ZJE - 147 1976

PROGRAM OF IN-CORE MEASUREMENTS
Part 4

MEASUREMENT OF RADIATION-INDUCED HEATING OF URANIUM IN FUEL ASSEMBLIES IN THE A 1 REACTOR UNDER OPERATION

J. Kott, V. Jiroušek, J. Němec, S. Teren, K. Černý

ŠKODA WORKS
Nuclear Power Construction Department, Information Centre
PLZEŇ - CZECHOSLOVAKIA

[illegible] OF IN-CO[illegible]

Part 4

MEASUREMENT OF RADIATION-INDUCED HEATING OF URANIUM IN FUEL ASSEMBLIES IN THE A1 REACTOR UNDER OPERATION

J. [illegible]ott, V. Jircušek, J. Němec, S. Teren, K. Černý

ŠKODA WORKS
Nuclear Power Construction Department, Information Centre
PLZEŇ, CZECHOSLOVAKIA

ABSTRACT

The article describes a special fuel assembly (type R-1, dia 112 mm) for the reactor of the 1st Czechoslovak nuclear power station, with built-in in-core instrumentation involving thermal detectors for measuring radiation-induced heating of uranium wherefrom the neutron flux may be determined. In addition to theoretical background the article presents some more particular data concerning the measurement in itself, the technique of processing the data obtained, as well as the results further used for thermal analysis and evaluation of reactor operational safety.

1. INTRODUCTION

The problems associated with operational in-core measurement of temperature fields and spacial distribution of neutron flux in the reactor of the 1st Czechoslovak nuclear power station have been solved in the framework of a so called "Program of in-core measurements", the overall technical philosophy of which has been described in ref. /1/. Verifying experiments of reactor radiation thermal detectors have been carried out on the R1 reactor (Yugoslavia), ref. /2/, and on Czechoslovak VVR-S reactor /3/. The results of measurements of thermal and technical parameters performed with special fuel assemblies (type 1-3) in the reactor A1 are presented in ref. /4/.

The report presented hereby represents a part of research and developmental effort /1, 3, 4/ accomplished with the aim of ensuring A1-reactor in-core measurements, and attaches importance to determining the generation of heat and therefrom derived neutron fields in reactor cores using thermometric technique.

2. DISCUSSION ON THE CHOICE OF THE TECHNIQUE OF MEASUREMENT

In the course of formulating the philosophy of in-core measurements insufficient experience with detectors for continual measurement of reactor radiation was available. After an overall evaluation of the possibility of a unified outlet of obtained information through the after-cooling zone where the fuel assembly is located by means of a suspension rod, generally two techniques have been adopted, i.e. the thermometric detection technique, and the beta-emission technique based on using Soviet beta-emission detectors of neutron flux /5/.

The thermometric detection technique is a straightforward method based on calorimetric principles. It should be added that this is valid under the understanding that the detector consists of a similar fissile material as the surrounding fuel rods in the assembly. The detector is then subjected to the same effect of the field of

Fig. 4. Distribution of thermal neutron flux in the cell H-09.

reactor radiation as [illegible], especially from the viewpoint of spectral distribution [illegible], or generally, from the viewpoint of the isotopic change of natural uranium derived therefrom.

In spite of the fact that the results of experimental research and development of reactor calorimeters with non-fissile detectors have been available, it has been necessary to develop theoretically and technologically a new type of detector with fissile materials in radial arrangement. Being it necessary [illegible] placing relatively great calorimeters into a rod-type fuel assembly /1/; nowadays the thermometric detector is miniaturized. Dimensional analysis has been originally limited by the following criteria:

- it has been necessary that the diameter of both the fuel rod and the detector were the same;
- minimum disturbance of the standard composition of fuel rods in the assembly;
- the maximum possible concordance in the operational temperatures of both the detector and the rod of natural uranium;
- not to exceed the limiting value of 600 °C of detector operating temperature;
- it has been necessary to perform the design in such a way as to enable assembly operations, which are rather difficult (mounting of calibrating heating spiral and mounting of sheathed thermocouples);
- it has been necessary that the calorimeter body were gas-tight.

Total number of detectors (the number was 5) in one fuel assembly of the type [illegible]-1 has been given by the number of thermocouples which could be led out through a special connector out of the reactor pressure vessel. Owing to the temperature conditions /5/ it has been necessary to use between the detector and the sheath a heat transmitting layer consisting of helium. During a long-term operation of the calorimetric detector it has been proved that if the detector is properly designed and made, then the helium layer may be suitable also for long-term (several months) [illegible] temperatures, pressures, and under the presence of radiation. For this reason the [illegible] of [illegible] played such an important role. Retrospectively, the results are satisfactory, because only one of [illegible] calorimeters operated in A1 reactor core suffered from loss of tightness, which resulted in a marked worsening of the transfer of heat.

2. [illegible]

[illegible] by reactor [illegible], [illegible] heating is caused mainly by [illegible] fission products and by various non-fissile interactions.

[illegible] non-fissile [illegible] can be expressed by means of the following formula:

$$H = C_K \sum_{i=1}^{m} \int_0^{\infty} \mu_i'(E) E B_i'(E) \varphi_\gamma(E) \cdot G[\mu'(E)R, E] B[\sigma_i'(E), \lambda_i', R] \, dE +$$

$$+ C_K \sum_{i=1}^{n} \frac{2 A_i N_i}{(A_i + 1)^2} \left[1 - \overline{\cos \vartheta}\right] \int_0^{\infty} \sigma_{si}'(E) \varphi_n(E) E G[\Sigma'(E)R, E] \, dE \qquad (1)$$

Equation /1/ neglects the influence of inelastic scattering, threshold reactions, radiative capture and non-fissile reactions. As has already been shown /Ref. 1./, by means of "clean" photon fields it is possible to measure the dose rates in reference materials with an accuracy much better than 10 %. Measuring carried out on several suitable reference detectors (where the absolute value of dose rate of fast neutrons is neglectable when compared with the overall error of the measurement) enables to determine the characteristic values of the most important low-energy component of the photon spectrum. In this way it is possible to obtain sufficient volume of numerical data concerning the photon field in the reactor, therefore [illegible] radiation-induced [illegible] in any material. Concerning [illegible] heating, the [illegible], it may be [illegible] in the following terms:

$$H_n = [illegible] - H_f$$

The radiation-induced heating caused by neutrons and photons may be also expected in fissile materials. Concerning fission, which [illegible] source of thermal energy in nuclear reactors, it is possible to write (for this volume of fissile material, e.g. natural uranium)

$$H_f = C_h Q^* \left[\int_0^{\infty} \sigma_f^s(E) N^s \varphi_n(E)\, dE + \int_{E}^{\infty} \sigma_f^e(E) N^e \varphi_n(E)\, dE \right] \qquad ([illegible])$$

After averaging the constants and coefficients one gets directly

$$H_f = \text{const} \cdot \varphi_n$$

which will be used in calculating the nuclear fission.

4. [illegible] -2000 [illegible]

Experimental assembly [illegible] yielded 24 pieces of information, wherefrom 20 concerned temperatures of calorimetrical samples and sheathes, inlet and outlet temperature of the gas, inlet gas pressure, the pressure drop on the [illegible] of the assembly. Because the measuring lines contained a thermostat maintaining the temperature of cold junctions of the thermocouples on 50 °C, a value of [illegible] has been added to the values of temperature. This has been also taken into consideration in the program for processing the data so obtained on the Hewlett-Packard calculator. The measurement of pressure and pressure drop has been intended to serve as a part of starting data for determining the throughflow of gas. On the basis of gas throughflow and the difference between the inlet and outlet enthalpy it has been possible to obtain the thermal output of the assembly (because the losses of heat into environment have been neglected). The thermal output of the experimental assembly served for comparison with another quantity obtained from an evaluation of measured thermal loading of uranium in the calorimeters.

5. EVALUATION OF DATA OBTAINED BY MEASUREMENT

The whole evaluation has been carried out by means of the calculator Hewlett-Packard 9100 B; in elaborating the computing programs it has been assumed that the accuracy of measurement ranges within ± 10 %, which is quite consistent with conditions existing on a power plant.

In computing the neutron fluxes, the following data of the detector have been used:

a) detector diameter 8 mm, therefrom the diameter of uranium core accounts for 6 mm;

b) average mass of aluminum in the sample: 7.054 g;

c) average mass of uranium in the sample: 1.335 g.

The amount of energy released in 1 second in 1 g of natural uranium is as follows

$$H_f = 3.323 \cdot 10^{-14} \varphi_n, \qquad (3)$$

self-shielding effect in the sample has been neglected, because voluminal concentration of uranium in the detector is only 5 % (basic material is aluminum), so that total energy released in conformity with the relations presented in chapter 3, may be expressed in the following manner:

$$H_{tot} = 1.335 \times 10^{-3} H_f + N_t \cdot Q \qquad (4)$$

Combining equations (3) and (4) yields an equation expressing the neutron flux

$$\varphi_n = 2.250 \cdot 10^{16} (H^{tot} - N_t \cdot Q) \qquad (5)$$

adiation-induced heating of non-fissile character has been determined for an effective energy of the photon field (in the A1 reactor) of 0.80 . 10^{-13} J by means of the results of the measurement carried out on the heavy water reactor RA in Yugoslavia /7/. In doing so, the ratio of photon field rates between the reactor A1 and reactor RA has been determined as $\phi_{A1} = 0.1274\ \phi_{RA}$. Furthermore, the values of $H_{\gamma}(Z)$ have been determined for individual positions of the calorimeters (assuming cosine distribution of the photon field along core height). These values have been also determined in ref. /8/. The first phases of measurement performed on A1 reactor have demonstrated that the distribution of neutron flux along core height is not cosine, so that the distribution of heat sources differed also from cosine distribution. As a consequence, the values of H_{γ} introduced in certain cases greater errors. From this reason further procedure consisted in selecting those measurements where the distribution of neutron flux along the height was almost cosine, and in averaging the ratios of detector heating due to fission and photon field. This enabled to find that the share of the photon heating accounts for 28.6 % of the total heating. On this basis equation (5) has been modified into the following shape:

$$\varphi_n = 1.606\ .\ 10^{13}\ H_{tot} \qquad (6)$$

The difficulties encountered in determining the value of H_n may be removed by building-in one calorimeter with non-fissile, photon-insensitive detector.

Measured values of heating induced by radiation in the detector have been taken advantage of in determining the value for the whole section of the assembly. This has been accomplished from the theoretically found radial distribution of the neutron flux in the assembly, with the influence of the calorimeter being properly taken into account /9/. Relative radial distribution of the rates of heating in the experimental assembly is given in table 1.

The calibration of technical detectors is described at full length [illegible].

Table 1

Radial position in the assembly	φ_i / φ_c
Calorimeter	[illegible]
2nd row of rods	[illegible].[illegible]0[illegible]9
3rd row of rods	[illegible].[illegible]9
4th row of rods	[illegible].[illegible]27
5th row of rods, φ_c	1.0000
average value in the assembly	[illegible].[illegible]0

5. C[illegible]
[illegible]

By the experimental assembly [illegible] type [illegible] possible to evaluate [illegible] sources in medium. By [illegible] N - 3 it is [illegible] possible to [illegible] temperatures [illegible] and radius. Thus, [illegible]. This may be e.g. accomplished by comparing the temperature distribution in the fuel [illegible] in the N - 3 assembly (obtained calculationally on the basis of found course of the distribution [illegible] within the framework of the experiment with N - 1 assembly), with the temperature distribution obtained by measurement. Moreover, the output of the N - 1 assembly may be compared with that of the N - 3 assembly obtained on the basis of temperature distribution and also on the basis of the balance of heat energy of the cooling gas. The results have been evaluated taking into consideration that the assemblies N - 1 (in the cell N-09) and N - 3 (in the cell N-08) have not been operated in identical conditions. Therefore the operational mode of both assemblies has been adjusted in such a way as to obtain the same outlet temperature of gas. The results of the measurement have been further compared for operational modes preceded by sufficiently long period without overcompensating the core and without any considerable change of the output. More particular data are given in table 2.

Table 2

Day of measurement	Hour of measurement	$\cdot 10^{17}$ $/m^{-2}s^{-1}/$	$N_\gamma \cdot 10^3$ $/Wkg^{-1}/$	N_E-1 /kW/	N_E-3 /kW/
27. 5. 75	6.03	5,02	16,74	2690	1744
30. 5. 75	18.05	6,07	20,22	3237	2325
	22.06	5,97	19,90	3185	2295
2. 6. 75	12.04	5,98	19,95	3191	2598
	18.08	6,03	20,10	3217	2602
5. 6. 75	4.04	5,96	19,88	3183	2405
	12.07	5,88	19,59	3136	2370
19. 6. 75	6.05	6,68	22,26	3564	3115
	14.06	6,74	22,40	3593	3191

7. COMMENTING SOME FURTHER RESULTS

Long-term measurement of the distribution of neutron flux using thermometric detectors in the cell H-09 of the A1 reactor /1/ involved all typical operational modes of the reactor, which have been further continuously analysed and, together with the data obtained with the assemblies E-3 /4/, E-2, E-4, and with a special assembly with self-powered detectors of neutron flux, they served as a basis for evaluating the operational and safety features of the reactor. The results offered some typical situations, beginning with the operational run of the assembly. In this case a marked vertical asymmetry has been noticed caused by excessive insertion of compensating rods into reactor core. Also changing the positions of the compensating rods resulted in interesting changes in the distribution of neutron fluxes. It should be also stated in this connection that of great importance are the measurements accomplished in the time of higher burnup of the fuel assembly. The results dealt with hereinafter are to be related to the positions of compensating rods as shown in fig. 1.

Fig. 2 shows the change in the vertical distribution of neutron flux density in dependence on [illegible] levels. Reactor thermal power has been [illegible] changed, and neighbouring compensating rods have been [illegible] deep into the reactor. [illegible] distribution of the neutron flux density has been [illegible] step-by-step removing of the three neighbouring compensating rods into positions indicated in fig. [illegible]. [illegible] presents also the distribution [illegible]. Operational values of neutron flux density [illegible] in fig. 1 and [illegible], have been taken from the period of bringing the reactor into power operation up to [illegible]. The operation was resumed in September 1975 (after performing some changes on the reactor core), second series of the measurements has been started, but without the possibility of relating the results with those obtained in the first series. During increasing the reactor thermal power in the second series, compensating rods have been progressively withdrawn. As shown in fig. 4, withdrawing the [illegible] neighbouring compensating rods caused a marked shift of the neutron field into the higher section of the fuel assembly. Progressive burnup of the fuel resulted in many cases in a shift of the maximum of fission heat sources distribution above the geometrical centre of the core. This is also shown in fig. 5. Comparison of relative course of data obtained with Soviet beta-emission detectors /6/ with absolute values of neutron flux density obtained by means of thermometric neutron detectors designed in ŠKODA WORKS, displays a good concordance and is illustrated in fig. 6. Calculation of neutron flux densities (chapter 3 and 5) has been based on using the relations given hereinbefore. These relations are valid for the whole run of the fuel assembly. Corrections for burnup have been done individually for every calorimeter beginning with burnup value of 5[illegible] M.W.t.

[illegible]. CONCLUSIONS

It may be said that a new, original technique of applying the principles of thermometric effects on continuous measurement of radiation-induced loading of uranium, has been brought to a successful conclusion. Research and developmental activities associated with this task have been performed on the A-1 Czechoslovak nuclear power station in the framework of the "Programme of in-core measurements", and took [illegible] years. A set of in-core instrumentation has been developed which measures the absolute amount of heat generated in uranium due to the influence of mixed reactor radiation.

The results of the measurement carried out on the A-1 reactor have demonstrated the advantage of the calorimetric method of measurement. The experiment enabled to find a very important quantity, i.e. the local thermal loading of the fuel and its distribution in the fuel assembly, which directly influences permissible loading of the assembly and, consequently, of the whole reactor. This is of utmost importance when considered from the viewpoint of the distribution of the maximum permissible temperatures of can and fuel.

The authors feel greatly indebted to the workers of the Nuclear Power Construction Department, as well as to the workers of Jaslovské Bohunice Nuclear Power Station who participated actively in the long-term experiment and helped to solve all demanding tasks.

LIST OF SYMBOLS

H	(W/kg)	dose rate
C_k	-	conversion constant, 1.602 . 10^{-13} J/MeV
$\mu_\gamma(E)$	(m^{-1})	macroscopic cross-section for photon capture
E	(MeV,J)	energy of particles, of photos
$\varphi_\gamma(E)$	($m^{-2}s^{-1}$)	photon flux density
$\varphi_n(E)$	($m^{-2}s^{-1}$)	neutron flux density
G	-	selfshielding factor
B	-	correction factor of electron surface emission
$\sigma_e(E)$	(m)	macroscopic cross-section for electron capture
λ_e	(m)	free path of electron
R	(m)	radius of calorimetric sample
H_f	(Wkg^{-1})	fission-induced heating of the material
H_γ	(Wkg^{-1})	photon-induced heating of the material
A	-	mass number
N	(m^{-3})	number of atomic nuclei
$\overline{\cos}\,\vartheta$	-	mean value of the cosine of the scattering angle in the centre-of-gravity system
σ_{se}	(m^2)	microscopic cross-section for elastic scattering of neutrons
Σ_{ef}	(m^{-1})	effective macroscopic cross-section for neutron transport
Q^+	(MeV, J)	mean kinetic energy of fission products
σ_f^5	(m^2)	macroscopic cross-section for the fission of U235
N^5	(m^{-3})	number of U235 nuclei
H_{tot}	(W)	total radiation-induced heating of the calorimetric sample
N_t	(MW)	reactor thermal power

N_{γ}	(W.MW^{-1})	specific radiation-induced heating of the calorimetric sample due to radiation for 1 MW of reactor power
N_{E-1}	(kW)	thermal output of experimental assembly with E-1 detectors
N_{E-3}	(kW)	thermal output of experimental assembly E-3

REFERENCES

1. V. Krett, J. Kott, V. Velechovský: Operational verification of A1 reactor core design parameters. Jaderná energie 20 (1974) No. 7, p. 217 (in Czech).
2. J. Kott, V. Jiroušek, B. Feren, J. Němec, K. Dach, R. Radak: Experience with the development of uranium reactor calorimeter. Jaderná energie 17 (1971) No. 8, p. 254 (in Czech).
3. V. Krett, J. Kott, J. Vlček et al: Irradiation experiment of functional components of special measuring assemblies of A1 reactor. Jaderná energie 20 (1974) No. 9, p. 289 (in Czech).
4. V. Krett, J. Vlček, J. Majer et al: Measuring thermometric parameters of A1 reactor under operation. Jaderná energie 21 (1975) No. 11, p. 408 (in Czech).
5. K. Dach, V. Jiroušek, J. Kott, J. Němec: Thermometric characteristics of an experimental uranium calorimeter. ŠKODA WORKS internal report.
6. S.R. Rana: In-core measurement of neutron fluxes in the KS-150 reactor using self-powered detectors. Program of in-core measurement on the A1 reactor, Symposion, Piešťany, 1973, (in Slovak).
7. J. Kott, V. Jiroušek, J. Němec, K. Dach, R. Radak: Spectrocalorimetric measurements in the core of heavy water reactor RA at Vinca. Bull. of the Boris Kidrich Institute of Nuclear Science 21 (1970) No. 2.

8. V. Jiroušek, J. Smolík, O. Teren: Calorimeters for the experimental program of in-core measurement. ŠKODA WORKS internal report.

9. H. Lehmann: Central and peripheral cell with a calorimetric detector. Not published.

10. Determination of absorbed dose in reactor. Technical report series No. 127, IAEA, 1971.

11. V. Jiroušek, J. Kott, J. Němec, J. Alánička, O. Teren: Results of the measurements performed with assembly type D-1, dia 112 mm in the cell H-09 in the A1 reactor. ŠKODA WORKS internal report.

Mutual position of compensating rods and the cell H-09

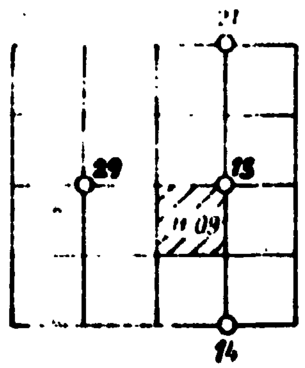

Fig. 1.
Arrangement of compensating rods around the E-1 assembly

Fig. 2. Distribution of thermal neutron flux in the cell H-09, assembly E-1, dia 112 mm. ▲ - 22nd May, 18 hours, 60 MW; x - 22nd May, 22 hours, 114 MW; o - 24th May, 10 hours, 168 MW

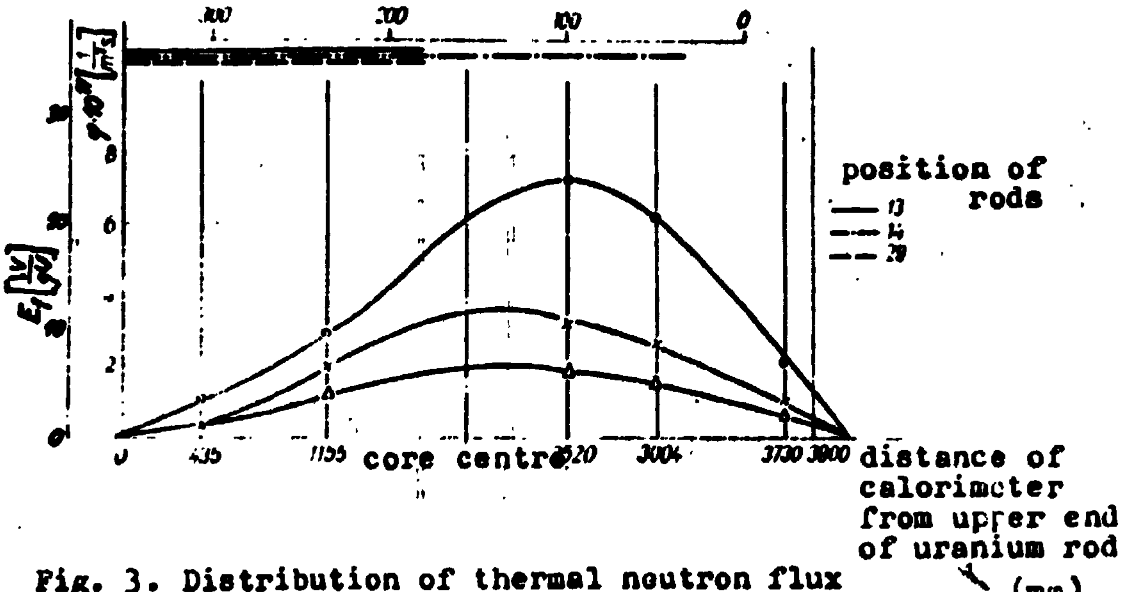

Fig. 3. Distribution of thermal neutron flux in the cell H-09, assembly E-1, dia 112 mm.
+ - 29th May, 2.03 hours; x - 29th May, 4.04 hours; o - 29th May, 5.03 hours.

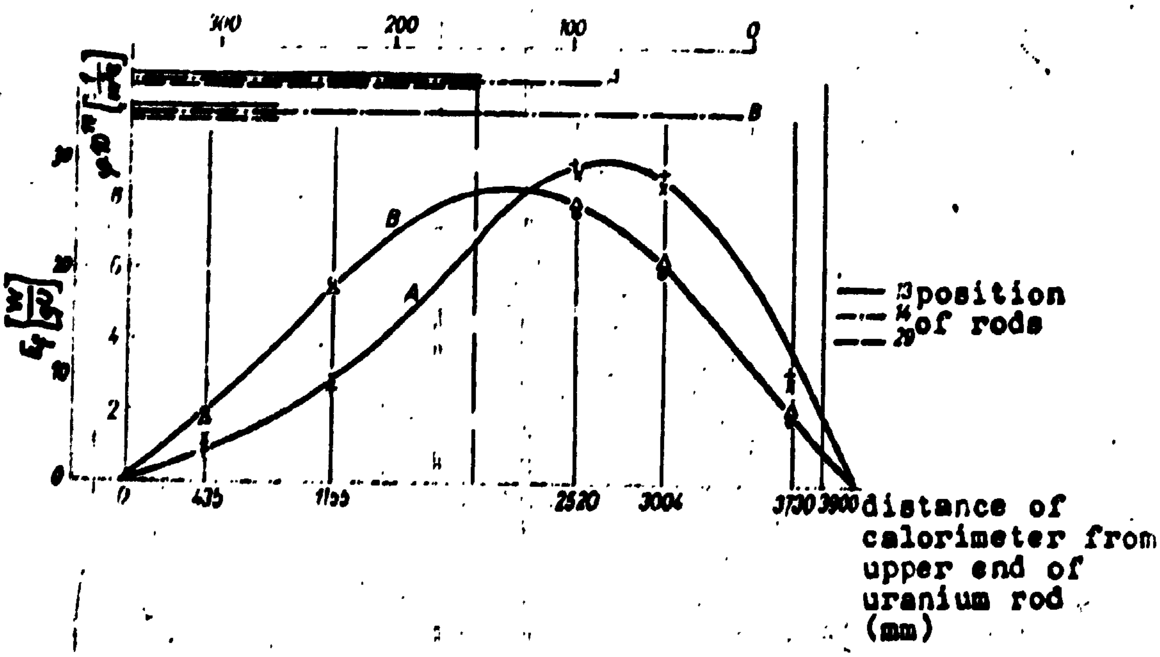

Fig. 4. Distribution of thermal neutron flux in the cell H-09, assembly E-1, dia 112 mm.
o - 3rd Oct., 00.10 hours, 247 MW; x - 4th Oct. 24.00 hours, 320 MW.

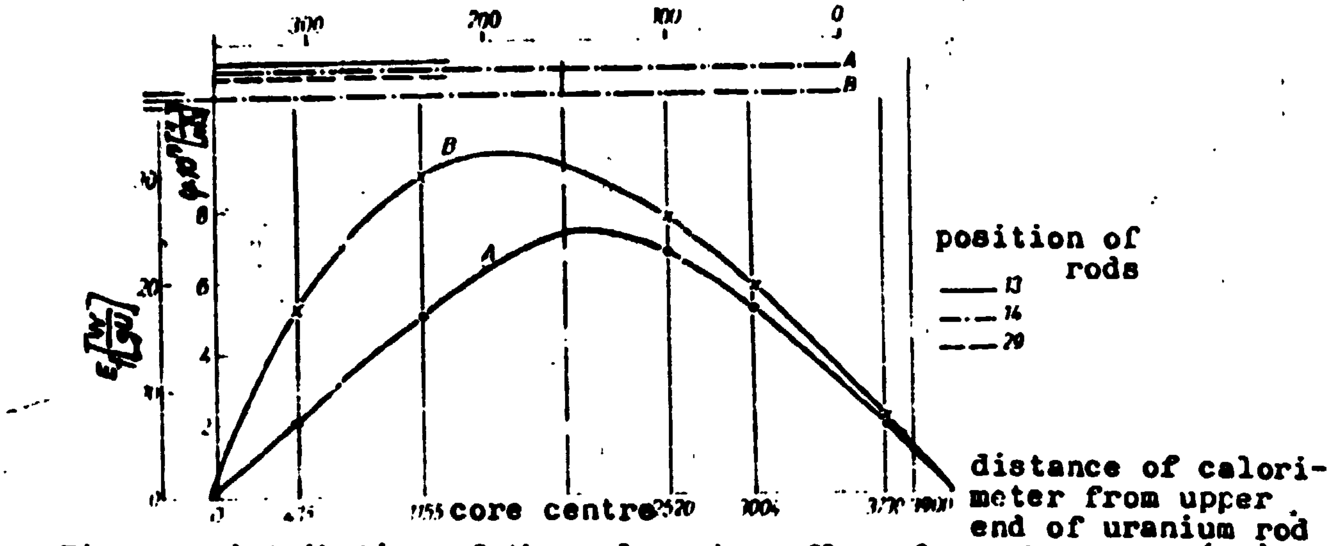

Fig. 5. Distribution of thermal neutron flux along the height of the cell H-09, assembly E-1, dia 112 mm.
x - 31st Oct.73, 12.06 hours, 348 MW; 2nd Nov. 73, 12.02 hours, 350 MW; ▲ - 3rd Nov. 73, 350 MW.

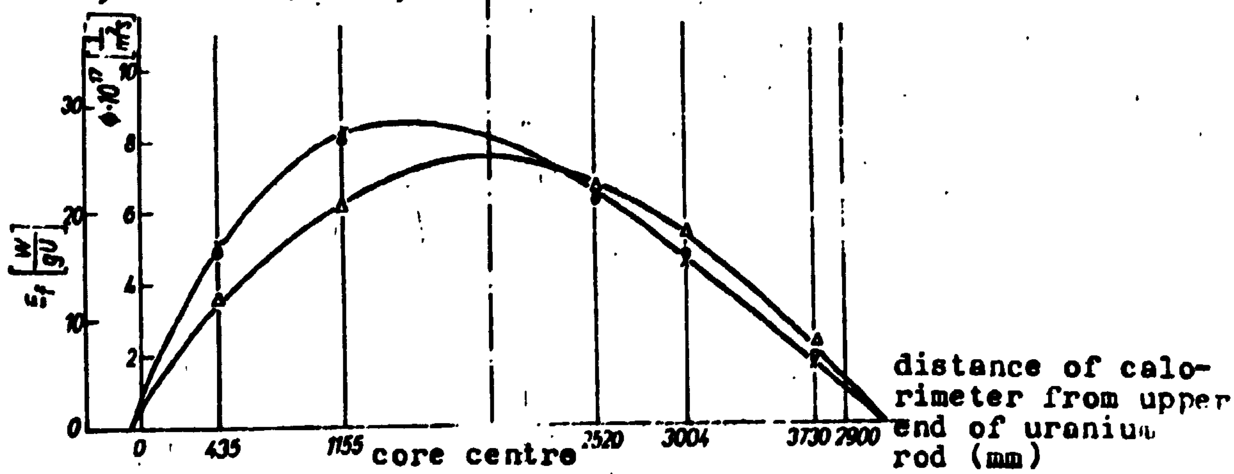

Fig. 6. Comparison of relative courses of data obtained from reactor calorimeters in the cell H-09 and rhodium beta-emitting neutron detectors in the cell D-08;
o - calorimeters in the cell H-09; x - neutroncoaxes in the cell D - 08