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HEAT GENERATION AND TEMPERATURE-RISE IN ORDINARY CONCRETE DUE TO CAPTURE OF THERMAL NEUTRONS

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ABSTRACT

The aim of this work is the evaluation of the heat generation and temperature-rise in local ordinary concrete as a biological shield due to capture of total thermal and reactor thermal neutrons. The total thermal neutron fluxes were measured and calculated. The channel number 2 of the ET-RR-1 reactor was used in the measurements as a neutron source. Computer code ANISN (VAX version) and neutron multigroup cross-section library EURLiB-4 was used in the calculations. The heat generation and temperature-rise in local ordinary concrete were evaluated and calculated.

The results were displayed in curves to show the distribution of thermal neutron fluxes and heat generation as well as temperature-rise with the shield thickness. The results showed that , the heat generation as well as the temperature-rise have their maximum values in the first layers of the shield thickness.

INTRODUCTION

The interactions by which radiations from a reactor core are attenuated in the shield materials surrounding the core inevitably result in a transfer of energy to the shield. This generation of heat within a shield can be a primary consideration in its design , especially when the shield is for high-power reactor in which rather large temperature increases can be expected. Allowable temperature increases are determined principally on the basis of thermal stresses.[1] The maximum allowable temperature may be limited by loss of strength or melting (as in lead) , boiling , or other change in properties (e. g. , loss of water in concrete). Thermal stress , caused by differential thermal expansion in temperature gradient , If the material is constrained. Thermal stress is most serious when the thermal conductivity is low (resulting in a large temperature gradient) and strength is poor , as in concrete. Stress , especially tensile stress, can cause cracking with loss of shielding effectiveness if radiation can stream through the cracks.[2]

High temperatures may also have detrimental effects on the physical , mechanical and nuclear properties of shielding materials.[3] Therefore it is important to prevent overheating of concrete shield , too , to avoid thermal stress , tensile stress , …… and loss of water essential for fast neutron attenuation.[2] Also , it is the responsibility of the shield designer to calculate this heating and required provision for cooling.[4]

Ordinary concrete , which is an excellent neutron shield material because of its water content and relatively high density , has from 8 to 15% of water content at normal temperatures. This water under natural environmental conditions , is retained in the concrete for periods of 20 to 50 years , even after 50 years as much as half the original water may still remain. Heating the concrete increases the molecular vibrations in the crystal of the material , which breaks the hydrated-water molecule bond ; the water begins to come off at a temperature of about 150 °F.[5] The ordinary concrete has one m $W.cm^{-3}$ maximum heating rate , maximum temperature rise of 6 °C and 0.8 °C cm^{-1} maximum temperature gradient.[2] The heat generation and temperature - rise in shielding materials were the subjects many works. Most of earlier published data has been based on theoretical calculations and limited geometries.[11,12]

In the present work the thermal neutron fluxes measured and calculated. The heat generation was estimated using measured and calculated fluxes. The temperature-rise obtained from the resulted heat generation was also estimated in ordinary concrete as biological shield.

EXPERIMENTAL DETAILS AND FLUX DETERMINATION

Experimental measurements have been carried-out using local ordinary concrete (2.3 g. cm^{-3}) blocks each of dimension 120 x 120 x 40 cm. Each block contains two holes 20 cm apart for housing the detectors. The concrete blocks were arranged in front of the horizontal channel number 2 of the ET-RR-1 reactor. The composition of the local ordinary concrete and a complete description of the shielding blocks are given elsewhere[6]. Indium foils $^{115}In\ (n,\gamma)\ ^{116}In$ of 0.15 mm and 0.18 mm diameters were used to measure the thermal neutron flux ($n.cm^{-2}.\ Sec^{-1}$). To measure the thermal neutron flux , bare and cadmium covered foils were used. The cadmium cover has 1mm thickness. The difference between measurements using bare indium foils and that covered with cadmium gives the thermal neutron fluxes. These measurements were carried out first using a direct beam (bare beam) and then repeated with filtered beam of the reactor by cadmium sheet of 1 mm thickness , and the difference between the two cases gives the reactor thermal neutrons.

These measurements were carried-out using end window Geiger-Muller tube and Ultra-Scalar with automatic timer. To reduce the statistical errors , every foil of indium was measured at least three times and the average was considered. The variation of the reactor power and the uncertainty of the time of irradiation during the measurements were estimated by $^{30}P(n,\ p)\ ^{31}P$ detectors as a monitor. The thermal neutron flux φ_{th} ($n.\ cm^{-2}.\ Sec^{-1}$) is given by ;

$$\varphi_{th}(x) = \frac{A_{bare} - A_{cd}}{\sigma_{th}} \quad n.\ cm^{-2}\ sec^{-1} \qquad (1)$$

where ;

A_{bare} : absolute activity for samples without cadmium ;

A_{cd} : absolute activity for samples covered by cadmium ;

σ_{th} : The thermal neutron cross-section (150 barn).

The relaxation lengths for total thermal and reactor thermal neutrons were evaluated by the conversion of the disc beam results to infinite monodirectional plane source by integrating the measured fluxes at different values of Z (beam direction) and R(normal to the beam direction).[7]

$$D(z) = C \int_{R=0}^{R=\infty} D(Z,R)R.\ dR \qquad n.\ cm^{-2}\ sec^{-1} \qquad (2)$$

Where c is a constant independent on the coordinate Z &R and D(Z,R) is the flux at a point Z and R. The relaxation lengths λ were estimated to have the values , 13.5 cm and 9.8 cm for the total thermal and reactor thermal neutrons respectively.

Estimation of the Heat Generation $H_O(x)$ ($W.cm^{-3}$)

If the simplifying assumption is made that , the gamma , beta or alpha radiation emitted during neutron capture is absorbed at once and the heat is released at the point of capture , the approximate volumetric heating rate at a point x due to neutrons conservatively estimated by the following.[1,5,8,9,10]

$$H_O(x) = 1.6 \times 10^{-13} \Sigma_C(E)\ \varphi(x)\ E_B \qquad W.cm^{-3} \qquad (3)$$

Where ;

$\Sigma_C(E)$: the macroscopic capture cross-section for neutrons in ordinary concrete;

$\varphi(x)$: given before by formula (1) ;

E_B : the binding energy for capture reaction ≈ 8 MeV.

Estimation of Temperature-Rise $\Delta T(x)$ (°C) for Gamma-Rays in Ordinary Concrete.

Once the rate of heat generation $H_O(x)$ per unit volume is known, an approximate formula for estimating temperature-rise $\Delta T(x)$ as a function of distance x into a shield, subject to radiation incident on one side, can be given as ; (1,5,8,9)

$$\Delta T(x) = \frac{H_O(x)\ \lambda^2}{K} \left(1 - e^{-x/\lambda}\right) \quad °C \qquad (4)$$

Where ;

- x : the distance into the shield (cm);
- $H_O(x)$: given before by formula (3) ;
- λ : the relaxation length of thermal neutrons in ordinary concrete ;
- K : thermal conductivity of ordinary concrete ($1.1\ W.\ m^{-1}\ °C^{-1}$).

Theoretical Calculation :

The one dimensional discrete ordinate computer code ANISN (VAX version) was used to calculate the attenuation of neutrons and hence the resulted gamma heating in ordinary concrete due to monoenergetic neutron source of thermal energy. The calculations were performed using neutron cross-sections for the thermal neutron energy extracted from the multi-group cross-section library EURLiB-4. Spherical geometry was assumed. ANISN derived fluxes were used to calculate gamma heating with the help of the formula (3).

RESULTS AND DISCUSSION

Experimental measurements have been carried-out to obtain the total thermal neutron fluxes as well as the reactor thermal neutron fluxes both along the beam direction (Z-direction) and perpendicular to the beam direction (R-direction). Also theoretical calculation have been carried out to obtain the total thermal neutron fluxes using the computer code ANISN (VAX version)

The heat generation and temperature-rise due to the measured and calculated thermal neutron fluxes as well as that due to reactor thermal neutron fluxes were evaluated and calculated using formula 3 and 4 respectively.

Figures 1 and 2 show measured and calculated total thermal neutron flux distributions and evaluated and calculated heat generation due to total thermal neutron fluxes respectively. From these figures , it is shown that , good agreement exists between measured and calculated total thermal neutron fluxes and heat generation. Also the figures show the attenuation of the total thermal neutron fluxes and the decrease of the heat generation with the increase of the shield thickness. The differences between measurements and calculations in figures 1 and 2 may be due to the approximation in modeling the experimental geometry which could not be considered spherical. Also it may be due to the cross-section data set used in the calculations.

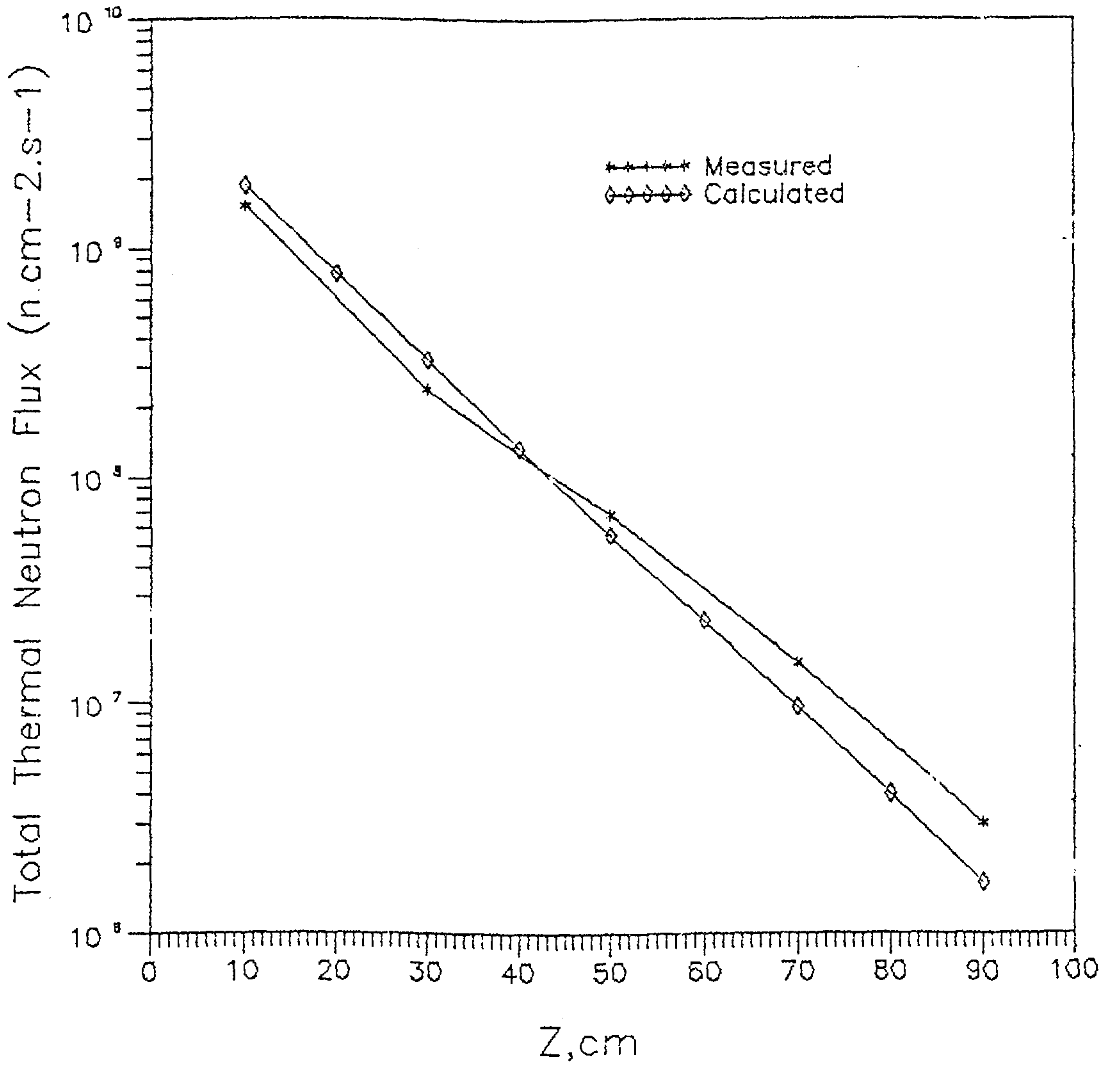

Figure 1 Measured and calculated total thermal neutron fluxes in ordinary concrete

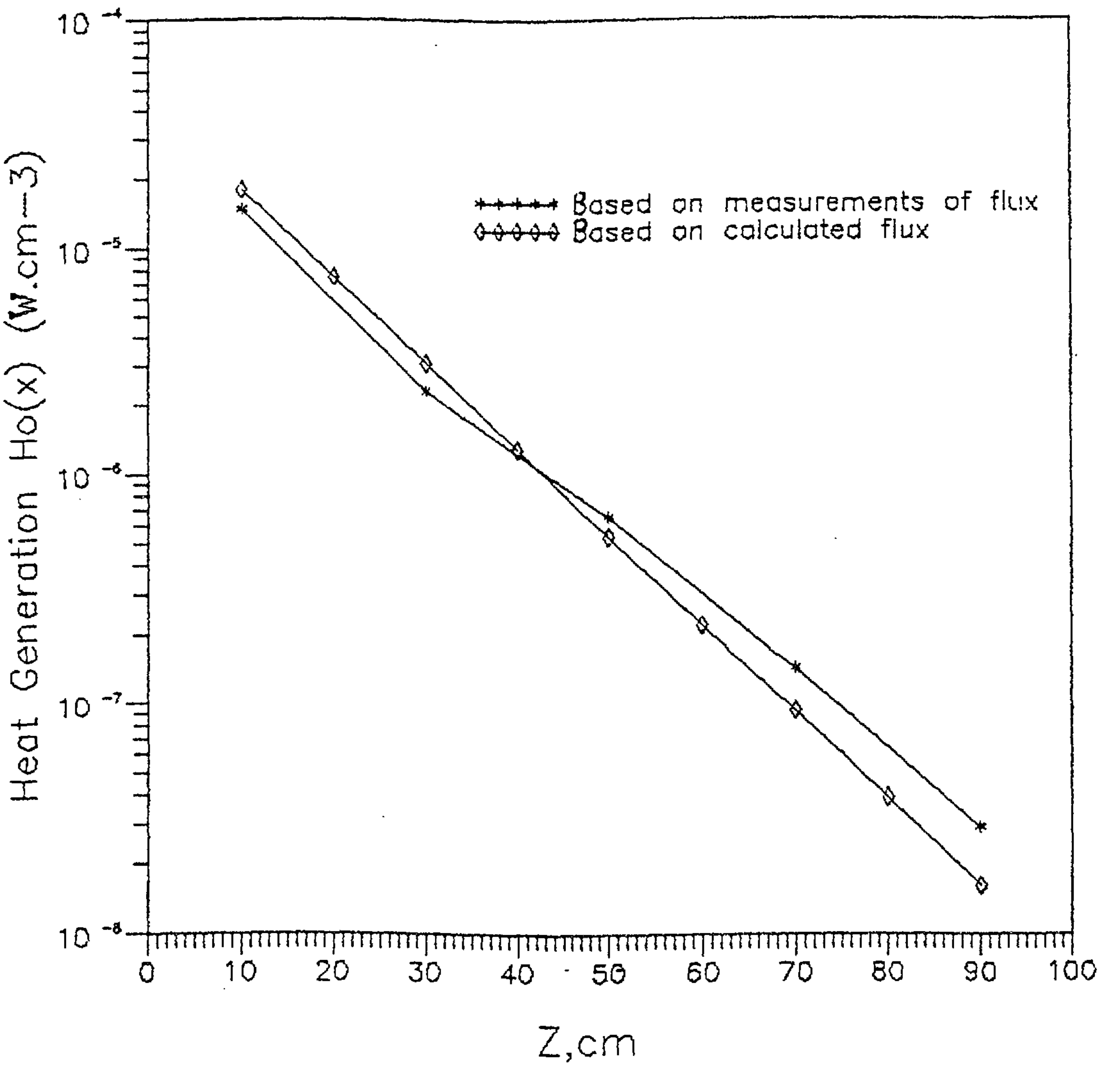

Figure 2 Heat generation in ordinary concrete evaluated using measured and calculated total thermal neutron fluxes

Figure 2 shows that the heat generation due to capture of total thermal neutrons can reach about 2×10^{-4} m W. cm^{-3} at 50 cm thickness in the shield which generates a temperature-rise about 0.005 °C. However at thickness 10 cm the heat generation reach of about 2×10^{-2} m W. cm^{-3} which generate a temperature-rise of about 0.17°C. This temperature-rise does not cause dehydration in ordinary concrete.

Figure 3 shows the heat generation due to both total thermal neutron and the reactor thermal neutron fluxes along the beam direction (Z-direction) and normal to the beam direction (R-direction). The figure shows a marked depression in the heat generation due to the reactor thermal neutron fluxes in both Z and R directions as compared with that due to the total thermal neutron fluxes. This depression in the heat generation reaches about 30% at 10 cm shield thickness in both Z and R directions. The depression in the heat generation reaches about 60% at 40 cm and 10 cm in the R and Z directions respectively. Along the beam direction the depression increases with increasing the shield thickness to reach about 80% at 50 cm and more than 90% at 70 cm thickness in the shield. This indicates that the heat generation due to thermal neutrons emitted directly from the reactor give the main contribution of the heat generation in the first layers of the shield. However at deep penetrations the contribution of the reactor thermal neutrons in the heat generation is about 20% or less.

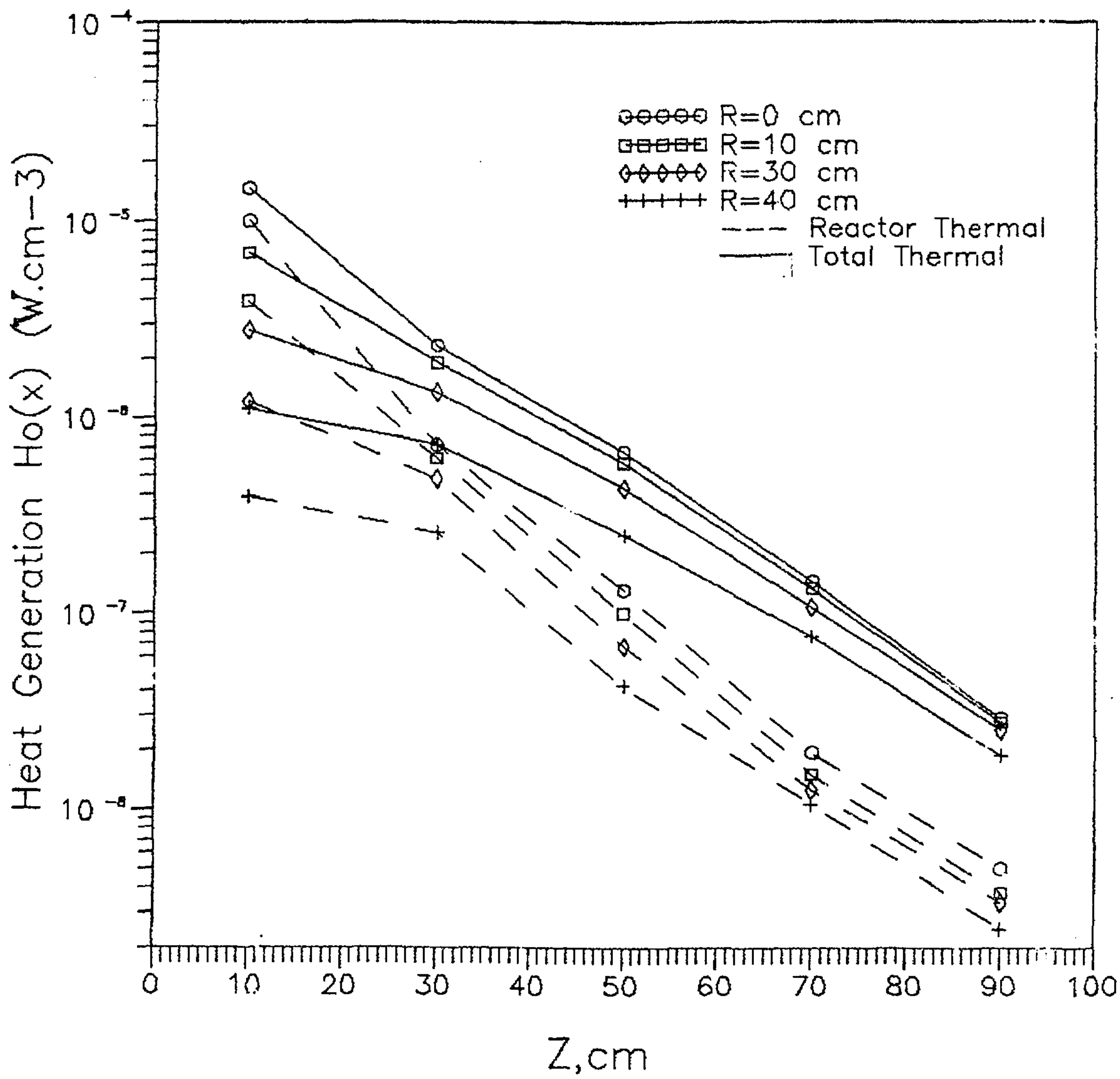

Figure 3 Heat generation in ordinary concrete evaluated using measured total and reactor thermal neutron fluxes

Figure 4 shows the temperature-rise in concrete because of gamma-rays due to capture of total thermal neutrons (measured and calculated) and that due to reactor thermal neutrons. The figures shows that , temperature-rise has its maximum value in the first layers of shield thickness and decreases rapidly atmdeep depths in the shield. The figure also shows that , the maximum evaluated temperature-rise is of about 0.17 °C, and this value of temperature increase has no effect in dehydration of the concrete. Also , from the figure it is noticed that , the temperature-rise because of gamma-rays due to capture of the reactor thermal neutrons has very small contribution as compared with total thermal neutrons.

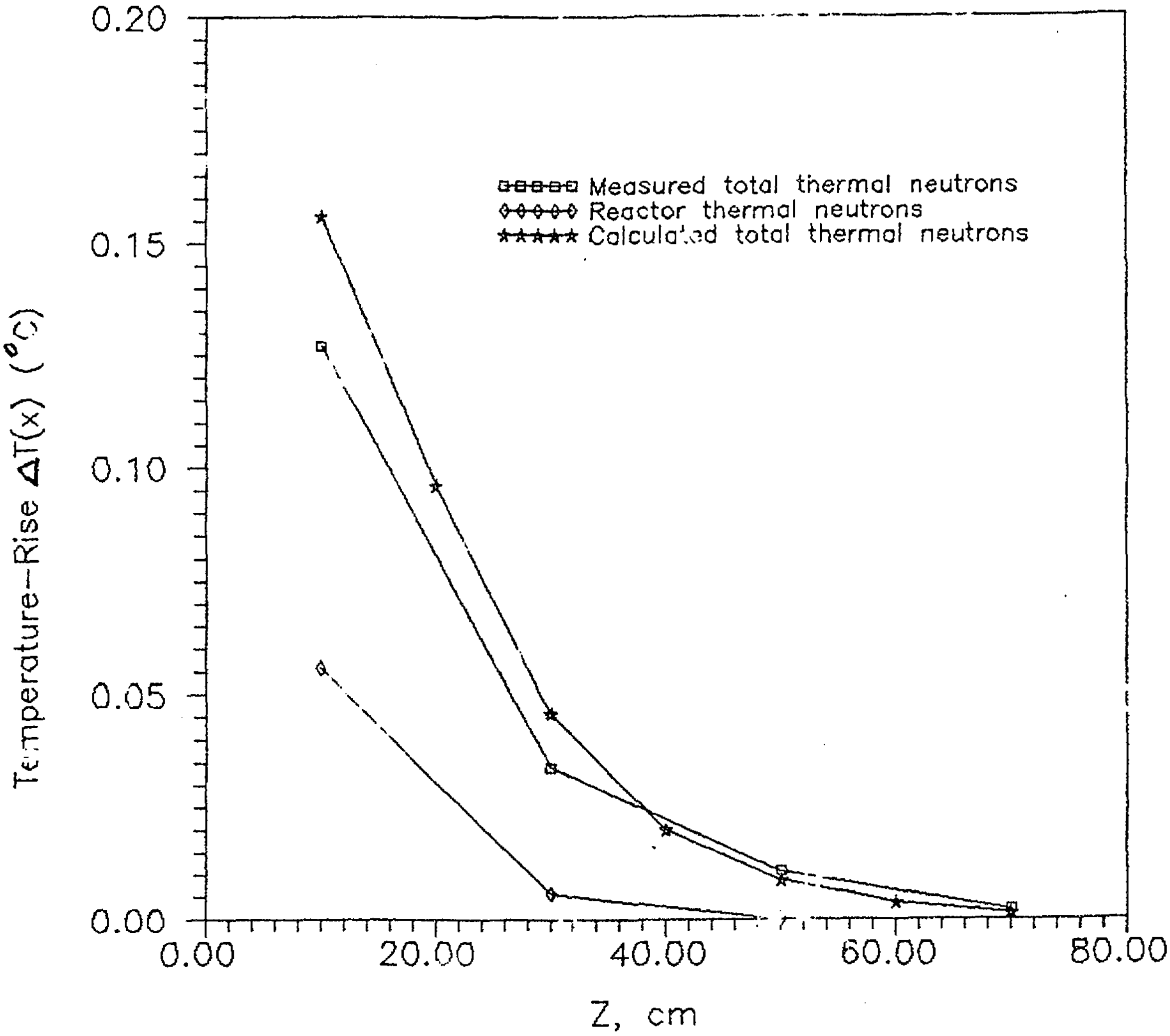

Figure 4 Temperature-rise distribution of gamma-rays due to capture of thermal neutrons in ordinary concrete

CONCLUSIONS

The measured and calculated total thermal neutron fluxes as well as evaluated and calculated heat generation and the temperature-rise in local ordinary concrete give the following conclusions ;

1- The total thermal neutron fluxes , measured and calculated , decrease with increasing the shield thickness as expected due to attenuation of neutrons in ordinary concrete of the same density.

2- The maximum value of the heat generation Ho(x) due to total thermal neutrons as well as that due to reactor thermal neutrons occurs in the first layers of the shield.

3- the temperature-rise $\Delta T(x)$ has its maximum value in the first layers of the shield and decreases rapidly with increasing the shield thickness.

4- The heat generation and temperature-rise due to the reactor thermal neutrons have a small contribution compared to the total thermal neutrons.

Because of the importance of the calculation of heat generation in the reactor shield due to fast neutrons and gamma-rays future work is planned to be done for different types of shields (concretes , iron , ...and so on).

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