results and those calculated by the CAROLINE I method is shown in figg. 3,4,5

IV. ORGEL EXPERIMENTAL PHYSICS PROGRAM

Measurements of the effective resonance capture integral and of the fine structure of the flux in ORGEL type fuel elements will be performed by the Neutron Physics Service, C.C.R. Euratom, Ispra, Italy, within the framework of the experimental program related to ORGEL lattice physics.

1. Effective resonance capture integral

The fuel element investigated is of the ECO type, and consists of an hexagonal array of 19 natural uranium metal rods, 1.2 mm dia., contained in an aluminum tube filled with organic liquid. Two geometries (corresponding to rod spacings of zero and 1 mm., respectively) and two organic liquids (diphyl and monoisopropyldiphenyl) will be tested.

The measurement will be performed by the activation technique, using gold as the standard. The irradiation facility selected is the Ispra-I reactor at the Euratom Center of Ispra (a 5 MW, 90% enriched uranium, CP-5 type reactor). Experimental work should start during the second semester 1962.

2. Fine structure measurement

The fuel elements investigated are :

A. Natural uranium metal rods, 44 mm dia., contained in cylindrical aluminum tubes filled with organic. The thickness of the organic layer in the element is 3 and 5 mm, respectively. Organic liquids employed will be diphyl and monoisoprophyldiphenyl.

B. ECO elements, with the same geometries and organic liquids as above.

The measurement will be performed by activation of dysprosium detectors placed in the fuel and in the organic. The irradiation will be carried out in the Aquilon II facility, CEN, Saclay, France (a natural uranium, heavy-water moderated critical assembly), replacing the central portion of the reference lattice with the lattice under consideration. Experimental work should start during the second semester 1962.

V. ORGEL CRITICAL EXPERIMENT (ECO)

ECO is a critical experiment conceived especially for buckling measurements by substitution method.
Its purpose is essentially the study of heavy water moderated and organic cooled lattices, the fuel is metallic uranium, uranium oxide or carbide.

Anyhow, according to the Commission's wishes, ECO may be used for buckling measurements on lattices using cooling fluids other than organic liquids.

Schematically ECO consists of an aluminium tank which can hold 20 t of heavy water and which lays on a 90 cm thick lower graphite reflector. Radially the tank is surrounded by a thermal insulator and a 90 cm reflector. The upper end of the fuel elements plunged in heavy water, hang on an automatic mechanism which allows a continual pitch variation of the lattice between 170 and 300 mm. The assembly is closed in a light concrete biological shielding having 1,70 m thickness; this allows experiments up to 1 KW power.

The ECO control is performed by the heavy water level adjustment (by means of pumps of different speeds and by means of a set of dosing vessels), as well as by a set of two vertical regulating plates sliding between the tank and the reflector and two horizontal regulating plates circulating in the lower reflector under the reactor tank.

ECO is an extremely flexible working instrument and it includes a large number of facilities allowing numerous physical experiments both inside and outside the reactor.

Besides the automatic lattice pitch drive we mentioned before, there is the possibility of using the reactor as a bare pile by means of the introduction of a boral skirt between tank and side reflector. The temperature coefficient measurements are made easier by the possibility of adjusting in a continuous way the moderator temperature between 10 and 80°C, whilst the coolant temperature of the 25 central fuel elements, forming the substitution lattice, may be controled between a low value corresponding to the moderators temperature and a high value of approximately 300°C.

A swinging channel being able to move inside the reactor between the lattice fuel elements allows the extraction of a neutron beam from the center of the reactor; this is intended for spectrum measurements by the time-of-flight-method.

An accelerator moving under the reactor shielding can send its beam on a target placed in the mid of the reactor.

A 10 m long axial channel crossing the reactor axially from bottom to top allows the oscillation of two fuel elements linked end to end.

In order to compare the irradiated or synthetic fuel to the unirradiated fuel a large number of experimental accessories is presently being studied to allow the physicists to obtain as much information as possible from this critical experiment; let us mention for instance the facilities allowing the periodical variation of organic liquid appearent density in the clusters and allowing oscillation of certain cluster elements, some pre-set automatic counting units, a generalized set for the processing of critical experiments data, some samples oscillators for cross section measurements, a source jerk facility for pulsed studies, a semi-hot cell for dismantling the lightly or moderately irradiated fuel elements, some sets of miniature drills and of activable detectors allowing flux mapping by different methods, a controled poisonning circuit and so on....

Progress Report

The building was studied by the Architecture Office of the C.C.R. - Ispra and an Italian firm is in charge of its realization. The contract has been signed on the 8th May 1962, but works already started in March and the foundations are now achieved.

At this moment the underfoundations of the real reactor bloc are being built.

A Dutch firm has been charged by a contract signed on December 15th, 1961 of the reactor engineering, the preliminary design of which had been executed at the Reactor Physics Department. The construction of the reactor itself will start practically against the end of May 1962.

Buildings should be achieved at the end of 1962 and the reactor circuits will be in a position to work with light water against May 1963, whilst the official divergence would take place at the 1st of July, 1963.

Nearby the ECO building, another building is simultaneously being constructed; it will contain laboratories for the groups assuring ECO operation and maintenance.

A first set of fuel elements constituting the reference lattice, (clusters of 19 uranium metal pencils) is presently being manufactured in a German firm and will be delivered to Ispra at intervals of time beginning on the 1st of July, 1962.

The reflector's graphite is delivered by a French firm and its nuclear control is made by CEA.

The Reactor Physics Department is now studying the transport and handling problems of the heavy water for the critical experiment and the Brussel's Free University is presently performing an hydraulic test on a full size mock-up in order to study the influence of the hydraulic circuits of the reactor on the thermal homogeneousness and moderator disturbance.

The oscillation mechanism of fuel elements in the axial channel works already as a reduced scale mock-up; the accelerator, a 1 Mev Van de Graaf, has been ordered and should be delivered at Ispra at the end of 1962.

The remaining accessories are presently being studied or realized.

VI. ESSOR PHYSICS CALCULATIONS

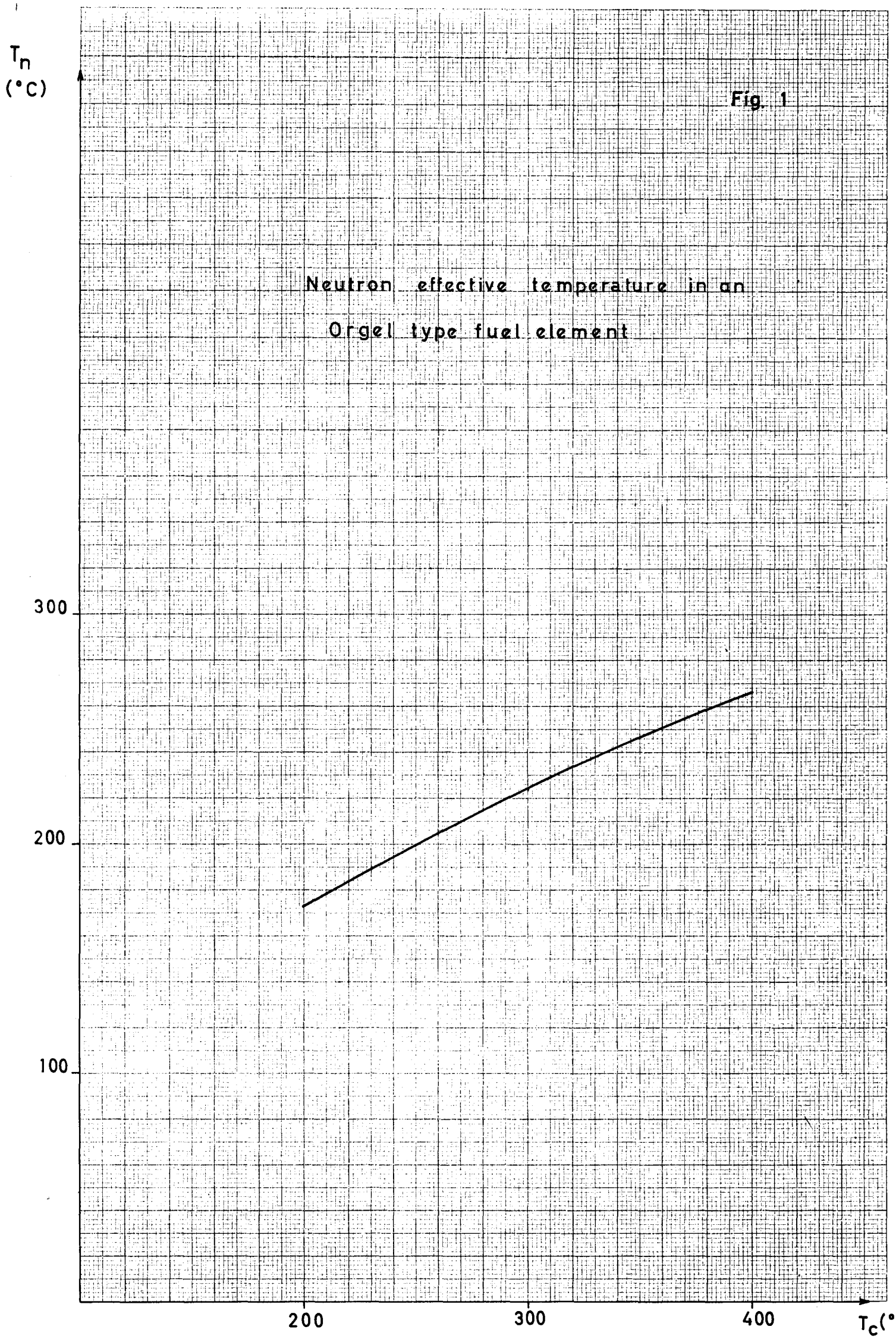
The work made for the ESSOR reactor may be divided into two parts.

In the first part we performed the neutronic study of a variant of the reactor. Namely, we studied a reactor in which the ORGEL channels were arranged on a circular ring and inside there were the feeding fuel elements; therefore an arrangement exactly opposite to the primitive one. The aim of this change was to reduce the power produced in the feeding zone and to increase the uniformity of the irradiation of the ORGEL channels. The main results of this study are reported in "Dossier de comparaison ESSOR" written by G.A.A.A. and INTERATOM in order to compare the two variants.

In the second part we studied the way of flattening the axial distribution of the neutron thermal flux in the ORGEL fuel elements in order to satisfy the following condition : the difference between the flux values at the centre and each extremity of a portion of fuel element 700 mm long cut into halves by the horizontal plane of symmetry of the core, must be smaller than 10 %.

The calculations made by us (ref. 9) showed that the use of feeding elements with a central zone without uranium is an effective way of flattening.

Fig. 1



Neutron effective temperature in an Orgel type fuel element

AQUILON II CRITICAL EXPERIMENTS

ORGEL FUEL TYPE ELEMENT

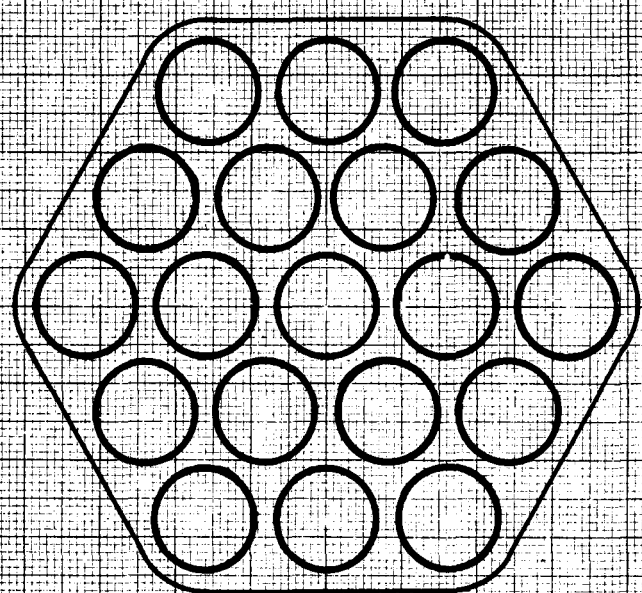
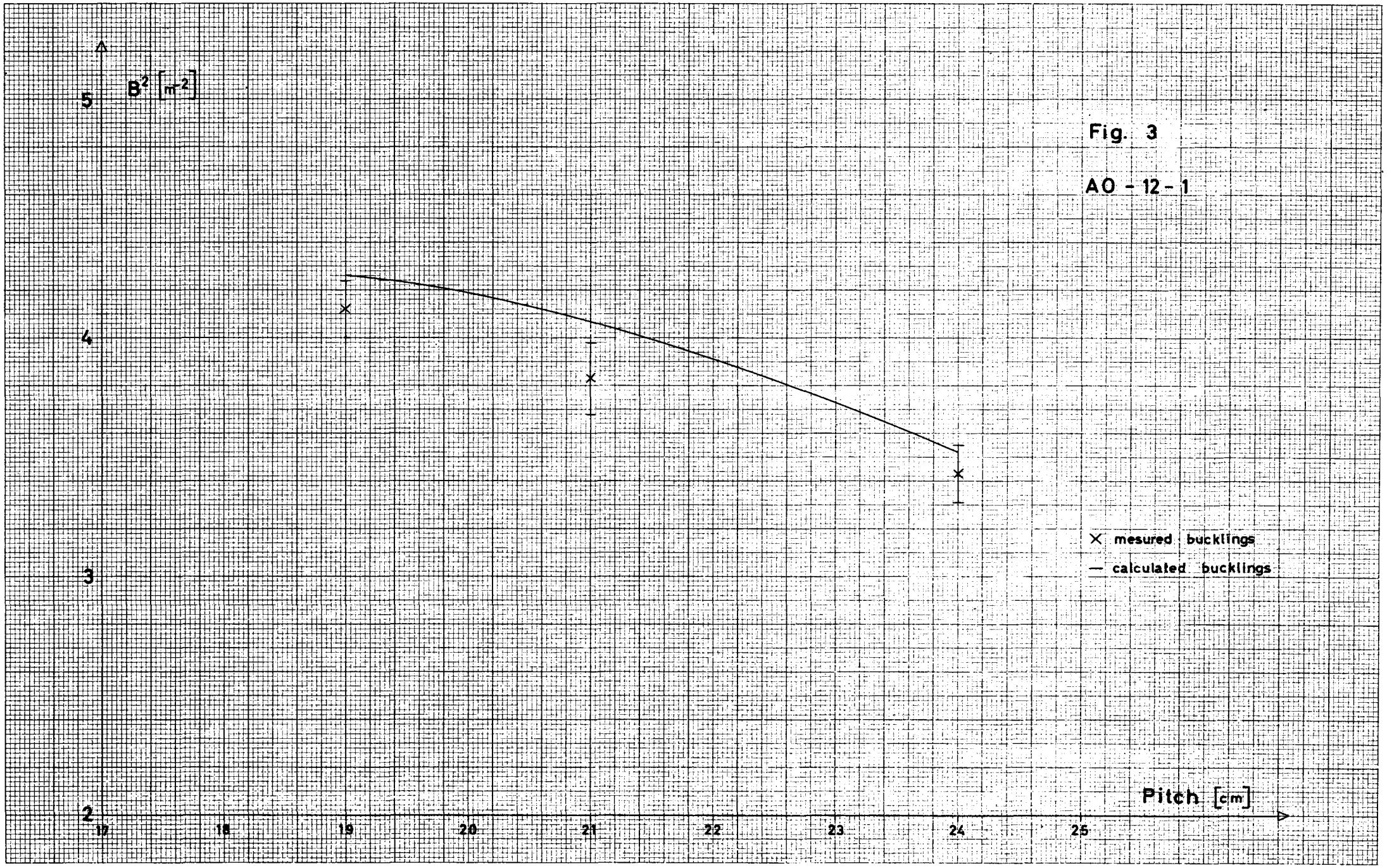


Fig. 2


B^2 [m^{-2}]
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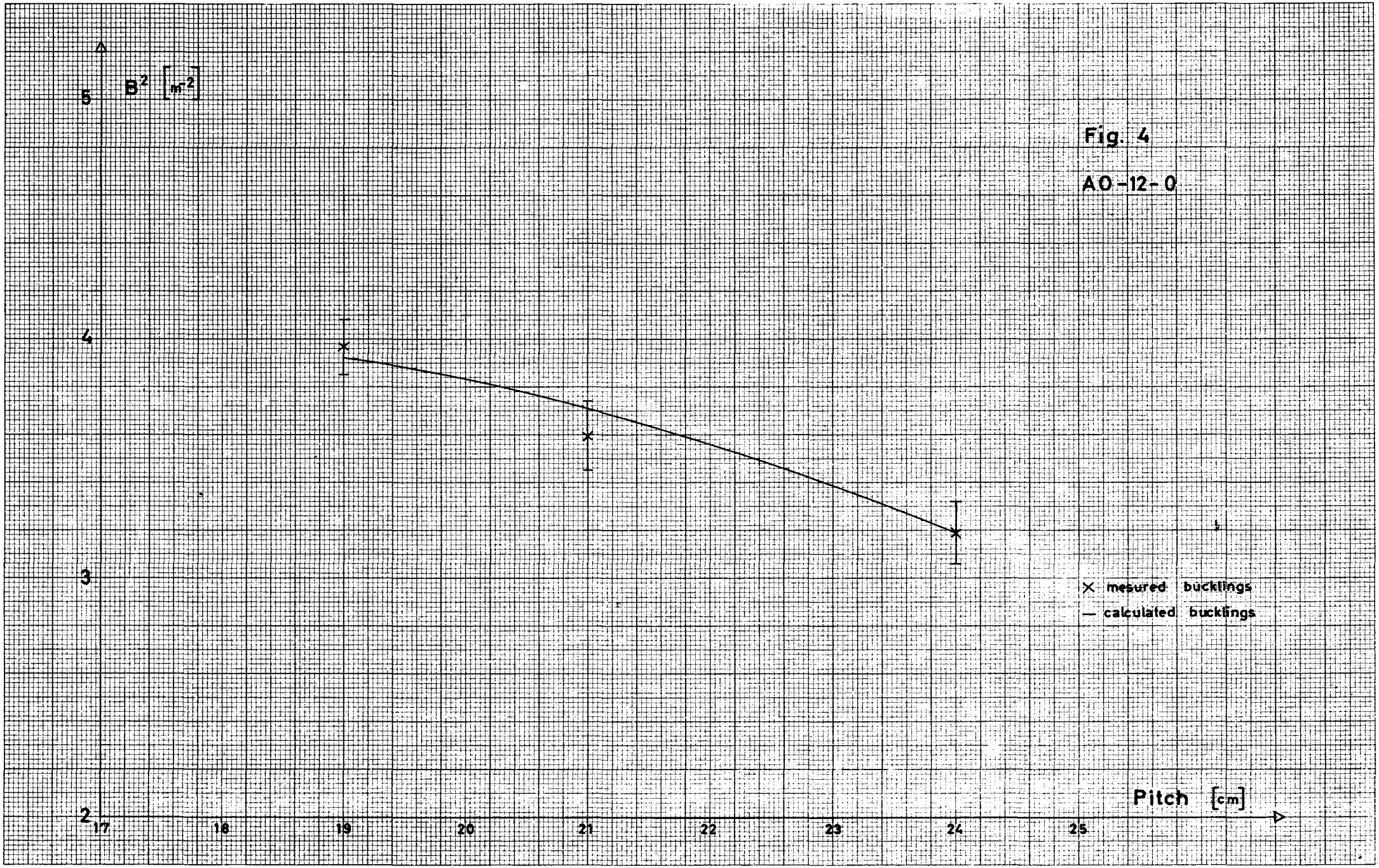
25
Pitch [cm]
× mesured bucklings
— calculated bucklings

Fig. 3

AO - 12 - 1


B² [m⁻²]
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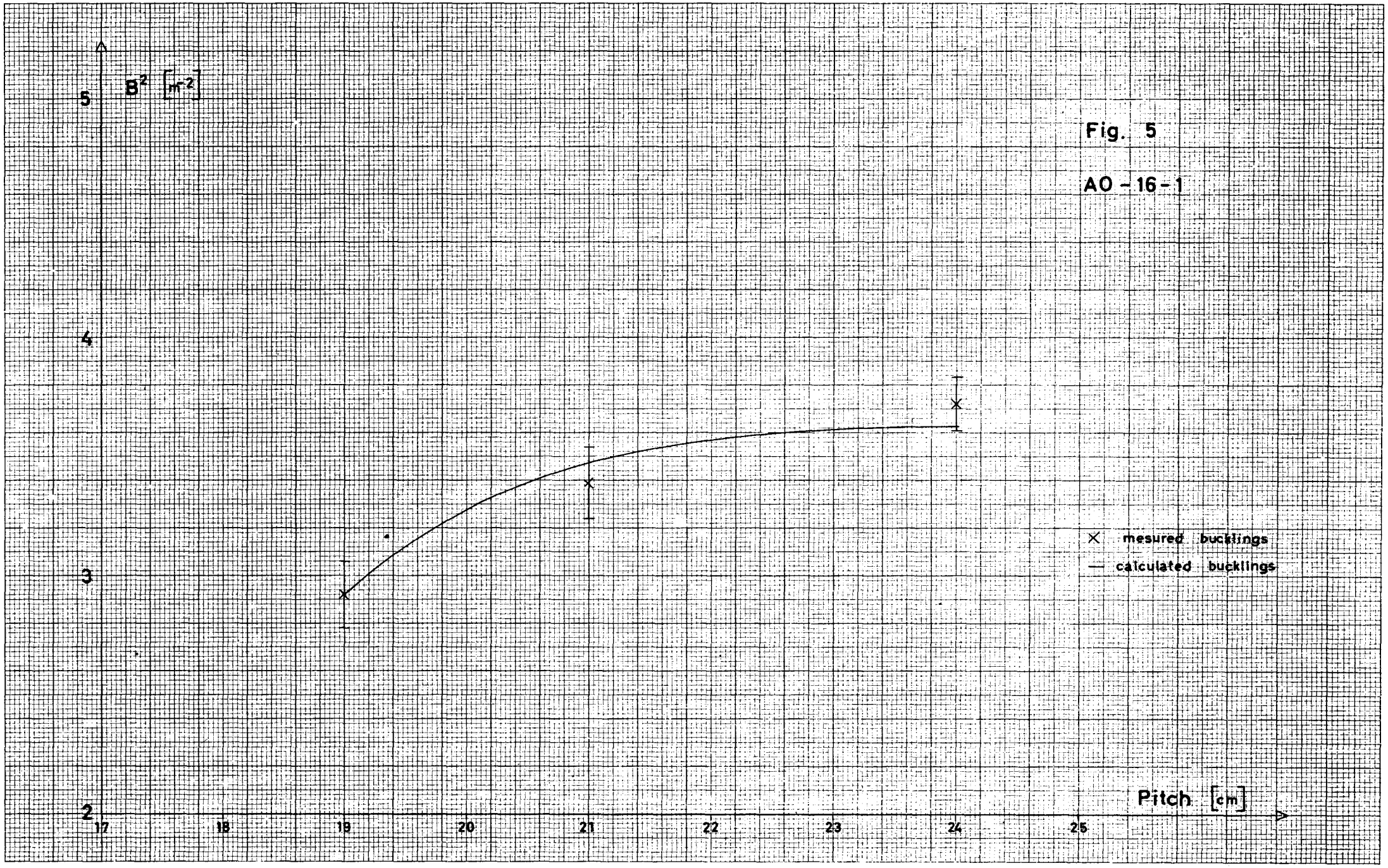
× mesured bucklings
— calculated bucklings

Fig. 4

AO-12-0


B^2 $[m^{-2}]$
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Pitch [cm]
× mesured bucklings
— calculated bucklings

Fig. 5

AO - 16 - 1

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