KfK 4142
Oktober 1986

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A Simple Method to Calculate Neutron Energy Deposition in ICF Targets

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ISSN 0303-4003

Abstract

In most ICF-target simulation calculations α's and other charged particles are supposed to be reabsorbed in the plasma whereas neutrons are often allowed to escape without interaction. However, in reactor size targets neutrons lose about the same amount of energy in the DT-fuel as α-particles and can considerably influence the burn characteristics of the target. In the present paper a simple method to account for neutron-fuel interaction is presented. Results are compared with Monte-Carlo and S_N transport calculations. The effect of neutron energy deposition in the fuel on the performance is shown on the 4 mg DT-target of the HIBALL reactor.

Eine einfache Methode zur Berechnung der Neutronen-Energie-Einlagerung in ICF-Targets

Zusammenfassung

In Trägheits-Fusions-Target-Simulationen wird häufig angenommen, daß α- und andere geladene Teilchen in dem Target wieder absorbiert werden, wogegen Neutronen das Plasma ohne Wechselwirkung verlassen. In größeren Targets deponieren Neutronen jedoch etwa die gleiche Menge an Energie wie die α-Teilchen. Dies beeinflußt das Targetverhalten beträchtlich. In der vorliegenden Arbeit wird eine einfache Methode zur Berechnung der Targetaufheizung durch Neutronen vorgestellt. Die Ergebnisse werden mit denen der Monte-Carlo- und S_N-Transport-Rechnungen verglichen. Der Einfluß der Neutronen-Energie-Einlagerung auf das Verhalten des 4 mg DT HIBALL Targets wird aufgezeigt.

To minimise the driver energy requirement, targets for inertial confinement fusion (ICF) are designed so that a small fraction of the fuel at a relatively low density is heated to ignition temperature keeping the rest of the fuel cold but compressed to a high density. The energy produced in the central spark region is reabsorbed in the fuel thus bootstrap heating all of the fuel to ignition temperatures. This requires the reabsorbed energy to be greater than the energy lost due to radiation and other processes. Whereas in most simulation calculations α's and other charged particles are supposed to be reabsorbed in the plasma, neutrons are often allowed to escape without interaction i.e. the energy imparted to neutrons in the fusion reactions does not contribute to the fuel heating. This may be a good approximation for small targets with fuel size at ignition much smaller than the neutron mean free path. The mass mean free path of 14 MeV neutrons is 4.75 g/cm^2 and for a 30% burn efficiency the required fuel size must be of the order of $\rho R = 3\ g/cm^2$. This means that about 30% of the neutrons make at least one collision with the fuel atoms before leaving the target.

In the present report a simple method to account for neutron-fuel interaction is presented. The method is required to be computing efficient in order to be usable in hydrodynamics codes for routine calculations. Some preliminary results of this method have been published previously [1,2] but the method has not been described so far. In the following the method is described in some detail. Calculations are presented for a 4mg DT-target published recently[2,3] and results are compared with those obtained with the neutron transport code ONETRAN[4]. A comparison of the results of this simple method with more elaborate time dependent neutron transport calculations shows that this is an adequate method to calculate integral target parameters[5].

METHOD

Following the work of Abdou and Maynard[6] the heating rate per unit volume in a scalar flux $\Phi(r,\varepsilon)$ can be written as

$$1)\qquad H(r) = \int \Phi(r,\varepsilon) \sum_{ij} N_i(r)\sigma_{ij}E_{ij}\, d(\varepsilon)$$

where $N_i(r)$ is the density of the particles interacting with neutrons, σ_{ij} is the cross section of the process j (i.e scattering, (n,2n), (n,γ) etc.) on the nucleus i and E_{ij} is the recoil energy of the nucleus i undergoing process j. The KERMA factor for the process j on nucleus i is defined as

$$(2)\qquad k_{ij}(\varepsilon) = \sigma_{ij}(\varepsilon)\, E_{ij}$$

The total KERMA factor for the fuel atoms interacting with the neutrons, i.e. the average kinetic energy released per collision in the fuel, is given by;

$$(3)\qquad k(\varepsilon) = \frac{\sum_{ij} N_i\, \sigma_{ij}(\varepsilon)\, E_{ij}}{\sum_i N_i\, \sigma_i(\varepsilon)}$$

where σ_i is the total cross section for the species i. If charged paricles are emitted in the reaction their contribution has to be added in equation (3).

The heating function per unit neutron can now be written as

$$(4)\qquad h(r) = \int P_c(\varepsilon,r)\, k(\varepsilon)d\varepsilon$$

where $P_c(\varepsilon,r)$ is the collision probability of the neutron with energy ε at r. For a uniformly burning sphere the total collision probability can be calculated using the average chord method to compute escape probabilities[7], i.e.,

$$(5) \quad P_e(R) = (\lambda/R_{av}) \int_0^{2R} \{1-\exp(-r/\lambda)\}\ \chi(r)dr$$

where $R_{av} = 4V/S$ is the mean chord length of the considered enclosure of volume V and surface area S, $\lambda=\lambda(\varepsilon)$ is the mean free path of neutrons of energy ε in the medium and $\chi(r)$ is the chord length distribution.

For a sphere of radius R;

$$R_{av} = 4R/3 \quad \text{and}$$

$$\chi(r)dr = rdr/2R^2$$

Thus the escape probability of a sphere is given by

$$(6) \quad P_e(R) = (3/8)(\lambda/R^3) \int_0^{2R} \{1-\exp(-r/\lambda)\}\ r\ dr$$

Upon integrating one obtains

$$(7) \quad P_e(R) = (3/8)(\lambda/R)^3\ \{2(R/\lambda)^2 - 1 + (1 + 2R/\lambda)\exp(-2R/\lambda)\}$$

The commonly used parameter in the ICF calculations is ρR, ρ being the density of the fuel. Density is assumed to be constant throughout the fuel. In case of nonconstant fuel density, the density is averaged over the fuel which is equivalent to the mass averaging of ρR i.e.

$$\langle\rho R\rangle = \frac{\int \rho r\, \rho r^2\, dr}{\int \rho r^2\, dr}$$

The constant density of the fuel is also implied by assuming that the neutron mean free path, $\lambda(\varepsilon)$, depends only on the neutron energy. To obtain the escape probability as a function of ρR equation (7) is multiplied by ρ both in the numerator and denominator. The quantity $\lambda(\varepsilon)\rho = \Lambda(\varepsilon)$ is now the mass mean free path of the neutrons of energy ε in the fuel of density ρ.

$$(8) \quad P_e(\rho R) = (3/8)(\Lambda/\rho R)^3 \{2(\rho R/\Lambda)^2 - 1 + (1 + 2(\rho R/\Lambda)\exp(-2\rho R/\Lambda)\}$$

The collision probability is given as

$$(9) \quad P_c(\varepsilon,\rho R) = 1 - P_e(\varepsilon,\rho R)$$

Equations 7 and 8 are correct for a uniformly burning sphere. The major contribution to the neutron energy deposition comes from the first collision (Fig. 2). which is calculated correctly by using Eqn.'s 4, 8 and 9. To be able to apply the above formalism to subsequent collisions, the scattered neutron source is assumed to be uniform and isotropic after each collision and treated in the same manner as the first collision. The neutron cross sections and KERMA factors for the n^{th} collision are averaged over the spectrum of neutrons having undergone (n-1) collisions. The anisotropy of the scattering is taken into account through the KERMA factors.

In Fig. 1 results for a uniformly burning sphere are shown as a function of ρR and compared with those of Monte-Carlo calculations of Beynon and Constantine[8]. The observed good agreement between the present results and the Monte-Carlo results indicate that the assumption of the uniformity of the source after each collision is a reasonably good approximation. This may be attributed to the fact that the geometry effect increases the scattered source density at the centre of the sphere whereas the anisotropy of scattering increases the source density towards the edge of the sphere.

In Fig. 2 the importance of multiple collision to the neutron energy deposition is demonstrated. For small targets of $\rho R = 2\ g/cm^2$ or less two collisions may be adequate to describe the neutron heating effect. For larger targets inclusion of a larger number of collisions is necessary.

Calculations based on the escape probability method have been performed previously by Southworth and Campbell[9]. They have used the Wigner rational approximation to calculate the neutron spectrum leaking from a bare DT sphere. This approximation, however, is known to be inaccurate by about 20% for enclosures of the size of neutron mean free path. Further in their calculations the anisotropy of scattering was not included and cross sections were assumed to be constant within an scattering interval.

An exact method (within the assumption of a uniform density sphere with an isotropic and spatially flat source) to calculate the spatial distribution is to compute the $\Phi(r)$ of equation (1) by solving the integral transport equation for neutrons

$$(10) \qquad \Phi(r) = \int \frac{\exp(-|r-r'|/\lambda)}{4\pi\,|r-r'|^2}\, q(r')dv'$$

For a uniform sphere of unit source and radius R $q(r') = \text{const} = 3/4\pi R^3$ and the uncollided flux at a point r is given by (see e.g.[10])

$$(11) \qquad r\Phi(r) = (3/8\pi r R^3) \int_{-R}^{+R} r'\, E_1(|r-r'|/\lambda)\, dr'$$

where $E_1(x)$ is the exponential integral[11]. On integrating one obtains;

$$(12) \qquad \Phi(r) = (3/8\pi R^3)\ [2\lambda + (R\lambda/r)\{E_2((R+r)/\lambda) - E_2((R-r)/\lambda)\} + (\lambda^2/r)\{E_3((R+r)/\lambda) - E_3((R-r)/\lambda)\}]$$

The collision probability in the cold fuel region is calculated assuming linear neutron attenuation. The escape probability for a definite collision forms the neutron source. It is assumed that neutrons make only one collision in the cold region.

$$(13) \qquad P_c(n) = P_e(\langle\rho R\rangle_H, n)\ \{\ 1 - \exp(-\rho\Delta R/\Lambda)\}$$

where ρ and ΔR the density and thickness of the shell and n denotes that the neutron has undergone n collisions in the hot burning region. $\langle\rho R\rangle_H$ is the size of hot burnig zone.

RESULTS and DISCUSSION

The above procedure has been implemented in the Karlsruhe version of the code MEDUSA[12,13]. Table 1 gives some of the target parameters of 4 mg DT target relevant for present study. The DT fuel is surrounded by a PbLi pusher of 67 mg and a Pb tamper of 256 mg. For detailed description of the target see Refs. [2,3]. Fig. 3 shows the neutron energy deposition in the target as a function of the time after ignition. These results of MEDUSA simulation calculations are compared with those obtained with the neutron transport code ONETRAN for two different time steps. It is seen that the agreement is quite good. For most parts of the burn, the energy deposited by neutrons in the fuel is comparable to that of α-particles.

Fig. 4 and 5 show the rate of neutron production and the target gain as a function of burn time with and without neutron heating included. It is seen that neutron heating reduces the burn propagation time from 44 ps to 29 ps. The ultimate gain of the pellet drops from 179(Ref.[3]) to 156.5.

It is pointed out that not all of the energy deposited by the neutron is utilised to heat the target. The neutrons deposit their energy primarily in form of recoil D or T ions. These ions being supra thermal may partly escape from the fuel reducing the effective heating. This has also the effect of enhanced depletion of the fuel reducing thereby the fusion rate. In the absence of charge particle transport in the code the exact magnitude of these effects can not be determined. These effects are, however, small at the start of ignition and increase with the time when increasing temperature makes the fuel more transparent to charged particles[14]. Another limitation of this method is that it considers only the steady state case. For time dependent treatment of neutron transport see Ref. [5].
To summarise it has been shown that neutron energy deposition in the reactor size targets is important and is of the same order of magnitude as the α-particle heating. It has a substantial effect on the burn charactristics of the ICF-target. The method presented here offers a simple and computing efficient way to calculate the neutron fuel interaction coupled to routine hydro calculations.

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TABLE 1 Some characteristics of the target

DT-mass		4 mg
Fuel size at ignition	($\Sigma\ \rho_i \Delta R_i$)	4.3 g/cm^2
	($< \rho R >$)	6.2 g/cm^2
Average energy of neutrons leaving target		10.04 MeV
Neutron multiplication		1.04
Neutron flytime accros the target		~ 2 ps
Burn duration		~ 80 ps

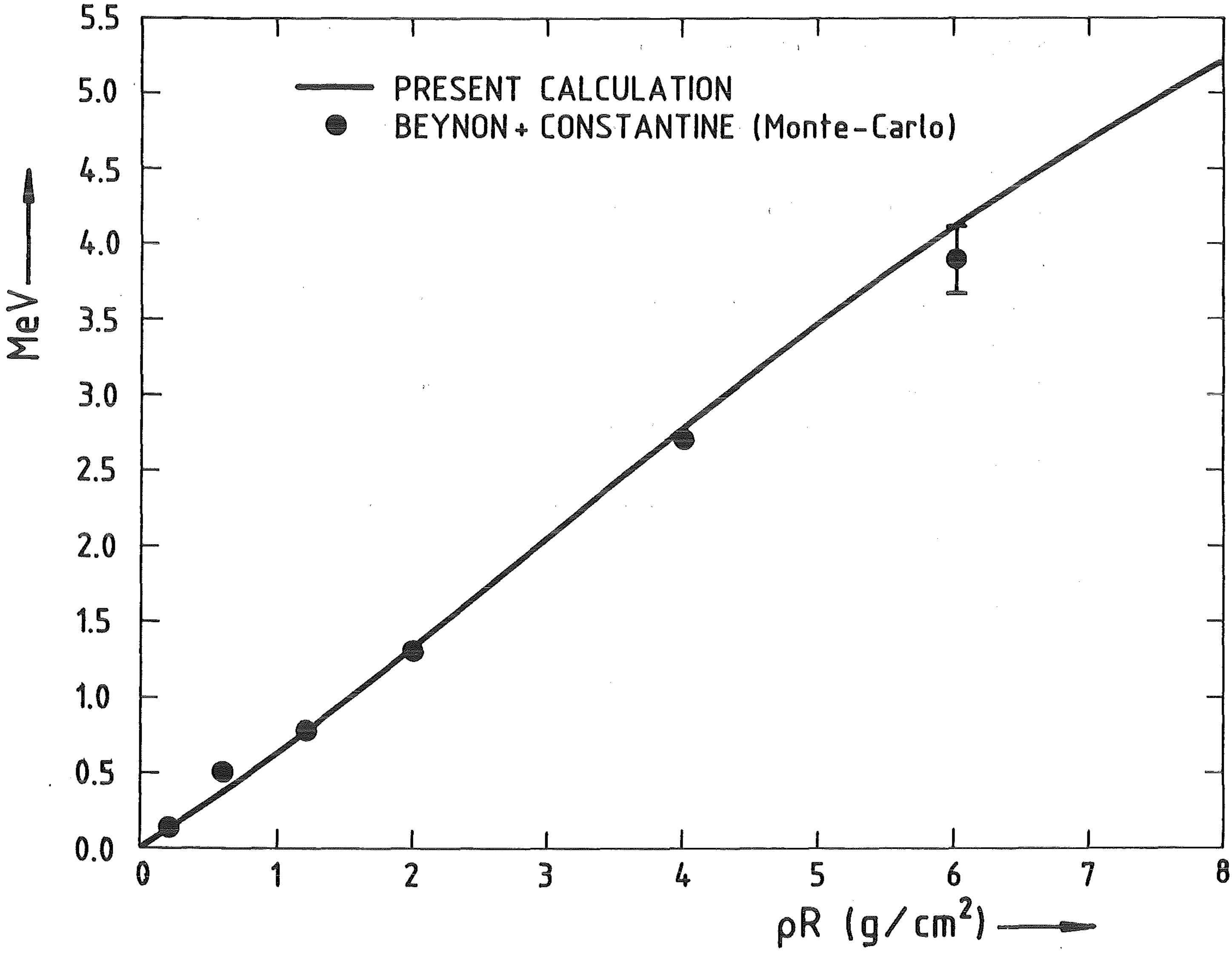

FIG. 1 AVERAGE NEUTRON ENERGY DEPOSITION IN HOT FUEL

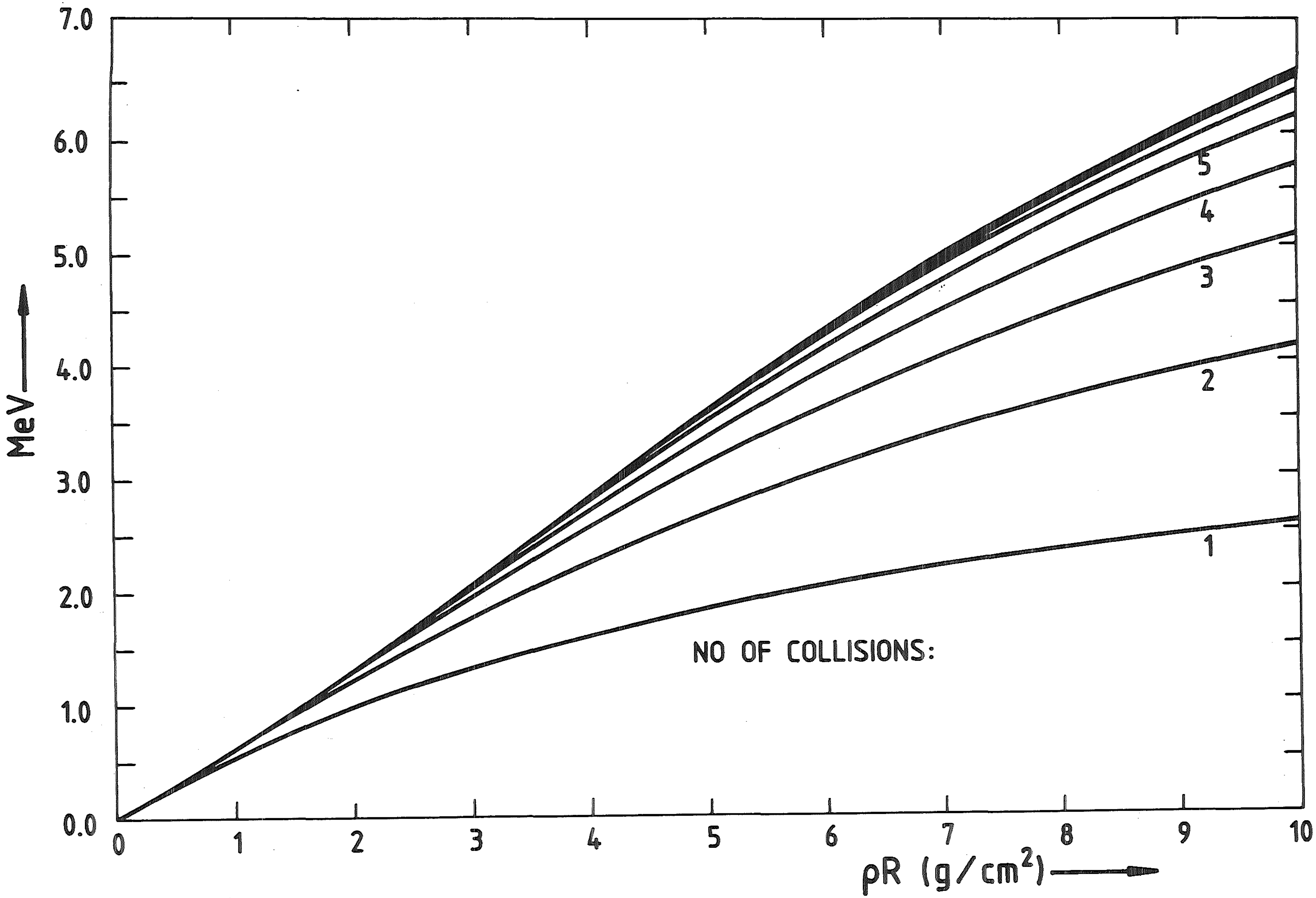

FIG. 2 AVERAGE NEUTRON ENERGY DEPOSITION AS FUNCTION OF NUMBER OF COLLISIONS

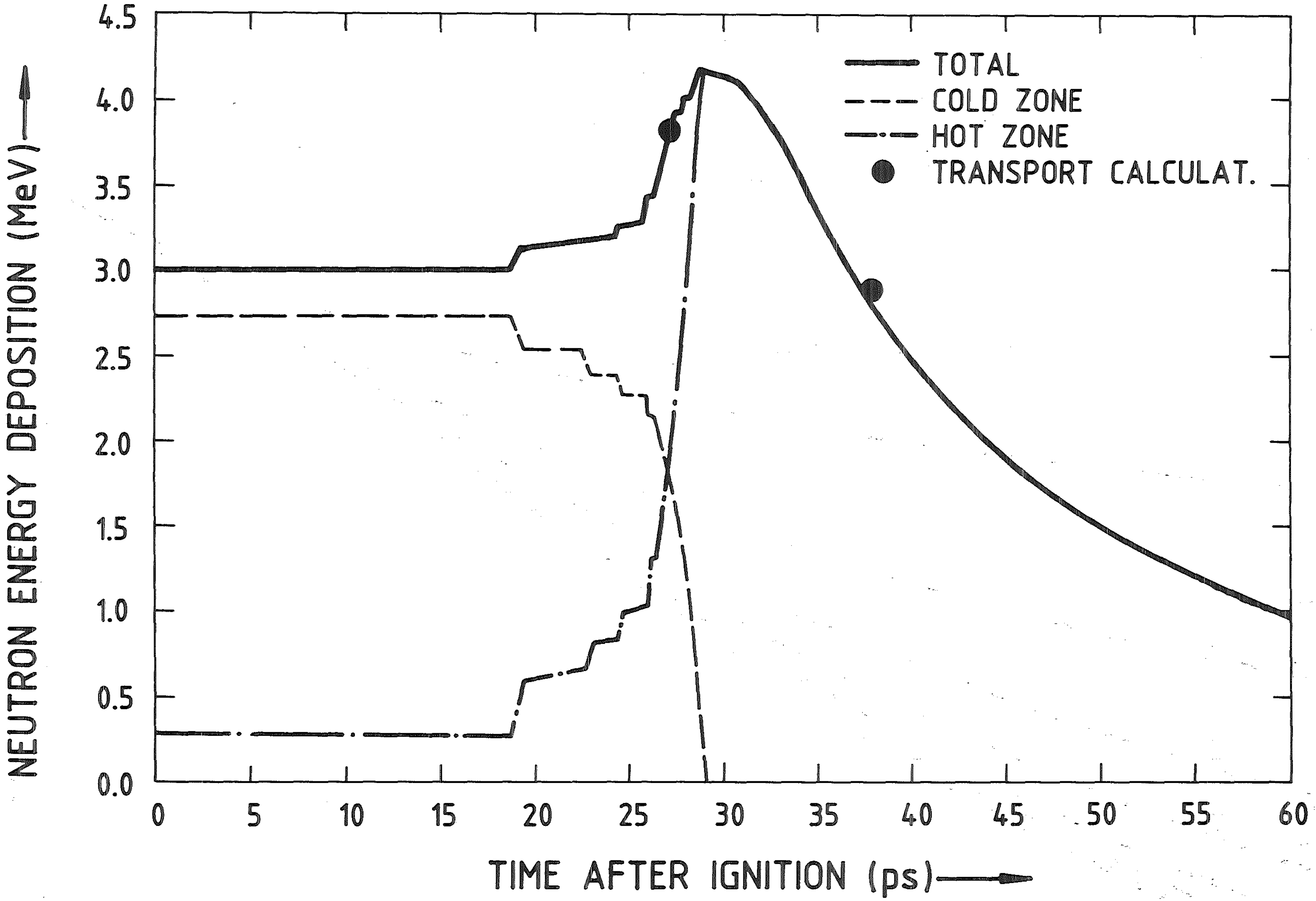

FIG. 3 NEUTRON ENERGY DEPOSITION IN FUEL OF THE HIBALL TARGET

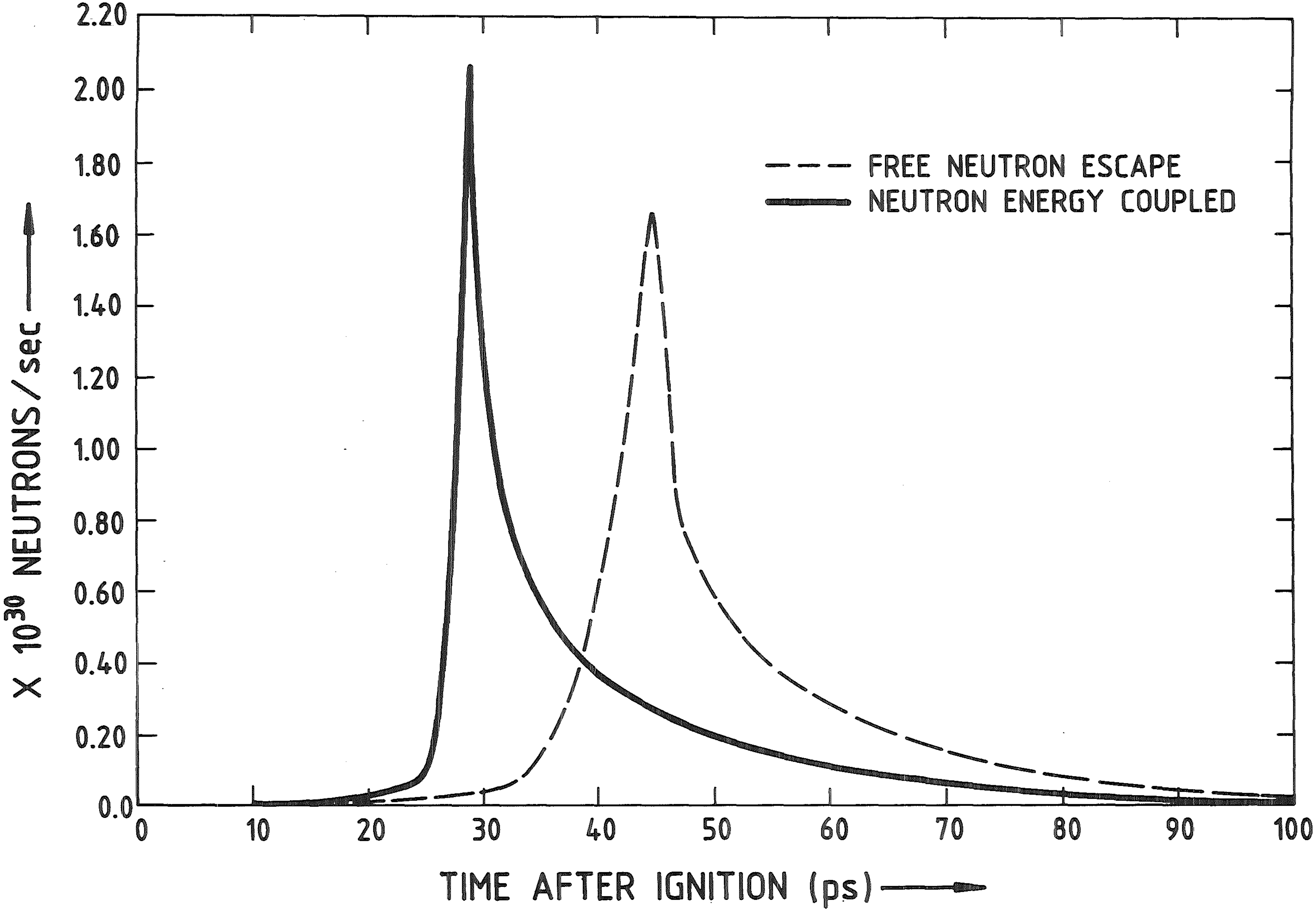

FIG. 4 RATE OF NEUTRON PRODUCTION

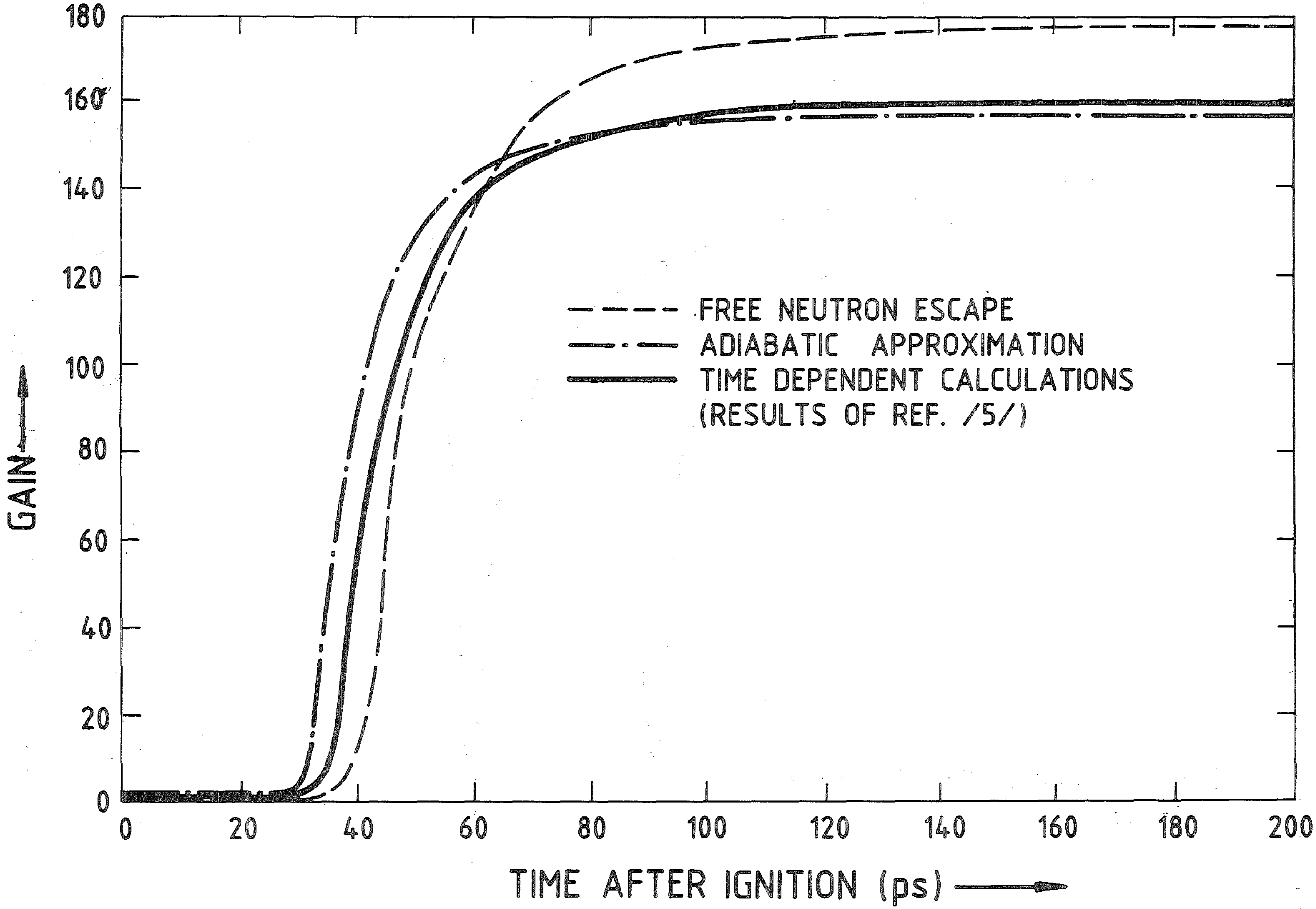

FIG. 5 CHANGE IN TARGET GAIN DUE TO NEUTRON HEATING