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MODERATION WITH RESPECT TO GRAPHITE DAMAGE

by

R. Nilson

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INTRODUCTION

The slowing down of fast neutrons in the moderator of a thermal reactor is a process whereby kinetic energy is transferred from the neutrons to the atoms of the moderator. For sometime it has been known that various types of damage in reactor materials are directly attributable to the moderation process. In particular, the problem of graphite contraction is of major concern both in assessing the continued life of the present reactors and in determining the expected behavior of the NPR graphite.

For several years graphite samples have been irradiated in various reactors, including our own production reactors, and an index of measurement has been adopted in the laboratory to describe the contraction rate. This index is $X\%$ linear contraction per 1000 MWD/AT. That is the contraction in a test sample is related to the number of MWD's in a ton of fuel elements in a fuel channel adjacent to the test sample.

Similarly, records of graphite contraction at the production reactors are now used to derive a figure relating to a pile average contraction for 1000 MWD/T. Not too surprisingly the laboratory data and the pile data when cast into equivalent form are in fair agreement.

While an index of $X\%/1000$ MWD/AT is useful in comparing data on different types of graphite all of the same density from irradiations in the same or similar reactors, it can be grossly misleading when it is used to predict contraction rates in a new reactor, to correlate data taken from dissimilar reactor irradiations, or to correlate contraction data on graphite samples of different densities even though irradiated in the same facility. The index can, however, be used in any of the above applications if used properly. It is the purpose of this brief report to summarize some of the succinct points in this regard and to especially derive some of the proper factors to be taken into account when, for example, one wishes to use contraction data from B, D, and F Reactors to predict the behavior of the NPR graphite.

SUMMARY AND CONCLUSIONS

Under simplifying assumptions, an expression is obtained relating the integrated kinetic energy transferred to moderator atoms to the thermal neutron exposure in adjacent fuel. Reactor-dependent factors are explicit in the expression which make it possible to compare graphite contraction data from different reactors or to estimate graphite damage in a new reactor based on observed damage in existent reactors. Graphite temperature, which may be an important factor in graphite contraction, is not considered as one of the variables in this report. Also, the damage dependency on neutron energy is not dealt with in great detail because of the relatively unknown facts in this regard.

A particular application of concern at the present is to determine the pertinent factors relating to fast neutron exposure in the NPR to assist in determining the integrated damage over the life of the pile.

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SUMMARY AND CONCLUSIONS (Contd)

The derived expression is termed the energy transfer function $R_1(E)$ and is defined as follows:

$R_1(E)$ = the integrated energy in Mev transferred to a cubic centimeter of graphite by neutron moderation per Mev of energy per 1000 MWD/AT.

The form of the expression is

$$R_1(E) = \frac{\text{Const} \sum_{s1}(E)}{\alpha_2 D(E)}$$

where

$\sum_{s1}(E)$ is the macroscopic scattering cross section of the graphite for which the energy transfer function is desired.

α_2 is the feet per ton of fuel element in the reactor supplying the neutrons.

$D(E)$ is the average slowing down power of the moderator of the reactor supplying the neutrons.

The constant takes into account reactor dependent factors such as heat of fission, lattice spacing, cell volumes, etc. and also various reactor independent constants. The detailed expression is given in Equation 16 of the report.

A particular application is made of the transfer function to determine the relative values of the function for the NPR and a BDF type reactor. Under two simple assumptions on the neutron energy dependence of the graphite damage, the following results were obtained for the integral damage ratio between the NPR (21.4 lb/ft concentric tube) and a BDF reactor with standard solid fuel element.

<u>Assumption</u>	<u>Damage Ratio for Same Exposure</u>
Damage independent of neutron energy	1.95
Damage varies directly as neutron energy	1.98

NPR
BDF

The significance of the above figures is that graphite damage due to fast neutron scattering will occur (as a function of exposure per adjacent ton of fuel) at approximately twice the rate in the NPR as in an original Hanford reactor.

DISCUSSION

The expression which is desired is one which relates the time-integrated fast neutron energy transferred per unit volume of moderator (at a given position and of a given moderator type) to an exposure-unit of 1000 MWD/AT. The expression must explicitly indicate the factors which differentiate one reactor from another. Short of costly experiments, the only method to carry out the complex calculations of neutron slowing down in a heterogeneous reactor is a Monte Carlo calculation. Such a calculation is underway. Short of this, much can be said based on a more qualitative treatment. Such is the treatment here.

Assumptions

The following assumptions must be made:

- (1) The reactor is represented by an infinite square array of infinitely long line-fission sources separated by a spacing $2b$ in a homogeneous moderator characterized by a slowing down power obtained by averaging the slowing down powers for the various moderators present.
- (2) Each line source emits N 2 Mev neutrons per second per unit length (cm).
- (3) At the higher energies p-scattering effects are included to determine the kinetics of scattering and energy transfer..
- (4) The continuous slowing down model is valid in the assumed homogeneous moderator.
- (5) No neutron absorption occurs during the slowing down process to 1 kev.
- (6) No neutron moderation process except elastic scattering in the moderators is considered. In particular, inelastic scattering in uranium is neglected.
- (7) The asymptotic energy distribution of neutron flux is valid at all energies considered.

Derivation of the Basic Expression

Simons¹ has shown that the neutron slowing-down density q for an entire reactor under assumption 1 can be written as

$$q(x, y; \tau) = -\frac{N}{4b^2} \mathcal{J}_3\left(\frac{\pi x}{2b}, \exp -\frac{\pi^2 \tau}{b^2}\right) \mathcal{J}_3\left(\frac{\pi y}{2b}, \exp -\frac{\pi^2 \tau}{b^2}\right), \quad (1)$$

¹L. Simons, A New Method For Calculating The Fine Structure In Heterogeneous Reactors, Proc. of the Int. Conf. on the Peaceful Uses of Atomic Energy, 1958, P/1100, vol. 16, p 644, United Nations, New York (1958).

Derivation of the Basic Expression (Contd)

which is equal to the number of neutrons slowing down past energy E Mev (corresponding to an age τ) at position (x, y) per unit volume per second.

N and b have been defined in the assumptions. The Fermi age is measured from 2 Mev. ϑ_3 is one of the four elliptic theta functions, namely

$$\vartheta_3(z, p) = 1 + 2 \sum_{n=1}^{\infty} p^{n^2} \cos 2nz. \quad (2)$$

The nature of the theta function is such that q can be approximated

$$q \approx \frac{N}{4b^2} \quad (3)$$

if $\tau \gg b^2/\pi^2$. For the NPR or B, D, and F Reactors, the lattice spacings are such that $\tau \gg b^2/\pi^2$ holds even for neutrons of small age, i.e., neutrons with energies close to the assumed 2 Mev source energy. Thus, the simplification (3) is assumed to be satisfactory for the treatment of the problem. This conclusion partially answers an important question concerning damage in the NPR vs. B, D, F graphite. In the NPR the critical graphite blocks which are the reactor supporting members are the tube blocks. In the older reactors the filler blocks are supporting. Thus, one would expect a relatively higher fast neutron flux in the NPR supporting graphite by virtue of its proximity to the fuel channels. The fact that q can be treated as spatially flat even for near source-energy neutrons relegates any position-effect to one of second order².

Under the assumption that the asymptotic energy distribution of neutrons slowing down in a non-capturing medium is valid, one obtains for the flux per unit energy E (Mev) the familiar inverse energy law

$$\phi(E) = \frac{S}{\bar{\xi} \Sigma_s E} \frac{N}{\bar{\xi} \Sigma_s 4b^2 E} \quad (4)$$

where $\bar{\xi} \Sigma_s$ is the average slowing down power in the homogeneous moderator.

²Recent multigroup calculations by Simpson and Yoshikawa bear out the spatial flatness of fast neutrons. Simpson's results indicate that the fast neutron flux above 0.24 Mev is flat to ± 12 percent in the NPR graphite. See HW-64501

Derivation of the Basic Expression (Contd.)

Since a reactor cell volume contains in addition to the moderator, voids and poor moderating materials such as cladding and tube material, the average slowing down power should be calculated to take into account these non-moderating regions. For the specific case of a Hanford reactor, it is sufficiently accurate to consider just graphite and water as significant moderators. Then,

$$\bar{\xi}\bar{\Sigma}_s = \frac{(\bar{\xi}\bar{\Sigma}_s)_w V_w + (\bar{\xi}\bar{\Sigma}_s)_l V_l}{V_{\text{cell}} - V_0}, \quad (5)$$

where the subscripts w, l, cell and 0 represent water, graphite, entire cell and uranium respectively. The V_l stand for actual volumes in a lattice cell of volume $8b^3$.

The slowing down density is energy-dependent through the energy dependence of the scattering cross sections $\bar{\Sigma}_s$. Also, the deviation from spherically symmetric scattering in the center-of-mass coordinate system (CMCS) at the higher energies introduces a forward bias to the scattering which changes the logarithmic energy loss, ξ . The n-H scattering is spherically symmetric in the CMCS well above neutron energies of importance here. Neutron-carbon scattering is fairly isotropic in the CMCS up to 1.5 Mev. Neutron-oxygen scattering however deviates from isotropy due to p-scattering at neutron energies as low as 0.4 Mev.

Values of $\xi\bar{\Sigma}_s$ for graphite and water for the energy range of interest are shown in Figure 1. These values are taken from References 3 and 4. The values plotted represent averages over the intervals indicated on the energy scale at the top of the graph.

Having the expression (4) for the flux per unit energy at energy E, one can calculate the collision density, F, for any unit volume of the moderator as a function of neutron energy

$$F(E) = \bar{\Sigma}_T(E) \phi(E) = \bar{\Sigma}_s(E) \phi(E). \quad (6)$$

The latter equality is true under our assumption of zero absorption.

$\bar{\Sigma}_s(E)$ represents the average macroscopic scattering cross section for a unit volume of the moderator

$$\bar{\Sigma}_s(E) = \frac{\bar{\Sigma}_{s,l}(E) V_l + \bar{\Sigma}_{s,w}(E) V_w}{V_l + V_w} \quad (7)$$

(3) Reactor Physics Constants, Argonne National Laboratory, (ANL-5800), p 157-159.

(4) Some Neutron Cross Sections for Multigroup Calculations, Nuclear Development Corp., N. Tralli, et al, June 30, 1958, (APEX-467).

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Derivation of the Basic Expression (Contd)

The units of F are neutron-moderator collisions per second per cm^3 of moderator per Mev at energy E .

The average fractional energy transferred to the moderator atom per neutron-moderator collision is $\alpha/2$ if the scattering is isotropic in the CMCS. The parameter α is the ratio of $E_{\min}/E$, where $E_{\min}$ is the lowest possible neutron energy after a collision (head-on collision) and E is the initial neutron energy before the collision. The average fractional energy exchange, $\bar{f}$, differs from $\alpha/2$ at higher neutron energies from the forward p-scattering. Again, at the energies under consideration, p-scattering effects become important only in oxygen.

Since our interest is in graphite which scatters very nearly isotropically in the CMCS for neutron energies to 1.5 Mev, it is not necessary to employ for $\bar{f}$ anything more exact than $\alpha/2$. For graphite $\frac{\alpha}{2} = .142$.

The energy transferred per collision is for a given moderator atom

$$\frac{dE}{dC} \Big|_i = \bar{f}_i E. \quad (8)$$

To obtain the total energy per second transferred to the graphite, expression (6) for $F(E)$ must distinguish between moderator type in the given unit volume.

$$F(E) = \frac{[\Sigma_{s1} V_1 + \Sigma_{sW} V_W] \phi(E)}{V_1 + V_W} \quad (9)$$

Is obtained using (6) and (7), and represents the collision rate per unit volume of moderator. Of a unit volume a fraction $V_1/(V_1 + V_W)$ is graphite and a fraction $V_W/(V_1 + V_W)$ is water (under our assumption of a homogeneous medium). Thus, for any unit volume of moderator and for neutrons of incident energy E ,

$$F_i^*(E) = \frac{\Sigma_{s1} V_1 \phi(E)}{V_1 + V_W}$$

neutron-graphite collisions per second take place.

For a unit volume of graphite the appropriate collision rate is

$$F_i(E) = F_i^*(E)(V_1 + V_W)V_1^{-1} = \Sigma_{s1} \phi(E). \quad (10)$$

Derivation of the Basic Expression (Contd)

To obtain the total energy transfer rate per unit energy at energy E in a unit volume of graphite, equations (4), (5), (8), and (10) are combined to yield.

$$\left. \frac{1}{V_i} \frac{dE}{dt} \right|_i = \frac{N \bar{\Sigma}_s(E) \bar{f}_i E}{4b^2 \bar{\Sigma}_s E} = \frac{N \bar{\Sigma}_s(E) \bar{f}_i (8b^3 - V_0)}{4b^2 [\bar{\Sigma}_s)_{wV_w + (\bar{\Sigma}_s)_{V_i}]} \quad (11)$$

It is now necessary to relate the energy transfer rate at energy E per unit volume of graphite per unit energy to an exposure unit of 1000 MWD/AT.

The number of fission neutrons born per second in unit length of fuel is given by the following expression

$$N = \bar{\phi}_{th} \bar{\Sigma}_f A_f (\nu_5 + \delta \nu_8) = \frac{\text{fast neutrons}}{\text{sec-cm}} \quad (12)$$

where $\bar{\phi}_{th}$ = the volume average true thermal flux in the fuel element,

$\bar{\Sigma}_f$ = the Maxwellian-averaged macroscopic fission cross section for the fuel element defined such that $\bar{\Sigma}_f \bar{\phi}_{th}$ gives the correct fission rate in the fuel element,

A_f = cross section area of fissile portion of fuel element,

ν_5 = number of fission neutrons per U^{235} fission,

ν_8 = number of fission neutrons per U^{238} fission, and

δ = U^{238} to U^{235} fission ratio.

N can be converted to a per foot basis if desired by multiplying the right side of Equation 2 by a constant α_1 , the number of centimeters per foot.

The MW/Ton in adjacent fuel can also be expressed in terms of the thermal flux in the fuel.

$$MW/AT = \bar{\phi}_{th} \bar{\Sigma}_f A_f H \alpha_1 \alpha_2 \quad (13)$$

where

H = heat of fission in megajoules/fission, and

α_2 = feet of fuel per ton

Derivation of the Basic Expression (Contd)

Thus,

$$\bar{\phi}_{th} = \frac{MW}{AT} \times \frac{1}{\sum_f A_f H \alpha_1 \alpha_2}$$

and

$$N = \left(\frac{MW}{AT} \right) \times \frac{\nu_5 + \delta \nu_8}{H \alpha_1 \alpha_2} = \text{fast neutrons/sec-cm.}$$

Thus,

$$\left[\frac{1}{V_1} \frac{dE}{dt} \right]_1 \left(\frac{MW}{AT} \right) \times \frac{\nu_5 + \delta \nu_8}{H \alpha_1 \alpha_2} \times \frac{\sum_s \bar{f}_1 (8b^3 - V_0)}{[(\sum_s Z_s)_w V_w + (\sum_s Z_s)_1 V_1] 4b^2} \quad (14)$$

Integrating (14) over 1000 days and dividing by MW/AT, we obtain the desired relationship for the total neutron energy transferred to a cm³ of graphite at energy E per unit energy for an exposure of 1000 MWD in an adjacent ton of fuel.

$$R_1(E) = \frac{(\nu_5 + \delta \nu_8) 10^3 \alpha_3 \sum_s \bar{f}_1 (8b^3 - V_0)}{4b^2 H \alpha_1 \alpha_2 [(\sum_s Z_s)_w V_w + (\sum_s Z_s)_1 V_1]} = \frac{\text{Mev transferred}}{\text{cm}^3 \text{-Mev-1000 MWD/AT}} \quad (15)$$

where $\alpha_3 = \text{sec/day}$.

Using the reactor-independent constants α_1 , α_3 , ν_5 , and ν_8 , as follows:

$$\alpha_1 = 30.4 \text{ cm/ft}$$

$$\alpha_3 = 86400 \text{ sec/day}$$

$$\nu_5 = 2.5$$

$$\nu_8 = 2.8$$

one obtains for $R_1(E)$ ⁽⁵⁾

$$R_1(E) = \frac{1.78 \times 10^6 \sum_s \bar{f}_1 (8b^3 - V_0) (1 + 1.12\delta)}{\alpha_2 H [(\sum_s Z_s)_w V_w + (\sum_s Z_s)_1 V_1] b^2} = \frac{\text{Mev transferred}}{\text{cm}^3 \text{-Mev-1000 MWD/AT}} \quad (16)$$

⁽⁵⁾ $R_1(E)$ is energy dependent through the energy dependence of the scattering cross sections, $\bar{f}_1$ and the $\sum$'s.

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APPLICATIONS

Expression (16) is not useful by itself. It must be further integrated over the energy range of interest. In the case of graphite contraction, the actual damage rate may also be a function of the energy transferred and hence to arrive at a damage figure for different reactors, one must assume some dependence between damage and energy transfer.

Equation (16) is plotted in Figure 2 for two cases of interest, a BDF type reactor and N reactor. The standard 1.33" OD solid slug is assumed for the BDF reactors since most of the irradiation history has occurred with this slug. The 21.4(avg) uranium pound per foot N reactor concentric tube fuel element is chosen for the NPR.⁽⁶⁾ The following constants are applicable to the two reactors.

TABLE I

CONSTANTS

<u>Constant</u>	<u>NPR</u>	<u>BDF</u>
α_2	93.4 ft/ton	174 ft/ton
2b	21.5 cm	21.2 cm
V_1	6695 cm ³	9248 cm ³
V_w	374 cm ³	53 cm ³
ρ_1	1.70 gm-cm ⁻³	1.656 gm-cm ⁻³
ρ_w	.830 gm-cm ⁻³	1.00 gm-cm ⁻³
δ	.073	.064
H	3.252 x 10 ⁻¹⁷ MJ/fission (203 Mev/fission)	3.220 x 10 ⁻¹⁷ MJ/fission (201 Mev/fission)
V_o	369 cm ³	198 cm ³

⁽⁶⁾ HW-62150 RD2, NPR System Parameters, Condotta and Simpson, 4-21-60, (Confidential).

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FIGURE 1
NEUTRON ENERGY TRANSFER FUNCTION $R_1(E)$

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APPLICATIONS (Contd)

A few comments on the curves in Figure 2 are worth mentioning. The first is the fact that, although the NPR contains considerably more water than a BDF reactor (almost six times as many water molecules per lattice), the beneficial effect of the water to the NPR is markedly lower at the higher neutron energies than at the lower energies. This is due to the poorer moderating property of the water (as compared to graphite) at the higher energies due to oxygen scattering competition with hydrogen scattering. The second feature is the small energy dependence on the energy transfer function over the range of interest. This is due principally to the fact that the fast flux is assumed $1/E$ in energy dependence and the energy transfer per collision is proportional to E . Thus, $R_1(E)$ depends on energy through rather slowly varying functions of energy.

Without specific knowledge of graphite damage as a function of neutron energy (except that neutrons above a certain threshold are necessary to cause any damage) it may be instructive to postulate two rather different damage-energy relationships and determine what damage ratio exists between a BDF reactor and NPR. Consider the following relationships.

1. The graphite damage is energy independent above an effective threshold of .001 Mev.
2. The graphite damage varies as E above .001 Mev.

Performing the necessary integration of

$$\Phi(E_1, E_2) = \int_{E_1}^{E_2} B(E) R_1(E) dE$$

where $B(E)$ is the damage-energy relationship and E_1 and E_2 are .001 and 2 Mev, respectively, the following ratios of Φ are obtained between NPR and BDF for the same MWD/AT.

TABLE II

<u>Assumption</u>	<u>Φ NPR / Φ BDF</u>
$B(E) = \text{Const}$	1.95
$B(E) = \text{Const } \times E$	1.98

The above results point out that the comparative integrated graphite damage over the neutron energy range of .001 to 2 Mev is fairly insensitive to the particular damage-energy relationship assumed.

A second application of the relationship given by Equation 16 is a qualitative comparison of damage ratios for graphite specimens indicated in various test reactors. If the pertinent reactor constants such as given in Table I are known for the test reactor, the use of Equation 16 will give the approximate

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APPLICATIONS (Contd)

energy transferred to a unit volume of the graphite. It must be remembered for a irradiation of a graphite sample in a foreign reactor (as opposed to the irradiation of graphite in a reactor for which the graphite is part of the original moderator) that the cross section Σ_s in Equation 16 is for the graphite sample. Since Σ_s depends on graphite density, it is immediately obvious that this factor automatically takes into account the density effect on the collision rate. Any density dependent characteristics pertaining to actual graphite stability, however, are naturally impossible to include in the formulation presented here.

If the test reactor is so small that the reduction into equivalent cells is not possible, or if the test samples are irradiated in positions which are abnormally near fast neutron sources, then the validity of the derived expressions is reduced. In the latter case, a position-dependent fast flux factor may be needed in Equation 16. A multigroup diffusion theory analysis would suffice to provide the needed spatial dependence of the flux. Monte-Carlo calculations presently being programmed by R. D. Cahoon will also serve to provide the spatial dependence of the energy transfer function R_1 .

RN:rf

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