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The Karlsruhe Fast-Thermal Argonaut Reactor Concept

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Abstract

The Fast-Thermal Argonaut Reactor STARK is a flexible zeropower facility suitable for various fast neutron experiments. A fairly subcritical uranium metal zone is driven by a surrounding thermal Argonaut core; according to four-group calculations, an asymptotic fast reactor spectrum is attained in the central region of the assembly, while a high degree of inherent reactor stability is guaranteed by the self-limiting properties of the light-water moderated Argonaut core, provided that the ratio of fission rates in both regions is properly limited.

Introduction

A coupled fast-thermal Argonaut Reactor (STARK) has been constructed at Karlsruhe by inserting a uranium metal fast core into the central part of the original thermal Argonaut+. This reactor will be used as a source reactor for testing fast neutron equipment and as a means for special investigations on coupled systems, e.g. spectrum measurements, material replacement, and kinetic experiments.

The basic structure of STARK is shown in Fig. 1: The central part of the assembly is a cylindrical region (37 cm dia., 61 cm height) consisting of enriched uranium metal and additional Al_2O_3 or graphite to simulate a fast reactor core. This test region is entirely enclosed in a 5 cm thick natural uranium casing acting as a semipermeable layer which absorbs thermal neutrons incident from the outside. The central fast region of the assembly is surrounded by a thermal driver region composed of an internal graphite annulus, the original thermal Argonaut core, and an external graphite reflector. Twelve Cd-plate control units are symmetrically arranged in the graphite reflector and one additional fuel-poison safety rod in the fast core thus giving a total control reactivity of at least 1.8 percent k.

+The preliminary version in operation since August 1964 contains a natural uranium fast core. The calculations presented in this report were made for the standard loading to be in operation in mid - 1965

The core regions of STARK differ with respect to material composition, neutron spectrum, and effective neutron lifetime. In the central fast region, the neutron spectrum is quite similar to that of a fast reactor (cf. Fig. 6); therefore, most fission processes are caused by fast neutrons having a rather short lifetime of the order of 10^{-7} sec. Fission processes in the annular Argonaut core, on the other hand, are mainly due to neutrons being thermalized before in the light-water moderator thus giving rise to a neutron lifetime of about 10^{-4} sec in that region. Each of the two regions is substantially subcritical if considered as isolated from the other, while the combined system is just critical due to a strong coupling by fast neutrons leaking out of the thermal region into the fast core and vice versa.

For reasons of reactor safety, the multiplication factor of the isolated fast core is kept relatively low ($k_{eff} < 0.8$) by a proper choice of fuel enrichment (≈ 10 percent), so that thermal fission processes are always necessary to maintain the chain reaction. This leads to the following consequences:

1. The overall neutron lifetime of the system is governed by the slowest part of the fission chain, namely the neutron lifetime in the thermal region. Hence, the general time behavior of the reactor is very similar to that of a thermal system.

2. The reactor can be shut down, in principle, by any mechanism acting only upon the thermal region. Therefore, we can take advantage of the negative temperature and void coefficient of the water-moderated core as an inherent safety mechanism; similarly, the cadmium plates of the original Argonaut can be used as control and shut-down devices, so that there is no need for safety rods acting directly on the fast core.

   Hence, under proper restrictions imposed on the multiplication of the fast region, STARK can be operated as safely as a thermal Argonaut with only minor changes of the control system being necessary.

1. Basic Considerations and Safety

1.1. General Properties of Coupled Reactors

Before evaluating the safety characteristics of STARK quantitatively, let us first consider the physics of coupled reactors in some more detail. Generally, a coupled system can be treated in the same way as a uniform reactor by applying usual multigroup calculation methods. From the resulting neutron flux and adjoint distribution, integral parameters (e.g. multiplication constant k, neutron lifetime l etc.) are derived by which a simple description of kinetic reactor behavior near critical is given. At larger deviations from critical, however, appreciable flux distortions may occur in a coupled system giving rise to a variation of these parameters so that a more refined integral theory is needed in such cases.

We are using the formalism developed by Avery (1), (2), which is based on the steady-state integral fission rates, S_1 and S_2, within both regions of the coupled system, index 1 and 2 referring to the fast and thermal core regions, respectively.

The fission rate S_i of region i is composed of two parts,

$$S_i = S_{i1} + S_{i2} \qquad (i = 1, 2), \qquad (1.1)$$

where S_{ij} represents that part of S_i which is initiated by neutrons born in region j. On this basis, partial multiplication constants k_{ij} of the coupled system at critical are defined by

$$k_{ij} = S_{ij}/S_j \qquad (i,j = 1, 2) \qquad (1.2)$$

Hence, k_{11} and k_{22} are the effective multiplication constants of the decoupled regions 1 and 2 with the flux distribution of the critical system maintained; k_{12} and k_{21} are coupling coefficients referring to neutrons born in one region and causing fission in the other.

According to eq. (1.2), the parameters k_{ij} are obtained in terms of the flux $\phi(v,r)$ of the critical reactor

$$k_{ij} = \frac{a_i \iiint \chi(v')\phi^+(v',\vec{r}) \left[\nu \Sigma_f(v,\vec{r})\right]_i \phi_j(v,\vec{r})\, d\vec{r}\, dv'dv}{a_j \iiint \chi(v')\phi^+(v',\vec{r}) \left[\nu \Sigma_f(v,\vec{r})\right]_i \phi(v,\vec{r})\, d\vec{r}\, dv'dv},$$

$$a_j = \iint \left[\nu\Sigma_f(v,\vec{r}\right]_j \phi(v,\vec{r})\, d\vec{r}\, dv\ , \tag{1.3}$$

where $\chi(v)$ = fission spectrum, Σ_f = fisssion cross section, and the symbol $[\ldots]_j$ indicates that the volume integral is to be taken only over region j. $\phi_k(v,\vec{r})$ and $\phi^+{}_k(v,\vec{r})$ are fictitious partial fluxes and adjoint functions obtained by a single iteration of the reactor equations with fission neutron sources given by the flux distribution of the critical reactor in region k and by zero in the other region. Correspondingly, effective neutron lifetimes l_{ij} are associated to the S_{ij} which are given by (1)

$$l_{ij} = \frac{\iint \phi_i^+(v,\vec{r})\, \frac{1}{v}\, \phi_j(v,\vec{r})\, d\vec{r}\, dv}{\iiint \chi(v')\, \phi^+(v',\vec{r}) \left[\nu \Sigma_f(v,\vec{r})\right]_i \phi_j(v,\vec{r})\, d\vec{r}\, dv'dv} \tag{1.4}$$

So far, the multiplication parameters k_{ij} are only defined for a critical system. Let us now consider deviatinns from critical by assuming the number ν_i of fission neutrons in region i being changed by small amounts $\delta\nu_i = \mathcal{E}_i\, \nu_i$. In this way, we have

$$\frac{\delta k_{11}}{k_{11}} = \frac{\delta k_{21}}{k_{21}} = \mathcal{E}_1 = \frac{\delta\nu_1}{\nu_1}\quad ,\quad \frac{\delta k_{12}}{k_{12}} = \frac{\delta k_{22}}{k_{22}} = \mathcal{E}_2 = \frac{\delta\nu_2}{\nu_2}, \tag{1.5}$$

and new parameters k_{ij} of the non-critical system are introduced by

$$k'_{ij} = k_{ij}\ (1 + \mathcal{E}_j). \tag{1.6}$$

From eq. (1.3) and (1.5), the usual static multiplication factor k of the combined system is obtained as a weighted average of the partial reactivity changes $\delta\nu_i/\nu_i$

$$k - 1 = \alpha_1 \frac{\delta\nu_1}{\nu_1} + \alpha_2 \frac{\delta\nu_2}{\nu_2} \qquad (1.7)$$

with weighting factors

$$\alpha_1 = \frac{\Delta_2}{\Delta_1 + \Delta_2}, \quad \alpha_2 = \frac{\Delta_1}{\Delta_1 + \Delta_2}, \quad \Delta_i = 1 - k_{ii} \qquad (1.8)$$

thus defining a partition of reactivity with respect to both core regions.

Finally, kinetic equations for the fission rates S_{ij} are established as follows (1):

$$l_{ij} \frac{dS_{ij}}{dt} = k'_{ij}(1-\beta^{(j)}) \sum_{K=1}^{2} S_{jK} - S_{ij} + k'_{ij} \sum_{K=1}^{6} \lambda_K C_{jK} \quad (i,j = 1,2),$$

$$\frac{dC_{jK}}{dt} = \beta_K^{(j)} \sum_{K=1}^{2} S_{jK} - \lambda_K C_{jK} \quad (j = 1,2;\ K = 1,---6) \qquad (1.9)$$

and a relationship between the k_{ij} and the reciprocal period ω of the exponential behavior of reactor power is obtained:

$$\left\{\Delta_1 + \omega l_{11} + \omega k'_{11} \sum_{K=1}^{6} \frac{\beta_K^{(1)}}{\omega + \lambda_K}\right\} \left\{\Delta_2 + \omega l_{22} + \omega k'_{22} \sum_{K=1}^{6} \frac{\beta_K^{(2)}}{\omega + \lambda_K}\right\} (1 + \omega l_{12}) \qquad (1.10)$$

$$(1 + \omega l_{21}) = k_{12} k_{21} (1 + \omega l_{11})(1 + \omega l_{22})\left(1 - \omega \sum_{K=1}^{6} \frac{\beta_K^{(1)}}{\omega + \lambda_K}\right)\left(1 - \omega \sum_{K=1}^{6} \frac{\beta_K^{(2)}}{\omega + \lambda_K}\right)$$

In case of small deviations from critical, this reduces to the wellknown inhour equation

$$\rho = \frac{\omega l}{k} + \omega \sum_{K=1}^{6} \frac{\beta_K}{\omega + \lambda_K} \qquad (1.11)$$

($\rho = (k - 1)/k$ = reactivity) if the following identifications are made:

effective neutron lifetime

$$l = \alpha_1 l_1 + \alpha_2 l_2 \qquad \text{where} \qquad \begin{aligned} l_1 &= k_{11} l_{11} + (1-k_{11})\, l_{12}, \\ l_2 &= k_{22} l_{22} + (1-k_{22})\, l_{21}, \end{aligned} \tag{1.12}$$

effective delayed neutron fractions

$$\beta_K = \alpha_1\, \beta_K^{(1)} + \alpha_2 \beta_K^{(2)}\ , \qquad \beta = \alpha_1 \beta^{(1)} + \alpha_2 \beta^{(2)} \tag{1.13}$$

Obviously, the effective parameters of the coupled system are obtained as average values weighted with the coefficients α_i of the reactivity partition.

1.2. Fundamental Safety Concept of STARK

The fast-thermal reactor STARK will be operated under conditions which ensure general safety properties equivalent to those of the thermal Argonaut. In particular, the following conditions must be fulfilled:

1. The time-behavior of the reactor must be equivalent to that of a thermal system.

2. A sufficient shut-down reactivity $\geqslant 1.2$ percent k must be provided by the control system for compensation of all credible positive reactivity increments during reactor operation and loading.

3. In case of any failure of the safety system, the reactor must be self-limiting to all credible power excursions; the maximum credible accident should not exceed that of the thermal Argonaut.

Let us now impose certain limits to the parameters of possible core configurations such as to fulfill these fundamental safety conditions.

1.2.1. Reactivity Partition and Neutron Lifetime

Control and safety of STARK are essentially based on shut-down mechanisms (Cd-control rods, negative temperature and void coefficients) which act only upon the thermal core. The effect of these devices on multiplication may be expressed, as a first approximation, as a corresponding fictitious change $\delta\nu_2/\nu_2$ of the number of fission neutrons in the thermal region. Hence, the resulting reactivity change $\Delta\varrho$ of the entire reactor is given by eq. (1.7)

$$\Delta\varrho = \alpha_2 \frac{\delta\nu_2}{\nu_2}$$

where $\delta\nu_2/\nu_2$ depends only slightly on the parameters of the fast core. Similarly, the effective neutron lifetime l is obtained by means of eq. (1.12) in terms of the given thermal neutron lifetime $l_2 \approx 10^{-4}$sec

$$l \approx \alpha_2 l_2 .$$

In order to guarantee minimum values of $\Delta\varrho$ and l with given $\delta\nu_2/\nu_2$ and l_2, the factor α_2 of the reactivity partition, which is largely dependent on the fast core parameters, must not fall below a certain limiting value α_{2min}

$$\alpha_2 \geqslant \alpha_{2min} \tag{1.14}$$

α_{2min} = 0,5 was adopted for the normal operational state of STARK leading to a minimum neutron lifetime $l \approx 0.6 \cdot 10^{-4}$sec, while α_2 = 0,59, $l = 0{,}78 \cdot 10^{-4}$ sec and a total reactivity worth of control rods of 2.0 percent k were found for the proposed first loading of the reactor (cf. Section 3.).

In addition to eq. (1.14), a certain degree of rigidity of α_2 is required with respect to possible deviations from the normal operational state, e.g. in the case of a reactor excursion. Let us therefore investigate the variation of α_2 if certain amounts of reactivity, say $\varepsilon_1 \delta\nu_1/\nu_1$ and $\varepsilon_2 = \delta\nu_2/\nu_2$, are added to both core regions. According to eq. (1.6), the new parameters k_{ij} are

$$k'_{11} = k_{11}\,(1+\varepsilon_1)\ , \qquad \Delta'_1 = \Delta_1 - \varepsilon_1 k_{11}\ ,$$

$$k'_{22} = k_{22}\,(1+\varepsilon_2)\ , \qquad \Delta'_2 = \Delta_2 - \varepsilon_2 k_{22}\ ,$$

and the factor α_2 changes to

$$\alpha'_2 \approx \alpha_2\,(1 - \varepsilon_1\alpha_1\,\frac{k_{11}}{\Delta_1} + \varepsilon_2\alpha_2\,\frac{k_{22}}{\Delta_1}). \tag{1.15}$$

Assuming a maximum credible reactivity increment $\varepsilon_1\alpha_1 = 0.025$ k added to the fast core, which is compensated by a corresponding reactivity decrease $\varepsilon_2\alpha_2 = -\,0.025$ k given to the thermal region, and a resulting overall variation of α_2 not higher than 25 percent, eq. (1.15) leads to the condition $0.05/\Delta_1 \leqslant 0.25$, i.e. a limitation of the fast core multiplication factor

$$k_{11} \leqslant 0.8 \tag{1.16}$$

is required. In this way, the reactor is definitely kept from any approach to critical as a <u>fast</u> system.

1.2.2. Inherent Safety

The inherent safety of STARK is basically due to the fairly high negative temperature and void coefficients of the water-moderated Argonaut core. According to eq. (1.7), the reactivity change $\Delta\rho$ of the coupled system resulting from a temperature variation $\Delta\vartheta$ and a formation of a relative void fraction $\Delta V/V$ in the moderator of the thermal core is given, as a first approximation, by

$$\Delta\rho = (\frac{d\rho}{d\vartheta})_2\,\alpha_2\,\Delta\vartheta + (V\,\frac{d\rho}{dV})_2\,\alpha_2\,\frac{\Delta V}{V} \tag{1.17}$$

where the temperature coefficient $(\frac{d\varrho}{d\vartheta})_2 \approx -10^{-4}$ k/°C

and the void coefficient $(V\frac{d\varrho}{dV})_2 \approx -1.8 \cdot 10^{-3}$ k/percent void of the isolated thermal core are obtained by extrapolation from experiments on the original thermal Argonaut reactor (3). With a thermal reactivity contribution α_2 not lower than 0.5, STARK behaves like a water-moderated thermal reactor having overall temperature and void coefficients $\alpha_2(\frac{d\varrho}{d\vartheta}) \approx -0.5\ 10^{-4}$ k/°C and $\alpha_2(V\frac{d\varrho}{dV}) \approx -10^{-3}$ k/percent void.

The general transient behavior of light-water moderated MTR-type reactors was extensively studied in the BORAX and SPERT experiments (4), (5). An instantaneous positive step rise $\Delta\varrho$ of reactivity was given to the critical reactor, while the time-variation of neutron flux and fuel plate temperature was measured during the following self-limiting reactor excursion. This self-limitation is caused by the prompt-acting part of the temperature coefficient and by a very rapid steam formation at the surface of the fuel plates which finally leads to an expulsion of moderator water out of the thermal core.

The maximum fuel plate temperature attained in such an excursion was found to be primarily a function of the reciprocal period ω of the power rise and the initial moderator temperature ϑ ; it proved to be almost independent of the neutron lifetime l of the particular reactor. At a moderator temperature $\vartheta = 20$ °C, the melting point of the aluminium can is reached at $\omega = 250$ sec^{-1}; with higher moderator temperatures, the maximum fuel temperature decreases appreciably, since the steam generation occurs already in an earlier stage of the excursion. At higher ϑ, therefore, excursions with still higher ω are stopped without melting of the can material.

The results of these experiments are applicable to STARK since the void coefficient of STARK is almost equal to that of BORAX II and the thermal conductivity of the Argonaut fuel plates is better compared with the MTR-plates of BORAX II (6). The maximum positive step reactivity $\Delta\varrho_{max}$ causing a self-limiting excursion without melting of the aluminum can material was determined for STARK ($l = 0.78 \cdot 10^{-4}$ sec) using eq. (1.11) and the extrapolation method of Luckow and Widdoes (7). The resulting $\Delta\varrho_{max}$ plotted in Fig. 2 shows an increase from

$$\Delta\varrho_{max} = 2.5 \text{ percent k} \qquad \text{at } \vartheta = 20\ ^{\circ}\mathrm{C} \quad \text{up to}$$
$$\Delta\varrho_{max} = 5.0 \text{ percent k} \qquad \text{at } \vartheta = 80\ ^{\circ}\mathrm{C}.$$

Therefore, we can compensate for the effect of lower thermal reactivity contribution by increasing the moderator temperature up to 80 °C, so that $\Delta\varrho_{max}$ is equal to the 5.0 percent k valid for the thermal Argonaut+.

So far, we have shown that instantaneous reactivity steps as high as $\Delta\varrho_{max}$ result in a self-limiting excursion of STARK without melting of thermal fuel plates. Beyond that, we must be sure that there are no consequences of such an excursion which might lead to a secondary excursion or any other dangerous situation.

First of all, penetration of large amounts of water into the fast core may produce a reactivity increase up to several percent k, because of the high positive reactivity coefficient ($\Delta\varrho = 0.63$ percent k/vol. percent H_2O, Tab. II). Therefore, moderator water, which may be pushed out of the thermal region during an excursion, must be kept from entering the fast core by means of an effective water sealing system (cf. Section 2.1).

Furthermore, a meltdown of the fast core is to be excluded for excursions with $\Delta\varrho < \Delta\varrho_{max}$ in the following way: The ratio s_1/s_2 of the fission power densities released during the excursion within the fuel materials of both regions must be limited by a proper choice of reactor parameters in such a way, that the temperature in the fast core remains below the melting point as long as the melting point of the aluminum in the thermal region is not reached.

In a fast excursion ($\omega \gtrsim 10\ \mathrm{sec}^{-1}$), the temperature rise ΔT_i of fuel material in region i is given by the fission energy density

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+The discussi n in the last paragraph is based on the separability of the flux in space and time. The validity of this assumption for coupled reactors will be investigated in a forthcoming report.

$s_i = \frac{1}{c} \Sigma_f \int \phi(t)\, dt$ (c = fission rate per unit power):

$$\Delta T_i \, C_i \, \rho_i = s_i \tag{1.18}$$

where C_i = specific heat, ρ_i = fuel density. Hence, the following condition must be fulfilled by the fission energy densities s_1 and s_2 liberated in the fast and thermal core regions, respectively,

$$\frac{s_1}{s_2} < \frac{w_1}{w_2} \tag{1.19}$$

where the critical values w_i are given by the heat capacity corrsponding to a temperature rise $(\Delta T_i)_{max}$ up to the melting point of the fuel:

$$w_i = C_i \; \rho_i \; (\Delta T_i)_{max}. \tag{1.20}$$

In the fast core of STARK, the fuel is concentrated in platelets made of U(20) metal, ρ_1 = 18.8 g/cm^3, C_1 = 0.035 cal/g^{o}C, $(\Delta T_1)_{max}$ = 1100 ^{o}C and we have

$$w_1 = 724 \text{ cal/cm}^3.$$

The active material of the thermal fuel plates is a mixture of 44.5 weight percent U_3O_8 with aluminium powder, ρ_2 = 3.23 g/cm^3, C_2 = 0.179 cal/g^{o}C, $(\Delta T_2)_{max}$ = 660 ^{o}C $-\vartheta$. If the heat transfer from fuel plates to the moderator in case of a relatively slow excursion is taken into account, the following formula is derived

$$w_2 = C_2 \, \rho_2 \, (\Delta T_2)_{max} + \frac{V_c - V_u}{V_u} \, C_3 \, \rho_3 \, (\Delta T_3)_{max} \tag{1.21}$$

where $(\Delta T_3)_{max}$ = 100 ^{o}C $-\vartheta$, C_3 = specific heat, ρ_3 = density of H_2O; V_c = Volume of thermal core, V_u = volume of fuel plates.

The maximum ratio of fission energy densities $(s_1/s_2)_{max}$ released in an excursion was determined for the standard loading of STARK on the basis of the stationary fission rate distribution, Fig. 8; the variation of the intrinsic fission rate distribution with ω , as calculated in Section 3.5, was taken into account. Fig. 3 shows $(s_1/s_2)_{max}$ as a function of water temperature ϑ compared with the permissible value w_1/w_2 given by eq. (1.20) and (1.21). Obviously, the safety condition eq. (1.19) is fulfilled at moderator temperatures higher than $30^\circ C$, so that a fast core meltdown is incredible at operational temperatures ϑ = 60 to $80^\circ C$.

2. Description of the Reactor

2.1. Core Regions

The fast core of the assembly (37 cm average dia., 61 cm height, Fig. 4) is composed of 37 vertical drawers made of 0.1 cm stainless steel square tubes, 5.1 x 5.1 cm^2 inner dimensions (Fig. 5) which are filled with platelets consisting of the core materials under question. These drawers are supported by a 10 cm stainless steel grid plate located at the bottom of the reactor; they are easily accessible for loading and experiments from top of the reactor. One of the eccentric drawers is used as a safety rod (cf. Section 2.4) thus keeping the central core position free for particular experiments.

Usually, 5.08 x 5.08 x 0.31 cm^3 platelets made of U(20) metal, natural uranium metal, and moderating material (Al_2O_3 or graphite) are used as core components. These platelets are sufficiently thin so as to allow the average U^{235}-enrichment as well as the neutron spectrum in the core to be varied by changing the volume ratio of the core components. During the standard experiments, the following fast core composition is envisaged:

40 vol. percent U(20),
40 vol. percent natural uranium,
20 vol. percent Al_2O_3.

The fast core is radially enclosed in a 5 cm-thick octagonal natural uranium casing which is built up of smaller blocks well fitted to each other in order to prevent direct penetration of thermal neutrons into the fast core. Correspondingly, 8-cm-thick natural uranium axial reflectors are arranged at the upper and lower end of the core drawers. Additional shielding of thermal neutrons is accomplished by an 0.8-cm-thick boral sheet located below the bottom grid plate and by small cadmium plates fixed at the upper end-plugs of the drawers.

The inner part of the reactor is surrounded by a 5-cm-thick graphite annulus which fits into a cylindrical aluminum tank, 56.4 cm in diameter. The tank walls lead up to the turnable top plug of the biological shield (Fig. 4) with a Perbunane seal pressing against the plug in order to prevent moderator water, which might be pushed out of the thermal core during an excursion, to enter the fast core. In addition, Perbunane seals are fixed at the upper end plugs of the drawers which are pressing against each other thus forming a second water-tight system at the top surface of the fast core.

The annular Argonaut core region is not changed in its basic dimensions and structure. However, the number of fuel plates, each of them containing 124 g U_3O_8, 20 percent enriched, has to be increased by 25 percent in order to maintain criticality (cf. Section 3). For reasons of symmetry, an annular loading is arranged in the thermal region with each of the 24 fuel element positions being occupied by either 11 or 12 fuel plates at the inner part and corresponding graphite dummy plates at the outer part of the annulus.

The thermal core is enclosed in a separate annular aluminum tank with an annular top lid in order to prevent evaporation of water at elevated moderator temperatures. A large part of the lid surface consists of a thin aluminum foil which allows expulsion of moderator water in case of a reactor excursion.

2.2. Light Water System

Some changes in the light water system of the Argonaut are necessary as a consequence of the moderator temperature $\vartheta_{\!f}$ chosen as high as 80°C. A 40 kW electric heater is installed in the storage tank to heat the water content of the system to the operational temperature within 2 hours. For reduction of heat losses the whole system is clad with a heat insulation.

An additional interlock system is established which prevents the reactor from going critical at temperatures lower than a preset limiting value. Hence, a second water return line with a magnetic valve is mounted at half the core height. When the moderator is pumped up into the core, the water level rises to the height of this lower return line. Only when the core region is heated to the lower limiting temperature can the magnetic valve be closed and the water rise to full operational level. In this way, the reactivity effect of $\sim 0.7\%$ k, caused by variation of the core temperature, is compensated without using any control plates of the safety system.

The moderator temperature during operation is kept constant within $< \pm 1\,^{\circ}C$ by a temperature control system; an incidental decrease of water temperature below a limiting value actuates the dump valve and shuts the reactor down.

2.3. Air Cooling System

In order to avoid appreciable heat losses from the thermal core and a resulting slow temperature rise in the adjacent colder parts of the reactor, the aluminum walls of the annular tank are clad at the outside with a 1-cm-thick styrofoam heat insulation. There are additional 0.5-cm-thick air gaps extending to the adjacent surfaces of the internal tank and the external graphite reflector. Air is sucked through these gaps at a rate of 200 m^3/h so as to guarantee a well-defined temperature distribution in the fast core and the graphite reflector.

2.4. Control System

As pointed out in Section 1.2, there is no need for a high-speed control system acting directly upon the fast core. Therefore, the original Argonaut control units can be used and no changes in the neutron instrumentation are necessary. However, the total reactivity worth of the control system has to be increased by a factor of 2 to compensate for the reduced reactivity contribution of the thermal core.

Twelve Argonaut control units are regularly arranged around the annular tank in the external graphite reflector; larger cadmium plates (21 x 21 cm^2) are used in these units so that a total reactivity worth of at least 1.5 percent k is guaranteed for the first loading (cf. Section 3.4).

Six of these plates are used as safety units which must be drawn out before water is pumped up into the core; 3 plates can be raised only when operational temperature and water level are attained. Normally, the remaining three plates worth 0.4 percent k are used as fine control units.

For a further increase of shut-down reactivity, an additional safety rod is mounted in the fast core at an eccentric position with a driving mechanism below the bottom of the tank. The rod consists of a normal drawer tube filled with core material in its lower part and boron carbide in the upper part. Reactivity worth is 0.5 percent k (cf. Section 3.4).

2.5. Operational Procedure

When a new core configuration of STARK is established, a conventional multiplication experiment is done in order to confirm the properties of the critical system previously determined by theory. Among the various possible ways of loading a two-region reactor, a specified procedure is chosen so as to ensure that the effective neutron lifetime and the reactivity worth of control plates never drop during the loading process substantially below those of the proposed critical configuration. Hence, the following loading scheme is adopted:

First of all, the fast core region is completely loaded with "dummy" drawers which are filled with natural uranium metal. Then, Argonaut fuel plates are added to the thermal core up to about 80 percent of the critical number predicted by calculations. In this stage, the reactor is still far below critical.

Now, the natural uranium dummy drawers are replaced in turn by normal drawers having the proposed fuel composition, the reactivity worth of each drawer being fairly below ß (Section 3.4). After the fast core is completely loaded in this way, the reactor is made critical by adding stepwise more Argonaut fuel plates to the thermal core. If criticality cannot be reached in this way within $\pm$ 20 percent of the calculated number of fuel plates, the enrichment in the fast core has to be changed accordingly.

As a routine procedure with the first loading experiment, control plates are calibrated as a standard of reactivity by conventional rod-drop or period measurements. After this, the distribution of fission rate, the temperature and void coefficients as well as reactivity worths of thermal fuel plates and drawers will be measured.

The reactor will be normally operated at a total fission power level about 1 watt; maximum permissible power is 10 watts. Therefore, no serious difficulties due to fission product activities are involved in handling the core materials during loading and unloading the reactor.

3. Four-Group Calculations on the First Loading of STARK

Four-group diffusion therory calculations have been carried out (10) on various possible loadings of STARK in order to find out critical configurations which are consistent with the fundamental safety conditions stated in eq. (1.14), (1.16), and (1.19). In the following section, results are given for a specified configuration having a core composition quite similar to that of the standard loading of STARK proposed in Section 2.1.

3.1. General Assumptions

The geometrical structure of STARK is idealized as a system of 5 concentric annular regions neglecting the presence of aluminum tank walls and air gaps between the various parts of the actual reactor. The material composition of these regions is considered to be a homogeneous mixture of their components as listed, as a function of radial coordinate r, as follows:

1. <u>Fast core ($0 < r < 18.9$ cm):</u>

   7.57 vol. percent U-235
   65.53 vol. percent U-238
   26.90 vol. percent Al_2O_3

2. <u>Natural uranium zone (18.9 cm $< r <$ 23.9 cm):</u>

   0.72 vol. percent U-235
   99.28 vol. percent U-238

3. <u>Internal graphite annulus (23.9 cm $< r <$ 30.5 cm):</u>

   100 vol. percent C

4. <u>Thermal core (30.5 cm $< r <$ 46.0 cm):</u>

   15.40 vol. percent fuel plates containing U_3O_8, 20 percent enriched
   35.96 vol. percent H_2O
   48.64 vol. percent C

5. <u>External graphite reflector (46.0 cm $< r <$ 86.0 cm):</u>

   100 vol. percent C.

Within all these regions, a cosine axial flux distribution was imposed corresponding to an effective core height H = 70 cm (4.5 cm reflector savings assumed on either side).

The energy groups used in our calculations are given in Tab. I. Four-group effective cross sections were partly taken from Avery (8), partly they were computed from other cross section data assuming

a fission spectrum in the 1^{st} group,
a 1/E-spectrum in the 2^{nd} and 3^{rd} group,
a hardened Maxwellian spectrum (kT = 0.065 ev) in the 4^{th} group.

3.2. Flux Distribution and Fission Rate

Four-group neutron fluxes $\varphi_i(r)$ and adjoint functions $\phi_i^+(r)$ defined by

$$\varphi_i(r) = \int_{E_{i-1}}^{E} \phi(E,r) \quad dE; \qquad \phi_i^+(r) = \phi^+(E,r) \qquad (E_{i-1} < E < E_i) \qquad (3.1)$$

were obtained from a source iteration code which calculates simultaneously $\varphi_i(r)$ and the multiplication factor k of the system, depending essentially on χ_i = fission spectrum and Σ_{ai}, Σ_{fi} = effective absorption and fission cross section in group i. In this way k = 0.997 was found for the coupled system given in Section 3.1 indicating that the reactor is very close to critical. As a check of four-group parameters, additional calculations were made on the original Argonaut loaded with 4.5 kg U^{235}, 20 percent enriched; k = 1.006+ was obtained in very good agreement with experimental data (3).

The resulting flux distribution of STARK normalized to 10 watt total reactor power is shown in Fig. 6. In the external region of the thermal zone ($r \gtrsim 35$ cm), the flux distribution is very similar to that of the original Argonaut. In the inner region ($r \lesssim 35$ cm), the neutron spectrum changes continuously into a fast reactor spectrum, in the course of which the thermal flux incident from the thermal core is attenuated by orders of magnitude within the natural uranium zone and the surface layers of the fast core. Finally, an almost asymptotic shape of the fast spectrum is attained within a 20 cm diameter

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+For this calculation, in contrast to the others, a reflector savings of 12.0 cm on each side was used.

central part of the fast core. The radial distribution of the adjoint function $\phi_i^+(r)$ given in Fig. 7, on the other hand, shows only minor spatial variations.

Finally, the radial distribution of the fission rate density s(r) was determined from

$$s(r) = \sum_{i=1}^{4} \varphi_i(r) \quad \Sigma_{fi}(r) \quad ; \tag{3.2}$$

the resulting curve plotted in Fig. 8 shows a pronounced peak of the fission rate density at the surface of the fast core which is due to thermal neutrons incident from the thermal core. From Fig. 8, the relative contributions of the various reactor zones to the total reactor power are found as follows:

fast core	23.2 percent
natural uranium zone	11.8 percent
thermal core	65.0 percent

3.3. Integral Reactor Parameters

3.3.1. Parameters of the Entire System

Throughout this section the reactor is considered as a unity and no partition is made into fast and thermal regions.

a) Neutron lifetime: The average neutron lifetime l of the critical reactor is written in the case of four-group theory and cylindrical geometry

$$, \; l = \frac{\int_0^R r dr \sum_{i-1}^{4} \phi_i^+(r) \frac{1}{v_i} \varphi_i(r)}{\int_0^R r dr \sum_{j=1}^{4} \chi_j \phi_j^+(r) \sum_{i=1}^{4} \nu \Sigma_{fi}(r) \varphi_i(r)} \tag{3.3}$$

with effective reciprocal group velocities $1/v_i$ defined as flux-weighted averages

$$\frac{1}{v_i} = \int_{E_{i-1}}^{E_i} \phi(E)\,\frac{dE}{v(E)} \Bigg/ \int_{E_{i-1}}^{E} \phi(E)\,dE.$$

Using the $1/v_i$-values listed in Tab. I, eq. (3.3) leads to an average neutron lifetime

$$l = 7.89 \cdot 10^{-5}\ \text{sec} \qquad \text{for STARK}$$
$$l = 2.38 \cdot 10^{-4}\ \text{sec} \qquad \text{for Argonaut.}$$

The latter value is in good agreement with the experimental result $l = 2.5 \cdot 10^{-4}$ sec (3) indicating that the kT-value of the thermal spectrum was properly chosen. Obviously, the neutron lifetime of STARK is smaller than that of the Argonaut by a factor of three,

$$l_{\text{STARK}} / l_{\text{Arg.}} = 0.331,$$

due to the decrease of thermal flux in the fast core and the additional fission processes occurring there.

b) <u>Effective delayed neutron fractions:</u> The effective fractions β_k^{eff} of the various delayed neutron groups are given as averages of β_k^{25} and β_k^{28}, the delayed neutron fractions of the fissionable isotopes under question. Since all delayed neutrons are born in group 2, we have

$$\beta_k^{eff} = \frac{\int_0^R r\,dr \sum_{i=1}^{4} \phi_2^+(r) \left\{ \beta_k^{28} \nu_i^{28} \Sigma_{fi}^{28}(r) + \beta_k^{25} \nu_i \ldots \right\} \varphi}{\int_0^R r\,dr \sum_{j=1}^{4} \chi_j \phi_j^+(r) \sum_{i=1}^{4} \nu\,\Sigma_{fi}(r)\,\varphi_i(r)} \qquad (3.4)$$

Inserting Keepin's data (9) into eq. (3.4), the following effective parameters are obtained:

group:	β_k^{eff}
k = 1	$2.474 \cdot 10^{-4}$
2	$1.500 \cdot 10^{-3}$
3	$1.398 \cdot 10^{-3}$
4	$3.104 \cdot 10^{-3}$
5	$1.160 \cdot 10^{-3}$
6	$2.891 \cdot 10^{-4}$

Hence, the total delayed neutron fraction is found to be $\beta^{eff} = 7.7 \cdot 10^{-3}$.

3.3.2. Parameters of the coupled System

For a more refined description of STARK we employ the integral theory of Avery (1). The reactor is divided into two regions: the fast region (index 1) consisting of the fast core and the surrounding natural uranium casing, and the thermal region (index 2) comprising the remaining outer part of the reactor.

For determination of the coupling constants k_{ij} and neutron lifetimes l_{ij} defined in eq. (1.3) and (1.4), the partial fluxes and adjoint functions $\varphi_{ki}(r)$ and $\phi^+_{ki}(r)$ were calculated in a single iteration assuming fission neutron sources in region k given by the flux distribution $\varphi_i(r)$ of the critical reactor, while fission neutron sources are put to zero in the other region.

From eq. (1.3) and (1.4) written in four-group notation, the following parameters wereobtained for STARK (10):

$k_{11} = 0.7610$	$l_{11} = 7.093 \cdot 10^{-6}$ sec
$k_{12} = 0.1316$	$l_{12} = 6.755 \cdot 10^{-5}$ sec
$k_{21} = 0.3023$	$l_{21} = 1.012 \cdot 10^{-4}$ sec
$k_{22} = 0.8330$	$l_{22} = 1.226 \cdot 10^{-4}$ sec ,

and coefficients of reactivity partition given by eq. (1.8) were found as

$$\alpha_1 = 0.411 \qquad\qquad \alpha_2 = 0.589 \; .$$

The thermal core neutron lifetime ($l_{22} = 1.226 \cdot 10^{-4}$ sec) is seen to be appreciably smaller than that of the thermal Argonaut, due to the increased number of fuel plates and greater leakage losses. The effective neutron lifetimes in the fast region, l_{11} and l_{12}, on the other hand, are greater than that of neutrons causing fast fission since thermal fission processes occur in the natural uranium casing. According to eq. (1.12), the overall lifetime $l = 7.89 \cdot 10^{-5}$ sec was found from k_{ij} and l_{ij}, which is in good agreement with the value obtained in the previous section.

3.4. Reactivity Contributions

The change of reactivity due to insertion of some typical materials into the fast core of STARK was calculated by a first order perturbation theory. The reactivity change $\Delta\rho$ in question is additively composed of various parts according to the variations of the individual cross sections, $\delta\Sigma_a$, $\delta\Sigma_f$, $\delta\Sigma_{i\to j}$, and the diffusion coefficient, δD, within the region where the sample is located. Hence, we have

$$\Delta\rho = \delta\rho_a + \delta\rho_f + \delta\rho_{in} + \delta\rho_D . \tag{3.5}$$

If a sample of a constant geometrical cross section f is completely inserted at $r = r_o$ parallel to the axis of the cylindrical reactor, we obtain a four-group notation

$$\delta\rho_a = -\frac{f}{F}\sum_{i=1}^{4} \delta\Sigma_{ai}\, \phi_i^+(r_o)\, \varphi_i(r_o),$$

$$\delta\rho_f = +\frac{f}{F}\sum_{j=1}^{4} \chi_j \phi_j^+(r_o) \sum_{i=1}^{4} \nu_i\, \delta\Sigma_{fi}\, \varphi_i(r_o)$$

$$\delta\rho_{in} = + \frac{f}{F} \sum_{j=1}^{4} \sum_{i=1}^{4} \left\{\phi_j^+(r_o) - \phi_i^+(r_o)\right\} \delta\Sigma_{i\rightarrow j} \varphi_i(r_o), \qquad (3.6)$$

$$\delta\rho_D = - \frac{f}{F} \sum_{i=1}^{4} \delta D \left(\frac{\partial\phi_i^+}{\partial r}\right)_{r_o} \left(\frac{\partial\varphi_i}{\partial r}\right)_{r_o},$$

where $$F = 2\pi \int_0^R r dr \sum_{j=1}^{4} \chi_j \phi_j^+(r) \sum_{i=1}^{4} \nu_i \Sigma_{fi} \varphi_i(r)$$

By means of these equations the reactivity change was calculated for the following cases:

1. Insertion of a drawer filled with natural uranium in exchange for a normal core drawer.

2. Insertion of a drawer filled with boron (density 2.5 g/cm^3) in exchange for a normal core drawer.

3. Insertion of a drawer filled with 20 percent enriched uranium metal in exchange for a normal core drawer.

4. Insertion of a drawer filled with H_2O in exchange for a normal core drawer.

5. Withdrawal of a normal core drawer without leakage effect taken into account.

The contributions of the various parts of $\Delta\rho$ are listed in Tab. II.

In the same way, the reactivity worth of the 12 cadmium control plates of STARK was determined relative to the effect of the same plates on the thermal Argonaut; with the location of plates at r_o = 50 cm we get

$$\frac{\Delta\rho_{STARK}}{\Delta\rho_{Arg.}} \approx \frac{(\phi_4^+(r_o)\ \varphi_4(r_o)\ /\ F)_{STARK}}{(\phi_4^+(r_o)\varphi_4(r_o)\ /\ F)_{Arg.}} \quad . \qquad (3.7)$$

Thus, from the experimental value $\Delta\rho_{Arg.} > 3.5$ percent k obtained for the thermal Argonaut, a minimum reactivity worth $\Delta\rho_{STARK} = 1.5$ percent k is derived for STARK.

3.5. General Time-Behavior of STARK

The time-behavior of neutron flux in the case of an excursion of STARK was investigated on the basis of Avery's integral theory using the coupling parameters k_{ij} and l_{ij} of the critical system as found in Section 3.3. The reactor behaviour is discussed in terms of the inverse stable period ω of the initial power rise resulting from a step variation of reactivity, .

In principle, there are many possible ways of changing the reactivity ρ of a coupled system by a proper change of the individual k_{ij}; hereof, four characteristic cases are considered:

a) Addition of reactivity to the fast region brought about by a corresponding variation of ν_1, the number of fission neutrons in that region. Hence, we have according to eq. (1.6)

$$k'_{11} = k_{11}(1 + \varepsilon_1) \quad , \quad k'_{21} = k_{21}(1 + \varepsilon_1)$$

b) Addition of reactivity to the thermal region in the analogous way:

$$k'_{22} = k_{22}(1 + \varepsilon_2) \quad , \quad k'_{12} = k_{12}(1 + \varepsilon_2)$$

c) Variation of the coupling parameter k_{12}:

$$k'_{12} = k_{12}(1 + \varepsilon_{12})$$

d) Variation of the coupling parameter k_{21}:

$$k'_{21} = k_{21}(1 + \varepsilon_{21})$$

For these cases, ε_i, ε_{ij}, and $\Delta\varsigma$ were calculated as a function of ω by means of eq. (1.7) and (1.10).

Fig. 9 gives a plot of $\Delta\varsigma$ versus ω for these cases in comparison with the $\Delta\varsigma$ obtained from the usual inhour equation using $l = 0.78 \cdot 10^{-4}$ sec. For small reactivities, $\Delta < 2$ percent k, ω is a definite function of $\Delta\varsigma$ which is well represented by the simple inhour equation. At larger reactivities, however, ω proves to be dependent on the individual k_{ij} which is a consequence of the variation of the flux distribution and the effective lifetime l.

The variation of the ratio S_1/S_2 of total fission rates with ω was calculated from the formula

$$\frac{S_1}{S_2} = \frac{1 + \omega l_{21}}{1 + \omega l_{22}} \; \frac{1 - k'_{22} + \omega l_{22} + \omega k'_{22} \sum_k \frac{\beta_k}{\omega + \lambda_k}}{k_{21}\left(1 - \omega \sum_k \frac{\beta_k}{\omega + \lambda_k}\right)} \tag{3.8}$$

which is easily derived from the kinetic equations (1.9). According to Fig. 10, there is a slight increase of S_1/S_2 with ω in the cases a) and c), while S_1/S_2 is almost constant in the other cases.

This calculation shows, that variations of the flux or fission rate distribution are only important for extremely short periods, according to the rather strong coupling between both core regions of STARK.

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Table I: Definition of Neutron Energy Groups:

Group	Energy Interval	Reciprocal Group velocity v_i^{-1}
1	$E > 1.353$ Mev	$4.1 \cdot 10^{-10}$ sec/cm
2	9.12 kev $< E <$ 1.353 Mev	$2.0 \cdot 10^{-9}$ sec/cm
3	0.4 ev $< E <$ 9.12 kev	$2.3 \cdot 10^{-7}$ sec/cm
4	$E < 0.4$ ev	$2.52 \cdot 10^{-6}$ sec/cm

Table II: Reactivity Change $\Delta\varrho$ in Percent k Caused by Replacement of Core Material by Other Substances:

	1	2	3	4	5
Material	Nat. Uranium	Boron	Oy(20)	H_2O	Void
r	0 cm	10.2 cm	0 cm	0 cm	0 cm
$\delta\varrho_a$	+ 0.092	- 0.209	- 0.673	+ 0.482	+ 0.487
$\delta\varrho_f$	- 0.409	- 0.646	+ 0.925	- 0.715	- 0.715
$\delta\varrho_D$	o	- 0.002	0	0	0
$\delta\varrho_{in}$	0.038	+ 0.214	- 0.032	+ 1.770	+ 0.076
$\Delta\varrho$	- 0.355	- 0.643	+ 0.220	+ 1.537	- 0.152

Fig. 1.

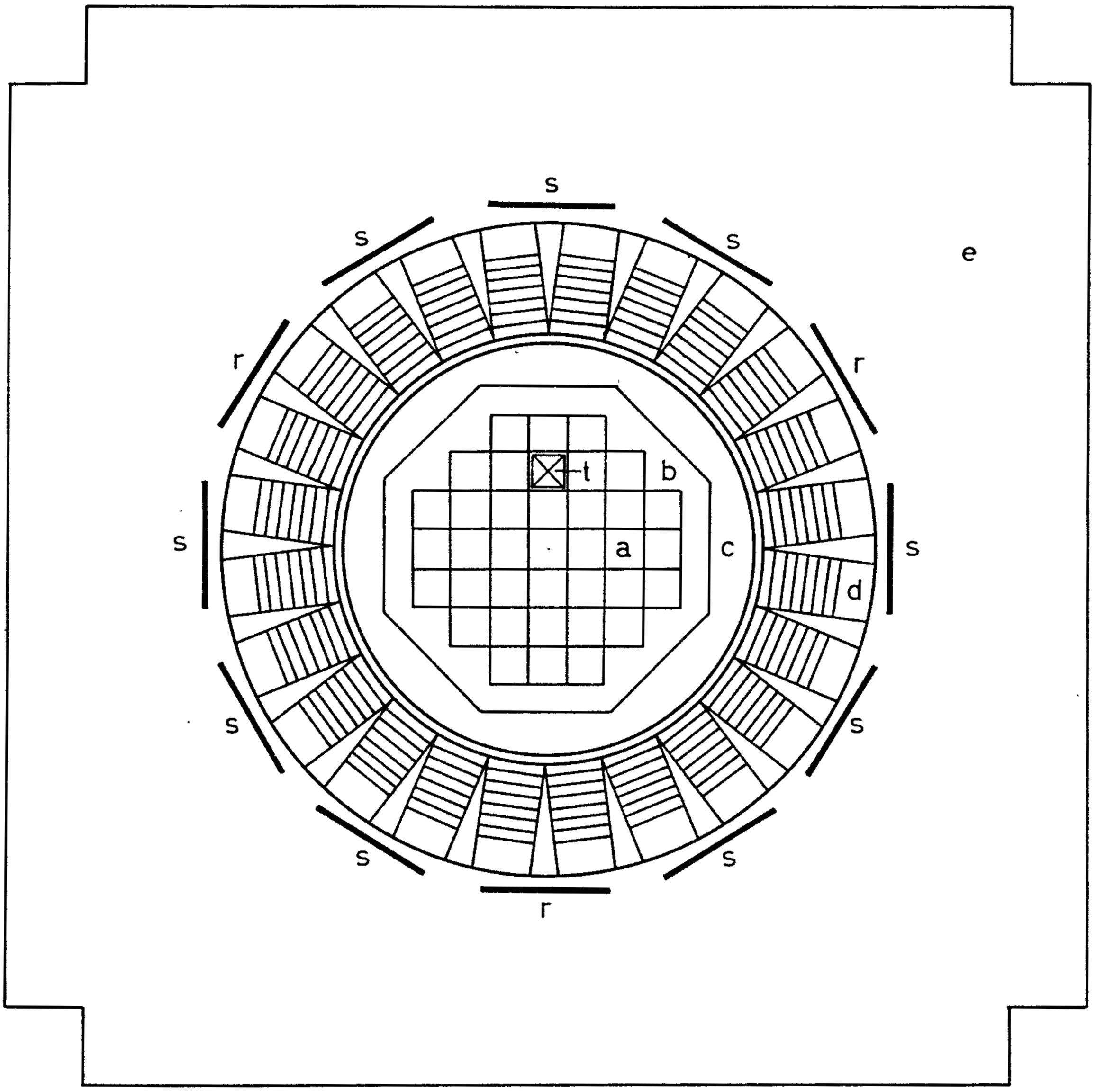


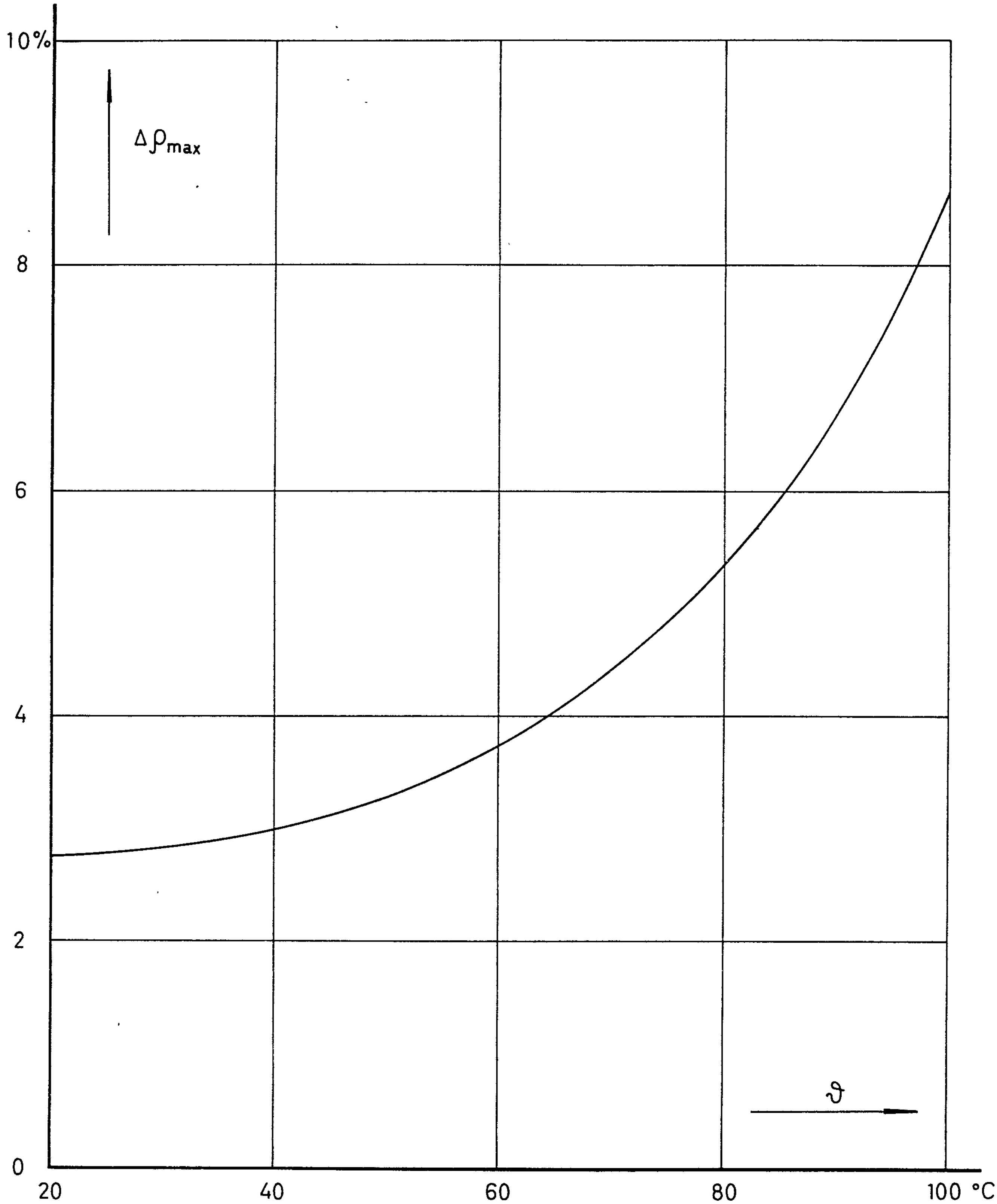
Fig. 1. Schematic cross section of STARK.

a- fast core, b- natural uranium casing, c- graphite region,

d- thermal Argonaut core, e- external graphite reflèctor,

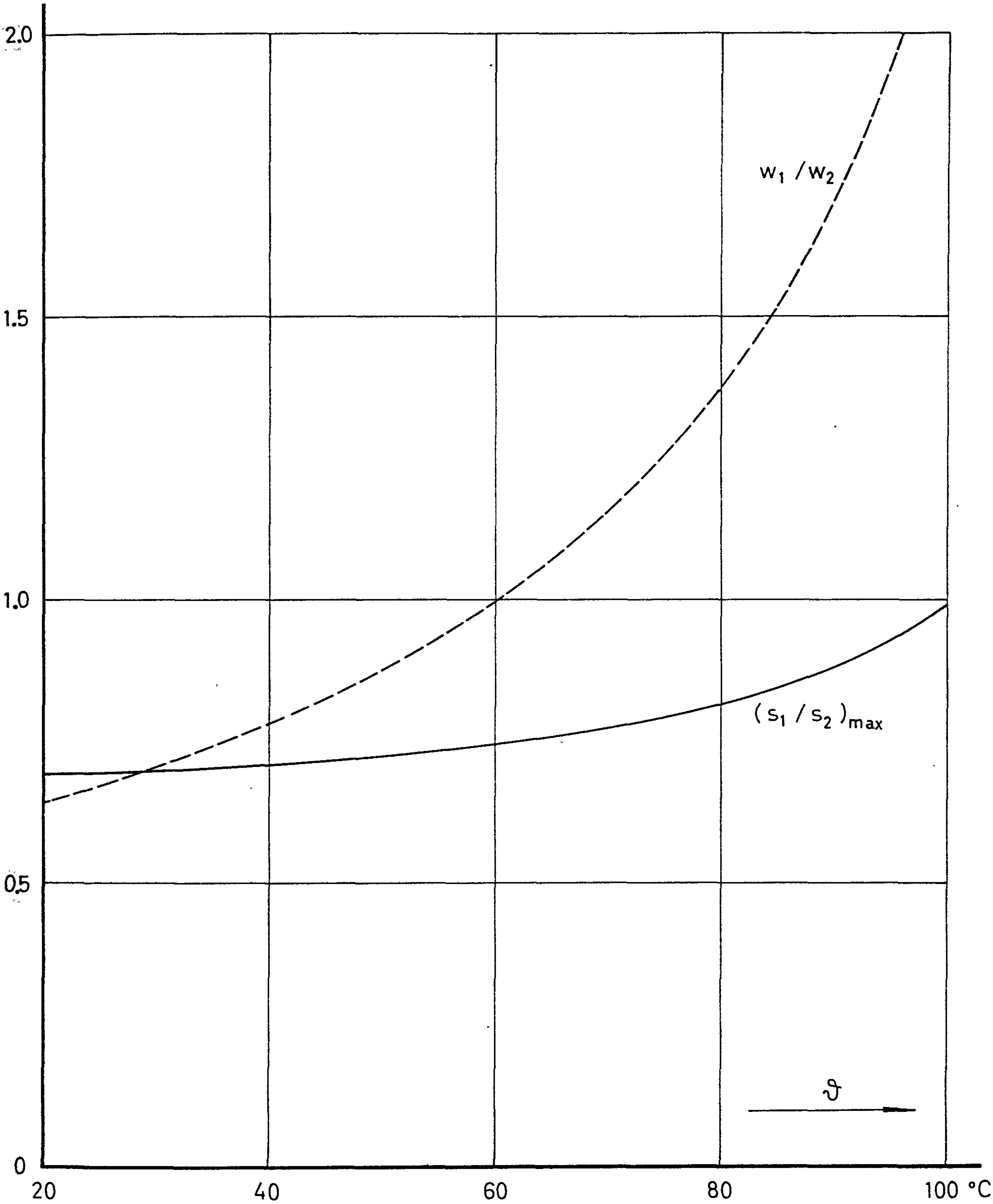
r- fine control plates, s- safety plates, t- safety rod of fast core.

Fig. 2.



Maximum step reactivity $\Delta\rho_{max}$ compensated in a self-limiting excursion without melting of fuel plates; ϑ = moderator temperature.

Fig. 3.



Maximum ratio of fission energy density $(s_1/s_2)_{max}$ released in fuel compared with permissible value w_1/w_2; ϑ = moderator temperature.

Fig. 4. Core regions of STARK

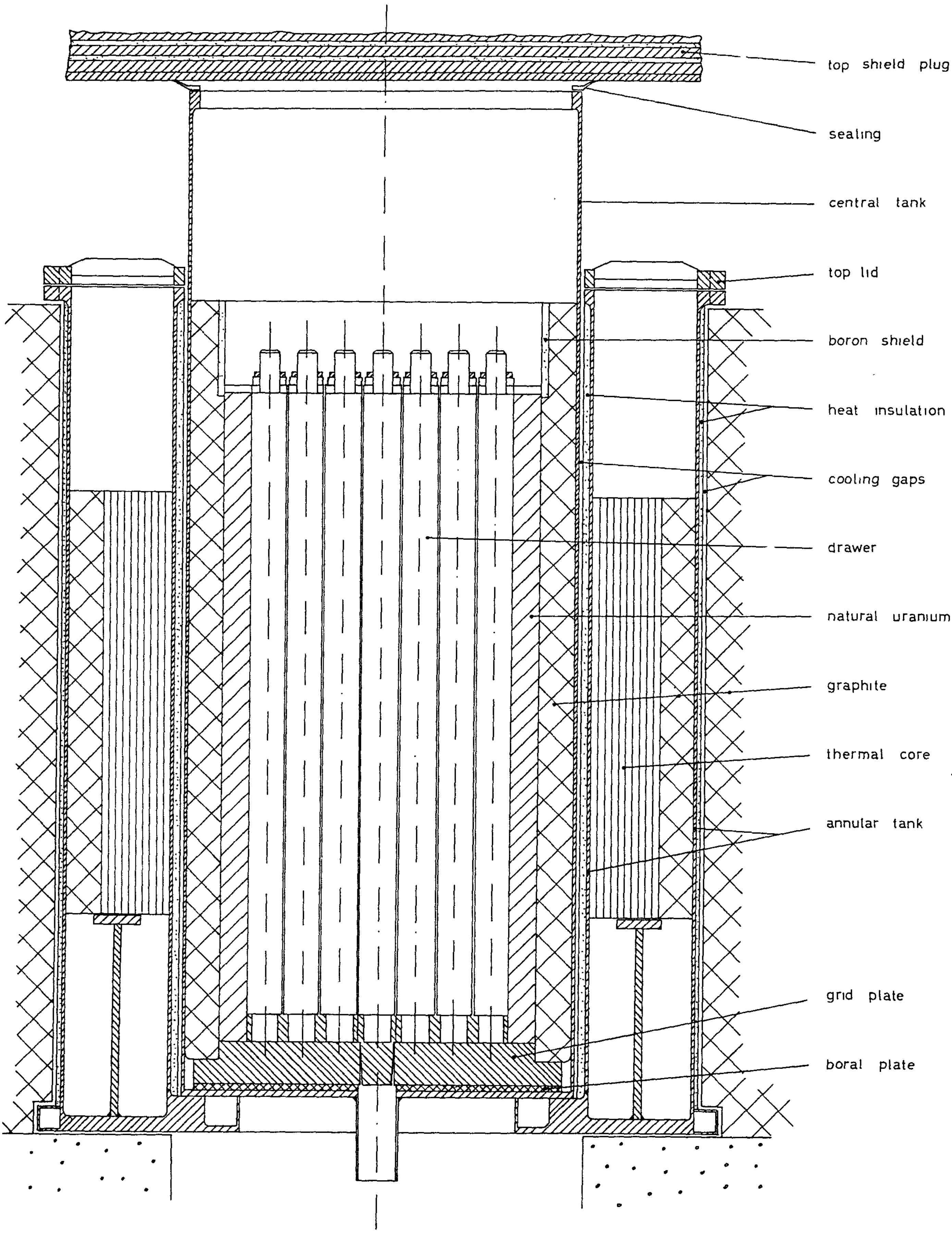


Fig. 5. Cross section of a fast-core drawer

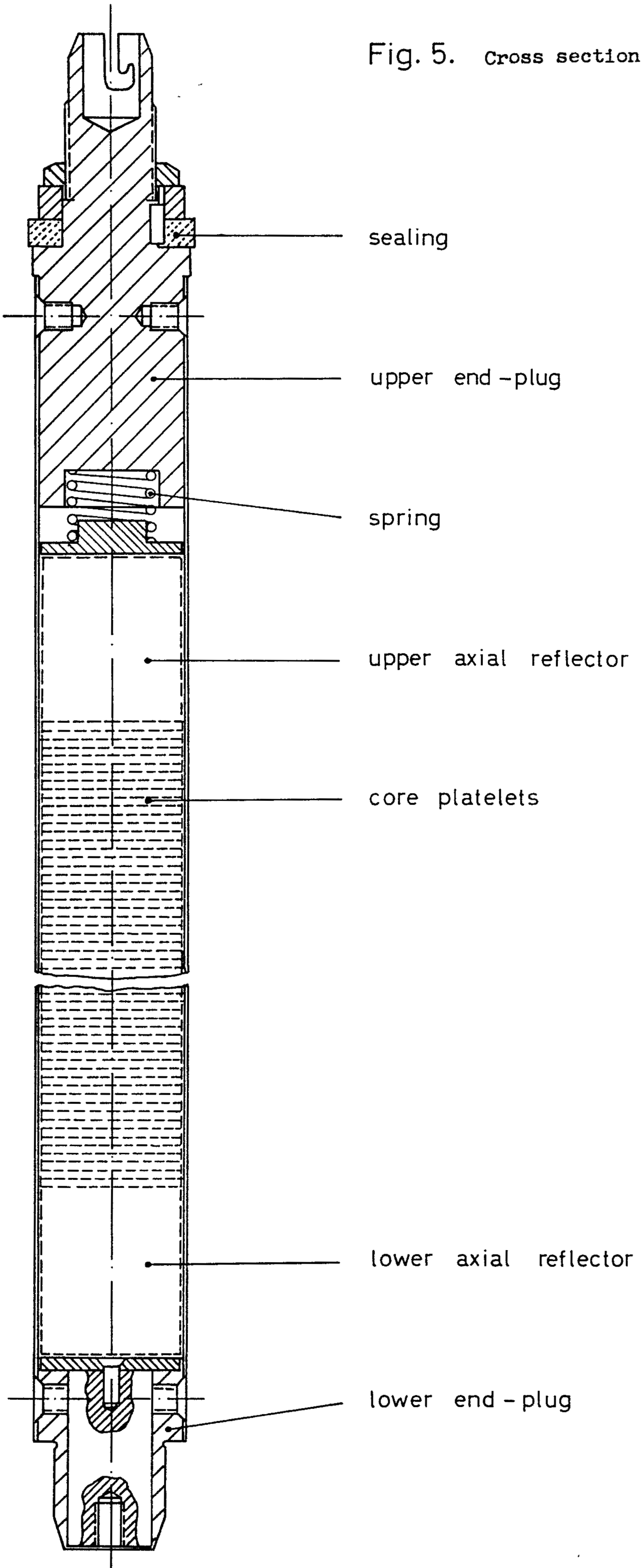


Fig. 6. Radial flux distribution of STARK, $\varphi_i(r)$, at lo Watts total reactor power

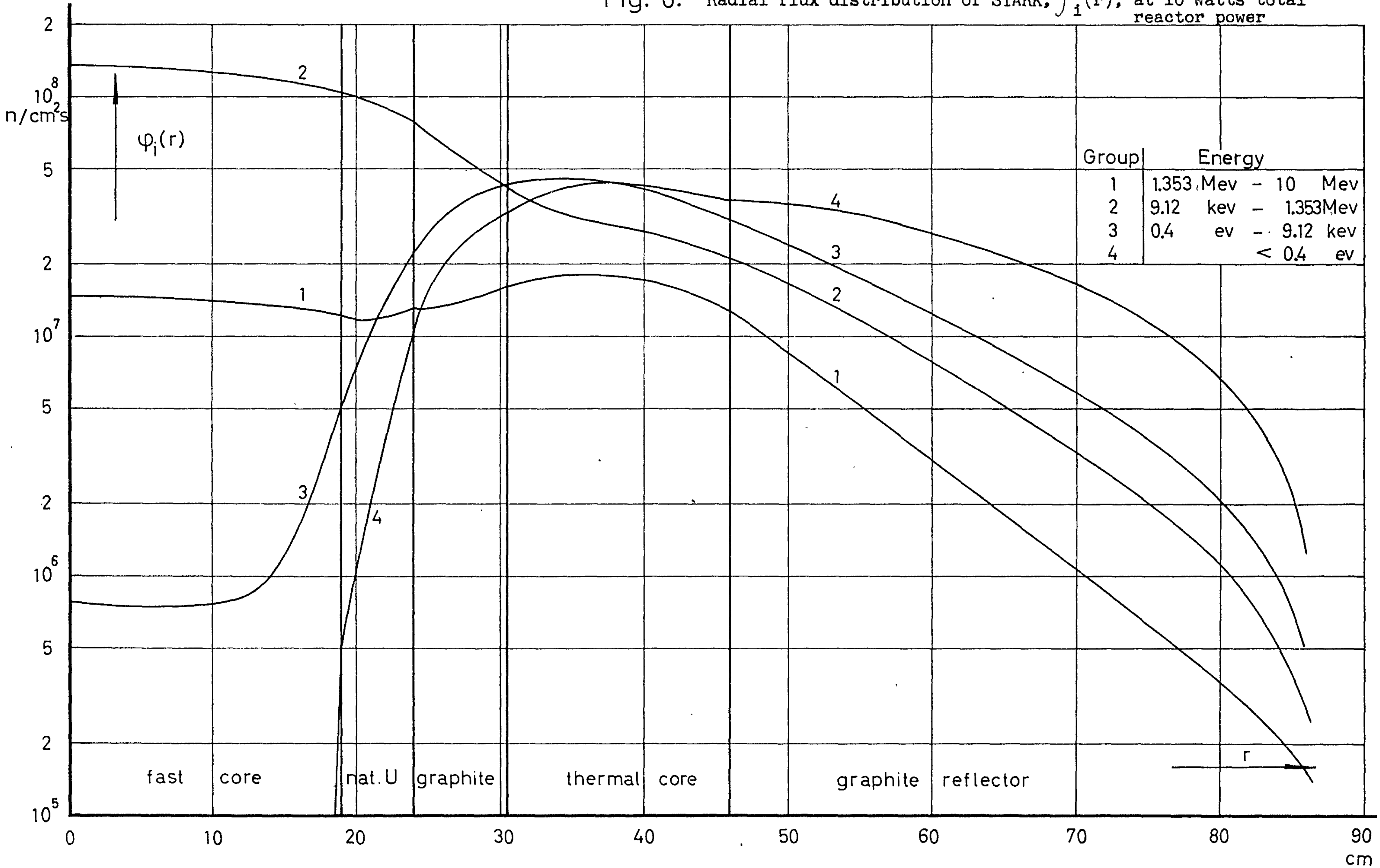


Fig. 7. Radial distribution of adjoint function, $\phi_i^+(r)$

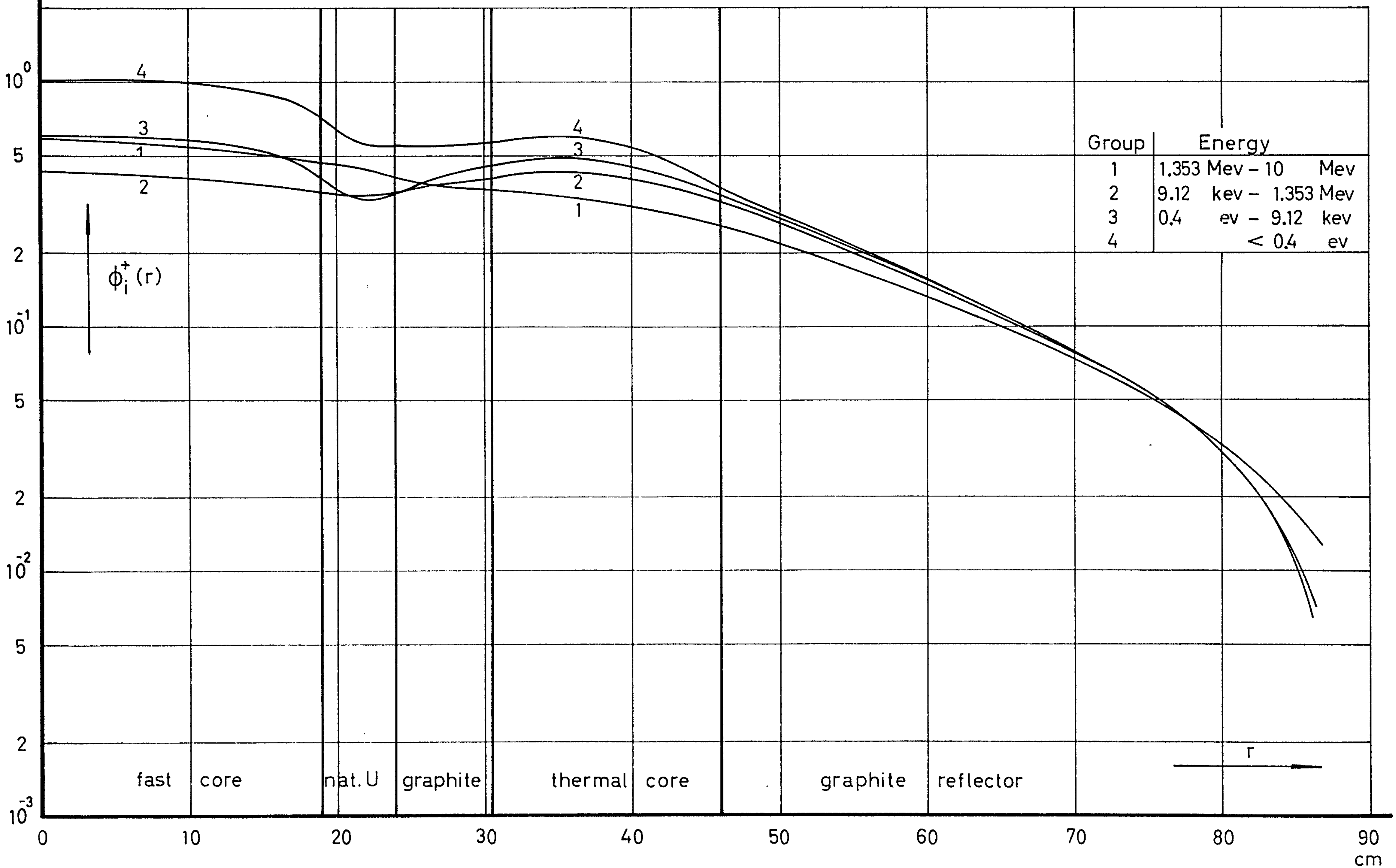
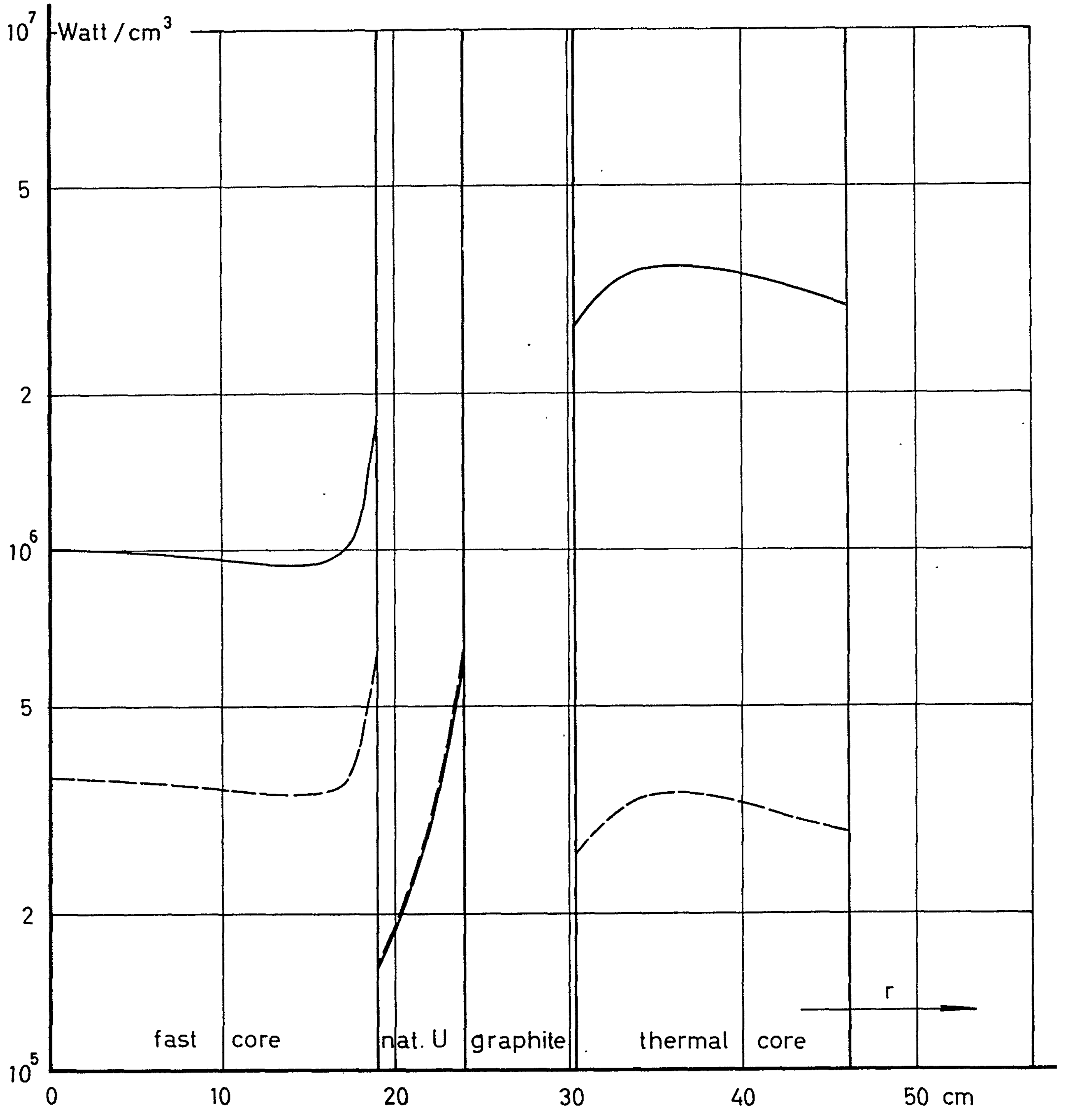


Fig. 8.



Radial distribution of fission rates in STARK at lo Watts total reactor power

———————fission rate per unit volume of fuel,

---------fission rate per unit volume of homogenized core

Fig. 9. Reactivity $\Delta\rho$ as a function of inverse period ω.
——— obtained for the coupled system using eq. (1.10), cases a) to d) of Section 3.5,
- - - - obtained from inhour equation (1.11)

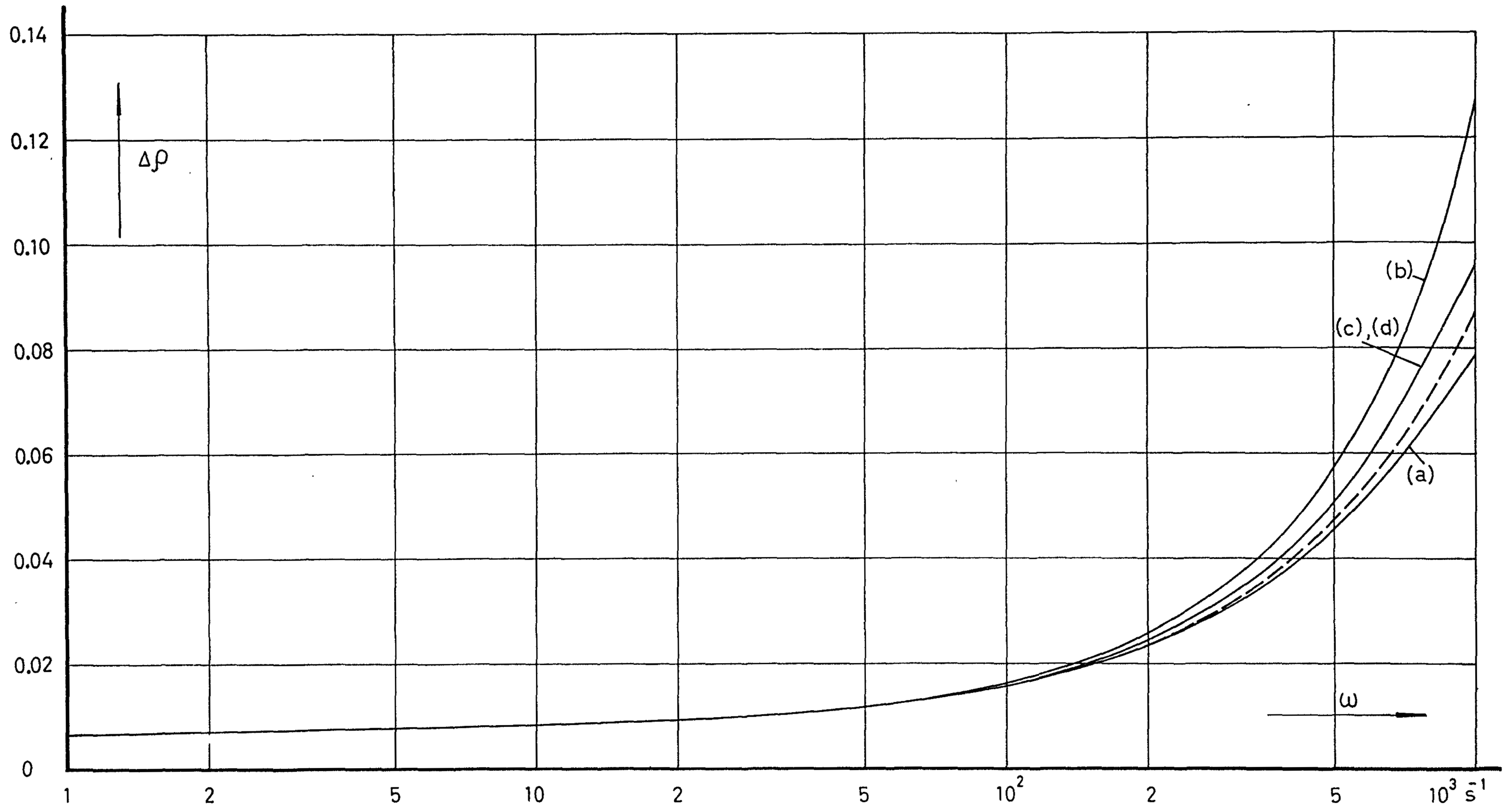


Fig. 10. Ratio of fission rates S_1/S_2 versus inverse period ω according to eq. (3.8), cases a) to d) of Section 3.5